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Site Profile for the West Valley Demonstration Project

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☐ New ☒ Total Rewrite ☐ Revision

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
06/22/2007	00	New approved Site Profile for the West Valley Demonstration Project. Incorporates formal internal and NIOSH review comments. There is an increase in assigned dose and a PER is required. Training required: As determined by the Task Manager. Initiated by Robert E. Burns, Jr.
08/17/2007	00 PC-1	Approved page change revision initiated to assign new document owner and to correct PER statement from Rev 00. No PER was required because Task 5 evaluation determined previously worked cases would not be adversely impacted (no compensability change) by the Rev 00 issuance of this TBD. No changes were needed as a result of formal internal and NIOSH review. Training required: As determined by the Task Manager. Initiated by Jackson R. Ellis.
10/07/2025	01	Revised to incorporate SEC-00252. Section 1.3 edited to update internal dose reconstruction limitations and partial dose reconstruction information introduced by SEC-00252. Section 5.1.1 was edited to include the limitations of internal dose reconstruction during the SEC period and includes information on partial dose reconstruction. Editing comments outside the NIOSH-directed scope of incorporating new language for the SEC discussion were archived for a later revision. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Initiated by Wade C. Morris and authored by Eugene M. Rollins.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
ARC	Acid Recovery Cell
ARPR	Acid Recovery Pump Room
ASDA	New York State Atomic and Space Development Authority
AWE	Atomic Weapons Employer
Bq	becquerel
CAA	Cell Access Aisle
CCR	CPC Crane Room
CEDE	committed effective dose equivalent
C.F.R.	*Code of Federal Regulations*
CHP	Certified Health Physicist
Ci	curie
cm	centimeter
COA	Chemical Operating Aisle
CPC	Chemical Process Cell
cpm	counts per minute
CR	Control Room
CUP	Cask Unloading Pool
CVA	Chemical Viewing Aisle
d	day
D&D	decontamination and decommissioning
DAC	derived air concentration
DOE	U.S. Department of Energy
DOG	Dissolver Off Gas
DOL	U.S. Department of Labor
dpm	disintegrations per minute
EDR	Equipment Decontamination Room
EDRVA	Equipment Decontamination Room Viewing Aisle
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EMOA	East Mechanical Operating Aisle
EPA	U.S. Environmental Protection Agency
F	fast (absorption type)
FRS	Fuel Receiving and Storage
ft	foot
g	gram
gal	gallon
GCR	GPC Crane Room
GOA	General Operating Aisle
GPC	General Purpose Cell
HEPA	high-efficiency particulate air
HEV	Head End Ventilation
HLW	high-level radioactive waste
HNS	Helgeson Nuclear Services

hr	hour
HTO	tritiated water
ICRP	International Commission on Radiological Protection
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
KPA	kinetic phosphorescence analysis
L	liter
lb	pound
LLLW	Low-Level Liquid Waste
LLW	low-level waste
LLWT	Low-Level Waste Treatment
LWA	Lower Warm Aisle
LXA	Lower Extraction Aisle
m	meter
M	moderate (absorption type)
MDA	minimum detectable amount
MDC	minimum detectable concentration
MDL	minimum detectable level
MeV	megaelectron-volt, 1 million electron-volts
MFAPs	mixed fission and activation products
MFPs	mixed fission products
mL	milliliter
mo	month
MOA	mechanical operating aisle
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MPLB	maximum permissible lung burden
mR	milliroentgen
mrad	millirad
mrem	millirem
MRR	Manipulator Repair Room
MSM	master-slave manipulator
MTU	metric ton of uranium
MWd	megawatt-day
nCi	nanocurie
NDA	NRC-licensed Disposal Area
NFS	Nuclear Fuel Services
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NYSERDA	New York State Energy Research and Development Authority
NYU	New York University
OGA	Off Gas Aisle
OGBR	Off Gas Blower Room
OGC	Off Gas Cell
ORAU	Oak Ridge Associated Universities

ORAUT	ORAU Team
PA	posterior-anterior
PAR	Programmed and Remote Systems
PCR	Process Chemical Room
PEA	Pulser Equipment Aisle
PER	program evaluation report
PMC	Process Mechanical Cell
PMCR	PMC Crane Room
ppb	parts per billion
PPC	Product Purification Cell
PPH	Product Packaging and Handling
PPS	Product Packaging and Shipping
PSC	Process Sample Cell
PSF	Plutonium Storage Facility
PUREX	plutonium-uranium extraction
R	roentgen
RHWF	Remote Handled Waste Facility
s	second
S	slow (absorption type)
scfm	standard cubic feet per minute
SDA	State-licensed Disposal Area
SEC	Special Exposure Cohort
SP	Storage Pool
SR	Scrap Removal
SRDB Ref ID	Site Research Database Reference Identification (number)
SSC	Sample Storage Cell
SST	Solvent Storage Terrace
STS	Supernatant Treatment System
Sv	sievert
SWP	Special Work Permit
t	ton
TBD	technical basis document
THOREX	thorium extraction
TLD	thermoluminescent dosimeter
TRU	transuranic
U.S.C.	*United States Code*
ULO	Uranium Loadout Area
UPC	Uranium Product Cell
UWA	Upper Warm Aisle
UXA	Upper Extraction Aisle
VEC	Ventilation Exhaust Cell
VOG	Vessel Off Gas
VWR	Ventilation Wash Room
wk	week
WMOA	West Mechanical Operating Aisle
WTF	Waste Tank Farm

wt%	weight percent
WVDP	West Valley Demonstration Project
WVNSC	West Valley Nuclear Services Company
XC1	Extraction Cell 1
XC2	Extraction Cell 2
XC3	Extraction Cell 3
XCR	Extraction Chemical Room
XSA	Extraction Sample Aisle
yr	year
µCi	microcurie
µg	microgram
µm	micrometer
µrem	microrem
§	section or sections

1.0 INTRODUCTION

Technical basis documents (TBDs) and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular U.S. Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies), such as changing scientific understanding of operations, processes, or procedures involving radioactive materials. These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of individual dose reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA).

In this document the word “facility” is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean, nor should it be equated to, an “AWE facility” or a “DOE facility.” The term “AWE facility” is defined in EEOICPA to mean “a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling.” 42 *United States Code* (U.S.C.) § 7384l(5). On the other hand, a DOE facility is defined as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located—(A) in which operations are, or have been, conducted by, or on behalf of, the [DOE] (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program); and (B) with regard to which the [DOE] has or had—(i) a proprietary interest; or (ii) entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services.” 42 U.S.C. § 7384l(12). The DOE determines whether a site meets the statutory definition of an AWE facility and the U.S. Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Under EEOICPA, a Part B claim for benefits must be based on an energy employee’s eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility’s designated time period and location (i.e., a “covered employee with cancer”). After DOL determines that a claim meets the eligibility requirements under EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and types of radiation exposure to be included in an individual dose reconstruction.

Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. 42 U.S.C. § 7384l(11). Also under EEOICPA, the types of exposure to be included in dose reconstructions for DOE employees are those radiation exposures incurred in the performance of duty. As such, NIOSH includes all radiation exposures received as a condition of employment at DOE facilities in its dose reconstructions for covered employees, which may include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. This is because NIOSH does not determine the fraction of total measured radiation exposure at a DOE facility that is contributed by the Naval Nuclear Propulsion Program at the DOE facility during a specified period of time for inclusion in dose reconstruction.

On the other hand, under EEOICPA, eligible employment at a AWE facility is categorized as employment either (1) during “a period when the employer was processing or producing, for the use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining and milling,” (i.e., the operational period); or (2) during a period that NIOSH has determined that “there is a potential for significant residual contamination outside of

the period in which weapons-related production occurred," (i.e., the residual contamination period). 42 U.S.C. § 7384l(3).

The EEOICPA definition for eligible AWE employment during the operational and the residual contamination periods creates two statutory frameworks directing types of radiation exposures to be included in individual dose reconstructions. For employment falling within an AWE facility's operational period, NIOSH includes radiation exposures incurred in the performance of duty, such as medical X-rays received as a condition of employment for participating in DOE projects at an AWE facility in dose reconstructions. Additionally, the total measured, occupational radiation exposure during an operational period may include radiation exposure contributed by the Naval Nuclear Propulsion Program at the AWE facility and any radiation exposure received from the production of commercial radioactive products that were concurrently manufactured by the AWE facility during the operational period.

In contrast, only two categories of radiation exposure are included in dose reconstructions for claims involving employment during the residual contamination period. 42 U.S.C. § 7384n(c)(4). First, NIOSH must include exposures to radiological contaminants resulting from activities that had a nuclear-weapon nexus or conducted by or on behalf of the DOE (with an exclusion of activities related to, among other things, the Naval Nuclear Propulsion Program) that took place during the operational period. 42 U.S.C. § 7384n(c)(4)(A). Second, radiation doses from sources not included in the first category but cannot be distinguished through reliable documentation should also be included in dose reconstructions. 42 U.S.C. § 7384n(c)(4)(B). Furthermore, NIOSH does not include doses from medical X-rays performed during the residual contamination period in dose reconstructions because all DOE-related activities have ceased during the residual contamination period, and thus any medical X-ray performed during the residual contamination period is not a condition of employment for participating in DOE-related activities [NIOSH 2007].

Likewise, NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment for DOE-related activities at a DOE or AWE facility. Therefore these exposures are not included in dose reconstructions for covered employees [NIOSH 2010]:

- Background radiation, including radiation from naturally occurring radon present in conventional structures, and

- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons.

1.1 PURPOSE

This site profile provides information about operations at the West Valley reprocessing plant operated initially by Nuclear Fuel Services (NFS) as an AWE site and subsequently by DOE as the West Valley Demonstration Project (WVDP). The operations addressed pertain to occupational radiation exposures for monitored or unmonitored workers.

1.2 SCOPE

Section 2.0 provides a description of the site and operations that pertain to possible radiation exposures and discusses radiation source terms. Section 3.0 provides guidance for the determination of occupational medical dose. Section 4.0 provides guidance for the determination of environmental dose to workers from external radiation and releases of radioactive materials to the environment. Section 5.0 provides guidance for the determination of internal doses based on internal monitoring for periods where workers were monitored for internal dose. Section 6.0 provides guidance for the

determination of external doses from measured doses or for periods for which records of measured doses are missing.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 7.0.

1.3 SPECIAL EXPOSURE COHORT

January 1, 1969, through December 31, 1973

The Secretary of the U.S. Department of Health and Human Services has added the following class of WVDP employees to the SEC:

> *All Atomic Weapons Employees who worked at the West Valley Demonstration Project in West Valley, New York, during the period from January 1, 1969, through December 31, 1973, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort* [Azar 2019].

The Secretary based this designation on the recommendation of the Advisory Board on Radiation and Worker Health (ABRWH), the Director of NIOSH, and the NIOSH Evaluation Report, which all found there is insufficient internal bioassay data and accurate workplace monitoring data to reconstruct internal dose from uranium from January 1, 1969, through December 31, 1973, and from mixed fission products for 1972 and 1973 [Azar 2019, NIOSH 2019].

Although NIOSH found that it is not possible to completely reconstruct internal radiation doses for all WVDP workers, NIOSH intends to use any internal and external monitoring data that might become available for an individual claim and that can be interpreted using existing NIOSH dose reconstruction processes or procedures to conduct partial dose reconstructions for employees who do not qualify for inclusion in the SEC [Azar 2019, NIOSH 2019].

2.0 SITE DESCRIPTION

2.1 SITE HISTORY

In 1962, the New York State Atomic and Space Development Authority (ASDA) and NFS partnered to build a privately owned nuclear fuel reprocessing plant. Construction of the plant began in June 1963 and took approximately 3 years to complete. The plant was built on 200 acres of land leased from ASDA by NFS. (The State of New York took title to 3,345 acres of land in 1961 and established the Western New York Nuclear Service Center.)

NFS was granted a license to receive and store fuel on May 27, 1965. The first fuel for reprocessing was received June 3, 1965, and placed in the Storage Pool (SP) on June 5. Approximately 60% of the fuel processed at NFS came from the U.S. Atomic Energy Commission (AEC), as part of its guarantee to supply fuel until an adequate commercial market grew. A majority of this AEC-supplied fuel came from the Hanford N Reactor [DOE 1999].

An operating license was granted to NFS for the reprocessing plant on April 19, 1966, and reprocessing began on April 22. The first three processing campaigns involved fuel from the Hanford Site N Reactor [Clark 1979].

The NFS processing plant was designed to be a multipurpose facility with the capability of reprocessing nuclear fuel from a number of different fuel cycles [Bailey 1990]. The plant primarily used the plutonium-uranium extraction (PUREX) process; however, one thorium extraction (THOREX) campaign was processed between November 1968 and January 1969.

From 1966 to 1972, NFS handled and reprocessed a total of 630 t of fuel from nine different reactors during 28 campaigns. Fuels processed included light-water reactor fuels (from both boiling- and pressurized-water reactors), fuels from AEC-owned reactors (such as the Hanford N Reactor), and a uranium-thorium fuel cycle core from the Indian Point 1 reactor. Plutonium and uranium recovered from irradiated fuels were delivered as nitrates. The recovered uranium was sent to the Fernald (Ohio) Plant, and the plutonium was sent to either Hanford or (later) to the ASDA Plutonium Storage Facility (PSF). In general, AEC plutonium was sent to Hanford, and utility-owned plutonium was retained by the utilities, sold to industry, or sold to NFS for later resale for use in plutonium recycle. NFS and the ASDA entered into a plutonium storage agreement on May 3, 1971, in which ASDA would store plutonium at its PSF. The PSF was approximately 2,200 m east-southeast of the NFS plant [ASDA/NFS 1971]. Storage of plutonium at the PSF was to be terminated by December 31, 1974 [NFS 1974a, p. 24].

In March 1972, the reprocessing plant was shut down to complete a series of improvements intended to increase capacity and meet new regulatory requirements. From 1973 to 1975, 756 spent light-water reactor fuel assemblies were shipped to the West Valley facility and placed in the SP in anticipation of resumption of reprocessing operations [from Bailey 1990]. However, after a decision by the AEC that a completely new licensing process would be required, and facing more stringent requirements on plant effluents, NFS concluded that reprocessing no longer made economic sense. In 1977 it transferred management of the facility to the New York State Energy Research and Development Authority (NYSERDA).

In 1980, the U.S. Congress passed the West Valley Demonstration Act, which directed DOE to solidify the high-level radioactive waste (HLW) and to decontaminate and decommission the tanks and facilities at WVDP. DOE assumed operational control of the West Valley site on February 26, 1982, with West Valley Nuclear Services Company (WVNSC) as its contractor [WVNSC 1983].

One of the first major projects under the WVDP was to address the hundreds of unreprocessed spent nuclear fuel assemblies that remained in the SP. During the early 1980s, a campaign was conducted to ship spent fuel assemblies back to their point of origin. When this initial shipping campaign was halted, 125 fuel assemblies still remained in the SP. Work was subsequently performed in the mid-1980s to remove the majority of empty fuel canisters and storage racks in the pool. The SP still contained 149 canisters and 11 rows of storage racks after this work was completed. By the early spring of 2001, all of the remaining 125 spent nuclear fuel assemblies had been transferred into two shipping casks for eventual shipment to the Idaho National Laboratory. By late 2001, the remaining 149 canisters and storage racks had been removed and decontamination of the SP began [excerpted from Jablonski et al. 2003].

Decontamination of the main reprocessing plant took place in the early 1980s to allow for utilization of the existing facilities for the HLW vitrification project to the extent practicable [Johnson 1991]. Decontamination activities took place in a number of former process cells from 1982 through 1987 [Hoffman 1997].

Pretreatment of the HLW began in 1988 to partition the cesium and strontium from the other constituents and thus create two waste streams: one high level and one low level. This partitioning reduced the amount of HLW that required vitrification. This waste pretreatment continued until 1995. The effluent (low-level) liquid from the pretreatment process was sent to the Cement Solidification

System to be solidified in cement. This low-level waste (LLW) was packaged in approximately 20,000 steel drums, which are being stored at West Valley pending disposal.

Vitrification of the HLW took place between June 1996 and November 2001. Followed by a program to flush residuals through it, the vitrification facility melter was shut down in September 2002. The HLW solidification campaign involved the vitrification of approximately 24 million Ci of activity into 275 glass logs. Each log contains approximately 2,000 kg of glass. The individual logs are encased in canisters. Individual canisters had a contact dose rate of approximately 2,600 rem/hr in 2003. They are stored in the Chemical Process Cell (CPC) [Petkus et al. 2003].

In 1999 Vitrification Expended Materials Processing was initiated to begin processing unserviceable equipment in the Vitrification Facility. The Remote Handled Waste Facility (RHWF) was subsequently developed to process large-scale, highly contaminated equipment excessed during decontamination and decommissioning (D&D) activities. Groundbreaking for the RHWF took place in 2000, as the WVDP restructured to focus on D&D.

In 2003, projects included removal and treatment of sodium-bearing wastewater from underground storage tanks and decontamination of the process cell used for purification of the plutonium product stream when the reprocessing plant was operating. Debris was removed from the General Purpose Cell (GPC) and Process Mechanical Cell (PMC) and packaged, and the Vitrification Cell Dismantlement Project was initiated.

Shipping LLW off the site was a major activity in 2004, with 104,427 ft^3 of waste being shipped for disposal. Footprint reduction began with the removal of 20 office trailers from the site. Decontamination of four of the major process cells continued: the Vitrification Cell, PMC, GPC, and Extraction Cell 2 (XC2). In November 2004, the Vitrification Melter was removed from the Vitrification Facility, packaged in a specially designed container, and staged on the site for potential offsite disposal. The Vitrification Melter was the last large component removed from the Vitrification Cell. The Vitrification Cell Dismantlement project was completed in mid-2005 [WVNSC/URS 2006, p. 7].

LLW shipments for offsite disposal increased significantly in 2005. Site footprint reduction activities escalated, and a number of site employees were moved to an offsite building.

In addition to the reprocessing plant, NFS operated a commercial radioactive waste disposal site at West Valley from 1963 to 1975. Disposal of solid material by land burial began in November 1963. During this time NFS buried waste from commercial generators totaling 2.4 million ft^3. These wastes, which came from nuclear power facilities, hospitals, and academic, industrial, and government facilities, and the reprocessing plant itself, were buried in soil trenches. The material ranged in activity from low-specific activity material to containers with exposure rates of 8,000 to 10,000 R/hr [Facilities description, no date]. This burial site, known as the State-licensed Disposal Area (SDA), encompasses approximately 15 acres. NYSERDA took over maintenance of the SDA in 1983.

NFS also operated a disposal (burial) area for high-level solid wastes generated during fuel reprocessing. These reprocessing wastes, which included fuel cladding, were buried in a 7-acre landfill known as the U.S. Nuclear Regulatory Commission (NRC)-licensed Disposal Area (NDA).

2.2 REPROCESSING PLANT OPERATIONS PERIOD, 1965 TO 1973

2.2.1 Plant and Process Description

The NFS reprocessing plant was on a 3,345-acre site bounded by a barbed-wire fence and posted as a restricted area. The plant itself and its waste storage operations were performed within a 300-acre security area at about the center of the site. The security area was defined by an 8-ft security fence.

The reprocessing plant consisted of a complex of cells with the various supporting and operating areas grouped around them [Nelson 1972, p. 4]. The plant was arranged in the shape of a U, with the Fuel Receiving and Storage (FRS) facility on one end and the product removal facilities on the other. The mechanical and chemical processing cells were in the middle [Runion 1970]. Most areas of the process building fall into one of three categories: shielded cells, operating aisles, and unshielded rooms [WVDP 2005]. The cells consisted of reinforced concrete walls several feet thick. The rest of the plant was of cinderblock construction. Chemical operations were directed from the Control Room, while mechanical operations were directed from operating aisles adjacent to viewing windows in the hot cells [Runion 1970].

Fuel reprocessing involved shearing and leaching of fuel elements on the front end, followed by solvent extraction to remove plutonium and uranium. Fuel arrived at the site in shipping casks. The casks were unloaded under water and the fuel was stored in a SP. These operations took place in the FRS facility. Fuel bundles would be inverted and passed through a transfer canal into the PMC. Shearing and sawing equipment removed hardware and segmented the fuel into fixed lengths. The segmented fuel fell through a chute into baskets in the GPC below. The baskets were then passed into the CPC where the segmented fuel was dissolved in acid. This process, known as leaching, dissolved the fuel, leaving behind the cladding and any structural components. This leftover material, referred to as “hulls,” was rinsed and disposed of in the NDA as HLW. The plutonium and uranium in the dissolved fuel were separated from the fission products and purified in a series of extraction cells. The purified products were then sent to their owners as nitrate solutions. After May 1971, plutonium solutions were sometimes also sent to the PSF, which was owned and operated by the State of New York on the West Valley site. High-level liquid waste generated in the process was stored in underground tanks. Some of the acids used in the process were recycled for reuse.

Thorium-uranium fuel from Indian Point Unit 1 was processed from November 15, 1968, to January 20, 1969 [Birchler 1970]. This is referred to as the “Con. Ed.” campaign in the vernacular of the day. Following the THOREX processing, the plant was thoroughly flushed to reduce the amount of high-enriched uranium in systems as much as possible. Processing of the thorium fuel required significant changes to the process to account for the high levels of thorium, ^{235}U, and ^{233}U present. The fuel also contained ^{239}Pu. The high concentrations of fissile species required the use of boron in the dissolving process. A number of systems normally used in processing (e.g., wash systems, silica beds) were not used due to criticality concerns. High ^{238}Pu activity required resin beds to be regenerated more frequently than usual.

Process Summary:

- Fuel receipt and storage;
- Mechanical preparation;
- Fuel dissolving;
- Solvent extraction;
- Plutonium product concentration, storage, and loadout; and
- Uranium product concentration, storage, and loadout.

As of 1970 the plant had processed fuel with burnups as high as 30,000 MWd/MTU [Runion 1970]. The last fuel reprocessing campaign at West Valley was completed in November 1971 [Nelson 1973a, p. 4]. The last plutonium scrap recovery operation took place in March 1972 [Nelson 1973b, p. 9]. Afterward, the plant was in a shutdown condition, with operations limited to fuel receipt and storage

and decontamination activities. Per the AEC, decontamination activities were significantly curtailed after May 4, 1973 [Nelson 1973b, p. 9].

2.2.2 Radiation Protection Program

Given the extreme radiological environment which existed while the plant was in operation, the radiation protection program focused on controlling personnel exposures relative to legal maximums. Exposure control was threshold-oriented, i.e., concerned with not exceeding quarterly exposure limits, allowable breathing zone air concentrations, maximum permissible body burdens (MPBBs), etc. [e.g., Keely and Wenstrand 1971].

NFS defined contamination areas using a system of four "zones" based upon removable contamination levels (Table 2-1). Zone 1 referred to equipment and areas with no smearable contamination. Zone 2 areas were those where removable contamination levels were such that special access controls were not deemed necessary. Zone 3 areas were those with contamination requiring special control, and Zone 4 areas required "stringent" control. Entry points into Zone 4 areas had buffer zones classified as Zone 3 areas. Entries into Zone 4 areas were covered under Special Work Permits (SWPs). Radiation Work Permits were used otherwise.

Table 2-1. NFS-defined contamination areas.[a]

Category	Removable alpha (dpm/100 cm^2)	Removable beta (dpm/100 cm^2)	Posting and barriers
Zone 1	≤10	≤100	"Clean" areas outside the security fence: not posted
Zone 2	10–50	100–500	"Clean" areas inside the security fence: not posted
Zone 3	50–500	500–50,000	Posted, rope barrier, step-off pad
Zone 4	≥500	≥50,000	Posted, rope barrier, step-off pad, entrance was airlock or tented

a. Source: NFS [1976].

All repair or maintenance activities in Zone 4 areas or in high radiation areas (exposure rates ≥100 mR/hr) were performed under SWPs. Standard Operating Procedures addressed contamination control, doffing, etc. Decontamination was performed after maintenance activities to reduce contamination to premaintenance levels. Radioactive wastes generated from maintenance activities were packaged and sent to one of the two burial sites for disposal.

Respiratory protection thresholds were as given in Table 2-2 [NFS 1974b]. These values are the maximum allowable concentration for the corresponding respiratory protection equipment.

Table 2-2. NFS respiratory protection thresholds (µCi/cm^3).[a]

Radionuclide	No respiratory protection	Full face canister mask	Supplied air mask (constant flow)
MFPs	1E−09	2E−08	1E−06
Pu-239	2E−12	4E−11	2E−09
Uranium	6E−11	1E−09	6E−08

a. Source: NFS [1974b].

It is of note that smoking was allowed in operating areas of the process building such as the mechanical operating aisles (MOAs). A Johnson & Higgins insurance inspection report from 1972 [Nelson 1972] echoes information seen in other documents that smoking in contaminated areas was an issue at West Valley. They recommended more extensive posting and enforcement of "no smoking" requirements in surface contamination areas. As of January 1, 1973, smoking, eating, drinking, or chewing were prohibited in Zone III or Zone IV areas, but smoking was still allowed in designated Zone II areas [NFS 1973a, p. 3].

2.2.3 Special Considerations for the Operations Period

Users of this site profile should bear in mind the West Valley reprocessing plant was an extreme radiological environment throughout the operations period. Fuel segmentation operations resulted in substantial quantities of high specific activity airborne particulate matter, resulting in significant operational difficulties associated with the plant ventilation systems and airflow issues. This, coupled with other unforeseen circumstances involving radioactivity in systems where it was not anticipated, or at unanticipated levels, meant radiological conditions encumbered operation of the facility from the outset. Routine, contact maintenance activities had to be performed in high dose rate environments. Dose rates in normally occupied areas were also high, and radiological contamination was substantial plant-wide from maintenance activities and spills (see Attachment A for details). High backgrounds compromised the effectiveness of contamination control measurements [1].

At its maximum, the plant employed a permanent work force of less than 200 persons during operations. Contract employees had to be brought in once the regular employees reached their quarterly dose limits. There are also reports of using persons “not normal[ly] assigned to radiation work” to complete high exposure tasks [Keely 1967a, p. 2]. A 1972 insurance survey report remarked, “To facilitate operations and meet AEC established radiation exposure criteria the plant utilizes about 1000 temporary laborers each year. Radiation exposure and contamination hazards are extensive and ever present in this type of operation” [Nelson 1972, p. 5].

Users of this site profile should recognize the principal focus of the radiation protection program at West Valley during its operations period was to minimize the number of individuals receiving exposures in excess of legal maximums [2]. This is particularly true for external exposures, which were controlled to a quarterly limit of 3 rem to the whole body. Quarterly skin and extremity limits were also enforced. Thresholds for internal monitoring were also applied, commensurate with the regulatory requirements of that period. Personnel exposures could not be optimized in accordance with normal radiation protection practice given the work environment. Therefore, any assumptions regarding the likelihood of personnel exposures at West Valley during its operations period based on job title or work area should be carefully considered [3]. In particular:

- For 1966 through 1973, users should not assume nonradiation workers or individuals who did not normally work in the process building did not receive appreciable occupational exposures without additional justification.

- Users should be aware that significant contamination or exposure rates (or both) could have been present in areas not normally associated with personnel exposures, such as stairways, lobbies, etc. Significant contamination could result from large spills of process solutions, in which case it would be persistent indefinitely, or from widespread tracking of contamination from maintenance activities, which would not be detected until it reached lower background areas of the building. This consideration applies primarily to the 1966 – 1973 time period.

In addition to the following discussions of personnel exposure mechanisms at West Valley during operations, users are also referred to Attachment A for detail on the radiological environment in the process building and other areas.

2.2.4 Chronic Sources of Personnel Exposures

In a March 1972 letter, the AEC criticized NFS for designing its radiation control program vis-à-vis maximum allowable exposure limits for workers over the course of its operating history rather than making efforts to maintain low levels of exposure, consistent with “universally accepted” radiation protection practice. In response, NFS acknowledged it appeared that the principal cause of these conditions was “performance of plant equipment at less than design expectations” [Miller 1972a, p. 4].

The overall largest contributors to the chronically high personnel exposures at West Valley during its operating years are discussed below.

2.2.4.1 Plant Ventilation Systems

The principal personnel exposure mechanism in the processing plant was “finely divided fuel particles” in direct-maintenance work areas adjacent to process cells, in analytical hot cells, in sample enclosures, solution storage areas, filter housings, and ventilation ducts. The particles were released during various head-end process operations, including shearing of fuel rods and transporting these from one process cell to another, dissolving of fuel, and handling the leftover cladding material after dissolving. The particles were a problem for both inhalation exposures and extremity exposures due to their high specific activity and significant beta dose rates [Keely and Wenstrand 1971].

The buildup of this high specific activity particulate matter in ducts and other parts of the ventilation system was a continuing problem. Following treatment in a washer (scrubber), ventilation cell air flowed through a 36-in. duct to filtration equipment at the base of the stack. Buildup of activity in that duct caused significant radiation levels in the Analytical Laboratory areas, the Lower Extraction Aisle (LXA), and the Control Room. Activity collecting in the washer caused exposure rates of 40 mR/hr in the Analytical Aisle, which was directly above the washer, during processing of high exposure fuels [NFS 1976]. Particulate activity in the ventilation system ducting resulted in exposure rates as high as 100 mR/hr in the LXA (below the duct), 10 to 20 mR/hr in the Analytical Laboratories, and 5 to 10 mR/hr in the Control Room [NFS 1976]. The plant design was such that periodic filter changes and other maintenance activities on ventilation system components were contact maintenance tasks, i.e., they could not be performed remotely. Personnel involved in these operations encountered considerable exposure rates.

In response to this problem the plant was modified in two ways. First, penetrations were made into the main ventilation system to allow periodic flushing to reduce built up particulate activity. These duct-flushing operations would temporarily reduce the exposure rates in personnel areas, but would also result in high airborne contamination in some areas while flushing was taking place. The second modification was to add an auxiliary ventilation system for the head-end process cells. This Head End Ventilation (HEV) system went into operation in October 1970. Following startup of the HEV system and decontamination of the original ventilation ducts and washer, radiation levels dropped by a factor of 5 to 10 in the Analytical Laboratories and in the Control Room [Miller 1972a, p. 6]. However, radiation levels in the 36-in. duct continued to be higher than desired even after the HEV system came on line.

Wenstrand [1972a] states the ventilation duct running between the ventilation washer and the Ventilation Exhaust Cell (VEC) continued to be a primary source of radiation on three levels of the plant. It says the duct and washer were extensively decontaminated after startup of the HEV system to reduce exposure rates in the operating aisles. However, as of March 1972 the radiation levels from the duct had doubled since the decontamination. The report points out the duct contained a number of 90-degree bends that interfered with airflow.

2.2.4.2 Acid Recovery System

Due to startup problems with the waste evaporator systems in 1966, radioactive contamination was spread throughout the acid recovery system. This unanticipated radioactivity in the recovered acid system resulted in significant exposure rates in numerous areas of the processing plant where such elevated levels were not expected.

Carryover of radioactivity from the waste evaporators and buildup of ruthenium on the inner walls of the process chemical feed tanks resulted in exposure rates up to 100 mR/hr in the vicinity of the

tanks. Ruthenium buildup also occurred in the recovered acid storage tanks located in the Hot Acid Cell. This cell was constructed of concrete block and did not afford adequate shielding. Exposure rates in the vicinity of the Hot Acid Cell therefore ranged from 100 mR/hr to 1,000 mR/hr. Activity in the recovered acid storage tanks caused exposure rates up to 10 mR/hr in the Control Room and Office areas [NFS 1976]. It also caused elevated exposure rates outside the plant, e.g., in front of the office building.

2.2.4.3 Fuel Receiving and Storage Facility

The FRS facility was a significant source of personnel exposure for workers in the head end of the plant. The primary source of these exposures (internal and external) was contamination of the pool water from ruptured fuel elements, and failure of the cleanup systems to reduce activity concentrations to desirable levels. A significant decontamination effort was undertaken in the FRS for both the pool water and surrounding floors after the plant shut down in March 1972.

2.2.5 Miscellaneous Sources of Personnel Exposures

This section describes miscellaneous sources resulting in personnel exposures in the reprocessing plant. Attachment A to this document provides detailed information on operating areas and associated radiological conditions.

Buildup of Contamination in Piping
Radioactivity would accumulate in steam lines in the LXA, Upper Extraction Aisle (UXA), and the Chemical Operating Aisle (COA). These steam lines were used to jet process solution. Exposure rates from the lines were 10 to 30 mR/hr and contributed to personnel exposures during sampling and valving operations [NFS 1976].

Duct Flushing Operations
Penetrations were made into the main ventilation system in 1967 to provide for periodic flushing of the ductwork to reduce exposure rates in working areas. These duct-flushing operations would cause elevated airborne radioactivity levels in numerous areas of the plant. In July 1970 the following areas showed concentrations that would have resulted in inhalation exposure in excess of 40 maximum permissible concentration (MPC)-hours [Wenstrand 1970a]:

- General Operating Aisle (GOA);
- MOA (east and west sides);
- Process Sample Cells (PSCs) 1, 2, and 3;
- Extraction Sample Aisle (XSA);
- Pulser Equipment Aisle (PEA), and
- South Stairs.

Radioactivity in Cooling Water, Steam and Condensate Piping
Radioactivity in cooling water, steam, and condensate piping were a source of elevated exposure rates in the following areas [NFS 1976]:

- COA,
- LXA,

- MOA,
- Off Gas Aisle (OGA),
- UXA,
- Utility Room,
- High Level Waste areas, and
- Control Room.

Analytical Activities
Elevated exposure rates existed in the vicinity of sampling stations from activity in sampling bottles and sample lines. Exposure rates were also elevated in the analytical laboratory areas from the handling and assay of samples [NFS 1976].

Extraction Cell Support Equipment
The Lower Warm Aisle (LWA), Upper Warm Aisle (UWA), and the PEA contained pumps, valves, and lines in shielded areas for the transfer of process solution within the process cells. These areas frequently became contaminated from maintenance activities.

2.2.6 Personnel Exposures Associated with Specific Tasks or Duties

This section contains personnel exposure information associated with specific tasks or job duties at the West Valley processing plant in its operations period.

2.2.6.1 Maintenance Workers

Maintenance employees received exposures associated with repair, maintenance, and replacement activities for contaminated plant equipment. These activities often required “personnel exposures higher than those normally received by Operations employees” [NFS 1976].

Analysis of personnel exposure data for maintenance workers for 1970 to 1971 showed the principal sources of exposure for these individuals were the crane rooms and the acid recovery system. Most of the exposure in the crane rooms was associated with decontamination and shielding of the area and equipment before maintenance activities. The total collective dose for prejob decontamination and shielding activities in the three crane rooms was 221 person-rem/yr [NFS 1976].

The maintenance task resulting in the highest personnel exposures was repair or replacement of the crane cable reels. Replacing or untangling of crane or hoist cables was also a significant source of personnel exposure for maintenance workers. Other tasks associated with significant dose to maintenance workers were repair or replacement of crane motors, repair of power manipulators, and retrieval of disabled cranes from the process cells. Collective dose for maintenance workers from decontamination and repair of master-slave manipulators (MSMs) was 10 person-rem/yr. Collective dose for maintenance workers in the Equipment Decontamination Room (EDR) was 22 person-rem/yr [NFS 1976].

Another major source of personnel exposure for maintenance workers was servicing the steam jets used to transfer recovered acid to the acid storage tanks. The jets suffered from high corrosion rates, prompting numerous entries into the Acid Recovery Pump Room (ARPR) to change them. This activity resulted in collective dose of 106 person-rem/yr for maintenance workers. Another significant source of personnel exposure for maintenance workers associated with the acid recovery system was

replacement of the heat exchangers in the Acid Recovery Cell (ARC). The heat exchangers became contaminated when seals failed, allowing contamination to enter the steam system. Repair and replacement of the heat exchangers resulted in 59 person-rem/yr of collective dose for maintenance workers [NFS 1976].

A cart mounted on a trolley was used for the transfer of samples of process solution between analytical cells. Repair and replacement of this cart after derailment was another source of personnel exposure for maintenance workers associated with the plants analytical areas. Cell entries would be required when the cart would derail, resulting in about 9 person-rem/yr associated with prejob decontamination of the cell and subsequent repair of the cart [NFS 1976].

Maintenance workers experienced high personnel exposures performing blower replacement activities in the Off Gas Blower Room (OGBR) and those associated with the Exhaust Cell Ventilation and HEV systems. These exposures were the result of the equipment being located in areas of elevated exposure rates. The exposure rate in the OGBR ranged from 1,000 mR/hr to 4,000 mR/hr from activity in the acid recovery system and the off gas treatment equipment. The VEC and HEV blowers were located in areas with a typical exposure rate of 100 mR/hr (from the filter banks) [NFS 1976].

Another source of personnel exposure for maintenance workers was repair or replacement of pumps in the UWA and LWA. The pumps were located in shielded locations adjacent to the process cells, and were used to transfer solutions within the cells.

The following is a chronology of radiation exposure conditions and events involving maintenance workers. A number of these involve skin and extremity exposures.

- *October 1965*. Maintenance was performed on a pump that had been used to empty a fuel cask. The pump read 11 R/hr and the exposure rate to the maintenance workers was a maximum of 1.2 R/hr [Loud 1965a].

- *March 1967*. “In general, the maintenance work in the Acid Recovery Cell necessitated exposures up to 1,000 mrem *to people not normal[ly] assigned to radiation work*” [emphasis added]. “These exposures began toward the end of March” [Keely 1967a, p. 2].

- *[Redacted]*. A [redacted] worker received an extremity exposure of [redacted] during work in the [redacted] [Wenstrand 1968a].

- *[Redacted]*. A [redacted] received an indicated [redacted] exposure to his hand as shown by a thermoluminescent dosimeter (TLD) worn as a finger dosimeter. Surveys of the work area concluded the exposure was from hot particles, as the maximum dose rate in the area was found to have been 16 rad/hr. The finger rings of [redacted] involved with the task showed [redacted] [Lewis 1968].

- *June 1968*. Installation of a new blower in the OGBR required 26 person-rem of collective dose. The typical working area exposure rate was 1 to 2 R/hr [Keely 1968a].

- *[Redacted]*. A [redacted] worker received an extremity exposure of [redacted] and a skin exposure [redacted] while working in the [redacted] (Keely 1968b). A different report says his skin exposure was [redacted], the [redacted] was to the front of his hand, and he received [redacted] to the back of his hand. His whole body exposure was [redacted] [Urbon 1968a, p. 4].

- *[Redacted]*. A [redacted] worker received an extremity exposure of [redacted] working in the [redacted] [Keely 1968c].

- *October 1968*. Changeout of the Dissolver Off Gas (DOG) blower in the OGBR required the use of 60 people and a collective dose of 48 rem whole body [Keely 1968d].

- *[Redacted]*. A [redacted] worker exceeded his quarterly extremity exposure limit while working in the [redacted]. He received a dose of [redacted] [Keely 1969a].

- *[Redacted]*. A [redacted] worker exceeded his quarterly extremity exposure limit when he received an exposure of [redacted] in the [redacted] [Keely 1969b].

- *[Redacted]*. A [redacted] worker received a skin exposure of [redacted] while changing a pump in the [redacted] [Keely 1969c, p. 2].

- *[Redacted]*. A [redacted] worker received an extremity exposure of [redacted] during repair work on the [redacted] [Keely 1969d].

- *[Redacted]*. {Redacted] received extremity exposures of [redacted] while working in the [redacted] [Keely 1969e].

- *[Redacted]*. [Redacted] "supplemental workers" received a collective [redacted] of dose changing a pump. NFS staff [redacted] received an additional [redacted] [Wenstrand 1971a].

NFS performed a review of 1970 and 1971 personnel exposure data for personnel involved in the "major work categories" of operations and maintenance. The average collective dose for operations and maintenance personnel over these 2 years was 985 person-rem and 1,002 person-rem, respectively. They assert incidents were minor contributors to these totals, which implies they were not biased by single events [NFS 1976].

Table 2-3 summarizes average annual collective dose for maintenance workers for typical tasks over the period 1970 to 1971 [NFS 1976]. All collective dose data in this site profile reflect external exposure only.

Table 2-3. Average annual collective dose for maintenance workers for typical tasks (person-rem/yr), 1970 to 1971.[a,b]

Plant area	Typical work performed	Collective dose
ARPR	Changing recovered acid jets	106
	Miscellaneous maintenance: replacing valves,	4
	instrument lines, etc.	2
	Valving	1
	Inspections	
ARC	Repair or replacement of 7C-3 heat exchanger	14
	Repair or replacement of 7C-5 heat exchanger	45
Analytical cells	Decontamination	5
	Repair, maintenance of transfer cart	4
CCR	Decontamination and shielding	53
	Repair or replacement of crane cable reels	40
	Repair or replacement of crane motors	27
	Repair PAR	20
	Replace or untangle crane cable	8
	Repair of misc. items	10
	Structural repair of crane	8
	Retrieval of disabled crane	2

Plant area	Typical work performed	Collective dose
EDR	Decontamination and cleanup	7
	Repair of CPC eductors, jumpers, etc.	10
	Repair of CPC PAR	5
GCR	Decontamination and shielding	106
	Repair or replacement of crane cable reels	70
	Repair of cell door hoist	32
	Retrieval of disabled crane	23
	Replace or untangle hoist cables	25
	Repair PAR	19
	Repair or replacement of crane motors	10
	Miscellaneous electrical repairs on crane	12
	Structural repair of crane	3
LWA	Changing pumps or valves	9
	Miscellaneous maintenance	3
	Valving and inspection	1
OGBR	Change blower	6
	Change off gas scrubber pumps	5
	Miscellaneous maintenance: replacing valves, instrument lines, etc.	5
HEV, PMCR	Replace blowers	3
	Decontamination and shielding	62
	Repair or replacement of crane cable reels	40
	Replace or untangle hoist cables	31
	Repair or replacement of crane motors	13
	Miscellaneous electrical repairs on crane	12
	PAR repairs	5
	Repair PMC cell equipment	2
	Structural repair of crane	2
UWA	Change pumps	12
	Miscellaneous maintenance	3
VEC	Maintenance and repair of blowers	7
	Miscellaneous maintenance	2
	Decontamination	6
VWR	Install new filters in washer	21
	Replacement and repair of pumps	9
MSM	Repair of manipulators	10
Plant	Miscellaneous maintenance in plant areas (operating aisles)	62
Total	**N/A**	**1,002**

a. Source: NFS [1976].
b. CCR – CPC Crane Room; PAR – Programmed and Remote Systems; PMCR – PMC Crane Room; VWR – Ventilation Wash Room.

