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**TRANSURANIC CONTAMINATED
WASTE FORM CHARACTERIZATION
AND DATA BASE**

VOLUME 2

Submitted to

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APPENDIX A
GLOSSARY OF
TERMS AND DEFINITIONS

TERA CORPORATION

Actinide Series: The series of elements beginning with actinium, element No. 89, and continuing through lawrencium, element No. 103, which together occupy one position in the periodic table. The series includes uranium, element No. 92, and all the man-made transuranium elements. The group is also referred to as the "actinides."

Activation: The process of making a material radioactive by bombardment with neutrons, protons, or other nuclear particles. Also called radio-activation. The resultant radioactive materials are commonly called activation products.

Activity: A measure of the rate at which a material is emitting nuclear radiations; usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time; the standard unit of activity is the curie (Ci).

Alpha Particle (α): A positively charged particle emitted by certain radioactive materials. It is made up of two neutrons and two protons; hence it is identical with the nucleus of a helium atom.

Alpha Waste: Waste material that is contaminated by radionuclides that emit alpha particles, particularly (in this text) the transuranium elements.

Anion: A negatively charged ion.

Aqueous Phase: The liquid phase in solvent extraction which consists of dilute nitric acid.

Back End of the Fuel Cycle: Includes reactors, spent fuel storage, fuel reprocessing, mixed-oxide fuel fabrication, and waste management.

Beta Particle (β): An elementary particle emitted from a nucleus during radioactive decay. It has a single electrical charge and a mass equal to $1/1837$ that of a proton. A negatively charged beta particle is physically identical to the electron. If the beta particle is positively charged, it is called a positron (β^+).

Burial Grounds: Areas designated for storage of containers of radioactive wastes by near-surface burial in geologic media.

Calcination: A process of solidifying high level liquid waste by evaporating to dryness on a granule and converting from the nitrate form to the oxide form.

Calcine: Material heated to a temperature below its melting point to bring about loss of moisture and oxidation to a chemically stable form.

Cask: A container that provides shielding and containment for transportation of retrievable storage of solid waste. The shielding is normally steel or concrete, and the waste is contained in a canister within the cask.

Cation: A positively charged ion.

Cermets: Ceramic-metal systems suitable for use at high temperatures. They have some of the desirable properties of both ceramics and metals, e.g., high melting point, high oxidation resistance, high thermal conductivity, and shock resistance.

Cladding Waste (hulls): Fuel rods in most nuclear reactors today are made up of fissionable materials clad in a protective alloy sheathing that is relatively resistant to radiation and the physical and chemical conditions that prevail in a reactor core. The spent fuel rods, after removal from the reactor and storage to permit radioactive decay of the short-lived fission products, are removed and in certain fuel cycle processing systems are chopped up, and the residues of the fissionable materials are leached out chemically. The remaining residues, principally the now radioactivated cladding material (zirconium alloys, stainless steel, etc.) and insoluble residues of nuclear fuel, fission products, activation products and transuranium nuclides, are left behind as cladding waste, which is a special category of transuranium radioactive waste.

Combustibles: A "material capable of being burned, except that if the only combustible content of a package is plastic lining or wrapping used for contamination control purposes around incombustible objects or materials, the contents of the package as a whole may be considered noncombustible." (DOE Manual 0511--Radioactive Waste Management)

A precise definition of the term "combustible" is difficult, if not impossible, to prepare. This difficulty arises because the combustible nature of a given material is dependent not only upon the form of the material but also on the conditions of ignition which must be considered. It is the intent of this definition that emphasis be placed on combustion. However, one should not ignore the long-term phenomena of pyrolysis and thermal decomposition.

This definition of "combustible" is applied to a material under the conditions existing (or possibly foreseen) in the disposal facility. Thus, extreme conditions and the method of packaging need not be included as factors in the determination of combustibility of the package. Further, the production of flammable gases need not be a consideration in determining combustibility of the waste material if the control of such gases can be accomplished by separate means. The overall intent of the definition is to consider those materials that may burn in the common sense of the word and that could lead to or add to a fire if not properly protected. The principal type of material considered as combustible should be the organics, which are composed mostly of hydrogen, carbon and oxygen. Those materials classed as fire-retardant, either by nature of the composition or because of treatment, should also be included since they will release heat and flammable gases if exposed to fire from other sources and can, therefore, add to the overall intensity of a fire.