Table 2-4 ranks the tasks that resulted in the highest exposures (collectively) to maintenance workers in 1970 to 1971 [NFS 1976].

Table 2-4. Tasks with the highest collective exposures to maintenance workers (person-rem), 1970 to 1971.[a,b]

Area	Task	Collective dose
ARPR	Changing recovered acid jets	106
GCR	Decontamination and shielding	106
GCR	Repair or replacement of crane cable reels	70
PMCR	Decontamination and shielding	62
Plant	Miscellaneous maintenance in operating aisles, etc.	62
CCR	Decontamination and shielding	53
ARC	Repair or replacement of 7C-5 heat exchanger	45
CCR	Repair or replacement of crane cable reels	40
PMCR	Repair or replacement of crane cable reels	40
GCR	Repair of cell door hoist	32
PMCR	Replace or untangle hoist cables	31
CCR	Repair or replacement of crane motors	27
GCR	Replace or untangle hoist cables	25
GCR	Retrieval of disabled crane	23
VWR	Install new filters in washer	21

a. Source: NFS [1976].
b. PMCR PMC Crane Room.

Tables 2-5 and 2-6 show collective dose data for pump changes for the period 1970 through 1971 for the UWA and LWA, respectively.

Table 2-5. UWA pump changeout data, 1970 and 1971.[a]

Date	Exposure rate (R/hr)	Collective dose (person-rem)
01/02/1970	2	0.30
03/26/1970	5	0.47
04/07/1970	10	2.46
05/13/1970	4	1.20
07/06/1970	15	1.44
07/20/1970	4	2.70
02/03/1971	5	2.67
03/31/1971	2	1.46
06/26/1971	7	0.91
07/15/1971	3–5	3.26
10/29/1971	5	5.78

a. Wenstrand [1974, p. 4].

Table 2-6. LWA pump changeout data, 1970 and 1971.[a]

Date	Exposure rate (R/hr)	Collective dose for changeout (person-rem)
03/23/1970	2	0.70
05/12/1970	3	1.20
05/20/1970	15	0.63
05/26/1970	10	0.88
06/16/1970	2	0.75
06/22/1970	2	0.73
06/30/1970	4	0.38
10/22/1970	7	1.81
11/25/1970	2	0.42
11/30/1970	10	2.85
06/07/1971	5	3.18
07/14/1971	3	1.05
08/03/1971	4	0.72

a. Wenstrand [1974, p. 5].

2.2.6.2 Operations

NFS' review of 1970 and 1971 personnel exposure data found the average collective dose for operations personnel over that period was 985 person-rem [NFS 1976]. NFS asserted incidents were minor contributors to the collective dose for operations workers (as well as maintenance workers). Table 2-7 shows the breakdown of collective dose to operations workers for typical tasks and duties [NFS 1976].

Table 2-8 shows dose rates and occupancy times for Operations workers for various areas of the plant for July 1971 [Wenstrand 1971e]. Table 2-9 shows a breakdown for analytical workers for the same period [Wenstrand 1971e].

Table 2-7. Average annual collective dose to operations workers for typical tasks and duties (person-rem), 1970 and 1971.[a,b]

Plant area	Typical work performed	General area	Ventilation systems	Acid recovery system	Contact maintenance areas	Totals
FRS	Fuel receiving, storage, transfer to PMC	250	No data	No data	No data	**250**
Labs	Sample analysis	50	90	No data	10	**150**
CR	Process surveillance	5	75	No data	No data	**80**
GOA	GPC operations	20	60	No data	No data	**80**
XCR	Mixing chemicals, valving, readings	12	8	60	No data	**80**
HEV	Filter changes	5	64	No data	No data	**69**
MOA	PMC operations	20	5	No data	25	**50**
Sample stations	Taking process samples	20	10	10	No data	**40**
SR	Decontamination of scrap casks and area	30	No data	No data	No data	**30**
VEC	Change filters	5	23	No data	No data	**28**
Various	Miscellaneous	27	No data	No data	No data	**27**
OGA	Chemical mixing, valving, readings	5	3	8	No data	**16**
OGBR	Change filters	No data	15	No data	No data	**15**
NDA	Waste burial	15	No data	No data	No data	**15**
NDA	Hull burial	10	No data	No data	No data	**10**
Offices	Supervision	No data	1	6	No data	**7**
COA	Valving, readings	4	1	No data	No data	**5**
PPS/PPH	Product loadout	5	No data	No data	No data	**5**
Yard	Interceptor, lagoons, etc.	5	No data	No data	No data	**5**
LLWT	Liquid waste treatment	4	No data	No data	No data	**4**
LXA	Valving readings	1	3	No data	No data	**4**
PEA	Readings, valving	2	No data	1	No data	**3**
WTF	Readings, valving	No data	3	No data	No data	**3**
CVA	CPC operations	2	No data	No data	No data	**2**
LWA	Valving	2	No data	No data	No data	**2**
UXA	Readings, valving	2	No data	No data	No data	**2**
PCR	Sampling, valving, readings	No data	No data	1	No data	**1**
Utility Room	Auxiliary services operation	1	No data	No data	No data	**1**
UWA	Valving	1	No data	No data	No data	**1**
Totals	Not applicable	**503**	**361**	**86**	**35**	**985**

a. Source: NFS [1976, p. 90].
b. CR – Control Room; CVA – Chemical Viewing Aisle; LLWT – Low-Level Waste Treatment; PCR – Process Chemical Room; PPH – Product Packaging and Handling; WTF – Waste Tank Farm.

Table 2-8. Occupancy times and dose rates for operations workers for various areas of the plant, July 1971.[a,b]

Area	Occupancy time (person-hr/shift)	Dose rate (mR/hr)	Annual collective dose (person-rem)
XCR	6	8	53
Control room	18	1.5	30
UXA	2	1	2
PCR	No data	100	No data
OGA	0.5	30	16
LXA	0.5	3	2
PPH/PPS	5	1	5
Sampling station	6	4	26
COA	3	1	3
WTF Shelter	0.5	5	3
LLLW	8	0.5	4
Interceptor	5	1	5
FRS	12	15	197
GOA	3	20	66
MOA	18	1.5	30
EDRVA	0.5	0.1	No data
CVA	8	0.1	1
Office building	7	1	7

a. Source: Wenstrand [1971e].
b. EDRVA – Equipment Decontamination Room Viewing Aisle; LLLW – Low-Level Liquid Waste; PCR – Process Chemical Room; PPH – Product Packaging and Handling; WTF – Waste Tank Farm.

Table 2-9. Occupancy times and dose rates for analytical workers, July 1971.[a]

Area	Occupancy time (person-hr/d)	Dose rate (mR/hr)	Annual collective dose (person-rem)
Quality Control Laboratory	12	1	5
Hot Laboratory	36	3	39
Shift office	18	4	26
Hot cells	6	4	9
Supervisor's office	3	5	5
Alpha Laboratory	3	3.5	4
Product Laboratory	15	2	11
Emission Laboratory	6	0.7	2
Mass Spectrometry Laboratory	18	0.7	5
Count Laboratory	9	3	10

a. Source: Wenstrand [1971e].

2.2.6.3 Ventilation System Filter Changeouts

A large contributor to personnel exposures within Operations duties was changeout of ventilation system filters. These included filters associated with the main ventilation system, the HEV system, the DOG system, and the Vessel Off Gas (VOG) system. The average annual collective dose associated with changing the filters (roughing and absolute) in the VEC was 23 person-rem for 1970 to 1971 [NFS 1976].

Tables 2-10 to 2-12 show personnel exposures associated with changeouts of main ventilation and HEV system filters during 1970 and 1971.

Table 2-10. Personnel exposure associated with main ventilation system filter changes (person-rem), 1970 and 1971.[a]

Change date	Collective dose (person-rem)
03/25/1970[b]	4.1
04/13/1970[b]	1.5
09/15/1970[b]	15.8
11/05/1970[b]	10.1
11/14/1970[b]	4.1
03/29/1971[c]	4.5
07/28/1971[c]	3.2
11/15/1971[c]	2.2

a. Source: NFS [1972a].
b. Before HEV operation.
c. After HEV operation.

Table 2-11. Collective dose for changeouts of HEV system prefilters (person-rem), 1970 and 1971.[a]

North train date	North train collective dose	South train date	South train collective dose
10/28/1970	0.2	10/29/1970	0.2
11/02/1970	0.2	11/02/1970	0.2
11/25/1970	0.65	11/28/1970	1.98
12/01/1970	1.23	12/05/1970	1.52
12/15/1970	4.25	12/23/1970	6.40
01/02/1970	1.96	01/22/1971	1.01
02/02/1971	1.40	02/24/1971	1.10
04/01/1971	4.70	04/15/1971	0.50
04/30/1971	1.92	05/18/1971	4.02
06/04/1971	0.75	06/22/1971	0.52
07/16/1971	4.17	08/04/1971	4.00
09/17/1971	1.30	09/08/1971	10.00
11/20/1971	1.37	10/07/1971	2.18
Not applicable	Not applicable	12/12/1971	4.58

a. Source: NFS [1972a].

Table 2-12. Collective dose for changeouts of HEV system roughing filters (person-rem), 1970 and 1971.[a]

Date	Collective dose
12/04/1970	2.29
12/14/1970	1.73
12/22/1970	2.76
03/01/1971	1.87
03/08/1971	2.15
06/21/1971	1.08
07/15/1971	3.20
10/14/1971	5.26

a. Source: NFS [1972a].

The November 1970 exposures are attributed to pre-HEV operation since the activity on the filters accumulated before the system was operating. The HEV system came online in late October 1970.

Additional information pertaining to filter changes follows.

- A memorandum dated December 9, 1966, says the plant ventilation system's roughing filters were changed "contact-wise in a radiation field of 5 to 10 R/hr" [Lewis 1966a].
- The VOG filter read 30 R/hr at 2 ft on November 2, 1967. TLD measurements showed 77 R/hr and 280 rad/hr at 18 in. [Wenstrand 1967a].
- The DOG filter read 200 R/hr at 2 in. and 30 R/hr at 2 ft on November 9, 1967 [Wenstrand 1967a].
- A DOG filter changed on 3/9/1968 read 1000 R/hr at 2 in. and 200 R/hr at 4 ft. The filter was placed in a lead cask that read 10 R/hr to 30 R/hr at 2 in. By March 25, 1968, the new filter read 500 R/hr at 6 in. and 50 R/hr at 4 ft. [Wenstrand 1968b].
- In May 1968 the typical exposure rate in the filter room of the VEC at the start of a filter change was 35 R/hr. Replacement of 30 each roughing and absolute filters required 16 person-rem of whole body dose [Keely 1968e].
- A DOG filter removed on November 27, 1968, read 1000 R/hr at 3 in. [Keely 1968f].
- Changeout of the main ventilation filters in November 1968 required "a total whole body exposure of 3 rem" [Keely 1968f].
- In January 1972 HEV prefilters were removed from the south plenum using a new method utilizing the roof of the Filter Change Room. Personnel dose was 1.2 rem to the whole body. The new method is said to virtually eliminate extremity and skin exposure problems [Wenstrand 1972b].

2.2.6.4 Overall Workforce

The *average* yearly whole body exposures for process, operations, and service employees working in or frequenting the normal access areas of the processing plant for 1968 through 1971 are as follows:

- 1968 = 2.74 rem/person,
- 1969 = 3.81 rem/person,
- 1970 = 6.76 rem/person, and
- 1971 = 7.15 rem/person.

These are average whole body exposures for individual workers. They are not collective dose [Miller 1972a]. The increase in 1970 and 1971 over previous years reflects the processing of higher burnup fuels, which resulted in significant increases in personnel exposure throughout the site.

The average weekly whole body exposure per person for 1967 for various responsibilities was as follows [Wenstrand 1967b]:

- Head End Operations = 115 mR/wk,

- Chemical Process Operations = 195 mR/wk,
- Maintenance = 110 mR/wk, and
- Health & Safety = 80 mR/wk.

Based on these data, NFS made the following projections for annual exposures per person in 1968 under the assumption that the weekly exposure date for 1967 did not change:

- Head End Operations = 5.8 rem/yr,
- Chemical Process Operations = 9.8 rem/yr,
- Maintenance = 5.5 rem/yr, and
- Health & Safety = 4.0 rem/yr.

Tables 2-13 through 2-17 show collective whole body, skin, and extremity dose for various work groups for 1969 through 1971.

Table 2-13. Collective whole body, skin, and extremity dose for various work groups (person-rem), 1967.[a]

Department	Number of workers	Whole body collective dose	Skin collective dose	Extremity collective dose
Analytical	28	66	175	176
Health & Safety	9	28	68	88
Maintenance	26	77	297	748
Production: Chemical	38	181	324	418
Production: Mechanical	29	77	243	418
Overhead + others	79	121	243	352
Total	**209**	**550**	**1,350**	**2,200**

a. Source: Wenstrand [1971b, p. 7].

Table 2-14. Collective whole body, skin, and extremity dose for various work groups (person-rem), 1968.[a]

Department	Number of workers	Whole body collective dose	Skin collective dose	Extremity collective dose
Analytical	34	77	130	331
Health & Safety	14	62	116	156
Maintenance	38	135	251	823
Production	107	368	503	1,268
Overhead	71	81	110	243
Total	**264**	**723**	**1,110**	**2,821**
Benz + non-NFS	**106**	**127**	**210**	**579**

a. Source: Wenstrand [1971b, p. 8].

Table 2-15. Collective whole body, skin, and extremity dose for various work groups (person-rem), 1969.[a]

Department	Number of workers	Whole body collective dose	Skin collective dose	Extremity collective dose
Analytical	29	122	164	763
Health & Safety	16	63	129	182
Maintenance	36	148	320	835
Production	74	404	727	1,649
Overhead	70	122	145	323
Total	**225**	**859**	**1,485**	**3,752**
Benz + non-NFS	**82**	**109**	**153**	**251**

a. Source: Wenstrand [1971b, p. 9].

Table 2-16. Collective whole body, skin, and extremity dose for various work groups (person-rem), 1970.[a]

Department	Number of workers	Whole body collective dose	Skin collective dose	Extremity collective dose
Analytical	24	165	173	572
Health & Safety	14	108	230	374
Maintenance	34	204	358	837
Operations	61	532	693	1,459
Overhead	24	46	74	150
Tech. Services	6	43	59	109
Total (NFS)	**163**	**1,098**	**1,587**	**3,501**
Benz + non-NFS	**472**	**433**	**1,043**	**2,630**

a. Source: Wenstrand [1971b, p. 10].

Table 2-17. Collective whole body, skin, and extremity dose for various work groups (person-rem), 1971.[a]

Department	Number of workers	Whole body collective dose	Skin collective dose	Extremity collective dose
Analytical	25	146	154	544
Health & Safety	15	116	236	361
Maintenance	34	248	517	910
Operations	66	589	900	1,665
Overhead	15	41	57	109
Tech. Services	7	32	47	78
Total (NFS)	**162**	**1,172**	**1,911**	**3,667**
Benz	**991**	**1,194**	**2,677**	**4,690**

a. Source: Wenstrand [1972c, p. 4].

Table 2-18 shows typical exposures per person for various job duties for 1970. The data are typical whole body, skin, and extremity dose per individual. They are not collective dose. The exposure data are for NFS employees. Exposures for contract workers are not included.

Table 2-18. Typical annual exposures per person for various job duties (rem), 1970.[a]

Job classification	Whole body	Skin	Extremities
Analytical: Process control supervisors	7.3	7.9	26
Analytical: Process control analysts	8.6	9.4	40
Analytical: Special laboratory supervisors	5.8	6.0	14
Analytical: Special laboratory analysts	6.3	6.4	15
Health & Safety: Supervisors	8.7	12	23
Health & Safety: A technicians	7.9	20	28
Health & Safety: B, C technicians	7.2	16	16
Maintenance: Management	8.9	15	52
Maintenance: Supervisors	8.2	15	32
Maintenance: Electricians	6.6	13	27
Maintenance: Instrument technicians	7.7	11	16
Maintenance: Mechanics	6.9	11	32
Operations: Management	8.6	10	24
Operations: Supervisors	10.5	15	32
Operations: A operators	8.5	11	21
Operations: B, C operators	7.6	10	21

a. Source: Wenstrand [1971b, p. 5].

Table 2-19 shows typical exposures per person for various job duties for 1971. The data are typical whole body, skin, and extremity dose per individual. They are not collective dose. Exposures for contract workers are not included.

Table 2-19. Typical exposures per person for various job duties (rem), 1971.[a]

Job classification	Whole body	Skin	Extremities
Analytical: Supervisors	9.2	5.6	11.6
Analytical: Process control	6.8	7.6	31.6
Analytical: Spec. labs	4.4	4.4	13.2
Health & safety: Supervisors	8.0	18.8	34.8
Health & safety: Technicians	7.6	16.0	24.0
Maintenance: Management	8.4	15.2	20.0
Maintenance: Supervisors	7.2	17.2	26.8
Maintenance: Electricians	7.2	18.0	27.2
Maintenance: Instrument technicians	8.0	11.6	13.6
Maintenance: Mechanics	8.0	16.8	34.8
Operations: Management	6.8	8.8	18.0
Operations: Supervisors	8.8	16.8	27.2
Operations: A operators	9.6	13.2	24.8
Operations: B, C operators	8.8	13.2	26.0

a. Source: Wenstrand [1972c, p. 5].

Table 2-20 shows exposure rate data for the processing plant as of March 15, 1972 (after shutdown). From review of the March 1972 report [Wenstrand 1972a] and a follow-up report from July of that year [Wenstrand 1972d], it appears sometimes the exposure rates given are for the source itself, and other times they are general exposure rates in the affected area. Exposure rates that appear to be for a specific source are in bold type. These distinctions were drawn by comparing the March 1972 survey report with the follow-up report from July 1972.

Table 2-20. Significant radiation sources in the reprocessing plant (mR/hr), March 15, 1972.[a,b]

Plant level and radiation source	Exposure rate	Affected areas
Below grade - Contamination around leaking window	20	GOA
Below grade - HEV Duct	10	GOA
100-ft elevation - FRS	10–20	EMOA
100-ft elevation - MRR	**100[c]**	EMOA
100-ft elevation - HEV Room	10	MSM Shop
100-ft elevation - Piping in window areas	**100[c]**	WMOA
100-ft elevation - Contamination around shear	2.5	No data
100-ft elevation - Contamination in southwest stairwell	10	Showers and locker room
114.5-ft elevation - Ventilation duct	**50[c]**	LXA
114.5-ft elevation - Contaminated piping and general contamination around the C-1 Sampler	10	COA
114.5-ft elevation - Contaminated piping in the COA	**50[c]**	COA
114.5-ft elevation - Ventilation washer	5	LXA
131-ft elevation - Ventilation duct	5–10	Laboratory area
131-ft elevation - Ventilation washer	10	Laboratory area
131-ft elevation - PMCR Penthouse	5	Laboratory area
131-ft elevation - Drain in hot laboratory	**10[c]**	No data
144-ft elevation - Ventilation duct	2.5	Control Room
144-ft elevation - Hot piping in UXA	**20[c]**	UXA
144-ft elevation - Ventilation filter bank	100	UXA
144-ft elevation - Hot acid cell	10	PCR
160-ft elevation - Hot acid tanks, lines, and floor contamination	1,000	XCR
160-ft elevation - Pump pot niche in the PEA	100	XCR
160-ft elevation - Ventilation duct	50	XCR
160-ft elevation - Hot acid tanks	**1,000[c]**	XCR
Waste Tank Farm - Ventilation line	**500[c]**	No data

a. Sources: Wenstrand [1972a,d].
b. EMOA – East Mechanical Operating Aisle; MRR – Manipulator Repair Room; PCR – Process Chemical Room; PMCR – PMC Crane Room; WMOA – West Mechanical Operating Aisle.
c. Exposure rate that appears to be for a specific source.

Table 2-21 shows exposure rate data for various locations extracted from a health physics logbook for a period between July and December 1973 [NFS 1974c]. More detailed information on exposure rates in a given area can be found in Attachment A.

Table 2-21. Exposure rate data for various locations extracted from a health physics logbook (mR/hr), July to December 1973.[a,b]

Area	Period	Exposure range low	Exposure range high
Analytical aisle	07/09–12/27	1–4	2–12
Analytical aisle–north	07/11–12/27	1–5	1–35
Counting room	08/03–12/27	1–2	1–6
Analytical supervisor's office	07/11–12/27	2–10	5–40
Hot laboratory	07/11–12/27	1	1–8
Emission spectrometry laboratory	07/11–12/27	<1	1
Mass spectrometry laboratory	07/11–12/27	<1	1.5
Product laboratory	07/11–12/27	1	1–30[c]
Quality Control Laboratory	07/11–12/27	1	1–13[d]

Area	Period	Exposure range low	Exposure range high
COA	07/09–12/31	1–3	1–10
CVA	07/09–12/31	1–18	1–50
Control room	07/09–12/31	1–3	2–6
EDR viewing aisle	07/09–12/31	<1	<1
LXA	07/09–12/31	1–20	2–35
FRS	07/15–12/27	2–10	5–60
GOA	07/10–12/29	2–6	2–20
Instrument shop	07/10–12/26	1–2	1–2
Laundry	07/10–12/29	0.5–1	1–3
EMOA	07/09–12/27	1–3	2–10
WMOA	07/09–12/27	1–3	1–15
MSM Shop	07/10–12/29	1–3	2–20
Personnel decontamination room	07/11–12/27	<1	<1
Stairs–east	07/10–12/31	1–2	1–5
Stairs–north	07/09–12/27	1–5	3–15
Ventilation supply room	07/10–12/31	1–5	1–20
UXA	07/09–12/31	1–15	3–30
LLWT building	07/11–12/12	1–5	1–18
WTF shelter	07/12–12/31	5–225	40–500
WTF: over tanks	07/12–12/31	<1–2	1–3
XCR	07/18–12/17	1–10	1–30
OGA	09/21–10/30	10–20	1,000[e]
Pulser equipment aisle	09/21	10–300	10–300[f]
PCR	07/11–10/30	1–5	1–10[g]
PSC-1	07/11	1–5	1–5
PSC-2	07/11–10/30	2–5	3–5
PSC-3	07/11–10/30	5–45	5–50
PPS	07/18–10/24	1–4	1–6
PPH	07/18–10/24	1–2	1–3
RER	08/06–10/24	1–35	1–50
Rooftops	09/18–12/26	<1–100	1–500
South stairs	Unknown–10/28	1–8	2–15[h]
ULO	08/18–10/24	1	1–2
Utility room	08/11–10/24	<1–2	<1–2
NDA	10/24	0.3–0.5	0.3–0.5[i]
XSA	07/11–11/3	1–6	1–10

a. Source: NFS [1974c].
b. LLWT – Low-Level Waste Treatment; PCR – Process Chemical Room; PPH – Product Packaging and Handling; ULO – Uranium Loadout Area; WMOA – West Mechanical Operating Aisle; WTF – Waste Tank Farm.
c. Maximum at hood lip.
d. Maximum at south hood.
e. Maximum was at NW corner; says 2–80 mR/hr "@ rope."
f. Says "west niche" open [only one measurement recorded for 1973]
g. "Area being cleaned."
h. First entry not dated.
i. Only one measurement recorded for 1973.

2.3 REPROCESSING PLANT RESIDUAL RADIATION PERIOD, 1974 TO FEBRUARY 25, 1982

This period is characterized by miscellaneous decontamination and plant improvement work. The level of activity was not as great as the period between shutdown in March 1972 and May 1973.

Activities at the plant consisted of minor maintenance, decontamination projects, and laboratory analyses, including some for NFS-Erwin [Nelson 1976, p. 10]. As of May 1975 they still received three

to five fuel shipments per week, usually by truck in NFS shipping casks. Railroad shipments were rarely made by this time. There was approximately 115 t of fuel in the SP, which was about 45% full [Johnson & Higgins 1975].

A May 12, 1975, Nuclear Insurance Survey report by Johnson & Higgins states the highest level of personnel exposure at West Valley was around 400 mrem/quarter received by the fuel unloading operators, and that maintenance workers received about 200 mrem/quarter [Johnson & Higgins 1975].

The DOG system went inoperative in November 1975. The VOG system remained in use to ventilate gases from the vessels and tanks in the process building [WVNSC 1994].

Fuel receipts had ceased by 1980, though the plant was still storing spent fuel [Duckworth 1980]. The fuel remained in storage until the early 1980s when it was removed by the DOE.

Table 2-22 shows exposure rate data for various locations extracted from a health physics logbook for a period between January and October 1974 [NFS 1974c].

Table 2-22. Exposure rate data for various locations from a health physics logbook (mR/hr), January to October 1974.[a,b]

Area	Period	Exposure range low	Exposure range high
Analytical aisle	01/04–10/20	1–7	2–10
Analytical aisle–north	01/04–10/20	1	1–15[c]
Counting room	01/04–10/20	1	1–4
Analytical supervisor's office	01/04–10/20	5–20	5–40
Hot Laboratory	01/04–10/20	1	1–4
Emission Spectrometry Laboratory	01/04–10/20	1	1
Mass Spectrometry Laboratory	01/04–10/20	<1	1
Product Laboratory	01/04–10/20	1	1–8
Quality Control Laboratory	01/04–10/20	1	1–2
COA	01/10–10/18	1–3	1–5
CVA	01/10–10/18	1–15	1–60[d]
Control room	01/10–10/18	1–3	1–6
EDR viewing aisle	01/10–10/18	<1	<1
LXA	01/10–10/18	1–35	1–65
FRS	01/03–10/18	5–20	5–35
GOA	01/03–10/18	2–7	2–20
Instrument shop	01/17–10/02	1	1–2
Laundry	01/02–10/18	<1	1
EMOA	01/03–10/18	1–4	1–10
WMOA	01/03–10/18	1–2	1–5
MSM shop	01/03–10/18	2–8	2–11
Personnel decontamination room	01/02–10/20	<1	1
Stairs–east	01/10–10/18	<1–3	1–4
Stairs–north	01/03–10/18	1–4	1–8
Ventilation supply room	01/10–10/18	1–12	1–20
UXA	01/10–10/18	1–20	1–30
UWA	05/10–10/18	1–15	1–20
LLWT building	01/13–10/31	<1	<1–5
WTF shelter	01/13–10/31	5–200	30–300
WTF: over tanks	01/13–10/31	<1–3	1–5
XCR	01/13–10/20	1–5	1–15
OGA	01/15–07/30	10	1,000[e]

Area	Period	Exposure range low	Exposure range high
Pulser equipment aisle	01/15–10/1	2–20	2–25
PCR	01/15–07/30	1–3	1–5
PSC-1	01/15–07/30	1–5	4–8
PSC-2	01/15–07/30	3–10	5–15
PSC-3	01/15–07/30	1–3	5–20
PPS	01/11–07/30	1	1–5
PPH	01/11–07/30	<1	1–2
RER	01/11–07/30	10–20	2–60
Rooftops	05/14–08/21	<1–100	<1–100
South stairs	05/14–07/30	1–3	2–5
ULO	01/11–07/30	<1	1
Utility room	02/12–07/30	1–2	1–3
NDA	02/12–09/30	0.3–0.5	0.5–1.5
XSA	01/15–07/30	1–5	1–8
Old instrument shop	05/20–07/30	20–75	50–100
VEC	06/03–07/30	5–250	5–250[f]
Southwest stairs	07/30	2–45	(g)

a. Source: NFS [1974c].
b. LLWT – Low-Level Waste Treatment; PCR – Process Chemical Room; ULO – Uranium Loadout Area; WMOA – West Mechanical Operating Aisle; WTF – Waste Tank Farm.
c. From 06/02/1974 forward, 20 mR/hr "in paint on PMC door".
d. 10/18/1974 entry says 1–4 mR/hr, but this is an outlier with respect to the other entries, which are reasonably consistent.
e. Maximum was at NW corner.
f. Start date might be 07/18.
g. Only one measurement recorded for 1974.

Table 2-23 shows exposure rate data for later in the residual radiation period. The data are cited by Golden et al. [1982].

Table 2-23. Exposure rate data for later in the residual radiation period (mR/hr).[a]

Area	Exposure rate	Date
Extraction chemical room	5	About 1981
Extraction cell 1	11,300	10/1978
Extraction cell 2	90	04/1978
Extraction cell 3	12	05/1978
Product purification cell	100	Not given
ARPR	500 to 1000	03/1976
Liquid waste cell	150 to 1000	03/1976
CPC	12,000 to 32,000	About 1981
Process mechanical cell	600,000 (maximum)	11/06/1978
GPC	650,000 (maximum)	11/06/1978
Miscellaneous areas	2 to 10	05/221980
ARC	100 to 150	02/1974
OGA	1 to 2	09/1980
Upper warm aisle	12 to 90	05/1981
LWA	5 to 40	04/16/1974
COA	10 to 20; 300 maximum	06/1981
LXA	10	09/18/1972
Upper extraction aisle	2.5 to 60	05/1981
MOA	0.1 to 10	05/1981
GCR	50 to 75	03/1978
Pulser equipment aisle	2 to 10	05/22/1980

Area	Exposure rate	Date
Southwest stairway	500 to 1000	Not given
EDR	10 to 100	1977
Scrap removal	1 to 10	1980

a. Source: Golden et al. [1982].

2.4 REPROCESSING PLANT U.S. DEPARTMENT OF ENERGY PERIOD, FEBRUARY 26, 1982, TO PRESENT

DOE took over operational control of the processing plant site on February 26, 1982 [WVNSC 1983]. This included the high-level burial area [4]. Responsibility for the LLW burial area transferred to the State of New York in 1983 [NYSERDA 2004].

Decontamination of the main reprocessing plant took place in the early 1980's to allow for utilization of the existing facilities for the HLW vitrification project to the extent practical. Decontamination and fuel handling activities in the early years of the WVDP resulted in higher personnel doses than experienced later in the project [Hoffman 1997].

As of 1988, the following areas of the main plant remained contaminated and represented sources of potential personnel intakes [Johnson 1991]:

- CPC,
- PMC,
- GPC,
- Extraction Cell 1 (XC1),
- XC2,
- Hot Acid Cell,
- Plutonium Product Handling, and
- Solvent Storage Terrace.

Inspections performed in 1986 found that much equipment in the head end cells (GPC and PMC) had fallen into disrepair. Cranes were not reliable, crane room doors were inoperable, and power manipulators and MSMs were inoperable or inaccessible [Vance 1986]. Exposure rate surveys inside the cells found general levels ranged from 40 R/hr to 340 R/hr at a height of 6 ft. Hot spots were found as high as 2,050 R/hr. Solids and liquids (in sump areas) were sampled, showing high levels of ^{137}Cs and ^{60}Co, plus fissile and transuranic (TRU) elements [Vance 1986].

The canisters of vitrified waste are stored in the CPC. An individual canister reads around 2,300 rem/hr at contact [Petkus et al. 2003]. The logs of vitrified material read around 5,000 R/hr before being loaded in the canisters. High backgrounds are still a confounding factor with respect to radiation contamination surveys, though not to the extent they were when the plant was operating [5].

2.5 STATE-LICENSED COMMERCIAL WASTE DISPOSAL AREA, 1963 TO 1983

The SDA occupies approximately 15 acres of the Western New York Nuclear Service Center immediately adjacent to the 175 acres DOE is using for the WVDP. NFS opened the SDA in 1963 and

continued to operate it until 1975, when disposal operations were terminated as a result of problems managing water that infiltrated the disposal trenches. Approximately 2.4 million ft^3 of LLW were disposed of in the SDA. This included material such as dry active waste from the NFS reprocessing plant. NYSERDA currently has responsibility for the monitoring and maintenance of the SDA [NYSERDA 2004].

During operations, packaged waste was placed in long trenches excavated in the soil that were then backfilled. Water from rain and snowmelt was able to infiltrate the trenches through cracks and discontinuities in the trench covers and upper levels of the soil. Because of the highly impermeable nature of the soil surrounding the trenches, water accumulated in them and became contaminated from contact with the wastes. In early 1975, this accumulation, coupled with the site operator's (NFS) inability to gain regulatory approval to remove, treat, and discharge the trench water on a controlled basis to an adjacent stream, led to uncontrolled seepage of contaminated water from some of the trenches. Waste burial operations promptly ceased and the SDA has remained shut down ever since [NYSERDA 2004].

Between 1975 and 1981, NFS pumped water out of the trenches several times and treated it before release to the adjacent stream. Redesigning and reworking of the covers reduced, but did not eliminate, water accumulation in the trenches. Water management problems continued to be a concern at the SDA. In 1983, NYSERDA assumed management responsibility for the SDA from NFS [NYSERDA 2004].

In June 1973 the New York State Department of Environmental Conservation published a study of the waste inventory buried in the SDA and the NDA [Kelleher and Michael 1973]. It said the greatest volume of waste came from waste disposal firms, most of which was associated with hospital, institutional and industrial activities. The use of the site by nuclear power plants was less than 10% of the total volume, though the authors noted the usage by nuclear power was increasing. ^{238}Pu was asserted to be the nuclide of greatest concern in terms of its quantity and toxicity. The radionuclide present in the greatest quantity in terms of activity was tritium.

The SDA was closed March 8, 1975, upon discovery of tritium and ^{90}Sr in surface waters from leakage from the SDA. Studies of water pumped from the trenches in 1973 showed positive indications of gross beta and gross alpha activity [Duckworth 1976]. A February 1974 report on radionuclide analyses on water from SDA trenches showed the following radionuclides: tritium, ^{90}Sr, ^{129}I, ^{125}I, ^{137}Cs, ^{134}Cs, ^{106}Ru, ^{54}Mn, ^{57}Co, ^{60}Co, ^{22}Na, ^{75}Se [NFS 1974d].

Tables 2-24 and 2-25 show estimated quantities of radioactive materials buried in the SDA through 1975. The authors did not decay-correct their estimates, which date back to 1963. The data are highly uncertain, as they were based upon assumptions of nuclide mix, waste volume ratios, etc. They are presented to give users an idea of the isotopes present in the SDA trenches more so than quantities or isotopic ratios. It is not clear what is meant by “Misc.” – other than miscellaneous – or why they lumped together ^{125}I with ^{226}Ra and ^{241}Am [Duckworth 1978a, p. 3].

Table 2-24. Estimated quantities of special nuclear material and source material in the SDA through 1975.[a]

Nuclide	Type	Mass	Activity (Ci)
Th-232	Source	15,399 lb	0.769
U-238 + natural uranium	Source	1,010,594 lb	151.590
Pu-238	Special nuclear material	2,136 g	37,339
Pu-239	Special nuclear material	2,018 g	123.906
U-235	Special nuclear material	51,202 g	0.103
U-233	Special nuclear material	255 g	2.422

a. Source: Greco [1978].

Table 2-25. Estimated radioactivity content in the SDA through 1975 (Ci).[a]

Tritium	C-14	Co-60	Cs-137	Sr-90	I-125, Ra-226, Am-241	Ru-106	"Misc."	Pu-238
171,000	1,200	227,000	12,000	29,000	500	3,000	232,000	35,000

a. Source: Duckworth [1978a, p. 3].

A review of solid waste disposal at the West Valley site through 1977 was generated as part of a Safety Analysis Report update in 1978. It showed a total of 2,349,000 ft^3 of waste totaling 710,600 Ci of activity had been buried in the SDA from 1963 through 1975. This included 314,400 ft^3 of waste from the reprocessing plant itself. A "special burial" of reactor parts took place at the SDA in 1973, involving an estimated 330,000 Ci of activity [Duckworth 1978b, pp. 3–4].

Radiological Data for the SDA

Three containers of HLW were received in September 1965 having a maximum exposure rate of 45 R/hr. The maximum exposure rate for the waste handlers was 200 mR/hr [Loud 1965b, p. 2].

Two containers of HLW were received in October 1965 showing a maximum exposure rate of 11 R/hr. The maximum exposure rate to waste handlers was 500 mR/hr. Twelve containers shipped as LLW were received during the month, showing exposure rates from 220 mR/hr to 750 mR/hr [Loud 1965a].

Four shipments of HLW were received in November 1965 with the maximum exposure rate from the containers being 10 R/hr. The maximum personnel exposure rate was 1.5 R/hr to individuals rigging one of the containers [Loud 1965c].

Seven shipments of HLW were received for disposal in December 1965. The highest exposure rate of those containers was 70 R/hr. The maximum personnel exposure rate was 3 R/hr to individuals performing rigging for one of the containers. On [redacted], the working area exposure rate for a [redacted] moving HLW from storage into a trench was [redacted]. The exposure rate for workers involved in removing HLW containers from transportation casks on December 31, 1965, was a maximum of 250 mR/hr. The exposure rate from the containers was a maximum of 5 R/hr [Loud 1966a].

High-level waste received for disposal in February 1966 showed exposure rates up to 170 R/hr. Personnel exposure rates for individuals working with these containers were up to 800 mR/hr. The fence around the HLW "compound" had to be relocated due to increasing exposure rates. The exposure rate through the concrete wall around the compound reached 190 mR/hr. The exposure rate at the barbed wire fence was less than 100 mR/hr, and at the chain link (outer fence) it reached 1 mR/hr [Loud 1966b, p. 2].

A container of HLW (received for disposal) in March 1966 read 240 R/hr, which exceeded NFS' license limit of 200 R/hr. They received permission to accept the container from the New York State Department of Labor [Loud 1966c].

2.6 HIGH-LEVEL WASTE DISPOSAL AREA, 1966 TO PRESENT

During the AEC/NRC-licensed period, material was buried at the NDA in holes (shafts) instead of trenches. Holes were dug 3 ft wide by 7 ft long by 50 ft deep. Waste containers were placed in the holes up to 6 to 8 ft below the ground level. They were then backfilled to a thickness of 4 ft above the original grade. Additional cover was added if necessary until a reading made with a Geiger-Müller survey instrument could not be distinguished from background. The holes were placed at least 6 ft apart. Equipment that was too large to fit in a standard-size hole was buried ad hoc. The principal material buried at the NDA was cladding hulls. However, a wide variety of other materials and items

were also disposed of in the NDA, including ruptured irradiated fuel assemblies [Kelleher and Michael 1973, pp. 9–10].

The review of solid waste disposal at the West Valley site through 1977 showed a total of 139,500 ft^3 of material was buried at the NDA, totaling 551,300 Ci of activity [Duckworth 1978b, p. 3]. Of this material, 6,400 ft^3 was hulls (pieces of cladding left over after the fuel was leached out), 130,200 ft^3 was miscellaneous items such as filters and equipment, and 2,900 ft^3 was designated as "Under 200 mR" [Duckworth 1978b, p. 4].

The WVDP built a Liquid Pre-Treatment System on the northeast corner of the NDA. The purpose of the system is to remove organics (solvent) from groundwater around the NDA. As of 2004 the system had never been used [WVDP 2005].

Radiological Data for the NDA
As of early 1968 hulls and scrap from the head end cells of the process plant were transferred for burial at least once a week [Runion 1968, p. 22]. The burial area routinely exceeded 100 mR/hr in the area occupied by the remote crane operator, who was located in a shielded cubicle 100 ft from the crane. Additional shielding was supposed to have been installed by June 1, 1968, to ensure that the crane operator was not exposed to exposure rates in excess of 300 mR/hr. As of July 24, 1968, a lead glass window had been installed at the NDA to reduce the exposure to the crane operator. The shielding was said to have reduced the exposure rate from the hulls to "about 2 mR/hr," which is said to be indistinguishable from background [Clark 1968a, p. 2].

In July 1969 a dissolver removed from the CPC read 30 R/hr at contact with its shield cask and 50 mR/hr at 50 ft. Fluid leaked from the dissolver while transferring it to the burial area, contaminating the road surface to 50 rad/hr. Cleanup involved scrubbing and removal of some of the surface [Keely 1969e].

In November 1971 a spill contaminated several acres of ground north of the HLW burial area. Levels of 1 mR/hr to 3 mR/hr required extensive cleanup [Wenstrand 1971c, p. 3].

On June 26, 1972, cement blocks were removed from the OGA for burial. They were wrapped in plastic and then brought through the roof of the OGA where they were transferred from the roof to the "Red Stake Truck." The blocks read 5 R/hr (general area), and were described as grossly contaminated by NFS staff, having been used to shield an acid recovery line in the OGA. Contact dose rate measurements showed 15 R/hr, 40 R/hr, and 250 R/hr for whole body, skin of whole body, and extremities, respectively. The self-reading dosimeter of the worker who unloaded the blocks showed [redacted] for the unloading operation. Afterward, the individual was found to be contaminated, and a [redacted] showed [redacted] and [redacted]. Chemical analysis of the [redacted] showed [redacted] and [redacted] [Nelson 1973a, p. 17]. Estimates from whole body counts were an exposure of [redacted] "of combined concentrations" [redacted] and [redacted].

[Redacted].

2.7 ISOTOPIC INFORMATION FOR PLANT AREAS

2.7.1 Operations and Residual Radiation Periods, 1966 to February 25, 1982

Little radionuclide-specific information is available for surface or airborne contamination encountered in the processing plant before the characterization activities conducted under the WVDP. When the plant was operating, alpha contamination was assumed to be ^{239}Pu and beta was treated as ^{90}Sr for radiation protection purposes [Clark 1968b]. In general, contamination within the process building can be characterized as being associated with product streams, feed streams, or waste streams [6].

Available nuclide mix information for product streams show that the plutonium produced at West Valley was in general approximately 75% ^{239}Pu by weight (Table 2-26). In terms of alpha activity, the data show that, in general, the dominant nuclide was ^{238}Pu [NFS 1969; Helgeson Nuclear Services (HNS) 1971]. This is consistent with the reactor-grade plutonium being separated. An exception is the mix asserted for the plutonium involved in a puncture wound incident in January 1972. This material was predominantly ^{239}Pu in terms of both mass and alpha activity [Wenstrand 1972e].