Combustion: The rapid and self-propagating oxidation of materials in air, with the evolution of significant quantities of heat.

Concrete Matrix: A process to fix calcine in a monolithic form by mixing with a concrete slurry.

Contamination: The deposition of radioactive material on a surface or the presence of fission products in a process stream

Contamination-Free: "A condition of the outer surfaces of stored containers, as determined by appropriate swipe surveys or direct radiation instrument surveys, sufficiently free of contamination so that under standard radiation work procedures for the site in question respiratory protection will not be required during container handling." (ERDAiv. 0511--Radioactive Waste Management)

Countercurrent: The opposed flow of the aqueous phase and organic phase in an extraction column.

Crib: A back-filled burial trench into which intermediate-level liquid waste is distributed through a perforated pipe.

Critical: The condition in which a material is undergoing nuclear fission at a self-sustaining rate; the critical mass of a material is the amount that will self-sustain nuclear fission when placed in an optimum arrangement in its present form.

Curie (Ci): A unit of radioactivity defined as the amount of a radioactive material that has an activity of 3.7×10^{10} disintegrations per second (d/s); millicurie (mCi) = 10^{-3} curie; microcurie (μ Ci) = 10^{-6} curie; nanocurie (nC) = 10^{-9} curie; picocurie (pCi) = 10^{-12} curie; femtocurie (fCi) = 10^{-15} curie.

Decay: The spontaneous radioactive transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. Every decay process has a definite half-life.

Decommissioning: The process of removing a facility or area from operation and decontaminating and/or disposing of it or placing it in a condition of standby with appropriate administrative and physical controls and safeguards.

Decontamination: The selective removal of radioactive material from a surface or from within another material.

Deep Geologic Formations: Geologic media beneath the continents and isolated from biologic species and phenomena. Distinguished from ice sheets and sea floor geologic media.

Defense Waste: Nuclear waste generated from government defense programs as opposed to commercial waste which is generated from civilian reactor programs.

Denitration: The chemical process of converting uranyl nitrates ($\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$) to uranium trioxide (UO_3).

Diluent: An organic compound used to dilute the solvent in a solvent extraction process. It is usually nonreacting.

Disposal: The planned release or placement of waste in a manner that does not include intent to recover.

Dose: The energy imparted to matter by ionizing radiation per unit mass or irradiated material at a specific location. The unit of absorbed dose is the rad.

Engineered Storage: The storage of radioactive wastes, usually in suitable sealed containers, into any of a variety of structures especially designed to protect them from water and weather and to help keep them from leaking to the biosphere by accident or sabotage.

Enriched Uranium: Uranium in which the percentage of the fissionable isotope uranium-235 has been increased above the 0.7% contained in natural uranium.

Extraction Column: A piece of process equipment which intimately mixes the organic phase and the aqueous phase in the solvent extraction process. Material is transported between the phases.

Extraction Cycle: The use of several extraction columns to separate a wanted product by solvent extraction from other constituents of the feed stream.

Federal Repository: A DOE-controlled facility to be used for the disposal of nuclear waste.

Fertile Material: A material, not itself fissionable by thermal neutrons, that can be converted into a fissile material by irradiation in a reactor. There are two basic fertile materials, uranium-238 and thorium-232. When these fertile materials capture neutrons, they are partially converted into fissile plutonium-239 and uranium-233, respectively.

Fines: Calcine dust having the consistency and flow properties of talcum powder.

Fissile Materials: While sometimes used as a synonym for fissionable material, this term has also acquired a more restricted meaning, namely, any material fissionable by neutrons of all energies, including (and especially) thermal (slow) neutrons as well as fast neutrons; for example, uranium-235 and plutonium-239.

Fission: The splitting of a heavy nucleus into two approximately equal parts (which are nuclei of lighter elements), accompanied by the release of a relatively large amount of energy and generally one or more neutrons. Fission can occur spontaneously, but usually is caused by nuclear absorption of gamma rays, neutrons, or other particles.

Fissionable Material: Commonly used as a synonym for fissile material. The meaning of this term also has been extended to include material that can be fissioned by fast neutrons only, such as uranium-238. Used in reactor operations to mean fuel.

Fission Products (F.P.): The nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclides formed by the fission fragments' radioactive decay.

Fluidized Bed: A bed of granular material expanded and agitated by an upflow of gas.

Fuel (nuclear, reactor): Fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor.

Fuel Assembly: An assembly of fuel elements.

Fuel Cycle: The complete series of steps involved in supplying fuel for nuclear power reactors. It includes mining, refining, the original fabrication of fuel elements, their use in a reactor, chemical processing to recover the fissionable material remaining in the spent fuel, re-enrichment of the fuel material, refabrication into new fuel elements, and management and disposal of radioactive waste.

Fuel Element: A tube, rod, or other form into which fissionable material is fabricated for use in a reactor.

Fuel Separation (fuel reprocessing): Processing of irradiated (spent) nuclear reactor fuel to recover useful materials as separate products, usually separation into plutonium, uranium, and fission products.

Gamma Rays (γ): High-energy, short-wavelength electromagnetic radiation emitted by a nucleus. Gamma radiation usually accompanies alpha and beta emissions and always accompanies fission.

Glass-ceramic: A fine-grained, crystalline material created by the controlled heating of a glass.

Half-life (T_{1/2}): The time in which half the atoms of a particular radioactive substance disintegrate to another nuclear form. Measured half-lives vary from millionths of a second to billions of years.

Half-life, Biological: The time required for a living organism to eliminate, by natural processes, half the amount of substance that has entered it.

Health Physics: The science concerned with recognition, evaluation, and control of health hazards from ionizing and nuclear radiations.

HEPA: High efficiency particulate air filter. A type of filter designed to remove 99.9% of particles down to 0.3 μ m in diameter from a flowing air stream.

High-Level Liquid Waste (HLLW): The aqueous waste resulting from the operation of the first-cycle extraction system (Purex Process), equivalent concentrated wastes from subsequent extraction cycles, or equivalent wastes from a process not using solvent extraction, in a facility for processing irradiated reactor fuels.

High-level Waste: (a) high-level liquid waste, or (b) the products from solidification of high-level liquid waste, or (c) irradiated fuel elements, if discarded without processing.

Immobilization: Treatment and/or emplacement of the wastes so as to impede their movement.

Interim Storage: The storage of wastes in a protected area while awaiting transfer to a deep geologic repository for extended storage or for disposal.

Ion Exchange: A chemical process involving the reversible interchange of various ions between a solution and a solid material, usually a resin. It is used to separate and purify chemicals, such as fission products, rare earths, in solution.

Ionizing Radiation: Any radiation displacing electrons from atoms or molecules, thereby producing ions. Examples: alpha, beta, and gamma radiation.

Isolation: A term encompassing both final storage and/or disposal in geologic formations.

Isotope: One of two or more atoms with the same atomic number (the same chemical element) but with different atomic weights. An equivalent statement is that the nuclei of isotopes have the same number of protons but different numbers of neutrons. Isotopes usually have very nearly the same chemical properties, but somewhat different physical properties.

Licensed Material: Source material, special nuclear material, or by-product material received, possessed, used, or transferred under a general or special license issued by the U.S. Nuclear Regulatory Commission or an Agreement State.

Long-lived Isotope: A radioactive nuclide that decays at such a slow rate that a quantity of it will exist for an extended period.

Long-lived Nuclides: Radioactive isotopes with half-lives greater than about 30 years. Most long-lived nuclides of interest to waste management have half-lives on the order of thousands to millions of years (Pu-239 - 24,400 years; Tc-99 - 2.1×10^5 years; I-12 - 1.6×10^7 years).

Man-rem: A unit in health physics to compare the effects of different amounts of radiation on groups of people. It is obtained by multiplying the average dose equivalent to a given organ or tissue (measured in rems) by the number of persons in that population.

Metal-Matrix: A process to fix calcine in a monolithic form by filling the void spaces with molten metal (either aluminum or lead).

Monolith: A dense, solid material incorporating nuclear waste.

Natural (Normal) Uranium: Uranium as found in nature. It is a mixture of the fertile uranium-238 isotope (99.3%), the fissionable uranium-235 isotope (0.7%), and a minute percentage of uranium-234.