Table 2-26. Isotopic mix data for West Valley plutonium product (%).[a]

Isotope	05/30/1969 weight fraction	05/30/1969 alpha activity fraction	1971 weight fraction	1971 alpha activity fraction	01/24/1972[b] weight fraction	01/24/1972[b] alpha activity fraction
Pu-238	0.7	59.9	0.6	58.1	0.03	6.0
Pu-239	76.8	24.1	76.1	24.7	90.7	69.9
Pu-240	13.7	15.9	14.1	17.1	8.4	24.1
Pu-241	7.7	0.1	7.9	0.1	0.8	0.02
Pu-242	1.1	0.02	1.2	0.02	0.1	0.00

a. Source: Wenstrand TK [1972e].
b. From a sampling needle responsible for a puncture wound.

Plutonium associated with head end operating areas (those upstream of the extraction cells) would likely have been in a metal or oxide form (absorption Type S) [7]. Plutonium associated with the process stream, such as would have been encountered in the process, analytical, and product handling areas, would have been in a nitrate form (absorption Type M) unless it was known to have been aged contamination [8]. This freshly separated plutonium would have had very little ^{241}Am [9]. If present, ^{241}Am should be considered as the same absorption type as the plutonium matrix in which it has ingrown. In other words americium should be treated as absorption Type S if it is associated with Type S plutonium.

If the chemical form of plutonium is unknown then either Type M or S should be selected based on which type maximizes dose to the organ of concern. Bioassay data should be used to determine absorption type in cases where sufficient data exist to establish excretion or retention. Because the plutonium isotopic mixture encountered at West Valley was predominantly ^{238}Pu on an activity basis, the Super S absorption type should not be applied [Oak Ridge Associated Universities (ORAU) Team (ORAUT) 2020]. This approach should be applied for West Valley plutonium cases over the entire time period of concern (see Section 5.5).

Alpha contamination from process areas upstream of the extraction cells (in terms of process flow) would have included other alpha species representative of irradiated nuclear fuel. These too would have been in metal or oxide form [10].

As of December 1966 approximately 11,000 kg of commercial reactor fuel had been processed. The first 1,190 kg of uranium produced from this fuel contained 24 ppb plutonium, which exceeded NFS' specification. NFS planned to blend this material with other uranium for which the plutonium content was within specifications to get an acceptable mixture. In addition, all of the plutonium product recovered the week ending December 16, 1966, exceeded specifications for fission product contamination [Lewis 1966b].

From November 15, 1968, through January 20, 1969, the processing plant ran a THOREX fuel cycle to reprocess thorium-uranium fuel from Indian Point Unit 1. Thus, during this time thorium, thorium decay products, and ^{233}U would have been encountered in the processing plant in addition to the other nuclides normally encountered. There were also higher than usual amounts of ^{235}U and ^{238}Pu present during this campaign [Birchler 1970].

Beta contamination at West Valley would have been a mixture of fission and activation products representative of irradiated nuclear fuel. The nuclides present would have included tritium, ^{58}Co, ^{60}Co, ^{90}Sr/Y, ^{95}Zr/Nb, ^{99}Tc, ^{106}Ru/Rh, ^{125}Sb, ^{134}Cs, ^{137}Cs, and ^{141,144}Ce [11]. In 1966, the principal gamma-emitters in the FRS pool water were found to be ^{58}Co, ^{60}Co, and ^{137}Cs [Loud 1966b]. In May 1972 isotopic measurements of FRS pool water included concentration data for ^{134}Cs, ^{137}Cs, ^{106}Ru/Rh, ^{144}Ce/Pr and ^{125}Sb [Jaroszeski 1972]. Isotopes of ruthenium, rhodium, zirconium, and niobium were large contributors to the radioactivity in the acid recovery system [NFS 1976]. ^{99}Tc was prominent in the Uranium Product Cell (UPC) [WVDP 2005].

Keely and Wenstrand [1971] state "entrainment" (resuspension) factors for radioactive particles found in "direct-maintenance" work areas in the plant ranged from 5×10^{-6}/m to 1×10^{-3}/m depending on work activity and airflow. The work areas they refer to included those adjacent to process cells, in analytical hot cells, in sample enclosures, solution storage areas, filter housings, and ventilation ducts. Resuspension factors were used in conjunction with smear surveys to predict work area air concentrations.

2.7.2 U.S. Department of Energy Period, February 26, 1982, to Present

One element of the WVDP was extensive characterization of plant areas and waste products. Characterization of HLW performed in 1986 determined the following radionuclides of concern in Table 2-27 [Johnson 1991].

Table 2-27. Radionuclides of concern, February 26, 1982 to present.[a]

Radionuclide	Chemical form
Am-241	All forms
Co-60	$Co(NO_3)_2$
Cs-137	All forms
Pu-238	PuO_2
Pu-239/240	PuO_2
Sr-90	$SrSO_4$
Th-228	$Th(NO_3)_4$
Th-232	$Th(NO_3)_4$
U-234	$UO_2(OH)_2$
U-235	$UO_2(OH)_2$
U-238	$UO_2(OH)_2$

a. Johnson [1991].

Strontium titanate was not present in the process building, so absorption Type S should not be applied.

Assessments for radiation protection purposes determined that the major beta-gamma nuclides were ^{90}Sr/Y, ^{137}Cs, ^{99}Tc, and tritium. Alpha emitters of concern were isotopes of uranium and plutonium, and other major actinide series elements (^{241}Am, ^{232}Th, et al.) and their decay products [Hoffman 1997].

Table 2-28 lists dominant radionuclides and conservative estimates of total activity for process building facilities and other site facilities. Some of the activity estimates give two values: one before deactivation and one after. All activity estimates are decay-corrected to September 30, 2004. It is implied the dominant radionuclides are listed in order of decreasing activity, though this is not explicitly stated in the reference [WVDP 2005].

Table 2-28. Dominant radionuclides and conservative estimates of total activity for process building facilities and other site facilities.[a,b]

Facility or area	Dominant radionuclides	Estimated total activity as of 09/30/2004
Acid recovery cell	Pu-241, Am-241, Pu-239, Cs-137, Pu-240	192 Ci before deactivation 66 Ci after deactivation
ARPR	Pu-241, Pu-238, Am-241, Cs-137, Pu-239	31 Ci
Analytical decontamination aisle	Cs-137, Sr-90, Pu-241, Am-241, Pu-239	0.1 Ci
Analytical hot cells	Cs-137, Sr-90, Pu-241, Am-241, Pu-239	39 Ci
CPC crane room	Sr-90, Cs-137, Pu-241, Am-241, Pu-239	5.5 Ci
COA	Cs-137, Sr-90, Pu-241, Am-241	Not significant
CPC	Cs-137, Sr-90, Pu-241, Am-241, Pu-238	116 Ci
Contact size reduction facility (former MSM repair shop)	Cs-137, Sr-90	Not significant
Control room	Cs-137, Sr-90	Not significant
EDR	Sr-90, Cs-137, Pu-241, Am-241, Pu-239	39 Ci
XC1	Sr-90, Cs-137, Pu-241, Am-241, Pu-239	4,900 Ci before deactivation 71 Ci after deactivation
XC2	Pu-241, Sr-90, Am-241, Cs-137, Pu-238	2.0 Ci
XC3	Sr-90, Cs-137, Pu-241, Pu-238, Am-241	1,274 Ci before deactivation 71 Ci after deactivation
FRS	Cs-137, Sr-90, Pu-241, Am-241, Pu-239	282 Ci before deactivation 192 Ci after deactivation
GPC	Cs-137, Sr-90, Pu-241, Am-241, Pu-238	6,000 Ci after deactivation
GCR and crane room extension	Cs-137, Sr-90, Pu-241, Am-241, Pu-239	7.1 Ci
GOA	Cs-137, Sr-90, Pu-241, Am-241, Pu-239	Not significant
HEV building and HEV crane room	Sr-90, Cs-137, Pu-241, Am-241, Pu-238	610 Ci (assuming fully-loaded filters)
Hot acid cell	Pu-241, Am-241, Pu-238, Pu-239, Pu-240	17 Ci (in tanks)
Liquid waste cell	Sr-90, Pu-241, Cs-137, Am-241, Pu-238	1,014 Ci, mostly in tanks 280 Ci after tanks are removed.
LWA	Pu-241, Am-241, Pu-238, Pu-239, Pu-240	85 Ci
MOA	Cs-137, Sr-90, Pu-241, Am-241	Not significant
Miniature cell	Cs-137, Sr-90, Pu-241, Am-241, Pu-239	8.5 Ci
OGA	Pu-241, Pu-238, Am-241, Cs-137, Sr-90	40 Ci
OGBR	Pu-241, Sr-90, Am-241, Cs-137, Pu-239	75 Ci
Off gas cell (OGC)	Pu-241, Pu-238, Am-241, Cs-137, Sr-90	459 Ci
PMC	Cs-137, Sr-90, Pu-241, Am-241, Pu-238	2,100 Ci
PMCR and MRR	Cs-137, Sr-90, Pu-241, Am-241, Pu-238	5.2 Ci

Facility or area	Dominant radionuclides	Estimated total activity as of 09/30/2004
Product purification cell (PPC)	Pu-241, Am-241, Pu-239, Pu-240, Pu-238	43 Ci
SR Room	Cs-137, Sr-90, Pu-241, Am-241, Pu-238	0.1 Ci
Southwest stairs	Pu-241, Pu-238, Am-241, Cs-137, Pu-239	5.2 Ci
UWA (and pump niches)	Pu-241, Am-241, Pu-238, Sr-90, Pu-239	26 Ci
ULO	Sr-90, Cs-137, Pu-241, Am-241, Pu-238	0.1 Ci (in tanks and pumps)
UPC	Cs-137, Sr-90, Tc-99, Pu-238, Am-241	12 Ci
VEC	Cs-137, Sr-90, Pu-241, Am-241, Pu-238	66 Ci
VWR	Cs-137, Sr-90, Pu-241, Am-241, Pu-238	74 Ci
LLWT building	Cs-137, Sr-90	None available
Permanent ventilation system building	Cs-137, Sr-90	None available
STS support building	Cs-137, Sr-90, Am-241, Pu-241, Cm-244 (98% of the activity is Cs-137 and Sr-90)	213 Ci (in the valve aisle)

a. Source: WVDP [2005].
b. LLWT – Low-Level Waste Treatment; MRR – Manipulator Repair Room; PMCR – PMC Crane Room; STS – Supernatant Treatment System; ULO – Uranium Loadout Area; VWR – Ventilation Wash Room.

3.0 OCCUPATIONAL MEDICAL DOSE

Presently, little information has been garnered on medical X-rays required as a condition of employment for West Valley workers. NFS [1976] indicated all radiation workers received preemployment physicals and annual physicals thereafter, with chest X-rays taken every 2 years. Review of claim files in the claims tracking system shows workers under DOE received preplacement examinations, which included chest X-rays, and that workers included in respiratory protection programs (e.g., asbestos workers) received routine follow-up exposures. However, no evidence has been found that these procedures were performed at the West Valley site.

Because no evidence has been identified to indicate the X-ray procedures were performed onsite at West Valley or at any other covered site, do not assign X-ray dose according to ORAUT-OTIB-0079 [ORAUT 2017].

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE

This section describes environmental internal and external radiation exposures to workers based on environmental monitoring at the West Valley reprocessing plant or WVDP site. “Site” in this context means within the security fence. It does not refer to the Western New York Nuclear Service Center property within which the plant site is located. NFS considered the boundary of the state property its site boundary during operations, not the area defined by its security fence [12].

4.1 INHALATION AND SUBMERSION DOSE

Airborne releases of radioactive material at West Valley were discharged to the atmosphere from the plant’s 200-ft main stack and a number of ground-level release points associated with specific facilities or operations.

4.1.1 Stack Releases

The following discussion focuses on airborne emissions from the West Valley stack during operations. Stack releases after the plant ceased operating were much smaller in magnitude. The airborne effluents during operations are assessed using maximizing assumptions of 2,600 hr/yr of occupancy, a breathing rate of 1.5 m^3/hr, and a particle size of 1 µm AMAD. The latter leads to maximized organ dose for the nuclides considered. These maximizing assumptions would not normally be used in a site profile. In most cases a best estimate approach would be provided utilizing 2,000 hr/yr of occupancy and a particle size of 5 µm AMAD. A maximizing approach is used here to emphasize the magnitude of West Valley's airborne effluents vis-à-vis dose to personnel who spent time outdoors onsite.

NFS' technical specification limits for airborne effluents were as follows [Runion 1970, Table 1]:

- A continuous release limit of 0.1 µCi/s for particulates (alpha plus beta),
- A daily release limit of 12,600 Ci/d for ^{85}Kr, and
- An annual release limit for ^{131}I of 3.3 Ci.

The daily limit for ^{85}Kr reflects the batched nature of the fuel dissolving process.

NFS performed both continuous monitoring and sampling of its stack effluent. Monitoring data were usually provided in monthly plant status reports. Results were typically reported as a fraction (percentage) of the applicable technical specification limit. Occasionally, total activity would be included in monthly effluent statements, but in general, total activity was only reported on an annual basis.

Particulate releases reported before March 1969 represent 1 day of decay for short-lived activity. Beginning in March 1969, particulate samples were decayed for 3 days before counting [Keely 1969b].

4.1.1.1 Particulates

Table 4-1 shows total particulate releases (alpha plus beta) from the main stack for 1966 through 1972. Fuel dissolving was carried out between 1966 and 1971. The processing plant shut down in 1972 after completion of plutonium recovery operations.

Table 4-1. Total particulate releases from West Valley's main stack, 1966 to 1972 (Ci).

Year	Total release
1966	0.15
1967	0.45
1968	1.14
1969	0.12
1970	0.19
1971	0.01
1972	0.15

Table 4-1 shows 1968 was the maximum year for particulate releases from the West Valley processing plant, despite higher burnup fuels being processed in later years. The relatively large particulate release of 1.14 Ci for 1968 was the result of numerous filter failures [North and Clark 1968; NFS 1968].

NFS reported total beta and alpha particulate releases of 1.86×10^{-1} Ci and 1.16×10^{-3} Ci, respectively, for 1970 [Miller 1972b]. These values give a beta-to-alpha ratio of 160. For 1971, NFS reported total beta and alpha particulate releases of 1.39×10^{-2} Ci and 1.92×10^{-4} Ci, respectively [Miller 1972b]. These values give a ratio of 72. NFS [1976] states the ratio of 72 for 1971 was "also typical of other years."

Applying the beta-to-alpha ratio of 72 to the particulate release for 1968 gives 1.12 Ci beta and 1.56×10^{-2} Ci alpha. These were the maximum particulate releases from the West Valley stack over its history. In terms of annualized release rate, the values are 3.55×10^{-8} Ci/s beta and 4.95×10^{-10} Ci/s alpha.

A calculation of site boundary alpha and beta particulate concentrations performed in about 1974 [Site boundary concentrations, no date] states the minimum dispersion (i.e., maximum concentration) is given by a dispersion coefficient of 2.5×10^{-7} s/m^3. This is a larger value than seen in more recent, albeit more formal, references. For example, the 1985 environmental monitoring report for the WVDP reports a maximum dispersion coefficient of 1.4×10^{-7} s/m^3 [WVNSC 1986, Table 4-1].

Since the annual average breathing zone concentration on the site property will be significantly less than that where the maximum dispersion coefficient occurs, using the maximum coefficient to estimate onsite concentration is favorable to the claimant. In this case, the maximum dispersion coefficient is 2.5×10^{-7} s/m^3.

Table 4-2 shows calculation of maximized annual intake from the particulate releases for 1968. The occupancy of 2,600 hr/yr reflects 50 hr/wk for 52 wk/yr.

Table 4-2. Maximized annual intake of particulate releases, 1968.

Type	Total release (Ci)	Annual release rate (Ci/s)	Dispersion coefficient (s/m^3)	Concentration (Ci/m^3)	Concentration (Bq/m^3)	Breathing rate (m^3/hr)	Occupancy (hr/yr)	Annual intake (Bq)
Beta	1.12	3.55E–08	2.5E–07	8.9E–15	3.3E–04	1.5	2,600	1.3E+00
Alpha	1.56E–02	4.95E–10	2.5E–07	1.2E–16	4.6E–06	1.5	2,600	1.8E–02

The magnitude of the maximum particulate intakes from stack emissions are such that selection of isotopic mix is largely immaterial. If one assumes Type M ^{239}Pu for 100% of the alpha intake, the resulting effective dose is 8.5×10^{-2} mrem for a particle size of 1 µm AMAD. The maximum organ dose is 2.7 mrem to the bone surface. Likewise, if one assumes Type S ^{106}Ru for 100% of the beta intake, the resulting effective dose is 7.4×10^{-3} mrem and the maximum organ dose is 3.4×10^{-2} mrem to the lung. These calculations reflect effective and organ dose coefficients from International Commission on Radiological Protection (ICRP) Publication 68 [ICRP 1994] and ICRP Publication 72 [ICRP 1996], respectively.

The calculations above show the magnitude of the particulate releases from the main stack, and associated personnel dose were of minimal consequence even under unrealistic, maximizing assumptions.

4.1.1.2 Krypton-85

Reports of the total activity of ^{85}K released for a given year have only been identified for 1966 through 1969. These are given in Table 4-3. The largest release for these years was 328,000 Ci in 1967.

Table 4-3. Total ^{85}Kr releases from main stack, 1966 to 1969 (Ci).

Year	Total release
1966	77,000
1967	328,000
1968	193,000
1969	300,000

Table 4-4 shows ^{85}Kr release data for 1967 and 1971 in terms of fraction of the daily release limit for each month. These are the only data available for 1971. No data have been found for 1970, but the 1971 releases should be comparable. There is little correlation between particulate and noble gas emissions, so particulate data are not a reliable means for estimating ^{85}Kr releases.

The curies per month data given in Table 4-4 were computed by multiplying the reported fraction by 12,600 Ci (the daily release limit) and then by the number of days in the respective month. The Ci/month values for 1967 sum to 327,537 Ci, matching the reported value of 328,000 Ci in Table 4-3.

The total ^{85}Kr released in 1971 corresponds to an annualized release rate of 0.04 Ci/s. Applying the maximum dispersion coefficient of 2.5×10^{-7} s/m^3 (Section 4.1.1.1) gives a maximum concentration on the site property of 1.0×10^{-8} Ci/m^3, or 370 Bq/m^3.

The ICRP Publication 68 effective dose rate factor for ^{85}Kr is 2.2×10^{-11} Sv/d per Bq/m^3 [ICRP 1994]. Applying this to the maximum ^{85}Kr concentration gives an effective dose rate of 8.14×10^{-9} Sv/d, or 8.14×10^{-4} mrem/d. Applying a maximizing assumption for occupancy of 2,600 hr/yr gives an effective dose of 0.09 mrem.

Table 4-4. Monthly ^{85}Kr release data, 1967 and 1971.[a]

Month	Kr-85 releases (% of daily limit) 1967	Kr-85 releases (% of daily limit) 1971	Kr-85 releases (Ci/mo) 1967	Kr-85 releases (Ci/mo) 1971
January	7.7	50.0	30,076	195,300
February	6.0	52.0	21,168	183,456
March	0.0	10.3	0	40,232
April	5.3	22.0	20,034	83,160
May	7.4	No fuel dissolved	28,904	0
June	7.2	30.0	27,216	113,400
July	16.4	40.0	64,058	156,240
August	17.9	42.0	69,917	164,052
September	0.0	7.3	0	27,594
October	4.9	41.0	19,139	160,146
November	5.0	27.0	18,900	102,060
December	7.2	Not applicable	28,123	0
Totals	Not applicable	Not applicable	**327,537**	**1,225,640**

a. Sources: For 1967, Lewis [1967a,b,c,d]; NFS [1967]. For 1971, Duckworth [1971a,b, 1972a]; Wilcox and Wenstrand [1971]; Bentley [1972]; NFS [1972b].

4.1.1.3 Iodine-131

The fuel processed at West Valley had decayed for a minimum of 12 months before dissolving. Longer cooling times were typical, particularly for fuel with high burnup (i.e., that from commercial power reactors). Currently, the data identified for airborne releases of radioiodines from West Valley during operations are incomplete. Data are available for 1966 through 1969 and for part of 1971 (July through December). The partial data for 1971 are shown in Table 4-5.

The data in Table 4-5 suggest the ^{131}I released from the West Valley stack was a small fraction of the technical specification limit of 3.3 Ci/yr. Further, the iodine releases data given in monthly summary reports for 1966 through 1969 are never more than a small fraction of the technical specification limit. These values are reported as “less than” some fraction of the limit, though the fractions vary from month to month. The largest value observed was less than 7% of the limit reported for August and

Table 4-5. Airborne releases of ^{131}I, July to December 1971 (Ci).[a]

Month	Discharged
July	1.66E–04
August	5.62E–04
September	7.80E–05
October	1.54E–04
November	2.37E–04
December	7.80E–05

a. Sources: NFS [1972b]; Duckworth [1972a]; Bentley [1972].

September 1968. However, 7% of 3.3 Ci is significantly larger than the releases given in Table 4-5 for 1971.

Given the variable and incomplete nature of the reported airborne iodine releases from West Valley, a maximizing evaluation was performed using the technical specification limit of 3.3 Ci/yr. In terms of release rate, 3.3 Ci/yr equates to 1.0×10^{-7} Ci/s. Table 4-6 shows the calculation of the associated maximized annual intake.

Table 4-6. Maximized annual intake of ^{131}I.

Type	Total release (Ci)	Annual release rate (Ci/s)	Dispersion coefficient (s/m^3)	Concentration (Ci/m^3)	Concentration (Bq/m^3)	Breathing rate (m^3/hr)	Occupancy (hr/yr)	Annual intake (Bq)
I-131	3.3	1.05E–07	2.5E–07	2.6E–14	9.7E–04	1.5	2,600	3.8E+00

The ICRP Publication 68 effective dose coefficient for ^{131}I in vapor form is 2.0×10^{-8} Sv/Bq [ICRP 1994]. Using this value, an intake of 3.8 Bq gives an effective dose of 7.6×10^{-8} Sv, or 7.6×10^{-3} mrem. The Publication 68 organ dose coefficient for the thyroid for ^{131}I in vapor form is 3.9×10^{-7} Sv/Bq. This yields a maximum organ dose of 1.5×10^{-1} mrem to the thyroid.

4.1.1.4 Tritium

NFS did not include tritium in its stack monitoring data during the operations period. Tritium was released from fuel assemblies during segmentation operations in the head end process cells. The U.S. Environmental Protection Agency (EPA) Office of Radiation Programs took measurements of tritium releases from West Valley over four processing campaigns from June through November 1971 [Cochran et al. 1972]. They correlated their tritium release data to fuel burnup and then used NFS’ processing records to estimate tritium releases over the past 6 years. They estimated a total of 2,800 Ci had been released, for an average of 460 Ci/yr. They determined the speciation was 76% HTO vapor and 24% gaseous.

For a maximizing assessment, it is assumed the entire 2,800 Ci of tritium was released in a single year and that it is 100% HTO vapor. Table 4-7 shows the calculation of the associated maximized intake.

Table 4-7. Maximized annual intake of tritium.

Type	Total release (Ci)	Annual release rate (Ci/s)	Dispersion coefficient (s/m^3)	Concentration (Ci/m^3)	Concentration (Bq/m^3)	Breathing rate (m^3/hr)	Occupancy (hr/yr)	Annual intake (Bq)
Tritium	2,800	8.88E–05	2.5E–07	2.2E–11	8.2E–01	1.5	2,600	3.2E+03

Applying the ICRP Publication 68 effective dose coefficient for tritium [1.8×10^{-11} Sv/Bq; ICRP 1994] gives an effective dose of 5.8×10^{-8} Sv, or 5.8×10^{-3} mrem. Multiplying by 1.5 to account for skin absorption gives 8.7×10^{-3} mrem.

4.1.2 Ground Level Releases

There were several sources of airborne effluents at or near ground level at West Valley over its history. These included:

- Discharges from laundry facilities,
- The FRS vent,
- The Extraction Chemical Room (XCR) vent,
- The XCR tank vent,
- The Process Chemical Room (PCR) tank vent, and
- The Low Level Waste Treatment (LLWT) Facility.

The above were sources of ground level airborne effluents while the plant was in operation. There were other sources of ground level releases during the WVDP, but these releases were much smaller in magnitude than those during operations. The following discussion focuses on the operations period.

Table 4-8 summarizes particulate discharges from the stack and ground level release points at West Valley in 1970 and 1971 [Miller 1972b].

Table 4-8. Particulate discharges (Ci), 1970 and 1971.[a]

Source	Gross alpha discharges for 1970	Gross beta discharges for 1970	Gross alpha discharges for 1971	Gross beta discharges for 1971
Stack	11.6E–4	18.6E–2	1.92E–4	1.39E–2
Laundry	6.1E–6	2.4E–5	9E–7	8.9E–5
LLWT	No data	No data	1E–7	1.7E–5
FRS vent	2.5E–6	1.7E–4	9.2E–6	2.5E–4
XCR vent	6.7E–6	1.7E–4	6.9E–5	6.6E–4
PCR tank vent	5E–8	3E–6	5E–8	3E–6
XCR tank vent	2.8E–6	8.5E–2	2.8E–6	8.5E–2
Totals	**1.178E–3**	**2.714E–1**	**2.74E–4**	**9.99E–2**

a. Source: Miller [1972b].

Table 4-8 shows the largest source of beta particulate emission other than the stack was the XCR tank vent, which released 8.5×10^{-2} Ci of activity for both 1970 and 1971. The XCR vent was the largest source of alpha particulate, with 6.9×10^{-5} Ci being released in 1971. In terms of annualized release rate, these values are 2.7×10^{-9} Ci/s and 2.2×10^{-12} Ci/s for beta and alpha, respectively.

The 1985 environmental monitoring report for the WVDP reports a maximum dispersion coefficient for a ground level release of 6.44×10^{-6} s/m^3. This value corresponds to a receptor azimuth of 337.5° at a distance of 805 m [WVNSC 1986, Table 4-2]. Table 4-9 shows use of this value with the above release rates to calculate intake.

Table 4-9. Intakes from ground-level releases, 1970 and 1971.

Type	Total release (Ci)	Annual release rate (Ci/s)	Dispersion coefficient (s/m^3)	Concentration (Ci/m^3)	Concentration (Bq/m^3)	Breathing rate (m^3/hr)	Occupancy (hr/yr)	Annual intake (Bq)
Beta	8.5E–02	2.70E–09	6.44E–06	1.7E–14	6.4E–04	1.5	2,600	2.5E+00
Alpha	6.9E–05	2.19E–12	6.44E–06	1.4E–17	5.2E–07	1.5	2,600	2.0E–03

Treating the beta intake as 100% Type S ^{106}Ru of 1 µm AMAD particle size gives an effective dose of 1.6×10^{-7} Sv, or 1.6×10^{-2} mrem. Treating the alpha as Type M ^{239}Pu with a particle size of 1 µm AMAD gives an effective dose of 9.4×10^{-8} Sv or 9.4×10^{-3} mrem.

4.2 AMBIENT DOSE

Nuclear-Chicago's "Explanation of Exposure Report Data" on the reverse of its exposure reports states ordinarily they considered any exposure on a control badge to be from a source other than that which caused exposure to a monitored individual. Control readings were therefore subtracted from readings of personnel badges [Wenstrand 1968c]. Likewise, the "Explanations and Remarks" on the reverse of the Landauer exposure reports says control readings were normally subtracted from personnel badge readings. However, it adds personnel badge readings may be normalized to Landauer's own control films (only) if it appeared the controls that accompanied the personnel badges had been "exposed differently than the personnel packet" [Landauer 1971–1972, p. 3, Item 6].

Exposure rates outside the process building during the operations period were significant with respect to exposure to control dosimeters stored in the Guard House. The exposure rate at the Guard House during operations was nominally 2 mR/hr from radioactivity in the plant's acid recovery system.

Heacker [1972] discusses the problem of dose accumulating on the control badges stored in the Guard House. He says the control badge for the week of December 20 to 26, 1971, showed 60 mrem because there was a drum reading 30 R/hr on a truck parked overnight at the cooling tower. The control badge reading for the following week was 20 mrem [Heacker 1972].

Per Heacker [1972], it is estimated control badges received a dose of 20 mrem/wk from elevated background during operations. Prorating this for 50 hr/wk of occupancy yields 6 mrem/wk which needs to be added to dosimeter and missed dose for monitored workers at West Valley over the period 1966 through 1973.

It is assumed the exposure rate at the storage location for the control badges was reduced after initial cleanup activities completed in 1973. An estimate of the dose rate to control badges in the Guard House for the period 1974 until 1982 can be made from later area monitoring data.

After DOE took over site operations in February 1982, dosimetry was only issued to persons who entered radiologically controlled areas. An area monitoring program served to monitor exposures in uncontrolled areas to demonstrate compliance with applicable limits. The program used TLDs stationed in numerous (indoor) locations about the plant site. Reported results were net values above background stated in terms of mrem/500 hr of occupancy. Control TLDs were stored in a shielded location [e.g., WVNSC 1999].

The earliest period for which area monitoring data are available for the Guard House is 1983 to 1985. The annual doses measured for that location were 14 mrem, 11 mrem, and 14 mrem for 1983 to 1985, respectively [Roberts 1986]. These values are based on 2,000 hours of occupancy. Adjusting for 2,600 hours of occupancy, the maximum annual value of 14 mrem becomes 18 mrem. This dose should be added to dosimeter and missed dose for monitored West Valley workers for the period 1974 to 1981.

Personnel dosimetry was required for entry into the plant security area during the operations and residual radiation periods. Users should not need to assign external dose for an unmonitored worker during this time. For the DOE period, monitoring data representative of what an unmonitored worker would have received from ambient exposure while outdoors on the West Valley site have not been identified. To be favorable to the claimant, it is assumed an unmonitored worker was exposed to an annual dose equal to the applicable administrative limit for the period in question. For 1982 through 1993 the administrative limit for annual dose to an unmonitored individual is assumed to have been 170 mrem [Roberts 1986]. For 1994 forward the annual limit was 100 mrem per 10 *Code of Federal Regulations* (C.F.R.) Part 835 [DOE 1993].

4.3 SUMMARY OF EXPOSURES FROM ENVIRONMENTAL SOURCES

The maximizing calculations described in Section 4.1 show intakes associated with airborne effluents at West Valley were negligible.

Monitored workers should be assigned an additional 312 mrem/yr to dosimeter or missed dose for the period April 1966 through 1973. For 1974 to 1981, 18 mrem/yr should be added. The added dose should reflect the monitoring period, i.e., 6 mrem/wk or 26 mrem/mo for April 1966 through 1973, and 0.35 mrem/wk or 1.5 mrem/mo for 1974 through 1981. The dose should be prorated as appropriate for partial years of employment. The additional dose should be treated as photons in the 30- to 250-keV energy group [ORAUT 2021].

Personnel dosimetry was required for entry into the plant security area between 1966 and February 1982, when DOE took over site operations [13]. Users should not need to assign external dose for an unmonitored worker during this time. For 1982 to 1993 170 mrem/yr should be assumed for unmonitored workers [14]. For 1994 forward, 100 mrem/yr should be assumed. This dose should be treated as photons in the 30- to 250-keV energy group [ORAUT 2021].

Table 4-10 summarizes external dose to be assigned to monitored and unmonitored workers at West Valley to account for excess background subtraction and onsite ambient exposures. All external dose assigned from Table 4-10 should be photons in the 30- to 250-keV energy group [ORAUT 2021] using the organ DCFs for Exposure (R) to Organ Dose (HT) and assuming isotropic geometry in accordance with DCAS IG-0001, *External Dose Reconstruction Implementation Guideline* [NIOSH 2007].

Table 4-10. Summary of external onsite environmental dose (mrem/yr).[a]

Year	Monitored workers additional dose	Monitored workers uncertainty	Unmonitored worker ambient dose	Unmonitored workers uncertainty
1965	312	Normal, ±30%	Not applicable	Not applicable
1966	312	Normal, ±30%	Not applicable	Not applicable
1967	312	Normal, ±30%	Not applicable	Not applicable
1968	312	Normal, ±30%	Not applicable	Not applicable
1969	312	Normal, ±30%	Not applicable	Not applicable
1970	312	Normal, ±30%	Not applicable	Not applicable
1971	312	Normal, ±30%	Not applicable	Not applicable
1972	312	Normal, ±30%	Not applicable	Not applicable

Year	Monitored workers additional dose	Monitored workers uncertainty	Unmonitored worker ambient dose	Unmonitored workers uncertainty
1973	312	Normal, ±30%	Not applicable	Not applicable
1974	18	Constant	Not applicable	Not applicable
1975	18	Constant	Not applicable	Not applicable
1976	18	Constant	Not applicable	Not applicable
1977	18	Constant	Not applicable	Not applicable
1978	18	Constant	Not applicable	Not applicable
1979	18	Constant	Not applicable	Not applicable
1980	18	Constant	Not applicable	Not applicable
1981	18	Constant	Not applicable	Not applicable
1982	0	Not applicable	170	Constant
1983	0	Not applicable	170	Constant
1984	0	Not applicable	170	Constant
1985	0	Not applicable	170	Constant
1986	0	Not applicable	170	Constant
1987	0	Not applicable	170	Constant
1988	0	Not applicable	170	Constant
1989	0	Not applicable	170	Constant
1990	0	Not applicable	170	Constant
1991	0	Not applicable	170	Constant
1992	0	Not applicable	170	Constant
1993	0	Not applicable	170	Constant
1994	0	Not applicable	100	Constant
1995	0	Not applicable	100	Constant
1996	0	Not applicable	100	Constant
1997	0	Not applicable	100	Constant
1998	0	Not applicable	100	Constant
1999	0	Not applicable	100	Constant
2000	0	Not applicable	100	Constant
2001	0	Not applicable	100	Constant
2002	0	Not applicable	100	Constant
2003	0	Not applicable	100	Constant
2004	0	Not applicable	100	Constant
2005	0	Not applicable	100	Constant
2006	0	Not applicable	100	Constant
2007	0	Not applicable	100	Constant

a. Additional dose should be prorated as appropriate for partial years of employment.

5.0 OCCUPATIONAL INTERNAL DOSE

5.1 INTERNAL DOSE CONTROL PROGRAM

5.1.1 Operations and Residual Radiation Periods, 1966 to February 25, 1982

NIOSH has determined that internal doses from uranium cannot be reconstructed for the operations period from January 1, 1969, through December 31, 1973, and from mixed fission products for 1972 and 1973 (see Section 1.3). Although it is not possible to completely reconstruct internal radiation doses for all WVDP workers, NIOSH intends to use any internal monitoring data that might become available for an individual claim and that can be interpreted using existing NIOSH dose reconstruction processes or procedures to conduct partial dose reconstructions for employees who do not qualify for inclusion in the SEC [NIOSH 2019].

The internal dose control program at West Valley during operations consisted of screening methods such as air sampling and nasal swabs as a method to estimate internal exposures. Bioassay was viewed as an evaluation aid in determining historical levels of exposure, and not as a means for exposure control [O'Reilly 1975]. The screening tools were based on the legal thresholds for mandatory personnel monitoring, i.e., 25% of the applicable AEC or NRC limits [O'Reilly 1975]. NFS was regulated and overseen by the AEC over the entire operations period. Regulatory authority transitioned to the NRC in 1974. Radiation protection standards for the site were therefore prescribed by 10 C.F.R. Part 20 throughout the operations and residual radiation periods [10 C.F.R. 20, 1991], until operational authority was transferred to DOE on February 26, 1982.

5.1.1.1 Nasal Swabs

As of 1968, and through at least 1974, NFS' action limits for nasal swabs were 50 dpm alpha and 5,000 dpm beta [O'Reilly 1975]. NFS stated nasal swab results were "used as a screening technique to determine when further evaluation is required. If the nasal swabs indicate the alpha activity is less than 50 dpm, the exposure is considered by NFS to be far enough below the limits specified in 10 CFR 20.103 that further evaluation is not required." They asserted their 50 dpm alpha criterion corresponded to "about 25%" of the Part 20 limit, and point out that personnel monitoring was not required until 25% of the limit was likely to be exceeded. "This specific procedure has been in effect at NFS since 1968" [O'Reilly 1975].

As of 1972, nasal swabs were taken after "each zone entry" and counted for alpha and beta. If activity was detected, urinalysis and a whole body count were performed. If alpha activity above 3 dpm was detected in the urinalysis, then fecal sampling was performed [Nelson 1973a, p. 9].

NFS' 1973 Safety Analysis Report says nasal swabs were obtained upon exiting from Zone 4 areas to screen for inhalations [NFS 1976].

1974: Nasal swabs were taken when "potential for inhalation of radioactive materials has increased." "Based on the activity found on nasal swabs, additional internal monitoring such as bioassay sampling or whole body counting may be required" [NFS 1974b].

5.1.1.2 Chest Counter

The chest counter was a 3- by 3-in. NaI detector formerly used as a thyroid monitor. Per Keely and Wenstrand (1971), the detector was collimated and used at a distance of 10 cm from the chest. (The detector was at 10 cm when the collimator was in contact with the subject being counted.) The chest counter was used at West Valley as a screening tool. Initially comparisons were made between the gross count rate from an individual suspected of an intake and that from a control subject. Later the procedure evolved to use a phantom as a control and a conversion factor for computing a measure of intake. The chest counter was in use at least as early as November 1966 [Helgeson 1966], and was used routinely back to at least 1967 [Miller 1972a].

As of 1971, the NFS procedure for use of the chest counter was to calibrate the system using ^{137}Cs photon energies, count a phantom, then count the individual of interest. The counts were performed with the collimator in contact with the subject. The gross count rates for a region of interest 15 channels (8 below and 7 above) about the ^{137}Cs peak were determined for the phantom and the individual. The net count rate was computed and multiplied by a conversion factor to get "relative lung burden" [HNS 1971].

As of September 1972 the chest counter had been replaced by an on-site whole body counter [Miller 1972a].

5.1.2 U.S. Department of Energy Period, February 26, 1982, to Present

Presently, the WVDP uses a combination of bioassay and air sampling data for internal dose assessment. Air sampling data are used in cases where bioassay methods lack requisite sensitivity, e.g., for the alpha constituents of the mixed fission and activation product source term. In these cases, the detection limits associated with bioassay methods are such that the corresponding dose could exceed the site investigation level if it is assumed the date of intake was 12 months earlier. Derived air concentration (DAC)-hr calculations are therefore used to assess exposures to airborne alpha activity. Workers for whom decision levels are established via alpha air monitoring must also participate in the alpha in vitro bioassay program [Kubiak 1997, p. 19].

The WVDP investigation level for internal dose assessment is 100 mrem committed effective dose equivalent (CEDE). If a DAC-hr calculation gives a CEDE of 10 mrem or less, no additional action is taken. However, all data and calculations are placed in the worker's dosimetry file. If the calculated CEDE exceeds 10 mrem, then additional sampling is required [Kubiak 1998, p. 47].

5.2 BIOASSAY PROGRAM

5.2.1 Operations and Residual Radiation Periods, 1966 to February 25, 1982

In vitro bioassay at West Valley during its operations period was a combination of routine and for cause analyses. Records show that ad hoc urine and fecal analyses were performed at least as early as September 1967. A routine urinalysis program was in place since at least 1968. It appears a routine fecal analysis program started sometime between 1968 and 1970 [Wenstrand 1968d] [15].

In vivo bioassay at West Valley during its operations period was performed using the chest counter (described above), a commercial mobile whole body counting service, and a phoswich counter operated by the New York University (NYU) Medical Center in Rochester, New York. The latter was used for follow-up in vivo counts for TRU elements after significant intakes. An onsite in vivo counter was installed at West Valley sometime in 1972 [Miller 1972a].

As of 1967, the NFS bioassay program for plutonium depended upon the location in the plant where an intake could have occurred. If a suspected intake occurred in a mechanical processing area, it was assumed the intake would have been accompanied by fission products. Therefore, for these areas, the internal dose control program consisted of a combination of nasal swabs to look for gross beta activity and in vivo counting using the NFS chest counter. Follow-up bioassay (fecal sampling or in vivo counting) would be ordered based on the results of the nasal swab and chest count. Plutonium in the mechanical processing (head end) areas would most likely have been in an insoluble form (metal or oxide) [Runion 1967a, p. 4].

If a possible plutonium intake occurred in a chemical processing area, the plutonium would have been in nitrate (soluble) form. Urinalysis was therefore used for these individuals [Runion 1967a].

In April 1968, Clark [1968b] asserted NFS' procedures required urine samples if:

- There was a positive nasal swab;
- Work was performed in an area with alpha contamination greater than 500,000 dpm/100 cm^2;
- There was a loss of air supply in an area with smearable alpha contamination greater than 500 dpm/100 cm^2 or smearable beta contamination greater than 50,000 dpm/100 cm^2; and
- Facial contamination was detected above 50 cpm alpha or 100 cpm beta (per probe area).

It is assumed that a “positive” nasal swab referred to a result in excess of the action limits given in Section 5.1.1.1.

As of 1968 fecal sampling was not part of the NFS bioassay program except for on an ad hoc basis when deemed necessary by their Health and Safety Department [Clark 1968b]. However, a February 29, 1968, memorandum discusses startup of a fecal monitoring program for workers who routinely entered alpha-contaminated areas [Wenstrand 1968d]. By 1970 this routine fecal sampling was an element of their program. A summary report from the bioassay technician for 1970 says 200 fecal samples were shipped out that year [Kester 1971].

As of 1972, “All permanent employees exposed to radiation receive an annual physical from the plant’s physician” which included blood counts and “checks for internal alpha (plutonium) exposure” [Nelson 1972]. NFS [1976] says a baseline radiation history was established for each radiation worker, which included a physical, blood counts, and urinalysis records. Blood counts and urinalysis were repeated semi-annually. It is not known how far back this practice dated.