Nonretrievable Solid Transuranic Waste: Any solid transuranic waste that contains, or is judged to contain, less than a specified quantity or concentration of transuranic elements (currently 10 nCi/g) and that may be disposed of in a nonretrievable manner, but still under conditions that will permit control in a disposal area and minimization of any migration of the transuranic elements. Included in this category are wastes contaminated with high beta-gamma emitters.

Normal Retrievable Waste: Wastes containing more than 10 nCi/g of transuranic isotopes and less than the recovery limit for that type of material as determined by the site.

Nuclide: A species of atom having a specific mass, atomic number, and nuclear energy state. These factors determine the other properties of the element, including its radioactivity.

Off-gas: Gas released by any process in fuel reprocessing or nuclear waste management.

Organic-Phase: The liquid phase in solvent extraction which consists of the organic solvent and the organic diluent.

Overpack: Secondary (or additional) external containment for packaged nuclear waste.

Partition: The chemical separation of actinides from other fission products.

pH: A measure of the hydrogen ion concentration in aqueous solutions. Acidic solutions have a pH from zero to 7. Basic solutions have a pH from 7 to 14.

Pit: An excavation for the disposal of miscellaneous aqueous and organic wastes, some of which may be contaminated with tritium, fission products, or activation products.

Plutonium (Pu): A heavy, radioactive, man-made, metallic element with atomic number 94. Its most important isotope is fissionable plutonium-239, produced by neutron irradiation of uranium-238. It is used for reactor fuel and in weapons.

Primary Wastes: As-generated forms and quantities of wastes.

Production Reactor: A nuclear reactor designed primarily for large-scale production of plutonium, tritium, and other radionuclides by neutron irradiation.

Process-Generated Wastes: Those wastes that are generated by, or are the result of, process operations in the enclosure defining the process lines and related equipment. In general, the TRU contamination levels will be significant.

Purex: A solvent extraction process in which uranium and plutonium are selectively separated from each other and from fission products by extraction from nitric acid solutions with tributylphosphate in a hydrocarbon diluent.

Rad (acronym for radiation absorbed dose): The basic unit of absorbed dose of ionizing radiation.

Radiation: The emission and propagation of energy through matter or space by means of electromagnetic disturbances, which display both wave-like and particle-like behavior; in this context the "particles" are known as photons. Also, the energy so propagated. The term has been extended to include streams of fast-moving particles (alpha and beta particles, neutrons, cosmic radiation, etc.). Nuclear radiation is that emitted from atomic nuclei in various nuclear reactions, including alpha, beta, and gamma radiation and neutrons.

Radioactive Contamination: Deposition of radioactive material in any place where it may harm persons, spoil experiments, or make products or equipment unsuitable or unsafe for some specific use. The presence of unwanted radioactive matter. Also radioactive material found on the walls of vessels in used-fuel processing plants, or radioactive material that has leaked into a reactor coolant. Often referred to only as contamination.

Radioactive Waste: Materials of no value consisting of, including, or contaminated with radioactive material in excess of the levels or concentrations permitted for unconditional release of such materials.

Radioactivity (often shortened to "activity"): The spontaneous decay or disintegration of an unstable atomic nucleus, usually accompanied by the emission of ionizing radiation. The word radioactivity is often used to refer to radioactive materials or radioactive nuclides, but this usage is not, strictly speaking, correct. Radioactivity is a process, not a substance.

Radioisotope: A radioactive isotope. An unstable isotope of an element that decays or disintegrates spontaneously, emitting radiation. More than 1,300 natural and artificial radioisotopes have been identified.

Radionuclide: An unstable nuclide of an element that decays or disintegrates spontaneously, emitting radiation.

Radwaste: See Radioactive Waste above.

Raffinate: The waste stream containing fission products from an extraction column.

Reactor: A device by means of which a fission chain reaction can be initiated, maintained, and controlled.

Recovery: The reprocessing of TRU contaminated materials for the purpose of reclaiming and reusing the transuranic elements which they contain.

Rem: A unit of measure for the dose of ionizing radiation that gives the same biological effect as 1 roentgen of x rays; 1 rem equals approximately 1 rad for X, gamma, or beta radiation.

Repository: A federally owned and operated facility for storage or disposal of specific types of radioactive waste from DOE sites and/or NRC licensees.

Resin: A synthetic organic polymer that can act as an ion exchanger.