As of September 1972 NFS had replaced the chest counter with a “shielded whole body counter,” and stated that “all plant personnel are now routinely counted by the whole body counter once yearly and at any other time there is reason to believe an uptake has occurred” [Nelson 1973a, p. 8]. The annual whole body counts replaced urinalyses for mixed fission products under an assumption that intakes of ^{90}Sr or ^{147}Pm would be accompanied by other, gamma-emitting isotopes [Nelson 1973b, p. 15].

NFS [1976] states urine samples were collected semi-annually on all radiation workers “to evaluate the potential long-term buildup of plutonium in the body.” It appears this practice was in place at least back to 1969 [e.g., Keely and Wenstrand 1971]. Fecal samples were also collected from radiation workers on an annual basis and analyzed for plutonium. It appears this practice began sometime between 1968 and 1969 [Clark 1968b; Keely and Wenstrand 1971].

As of October 1973 the frequency for plutonium urinalysis had been decreased to annually for radiation workers [Nelson 1973b, 15]. NFS’ calculations showed the program would detect a single intake of soluble plutonium of 20% of a MPBB 12 months after intake. “Uranium exposures are evaluated by using plutonium as a tracer.” “Commercial waste burial contractor employees” received annual whole body counts and annual urinalyses for mixed fission products and tritium [Nelson 1973b, p. 16]. Tritium had accumulated in water in burial trenches up to 0.1 $\mu Ci/cm^3$. In an inspection report the AEC remarked on the fact annual urinalysis for tritium would represent a high limit of detection. Other contractor employees received bioassay evaluations if “nasal smears are positive and exceed specified limits” [Nelson 1973b, p. 16].

As of 1974 the internal monitoring program for radiation workers in the process building remained as it was in about October 1973. Annual urine samples were collected and analyzed for plutonium and all radiation workers were whole body counted annually for fission products [NFS 1974b].

A May 18, 1977, memorandum says NFS’ procedures had been modified to implement entrance and termination whole body counts. This was apparently not being done before this time [Oldham 1977].

NFS sometimes experienced delays in processing bioassay samples. A December 28, 1967, letter from the AEC stated, “The bioassay program needs modification and improvements. Specifically the procedure for issuance, follow-up and retrieval of filled urinalysis sample bottles from employees as well as records thereof have not been effective in some instances ... A better effort is necessary in obtaining more timely samples after evidence of possible internal depositions.” Earlier that year (September), 38 bioassay samples were lost due to an error by the parcel delivery service used [Clark 1968b]. In 1974 a worker involved with possible inhalation exposure submitted urine samples on

February 23 and 24 which were not sent to the vendor for analysis until May 16, 1974 [O'Reilly 1975]. It is not known how NFS stored its samples or if preservatives were added.

5.2.1.1 Bioassay Records

As of April 1968 a "Bioassay Log" was kept documenting the names of employees involved in the bioassay program and dates their samples were issued and shipped [Clark 1968b].

As of April 1972 exposure to airborne radioactive material was recorded on a "Lapse of Radiation Control" form. "Bioassay results and whole body counts are recorded on appropriate forms in the individual personnel folders and are available for examination by the employee" [Miller 1972a, p. 19]. A September 7, 1972, memorandum confirms this practice [Duckworth 1972b].

In its 1973 Safety Analysis Report, NFS said dosimetry records were maintained by the Health and Safety Department for each employee or worker [NFS 1976]. These records included previous radiation exposure, current radiation exposure, urinalysis results, fecal analysis results, whole body counting data, and "special reports," such as exposure estimates for lost or damaged film badges. These reports were submitted to the individuals and to the AEC as required.

5.2.2 U.S. Department of Energy Period, February 26, 1982, to Present

Bioassay at the WVDP comes under six categories [Kubiak 1997]:

- Baseline upon initial employment,
- Re-entry, upon resumption of bioassay for an employee previously monitored,
- Periodic routine bioassay,
- Termination bioassay upon termination of employment or radiation worker status,
- Discontinue bioassay upon transfer from monthly to quarterly external monitoring or removal from the respiratory protection program (in vitro only), and
- Special bioassay.

5.2.2.1 Routine Bioassay

The routine bioassay program at the WVDP includes both in vivo and in vitro monitoring.

All radiation workers are included in the in vivo bioassay program. In addition, any female who declares a pregnancy receives both in vivo and in vitro bioassay at the time of declaration and when the pregnancy is declared ended [Kubiak 1998, p. 42].

All dosimetry qualified personnel monitored on a monthly basis or individuals with respiratory protection qualifications for work in radiological areas are included in the WVDP in vitro bioassay program. The in vitro program required baseline, annual, and termination sampling [Kubiak 1998, p. 38].

As of July 31, 1990, routine in vitro analyses for the WVDP were to include [WVNSC 1990a]:

- Plutonium isotopic,

- Total uranium,
- ^{241}Am, and
- ^{90}Sr.

As of December 7, 1990, routine in vitro analyses for the WVDP were to include [WVNSC 1990b]:

- Plutonium isotopic,
- Total uranium, and
- ^{90}Sr.

5.2.2.2 Special Bioassay

Special bioassay is required if action levels established for workplace surface and airborne contamination levels and personnel contamination are reached or exceeded, or if otherwise requested by the RP Department management or technical staff. Special bioassay can be in vivo and/or in vitro. The action levels for special bioassay are [Kubiak 1998, pp. 39–40]:

- A routine bioassay measurement is determined to be positive for radioactive content;
- Facial contamination or detectable radioactivity on nasal or mouth swabs indicates a potential for intake;
- A respiratory protection factor for a respirator is exceeded and exposes the worker to airborne concentrations exceeding the DAC;
- Airborne monitoring indicates the potential for internal dose in excess of 100 mrem CEDE; or
- Wound contamination (in or around a skin break), which is:
  - Greater than 100 dpm/100 cm2 alpha;
  - Greater than 10,000 dpm/100 cm2 beta/gamma; or
- Extensive personnel decontamination is required.

5.2.3 Summary

Table 5-1 gives a summary of ad hoc and routine in vivo monitoring carried out at West Valley from 1966 to the present.

Table 5-1. Summary of in vivo bioassay frequencies for West Valley radiation workers.[a]

Period	Ad hoc	Routine
1966–September 1972	Chest counter used for onsite screening Phoswich counter at NYU-Rochester used for ad hoc counts for TRU (postincident follow-ups)	Commercial, mobile whole body counter used for annual counts of selected workers Chest counter used for routine counts (frequency unclear)
September 1972–1977	Onsite whole body counter used for screening of suspected intakes involving MFPs	Radiation workers counted annually for MFPs using onsite whole body counter
1977–present[b]	Onsite whole body counter used for screening of suspected intakes involving MFPs	Radiation workers given baseline, annual, and termination counts for MFPs using onsite whole body counter

a. MFPs – mixed fission products.
b. In 1993 the system changed from a NaI detector and a bed geometry to an intrinsic germanium detector in a standup geometry.

Table 5-2 gives a summary of routine in vitro monitoring carried out at West Valley from 1966 to the present. In vitro bioassay was also performed ad hoc based on nasal smears or other indicators of possible intake. The type of bioassay (urinalysis or fecal analysis) was selected based on the chemical form (soluble or insoluble) of suspected material.

Table 5-2. Summary of routine in vitro bioassay frequencies for West Valley radiation workers.[a]

Period	Routine urinalysis	Routine fecal analysis
1966–1972	Semiannual for all radiation workers for plutonium Annual for selected workers for MFPs, might have been more frequent than annually	Annual for all radiation workers beginning about 1968
1973–February, 1982	Annual for all radiation workers for plutonium Annual for MFPs and tritium for SDA workers (through 1975)[b,c]	None known
March 1982–late 1990	Baseline, annual, and termination for workers with monthly TLD exchange or with respiratory protection qualifications; analytes were plutonium isotopic, total uranium, Am-241 and Sr-90	None known
Late 1990–present	Baseline, annual, and termination for workers with monthly TLD exchange or with respiratory protection qualifications; analytes were plutonium isotopic, total uranium, and Sr-90	None known

a. MFPs – mixed fission products.
b. Annual whole body counts replaced routine urinalysis for mixed fission products (MFPs) for workers in the process building.
c. The SDA was closed in May 1975.

5.3 IN VITRO ANALYSES

5.3.1 Operations and Residual Radiation Periods, 1966 to February 25, 1982

In vitro bioassay at West Valley during its operating period was generally performed by commercial vendors. However, it appears nuclide-specific urinalyses (i.e., those not analyzed for mixed fission products) were performed in house in 1970 [Kester 1971]. It is not known when this practice actually began or how long it continued.

As of June 1967 the MDA (given as minimum detectable "concentration") for the plutonium urinalysis vendor was 0.45 dpm/d [Clark 1968c].

A letter dated March 19, 1968, states the minimum detectable excretion rate for the bioassay laboratory performing their urinalyses for plutonium was "0.3 dpm/L" [sic] [Monroe 1968]. Keely and Wenstrand [1971] show a value of 0.3 dpm/d for plutonium nitrate in urine.

NFS did not normalize its urinalysis results to a reference excretion rate of 1.4 L/d. Activity per 24 hours was computed by taking the product of the analysis result in activity per unit volume and the total volume collected and dividing the resulting activity by the sampling interval [e.g., Personnel exposure 1974]. For example if 0.85 L was collected over a period of 12 hours and showed 0.12 dpm/L, the result would be recorded as:

$$0.12 \text{ dpm/L} \times 0.85 \text{ L} \div 0.5 \text{ d} = 0.20 \text{ dpm/d} \quad (5\text{-}1)$$

The same result collected over 24 hours would have been reported as 0.10 dpm/d.

In general 24-hour urine samples collected at West Valley during the operations and residual radiation periods had an excretion volume of less than 1 L. Some instances were noted of greater than 1 L collections for periods greater than 24 hours in the NOCTS files, but these were uncommon. Users should therefore multiply urinalysis results reported in terms of activity excreted per liter by 1.4 to correct them to a reference excretion rate of 1.4 L/d.

The "Yearly Report from Bioassay Technician" for 1970 implies urinalyses for other than mixed fission products were performed onsite, and that counts were often lost "due to electrical storms." His numbers show approximately 10% of the counts were lost. The onsite counting in 1970 included samples collected in 1969 because they lost 3 months of counting due to a strike. Counting began in April 1970. For the year they counted 393 urine samples in house, shipped out approximately 150 samples for urinalysis for mixed fission products, and shipped out approximately 200 fecal samples [Kester 1971].

Sometimes a batch of urine samples from an individual would be submitted to the analysis laboratory with instructions to analyze the most recent sample first. The remaining samples were only to be analyzed if the initial analysis showed an activity concentration in excess of some threshold. Otherwise the laboratory was instructed to discard the remaining samples [WNYNSC, no date, pp. 543, 546].

As of 1973 plutonium urinalysis was performed by a commercial vendor via plutonium separation and alpha spectroscopy. It was not performed onsite. The detection limit was 0.2 dpm/L [Nelson 1973b, p. 16]. NFS' calculations showed the program would detect a single intake of soluble plutonium of 20% of a MPBB 12 months after intake. "Uranium exposures are evaluated by using plutonium as a tracer."

In its report summarizing an inspection performed in October 1973 the AEC assumed a detection limit for tritium urinalysis of 0.02 µCi/L. It is not explicitly clear this was a representative minimum detectable concentration (MDC) [Nelson 1973b, p. 16].

Table 5-3 summarizes analytes, vendors, and observed reporting levels for in vitro analyses at West Valley during the operations and residual radiation periods. The reporting levels represent "less than" values given in analysis reports, e.g., "< 0.3 dpm/L." In many cases reporting levels can be inferred from the analysis reports from the various vendors. Some reporting levels are a priori values, others are *a posteriori*. Reporting levels are not apparent in analysis reports for samples associated with incidents or which otherwise showed positive activity for all samples.

Table 5-3. In vitro analyses and reporting levels for West Valley during the operations and residual radiation periods.[a]

Year	Analyte	Vendor	Reporting level
1967	MFP in urine	Controls for Radiation	10 to 50 dpm/L
1967	Gross beta in urine[b]	Teledyne	7770 dpm/L[c]
1967	Total U in urine	Controls for Radiation	0.1 µg U/L
1967	Total U in urine	Teledyne	0.1 µg U/L
1967	Total Pu in urine	Controls for Radiation	0.2 to 0.4 dpm/L
1967	Total Pu in urine	Teledyne	0.3 dpm/L
1968	MFP in urine	Teledyne	30 dpm/L
1968	Total U in urine	Teledyne	0.1 µg U/L
1968	Total Pu in urine	Teledyne	0.3 dpm/L
1968	Pu-239 in feces	Eberline	0.9 dpm/sample[d]
1968	Total Pu in feces	Teledyne	0.3 dpm/g
1968	Total Pu in feces	Eberline	0.3 dpm/g[e]
1969	MFP in urine	No data	50 dpm/L[f]
1969	Gross beta in urine	Eberline	7770 dpm/L[c]
1969	Sr-90 in urine	Eberline	444 dpm/L[c]
1969	U-232 in urine	No data	0.22 dpm/L[g]
1969	U-235 in urine	Eberline	0.22 dpm/L[g]
1969	Total Pu in urine	No data	0.3 dpm/L[h]
1969	MFP in feces	No data	22 dpm/sample[i]
1969	Pu-239 in feces	Eberline	0.9 dpm/sample[d]
1969	Total Pu in feces	Eberline	0.3 dpm/g[e]
1970	MFP in urine	No data	50 dpm/L[f]
1970	Pu-239 in feces	Eberline	0.9 dpm/sample[d]
1970	Total Pu in feces	Eberline	0.3 dpm/g[e]
1971	MFP in urine	No data	50 dpm/L[f]
1971	Total Pu in feces	No data	0.3 dpm/g[e]
1972	MFP in urine	No data	50 dpm/L[f]
1972	Pu-239/240 in urine	Ledoux & Company	0.09 to 0.20 dpm/L
1972	Total Pu in urine	Ledoux & Company	0.20 to 0.40 dpm/L
1972	Total Pu in feces	No data	0.3 dpm/g[e]
1973	Pu-239/240 in urine	No data	0.3 dpm/L[h,j]
1973	Total Pu in urine	No data	0.3 dpm/L[h,j]
1974	MFP in urine	Ledoux & Company	50 dpm/L[f]
1974	Tritium in urine	Ledoux & Company	0.004 µCi/L
1974	Pu-238 in urine	Ledoux & Company	0.07 to 0.58 dpm/L
1974	Pu-239/240 in urine	Ledoux & Company	0.03 to 0.43 dpm/L
1974	Total Pu in urine	Ledoux & Company	0.07 to 0.72 dpm/L
1975	Tritium in urine	Ledoux & Company	0.004 µCi/L[k]
1975	Pu-238 in urine	Ledoux & Company	0.05 to 0.10 dpm/L
1975	Pu-239/240 in urine	Ledoux & Company	0.04 to 0.07 dpm/L
1975	Total Pu in urine	Ledoux & Company	0.09 to 0.16 dpm/L
1976	MFP in urine	Eberline	50 dpm/L[h]
1976	Tritium in urine	Ledoux & Company	0.002 µCi/L
1976	Pu-238 in urine	Ledoux & Company	0.06 to 0.10 dpm/L
1976	Pu-239/240 in urine	Ledoux & Company	0.05 to 0.08 dpm/L
1976	Total Pu in urine	Ledoux & Company	0.07 to 0.14 dpm/L
1976	Pu-238 in urine	Eberline	0.04 to 0.10 dpm/sample
1976	Pu-239 in urine	Eberline	0.05 to 0.07 dpm/sample
1977	Pu-238 in urine	Ledoux & Company	0.07 to 0.58 dpm/L[l]
1977	Pu-239/240 in urine	Ledoux & Company	0.03 to 0.43 dpm/L[l]
1977	Total Pu in urine	Ledoux & Company	0.07 to 0.72 dpm/L[l]

a. MFPs – mixed fission products.

b. Result includes a value of 2 to 3 dpm/L for average ^{40}K excretion for an unexposed individual. This implies ^{40}K was not separated before counting for gross beta.
c. Selected from ORAUT-TKBS-0007-5 for occupational internal dose for Argonne National Laboratory-West) after a review of TBDs showed it to be the highest value for the period of interest [ORAUT 2010a]. This is an MDC rather than a reporting level.
d. Selected from ORAUT-TKBS-0007-5 for occupational internal dose for Idaho National Laboratory after a review of TBDs showed it to be the highest value for the period of interest [ORAUT 2010a]. This is an MDA rather than a reporting level.
e. Assumed same as reported by Teledyne.
f. Assumed high end of range reported by Controls for Radiation in 1967. This bounds the 30-dpm/L value reported by Teledyne in 1968.
g. Reflects the MDCs given in ORAUT-TKBS-0007-5 for occupational internal dose for Argonne National Laboratory-West) and ORAUT-TKBS-0007-5 for occupational internal dose for Idaho National Laboratory) for uranium isotopic analyses [ORAUT 2010a].
h. Assumed the value given by Teledyne in 1968, which should be bounding for later analyses.
i. Reflects the MDA given in ORAUT-TKBS-0007-5 for occupational internal dose for Idaho National Laboratory) for ^{60}Co, ^{90}Sr and ^{134}Cs [ORAUT 2010a].
j. Nelson [1973b, p. 16] reports a minimum detectable level (MDL) of 0.2 dpm/L for plutonium in urine in 1973.
k. Assumed value reported for 1974.
l. Assumed range reported for 1974.

The information in Table 5-3 comes from claims tracking system and other records. The analyses listed should not be interpreted as complete or inclusive. Analytes in addition to those in the table are likely.

5.3.2 U.S. Department of Energy Period, February 26, 1982, to Present

The WVDP continues to use vendors for analysis of in vitro bioassay samples. Required detection levels are specified by the WVDP for different analytes based on the requested turnaround time.

Table 5-4 shows the WVDP required detection levels for in vitro analyses as of September 12, 1990, for different turnaround times. “Priority 1 Processing” means immediate turnaround, Priority 2 was approximately 30 days, and routine processing was approximately 3 months [WVNSC 1990c].

Table 5-4. Required detection levels for WVDP in vitro analyses (dpm/sample unless otherwise noted), September 12, 1990.[a]

Analytical method	Analytes	Priority 1 processing urine	Priority 1 processing fecal	Priority 2 processing urine	Priority 2 processing fecal	Routine processing urine
Pu isotopic	Pu-238/239/240	8.0E–02	6.0E+01	2.0E–02	6.0E+01	2.0E–02
Am-241	Am-241	8.0E–02	3.3E+01	2.0E–02	3.3E+01	2.0E–02
Cm isotopic	Cm-243/244/245	1.2E+00	3.4E+01	2.0E–02	3.4E+01	2.0E–02
Th isotopic	Th-228/230/232	1.0E–01	3.0E+00	1.0E–01	3.0E+00	No data
Sr-90	Sr-90	3.4E+01	2.2E+04	3.4E+01	2.5E+04	3.4E+01
Sm-151	Sm-151	1.2E+02	3.1E+05	1.2E+02	3.1E+05	No data
Eu isotopic	Eu-154/155	8.0E+00	2.2E+04	8.0E+00	2.2E+04	No data
Total uranium	Total U	1.0 µg/sample	240 µg/sample	1.0 µg/sample	240 µg/sample	1.0 µg/sample
U isotopic	U-233/234/235/238	1.0E–01	3.0E+01	9.0E–02	3.0E+01	No data

a. Source: WVNSC [1990c].

Table 5-5 shows the WVDP required detection levels for priority and routine in vitro samples that were to be implemented “on or about” August 1, 1997 [Kubiak 1997, pp. 48–49].

Table 5-5. Required detection levels for priority and routine in vitro samples, August 1, 1997.[a,b]

Analytical method	Urinalysis priority	Urinalysis routine	Fecal analysis priority
Pu isotopic	2.0E–01 dpm/L	2.0E–02 dpm/L	6.0E+01 dpm/sample
Am-241	2.0E–01 dpm/L	2.0E–02 dpm/L	3.3E+01 dpm/sample
Th isotopic	1.0E–0 dpm/L	N/A	3.1E+00 dpm/sample
Sr-90	8.9E+00 dpm/L	1.1E+00 dpm/L	3.1E+00 dpm/sample
Total U	2.0E–04 µg	2.0E–04 µg	5.3E+08 dpm/sample
U isotopic	1.0E–01 dpm/L	N/A	3.1E+01 dpm/sample

a. Source: Kubiak [1997, pp. 48–49].
b. There was no routine in vitro fecal analysis.

Table 5-6 shows the WVDP required detection levels for in vitro analyses as of March 14, 1998, for different turnaround times [WVNSC 1997].

Table 5-6. Required detection levels for WVDP in vitro analyses, March 14, 1998.[a,b,c]

Analytical method	Analytes	Urinalysis priority (dpm/L)	Urinalysis routine (dpm/L)	Urinalysis follow-up (dpm/L)	Fecal analysis priority (dpm/g)
Pu isotopic	Pu-238	2.2E–02	2.2E–02	2.1E–02	2.2E–03
Pu isotopic	Pu-239/240	Not applicable	Not applicable	Not applicable	Not applicable
Sr-90	Sr-90	8.9E+00	1.1E+00	9.1E–01	2.2E–02
KPA	Total uranium	4.4E+05	4.4E+05	4.4E+05	Not applicable
Am-241	Am-241	2.2E–02	2.2E–02	2.1E–02	2.2E–03
Th isotopic	Th-228	2.2E–02	2.2E–02	2.1E–02	3.8E–03
Th isotopic	Th-230	Not applicable	Not applicable	Not applicable	Not applicable
Th isotopic	Th-232	Not applicable	Not applicable	Not applicable	Not applicable
U isotopic	U-233/234	2.2E+01	3.3E–02	3.3E–02	4.4E–03
U isotopic	U-235	Not applicable	Not applicable	Not applicable	Not applicable
U isotopic	U-238	Not applicable	Not applicable	Not applicable	Not applicable
Gamma spec	Cs-137	3.8E+01	3.8E+01	Not applicable	Not applicable

a. Source: WVNSC [1997].
b. KPA – kinetic phosphorescence analysis.
c. There was no routine in vitro fecal analysis.

As of September 12, 1990, if WVDP received an in vitro result showing a concentration greater than the required detection level, an evaluation was performed to determine if the result should be considered positive. A result greater than three times the 1-sigma error was considered positive and a confirmatory sample was collected. If the confirmatory sample also showed positive activity, then a dose assessment was performed. Results less than three times the 1-sigma error were considered nondetections [WVNSC 1990c]. This practice continued until July 1997.

In July 1997 WVDP revised its decision criterion for positive in vitro results to one based on the standard deviation observed for a series of blank samples. An in vitro result greater than two times the standard deviation for the representative blank was considered positive [Kubiak 1997]. This was subsequently revised further to assert decision levels as a fraction of minimum detectable amount (MDA). The WVDP decision levels as a fraction of MDA are given in Table 5-7. These values are as of November 23, 1998 [Kubiak 1998].

Table 5-7. WVDP decision levels for positive in vitro samples, November 23, 1998.[a]

Analysis	Urine detection limit	Fecal detection limit
Sr-90	≥0.93 MDA and ≥4.9E–10 µCi/mL	Not performed
Total uranium	≥3.8E–05 µg/mL	Not performed
Pu isotopic	Pu-238 ≥0.77 MDA	Pu-238 ≥0.71 MDA
Am-241	≥0.71 MDA	Not performed
Th isotopic	Any Th result ≥0.71 MDA	Not performed
U isotopic	U-234 and U-238 ≥0.71 MDA	Not performed
U isotopic	U-234/U-238 >3 or	Not performed
U isotopic	U-234 ≥0.71 MDA	Not performed
U isotopic	U-238 <0.71 MDA	Not performed

a. Source: Kubiak [1998].

5.4 IN VIVO ANALYSES

5.4.1 Operations and Residual Radiation Periods

5.4.1.1 Chest Counter

In a December 1966 letter Helgeson states he felt that the NFS chest counter would see levels “much less” than 1 µCi for typical fission and activation products, and most likely levels “less than a few tens of nanocuries” [Helgeson 1966].

Results from the NFS chest counter are sometimes reported in nanocuries of ^{137}Cs and fraction of maximum permissible lung burden (MPLB). Other reports of chest counter data show total counts, a background count, the net count rate, and "relative lung burden" computed from the net count rate. Example values of relative lung burden are 20.17 and 13.17. Relative lung burden might be the same as percent MPLB, but this has not been verified.

5.4.1.2 Commercial Mobile Whole Body Counting

December 1, 1966
Two West Valley workers were counted on December 1, 1966. Positive activity was reported for ^{40}K, ^{60}Co, ^{95}Zr/Nb, and ^{137}Cs for both individuals. Positive ^{106}Ru/Rh was reported for one of them. No ^{131}I was observed in either individual [Helgeson 1966].

April 17 to 20, 1967
Helgeson Nuclear Services performed whole body counts on 160 radiation workers at West Valley April 17 to 20, 1967 [Wenstrand 1967c]. “Trace quantities of gamma emitting nuclides other than potassium-40 and cesium-137 were detected in 22% of the individuals counted. The maximum deposition was from Zr/Nb-95.” Nuclides identified were ^{60}Co, ^{65}Zn, ^{58}Co (^{134}Cs), ^{95}Zr/Nb, ^{137}Cs, and $^{103/106}$Ru. It was stated “All Co-58 results may be interpreted as Cs-134” [HNS 1967]. Approximately one-third of Production, Maintenance, and Health and Safety personnel showed detectable internal deposition of radionuclides [Keely 1967b].

The mobile whole body counter used was a scanning bed counter in a 24-ft van with an 8- by 4-in. NaI detector. Individuals were counted for approximately 8 minutes [HNS 1967].

Table 5-8 shows sensitivity data for the April 1967 whole body counting for “typical” background at 2.326 sigma (99% confidence) [HNS 1967].

Table 5-8. In vivo counting sensitivity data, April 1967.[a]

Isotope	Background (cpm)	MDA
K-40	72.4	8.8 g
Co-60	38.7	0.81 nCi
Zn-65	57.8	2.73 nCi
Co-58	73.9	1.70 nCi
Zr/Nb-95	93.1	1.28 nCi
Cs-137	129.8	1.89 nCi
Ru-103/106	333.6	2.22 nCi

a. Source: HNS [1967].

The format of the reports for the April 1967 in vivo counting is as follows:

- Last name, First name + Middle initial, Social Security Number;
- Weight in pounds followed by seven values of net count rate (cpm) for ^{40}K, ^{60}Co, ^{65}Zn, ^{58}Co, ^{95}Zr/Nb, ^{137}Cs, and $^{103/106}$Ru, respectively;
- ^{40}K result in grams of potassium;
- ^{60}Co result in nanocuries;
- ^{65}Zn result in nanocuries;
- ^{58}Co result in nanocuries;
- ^{95}Zr result in nanocuries;
- ^{137}Cs result in nanocuries; and
- $^{103/106}$Ru result in nanocuries.

Note that reported results might be negative.

April 15 to 17, 1968

One hundred twenty-eight (128) people were counted by the mobile whole body counting vendor from April 15 to 17, 1968. “The very high background encountered caused severe disturbances in the low energy end of the spectrum, below approximately 0.4 MeV.” This affected the sensitivity for nuclides with primary gamma lines below this energy, such as ^{131}I, ^{51}Cr, and $^{141/144}$Ce. The vendor estimated, based on visual inspection of the spectra, that none of those isotopes was present at a level greater than or equal to 10% of the MPBB. “The background at the whole-body counter was the highest in our experience.” This was said to render their simultaneous equations for evaluating results “useless” [HNS 1968].

The format for 1968 in vivo counting reports is as follows [HNS 1968]:

- Name and Social Security Number; and
- Results: nuclide, units; result; two sigma confidence level; percent of MPBB or lung burden if determined to be insoluble. For potassium, the last value is the ratio of the grams of potassium to the individual’s body weight in kilograms.

April 17 to 18, 1969

The commercial provider was on site for mobile whole body counting from April 17 to 18, 1969. Only 29 people were counted. “Low levels of typical mixed fission and corrosion products were seen in a number of people” [HNS 1969a]. “Eight people were measured for americium-241 and small photopeaks were visualized in the graphs in each of the eight. In every case, however, the level of activity was below what we define as our minimum sensitivity of 0.5 nanocurie.” One individual was also counted for ^{235}U. The ^{241}Am and ^{235}U counts were performed using a 0.5- by 8-in. NaI detector. These were lung counts as opposed to whole body counts [HNS 1969a]. The sensitivity for ^{235}U was asserted to be 50 µg [HNS 1969a]. The 50-µg sensitivity for ^{235}U was asserted to be at a confidence level of “about 80%” [HNS 1969b]. The ^{241}Am sensitivity is said to be 0.5 nCi ±0.5 nCi at 2 sigma.

The in vivo counts (whole body and lung) for April 1969 were performed at the “schoolhouse,” which afforded a lower background than that experienced in previous years [HNS 1969a].

The results reports for 1969 should be self-explanatory. Table 5-9 shows sensitivity data for an 8-minute count using a background measured April 17, 1969, and assuming 140 g of potassium plus 10 nCi of ^{137}Cs in an individual. Sensitivity was defined as three times standard deviation of background divided by the calibration factor [HNS 1969a].

Table 5-9. Sensitivity data for April 1969 in vivo counts.[a]

Isotope	Background (cpm)	Sensitivity
Potassium	90.81	22.5 g
Sb-124	41.93	2.3 nCi
Sb-125	224.88	7.4 nCi
Ba/La-140	56.71	2.0 nCi
Ce-144	497.99	27.3 nCi
Cs-134	153.32	2.3 nCi
Cs-137	186.77	2.6 nCi
Cr-51	351.97	29.0 nCi
Co-58	147.61	2.3 nCi
Co-60	82.12	1.7 nCi
Au-198	335.80	3.0 nCi
I-131	282.42	2.4 nCi
Ir-192	253.41	2.1 nCi
Mn-54	150.29	2.1 nCi
Hg-203	547.64	5.1 nCi
Ru-106	299.41	12.7 nCi
Se-75	247.67	5.2 nCi
Ag-110m	126.29	2.2 nCi
Sr-85	299.41	2.9 nCi
Ta-182	118.08	2.4 nCi
Tc-99m	473.99	1.9 nCi
Sn-113	185.98	3.0 nCi
Zn-65	103.94	4.5 nCi
Zr/Nb-95	167.63	2.7 nCi

a. Source: HNS [1969].

May 10 to 11, 1971

The commercial provider was on site for mobile whole body counting from May 10 to 11, 1971. Only 11 people were counted [HNS 1971]. It appears they were all counted for mixed fission and activation products (MFAPs), ^{241}Am, and ^{239}Pu. Two of the 11 persons showed positive lung burdens of ^{239}Pu/^{241}Am. The counts for MFAPs were still performed using the scanning bed counter. The

^{239}Pu/^{241}Am counts were lung counts performed using a pair of 5-in.-diameter dual-scintillator (phoswich) detectors.

The "sensitivities" reported for mixed fission and activation products for the May 1971 in vivo counting were essentially the same values as from 1969 [HNS 1971]. Sensitivities for plutonium counts were asserted for each individual based on estimates of chest wall thickness.

July 17 to 18, 1972
The commercial provider was on site for mobile whole body counting July 17 to 18, 1972. Eleven individuals were lung counted for ^{239}Pu and ^{241}Am, and four were counted for MFAPs (whole body). Due to a miscommunication, the persons counted for TRU elements were not counted for MFAPs, though they were supposed to have been. The lung counts for TRU elements were performed using the same phoswich detectors used in 1971. A 0.5- by 8-in. NaI was also used "for some studies in the lower energy regions" [HNS 1972]. The initial results indicated one positive ^{241}Am count and two possible plutonium depositions [Wenstrand 1971d].

The sensitivity values for the July 1972 counts do not differ significantly from those given above for the 1969 counting.

5.4.1.3 New York University Medical Center Phoswich Counter

West Valley sometimes sent individuals involved in significant plutonium intakes for follow-up in vivo counting using the phoswich system operated by the NYU Medical Center in Rochester, New York. The Medical Center had two counters: an 8- by 4-in. NaI and a CsI-NaI phoswich detector. Individuals were counted for plutonium only unless sufficient time had elapsed since intake for ingrowth of ^{241}Am. Counts were performed of the head, chest, or liver.

In November 1967 the minimum sensitivity of the NYU plutonium counter was said to be an equivalent of 8 nCi of plutonium in the whole body based on a skull count [Lewis 1967e]. As of July 1968 NYU stated the detection limit of their counter for plutonium was 1.0 nCi in the skull, corresponding to approximately 10 nCi in the whole body, and 2.0 nCi in the chest or liver [Laurer 1968].

5.4.2 U.S. Department of Energy Period

When DOE took over responsibility for operations at West Valley in February 1982, a bed-type lung counter was in use for in vivo monitoring. The counter used a single 4- by 4-in. NaI detector and a 256-channel MCA. The reported MDAs were 35 nCi for ^{137}Cs and 25 nCi for ^{60}Co. The 4-by-4 detector was used through at least February 1987. Later versions of in vivo monitoring procedures describe the lung counter as a 5- by 3-in. NaI detector surrounded by a 1-in. lead shield. This detector was in use by November 1988 [WVNSC 1988]. Johnson [1991] reports MDAs for this system (for lung counts at contact) of 190 Bq (5.1 nCi) for ^{137}Cs and 55 Bq (1.5 nCi) for ^{60}Co. The lung counting system was only calibrated for ^{137}Cs and ^{60}Co.

As of May 12, 1993, the in vivo monitoring program had transitioned to a high resolution, standup scanning whole body counter using hyperpure germanium detectors [WVNSC 1993]. The MDA for this system is 7 nCi of ^{137}Cs, which is used as the tracer analyte in the WVDP in vivo monitoring program [Kubiak 1997, p. 26]. Currently count-specific MDAs are reported for the nuclides in the counting library used (e.g., mixed fission and activation products). Hoffman [1997] states the in vivo counter can detect internal deposition of gamma-emitting nuclides "at a level much less than 10 nanocuries" and less than 1% of the annual limit on intake. At the time the in vivo laboratory participated in intercomparison studies.

5.5 INTERPRETATION OF BIOASSAY MEASUREMENTS

Tables 5-10, 5-11, and 5-12 show nuclide mix information for the MFAP, uranium, and plutonium source terms at West Valley as of January 1, 2000. These data were derived from [Kubiak 1999] by converting CEDE fractions to activity fractions. The alpha activity fractions for the plutonium mixture given in Table 5-12 were approximated by removing the ^{241}Pu activity and renormalizing the remaining constituents.

Table 5-10. Nuclide activity fractions for West Valley MFAP source term (%), January 1, 2000.[a]

Nuclide	Activity fraction
Sr-90	46.73
Cs-137	51.87
Pu-238	0.05
Pu-239	0.01
Pu-240	0.03
Pu-241	0.42
Am-241	0.74
Am-243	0.02
Cm-244	0.12

a. Source: Kubiak [1999].

Table 5-11. Nuclide activity fractions for West Valley uranium source term (%), January 1, 2000.[a]

Nuclide	Activity fraction
Ra-226	0.00
Th-228	0.97
Th-229	0.02
Th-230	0.02
U-232	0.93
U-233	5.63
U-234	61.87
U-235	2.44
U-236	8.88
U-238	19.24

a. Source: Kubiak [1999].

Table 5-12. Nuclide mix information for West Valley plutonium source term (%), January 1, 2000.[a]

Nuclide	Activity fraction	Alpha activity fraction
Pu-238	10.55	44.65
Pu-239	2.86	12.10
Pu-240	2.18	9.23
Pu-241	76.36	0
Am-241	8.04	34.02

a. Source: Kubiak [1999].

The plutonium mixture in Table 5-12 is representative of aged material. Freshly separated plutonium during the operations period would not show appreciable ^{241}Am content and would be best represented by the activity fractions given in Table 2-26. Table 5-13 summarizes the alpha activity fractions to use for fresh and aged plutonium mixtures at West Valley. The mixture for the fresh

material is the 1971 data from Table 2-26. Absorption Type M or S should be selected based on what is known about the material involved in an intake. If the chemical form is unknown, then both Type M and S should be used to see which results in the largest organ dose. The absorption type indicated by bioassay measurements should be used in cases where there are a sufficient number of measurements to make a determination. The absorption type selected for ^{241}Am should be the same as for the corresponding plutonium matrix, i.e., either Type M or Type S.

Table 5-13. Nuclide mix for freshly separated and aged plutonium, alpha activity fraction (%) at West Valley.

Nuclide	Alpha activity fraction for plutonium mixtures, freshly separated material	Alpha activity fraction for plutonium mixtures, aged material
Pu-238	58.1	44.65
Pu-239	24.7	12.10
Pu-240	17.1	9.23
Pu-241	0.1	0
Pu-242	0.02	Not reported
Am-241	0	34.02

The same approach described above for selection of absorption type for plutonium should be applied for uranium and thorium. Absorption type should be selected based on which one yields the higher dose to the organ of interest unless there is sufficient bioassay data with which to make a determination. For uranium, users should select between absorption Types F, M or S. For thorium users should select between absorption Types M or S.

The nuclide mix information for MFAPs given in Table 5-10 is representative of aged material. Fresh material would show additional, shorter-lived nuclides. These are discussed in Section 2.7.1. Specific nuclide mix information that would aid in the evaluation of in vitro analysis results for gross beta or mixed fission products (MFPs) is not available. Keely and Wenstrand [1971] describe an incident involving inhalation of MFAPs by two workers. In vivo counting performed 3 days after intake showed the activity to be predominantly ^{95}Zr/Nb. The next largest contributor was ^{137}Cs, followed by ^{106}Ru/Rh and ^{60}Co. However, these in vivo results do not represent the activity mix of the inhaled material or consider other contributors such as isotopes of strontium, yttrium, cerium, and promethium.

The internal monitoring program at West Valley during operations was designed to detect chronic levels of intake corresponding to a threshold of 25% of applicable legal (10 C.F.R. pt. 20) limits. Radiation workers were included in routine in vitro and in vivo analysis programs, but the frequency and sensitivity of this monitoring was established relative to the 25% threshold for mandatory personnel monitoring. The routine in vivo monitoring program made use of the chest counter until mid-1972. The counter was calibrated to count ^{137}Cs activity, which is cleared rapidly from the body after intake. The in vitro monitoring program for MFPs would also have had limited sensitivity for non-chronic exposures.

Dose reconstructors can use ORAUT-OTIB-0018 [ORAUT 2022] as appropriate for assigning internal dose for unmonitored workers or those for whom monitoring data are missing or incomplete. However, limitations set forth by the SEC (see Section 1.3 above) state that there is insufficient internal bioassay data and accurate workplace monitoring data to reconstruct internal dose from uranium from January 1, 1969, through December 31, 1973 and from mixed fission products for 1972 and 1973 [Azar 2019, NIOSH 2019]. Therefore, under Section 6.3 of ORAUT-OTIB-0018 [ORAUT 2022], do not use ORAUT-OTIB-0018 to bound internal doses from uranium during the SEC period from January 1, 1969, through December 31, 1973 and from mixed fission products for 1972 and 1973; use this document according to its stated applicability and limitations.

6.0 OCCUPATIONAL EXTERNAL DOSE

6.1 BASIS OF COMPARISON

From review of the available reference material, it appears external whole body dose data for persons monitored at West Valley have always been reported in rem or fractions thereof. This includes dose to the skin of the whole body. However, the calibration methods used by the commercial dosimetry vendors or by the WVDP before becoming accredited under the DOE Laboratory Accreditation Program are currently unknown. The WVDP external dosimetry program first became DOE-accredited for whole body dose in January 1990 [16]. Users should therefore compute organ dose using conversion factors from exposure through 1989. From 1990 forward organ dose conversion factors from deep dose equivalent should be used. Skin dose measurements, if needed, should be used as reported.

6.2 SITE HISTORICAL ADMINISTRATIVE PRACTICES

NFS used vendors to process its whole body personnel monitors from the onset of monitoring throughout its tenure at the site. No definite date has been established for when personnel monitoring began. The transition of responsibility for the site from NFS to DOE under the West Valley Demonstration Act also saw a transition to on-site processing of personnel whole body dosimetry. This practice continues to the present. A vendor is used for extremity monitoring.

6.2.1 Site Access Requirements

NFS Radiation Work Permits prescribe the following personnel dosimetry requirements [NFS 1973a, p. 3]:

- For plant entry: film badge;
- For entry into Radiation Areas: film badge, dosimeters and current dose rate survey; and
- For entry into High Radiation Areas: film badge, dosimeters and current dose rate survey.

NFS [1976] states all “employees and workers” entering radiation areas were required to wear film badges in addition to the indirect reading dosimeters. NFS [1974b, p. 40] states, “Each permanent employee is issued a film badge (and dosimeter) ... at the Guard House as he enters [the] exclusion area and he is to wear it all times in the plant areas.” This is echoed in [Nelson 1973b, p. 14], which says, “First line control over the issuance of personnel monitoring devices is exercised at the Security Area Guard House.” Dosimeters and badges worn by people “normally assigned to radiation work” were to be returned to the badge racks after each shift [NFS 1974b, p. 40]. Temporary badges were supposed to be assigned if a worker lost his badge [Runion 1967b, p. 3].

It appears the same dosimetry requirements applied to workers in the waste disposal areas. Workers in the area now known as the SDA wore “radiation badges and gamma dosimeters” (i.e., film badges and pocket meters, respectively) [Facilities description, no date].

NFS [1976] states neutron exposures were encountered in the FRS and the plutonium loadout area. “Neutron film badges” were therefore used in those areas. NFS [1974b, p. 61] states, “Special film badges sensitive to neutrons will be worn by personnel working in plutonium loadout and shipping areas when directed to do so. A satisfactory alternate procedure will be to place neutron films in typical locations and calculate exposures” using occupancy times. “On lower burnup fuels, Health and Safety will waive these requirements.”

NFS' practice was to assign extremity dosimeters (TLDs) when extremity dose rates were expected to exceed those for the skin of the whole body [Duckworth 1973].

A memorandum titled "Personnel Accountability and Dosimetry" [Heacker 1972] discusses concerns about personnel accountability at West Valley during an accident scenario vis-à-vis their film badges. It states they could not be certain an individual had on the right badge or if they had a badge at all, and that the sign-in procedure was not rigorously followed. It mentions a "brown and white nuisance badge," but says not all employees wore them. It also says badges were checked for contamination each shift at the guardhouse, but they could still have undetected contamination inside them. The impetus for these statements was to emphasize the need, in Heacker's view, for photo badges such as those used at AEC sites. He used the photo badge in use at the time at Oak Ridge National Laboratory as an example. NFS' records show isolated cases of personnel picking up the wrong dosimeter at the storage rack on their way in to the plant, but it does not appear this was a common occurrence or that it happened at a disproportionate frequency relative to other facilities where a similar number of people were badged. Once recognized, cases where individuals wore the wrong dosimeter were addressed by sending the badges for processing and assigning new badges to the affected individuals. After processing, the reported dose was assigned to the appropriate individual. Dose could also be assigned through the use of survey data or pocket chamber readings. Concurrence was sought from the affected individuals when such dose estimations were made.

Under the current program, dosimetry is issued to radiation workers as follows, as "determined by the employee's potential exposure" [Hoffman 1997]:

- Radiation Worker I, quarterly exchange;
- Radiation Worker II, quarterly exchange;
- Radiation Worker I, monthly exchange; and
- Radiation Worker II, monthly exchange.

Dosimetry is only issued to workers who meet the following criteria:

- Have completed a baseline whole body count or bioassay (or both),
- Are included on the periodic whole body count or bioassay roster (or both),
- Have completed radiation worker training,
- Have completed General Employee Training,
- Have no radiological restrictions, and
- Have their radiation dose history on file or on request by the Radiation Protection Department.

Workers who are badged monthly are also issued direct-reading dosimeters, which along with the TLDs, are worn continuously by workers in Radiation Control Areas. In some cases, electronic dosimeters are used in addition to the TLDs and direct-reading dosimeters [Hoffman 1997]. These are used to supplement external dose information obtained from TLDs, and do not serve as a primary dosimeter.

Visitor badges are issued to those having a "valid need" to enter areas requiring dosimetry.

Supplemental dosimetry, if used, is placed at the location of the highest anticipated dose rate to monitor localized exposures (to extremities, for instance).

During construction of the RHWF, all construction workers were trained as Radiation Workers and were issued a TLD while they were in the RHWF Radiological Buffer Area. Individuals who were expected to spend less than 80 hr/quarter in the buffer area were classified as visitors and were not Radiation Worker trained or issued dosimetry. Some means of tracking the total time spent in the RHWF buffer area was supposed to be in place and if a visitor reached the 80-hour limit, they could be subject to Radiation Worker training and issued dosimetry [Crotzer 2000, p. 2].

6.2.2 Personnel Exposure Control

Personnel exposures at West Valley during plant operation were controlled relative to the legal maximums for *quarterly* exposure at that time:

- 3 rem to the whole body (or 5 rem/yr for “younger employees” [Keely 1968a],
- 7.5 rem to the skin of the whole body, and
- 18.75 rem to the extremities.

Personnel exposures were such that NFS had to utilize a substantial number of contract employees each quarter once the regular employees reached their limits. NFS used upwards of 1,000 contract employees per year, relative to a permanent staff of under 200. Controlling exposures to lower limits would have required substantial increases in personnel [e.g., Wenstrand 1971e].

In response to a number of extremity overexposures, NFS revised its SWPs in the third quarter of 1967 to include limits for skin and extremity exposures [Lewis 1967e]. Their Health and Safety Manual had been amended to include daily limits for skin and extremity exposure in April of that year [Keely 1967b]. The individuals receiving extremity overexposures were maintenance or mechanical operators performing decontamination and maintenance on mechanical processing equipment such as manipulators. Lead gloves therefore procured for such tasks. Keely and Wenstrand [1971] says 20,000-V lineman’s gloves were also used, though he does not indicate when this practice began. He also says if the gloves were too cumbersome to wear, workers changed their cotton gloves “every few minutes” to reduce the build up of contamination.

The individuals affected by the 1967 extremity overexposures were “removed from work which could give any further significant extremity dosage,” so it does not appear that they were removed from radiation work altogether [Lewis 1967e].

In a February 24, 1971, letter to its headquarters, NFS points out the majority of over-exposures at West Valley over its operating history were from chronic exposure and not acute events. In September 1969 they made efforts to reduce the rate of accumulated dose, but over-exposures still occurred. The practice was modified in the first quarter of 1970 by putting workers on dose rate restrictions once they exceeded 80% of their quarterly limit [Duckworth 1971c]. Thus, individuals were placed on dose rate restriction when they reached 2,400 mrem whole body, 6,000 mrem skin, or 16,000 mrem extremity dose in an effort to “help control chronic over-exposures” [Wenstrand 1970b]. Exposure control practices were changed again in the second quarter of 1970 to where workers were placed on daily dose restrictions once they exceeded 80% of their limit and total radiation work restriction when they reached 90% [Duckworth 1971c].

As of May 1971, NFS controlled dose to contract employees to the following quarterly limits [Metzler 1971]:

- 2 rem to the whole body,
- 4 rem to the skin of the whole body, and
- 13 rem to the extremities.

NFS began a practice of summing and recording dose results from pocket chambers and extremity TLDs for contract employees for comparison with these limits. The impetus was frequent occurrences of these individuals being called in for work who were already on dose restrictions [Metzler 1971].

NFS points out the lag time between collection and processing of film badges contributed to personnel exposure problems. They began sending their film badges to the processor by Special Delivery, which reduced the lag time to about 2 weeks [Duckworth 1971c]. In 1970, NFS was encouraged by its insurance carrier to find ways to reduce the delay time for the processing of film badges and to consider doing its own processing. Accumulated dose (for the purpose of determining individuals to place on work restriction) was based on film badge data when available and "dosimeter" (i.e., pocket chamber) data if not [Keely 1970a]. A May 1972 insurance inspection report [Nelson 1972, p. 10] also mentions the long delay times for processing personnel dosimetry, saying the turnaround is "about three weeks from mailing to receipt of results." This is said to be a decrease from what delays used to be [Nelson 1972, p. 10].

6.2.3 Evaluation and Recording of Personnel Exposures

6.2.3.1 Dose Recording Practices

NFS determined skin and whole body dose from film badges, and extremity dose from TLD finger dosimeters [NFS 1974b, p. 20]. The film badge was the dosimeter of record. If the film had "an obviously incorrect exposure," supplementary data, such as dosimeter readings and surveys were used to determine the exposure [NFS 1974b, p. 40].

Per NFS, "All persons entering radiation areas or restricted areas identified in 10 CFR 20.202 are required to wear two indirect reading dosimeters with a range of 200 mR" [NFS 1976, p. 75]. These dosimeters were read at the end of each shift and the reading recorded. These readings were totaled each week and transferred to the individual's exposure card. The exposure card contained all of the information required on an AEC Form 5 [NFS 1976]. Self-reading dosimeter results were supposed to have been recorded on the SWP if they exceeded 100 mR. A weekly dosimeter log was also kept [O'Reilly 1975, pp. 12–13]. The indirect reading dosimeter totals provided a temporary value for whole body exposure until film badge results were received [NFS 1976].

Extremity TLDs were issued to those receiving exposures where skin dose was anticipated to exceed the whole body dose. The extremity TLDs issued were to be returned to the Hot Lobby for processing upon completion of the work [Duckworth 1973, p. 2].

For dose to the skin of the whole body, the TLD results were used as a temporary measure until the film badge readings came back. For dose to the extremities, the TLD reading was the dose of record [NFS 1976]. At one point NFS added a TLD disk to the film badge that could be read locally in an effort to predict the film badge reading. They say the TLD disk and the film badge generally agreed to within 30%.

Whole body exposure data was received from the film badge supplier and recorded on an individual's exposure card. Exposure data from pencil dosimeter and TLD readouts was also compiled daily and recorded on an individual's card. Lifetime totals were summarized quarterly [Duckworth 1972b].

Keely [1970b] suggests there were discrepancies between the "dosimeters" (pocket meters) and the film badges. "Overexposures" indicated by the pocket meters were not treated as true dose until the film badge result was available.

Review of "Finger Dosimeter Records" forms from April 1972 showed entries under "Name" also included an indication of whether the dose was "skin" or "ext" (i.e., extremity) [NFS 1972c].

The NFS records include several requests from the site asking the dosimetry vendors to change personnel dose results based on "dosimeter" (pocket chamber) readings [for examples, Keely 1968g,h]. Both ask for personnel monitoring results to be changed based on "dosimeter" readings. The latter asks that a reported dose of 2,290 mR be changed to 165 mR. It is not known if the dosimetry vendors actually made these changes, and if so, where they would appear.

The requests from NFS for changes to film badge readings did not specify what prompted them. None of the requests seen had the corresponding dosimetry reports attached. An April 4, 1967, letter from Runion to the AEC says most rejections of film badges are due to overheating, accidental exposure to light, water damage, uneven exposure, or exposure through the back of the badge. It states if a badge is rejected for any reason, the person's pocket dosimeter readings are summed and added to his previous film badge dose [Runion 1967c, p. 3].

NFS made distinctions between NFS employees and non-NFS (e.g., Benz Construction workers) in its radiation exposure summaries. In the 1970 to 1971 period, Benz workers outnumbered NFS employees by approximately 5:1. NFS often gave detailed breakdowns of radiation exposure associated with specific job functions or tasks, but only for NFS employees. Data for Benz workers was all lumped together. They also appear to have distinguished between contract employees who worked at the reprocessing plant and those who worked at the SDA [Nelson 1973b, p. 15]. This reference states, "Contractor employee exposures are monitored by H&S [health and safety] by daily review of their dosimeter records by a member of the H&S staff."

Much of the external dose information in the claims tracking system claim files for individuals who worked at West Valley under NFS consists of a single sentence asserting total whole body, skin, and extremity dose for their period of employment. No individual monitoring data are provided. Dose reconstructors will therefore need to fractionate the reported total dose over the employment period a number of different ways so that latency is accounted for in the calculation in a manner most favorable to the claimant. This can be done, for example, by loading the reported total dose over the first few years of employment, the last few years, and evenly over the employment period and taking the result that yields the largest probability of causation. However, in doing so the dose assigned for any individual year should not exceed the corresponding legal maximum. Annual legal maximums for the operations period are as follows [e.g., Wenstrand 1971b].

- 12 rem whole body,
- 30 rem skin, and
- 75 rem extremity.

6.2.3.2 Nuclear-Chicago Film Badge Exposure Reports

Nuclear-Chicago reported the "current reading" for an individual under either the heading "X-G-N" or "BETA," with X-G-N referring to X-ray, gamma, and neutron. X-ray plus gamma results and neutron results (if any) were reported separately under the "X-G-N" column, with the X-ray plus gamma result listed above the neutron value. Results in this column were in rem, with a vertical dashed line representing the decimal point. So for example, an entry of "123" to the right of the dashed line would be read as 123 mrem. In contrast, beta dose was reported to the nearest 10 mrem, so only two digits appeared to the right to the right of the vertical "decimal point" line. Hence, an entry of "25" to the right of the line would be read as 250 mrem.

Nuclear-Chicago used an 8000 series number to identify neutron films under the "Film Number" column. Later Nuclear-Chicago exposure reports gave X-ray plus gamma and neutron results in separate columns.

Later Nuclear-Chicago exposure reports indicated a badge type in the second column. These were as shown in Table 6-1.

Table 6-1. Nuclear-Chicago film badge type codes.[a]

Type code	Type
1	X-ray, beta, gamma clip-on
2	X-ray, beta, gamma wrist
3	X-ray, beta, gamma ring
5	Extended range X-ray, beta, gamma
14	X-ray, beta, gamma, and neutron clip-on
24	X-ray, beta, gamma, and neutron wrist

a. Source: Nuclear-Chicago [1970, p. 2].

Frequency codes identified the exchange frequency. It appears a code of "1" indicated a monthly exchange and "5" indicated quarterly.

On the older Nuclear-Chicago reports, values in columns labeled "H" indicated the total number of readings used to arrive at a given total. So for example, the value should be "3" at the end of a calendar quarter for a monthly subscriber. The "H" column only accommodated a single character, so the letters O, A, B, and C were used for 10, 11, 12, and 13, respectively.

Entries in Column "D" indicated cases where badges could not be read at all or evaluated accurately. The codes shown in Table 6-2 were used.

Table 6-2. Discrepancy codes and explanations.[a]

Discrepancy codes used in Column "D" of Nuclear-Chicago Exposure Reports	Explanation
1	The film was not returned within 90 d. Evaluation was not possible due to aging of the sensitive film.
2	Film was apparently exposed to light. No reading was possible.
3	The film was blacked out, indicating either exposure to light or chemical fumes, or an exposure in excess of 600 R.
4	The film was soaked in water, over-heated, or otherwise mishandled in such a way that reading was not possible.
5	The film was exposed unevenly or partially obliterated, meaning the reading will reflect less information than usual, with a corresponding loss of accuracy.
6	Exposure occurred through the back of the badge, meaning a loss of accuracy due to self-shielding.

a. Source: Landauer [1982].

Later Nuclear-Chicago reports include a column for skin dose. This was the sum of the X-ray plus gamma dose and any beta dose.

6.2.3.3 Landauer Radiation Dosimetry Reports

The personnel exposure data in the Landauer reports are largely self-explanatory. Their reporting levels around 1971 [Landauer 1971–1972, p. 3, Item 2] were 10 mrem for X- or gamma rays, 40 mrem for hard beta, 20 mrem for fast neutron, and 10 mrem for thermal neutron. Column 5 indicated the dosimeter type and Column 6 the exposure type.

The dosimeter type codes typically seen for West Valley workers are "J" and "N," corresponding to "Regular beta-gamma body film badge" and "Fast neutron-only body film badge." See Landauer [1971–1972, p. 3] for other dosimeter type codes that might be encountered. Table 6-3 lists the exposure type codes.

Table 6-3. Landauer exposure type codes.[a]

Code	Exposure type
1	Total body
2	Skin dose
3	Right finger
4	Left finger
5	Right wrist
6	Left wrist
7	Other extremity
8	Other total body

a. Source: Landauer [1982, p. 2].

Codes listed in column 4 for discrepancies or other notes are too numerous to include here; see Landauer [1971–1972, p. 3].

6.3 SITE DOSIMETRY TECHNOLOGY

6.3.1 Whole Body Monitoring

Whole body monitoring was performed using film badges from the onset of personnel monitoring through August 1982. At least two film badge vendors were used: Nuclear-Chicago and Landauer. In

October 1970 NFS discontinued film badge service with Nuclear-Chicago and ordered film badges from Eberline Instrument Corporation for a comparison evaluation with the Landauer badges [Keely 1970c]. However, it is not known if another vendor was actually used.

The changeover from film badges to a Harshaw TLD system began in August 1982. The Harshaw TLD system was replaced in September 1986 with a Panasonic TLD system, which is still in use. The external dosimetry program is presently accredited by the DOE Laboratory Accreditation Program [Hoffman 1997].

6.3.2 Extremity Monitoring

NFS used TLD chips for extremity monitoring, which were read on site. These “finger dosimeters” went into service in March 1967 “on personnel exposed to high extremity dose rates” [Keely 1967a, p. 2]. A 1968 memorandum describes using TLD disks in black pouches attached with adhesive tape for extremity monitoring. It states they were good for single use in high exposure rate areas, but did not work well for continuous wearing by analytical laboratory workers [Keely 1968i]. Wenstrand [1968e] says they are “using LiF TLD disks to monitor extremity exposure to laboratory analysts. The analyst is assigned two dosimeters and tapes at the beginning of each shift week and wears the dosimeters during the entire week. During the week, the analyst must change tapes on the dosimeters several times as the adhesive wears off.” NFS is requesting a “plastic ring dosimeter holder” to replace the “tapes.”

A statistical evaluation was made of the extremity exposure of 76 work teams in the MSM Repair Shop during the fourth quarter of 1971. The 76 pairs of extremity dosimeters showed an average difference of 88% with a maximum of 275%. The minimum detectable dose for the extremity TLD was said to be 10 mrem [Duckworth 1973, p. 4].

6.4 WORKPLACE RADIATION FIELDS

6.4.1 Beta and Photon

The West Valley reprocessing plant used the same (PUREX) process as at AEC plutonium production facilities, notably, the Hanford site. The workplace photon energy spectra, therefore, would likewise have been dominated by high-energy photons from mixed fission and activation products, with a low-energy component representing down-scatter. This is shown in the photon energy distribution information provided in the Hanford site profile for the separation plants. The photon energy selections established for the separations plants are 75% >250 keV and 25% 30 to 250 keV [ORAUT 2010b]. The same selections should be used for photon exposures in the reprocessing plant at West Valley. Likewise, beta (electron) exposures should be treated as 100% >15 keV. These selections should be applied for all facilities and areas over the entire time period of concern (1965 – present).

Workers in the waste burial areas received exposures from handling and emplacement of waste containers and items. These included waste materials and items from commercial entities (in the case of the SDA) and HLW from the reprocessing plant. LLW from the reprocessing plant were also disposed of at the SDA. Users should assume the energy groups for photons and electrons as given above for the processing plant [17].

6.4.2 Neutron

Neutrons could have been encountered in the FRS (cask handling operations), the Product Packaging and Handling (PPH), and the Product Packaging and Shipping (PPS) areas from spontaneous fission sources and alpha-neutron reactions. Given a lack of information on workplace neuron spectra at

West Valley, users should assume 100% of any neutron dose was from the 0.1 to 2.0 MeV energy group [18].

Any positive neutron dose reported for West Valley workers based on personnel dosimetry or survey data should be doubled to reflect ICRP Publication 60 radiation weighting factors [ICRP 1991]. This would apply to any missed neutron dose assigned as well.

6.4.3 Summary

Table 6-4 summarizes the Interactive RadioEpidemiological Program (IREP) energy group selections that should be made for West Valley cases.

Table 6-4. Selection of radiation energies for West Valley facilities.

Area	Radiation type	Energy selection	Fraction (%)
All areas	Photon	30–250 keV	25
All areas	Photon	>250 keV	75
All areas	Electron	>15 keV	100
FRS, PPH, PPS	Neutron	0.1–2.0 MeV	100

6.5 MISSED DOSE

Exchange frequencies, minimum detectable levels (MDLs), and maximum annual missed dose are given in the tables below for the personnel dosimetry badges used at West Valley. Table 6-5 provides data for deep dose. Table 6-6 is for nonpenetrating radiation. The MDLs for deep and nonpenetrating dose for the commercial film badge reflect the guidance provided in [ORAUT 2006] and [ORAUT 2005], respectively. The MDLs for the Harshaw and Panasonic TLDs are given in DOE-period dosimetry reports in claims tracking system claim files.

Table 6-5. Exchange frequencies, MDLs, and maximum annual missed deep dose for personnel dosimeters.[a]

Period	Dosimeter	MDL (rem)	Exchange frequency	Maximum annual missed dose (rem)
Through 1981	Commercial film badge	0.040	Weekly (n = 52)	1.040
Through 1981	Commercial film badge	0.040	Monthly (n = 12)	0.240
1982–08/1986	Harshaw TLD	0.015	Monthly (n = 12)	0.090
1982–08/1986	Harshaw TLD	0.015	Quarterly (n = 4)	0.030
09/1986–present	Panasonic TLD	0.010	Monthly (n = 12)	0.060
09/1986–present	Panasonic TLD	0.010	Quarterly (n = 4)	0.020

a. Source: Slawson [1999, p. 124].

Table 6-6. Exchange frequencies, MDLs, and maximum annual missed nonpenetrating dose for personnel dosimeters.[a]

Period	Dosimeter	MDL (rem)	Exchange frequency	Maximum annual missed dose (rem)
Through 07/1982	Commercial film badge	0.050	Weekly (n = 52)	1.300
Through 07/1982	Commercial film badge	0.050	Monthly (n = 12)	0.300
08/1982–08/1986	Harshaw TLD	0.030	Monthly (n = 12)	0.180
08/1982–08/1986	Harshaw TLD	0.030	Quarterly (n = 4)	0.060
09/1986–present	Panasonic TLD	0.030	Monthly (n = 12)	0.180
09/1986–present	Panasonic TLD	0.030	Quarterly (n = 4)	0.060

a. Source: Slawson [1999, p. 124].

An asserted MDL for the commercial film badges used at West Valley has not been identified. Landauer's reporting limit for its film badge was 40 mrem for hard beta.

NFS asserted the MDL for its extremity TLD chips was 10 mrem. It is assumed this value is for penetrating radiation. Slawson (1999) reports the MDL for the Landauer Type U extremity dosimeter ranges from 5.1 to 5.9 mrem and averages 5.5 mrem. Claim files in claims tracking system give a MDL for the Landauer Type U finger ring of 30 mrem/exchange. It is assumed the difference reflects that between penetrating and nonpenetrating radiation. The extremity dosimeters were used on an ad hoc basis, so no annual missed dose can be asserted.

Practices for assigning personnel neutron dosimetry at West Valley during operations have not been identified. If necessary, users should assume a neutron MDL of 80 mrem, consistent with that asserted for neutron film dosimeters in previous site profiles (e.g., Hanford). Any neutron missed dose assigned using this MDL should be doubled to account for ICRP Publication 60 radiation weighting factors [ICRP 1991].

6.6 UNCERTAINTY

Film badges normally overrespond in the workplace except for where plutonium or americium are the primary contributors to external dose [ORAUT 2006]. Dose contributions from low-energy photons are not believed to ever have been prevalent at West Valley given the predominance of photon exposures from the ventilation and acid recovery systems. Use of recorded film badge dose for photon exposure is therefore believed to be favorable to the claimant.

Specific information pertaining to the uncertainty associated with recorded personnel dose at West Valley has not been identified. Uncertainties to assign for positive recorded dose are given in Table 6-7. These have been reproduced from information provided in the external dose TBDs for the Y-12 and X-10 sites [ORAUT 2009, 2007b]. Morgan [1961] provides additional discussion of the uncertainties in film badge data.

Table 6-7. Uncertainties for positive recorded dose (%) at West Valley.

Dosimeter	Period	Photon	Beta	Neutron
Commercial film badge	1965–1982	±30	±30	±50
Commercial TLD	1982–present	±15	±15	±15

7.0 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item.

This site profile was authored by Robert E. Burns, Jr., Certified Health Physicist (CHP). Many of the following attributions, therefore, represent conclusions and impressions drawn from detailed review of records from when the West Valley site was operating.

[1] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
Elevated exposure rates existed throughout the process building and exterior to it during the operations period. Review of operating records showed numerous instances of removable contamination being spread throughout multiple areas of the process building. Contamination incidents sometimes would not be recognized until it reached radiological buffer areas at personnel ingress/egress points.

[2] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
This is apparent through review of plant records and AEC inspection reports. The AEC often

remarked about the radiation protection program being implemented vis-a-vis legal maximums.

[3] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
Elevated exposure rates from acid recovery system components and contamination from major spills affected all areas of the plant. Plant records indicate persons not normally involved in radiation work were sometimes brought in to support tasks in the process building when personnel exposure limits became a constraint. Table 2-7 shows acid recovery supervisors collectively averaged 6 person-rem/yr. between 1970 and 1971. The work area for these individuals was “offices.”

[4] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The NDA is within the DOE fence line, and the WVDP built a Liquid Pre-Treatment System on the northeast corner of the NDA ca. 2004. The purpose of the system was to remove organics (solvent) from groundwater around the NDA.

[5] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The information on the contact dose rate from the unshielded glass logs and the impact of elevated exposure rates on release surveys is from discussions the author had with [redacted] during a site visit in June 2006.

[6] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
These three streams represent the three principal processes when the plant was operating: mechanical and chemical preparation of the fuel, separation of the uranium and plutonium products, and handling of the associated chemical wastes.

[7] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The fuels processed at West Valley consisted of both metal and oxide forms.

[8] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The plutonium product from the PUREX process is in an aqueous nitrate form. Plutonium contamination will undergo oxidation as it ages, reducing its solubility.

[9] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The PUREX process separates the americium present in the feed stream from the plutonium product stream. 241Am slowly grows back in over time from decay of 241Pu.

[10] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The fuels processed at West Valley consisted of both metal and oxide forms.

[11] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The nuclides listed represent those observed in in vivo counting data for operations period workers.

[12] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
This is apparent from review of NFS documents discussing atmospheric releases and correspondence between NFS and the AEC.

[13] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
No references have been identified suggesting NFS ever changed its site access requirements for personnel monitoring for external exposure.

[14] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
170 mrem/yr was the administrative limit for annual dose to unmonitored workers as of 1986 [Roberts 1986]. To be favorable to the claimant, it is being assumed unmonitored individuals received an annual dose equal to this limit for the period 1982 – 1993. Likewise, for 1994 forward an annual dose equal to the 100 mrem/yr limit of 10 CFR Part 835 is being assumed [DOE 1993].

[15] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
[Wenstrand 1968d] is a letter from NFS to Eberline requesting information on their ability to support a fecal monitoring program. Plant records show a routine fecal analysis program was in place as of 1970.

[16] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
Communication between Bob Burns and Paul Szalinski, CHP, Integrated Environmental Management, on March 15, 2007.

[17] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
The photon energy spectra at the waste burial sites are not expected to differ from those in the processing plant in a way that would not be favorable to the claimant.

[18] Burns, Jr. Robert E. ORAU Team. Senior Health Physicist. March 2007.
This neutron energy group is representative of fission neutron spectra and is also most favorable to the claimant.

REFERENCES

10 C.F.R. 20, [1991]. Standards for protection against radiation. [SRDB Ref ID: 89171]

ASDA/NFS [1971]. Plutonium storage agreement. New York: New York State Atomic and Space Development Authority. West Valley, NY: Nuclear Fuel Services. May 3. [SRDB Ref ID: 194482]

Azar AM II [2019]. U.S. Department of Health and Human Services designation of additional members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation Program Act of 2000 designating a class of employees from West Valley Demonstration Project West Valley, New York. Washington, DC: U.S. Department of Health and Human Services, Office of the Secretary. October 25. [SRDB Ref ID: 192285]

Bailey WJ [1990]. West Valley Facility spent fuel handling, storage, and shipping experience. Pacific Northwest Laboratory, Richland, WA: Battelle Memorial Institute. PNL-7436, November. [SRDB Ref ID: 30229]

Bentley W [1972]. 1971 annual report of environmental radiation in New York State. Memorandum to Commissioner Diamond. Albany, NY: State of New York Department of Environmental Conservation. July 7. [SRDB Ref ID: 159051]

Birchler JE [1970]. Plant cleanout between processing of a thorium fuel and a low enriched uranium fuel. Memorandum to J.R. Clark. West Valley, NY: Nuclear Fuel Services. December 16. [SRDB Ref ID: 194486]

Clark JR [1968a]. USAEC compliance inspection. Memorandum to E.D. North. West Valley, NY: Nuclear Fuel Services. July 24. [SRDB Ref ID: 194487]

Clark JR [1968b]. Compliance with AEC letter of December 28, 1968. Memorandum to E.D. North. West Valley, NY: Nuclear Fuel Services. April 8. [SRDB Ref ID: 194488]

Clark JR [1968c]. Untitled memorandum to E.D. North West Valley, NY: Nuclear Fuel Services. January 26. [SRDB Ref ID: 28343]

Clark JR [1968d]. Compliance with AEC letter of December 28, 1968. Memorandum to E.D. North. West Valley, NY: Nuclear Fuel Services. May 21. [SRDB Ref ID: 194489]

Clark JR [1968e]. Criteria for removal of DOG filters. Memorandum to Distribution. West Valley, NY: Nuclear Fuel Services. April 8. [SRDB Ref ID: 194490]

Clark JR [1979]. Historical summary of spent fuel receipts and reprocessing at West Valley. Memorandum to C.B. Woodhall. Rockville, MD: Nuclear Fuel Services. November 5. [SRDB Ref ID: 22767]

Cochran JA, Griffin WR Jr., Troianello EJ [1972]. The observation of airborne tritium waste discharge from a nuclear fuel reprocessing plant. Winchester, MA: U.S. Environmental Protection Agency, Office of Radiation Programs. EPA/ORP 73-1, May.[SRDB Ref ID: 139643]

Crotzer ME [2000]. Thermoluminescent dosimeter (TLD) badge requirements during Remote Handled Waste Facility (RHWF) construction. Memorandum to D.K. Pletz. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. July 6. [SRDB Ref ID: 25642]

Daugherty HF [1986]. Handling and disposition of radioactively contaminated soil. Memorandum to J.L. Knabenschuh. West Valley, NY: West Valley Nuclear Services Company. August 28. [SRDB Ref ID: 25992]

DOE [1993]. Federal Register Vol. 58 No. 238 December 14, 1993 10 C.F.R. Part 835 Occupational radiation protection. Final Rule. Fed Regist *58*(238):65458. [SRDB Ref ID: 176022]

DOE [1999]. Plutonium & uranium recovery from spent fuel reprocessing by Nuclear Fuel Services at West Valley, New York from 1966 to 1972. Washington, DC: U.S. Department of Energy, Office of Defense Programs. Revised, November. [SRDB Ref ID: 139744]

Duckworth JP [1970]. Corrective actions taken in response to routine USAEC inspections conducted at West Valley on December 17, 1969, January 21, February 3, and April 17, 1970. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. West Valley, NY: Nuclear Fuel Services. August 28. [SRDB Ref ID: 194491]

Duckworth JP [1971a]. Quarterly report for January 1, 1971 through March 31, 1971. West Valley, NY: Nuclear Fuel Services. July 29. [SRDB Ref ID: 93367]

Duckworth JP [1971b]. Quarterly report for April 1, 1971 through June 30, 1971. West Valley, NY: Nuclear Fuel Services. July 29. [SRDB Ref ID: 93342]

Duckworth JP [1971b]. Radiation exposure at West Valley plant. Memorandum to E.R. Oppel. West Valley, NY: Nuclear Fuel Services. February 24. [SRDB Ref ID: 194492]

Duckworth JP [1972a]. Quarterly report for October 1, 1971 through December 1, 1971. West Valley, NY: Nuclear Fuel Services. January 31. [SRDB Ref ID: 151078]

Duckworth JP [1972]. Proposed 20.408 compliance procedure. Memorandum to E.J. Heerwagen. Rockville, MD: Nuclear Fuel Services. September 7. [SRDB Ref ID: 194493]

Duckworth JP [1973]. Response to violations identified during inspections conducted on March 15-17 and May 16-19, 1972. Letter to J.P. O'Reilly, U.S. Atomic Energy Commission, Directorate of Regulatory Operations. West Valley, NY: Nuclear Fuel Services. January 10. [SRDB Ref ID: 194494]

Duckworth JP [1976]. Pumpout data of waste burial trenches. Memorandum to W.A. Oldham. West Valley, NY: Nuclear Fuel Services. October 20. [SRDB Ref ID: 194495]

Duckworth JP [1978a]. Estimate input curie content in the commercial burial area. Memorandum to File. West Valley, NY: Nuclear Fuel Services. June 19. [SRDB Ref ID: 194496]

Duckworth JP [1978b]. NFS waste burial history. Memorandum to W.A. Oldham. West Valley, NY. March 27. [SRDB Ref ID: 194497]

Duckworth JP [1980]. Fifty-second quarterly report for the period January 1, 1980 - March 31, 1980. West Valley, NY: Nuclear Fuel Services. April 28. [SRDB Ref ID: 194498]

Facilities description [no date]. [SRDB Ref ID: 194484]

G. [author's initial] [1974]. FRS air samples - week ending 5-15-74. May 27. [SRDB Ref ID: 194499]

Golden MP, Frank DE, Prina ER [1982]. NFS, WVNY, summary of decontamination information. Memorandum to W.R. Jacoby. New York: Westinghouse. January 14. [SRDB Ref ID: 25980]

Greco SR [1978]. Special nuclear and source material in commercial burial trenches. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. June 20. [SRDB Ref ID: 194500]

Heacker HV [1971]. PPC air circulation and filtration. Memorandum to C. Alday. West Valley, NY: Nuclear Fuel Services. November 11. [SRDB Ref ID: 194501]

Heacker HV [1972]. Personnel accountability and dosimetry. Memorandum to B.E. Knight. West Valley, NY: Nuclear Fuel Services. January 21. [SRDB Ref ID: 194502]

Helgeson GL [1966]. Whole body counting results. Letter to R.B. Keely, Nuclear Fuel Services. Pleasanton, CA: Helgeson Nuclear Services. December 8. [SRDB Ref ID: 194503]

HNS [1967]. Report of whole body counting for Nuclear Fuel Services, Inc. Pleasanton, CA: Helgeson Nuclear Services. April 21. [SRDB Ref ID: 194504]

HNS [1968]. Report of whole body counting for Nuclear Fuel Services, Inc. Pleasanton, CA: Helgeson Nuclear Services. May 3. [SRDB Ref ID: 194505]

HNS [1969a]. Report of in vivo counting for Nuclear Fuel Services, Inc. Pleasanton, CA: Helgeson Nuclear Services. June 6. [SRDB Ref ID: 194506]

HNS [1969b]. Announcement of new services development of counting techniques for in vivo U-235 and Am-241. Pleasanton, CA: Helgeson Nuclear Services. January 1. [SRDB Ref ID: 194507]

HNS [1971]. Report of in vivo counting for Nuclear Fuel Services, Inc. Pleasanton, CA: Helgeson Nuclear Services. September 16. [SRDB Ref ID: 194599]

HNS [1972]. Report of in vivo counting for Nuclear Fuel Services. Pleasanton, CA: Helgeson Nuclear Services. October 18. [SRDB Ref ID: 194508]

Hoffman RL [1997]. Radiation safety at the West Valley Demonstration Project. Washington, DC: US Department of Energy, Office of Nuclear Energy. DOE/NE/44139-75, May 6. [SRDB Ref ID: 30272]

ICRP [1991]. ICRP 60 - 1990 recommendations of the International Commission on Radiological Protection. ICRP Publication 60. Ann ICRP *21*(1-3). [SRDB Ref ID: 11517]

ICRP [1994]. ICRP 68 - Dose coefficients for intakes of radionuclides by workers. ICRP Publication 68. Ann ICRP *24*(4). [SRDB Ref ID: 22731]

ICRP [1996]. Age-dependent doses to the members of the public from intake of radionuclides: Part 5 Compilation of ingestion and inhalation dose coefficients. ICRP Publication 72. Ann ICRP *26*(1). [SRDB Ref ID: 11513]

Jablonski JF, Al-Daouk A, Moore H [2003]. Progress on fuel receiving and storage decontamination work at the West Valley Demonstration Project. Waste Management 2003 Conference, February 23-27, 2003, Tucson, AZ. Abstract No. 323. [SRDB Ref ID: 30291]

Jaeger P [no date]. Contamination to Zone III. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 194509]

Jaroszeski RA [1972]. Identification of fission products in FRS pool water. Memorandum to J.M. Jump. West Valley, NY: Nuclear Fuel Services. May 11. [SRDB Ref ID: 194510]

Johnson & Higgins [1975]. Nuclear insurance survey of West Valley Plant. West Valley, NY: Nuclear Fuel Services. May 12. [SRDB Ref ID: 194512]

Johnson P [1991]. Development of derived investigation levels for use in internal dosimetry at the West Valley Demonstration Project. Dissertation. Pittsburgh, PA: University of Pittsburgh. [SRDB Ref ID: 30275]

Keely RB [1966]. Radiation-contamination status report September 1966. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. October 12. [SRDB Ref ID: 194513]

Keely RB [1967a]. Radiation-contamination status report March 1967. Memorandum to R. Wischow. West Valley, NY: Nuclear Fuel Services. April 7. [SRDB Ref ID: 194514]

Keely RB [1967b]. Radiation-contamination status report April 1967. Memorandum to R.P. Wischow. West Valley, NY: Nuclear Fuel Services. May 3. [SRDB Ref ID: 194515]

Keely RB [1967c]. Radiation-contamination status report February 1967. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. April 7. [SRDB Ref ID: 194516]

Keely RB [1967d]. Radiation-contamination status report January, 1967. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. February 6. [SRDB Ref ID: 194517]

Keely RB [1967e]. Radiation-contamination status report December, 1966. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. January 6. [SRDB Ref ID: 194518]

Keely RB [1967f]. Radiation - contamination report June 1967. Memorandum to R.P. Wischow. West Valley, NY: Nuclear Fuel Services. July 6. [SRDB Ref ID: 194519]

Keely RB [1968a]. Radiation-contamination status report June 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. July 1. [SRDB Ref ID: 194520]

Keely RB [1968b]. Radiation-contamination status report August 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. September 2. [SRDB Ref ID: 194521]

Keely RB [1968c]. Radiation-contamination status report September 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. October 1. [SRDB Ref ID: 194522]

Keely RB [1968d]. Radiation-contamination status report October 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. November 5. [SRDB Ref ID: 194523]

Keely RB [1968e]. Radiation-contamination status report May 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. June 3. [SRDB Ref ID: 194524]

Keely RB [1968f]. Radiation-contamination status report, November 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. December 5. [SRDB Ref ID: 194525]

Keely RB [1968g]. Request to change faulty reading for [redacted] badge [redacted] week of [redacted]. Letter to Mrs. Wilson, R. S. Landauer Jr. & Company. West Valley, NY: Nuclear Fuel Services. August 27. [SRDB Ref ID: 194526]

Keely RB [1968h]. Correction to results for [redacted] badge [redacted] week of [redacted]. Letter to Mrs. McDonald, R.S. Landauer Jr. & Company. West Valley, NY: Nuclear Fuel Services. June 26. [SRDB Ref ID: 194527]

Keely RB [1968i]. Purchasing plastic ring badges to use as holders for TLD discs. Letter to Nuclear Service Laboratories. West Valley, NY: Nuclear Fuel Services. October 8. [SRDB Ref ID: 194528]

Keely RB [1968j]. Health and Safety Department report for April 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. May 1. [SRDB Ref ID: 194529]

Keely RB [1968k]. Radiation - contamination status report July 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. August 2. [SRDB Ref ID: 194530]

Keely RB [1969a]. Radiation - contamination status report December 1968. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. January 2. [SRDB Ref ID: 194531]

Keely RB [1969b]. Radiation - contamination status report March 1969. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. April 2. [SRDB Ref ID: 194532]

Keely RB [1969c]. Radiation-contamination status report May 1969. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. June 3. [SRDB Ref ID: 194533]

Keely RB [1969d]. Radiation - contamination status report June 1969. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. July 7. [SRDB Ref ID: 194534]

Keely RB [1969e]. Radiation-contamination status report July 1969. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. August 5. [SRDB Ref ID: 194535]

Keely RB [1969f]. Radiation - contamination status report February 1969. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. February 28. [SRDB Ref ID: 194536]

Keely RB [1969g]. Radiation - contamination status report April 1969. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. May 5. [SRDB Ref ID: 194537]

Keely RB [1969h]. Radiation - contamination status report January 1969. Memorandum to W.G. Urbon. West Valley, NY: Nuclear Fuel Services. February 3. [SRDB Ref ID: 194538]

Keely RB [1970a]. Safety & industrial relations monthly report - August 1970. Memorandum to J. P. Duckworth. West Valley, NY: Nuclear Fuel Services. September 8. [SRDB Ref ID: 194539]

Keely RB [1970b]. Safety and industrial relations monthly report - March 1970. Memorandum to J. P. Duckworth. West Valley, NY: Nuclear Fuel Services. April 6. [SRDB Ref ID: 194540]

Keely RB [1970c]. Safety and industrial relations monthly report - October 1970. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. November 6. [SRDB Ref ID: 194541]

Keely RB, Wenstrand TK [1971]. Impact of fuel reprocessing on the health physics profession. Health Phys *20*(143–151). [SRDB Ref ID: 30270]

Kelleher WJ, Michael EJ [1973]. Low level radioactive waste site burial inventory for the West Valley Site, Cattaraugus County, N.Y. Albany, NY: New York State Department of Environmental Conservation, Bureau of Radiological Pollution Control. June 20. [SRDB Ref ID: 194542]

Kester RT [1971]. Yearly report from bioassay technician. Memorandum to D.P. Wilcox. West Valley, NY: Nuclear Fuel Services. February 22. [SRDB Ref ID: 194543]

Kubiak MS [1997]. WVDP internal dosimetry program manual and technical basis document. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. WVDP-070 Rev. 5, July 28. [SRDB Ref ID: 24027]

Kubiak MS [1998]. WVDP internal dosimetry program manual and technical basis document. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. WVDP-070 Rev. 6, September 14. [SRDB Ref ID: 24028]

Kubiak M [1999]. Interpretation of bioassay measurements. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. RC-DOS-30 Rev. 2 PC5, July 8. [SRDB Ref ID: 24209]

Landauer [1971–1972]. R. S. Landauer Jr. and Co. radiation dosimetry report. Glenwood, IL: R. J. Landauer, Jr. & Company. [SRDB Ref ID: 144318]

Landauer [1982]. Landauer radiation dosimetry report. Glenwood, IL: R. J. Landauer, Jr. & Company. September 10. [SRDB Ref ID: 144318]

Laurer GR [1968]. Results of whole body counting measurements. Letter to R. Keely, Nuclear Fuel Services. New York, NY: New York University Medical Center Institute of Environmental Medicine. August 23. [SRDB Ref ID: 194544]

Lewis WH [1966a]. Status report of West Valley Plant for week ending December 9, 1966. Memorandum to T.C. Runion. West Valley, NY: Nuclear Fuel Services. December 9. [SRDB Ref ID: 194545]

Lewis WH [1966b]. West Valley status report for week ending December 16, 1966. Memorandum to T.C. Runion. West Valley, NY: Nuclear Fuel Services. December 19. [SRDB Ref ID: 194546]

Lewis WH [1967a]. NFS third quarterly report covering activities from October 20, 1966 through January 19, 1967. Letter to J.A. McBride, U.S. Atomic Energy Commission, Materials Licensing Division. West Valley, NY: Nuclear Fuel Services. February 19. [SRDB Ref ID: 93334]