Retrievable Solid Transuranic Waste: Any solid transuranic waste that contains, or is judged to contain, more than a specified quantity or concentration of transuranic elements (currently 10 nCi/g) and must, as a result, be placed into interim storage for eventual disposition.

Retrievable Storage: "Solid waste generated at DOE sites and containing significant U-233 or transuranium nuclide contamination shall be stored at DOE sites, segregated from other radioactively contaminated solid waste and with combustible and noncombustible transuranium-contaminated waste packaged separately. The packaging and storage conditions shall be such that the packages can be readily retrieved in an intact, contamination-free condition for 20 years. The packages shall be suitably labeled so the waste they contain can be identified by cross reference to permanent records." (DOE Manual 0511—Radioactive Waste Management)

Roentgen (abbreviation r): A unit of exposure to ionizing radiation. It is that amount of gamma or X rays required to produce ions carrying one (1) electrostatic unit of electrical charge (either positive or negative) in one (1) cubic centimeter of dry air under standard conditions. Named after Wilhelm Roentgen, German scientist who discovered X rays in 1895.

Room-Generated Wastes: Those wastes that are generated by, or are the result of, support operations in the room or area containing the transuranic processing facilities. These materials are usually of considerably lower contamination level than wastes generated in process lines.

Salt Cake: The remaining product of high-level liquid waste when evaporated to dryness without agitation.

Scrap: The feed materials for a TRU recovery process, that is, the transuranic-contaminated materials which are discarded from a process but which contain sufficient transuranic elements to warrant recovery.

Secondary Wastes: Forms and quantities of all wastes that result from applying waste treatment technologies to primary wastes.

Separations: Chemical processes used to separate nuclear products from by-products and from each other.

Short-lived Isotope: A radioactive nuclide that decays so rapidly that a given quantity is transformed into its daughter products within a short period (usually those with a half-life of days or less).

Short-lived Nuclides: Radioactive isotopes with half-lives no greater than about 30 years, e.g., Cs-137 and Sr-90.

Solidification: Conversion of radioactive waste to a dry, stable solid.

Solid Radioactive Waste: Radioactive waste material that is essentially dry but may contain sorbed radioactive fluid in sufficiently small amounts to be immobile when buried in dry soil.

Solvent Extraction: A process in which materials are selectively removed from an aqueous solution by contact with an immiscible organic solvent.

Special Nuclear Material (SNM): Plutonium, uranium-233, uranium-235, or uranium enriched to a higher percentage than normal of the -233 or -235 isotopes.

Storage: Retention of radioactive waste in some type of man-made engineered device (e. g., tank) in a manner permitting retrieval.

Storage Area: The area in which the storage facility is located (i.e., the site of the storage facility).

Storage Facility: The actual place of storage of wastes. This may be an engineered structure such as a building or vault or a protected area in any other type of facility.

Tank: A large metal container located above ground or underground for storage of liquid wastes.

Tank Farm: An installation of interconnected underground containers (tanks) for storage of high-level waste.

Transmutation: Conversion of a radioactive nucleus to another isotope by bombarding it with radiation or nuclear particles.

Transportation: Movement of materials between sites. Intrasite movement is not considered. Includes alternative methods for packaging, handling, and transport of waste materials. Concepts include all conventional methods of land and water transport required by the waste management system.

Transuranic Waste: Any waste material measured or assumed to contain more than a specified concentration (e.g., presently 10 nanocuries of trans-uranium activity per gram of waste) of transuranic elements.

Transuranium: Nuclides having an atomic number greater than that of uranium (i.e., greater than 92). The principal transuranic radionuclides of concern in radioactive waste management are tabulated below with their half-lives:

<u>Nuclide</u>	<u>Half-Life (Years)</u>	<u>Principal Decay Modes</u>
neptunium-237	2,140,000	alpha
plutonium-238	86	alpha, spontaneous fission
plutonium-239	24,390	alpha, spontaneous fission
plutonium-240	6,580	alpha, spontaneous fission
plutonium-242	379,000	alpha
americium-241	458	alpha
americium-243	7,950	alpha
curium-245	9,300	alpha
curium-246	5,500	alpha, spontaneous fission

The transuranium nuclide produced in largest amounts is plutonium-239; americium-241 is also produced in significant amounts. For further discussion see Appendix B.