Lewis WH [1967b]. Quarterly report for January 20, 1967 through April 19, 1967. West Valley, NY: Nuclear Fuel Services. May 15. [SRDB Ref ID: 93341]

Lewis WH [1967c]. Quarterly report for April 20, 1967 through June 30, 1967. West Valley, NY: Nuclear Fuel Services. August 14. [SRDB Ref ID: 93355]

Lewis WH [1967d]. Quarterly report for July 1, 1967 through September 30, 1967. West Valley, NY: Nuclear Fuel Services. October 25. [SRDB Ref ID: 93316]

Lewis WH [1967e]. Plutonium inhalation incident on September 9, 1967 (Supplement). Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. December 20. [SRDB Ref ID:194547]

Lewis WH [1967f]. Extremity overexposures (June 30, 1967 to September 25, 1967). Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. West Valley, NY: Nuclear Fuel Services. September 27. [SRDB Ref ID: 194548]

Lewis WH [1968]. Extremity exposure of a [redacted] worker at Nuclear Fuel Services reprocessing facility on December 27, 1967. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. West Valley, NY: Nuclear Fuel Services. January 31. [SRDB Ref ID: 194549]

Loud GC [1965a]. Radiation-contamination status report October, 1965. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. November 9. [SRDB Ref ID: 194550]

Loud GC [1965b]. Radiation-contamination status report September, 1965. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. October 4. [SRDB Ref ID: 194551]

Loud GC [1965c]. Radiation-contamination status report November, 1965. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. December 10. [SRDB Ref ID: 194552]

Loud GC [1966a]. Radiation-contamination status report December 1965. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. January 6. [SRDB Ref ID: 194553]

Loud GC [1966b]. Radiation-contamination status report February, 1966. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. March 7. [SRDB Ref ID: 194554]

Loud GC [1966c]. Radiation-contamination status report March, 1966. Memorandum to W.H. Lewis. West Valley, NY: Nuclear Fuel Services. April 5. [SRDB Ref ID: 134533]

Loud GC [1966d]. Radiation-contamination status report May, 1966. Memorandum to W.H. Lewis, West Valley, NY: Nuclear Fuel Services. June 13. [SRDB Ref ID: 194555]

Metzler GH [1971]. Limiting exposure on Benz personnel. Memorandum to Health & Safety Technicians. West Valley, NY: Nuclear Fuel Services. May 25. [SRDB Ref ID: 194556]

Miller RN [1972a]. NFS responses to safety items identified by AEC. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. Rockville, MD: Nuclear Fuel Services. April 21. [SRDB Ref ID: 194557]

Miller RN [1972b]. Responses of Nuclear Fuel Services, Inc. to AEC noncompliance and nonconformance items. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. Rockville, MD: Nuclear Fuel Services. April 7. [SRDB Ref ID: 194558]

Monroe HL [1968]. Interim report on two cases of [redacted] at West Valley September 9 and 28, 1967. Letter to W.T. Doran, U.S. Atomic Energy Commission. West Valley, NY: Nuclear Fuel Services. March 19. [SRDB Ref ID: 194559]

Morgan KZ [1961]. Dosimetry requirements for protection from ionizing radiation. In: Selected topics in radiation dosimetry. Proceedings of the symposium on selected topics in radiation dosimetry. Vienna, Austria: International Atomic Energy Agency. [SRDB Ref ID: 8345]

Morrow PK [1972a]. Health and safety monthly report (September). Memorandum to J.P. Duckworth, West Valley, NY: Nuclear Fuel Services. October 5. [SRDB Ref ID: 194560]

Morrow PK [1972b]. Health and safety monthly report (October). Memorandum to J.P. Duckworth, West Valley, NY: Nuclear Fuel Services. November 7. [SRDB Ref ID: 194561]

Morrow PK [1972c]. Health and safety monthly report - May, 1972. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. June 7. [SRDB Ref ID: 194563]

Morrow PK [1972d]. Health and safety monthly report - November. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. December 6. [SRDB Ref ID: 194562]

Nelson HA [1972]. Nuclear loss control evaluation Nuclear Fuel Services, Inc., fuel reprocessing plant West Valley, New York. Memorandum to J.F. Miller. West Valley, NY: Nuclear Fuel Services. August 11. [SRDB Ref ID: 194511]

Nelson PR [1973a]. USAEC RO Inspection Report 50-201/72-03 NFS License CSF-1 conducted September 19-23 and 26-29, 1972. Letter to R.N. Miller, Nuclear Fuel Services. Newark, NJ: U.S. Atomic Energy Commission, Directorate of Regulatory Operations. Inspection Report No. 72-03, February 5. [SRDB Ref ID: 194564]

Nelson PR [1973b]. USAEC RO Inspection Report 50-201/73-04 NFS License CSF-1 conducted October 29-31, 1973. Letter to R.N. Miller, Nuclear Fuel Services. King of Prussia, PA: U.S. Atomic Energy Commission, Directorate of Regulatory Operations. Inspection Report No. 73-04, December 7. [SRDB Ref ID: 194565]

Nelson PR [1976]. USNRC IE Inspection Report 50-201/75-09 NFS License CSF-1 conducted December 22-23, 1975. Letter to R.W. Deuster, Nuclear Fuel Services. King of Prussia, PA: U.S. Atomic Energy Commission, Directorate of Regulatory Operations. Inspection Report No. 75-09, January 21. [SRDB Ref ID: 194566]

NFS [1967]. Environmental report January to September, 1967. West Valley, NY: Nucelar Fuel Services. January 24. [SRDB Ref ID: 28756]

NFS [1968]. Report on the failure of a filter in the main ventilation system on September 4, 1968. West Valley, NY: Nuclear Fuel Services. September 27. [SRDB Ref ID: 194771]

NFS [1969]. Plutonium shipping record. West Valley, NY: Nuclear Fuel Services. June 3. [SRDB Ref ID: 194567]

NFS [1972b]. Environmental report 11 July - December 1971. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 93318]

NFS [1972a]. Head End Ventilation system annual performance report. West Valley, NY: Nuclear Fuel Services. May 1. [SRDB Ref ID: 194568]

NFS [1972c]. Finger dosimeter records. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 30262]

NFS [1973a]. Radiation area extended work procedure ANA-1. West Valley, NY: Nuclear Fuel Services. RWP No. ANA-1 (Rev. 1), January 1. [SRDB Ref ID: 194569]

NFS [1973b]. Environmental report, NFS' reprocessing plant, West Valley, New York. Rockville, MD: Nuclear Fuel Services. Docket Number 50-201. [SRDB Ref ID: 29128]

NFS [1974a]. Environmental report, NFS' UF_6 plant, West Valley, New York. Rockville, MD: Nuclear Fuel Services. Docket Number 70 - 071. [SRDB Ref ID: 28431]

NFS [1974b]. Nuclear Fuel Services fuel reprocessing plant, West Valley, New York, health and safety manual. West Valley, NY: Nuclear Fuel Services. Rev. 3, April 30. [SRDB Ref ID: 194570]

NFS [1974c]. Health and safety logbook beginning March 27, 1972. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 24410]

NFS [1974d]. Monthly report radioactive waste burial study. West Valley, NY: Nuclear Fuel Services. February 15. [SRDB Ref ID: 194571]

NFS [1976]. Safety analysis report, NFS' reprocessing plant, West Valley, New York. Vol. I. Rockville, MD: Nuclear Fuel Services. February. [SRDB Ref ID: 27086]

NIOSH [2007]. External dose reconstruction implementation guideline. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. OCAS-IG-001 Rev. 3, November 21. [SRDB Ref ID: 38864]

NIOSH [2010]. Radiation exposures covered for dose reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. DCAS-IG-003 Rev. 1, October 5. [SRDB Ref ID: 88929]

NIOSH [2019]. Special Exposure Cohort petition evaluation report petition SEC-00252 Rev. 0 West Valley Demonstration Project qualified July 31, 2019. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. Petition SEC-00252 Rev. 0, July 31. [SRDB Ref ID: 177848]

North ED, Clark JR [1968]. A study of particulate release rate during the period March 8 - 10, 1968. West Valley, NY: Nuclear Fuel Services. March 24. [SRDB Ref ID: 30271]

Nuclear-Chicago [1970]. Nuclear-Chicago radiation exposure report. Des Plaines, IL. August 14. [SRDB Ref ID: 24460]

NYSERDA [2004]. The Western New York Nuclear Service Center (The West Valley Site). Albany, NY: New York State Energy Research and Development Authority. [SRDB Ref ID: 30288]

Oldham WA [1977]. Recommendation that employees be given an entrance and termination whole body count. Letter to M.W. Stradley, NEL-PIA. West Valley, NY: Nuclear Fuel Services. May 18. [SRDB Ref ID: 194572]

ORAUT [2005]. Interpretation of dosimetry data for assignment of shallow dose. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0017 Rev. 01, October 11. [SRDB Ref ID: 19434]

ORAUT [2006]. A standard complex-wide methodology for overestimating external doses measured with film badge dosimeters. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0010 Rev. 01, June 5. [SRDB Ref ID: 29953]

ORAUT [2006b]. Occupational ambient dose reconstruction for DOE sites. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-PROC-0060 Rev. 01, June 28.

ORAUT [2007a]. Oak Ridge National Laboratory – occupational medical dose. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0012-3 Rev. 02, July 21. [SRDB Ref ID: 35194]

ORAUT [2007b]. Oak Ridge National Laboratory – occupational external dosimetry. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0012-6 Rev. 01, September 10. [SRDB Ref ID: 34945]

ORAUT [2009]. Y-12 National Security Complex – occupational external dosimetry. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0014-6 Rev. 02, December 18. [SRDB Ref ID: 77701]

ORAUT [2010a]. Idaho National Laboratory and Argonne National Laboratory-West – occupational internal dose. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0007-5 Rev. 03, March 2. [SRDB Ref ID: 79571]

ORAUT [2017]. Guidance on assigning occupational x-ray dose under EEOICPA for x-rays administered off site. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0079 Rev. 02, June 15. [SRDB Ref ID: 166967]

ORAUT [2010b]. Hanford Site – occupational external dose. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0006-6 Rev. 04, January 7. [SRDB Ref ID: 77581]

ORAUT [2019]. Dose reconstruction from occupational medical X-ray procedures. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0006 Rev. 06, September 27. [SRDB Ref ID: 178310]

ORAUT [2020]. Estimating doses for plutonium strongly retained in the lung. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0049 Rev. 02, September 1. [SRDB Ref ID: 178329]

ORAUT [2021]. External dose reconstruction. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0088 Rev. 02, June 11. [SRDB Ref ID: 187044]

ORAUT [2022]. Internal dose overestimates for facilities with air sampling programs. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0018 Rev. 02, July 1. [SRDB Ref ID: 193095]

O'Reilly JP [1975]. USAEC RO Inspection Reports 74-10 and 74-02 NFS Licenses CSF-1 and SNM-984 conducted November 18-20, 1974. Letter to R.W. Deuster, Nuclear Fuel Services. King of Prussia, PA: U.S. Atomic Energy Commission, Directorate of Regulatory Operations. January 15. [SRDB Ref ID: 194573]

Personnel exposure following removal of masks during plutonium transfer January 8, 1974 [1974]. Memorandum to G. [illegible]. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 194485]

Petkus LL, Paul J, Valenti PJ, Houston H, May J [2003]. A complete history of the high-level waste plant at the West Valley Demonstration Project. Submission to the Waste Management 2003 Conference, February 23-27, 2003. Tucson, AZ. Abstract #244, February. [SRDB Ref ID: 29382]

Preliminary safety analysis report of the Head End Ventilation system [1969]. Third revision, January 15. [SRDB Ref ID: 29127]

Roberts RS [1986]. Area radiation monitoring. Memorandum to D.J. Harward. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. April 24. [SRDB Ref ID: 156065]

Runion TC [1967a]. Response to letter regarding inspections conducted April and June 1967 not being in full compliance with provisional operating license CSF-1 and SNM-984. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. Wheaton, MD: Nuclear Fuel Services. September 18. [SRDB Ref ID: 194574]

Runion TC [1967b]. Response to request for additional information and/or explanation in relation to the activities of NFS under the provisional operating license CSF-1. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. Wheaton, MD: Nuclear Fuel Services. April 4. [SRDB Ref ID: 194575]

Runion TC [1968]. Confirmation of NFS program to insure the health and safety of employees and the public. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. Wheaton, MD: Nuclear Fuel Services. January 13. [SRDB Ref ID: 194576]

Runion TC [1970]. Statement before the Joint Committee on Atomic Energy, part II, hearings on the environmental effects of producing electric power. Wheaton, MD: Nuclear Fuel Services. February 25. [SRDB Ref ID: 30228]

Site boundary concentrations of alpha and beta releases from FRP stack annual beta sector width averaged [no date]. [SRDB Ref ID: 102209]

Slawson JW [1999]. WVDP external dosimetry technical basis document, program manual, and quality assurance plan. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. WVDP-071 Rev. 7, November 12. [SRDB Ref ID: 24213]

Urbon WG [1968a]. Nuclear Fuel Services Reprocessing Plant personnel exposure August 1968 Gorecki, Casmir. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. West Valley, NY: Nuclear Fuel Services. September 9. [SRDB Ref ID: 194578]

Urbon WG [1968b]. Nuclear Fuel Services Reprocessing Plant personnel exposure first quarter 1968. Letter to L.D. Low, U.S. Atomic Energy Commission, Division of Compliance. West Valley, NY: Nuclear Fuel Services. May 6. [SRDB Ref ID: 194579]

Vance RF [1986]. Characterization of the head end cells at the West Valley Nuclear Fuel Reprocessing Plant. Washington, DC: U.S. Department of Energy, Office of Nuclear Energy. DOE/NE/44139-31, November. [SRDB Ref ID: 162693]

Wenstrand TK [1967a]. Radiation - contamination status report November 1967. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. December 15. [SRDB Ref ID: 194580]

Wenstrand TK [1967b]. Yearly whole body exposure for 1967. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. December 7. [SRDB Ref ID: 194581]

Wenstrand T [1967c]. Supplementary analysis of whole body counts. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 194582]

Wenstrand TK [1967d]. Radiation - contamination status report October 1967. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. November 16. [SRDB Ref ID: 194583]

Wenstrand TK [1968a]. Radiation - contamination status report December 1967. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. January 8. [SRDB Ref ID: 194584]

Wenstrand TK [1968b]. Radiation - contamination status report March 1968. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. April 1. [SRDB Ref ID: 194585]

Wenstrand TK [1968c]. Film badge exposure report Walegura (Waligura). Memorandum to H.T. Daw. West Valley, NY: Nuclear Fuel Services. February 14. [SRDB Ref ID: 194590]

Wenstrand TK [1968d]. Plans for fecal sample program for plant workers entering contaminated alpha areas. Letter to J.J. Mueller, Eberline Instrument Company. West Valley, NY: Nuclear Fuel Services. February 29. [SRDB Ref ID: 194586]

Wenstrand TK [1968e]. Request for plastic ring dosimeter holder for LiF TLD discs to replace current type. Letter to Isotopes. West Valley, NY: Nuclear Fuel Services. September 23. [SRDB Ref ID: 194587]

Wenstrand TK [1968f]. Radiation - contamination status report February 1968. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. March 1. [SRDB Ref ID: 194588]

Wenstrand TK [1968g]. Radiation - contamination status report January 1968. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. February 6. [SRDB Ref ID: 194589]

Wenstrand TK [1968h]. Radiation-contamination status report April 1968. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 194483]

Wenstrand TK [1969a]. Radiation - contamination status report, September 1969. Memorandum to R.B. Keely. West Valley, NY: Nuclear Fuel Services. October 6. [SRDB Ref ID: 194591]

Wenstrand TK [1969b]. Radiation contamination status report August 1969. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. September 3. [SRDB Ref ID: 194592]

Wenstrand TK [1969c]. Radiation-contamination status report - October 1969. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. November 6. [SRDB Ref ID: 194593]

Wenstrand TK [1970a]. Input for duct flush TOP. Memorandum to J. Birchley. West Valley, NY: Nuclear Fuel Services. July 8. [SRDB Ref ID: 194577]

Wenstrand TK [1970b]. Radiation work restrictions. Memorandum to S.A. Kapple et al. West Valley, NY: Nuclear Fuel Services. April 10. [SRDB Ref ID: 194594]

Wenstrand TK [1971a]. Health and safety monthly report - October 1971. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. November 5. [SRDB Ref ID: 194595]

Wenstrand TK [1971b]. Summary of 1970 radiation exposures. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. March 9. [SRDB Ref ID: 194596]

Wenstrand TK [1971c]. Health and safety monthly report - November 1971. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. December 8. [SRDB Ref ID: 194597]

Wenstrand TK [1971d]. Health and safety monthly report - May 1971. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. June 7. [SRDB Ref ID: 194598]

Wenstrand TK [1971e]. Reduction of whole body exposures to 5 rems/yr. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. July 1. [SRDB Ref ID: 194600]

Wenstrand TK [1971f]. Health and safety monthly report - April 1971. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. May 7. [SRDB Ref ID: 194601]

Wenstrand TK [1971g]. Contamination around HEV. Memorandum to B.E. Knight. West Valley, New Y: Nuclear Fuel Services. October 12. [SRDB Ref ID: 194602]

Wenstrand TK [1972a]. Plant radiation status March 15, 1972. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. March 17. [SRDB Ref ID: 194603]

Wenstrand TK [1972b]. Health and safety monthly report - January 1972. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. February 8. [SRDB Ref ID: 194604]

Wenstrand TK [1972c]. Radiation exposure for 1971. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. June 1. [SRDB Ref ID: 194605]

Wenstrand TK [1972d]. Status of plant decontamination. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. July 27. [SRDB Ref ID: 194606]

Wenstrand TK [1972e]. Plutonium nitrate contaminated puncture wound. West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 71671]

Wenstrand TK [1972f]. Health and safety monthly report - February 1972. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. March 8. [SRDB Ref ID: 194607]

Wenstrand TK [1972g]. Health & safety monthly report. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. August 7. [SRDB Ref ID: 194608]

Wenstrand TK [1972h]. Health and safety monthly report - December 1971. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. January 10. [SRDB Ref ID: 194609]

Wenstrand TK [1972i]. Health & safety monthly report - June 1972. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. July 10. [SRDB Ref ID: 194610]

Wenstrand TK [1972j]. Health and safety monthly report - April 1972. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. May 11. [SRDB Ref ID: 194611]

Wenstrand TK [1972k]. Health & safety monthly report. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. September 12. [SRDB Ref ID: 194613]

Wenstrand TK [1972l]. Health and safety monthly report - March 1972. Memorandum to J.P. Duckworth. West Valley, NY: Nuclear Fuel Services. April 13. [SRDB Ref ID: 194612]

Wenstrand TK [1973]. Radiation levels at WTF shelter. Memorandum to B.E. Knight, West Valley, NY: Nuclear Fuel Services. April 19. [SRDB Ref ID: 194614]

Wenstrand TK [1974]. LWA and UWA pump changes analysis of exposures. Memorandum to J.R. Clark. West Valley, NY: Nuclear Fuel Services. December 5. [SRDB Ref ID: 194615]

Wilcox DP, Wenstrand TK [1971]. Environmental report 10 January - June 1971. West Valley, NY: Nuclear Fuel Services. August [SRDB Ref ID: 93299]

WNYNSC [no date]. Investigation of high bioassay results - Nuclear Fuel Services, Inc. Western New York Nuclear Service Center, West Valley, NY: Nuclear Fuel Services. [SRDB Ref ID: 24457]

WVDP [2005]. WVDP decommissioning plan Appendix C, summary of facility radiological status by area. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. Rev. 0 Draft 2. [SRDB Ref ID: 30280]

WVNSC [1983]. 1982 environmental monitoring program report for the West Valley Demonstration Project. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. May. [SRDB Ref ID: 23601]

WVNSC [1986]. 1985 environmental monitoring program report for the West Valley Demonstration Project. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. WVDP-040 Rev. 0, March. [SRDB Ref ID: 23604]

WVNSC [1988]. Personnel in vivo counting. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. RC-DOS-6 Rev. 0, November 2. [SRDB Ref ID: 24183]

WVNSC [1990a]. In vitro monitoring. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. RC-DOS-7 Rev. 1, July 31. [SRDB Ref ID: 24193]

WVNSC [1990b]. In vitro monitoring. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. RC-DOS-7 Rev. 2, December 7. [SRDB Ref ID: 24194]

WVNSC [1990c]. Interpretation of in vitro bioassay measurements. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. RC-DOS-30 Rev. 0, October 1. [SRDB Ref ID: 24207]

WVNSC [1993]. Personnel in vivo counting. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. RC-DOS-6 Rev. 2, May 12. [SRDB Ref ID: 24186]

WVNSC [1994]. Sealed rooms paper characterization. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. WVDP-RFI-016 Rev. 0, June 7. [SRDB Ref ID: 30284]

WVNSC [1999]. Area and emergency dosimetry badges. West Valley Demonstration Project, West Valley, NY: West Valley Nuclear Services Company. RC-DOS-34 Rev. 6, October 29. [SRDB Ref ID: 102211]

WVNSC/URS [2006]. West Valley Demonstration Project annual site environmental report CY2005. West Valley, NY: West Valley Nuclear Services Company. URS Group. September. [SRDB Ref ID: 139453]

GLOSSARY

external dose
Dose from radiation (e.g., photons, electrons, and neutrons) that originates outside the body including that from medical screening examinations.

internal dose
Dose from radioactive material in the body (e.g., plutonium or uranium) that was inhaled, ingested, absorbed, or injected through a wound.

limit of detection
Minimum level at which a particular device can detect and quantify exposure or radiation. Also called lower limit of detection and detection limit or level.

minimum detectable amount (MDA)
Smallest amount (activity or mass) of an analyte in a sample that can be detected with a probability ß of nondetection (Type II error) while accepting a probability a of erroneously deciding that a positive (nonzero) quantity of analyte is present in an appropriate blank sample (Type I error).

minimum detectable concentration (MDC)
Minimum detectable activity (or amount) in units of concentration. See *minimum detectable activity*.

minimum detectable level (MDL)
See *minimum detectable amount*.

occupational environmental dose
Dose received from radiation site-related activities (i.e., above-normal background levels) while on a site, which is often recorded by monitoring stations in specific areas or along the boundaries of facilities (e.g., plant stack emissions).

occupational external dose
External ionizing radiation dose from exposure during employment. It does not include dose from background radiation or medical diagnostics, research, or treatment but does include dose from occupationally required radiographic examinations that were part of medical screening.

occupational internal dose
Dose from radioactive material in the body (e.g., plutonium or uranium) that was inhaled, ingested, absorbed, or injected through a wound during employment.

occupational medical dose
Dose from X-ray procedures performed for medical screening of workers as part of an occupational health program and taken on the site. Doses from X-rays used to diagnose diseases or injuries, even if incurred on the job, are not considered occupational and are therefore not eligible to be included in dose reconstruction under the Energy Employees Occupational Illness Compensation Program Act of 2000.

working level (WL)
Unit of concentration in air of the short-lived decay products of ^{222}Rn (^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) and ^{220}Rn (^{216}Po, ^{212}Pb, ^{212}Bi, ^{212}Po) defined as any combination of the short-lived radioactive progeny of radon or thoron in 1 liter of air, without regard to the degree of

equilibrium, that results in the ultimate emission of 1.3×10^5 megaelectron-volts of alpha energy; 1 working level equals 2.083×10^{-5} joule per cubic meter.

working level month

Unit of exposure to radon progeny defined as exposure for 1 working month (170 working hours) to a potential alpha energy concentration of 1 working level; 1 working level month equals 1 working level times 170 hours, which is 0.00354 joule-hour per cubic meter. See *working level*.

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

This attachment describes specific facilities and areas where personnel exposures occurred, with an emphasis on radiological conditions and occurrences.

The ground level of the reprocessing plant was the 100-ft elevation. There was one level below the ground level and four levels above.

A.1 ACID RECOVERY CELL

The ARC was in the southwest corner of the plant on the 114.5-ft level. It housed the fractionator and other equipment used for recovery of process acid. The recovered acid would be mixed with chemicals in the XCR before reuse. Significant dose rates existed in the vicinity of acid recovery equipment from the buildup of ruthenium and carryover from the waste evaporators.

The ARC equipment was known to leak during operation. A spill in the ARC in 1967 required removal of nearly 24 in. of the affected floor. Six additional inches of concrete had to be poured over the original floor thickness to provide shielding. The spill also affected the cells on the ground level below, and reached the soil beneath the process building. It was determined to be the source of a groundwater plume that was discovered years later [WVNSC 1994, 2005].

A.1.1 Radiological Conditions

A.1.1.1 External Exposure Data

In November 1967 piping from the ARC read 3.5 R/hr at 12 in.

On March 5, 1968, a tank in the ARC read 30 R/hr on its sides and 100 R/hr at the bottom [Wenstrand 1968b].

In April 1968 chemical decontamination of the ARC reduced exposure rates from a general level of 2 R/hr to 4 R/hr to 125 mR/hr. The maximum exposure rate was reduced from 50 R/hr to 3 R/hr. Shielding of localized activity reduced the general level further to 120 mR/hr [Keely 1968j].

Decontamination efforts in the ARC in September 1969 reduced radiation levels by a factor of 20 [Wenstrand 1969a].

A.2 ACID RECOVERY PUMP ROOM

The ARPR was located in the southwest corner of the plant on the ground level. It contained equipment that supported operation of the acid recovery system and a sampler that collected a 24-hour composite sample of the fractionator condensate. This condensate was discharged to the plant's liquid effluent interceptor system.

Decontamination work in the ARPR was performed between April 28, 1976, and May 6, 1976. On May 6, 1976, the door between the ARPR and the South Stairs was sealed using a plywood cover with its edges caulked over. A concrete sill had been installed at the door on May 1, 1976 [WVNSC 1994].

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.2.1 Radiological Conditions

The ARPR was affected by the high exposure rates associated with the acid recovery system throughout the plant and was chronically contaminated from maintenance operations. In addition, leaks from ARPR equipment resulted in erosion of the concrete walls and floor.

A.2.1.1 External Exposure Data

In early February 1967 operators taking samples in the ARPR had only 15 minutes before they reached their allowable weekly exposure limit. By February 19, 1967, exposure rates had increased to the point where the operators had only 2 minutes before they reached their weekly limit [Keely 1967c].

“Easily” removable contamination continued to be a problem in the ARPR (and the OGBR) in April 1968, giving skin dose rates of 120 mrad/hr [Keely 1968j].

In February 1972 replacement of the 7G-5 jet in the ARPR required 40 rem of whole body exposure. The initial working area exposure rate was 60 R/hr. Shielding was installed to reduce the exposure rate to where a reasonable working time could be established [Wenstrand 1972f].

An exposure rate of 30 R/hr was encountered in the ARPR in September 1972 [Morrow 1972a].

In October 1972 exposure rates of 20 R/hr were encountered during cleanup work in the ARPR. This is referred to as an area radiation dose rate. It was not a hot spot [Morrow 1972b].

A.2.1.2 Removable Contamination Data

Smears from the ARPR in February 1968 showed typical levels of 1.8×10^6 dpm/100 cm^2 beta [Wenstrand 1968f].

A.3 ANALYTICAL AISLE, LABORATORIES, AND NORTH OPERATING AISLE

The analytical level of the plant (131-ft elevation) included six radiochemical labs:

- Hot Laboratory,
- Quality Control Laboratory,
- Alpha Laboratory,
- Product Laboratory (also sometimes referred to as the Uranium Laboratory),
- Mass Spectrometry Laboratory, and
- Emission Spectrometry Laboratory.

There was also a counting room north of the Mass Spectrometry Laboratory and a chemical storage room. Access to the laboratory areas was via the Analytical Aisle. The Analytical Aisle contained viewing windows and manipulators for operations within the analytical hot cells and Sample Cell 2C.

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

Under the WVDP the laboratories were modified somewhat to support HLW vitrification operations. Some of the laboratories were expanded and renamed. One was called the Radiochemistry Laboratory. Another was the Vitrification Laboratory.

The North Operating Aisle was on the north end of the analytical level east of the CPC Crane Room (CCR). On the east side of the North Operating Aisle was the access to the hoist mechanism for the shield door between the PMC and the PMC Crane Room (PMCR). This area was known as the penthouse.

NFS decontaminated all of the operating period laboratories after shutdown.

A.3.1 Radiological Conditions

A.3.1.1 External Exposure Data

A chronic source of exposure in the Analytical Aisle was activity buildup in the ventilation system air washer. Activity in the washer caused exposure rates up to 40 mR/hr in the Analytical Aisle, which was located directly above the washer cell. NFS (1976) states that this exposure rate could be kept at 20 mR/hr with repeated decontamination of the washer and the addition of lead shielding to the floor of the Analytical Aisle.

In addition to the air washer, the ventilation system ducting was another chronic source of external personnel exposure in the analytical laboratories. Buildup of activity in the duct would produce exposure rates in the analytical laboratories of 10 mR/hr to 20 mR/hr. Lead shielding was added to the floors of the labs to the extent allowed by floor loading constraints [NFS 1976].

Another source of exposure was buildup of contamination from maintenance activities in the penthouse area resulted in exposure rates of 5 mR/hr to 10 mR/hr in the adjacent analytical areas [Wenstrand 1972a].

Routine handling of samples was also an exposure mechanism. Samples having dose rates up to 500 mrem/hr were analyzed in the laboratories in hoods. Shadow shielding was apparently used [NFS 1973b, p. 334].

In early 1968 the Hot Laboratory contained a high radiation area caused by solution leaking from a faulty drain [Runion 1968, p. 35].

In October 1969 the exposure rate in the Analytical Aisle was 50 mR/hr.

In 1990 dose rates in the North Operating Aisle ranged up to 8 mrem/hr. In 2004 the general level was around 1.5 mrem/hr, and in 2005 it was typically less than 1 mrem/hr except for one area by the east doorway that ranged up to 2 mrem/hr [WVDP 2005].

As of 2005 dose rates in the analytical area were low, coming primarily from fume hoods and the ventilation duct. Typical levels were said to be 0.1 mrem/hr to 2 mrem/hr, with a level of 2.9 mrem/hr being reported in the Analytical Aisle in May 2005 [WVDP 2005].

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.3.1.2 Removable Contamination Data

On November 29, 1966, particulate activity from the PMC leaked into the analytical chemistry laboratories resulting in "gross contamination of all laboratories." Cleanup took 4 days.

An inspection by the plant housekeeping committee on January 27, 1967, pointed out a Zone IV leak in the Analytical Aisle that was asserted to have been there for months.

In June 1969 a sample bottle exploded in the Alpha Laboratory causing removable contamination levels of up to 2×10^7 dpm/100 cm^2 alpha. The report states, "No personnel internal exposure to contamination occurred" [Keely 1969d].

A.4 CELL ACCESS AISLE

The Cell Access Aisle (CAA) was on the ground level north of XC2 and Extraction Cell 3 (XC3) (and west of the UPC. It contained shield doors that provided access to XC2 and XC3, the Liquid Waste Cell, and the UPC. The CAA served as an airlock when contact maintenance was being performed in these areas [WVDP 2005].

A.4.1 Radiological Conditions

A.4.1.1 Removable Contamination Data

Smears from the CAA taken January 15, 1968, showed typical removable levels of 5,000 dpm/100 cm^2 alpha [Wenstrand 1968g].

In June 1968 the typical removable contamination in the CAA was said to remain at 800,000 dpm/100 cm^2 alpha [Keely 1968a]. The same level is reported for July 1968 also [Keely 1968k].

In February 1969 smearable contamination in the CAA was reduced from 200,000 dpm/100 cm^2 "to a low Zone 3" [Keely 1969f].

A.4.1.2 Airborne Contamination Data

The CAA became an airborne contamination area during January 1967 [Keely 1967d, p. 2].

A.5 CELL CRANE ROOMS

The remotely operated cranes and power manipulators used in the PMC, CPC, and GPC required periodic maintenance. Access to this equipment was via crane rooms located at one end of each cell. A shield door separated the crane rooms from the adjoining process cells. The equipment was first remotely decontaminated using spray systems before contact maintenance was performed in the crane rooms. Rubber gloves and mats were sometimes used to reduce beta exposures in these areas [NFS 1976].

A.5.1 Impact of the Thorium Extraction Campaign on Airborne Activity Levels in the Cell Crane Rooms

In November 1968 airborne activity in the cell crane rooms was noted to be increasing due to alpha-emitting thorium daughters from processing of thorium-uranium fuel from Indian Point 1. NFS

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

developed methods to correct count data to determine the concentration of the long-lived alpha emitters [Keely 1968f]. By December 1968 airborne activity in the cell crane rooms was said to remain "at maximum levels" (1,000 MPC) during Con-Ed (Indian Point 1) Core A processing "because of short-lived Thoron decay product." Occasional water fogging was used to reduce airborne levels for entries [Keely 1969a]. In February 1969 the airborne activity in the crane rooms "remains at maximum levels" (1,000 MPC) "as thorium contamination in cells still provides a source for short-lived alpha-emitting daughters." The maximum concentration during the month was 50,000 MPC in the CCR [Keely 1969f].

A.5.2 General Purpose Cell Crane Room and Extension

The GCR was directly west of the GPC on the same level (below grade). In 1971 the GCR was lengthened to accommodate storage of the crane and the power manipulator. This area is referred to as the GCR Extension.

The GCR was used for contact maintenance work on the GPC crane, the GPC power manipulator, and the mechanism for the door between the cell and the GCR. This equipment would typically be decontaminated before maintenance activities [WVNSC 1994]. Water would reportedly infiltrate the area on occasion during heavy rains [WVDP 2005].

In 2001 the GCR was used by the WVDP to support cleanup activities in the GPC [WVDP 2005].

A.5.2.1 Radiological Conditions

A.5.2.2 External Exposure Data

During operations, the typical and maximum exposure rates observed in the GCR before decontamination were 8 R/hr and 25 R/hr, respectively [NFS 1976].

In 1999, the general area dose rate in the GCR was approximately 100 mrem/hr, except for the south side, which showed approximately 140 mrem/hr. The highest level measured was 2,200 mrem/hr on the floor near the sump [WVDP 2005].

In 2002 the general dose rate in the GCR extension was in the range of 15 mrem/hr to 30 mrem/hr [WVDP 2005].

A.5.2.3 Removable Contamination Data

A smear survey of the GCR airlock on February 24, 1968, showed removable beta levels ranging from 11,000 dpm/100 cm^2 to 200,000 dpm/100 cm^2 [Wenstrand 1968f, p. 4].

Removable contamination surveys in the GCR in 1999 and 2000 showed levels up to 28,000 dpm/100 cm^2 alpha and 1.25×10^6 dpm/100 cm^2 beta [WVDP 2005].

A.5.2.4 Airborne Contamination Data

In January 1968 airborne activity in the GCR exceeded 1000 MPC on 3 successive days (January 23 to 25). The maximum concentration during this time was 30,000 MPC, or 6×10^{-8} µCi/cm^3 alpha [Wenstrand 1968f, p. 3].

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On February 6, 1968, airborne activity in the GCR was 9.4×10^{-11} µCi/cm^3 alpha and 1.2×10^{-8} µCi/cm^3 beta [Wenstrand 1968f, p. 3].

In June 1968 airborne activity in the GCR remained above 1000 MPC for 3 days, preventing entry [Keely 1968a]. (1000 MPC was NFS' limit for supplied air respiratory protection.)

A.5.3 Process Mechanical Cell Crane Room

Access to the PMCR, which was on the 114.5-ft elevation of the plant, was via a hatch in the ceiling of the Manipulator Repair Room (MRR) below.

A.5.3.1 External Exposure Data

During operations, the typical and maximum exposure rates observed in the PMCR before decontamination were 7 R/hr and 20 R/hr, respectively [NFS 1976].

In 2003 the general area dose rate in the PMCR ranged from 45 mrem/hr to 100 mrem/hr 3 ft above the floor [WVDP 2005].

A.5.3.2 Removable Contamination Data

On February 13, 1968, removable beta contamination in the PMCR airlock ranged from 16,000 dpm/100 cm^2 to 600,000 dpm/100 cm^2 [Wenstrand 1968f, p. 4].

In 2003 removable contamination on the PMCR door hoist mechanism was up to 14,000 dpm/100 cm^2 alpha and 263,000 dpm/100 cm^2 beta. No data were available for the PMCR itself, but the levels were expected to be high [WVDP 2005].

A.5.3.3 Airborne Contamination Data

In February 1968 "typical" airborne activity in the PMCR was 6.6×10^{-7} µCi/cm^3 beta and 8.2×10^{-10} µCi/cm^3 alpha [Wenstrand 1968f, p. 3].

A.5.4 Chemical Process Cell Crane Room

The CCR was at the north end of the CPC oriented 90 degrees to its west wall. It was two levels above the ground level of the CPC on the 131-ft elevation.

A.5.4.1 External Exposure Data

During operations, the typical and maximum exposure rates observed in the CCR before decontamination were 3 R/hr and 6 R/hr, respectively [NFS 1976].

A.5.4.2 Removable Contamination Data

On February 13, 1968, typical removable beta contamination in the CCR airlock ranged from 120,000 dpm/100 cm^2 to 400,000 dpm/100 cm^2 [Wenstrand 1968f, p. 3].

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.5.4.3 Airborne Contamination Data

On February 23, 1968, airborne activity in the CCR was 1.5×10^{-10} µCi/cm^3 alpha and 7.2×10^{-7} µCi/cm^3 beta [Wenstrand 1968f, p. 3].

During February 1969 the concentration of thoron decay progeny in the CCR reached a maximum of 50,000 MPC [Keely 1969f]. This was despite processing of thorium fuel from Indian Point Unit 1 having been completed on January 20, 1969.

A.6 CHEMICAL OPERATING AISLE AND CHEMICAL VIEWING AISLE

The COA was east of the CPC on the 114.5-ft elevation, between the CPC and the PMC. The Chemical Viewing Aisle (CVA) was west of the CPC, between it and the 2nd floor offices in the Office Building.

The COA contained Sampling Station 1-C, which was used to remotely sample the process vessels in the CPC.

A.6.1 Radiological Conditions

A.6.1.1 External Exposure Data

The COA contained a high radiation area caused by backup of solution in a steam line [Runion 1968, p. 31].

A July 27, 1972, summary of plant decontamination activities said radiation levels in the COA had been reduced through various decontamination efforts, both in the COA and in the extraction cells. Remaining sources of radiation were contamination on the floor and residual activity in piping and instrument lines. A steam line was said still to require heavy shielding and should be replaced to reduce radiation levels further. The report also mentions exposure rates in the CVA originating from the Southwest Stairwell have not changed from a previous survey in March 1972 [Wenstrand 1972d].

A.6.1.2 Removable Contamination Data

Contamination would be carried into Sampling Station 1-C from the manipulator used to perform the remote sampling. This contamination was said to be limited to the immediate vicinity of the sampling station [Runion 1968, pp. 30–31].

A.7 CHEMICAL PROCESS CELL

The CPC is where the leaching (dissolving) process was performed on small pieces of fuel after mechanical segmentation in the PMC. The CPC contained two 4,000-L dissolvers. The dissolved fuel was chemically adjusted as necessary in the CPC to create the feed solution for the solvent extraction process carried out in the extraction cells.

A.7.1 Radiological Conditions

The CPC was never a routinely occupied area.

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In May 1968 a remote survey just inside the CPC showed 220 R/hr. [Keely 1968e]. In 1981 the exposure rate in the CPC was said to range from 12 R/hr to 32 R/hr [Golden et al. 1982].

A.8 CONTROL ROOM

The Control Room was located on the fourth floor of the processing plant on the 144-ft elevation. It was used for the monitoring and control of plant processes. Adjacent to the Control Room were the Control Room Office and an area containing water distillation equipment.

A.8.1 Radiological Conditions

There were two chronic sources of external personnel exposure in the Control Room:

- The plant's main ventilation duct, and
- The recovered acid storage tanks in the Hot Acid Cell.

A.8.1.1 External Exposure Data

In January 1967 exposure rates in the Control Room were 10 mR/hr to 15 mR/hr due to activity in the ventilation duct (between the air washer and the stack) [Keely 1967d, p. 3].

On February 26, 1968, the exposure rate in the Control Room ranged from 1 mR/hr to 5 mR/hr [Wenstrand 1968f, p. 3].

Exposure rates in the Control Room increased from 2 mR/hr to 10 mR/hr in June 1969 [Keely 1969d]. The exposure rate dropped back to 2.5 mR/hr in July after flushing of the ventilation duct [Keely 1969e].

The July 27, 1972, summary of plant decontamination activities says reduced activity in the ventilation duct (post shutdown) resulted in "some reduction" of radiation levels in the Control Room, however, a spill in the northwest corner was said to have increased levels in that area [Wenstrand 1972d, p. 4].

NFS [1976] says the exposure rate in the Control Room from radioactivity in the ventilation ducting was typically 5 mR/hr to 10 mR/hr.

In May 2005 dose rates in the Control Room area were generally less than 1 mrem/hr with a high of 1.4 mrem/hr [WVDP 2005].

A.9 EQUIPMENT DECONTAMINATION ROOM AND VIEWING AISLE

The EDR was on the ground level in the northwest corner of the process building, adjacent to the CPC and the Scrap Removal (SR) area. It contained a decontamination pool known as the "soaking pit" and equipment for decontamination of large vessels and equipment.

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.9.1 Radiological Conditions

A.9.1.1 External Exposure Data

The general area exposure rate in the EDR in the first quarter of 1968 was 30 mR/hr to 40 mR/hr. Extremity dose rates were estimated at 590 mrad/hr to 650 mrad/hr [Urbon 1968b].

A.9.1.2 Removable Contamination Data

Removable beta contamination in the EDR ranged up to 2.1×10^6 dpm/100 cm^2 during the week of February 12, 1968 [Wenstrand 1968f, p. 3].

In September 1968 the EDR was reduced to “Zone 3” levels for removable contamination during the month (meaning it was Zone IV previously) [Keely 1968c].

A.9.1.3 Airborne Contamination Data

Airborne activity in the EDR exceeded 1,000 MPC on one sample during March 1968 [Wenstrand 1968b].

A.10 EXTRACTION AND PRODUCT PURIFICATION CELLS

Solvent extraction was carried out in a series of four cells: XC1 to XC3 and the Product Purification Cell (PPC). The four cells were approximately 60 ft high each, so they passed through every elevation of the process building from ground level up to the 160-ft elevation. The cells contained extraction columns and evaporators.

The first two partition cycles, separation of uranium and plutonium from fission products, followed by separation of the plutonium from the uranium, were carried out in XC1. The final partition cycle and first uranium and plutonium extraction cycles were conducted in XC2. The second (final) plutonium and uranium extraction cycles and cleanup were performed in XC3. Final purification of plutonium and uranium nitrate process streams was performed in the PPC. Uranium was processed on the north side of the PPC and plutonium on the south side.

Sometimes the plant records will contain references to the Plutonium Purification Cell or just the Plutonium Cell. These are references to the south (plutonium) side of the PPC.

A.10.1 Radiological Conditions

The extraction cells presented both internal and external radiation exposure hazards from high airborne contamination levels and high dose rates. The PPC was subject to “significant” spills of radioactive liquids including one that contaminated the south wall of the south (plutonium) side to a height of 35 ft [WVDP 2005]. During the operations period, entries into the PPC would cause airflow reversal issues on other levels of the plant, resulting in area contamination.

A.10.1.1 External Exposure Data

Surveys in XC3 on February 29, 1968, showed working exposure rates from 300 mR/hr to 500 mR/hr, with a localized reading of 5 R/hr at 12 in. from a particular component [Wenstrand 1968f, p. 3].

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In October 1969 a survey at the entrance to XC2 showed a whole body exposure rate of 5 R/hr.

A manned entry was made into XC1 on June 29, 1972, after water flush activities. A maximum exposure rate of 50 R/hr was indicated near tank 4D-1 [WVNSC 1994]. Several more entries were made into XC1 in July 1972, identifying exposure rates up to 20 R/hr at Tank 4-D-1 [Wenstrand 1972g]. A survey in XC1 on February 21, 1973, found exposure rates 30 R/hr to 50 R/hr at the second stair landing and 10 R/hr to 50 R/hr at the first landing [WVNSC 1994].

Decontamination efforts in XC2 in May 1972 reduced exposure rates from the floor from 100 R/hr to 3 R/hr [Morrow 1972c].

The contamination on the south wall of the south side of the PPC read up to 53 mrem/hr as of 2003. The dose rate on the north side ranges from 1 mrem/hr to 3 mrem/hr [WVDP 2005].

A.10.1.2 Removable Contamination Data

On February 29, 1968, removable alpha contamination in XC3 ranged from 1.5×10^6 dpm/100 cm^2 to 6.5×10^6 dpm/100 cm^2. Removable beta ranged from 3.5×10^6 dpm/100 cm^2 to 6.4×10^6 dpm/100 cm^2 [Wenstrand 1968f, p. 4].

A survey in 1986 showed a maximum of 400,000 dpm/100 cm^2 removable alpha on the floor of the north side of the PPC before application of strippable coating. Levels on the walls of the north side showed a maximum of 6,700 dpm/100 cm^2 removable alpha. Removable levels in the PPC remained high as of 2003.

A.10.1.3 Airborne Contamination Data

Air samples from the PPC showed concentrations of 2.0×10^{-9} µCi/cm^3 alpha on February 23, 1968, and 6.7×10^{-11} µCi/cm^3 alpha on February 26, 1968 [Wenstrand 1968f, p. 3].

In April 1968 modification work in the PPC (plutonium side) had to be interrupted on several occasions due to excessive airborne activity levels (>1,000 MPC). Smearable contamination reached 5×10^6 dpm/100 cm^2 alpha [Keely 1968j]. On April 19, 1968, the concentration in the PPC reached 5000 MPC [Wenstrand 1968h].

Airborne activity in the PPC reached 30,000 MPC during an entry on October 30, 1968. Personnel had to be evacuated until remote decontamination reduced the level to 1000 MPC [Keely 1968d].

An entry was made into the PPC in May 1969 after an isotopic evaluation showed the plutonium fraction of the airborne contamination was 4% before cell entry, 30% after resin dump, and 7% at completion. Entry had been prevented at first when an assumption of 100% plutonium for the alpha in air meant the concentration exceeded limits for supplied air respiratory protection [Keely 1969d, p. 2].

A November 11, 1971, memorandum alludes to the problems of airborne plutonium levels in the PPC and that workers have to wait for it to settle before making entries. Surface contamination and resuspension are also issues [Heacker 1971].

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A.11 EXTRACTION CHEMICAL ROOM

The XCR was above the extraction cells on the 160-ft elevation of the plant. It is where chemical solutions for the solvent extraction process were prepared.

A.11.1 Radiological Conditions

Some steps of the extraction chemical makeup process made use of acid from the acid recovery system. High exposure rates were therefore an issue. In addition, the XCR was chronically contaminated from leaks, spills, and equipment maintenance or replacement activities.

A.11.1.1 External Exposure Data

The typical exposure rate in the XCR during the month of February 1968 ranged from 2.5 mR/hr to 500 mR/hr [Wenstrand 1968f, p. 3].

Dose rates of 50 R/hr were measured in the vicinity of the XCR in December 1971. The source was recovered acid in a tank [Wenstrand 1972h].

The July 27, 1972, summary of plant decontamination activities says flushing of tanks and removal of contaminated concrete in the XCR had reduced radiation levels by a factor of 10 to 100. The principal remaining radiation source was a tank reading 20 mR/hr to 30 mR/hr at contact. This tank read 1 R/hr to 2 R/hr before decontamination [Wenstrand 1972d, p. 4].

A.11.1.2 Removable Contamination Data

Removable beta under the tanks in the XCR ranged from 21,000 dpm/100 cm^2 to 240,000 dpm/100 cm^2 on January 22, 1968 [Wenstrand 1968g, p. 4].

Smearable contamination in the XCR area was well into Zone 4 levels in December 1971 [Wenstrand 1972h].

A.12 EXTRACTION SAMPLE AISLE AND PROCESS SAMPLE CELL 2

The XSA was located on the 131-ft elevation of the plant, which was known as the analytical level. The XSA was adjacent to the north walls of the four extraction cells, between the cells and the laboratories. It included PSC-2, which was equipped with gloveboxes and sampling stations for sampling the adjacent extraction cells. The gloveboxes were removed in 1986 [WVDP 2005].

A.12.1 Radiological Conditions

The XSA contained two sample stations that suffered from contamination problems purported to be from insufficient bagging and airflow reversals caused by entries into the PPC [Runion 1968, p. 28]. The contamination problems from entries into the PPC were such a problem that, as of early 1968, NFS required the XSA to be cleared of personnel and the door locked before a PPC entry was made. Subsequently, the XSA had to be surveyed once work in the PPC was completed [Runion 1968, p. 29]. At the time they were enacted (early 1968), these measures were intended to be temporary until the sampling stations and gloveboxes could be improved to mitigate the contamination problems.

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A.12.1.1 Removable Contamination Data

General removable alpha levels in the XSA ranged from 5,000 dpm/100 cm^2 to 10,000 dpm/100 cm^2 on November 20, 1967.

A.12.1.2 Airborne Contamination Data

In October 1967 air samples "taken on each shift" from the XSA and PSC-1 showed long-lived airborne alpha concentrations of 2×10^{-12} µCi/cm^3 and 5.4×10^{-11} µCi/cm^3 [Wenstrand 1967d].

Airborne activity levels rose to MPC in the XSA during a shutdown of the ventilation system after the filter failure incident on September 4, 1968 [Keely 1968c].

A.13 FUEL RECEIVING AND STORAGE

The FRS facility was used to handle and decontaminate shipping casks for nuclear fuel and to move fuel using service crane bridges. It consisted of the fuel SP, the cask receiving area, the cask decontamination area (pit), and the pool water treatment area. Fuel was transferred from the FRS to the PMC via a transfer tunnel in the southwest corner of the SP.

Other facilities associated with the FRS were a radioactive waste processing building (known as the Hittman Building) and the resin pit.

The radioactive waste processing building was located in a yard area north of the FRS. It was used for dewatering and storage of filter media from FRS pool cleanup systems and ion exchange resins. Spent resin was dewatered and stored in High Integrity Containers located 50 ft north of the FRS Building [WVDP 2005].

The resin pit was an in-ground concrete vault located outside the southeast corner of the FRS Building. Spent filter media and ion exchange resin from the pool water treatment area were flushed to a drum in the pit. A drain line went from the pit to the Interceptor system.

Fuel was stored in the FRS pool until 2001. In 2002 and 2003 the pool was drained, empty canisters and storage racks were removed, and the SP and Cask Unloading Pool (CUP) were decontaminated and sealed [WVDP 2005].

A.13.1 Radiological Conditions

Receipt and storage of ruptured fuel assemblies resulted in significant buildup of fission products in the SP. The radioactivity concentration in the pool eventually reached 0.1 µCi/cm^3, prompting the installation of auxiliary cleanup systems to augment the primary system. However, even with the auxiliary system the contamination level of the pool water could only be reduced to 0.01 µCi/cm^3. Hence, significant levels of contamination would be found "Whenever items were removed from the pool." Radiation levels would reach 20 mR/hr to 100 mR/hr during routine operations in some areas of the FRS [NFS 1976].

Personnel exposures in the FRS area would also result from preparation of casks for loading into the pool. Workers were exposed to the surface dose rates from the casks while they were prepared for immersion [NFS 1976]. Incoming casks could have a contact dose rate up to 200 mR/hr (limit per

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shipping regulations). Personnel were routinely in close proximity to the casks while preparing them for immersion into the pool and unloading of the fuel [Runion 1968, p. 15]. FRS workers removing and cleaning fuel shipping cask internals worked in exposure rates up to 300 mR/hr [Loud 1965b, p. 3]. The average collective dose for FRS workers over 1970 and 1971 was 250 person-rem. A May 12, 1975, Nuclear Insurance Survey report by Johnson & Higgins states the highest level of personnel exposure at West Valley was around 400 mrem/quarter received by the fuel unloading operators [Johnson & Higgins 1975].

Liquids would be spilled into work areas in the FRS, resulting in airborne contamination and personnel intakes. There was also potential for exposure to fission gases released when cask lids were removed (in the pool) [Bailey 1990]. Other routes of personnel exposure in the FRS were handling of contaminated tools, drumming of filter and demineralizer media from the pool water treatment system, and shipping cask decontamination operations. There were multiple instances of personnel exposures associated with flushing of shipping casks.

Underwater tools and the crane cables and blocks allowed highly contaminated pool water to drip onto the service bridges. Casks were contaminated to “several hundred thousand” dpm/100 cm^2 beta from the pool water upon being removed from the pool. Unloaded casks were moved from the SP to the decontamination pit, where they were cleaned using spray and scrub techniques to return them to acceptable contamination levels for shipping. The FRS service crane bridges were repeatedly contaminated from FRS operations. The bridges were isolated and controlled via rope and step-off pads [Runion 1968, pp. 15–16].

A.13.1.1 Pool Water Concentration Data

Table A-1 shows gross beta and gross alpha concentration data for the FRS CUP and SP for December 1965.

Table A-1. Gross beta and gross alpha concentration data for the CUP and SP, December 1965 (µCi/cm^3).

Pool	Gross beta concentration	Gross alpha concentration
CUP	3.1E−5	2.5E−9
SP	2.8E−5	4.8E−9

Table A-2 shows concentration data for the CUP and SP for January to June 1966. Data are given for soluble gross beta, gross alpha and tritium; and insoluble gross beta and gross alpha.

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Table A-2. Radioactivity concentration data for the CUP and SP, January to June 1966 (µCi/cm^3).

Date	Pool	Gross beta, soluble	Gross alpha, soluble	Tritium, soluble	Gross beta, insoluble	Gross alpha, insoluble
January 1966	CUP	8.0E−5	1.6E−7	6.5E−7	8.8E−7	1.0E−8
January 1966	SP	8.8E−5	3.0E−8	5.0E−6	2.0E−6	1.0E−8
February 1966	CUP	2.7E−5	<1E−9	6.0E−6	1.9E−6	2.6E−9
February 1966	SP	2.8E−5	3.7E−9	9.5E−6	1.4E−6	9.4E−9
March 1966	CUP	5.9E−5	3.3E−7	3.9E−6	5.5E−6	<1E−8
March 1966	SP	6.8E−5	<1E−8	4.3E−6	2.0E−6	<1E−8
April 1966	CUP	4E−5	1E−8	1E−5	3E−6	7E−10
April 1966	SP	5E−5	<1E−8	9E−6	3E−6	4E−9
May 1966	CUP	2E−4	6E−8	1E−4	6E−6	9E−9
May 1966	SP	2E−4	5E−8	1E−4	5E−6	1E−9
June 1966	CUP	1E−4	<1E−8	4E−4	4E−5	9E−9
June 1966	SP	2E−4	<1E−8	3E−3	1E−5	6E−10

For the February 1966 samples, “Gamma analysis showed that Co-60, Co-58, and Cs-137 are the principal contaminants” [Loud 1966b, p. 3].

Table A-3 shows gross beta concentration data for the CUP and SP over the period September 1965 to October 1969. Values reported for a month (rather than a specific date) are typical or representative values. Values are for the SP unless otherwise specified.

After reprocessing operations ceased in 1972, the FRS pool was drained and cleaned. After refilling, the contamination level of the pool water was reduced to 5×10^{-3} µCi/cm^3 [NFS 1976].

Table A-3. Gross beta concentration data for the CUP and SP, September 1965 to October 1969 (µCi/cm^3).

Date	Gross beta concentration	Remarks
September 1965	1.1E−04	CUP
September 1965	7.8E−05	SP
September 16, 1965	1.6E−04	CUP concentration due to ruptured fuel assembly; previously 5.4E−6 µCi/cm^3
October 1965	3.1E−04	CUP
October 1965	2.8E−04	SP
November 1965	1.4E−04	"No detectable alpha"
August 1966	2.0E−03	Small cracks noted in pool, causing contamination on nearby floor areas
July 1, 1967	4.0E−03	Maximum; low was 9E−5 µCi/cm^3
September 6, 1967	7.5E−03	None
October 8, 1967	1.8E−02	None
March 1969	1.0E−01	Waiting for removal of ruptured N Reactor fuel before attempting cleanup
April 1969	1.0E−01	None
May 1969	1.2E−01	None
June 1969	1.2E−01	None
July 1969	8.0E−02	None
August 1969	1.7E−02	None
September 1969	8.0E−03	None
October 1969	4.0E−03	None

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On May 11, 1972, isotopic measurements were made on FRS pool water to examine the effectiveness of cleanup systems. Concentration data were given for ^{134}Cs, ^{137}Cs, ^{106}Ru/Rh, ^{144}Ce/Pr, and ^{125}Sb [Jaroszeski 1972].

A.13.1.2 External Exposure Data

Table A-4 shows exposure rates over the top of the FRS pools for November 1965 to May 1969.

Table A-4. Exposure rates over the top of the FRS pools (mR/hr), November 1965 to May 1969.

Date	Exposure rate (mR/hr)	Remarks
11/1965	0.2	None
01/1966	0.1–0.15	None
02/1966	0.1–0.15	7 fuel casks received during the month
03/1966	0.1–0.15	10 fuel casks received during the month
04/1966	0.2	15 fuel casks received during the month
05/1966	0.2	None
06/1966	0.4	None
05/1969	30 whole body 100 skin	Average working level on bridges above pool

In December 1965 the FRS pool demineralizer was placed in service “for a few days” and showed an exposure rate “at the side” of 1.5 R/hr [Loud 1966a].

In September 1969 an expansion box on a fuel cask received read 5 R/hr on contact. Investigation found it contained high-activity primary coolant. Attempts to clean up the coolant using an ion exchange unit were hampered when the unit read 350 R/hr after 3 hours of operation [Wenstrand 1969a].

On December 5, 1969, a line venting the cask leaked coolant onto the concrete outside the CUP, which resulted in a high radiation area that persisted for days. The contaminated concrete had to be removed and lead fill added [Duckworth 1970].

In November 1971 shrouds were removed from the FRS pool resulting in exposure rates up to 40 R/hr. Shielding was provided for the FRS crane operator, reducing his work area exposure rate to 300 mR/hr [Wenstrand 1971c, p. 2].

In June 1972 decontamination efforts on the CUP reduced exposure rates to approximately 5 R/hr [Wenstrand 1972i].

During October 1972 exposure rates of 20 R/hr were encountered while handling spent resin in the FRS. This is referred to as an “area radiation dose rate”; that is, it was not a hot spot [Morrow 1972b].

During November 1972 exposure rates of 30 R/hr were encountered during removal of wastes from the FRS [Morrow 1972d].

Representative exposure rates for the FRS in 2002 and 2003 were [WVDP 2005].

- SP walls and floor = 6.7 mrem/hr to 8.5 mrem/hr;

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- CUP walls and floor = 25.7 mrem/hr to 500 mrem/hr;
- Water treatment area = 5.4 mrem/hr on face of shield wall; 195 mrem/hr on middle of demineralizer column;
- Cask Decontamination Area = 0.17 mrem/hr in stall; 24 mrem/hr in pit sump;
- Hittman Building = 0.5 mrem/hr to 0.8 mrem/hr;
- Top of resin pit = 18 mrem/hr; and
- Bottom of FRS pool transfer tunnel = 2,000 mrem/hr.

A.13.1.3 Removable Contamination Data

Smears around the FRS Decontamination Pit showed removable levels from 3,000 dpm/100 cm^2 to 8×10^6 dpm/100 cm^2 beta on 1/24/1968 [Wenstrand 1968g, p. 4].

Smearable contamination on an incoming cask was 1×10^6 dpm/100 cm^2 beta in July 1968 [Keely 1968k].

In September 1969 smearable contamination in excess of 50,000 dpm/100 cm^2 beta was found in the FRS rail track area [Wenstrand 1969a].

Smearable contamination on an incoming fuel cask showed a maximum of 7.5×10^7 dpm/100 cm^2 April 1971. This is assumed to be removable beta contamination, though the reference does not explicitly say so [Wenstrand 1971f, p. 2].

NFS received fuel casks with removable contamination levels in excess of the 10 C.F.R. Part 20 limit of 22,000 dpm/100 cm^2 [10 C.F.R. 20, 1991] five times between July 2 and November 7, 1974 [O'Reilly 1975, p. 7].

After the decontamination activities in 2002 to 2003, removable contamination levels in the FRS area were less than 2,000 dpm/100 cm^2 alpha and 20,000 dpm/100 cm^2 beta [WVDP 2005].

A.13.1.4 Airborne Contamination Data

There was high airborne beta activity in the FRS on December 20, 1966, attributed to removable activity in a drain [Keely 1967e].

April 1969: The "maximum weekly" long-lived air activity above the FRS pool for April 1969 was 4×10^{-10} µCi/cm^3 beta [Keely 1969g].

January 1972: airborne concentrations in the FRS have decreased by a factor of 10 due to the cleanup effort [Wenstrand 1972b].

A lid blew off a 55-gal drum during decanting of the FRS settling tank on May 15, 1974. Resulting airborne contamination levels were 1.56×10^{-11} µCi/cm^3 alpha and 2.66×10^{-9} µCi/cm^3 beta [G. 1974].

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A.14 GUARD HOUSE

The Guard House was the main point of entry to and exit from the plant, and where employee film badges were stored. It was also referred to as the Gate House.

A.14.1 Radiological Conditions

In August 1970 the AEC expressed concerns over exposure rates at the gatehouse. NFS said that while the exposure rates in that area were “higher than desired”, they did not consider it to be an unrestricted area vis-à-vis 10 C.F.R. Part 20 [10 C.F.R. 20, 1991]. The only persons routinely in that area were the guards, who were described as being “low in radiation exposure.” NFS said radiation levels in that area would be “reduced to more desirable levels as soon as possible” [Duckworth 1970]. The discussion pertains to the fact exposure rates at the Guard House exceeded 10 C.F.R. Part 20 limits for an unrestricted area.

A.15 GENERAL OPERATING AISLE

The GOA was below grade on the north side of the GPC. It contains four shielded viewing windows that were used by workers operating manipulators and other equipment in the GPC.

A.15.1 Radiological Conditions

A.15.1.1 Exposure Rate Data

A July 27, 1972, summary of plant decontamination activities says radiation levels in the GOA were reduced by a factor of two due to lower radiation levels in the HEV duct. It also says contamination around the west window was still causing a general area exposure rate of 20 mR/hr [Wenstrand 1972d].

The reported dose rate in the GOA as of 2005 was less than 1 mrem/hr [WVDP 2005].

A.15.1.2 Airborne Contamination Data

The GOA was one of the areas that showed airborne contamination levels in excess of 40 MPC-hr during flushing of the plant ventilation system.

A.16 GENERAL PURPOSE CELL

The GPC was 25 ft below the ground level of the plant, and ran perpendicular between the PMC and the CPC. The GPC served to transfer baskets containing 200 kg (design basis) of segmented fuel between the PMC and the CPC for dissolving. Afterward, they would be moved back into the GPC so the remaining cladding and fuel hardware could be packaged for transfer into the SR Area for disposal. The GPC would also be used for packaging of other process wastes and damaged equipment for transfer into the SR Area.

The GPC had three shielded viewing windows in its north wall, denoted east to west as GPC-A, -B, and -C [Vance 1986]. The west window was a source of contamination into the GOA, which ran adjacent to the north wall of the GPC.

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There was another cell on the other side of east wall of the GPC called the Miniature Cell. This cell was never used.

Vance [1986] says discussions with former employees indicated there had been a metal fire in the GPC during the operations period when a batch of leached cladding spontaneously ignited when it was dumped. It is not known specifically when this occurred. WVDP [2005] says three cladding fires occurred during operations.

A.16.1 Radiological Conditions

The GPC was never a routinely occupied area.

A.16.1.1 External Exposure Data

In 1986 dose rates 6 ft above the floor ranged from 40 rem/hr to 340 rem/hr. At 2 ft above the floor the dose rates ranged up to 650 rem/hr.

A characterization survey performed in November 2004 showed the following dose rates:

- 3 ft above the floor = 4 rem/hr to 200 rem/hr;
- 6 ft above the floor = 3 rem/hr to 45 rem/hr; and
- 9 ft above the floor = 2 rem/hr to 32 rem/hr.

A.16.1.2 Removable Contamination Data

Smear samples collected in 1986 showed removable ^{137}Cs up to 388 µCi/smear.

A.17 HEAD END VENTILATION BUILDING AND CRANE ROOM

The HEV Building was located on the east side of the MSM Shop. It was built in 1970 to accommodate the HEV system. The HEV Building had two levels. The lower level housed filters, blowers, ductwork, and other equipment. The upper level contained a crane and filter change equipment. This area was often referred to as the HEV Crane Room. Like the main ventilation system, the HEV system could be operated on steam or electric power.

A.17.1 Radiological Conditions

A.17.1.1 External Exposure Data

On October 12, 1971, extensive contamination was found in a surface water drainage ditch from the HEV area. Typical exposure rates ranged from 60 mR/hr to 100 mR/hr, and up to 150 mR/hr inside drains [Wenstrand 1971g].

In July 1972 decontamination of the floor and dock area of the HEV Crane Room reduced radiation levels in these areas by a factor of 10 to 12. Radiation fields from “shielding flaws at the sides of the HEV” still required shielding, though levels were lower due to less activity in the filters and decontamination of the filter chambers [Wenstrand 1972d, p. 3].

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In 2002 dose rates from the HEV prefilters ranged up to 50 rem/hr. The roughing filters ranged up to 18 rem/hr and the first stage high-efficiency particulate air (HEPA) filter bank up to 2.3 rem/hr. Ductwork in the adjacent operating aisle ranged from 3 mrem/hr to 50 mrem/hr. Dose rates up to 61 mrem/hr were measured in the lower level of the HEV Building in May of 2005 [WVDP 2005].

A.17.1.2 Removable Contamination Data

No specific information has been identified. The 2005 draft of the WVDP Decommissioning Plan says removable levels in the HEV Building "are expected to be significant" [WVDP 2005].

A.18 HOT ACID CELL

The Hot Acid Cell was located on top of the CPC at an elevation of 148 ft. It contained two storage tanks (7D-11 and 7D-12) for recovered acid. It also contained a pump in a niche.

A.18.1 Radiological Conditions

A.18.1.1 External Exposure Data

On February 14, 1994, exposure rates of 5 mR/hr to 10 mR/hr were present near the floor and 10 mR/hr to 15 mR/hr at the base of the acid tanks [WVNSC 1994].

In 2001 dose rates inside the Hot Acid Cell ranged from 0.3 mrem/hr to 80 mrem/hr. Dose rates from piping, valves, and the floor drain ranged from 5 mrem/hr to 60 mrem/hr [WVDP 2005].

A.18.1.2 Removable Survey Data

A 1994 removable contamination survey showed levels up to 450 dpm/100 cm^2 alpha and 42,000 dpm/100 cm^2 beta [WVDP 2005].

A.19 HOT SHOP

The Hot Shop was a portion of the "old maintenance shop adjacent to the new laundry" [Runion 1968, p. 21]. The area was used for decontamination, repair, and maintenance of equipment such as MSMs and power manipulator parts. The Hot Shop was provided in 1967 to permit repair of contaminated equipment. A new Decontamination and MSM Repair Facility was under construction as of early 1968 with completion expected July 1 [Clark 1968b].

A.19.1 Radiological Conditions

Equipment was bagged before transfer to the Hot Shop, but maintenance activities there resulted in contamination of it and surrounding areas. Hot Shop personnel wore respiratory protection.

A.19.1.1 External Exposure Data

On November 3, 1967, the Hot Shop showed a general background of 10 mR/hr to 20 mR/hr. Some manipulator parts read up to 21 rad/hr.

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.19.1.2 Removable Contamination Data

Smears of the Hot Shop on January 22, 1968, showed removable beta levels of 240,000 dpm/100 cm^2 to 4 × 10^6 dpm/100 cm^2 and alpha levels of 10,000 dpm/100 cm^2 to 12,000 dpm/100 cm^2 [Wenstrand 1968g, p. 4].

Typical removable beta contamination in the Hot Shop was 100,000 dpm/100 cm^2 on February 27, 1968 [Wenstrand 1968f, p. 3].

Removable contamination increased above allowable limits in the Hot Shop on 3/3/1968 [Wenstrand 1968b, p. 3].

A.20 INSTRUMENT SHOP/OLD INSTRUMENT SHOP

The Instrument Shop was located in the Ventilation Supply Room, which was east of the PMC on the 114.5-ft elevation. It was used for the repair of uncontaminated instruments until this function was relocated sometime in 1967. After this time the area was referred to as the Old Instrument Shop. The area itself was physically removed in 1985.

A.20.1 Radiological Conditions

A.20.1.1 External Exposure Data

In January 1967 exposure rates in the Instrument Shop were 10 mR/hr to 40 mR/hr from activity in the ventilation duct (between the air washer and the stack) which passed adjacent to the area [Keely 1967d, p. 3].

A survey in the Old Instrument Shop on February 26, 1968, showed exposure rates from 250 mR/hr to 400 mR/hr [Wenstrand 1968f, p. 2].

As of 2005 dose rates adjacent to the ventilation duct in the Ventilation Supply Room ranged up to 7 mrem/hr [WVDP 2005].

A.20.1.2 Removable Contamination Levels

Removable contamination in the Old Instrument Shop was reduced from “high Zone 4 levels to low Zone 3 levels” during the month of April 1969 [Keely 1969g]. However, a leaking duct recontaminated the area back to Zone 4 levels the following month [Keely 1969c, p. 2].

A.21 INTERCEPTOR AND NEW INTERCEPTOR

The original Interceptor was located within the plant security fence just east of the process building. Its purpose was to collect liquid wastes from floor drains, sink drains, blowdown from process water and steam, and cask wash water from the FRS. These liquids were collected in batches and analyzed before transfer to the lagoon system. Wastes were then stored in Lagoons 1 and 2 until they were processed in the LLWT Facility. Before May 1971, the wastewater went from Lagoon 2 to Lagoon 3 before being discharged to surface streams.

Potentially radioactive condensates in the processing plant were collected and sampled in batches before disposal. If the gross beta concentration exceeded 3 × 10^{-7} µCi/cm^3, the waste was transferred

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

to the Interceptor system. Otherwise, it was discharged directly to surface streams. Secondary cooling water used in the High Level Waste Storage Facility and in the FRS cooler was also discharged directly to surface streams.

A new Interceptor was installed by the WVDP in the late 1990s consisting of twin, open-top concrete storage tanks. The new Interceptor is located just east of the old one. The tanks are used to store radioactive liquids before treatment in the LLWT Facility [WVDP 2005].

In 2005 the old Interceptor was being used for storage of radioactive liquids that exceeded the effluent limit of 5×10^{-3} µCi/cm^3. Liquids can be transferred from the old Interceptor to the new Interceptor via steam jet [WVDP 2005].

A.21.1 Radiological Conditions

A.21.1.1 External Exposure Data

Exposure rate surveys performed in August 2003 on the old Interceptor showed a range of 0.4 mR/hr at the top of the tank to 408 mR/hr near the bottom at a distance of 1 ft from the tank wall [WVDP 2005].

A valve in the new Interceptor valve pit showed 0.3 mR/hr in October 2003 [WVDP 2005].

A.22 LAUNDRY

A new laundry facility went into operation in 1967 in an effort to lower workplace contamination experienced by laundry workers.

A.22.1 Radiological Conditions

A.22.1.1 External Exposure Data

As of early 1968 clothing articles with dose rates up to 50 mR/hr could be sent to the plant laundry.

An AEC inspection report from 1972 summarizes external dose received by a [redacted] worker at West Valley between [redacted]. His total whole body dose over this period was [redacted], and he had no quarterly dose in excess of [redacted]. The report states the worker's "extremity and skin exposure" over the [redacted] was [redacted]. It is not known why the two were summed, if that was the case [Nelson 1973a, p. 17]. The whole body dose averages to [redacted].

A.22.1.2 Removable Contamination Data

Waste from the laundry facility backed up in June 1967, contaminating the First Aid Room to Zone IV levels [Keely 1967f].

A.22.1.3 Airborne Contamination Data

On February 9, 1967, the airborne alpha activity in the laundry area increased "to levels requiring personnel to wear masks" [Keely 1967a, p. 2].

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

As of early 1968 laundry personnel wore canister masks and worked under a hood flowing at "about 1,300 scfm" while sorting contaminated clothing before washing [Runion 1968, p. 20]. The laundry room also contained a continuous air monitor [Clark 1968b].

A.23 LOAD-IN FACILITY

The Load-in Facility (LIF) was built by the WVDP adjacent to the west wall of the EDR. It was built to make use of the EDR as the primary access for moving canisters and replacement equipment into the Vitrification Cell or the CPC [WVDP 2005].

The dose rate in the LIF is less than 0.1 mrem/hr [WVDP 2005].

A.24 LOW LEVEL WASTE TREATMENT FACILITY

The LLWT facility began operating on May 24, 1971. It utilized ion exchange methods to remove cesium and strontium from liquid wastes collected by the Interceptor system. The effluent from the LLWT facility was stored in Lagoons 4 or 5. It would then be analyzed and transferred to Lagoon 3 for discharge to surface streams. If the analyses of wastes in Lagoons 4 or 5 exceeded concentration limits, the waste was recycled back to Lagoon 2. NFS [1976] says the LLWT facility removed "greater than 95% of the cesium and strontium" from liquid wastes.

A.24.1 Radiological Conditions

A.24.1.1 External Exposure Data

The area around the LLWT facility's ion exchange beds read 30 mR/hr at contact and 5 mR/hr in the aisle shortly after it began operating in May 1971. Exposure rates from the first drums of sludge were 50 mR/hr contact 5 mR/hr at 3 ft [Wenstrand 1971d].

A.25 LOW LEVEL WASTEWATER TREATMENT BUILDING

The Low Level Wastewater Treatment Building was built by the WVDP and put into service in 1998. Its purpose is to remove contaminants (^{90}Sr and ^{137}Cs primarily) from radioactive wastewater via ion exchange and filtration. It treats water from Lagoon 2 and groundwater from wells on the North Plateau [WVDP 2005].

Spent resin from the ion exchange modules is dewatered in a packaging area in the northeast end of the building. The liquid from the dewatering operation goes to a sump to be processed as wastewater. The spent resin is packaged and transferred out for storage and disposal [WVDP 2005].

A.25.1 Radiological Conditions

A.25.1.1 External Exposure Data

Survey data show contact readings on one of the ion exchange modules up to 11 mrem/hr. A 2003 survey of the sump exit piping showed 4 mrem/hr. General area dose rates were less than 1 mrem/hr in 2005 [WVDP 2005].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.25.1.2 Removable Contamination Data

Most removable contamination surveys do not show any detectable activity. A smear of the sump grating cover showed 1,815 dpm/100 cm^2 removable beta in 2003 [WVDP 2005].

A.26 LOWER EXTRACTION AISLE

The LXA was immediately north of the extraction cells on the 114.5-ft elevation of the plant. It contained two PSCs with gloveboxes. The LXA was an operations and maintenance area that provided access to piping and instrumentation penetrations into the extraction cells and the UPC.

A.26.1 Radiological Conditions

As with numerous areas in the process building, the ventilation duct passing through the LXA was a chronic source of external personnel exposure.

A.26.1.1 External Exposure Data

The July 27, 1972, summary of plant decontamination activities says the principal radiation sources in the LXA were the ventilation duct and ventilation washer [Wenstrand 1972d, p. 4].

In early 1968 the LXA had an area of elevated exposure rates (in excess of 100 mR/hr) due to backup in a steam supply line that supplied a jet in one of the process cells. Shielding was supposed to have been installed on the line by March 1, 1968 [Runion 1968, p. 25].

The exposure rate under the ventilation duct in the LXA was 200 mR/hr before it being flushed on October 3, 1969.

As of 2004 the general area dose rate in the LXA was typically 0.1 mrem/hr with higher levels present near ventilation ducts [WVDP 2005].

A.27 LOWER WARM AISLE

The LWA was immediately adjacent to the extraction cells on the ground level. It contained shielded niches (concrete enclosures) for equipment such as air-operated pumps, valves, etc. that serviced the extraction cells. The LWA would become contaminated during replacement or repair of this equipment and from movement of contaminated filters from the OGBR through the LWA to the waste truck for disposal [Runion 1968, pp. 23–24].

General decontamination was performed on the LWA between 1972 and 1974. Additional cleanup and modifications (to support the WVDP) occurred in the early 1980s [WVDP 2005].

A.27.1 Radiological Conditions

A.27.1.1 External Exposure Data

On November 7, 1967, a “leaking drain pipe” on the west wall of the LWA read 5 R/hr at 6 in. [Wenstrand 1967a].

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

DOG filters being stored in the LWA on March 29, 1968, were causing exposure rates from 1 R/hr to 50 R/hr in the aisle [Wenstrand 1968b].

In 1987 the dose rate in LWA Niche 1 ranged up to 2,700 mrem/hr. Rates for Niche 2 were up to 758 mrem/hr and up to 96 mrem/hr in Niche 3 [WVDP 2005].

Typical dose rates in the LWA (not in the niches) as of 2005 are 0.1 mrem/hr to 0.3 mrem/hr [WVDP 2005].

A.27.1.2 Removable Contamination Data

Removable beta contamination in the LWA was 100,000 dpm/100 cm^2 to 600,000 dpm/100 cm^2 on November 9, 1967. Removable alpha was up to 18,000 dpm/100 cm^2.

On November 23, 1967, the LWA showed maximum levels of removable contamination on the floor of 750,000 dpm/100 cm^2 beta and 140,000 dpm/100 cm^2 alpha. Horizontal surfaces showed removable up to 1.8×10^6 dpm/100 cm^2 beta and 4.5×10^6 dpm/100 cm^2 alpha.

Removable contamination in the Product Packing and Shipping and LWA airlock ranged from 18,000 dpm/100 cm^2 to 6×10^6 dpm/100 cm^2 beta on January 16, 1968. On January 20, 1968, the LWA showed removable levels in equipment niches of 500,000 dpm/100 cm^2 to 9×10^6 dpm/100 cm^2 beta and 1×10^6 dpm/100 cm^2 to 9×10^6 dpm/100 cm^2 alpha [Wenstrand 1968g, p. 3].

In May 1968 “aggressive” decontamination efforts reduced removable beta contamination levels to Zone 3 limits in both the UWA and LWA. Removable alpha contamination in the LWA was still at Zone 4 levels [Keely 1968e].

A spill on June 30, 1969, resulted in removable contamination levels in the LWA of 1×10^6 dpm/100 cm^2 [Keely 1969d].

A.27.1.3 Airborne Contamination Data

In April 1968 the Plutonium Cell and the LWA had to be “fogged” with water to reduce airborne radioactivity levels to less than 1000 MPC so entry could be made using supplied air (Keely 1968j). The excessive airborne concentration in the LWA resulted from a filter change on April 4, 1968 [Wendstrand 1968h, p. 3].

A.28 MAIN OFFICE BUILDING

The main Office Building served as the main entrance to the processing plant. It had three floors and was located west of the CPC. Between the Office Building and the CPC was the Chemical Viewing Aisle.

The first floor of the Office Building included the Hot Lobby and locker room areas. The Hot Lobby was a dressout area where workers donned basic protective clothing. It served as a buffer area between the Office Building and the processing plant. The office building had another lobby area that was differentiated from the Hot Lobby, i.e., they were separate areas.

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.28.1 Radiological Conditions

The ventilation and acid recovery systems were chronic sources of external exposure in the office building. Contamination in the Southwest Stairwell was also a contributor.

A.28.1.1 External Exposure Data

Radiation levels in the third floor offices increased to 5 mR/hr in September 1966. Personnel were moved to the Annex Building to decrease their exposure [Keely 1966].

“Background in the Control Room and Office areas sometimes rose to 10 mR/hr” from activity in the recovered acid storage tanks in the Hot Acid Cell [NFS 1976].

The July 27, 1972, summary of plant decontamination activities says radioactive piping was removed from the northwest corner of the OGA, resulting in “some reduction” in radiation levels in the third-floor office [Wenstrand 1972d, p. 4]. The report also says radiation levels in the Office Building originating from the Southwest Stairwell have not changed from a previous survey in March 1972 [Wenstrand 1972d, p. 3].

Most areas in the Office Building showed dose rates less than 32 µrem/hr in April and May 2005. A maximum of 1.4 mrem/hr was measured on the first floor [WVDP 2005].

A.28.1.2 Removable Contamination Data

The OGA and Southwest Stairwell were contaminated to Zone 4 levels by an acid leak in the OGA on October 23, 1969. Contamination was spread to the Office Building and Chemical Viewing Aisle [Wenstrand 1969b]. The spill also contaminated an area beneath the floor of the men’s shower room on the ground level [WVDP 2005].

A.29 MAINTENANCE SHOP AND WAREHOUSE

These facilities were separate from the process building. Maintenance or repair operations on contaminated components were sometimes carried out in the Maintenance Shop. However, this was not routinely a radiological area. No indication of radioactive material being present in the warehouse has been found in references reviewed to date.

A.29.1 Radiological Conditions

A temporary contamination zone was established in the Maintenance Shop in March 1968 for the repair of a PAR Manipulator [Wenstrand 1968b].

An AEC inspector looked at the “new” Maintenance Building and the Warehouse during an October 1973 inspection. No remote area monitor was found in the warehouse. Two fuel canisters were found in the maintenance building, one of which appeared to have been modified by having a new section welded on. The contact exposure rate was 1 mR/hr (gross). Smear surveys and nasal swabs had been collected during the welding [Nelson 1973b, pp. 22–23].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.30 MANIPULATOR REPAIR ROOM

The MRR was located north of the MOA and below the PMCR on the ground (100-ft) level of the plant. A ceiling hatch provided access to the PMCR for repair and adjustment of the power manipulator used in the PMC. The manipulator would be positioned in the PMCR so its hand, wrist, and arm could be extended through the hatch into the MRR. This allowed it to be repaired without workers being directly exposed to the radiation in the PMCR or from the manipulator's bridge.

A.30.1 Radiological Conditions

A.30.1.1 External Exposure Data

Exposure rates in the MRR were significant during operations. The shielding of the MRR was found to be inadequate, causing exposure rates up to 50 mR/hr in adjacent areas.

In 2003 the dose rate in the MRR ranged from 10 mrem/hr to 12 mrem/hr [WVDP 2005].

A.30.1.2 Removable Contamination Data

On February 13, 1968, the MRR airlock showed removable beta contamination ranging from 76,000 dpm/100 cm^2 to 900,000 dpm/100 cm^2 [Wenstrand 1968f, p. 2].

Removable contamination up to 4500 dpm/100 cm^2 alpha and 112,500 dpm/100 cm^2 beta were measured in the MRR in 2003 [WVDP 2005].

A.31 MECHANICAL OPERATING AISLE (NORTH, EAST AND WEST)

Head-end mechanical operations were directed from operating aisles adjacent to the shielded viewing windows of the hot cells. The MOA surrounded three sides of the PMC on the ground (100 ft) level of the plant. The two sections of the MOA that ran adjacent to the east and west sides of the PMC were therefore often referred to independently as the East Mechanical Operating Aisle (EMOA) and the West Mechanical Operating Aisle (WMOA). Occasionally there will also be a reference to the north MOA, usually in relation to the MSM Repair Shop that was north of the PMC (i.e., on the other side of the MOA).

A.31.1 Radiological Conditions

A.31.1.1 External Exposure Data

An inspection by the plant housekeeping committee on January 27, 1967, found the plant to be in a "deplorable" condition. Radiological items noted in the inspection report include smoking in the WMOA, which apparently was not prohibited at the time, and an MSM in the aisle with fingers reading 5 R/hr contact and 10 mR/hr at 6 ft. The report says the MSM had been there for some time, and should either be decontaminated or moved to a less-traveled area. Shear filters in the EMOA were reading 10 mR/hr through the concrete wall at 10 in. This radiation area was not posted.

Air cylinders used in the PMC to operate various remote tools exhausted into a header that was located in the MOA approximately 10 ft above the floor. Activity accumulating in the header resulted in contact readings of 10 mR/hr to 300 mR/hr. General area exposure rates in the MOA were said to have been "less than 100 mR/hr" [Runion 1968, p. 17]. In March 1968 filters were installed in

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

pneumatic piping in the PMC, reducing exposure rates in the MOA to where only a "small area" was still designated a radiation area [Clark 1968d]. A July 1968 report said the filters prevented all but a 3-ft section of the piping from being a radiation area. The 3-ft section was shielded so that the exposure rate at 18 in. was "typically 10 mR/hr" [Clark 1968a, p.3]. NFS [1976] states buildup of radioactivity in the plant air exhaust lines from cylinders in the PMC caused exposure rates up to 24 mR/hr next to the WMOA windows.

The MRR was not adequately shielded and exposure rates in the North Mechanical Operating Aisle "reached 50 mR/hr on occasions."

A July 27, 1972, summary of plant decontamination activities mentions removal of contaminated piping and other sources from the WMOA and cleanup of fixed contamination on floors and walls in the EMOA. The report says the filter units in operation on the west end of the FRS were causing "significant dose rates" in the EMOA. It also says decontamination efforts in the MRR had resulted in a significant reduction of dose rates in the EMOA [Wenstrand 1972d, p. 2]. Another report says the decontamination efforts in the MRR reduced the exposure rate in the MOA by a factor of four [Wenstrand 1972j].

Typical dose rates in the MOA in 2005 range from 0.1 mrem/hr to 5.4 mrem/hr [WVDP 2005].

A.31.1.2 Removable Contamination Data

On October 24, 1969, at 5:00 pm the EMOA and WMOA, along with numerous other areas of the plant, became contaminated to Zone III levels. The MOAs were contaminated to 2000 to 3000 dpm/100 cm^2 beta. Maintenance work was postulated as the cause of the widespread tracking of contamination, but no definite cause was established. The report mentions that high backgrounds make detection of "low" levels of contamination (800 to 1,000 dpm beta) difficult. Overall, the cause was attributed to either failure of persons to self-monitor (using hand and foot monitors) or ineffectiveness of these monitors [Jaeger no date].

A.31.1.3 Airborne Contamination Data

In December 1966 airborne alpha activity in the MOA rose above NFS' concentration limit of 4×10^{-11} µCi/cm^3 for canister masks. The activity was attributed to contamination in the CAA that resulted from maintenance work in the Plutonium Cell [Keely 1967e].

A report dated July 8, 1970, states the east and west MOAs are some of the areas that show air concentrations greater than "40 MPC-hr" [sic] during duct flushing operations [Wenstrand 1970a].

A.32 MASTER-SLAVE MANIPULATOR SHOP AND CONTACT SIZE REDUCTION FACILITY

Construction of a new MSM Repair and Decontamination shop (to replace the Hot Shop) was under way in May 1968 [Clark 1968d]. It was completed within 1 year. The MSM Shop was used for the repair of MSMs from the mechanical and analytical cells. The MSMs were decontaminated remotely while still in place to reduce the exposure rate to about 1 R/hr. The MSMs were then packaged and transferred to the MSM Shop for further decontamination and repair.

The MSM Shop was decontaminated and renovated between July 1982 and June 1983. Contamination had penetrated the original floor to the extent that it had to be removed and repoured.

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DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

As of 1994 the MSM Shop was used to repair and adjust 25 to 30 manipulators a year. The manipulators were first decontaminated in an area adjacent to the MSM Shop.

The MSM Repair Shop transitioned to the Contact Size Reduction Facility beginning in 1987.

A.32.1 Radiological Conditions

A.32.1.1 External Exposure Data

Exposure rates up to 50 mR/hr were present in the MSM Shop from activity in the adjacent MRR.

A July 27, 1972, summary of plant decontamination activities said the “background” in the MMR area of the MSM Shop originated from the HEV System filters, the demister and the duct located on the east wall. It stated the “small room in the southwest corner of the MSM Shop” contained several sources that “should be packaged and removed ... or buried”, and that the drain line from the PMCR needed to be shielded. It said radiation levels in the MSM Shop originating from the MRR had been reduced, but other sources had been added, and that the north MSM Shop contained a “high radiation source” located by the double doors that needed to be packaged for storage or shielded [Wenstrand 1972d, p. 3].

A.32.1.2 Removable Contamination Data

Removable contamination levels exceeded 50,000 dpm/100 cm^2 in the “new” MSM Repair Area in May 1969 [Keely 1969c, p. 2].

A.33 OFF GAS AISLE AND PROCESS SAMPLE CELL 3

The OGA was in the southwest area of the process building on the 131-ft elevation. It was also known as the Off Gas Operating Aisle or the Acid Recovery Aisle. The ARC was in the southwest corner of the process building on the same level.

The OGA was used for monitoring and controlling operations in the OGC, the ARC, and part of the CPC. It also provided access to PSC-3 and acted as a passageway between laboratories, the Office Building and the south and southwest stairs [WVDP 2005]. PSC-3 contained three samplers for fission product solutions in XC1 [Runion 1968, p. 30].

A.33.1 Radiological Conditions

The OGA was impacted by the substantial spill of contaminated acid from the ARC in 1967 (Section A.1).

A.33.1.1 External Exposure Data

On November 12, 1967, the exposure rate in PSC-3 was 70 mR/hr to 150 mR/hr. The west side of the glove box read 11 rad/hr.

Radioactivity in the acid recovery system created significant exposure rates in the OGA [Runion 1968, p. 30; NFS 1976].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

On June 26, 1972, cement blocks were removed from the OGA for burial. The blocks read 5 R/hr (general area) [Nelson 1973a, p. 16]. Dose rates as high as 20 rem/hr were recorded in the OGA in 1972. Concrete was removed from this area reducing the dose rate to a maximum of 400 mrem/hr [WVDP 2005].

The July 27, 1972, summary of plant decontamination activities says the radiation level on the OGA roof around the condenser had been reduced by a factor of 10 from decontamination efforts. The major source of activity then became the backup filters [Wenstrand 1972d, p. 4].

Work was performed in the OGA the week of February 20, 1974, to remove a contaminated stainless steel pipe embedded in the floor [O'Reilly 1975]. The general area exposure rate at the start of the work was 15 mR/hr to 20 mR/hr. The exposure rate in contact with the pipe was 400 mR/hr.

As of 2005 dose rates in the OGA generally range from 0.1 mrem/hr to 0.7 mrem/hr [WVDP 2005].

A.33.1.2 Removable Contamination Data

Smearable contamination reached Zone 4 levels due to the leak in the DOG vent pipe in January 1969 [Keely 1969h, p. 2–3].

PSC-3 became contaminated from leakage of recovered acid from a valve in a sampler. Contamination also resulted from penetrations in the floor and airflow into the area from the OGA.

Staff stated the blocks removed from the OGA on June 26, 1972, were grossly contaminated, having been used to shield an acid recovery line. Analysis showed the contamination on the blocks included ^{106}Ru, ^{90}Sr, ^{239}Pu, and ^{95}Zr/Nb [Nelson 1973a, p. 17].

During decontamination work the week of February 20, 1974, removable levels were generally 4000 dpm/100 cm^2 to 5000 dpm/100 cm^2 with a maximum of 200,000 dpm/100 cm^2. Potential nuclides in the contamination included mixed fission products, uranium, and plutonium [O'Reilly 1975]. The report does not say whether the stated removable levels were alpha or beta.

A.33.1.3 Airborne Contamination Data

In January 1969 airborne activity in the OGA reached 20 MPC after failure of a temporary repair of the DOG vent pipe.

A.34 OFF GAS BLOWER ROOM

The OGBR was in the southwest section of the plant on the ground level, directly north of the ARPR. It contained blowers, filters, and scrubber recirculation pumps for the process off gas system.

The OGBR was decontaminated between November 1975 and June 23, 1976. Equipment (blowers and filters) was removed and new concrete was poured to repair parts of the floor that had been eroded away by spills from the ARC above [WVNSC 1994].

The OGBR was entered several times in the 1980s to replace HEPA filters. As of 1994 the OGBR had not been entered since the fall of 1992 [WVNSC 1994].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.34.1 Radiological Conditions

A.34.1.1 External Exposure Data

A survey inside the 6 G-3 scrubber pump niche in the OGBR on January 4, 1968, showed a whole body exposure rate of 4 R/hr at 18 in. The extremity dose rates ranged from 18 rad/hr to 20 rad/hr [Wenstrand 1968g, p. 3].

On January 15, 1968, the DOG filter in the OGBR read 600 R/hr on contact and 100 R/hr at 12 in. It read 1000 rad/hr at 10 in. [Wenstrand 1968g, p. 3]. Changing of the DOG and VOG filters was a contact maintenance activity.

As of early 1968 an operator had to enter the OGBR once a day for operation of manual valves to change solutions in the off gas scrubbers. The exposure rate in this area was 300 mR/hr [Runion 1968, p. 32].

In April 1968 “easily” removable contamination continued to be a problem in the OGBR (as well as the ARPR), giving skin dose rates of 120 mrad/hr [Keely 1968j].

In May 1968 a TLD disk was used to measure the dose rate from the floor of the OGBR, giving a result of 165 rad/hr [Keely 1968e].

In June 1968 the typical working area exposure rate in the OGBR was 1 to 2 R/hr [Keely 1968a].

Exposure rates of 15 R/hr were encountered during cleanup of the “niche” in the OGBR in November 1972 [Morrow 1972d].

HEPA filters removed for replacement in 2000 and 2001 read 55 mrem/hr [WVDP 2005].

In 2003 dose rates in the southwest part of the OGBR ranged from 70 mrem/hr to 700 mrem/hr [WVDP 2005].

As of 2005 the general area dose rate in the OGBR ranged from 10 mrem/hr to 80 mrem/hr [WVDP 2005].

A.34.1.2 Removable Contamination Data

As of early 1968 the OGBR was chronically contaminated from replacement of the DOG and VOG filters [Runion 1968, p. 32].

In May 1968 cutie pie readings on smears from the OGBR indicated a maximum removable level of 4×10^8 dpm/100 cm^2 [Keely 1968e].

A.35 OFF GAS CELL

The OGC was in the southwest section of the plant on the ground level. It was directly north of the OGBR. The OGC contained equipment for the VOG and DOG systems.

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.35.1 Radiological Conditions

The OGC was another one of the areas affected by the 1967 acid spill from the ARC above (Section A.1). This lead to high radiation levels in the northwest corner of the cell [WVDP 2005].

A.35.1.1 External Exposure Data

In August 1972 an entry was made into the OGC for a radiation survey for the first time since fuel reprocessing began. The general area exposure rate was between 10 R/hr and 20 R/hr [Wenstrand1972k]. The radiation levels along the floor were 20 R/hr to 25 R/hr and as high as 50 R/hr in the northwest corner of the cell.

In 2002 dose rates in the area of the DOG scrubber ranged from 94 mrem/hr at 28 ft from the floor to 1,600 mrem/hr at floor level. Dose rates in the northwest corner of the cell ranged from 270 mrem/hr at 28 ft from the floor to 1,500 mrem/hr at the floor.

A.36 PERMANENT VENTILATION SYSTEM BUILDING

The Permanent Ventilation System Building was constructed by the WVDP to provide ventilation to the Supernatant Treatment System (STS) Support Building, the STS Valve Aisle, and the STS Pipeway. The building contains equipment (filters primarily) to treat ventilation exhaust air before release to the atmosphere. The Permanent Ventilation System Building has its own stack.

A.36.1 Radiological Conditions

A.36.1.1 External Exposure Data

Exposure rates are reported to be 80 mrem/hr to 170 mrem/hr on contact with the filter housing [WVDP 2005].

A.36.1.2 Removable Contamination Data

Routine surveys do not show detectable removable contamination. Removable beta of 2.5×10^6 dpm/100 cm^2 has been measured in the inlet plenum [WVDP 2005].

A.37 PROCESS CHEMICAL ROOM

The PCR was above the north end of the CPC on the 144-ft elevation of the plant. It was directly north of and adjacent to the Hot Acid Cell. It contained four chemical makeup tanks and associated pumps.

A.37.1 Radiological Conditions

A.37.1.1 External Exposure Data

In June 1969 the general exposure rate in the PCR increased from 30 mR/hr to 200 mR/hr with localized spots around the sampler reading as high as 5 R/hr [Keely 1969d].

The July 27, 1972, summary of plant decontamination activities says radiation levels in the PCR were significantly reduced when the Hot Acid Cell was decontaminated.

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

As of 2005 the dose rate in the PCR was around 1 mrem/hr [WVDP 2005].

A.37.1.2 Removable Contamination Data

The July 27, 1972, summary of plant decontamination activities says radiation levels in the PCR were significantly reduced when the Hot Acid Cell was decontaminated. The reduced levels allowed the detection of contamination in the PCR that had previously been "obscured" by the high background [Wenstrand 1972d, p. 4].

A.38 PROCESS MECHANICAL CELL

The PMC is where extraneous hardware was removed from fuel assemblies, which were then sheared into small pieces for dissolving. The sheared fuel dropped through a discharge chute into baskets in the GPC below.

The PMC included high-workload manipulators for handling fuel elements. Maintenance and repair of these manipulators (performed after they were removed from the cell) was a significant source of personnel exposures because of the substantial contamination from fuel particles. Exhaust from air cylinders used in the PMC to operate various remote tools was a source of contamination and elevated exposure rates in other areas of the plant, such as the MOA.

A general cleanup of the PMC was conducted between October 22, 1971, and September 15, 1972 [WVNSC 1994]. The WVDP completed deactivation of the PMC in 2004 [WVDP 2005].

A.38.1 Radiological Conditions

The PMC was never a routinely occupied area.

A survey of the PMC in April 1972 showed exposure rates ranging from 1,500 R/hr to 4,500 R/hr [Wenstrand 1972j].

In 1986 gamma exposure rates in the PMC ranged from 0.8 R/hr to 40 R/hr at 6 ft from the floor. Beta-gamma levels ranged from 40 rem/hr to 240 rem/hr [WVDP 2005].

Surveys performed after the 2004 cleanup in grid locations 3 ft above the floor showed dose rates ranging from 1.5 rem/hr to 40 rem/hr [WVDP 2005].

A.39 PRODUCT PACKAGING AND SHIPPING, PRODUCT PACKAGING AND HANDLING, URANIUM PRODUCT CELL, AND URANIUM LOADOUT AREAS

The aqueous plutonium and uranium nitrate products from fuel reprocessing were packaged in 10 L shipping containers in the PPH area. They were then shipped by truck from the PPS area above. The two were connected via an airlock [WVDP 2005]. During operations the PPH contained glove boxes used for filling of product bottles. One was used for plutonium and the other for uranium.

The PPH became the Waste Reduction and Packaging Area under the WVDP [WVNSC 1994]. In 2005 it was being used for compressing LLW into metal storage containers using a 50-t hydraulic compactor [WVDP 2005].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

The UPC was on the ground level directly north of the PPC. It contained tanks for the collection and storage of uranyl nitrate before analysis and shipment. The function of these tanks was modified in 1988 to where they became hold tanks for the WVDP Low-level Waste Treatment System [WVDP 2005].

The Uranium Loadout Area (ULO), also on the ground level, was directly east of the UPC. The ULO contained a 4,000-gal weigh tank used to measure shipments of uranyl nitrate before transfer to tanker trucks or to the back of the cell for further processing [WVDP 2005].

A.39.1 Radiological Conditions

A.39.1.1 External Exposure Data

May 1966: Several shipments of uranium product were made and two bottles of plutonium product were loaded out. The maximum radiation level on the Pu bottles was 130 mR/hr [Loud 1966d].

Exposure rates at 6 in. from the PPH loading dock on January 20, 1968, were 2 R/hr whole body and 17 rad/hr extremity [Wenstrand 1968g, p. 3].

In 1996 dose rates in the UPC ranged from 10 mrem/hr to 450 mrem/hr. Surveys in 2003 showed dose rates of 120 mrem/hr on the underside of one of the waste tanks and 110 mrem/hr between the two tanks [WVDP 2005].

As of 2004 the dose rate in the PPH area was less than 1 mrem/hr [WVDP 2005].

As of 2005 the dose rate in the ULO area was typically less than 1 mrem/hr, with levels up to 29 mrem/hr in the pump room [WVDP 2005].

A.39.1.2 Removable Contamination Data

Decontamination efforts in the UPC in April 1972 reduced the typical removable alpha contamination levels from 10^7 dpm/100 cm^2 to 10^5 dpm/100 cm^2 [Wenstrand 1972j].

In 1996 surveys showed removable levels up to 32,000 dpm/100 cm^2 alpha and 348 dpm/100 cm^2 beta in an area north of one of the tanks [WVDP 2005].

Low levels of removable contamination were present in the ULO area as of April 2005. Removable was not detected elsewhere in the area [WVDP 2005].

A.39.1.3 Airborne Contamination Data

The airborne activity levels in the PPS area exceeded the MPC during the week of March 18, 1968 [Wenstrand 1968b].

In June 1969 long-lived alpha air activity in the PPH area exceeded the MPC for “a major portion of the month.” The air activity was said to originate from contamination present on the north wall by the High Enriched Loadout Box [Keely 1969d].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.40 PULSER EQUIPMENT AISLE

The PEA was immediately south of the XCR on the 160-ft elevation of the process building. The PEA included air pulsers for the extraction columns in the extraction cells and other equipment. This equipment was removed in 1980 and the area was reconfigured to support operation of the LLW treatment system. The former PEA no longer exists [WVDP 2005].

A.40.1 Radiological Conditions

A.40.1.1 Airborne Contamination Data

The airborne activity levels in the PEA exceeded the MPC during the week of March 18, 1968. The same document states this condition existed in both the PEA and the PPS area, suggesting the MPC of concern was that for alpha (^{239}Pu) [Wenstrand 1968b, p. 2].

A.41 REMOTE HANDLED WASTE FACILITY

The RHWF was built by the WVDP. It went into operation in June of 2004. It is used for size reduction and packaging of equipment containing large amounts of radioactive materials.

A.42 SAMPLE STORAGE CELL, ANALYTICAL HOT CELLS, AND ANALYTICAL DECONTAMINATION AISLE

The Sample Storage Cell (SSC) was on the east side of the CPC at the 131-ft elevation (i.e., the analytical level of the plant). It contained a conveyor-elevator for transferring samples to the Analytical Hot Cells or laboratories. The SSC had three shielded viewing windows and three sets of manipulators in its north wall. As of 2005 the SSC remained highly contaminated and showed dose rates from 126 mrem/hr up to nearly 20 rem/hr [WVDP 2005, p. 51].

Entries were sometimes made into the SSC for periodic maintenance of equipment in the Analytical Hot Cells that could not be removed. These entries were made via the Analytical Decontamination Aisle (ADA). The SSC would be remotely decontaminated before these entries, and internally contaminated items such as sample bottles would be removed via the conveyor system.

On the east side of the ADA were five Analytical Hot Cells and Sample Cell 2C. The ADA was sometimes referred to as the Analytical Cell Decontamination Area.

A.43 SCRAP REMOVAL AREA

The SR area was on the ground (100 ft) level of the process building northeast of the CPC. This area is also sometimes referred to as the SR Cell or SR Room.

Leached hulls and process wastes (including failed equipment) were placed in 30-gal drums for burial. The drums were loaded inside the GPC, then passed through a hatch to the SR area. They were sprayed with water as they passed through the hatch and then transferred to a shielded cask inside the SR area. Excess water was allowed to drip off before the drums were placed in the transfer cask. The cask was placed on a trailer (which had been backed into the SR area) and both were decontaminated before being released to go to the HLW burial area.

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

Decontamination of the SR area took place in 1972 and again in 1980. D&D of the SR Room took place between March and September 1986. The SR Room now contains Tank 7D-13, which formerly received liquids from the plant laundry and analytical laboratory drains. As of 1994 the tank received liquids from laboratory drains and from drum flushing operations at the Concrete Solidification System [WVNSC 1994].

A.43.1 Radiological Conditions

A.43.1.1 External Exposure Data

A dissolver removed from the CPC in July 1969 read 30 R/hr at contact with its shield cask and 50 mR/hr at 50 ft (upon removal from the SR area) [Keely 1969e].

The 1986 D&D of the SR area reportedly reduced exposure rates in the SR Room to less than 1 mR/hr [WVNSC 1994]. However, other references report levels from 1 mrem/hr to 40 mrem/hr at 1 ft from the floor in 2003, and up to 10 mrem/hr at 3 ft from the floor [WVDP 2005].

A.43.1.2 Removable Contamination Data

The SR area was a chronic source of contamination because of the waste packing and removal operations. Typical contamination levels for the drums is said to have been in excess of 500,000 dpm/100 cm^2 as they came from the GPC to the SR area [Runion 1968, pp. 18–19].

A.43.1.3 Airborne Contamination Data

In January 1967 repair of the seal around the SR door resulted in differential pressure between the GPC and the SR Area, with the latter becoming an airborne activity area on several occasions thereafter [Keely 1967d, p. 2].

A.44 SOLVENT STORAGE TERRACE

The Solvent Storage Terrace (SST) was located on the south side of the process building on the roof of the UWA at the 131-ft elevation. Storage tanks for recovered acid and solvent wastes were located on the SST. The Recovered Acid Storage Tank received acid from the bottom of the acid fractionator located in the ARC. The solvent waste tanks received solvent from the extraction cells [WVNSC 1994].

A.44.1 Radiological Conditions

The SST became contaminated from spills and leaks from the tanks. Liquids from the SST drained to an unlined holding pond near Lagoon 1 known as the “solvent dike” [WVNSC 1994].

A.44.1.1 External Exposure Data

On November 4, 1967, the solvent storage tank read 2.5 R/hr to 3 R/hr on contact.

In June 1968 a survey of the SST showed hot spots on the floor reading up to 100 R/hr. The exposure rate at the bottom of the solvent storage tank was 300 R/hr [Keely 1968a].

As of 2005 dose rates on the SST were typically 0.1 mrem/hr or less [WVDP 2005].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.44.1.2 Removable Contamination Data

In July 1968 smears from the SST indicated removable beta contamination levels of 1×10^8 dpm/100 cm^2 [Keely 1968k].

In March 1972 a leaking connection during solvent loadout caused removable beta contamination up to 10^7 dpm/100 cm^2 on the SST [Wenstrand 1972l].

A.45 SUPERNATANT TREATMENT SYSTEM SUPPORT BUILDING

The STS Support Building was built by the WVDP to support STS operations. It is a two-story building located adjacent to and on top of the HLW storage tank 8D-1. The first level contains the valve aisle, operating aisle, and control room. The second story houses tanks and delivery systems. The STS Support Building supported STS operations from 1988 through 1995. Subsequently it was used to treat vitrification waste streams and for processing of sodium-bearing wastewater.

A.45.1 Radiological Conditions

A.45.1.1 External Exposure Data

In May 2002 dose rates in the valve aisle ranged from 0.9 rem/hr to 8.2 rem/hr 4 ft from the back wall. The dose rate ranged from 0.5 rem/hr to 3.3 rem/hr 12 in. from the viewing windows [WVDP 2005].

A.45.1.2 Removable Contamination Data

Surveys on the first and second floors outside the valve aisle typically do not show detectable removable contamination. Removable contamination is said to be “very high” inside the valve aisle [WVDP 2005].

A.46 UPPER EXTRACTION AISLE

The UXA was immediately north of the extraction cells on the 144-ft elevation of the process building. It served as an operations and maintenance area for extraction cell and other equipment. The area included valves, piping, instrumentation, ventilation ducts, and stack monitoring equipment.

A.46.1 Radiological Conditions

A.46.1.1 External Exposure Data

As of early 1968 a high radiation area existed on the UWA in the area of a recovered acid line [Runion 1968, p. 25].

Radiation levels around the 7C-1 jet in the UXA reached 10 R/hr at 18 in. twice during the month of May 1969 [Keely 1969c, p. 3].

The July 27, 1972, report on plant decontamination status said radiation from the west wall of the UXA appeared to be caused by “a shielding flaw between the aisle and the steam ventilation filter bank” [Wenstrand 1972d, p. 4].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

In 2004 the general area dose rate in the UXA ranged from 0.1 mrem/hr to 6 mrem/hr. In 2005 the range was 0.1 mrem/hr to 4.0 mrem/hr [WVDP 2005].

A.46.1.2 Removable Contamination Data

Horizontal surfaces in the UXA showed removable levels up to 280,000 dpm/100 cm^2 beta on January 1, 1968 [Wenstrand 1968g, p. 3].

A jet was mistakenly left open on May 25, 1972, allowing an estimated 1.5 to 2 gal of contaminated acid and water to leak onto the floor of the UXA. Surveys of the surrounding floor area showed 50,000 cpm alpha. The activity in the puddle was determined to be ^{239}Pu.

A.47 UPPER WARM AISLE AND PROCESS SAMPLE CELL 1

The UWA was immediately above the LWA, adjacent to the extraction cells on the 114.5-ft elevation. Like the LWA, it contained concrete niches that housed pumps, valves, and other equipment used in the solvent extraction process. There were six of these niches extending into the UWA. The equipment in the niches required routine maintenance or replacement, resulting in contamination of the area [Runion 1968, p. 24].

The UWA also contained PSC-1, which was on the northeast end of the aisle. PSC-1 was a shielded stainless-steel box that housed a glovebox and samplers for sampling purified plutonium and uranium solutions from the PPC.

NFS decontaminated the UWA, including the niches, in the 1972 to 1974 period. They also replaced several of the pumps. The aisle and niches were also decontaminated by the WVDP in 1985 [WVDP 2005]. Equipment niches and associated equipment were removed from the UWA in 1986 [WVNSC 1994].

A.47.1 Radiological Conditions

A.47.1.1 External Exposure Data

A September 18, 1967, letter from NFS to the AEC says the UWA area in general (i.e., outside the pump niches) only became a high radiation area when maintenance work was being performed [Runion 1967a, p. 3].

On November 13, 1967, the exposure rate at the top of the 13G-2 pump niche in the UWA was 2.5 R/hr to 3.0 R/hr. The pump read 60 rad/hr and 12 R/hr at 2 in. The 13G-1 pump niche read 6 R/hr at the top and 20 R/hr to 30 R/hr at the bottom. The pump stand read 50 R/hr at contact.

Surveys in the UWA on January 20, 1968, showed exposure rates of 10 R/hr and dose rates up to 35 rad/hr [Wenstrand 1968g, p. 3].

Typical dose rates in the UWA as of 2005 ranged from 0.1 mrem/hr to 0.5 mrem/hr for the general area. There was a hot spot up to 35 mrem/hr on a drain line and up to 15 mrem/hr on the south wall.

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

Surveys inside pump niches 1 through 5 in December 2003 to January 2004 showed the following maximum levels, each near the floor [WVDP 2005]:

- UWA pump niche 1 = 40 mrem/hr;
- UWA pump niche 2 = 1.8 mrem/hr;
- UWA pump niche 3 = 45 mrem/hr;
- UWA pump niche 4 = <1 mrem/hr; and
- UWA pump niche 5 = <1 mrem/hr.

A.47.1.2 Removable Contamination Data

On November 17, 1967, removable alpha in PSC-1 ranged from 150 dpm/100 cm^2 to 3700 dpm/100 cm^2 with hot spots on the floor as high as 320,000 dpm/100 cm^2.

Sample bottles in PSC-1 were bagged and transferred into a glovebox. The sampling station would periodically become contaminated from these operations, allegedly from inadequate bagging. Entries into the PPC also resulted in contamination of the sampling station from airflow reversals into the glovebox through its vent.

Removable beta contamination levels in the UWA ranged from 200,000 dpm/100 cm^2 to 1.5×10^6 dpm/100 cm^2 on January 22, 1968 [Wenstrand 1968g, p. 4]. In February 1968 smears of the UWA showed removable beta levels of 1.2×10^6 dpm/100 cm^2 [Wenstrand 1968f, p. 2].

In May 1968 “aggressive” decontamination efforts reduced removable beta contamination levels to Zone 3 limits in both the UWA and LWA [Keely 1968e].

In August 1968 removable contamination in the UWA showed a maximum of 8×10^6 dpm/100 cm^2 [Keely 1968b].

On October 29, 1968, smearable alpha contamination in PSC-1 reached 250,000 dpm/100 cm^2 during maintenance work [Keely 1968d].

Contamination in the UWA was reduced from Zone 4 to low Zone 3 during the month of February 1969 [Keely 1969f].

In August 1969 removable alpha contamination was 1×10^6 dpm/100 cm^2 in PSC-1 after modifications to sample lines and installation of a glove box [Wenstrand 1969c].

Removable contamination levels in the UWA before decontamination in 1985 ranged up to 2,070 dpm/100 cm^2 alpha and 19,800 dpm/100 cm^2 beta [WVDP 2005].

As of 2005 removable levels on the UWA were nondetectable, but levels up to 312,000 dpm/100 cm^2 beta-gamma had been measured in the niches [WVDP 2005].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.47.1.3 Airborne Contamination Levels

In October 1967 air samples "taken on each shift" from the XSA and PSC-1 showed long-lived airborne alpha concentrations of 2×10^{-12} µCi/cm^3 and 5.4×10^{-11} µCi/cm^3 [Wenstrand 1967d, p. 3].

A.48 UTILITY ROOM AND STAIRWELLS

The Utility Room was a concrete block structure adjacent to the process building on its south side. It shared its north wall with the process building. This wall was reinforced concrete [WVDP 2005].

The process building contained four main reinforced concrete stairwells having landings on the various levels. These were:

- *East Stairs*. Near the PPC extending from the ground level to the 160-ft elevation;
- *North Stairs*. Near the GOA extending from 76.5 ft (below ground level, i.e., the level of the GOA) up to the 131-ft (analytical) level;
- *South Stairs*. North of the Utility Room extending from ground level up to the 160-ft elevation; and
- *Southwest Stairs*. South of the office area extending from ground level to an elevation of 152 ft.

A.48.1 Radiological Conditions

The stairways (South and Southwest in particular), as well as the Utility Room, became contaminated from spills and frequent airflow reversals that would occur when doors were opened. These reversals resulted in flow of contamination from acid recovery and off gas areas into the stairwells and the Utility Room.

A.48.1.1 External Exposure Data

In June 1969 exposure rates in the Southwest Stairwell increased from 20 mR/hr to 90 mR/hr during the "Yankee campaign"; that is, higher burnup fuel [Keely 1969d].

Exposure rates up to 6 R/hr were measured in the lower levels of the Southwest Stairs in the early 1970s [WVDP 2005].

The July 27, 1972, summary of plant decontamination activities says the radiation levels in the shower and locker room areas originating from the Southwest Stairwell remained "essentially the same" as they did in March 1972 [Wenstrand 1972d, p. 3].

A typical dose rate of 30 mrem/hr with hot spots up to 125 mrem/hr are reported for the North Stairs on the east wall south of a viewing window into the PMCR, but no time frame is given for when or how long these exposure rates existed. Levels up to 20 mrem/hr gamma and 130 mrem/hr beta-gamma are reported for "different times in other areas" of the north stairwell [WVDP 2005].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

In 1990 the dose rate in the south stairs at the Control Room landing (144-ft level) ranged up to 9 mrem/hr [WVDP 2005].

Typical dose rates for the stairwells in 2004 were [WVDP 2005]:

- East stairs = 0.1 mrem/hr to 0.5 mrem/hr;
- North stairs = 0.1 mrem/hr to 0.5 mrem/hr;
- South Stairs = 0.1 mrem/hr to 5.0 mrem/hr; and
- Southwest Stairs = 0.1 mrem/hr to 14 mrem/hr.

Typical dose rates for the stairwells in 2005 were [WVDP 2005]:

- East stairs = <0.04 mrem/hr to 1.1 mrem/hr;
- North stairs = <0.04 mrem/hr to 0.6 mrem/hr;
- South Stairs = 0.2 mrem/hr to 4.8 mrem/hr; and
- Southwest Stairs = 0.1 mrem/hr to 22 mrem/hr.

The elevated dose rates for the Southwest Stairs were at the second landing.

A.48.1.2 Removable Contamination Data

In January 1967 gamma analysis on sand from the Utility Room trench, which was reading 20 mR/hr to 50 mR/hr, showed “mostly Zr/Nb-95” [Keely 1967d, p. 3].

As of early 1968 the South Stairwell had become contaminated from an airflow reversal from the OGBR, which was located at the base of the stairway. These reversals occurred whenever the door at the base of the stairway was opened [Runion 1968, p. 33]. Subsequently, the OGA and Southwest Stairway became contaminated since they received air from the South Stairs [Runion 1968, p. 34].

In October 1968 removable contamination on a wall in the Southwest Stairwell was 5×10^6 dpm/100 cm^2 around a leak through the wall of the ARC [Keely 1968d].

The Southwest Stairwell became contaminated to Zone 4 levels by an acid leak in the OGA on October 23, 1969 [Wenstrand 1969c].

A.48.1.3 Airborne Concentration Data

On July 8, 1970, the East and North stairs showed concentrations “close to” 40 MPC-hr [sic] during flushing of the ventilation ducts [Wenstrand 1970a].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

An air sample collected from the Utility Room in January 1974 showed an alpha concentration of 5.8×10^{-12} µCi/cm^3. Isotopic analysis of the alpha activity showed the following:

- Pu-238 = 7.6 wt%;
- Pu-239 = 76.1 wt%;
- Pu-240 = 11.2 wt%;
- Pu-241 = 4.3 wt%; and
- Pu-242 = 0.8 wt%.

A.49 VENTILATION EXHAUST CELL

The VEC was located on the roof of the CPC at an elevation of 148 ft. It contained ductwork, filters, blowers, and controls for the main ventilation system. The VEC contained two parallel ventilation systems: electric powered for routine operations and steam powered as a backup. The two filter banks were therefore identified as “steam” or “electric.”

A.49.1 Radiological Conditions

A.49.1.1 External Exposure Data

During a filter changeout in 1994 the roughing filters read 1.8 rem/hr gamma at 2 in. The HEPA filters read 400 mrem/hr to 500 mrem/hr at 2 in. The floor of the filter housing ranged up to 1 rem/hr gamma [WVDP 2005].

As of 2005 general dose rates in the VEC ranged from 1 mrem/hr to 100 mrem/hr [WVDP 2005].

A.50 VENTILATION SYSTEMS

Overall, the single largest source of personnel exposures at West Valley during its operations period was the plant ventilation system. The unanticipated magnitude of the problem of high specific activity fuel particles, coupled with plateout of these particles within the system and its failure to maintain necessary pressure differentials, was both a direct and indirect source of personnel exposure throughout the plant. The main ventilation system serviced the entire reprocessing plant from startup until October 25, 1970. On October 25, 1970, a new HEV system came on line that provided additional ventilation capacity for the head end process cells and operating aisles.

A.50.1 Main Ventilation System

The main ventilation system consisted of an air washer followed by roughing and absolute filters along with associated blowers and ducting. After exiting the washer, the ventilation air was routed through a duct along the ceiling of the LXA, around the outside of the laboratories, over the top of the Control Room, and into the filter plenum in the VEC.

Particulate activity collecting on the interior of the ducting gave contact exposure rates from 100 mR/hr to 800 mR/hr. The system was modified in 1967 to allow periodic flushing of the ducts, which

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

would offer temporary relief until the contamination built up again. These duct-flushing operations would also result in high airborne activity levels throughout the plant. In addition to the chronic local and area exposure rate problems associated with the ventilation system ducting, another significant source of personnel exposure was that associated with changing the filters (roughing and absolute) in the VEC.

After filtration, the process building ventilation system air was either exhausted through the stack or recycled. Recycled air was used in the LLWT Plant, the HLW Storage Facility, and the FRS. The recycle air passed through HEPA filters before reuse.

Other ventilation systems, which served to keep process vessels at negative pressure relative to the cells, which were at negative pressure relative to the aisles, included:

- The DOG system,
- The VOG system (for process vessels), and
- The Waste Tank Farm ventilation system.

The DOG system treated gases from the vacuum dissolvers in the CPC and purge gases from the fuel bundle shear in the PMC. The DOG system included scrubbers to reduce nitrogen oxides, which were also effective for removing particulates and iodines. North and Clark [1968] state silver nitrate reactors in the DOG system “have not been put into operation because the long-cooled fuel contains an insignificant amount of iodine-131.” The DOG system effluents first passed through HEPA filters before discharge via the plant stack.

The VOG system ventilated gases from process vessels and tanks to the OGC, where it passed through scrubber and filters before discharge to the main ventilation system [WVNSC 1994].

The removal efficiency of the off gas scrubbers was lower than expected [NFS 1976]. This resulted in higher than expected activity in the DOG and VOG absolute filters. Periodic changing of these filters, which was a contact maintenance activity, was a significant source of personnel exposures.

A document from ca. 1967 gives the criteria for replacement of the DOG filters, saying a filter shall be replaced whenever the exposure rate at 2 in. exceeds 500 R/hr. This measurement was to be taken whenever the exposure rate through the shield plug exceeded 6 R/hr, i.e., once it got to 6 R/hr the plug was to be removed and a measurement taken at 2 in. [Clark 1968e].

Following the modification of the plant to add the dedicated HEV system (see next section), the original building ventilation system serviced the nonprocessing areas of the plant, including the operating aisles, office areas, chemical makeup area, maintenance area, FRS pool area and the laboratories [NFS 1973b, p. 336].

A.50.2 Head End Ventilation System

Before 1970, the plant’s main ventilation system served all of the plant, including the head end process cells and operating areas. In an effort to relieve the main ventilation system of the substantial airborne radioactivity generated in the head end processing, the HEV system was activated in 1970. The radioactivity removed by the HEV was greater than expected, resulting in dose rate problems in

several areas adjacent to the HEV building. In addition, periodic replacement of the HEV filters was a source of significant personnel exposures.

The HEV system was added to the plant in response to the operational difficulties encountered from the high-activity particulate contamination associated with mechanical segmentation of fuel assemblies. The HEV system was expected to increase the differential pressure between occupied and contaminated areas, to increase the airflow through contaminated areas where contact maintenance was performed (the MSM shop in particular), and to divert the bulk of the particulate contamination to a dedicated filter system instead of the main ventilation system filters [Preliminary safety 1969]. NFS felt their existing stack sampling and monitoring equipment could handle the increased flow.

The HEV system was licensed on October 20, 1970, and placed into operation October 25, 1970 [NFS 1972a, p. 1]. It serviced the cells and operating aisles involved in the mechanical processing of fuel. The air flowed into filter trains in the HEV Building and then out to the plant main stack [WVNSC 1994].

NFS states the HEV system reduced the exposure rates from the main ventilation system components, thus reducing personnel exposures associated with the main ventilation system ducting, air washer, etc. Exposure rates were reduced “significantly” in the Control Room, analytical aisles, and extraction aisle. Personnel exposures associated with changeout of main ventilation system filters were reduced, and the change frequency also decreased. The HEV system also reduced contamination levels in the airlocks and crane rooms, and reduced particulate emissions from the main stack. However, the location of the HEV exhaust duct near the fuel shearing operation resulted in higher than expected loading of the HEV filters and buildup of activity in the duct. The activity in the HEV duct and filters was responsible for 69 person-rem/yr of collective dose.

A.51 VENTILATION WASH ROOM

The Ventilation Wash Room (VWR) was on the 114.5-ft elevation of the plant, directly south of the PMC. It contained the air washer for the main ventilation system. The purpose of the washer was to intercept particulate activity from the saw and shear in the PMC.

A.51.1 Radiological Conditions

Buildup of particulates in the ventilation air washer filters after several years of plant operation necessitated replacing the filters. Changeout of these filters and the air washer pump were a significant source of exposure for maintenance workers. Per NFS [1976], these two sources resulted in collective dose for maintenance workers of about 30 person-rem/yr from the VWR. However, this value likely reflects exposures after the HEV system came on line in October 1970. Personnel exposures before the HEV being in operation would have been higher. The air washer was decontaminated after the plant shut down in 1972.

In 2001 the dose rate on the ventilation duct in the VWR upstream of the inlet plenum was 1.5 rem/hr. The average for the entire length of the duct from the VWR to the VEC averaged 34 mrem/hr at 6 in. with a maximum of 130 mrem/hr [WVDP 2005].

ATTACHMENT A
DESCRIPTIONS OF OPERATIONS AREAS AND SYSTEMS WITH EMPHASIS ON RADIOLOGICAL CONDITIONS (CONTINUED)

A.51.1.1 External Exposure Data

The working area exposure rate for a pump changeout in the VWR was in October 1969 was 10 R/hr to 15 R/hr.

1983 estimates of exposure rates in the VWR were general levels of 200 mR/hr to 1,000 mR/hr and in excess of 5 R/hr at the washer itself [WVNSC 1994, p. 99]. In 2001 the dose rate on the floor ranged from 10 mrem/hr to 1,200 mrem/hr gamma. The highest dose rate was in an area around the floor drain that read 2,500 mrem/hr beta-gamma [WVDP 2005].

A.51.1.2 Airborne Contamination Data

An air sample taken in the VWR on October 23, 1967, showed 4×10^{-10} µCi/cm^3 of “long-lived alpha” and 3×10^{-10} µCi/cm^3 of “long-lived beta” [Wenstrand 1967d, p. 2].

A.52 WASTE TANK FARM

The Waste Tank Farm (WTF) area included the underground tanks used to contain high-level liquid wastes from fuel reprocessing. There were separate tanks for PUREX waste (neutralized) and THOREX wastes (not neutralized). The PUREX tank was 750,000 gal. The THOREX tank was 15,000 gal. Separation and solidification of these high-level liquid wastes were a major focus of the WVDP.

A.52.1 Radiological Conditions

A.52.1.1 External Exposure Data

An April 19, 1973, memorandum [Wenstrand 1973, p. 2] describes “major problems” associated with radiation levels at the WTF shelter. It says the exposure rates make the area a high radiation area, so it must be locked. The sources of the exposure rates were vent lines, piping, and resin units.