Treatment: Operations intended to benefit safety or economy by changing the waste characteristics. Four basic treatment concepts are defined:

- Volume reduction
- Immobilization of radioactivity
- Change of composition
- Removal of radioactivity from the waste

Trench: A long and narrow excavation in the ground for solid waste. Unless qualifying descriptions are given, a trench is unlined, and its walls are unsupported. After the solid wastes are placed in position, the trench is filled to grade level with some of the removed soil.

Uranium: A radioactive element with the atomic number 92 and, as found in natural ores, an average atomic weight of approximately 238. The two principal natural isotopes are U-235 (0.7 percent of natural uranium), which is fissionable, and U-238 (99.3 percent of natural uranium), which is fertile. Natural uranium also includes a minute amount of U-234.

Vitrification: The process of forming an amorphous as opposed to crystalline structure—a glassing process.

Waste Container: The device or devices, including liners and closures, used in the packaging of wastes.

Waste Package: The final configuration of the waste in its container as it is ready to be placed into interim storage or final geologic disposal.

ABBREVIATIONS

AEC	United States Atomic Energy Commission
ALARA	As low as reasonably achievable
ANL	Argonne National Laboratory
BNL	Brookhaven National Laboratory
DOE	United States Department of Energy
DOT	United States Department of Transportation
DF	Decontamination factor
EPA	United States Environmental Protection Agency
ERDA	United States Energy Research and Development Administration
HEDL	Hanford Engineering Development Laboratory
HLLW	High-level liquid waste
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission of Radiological Protection
ILLW	Intermediate-level liquid waste
INEL	Idaho National Engineering Laboratory
LASL	Los Alamos Scientific Laboratory
LLNL	Lawrence Livermore National Laboratory
MTU	Metric ton uranium
NRC	United States Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
RFP	Rocky Flats Plant
SL	Sandia Laboratory
SRP	Savannah River Plant
TBP	Tributyl phosphate
WCF	Waste Calcining Facility

APPENDIX B

TRANSURANIUM RADIONUCLIDES^{2,28}

**AN OVERVIEW OF CHEMICAL, PHYSICAL
AND RADIOLOGICAL PROPERTIES**

TERA CORPORATION

INTRODUCTION

The transuranium radionuclides are those nuclides with atomic numbers 93 to 103, inclusive. Table B-1 gives the atomic numbers, names and chemical symbols of this group of elements. Table B-2 presents various physical characteristics of alpha-emitting TRU isotopes with half-lives greater than 100 days.

No known nuclides of the transuranium elements are stable. In general, the higher the atomic number of the element, the shorter the half-life.

Plutonium is the best known of the transuranium radionuclides and it is common to compare the properties of this group of radionuclides to plutonium. This group of radionuclides is highly radioactive and has a high specific activity, i.e. there are many curies, Ci, of radioactivity for each gram of nuclide.

The relatively high energy release per each nuclear decay and the general tendency of these nuclides to concentrate in bone tissue leads to a highly toxic effect when these nuclides are present within the human body. On a Ci or radioactivity basis, the transuranium radionuclides are not greatly different from plutonium in their toxicity. On a mass or gram basis, however, some of these radionuclides are a factor of 10^4 times more toxic than plutonium. Their extremely high toxicity per unit mass requires that extraordinary precautions and care need to be taken when working with these materials. These radionuclides require a skill and a specially designed complement of equipment for handling that far exceeds that required for the more common fission product radionuclides. It is necessary to plan diligently with expert knowledge to design and operate safe transuranium processing and handling facilities.

Experience in working with fission products or uranium is helpful in planning transuranium radionuclide work and activities, but such experience cannot be translated to working with these high specific activity materials. The contamination control requirements are 10^3 to 10^6 times more rigid for the transuranium radionuclides than for the typical fission product radionuclides.

TABLE B-1
TRANSURANIUM ELEMENTS

<u>Atomic Number</u>	<u>Element Name</u>	<u>Chemical Symbol</u>
93	Neptunium	Np
94	Plutonium	Pu
95	Americium	Am
96	Curium	Cm
97	Berkelium	Bk
98	Californium	Cf
99	Einsteinium	Es
100	Fermium	Fm
101	Mendelevium	Md
102	Nobelium	No
103	Lawrencium	Lw

TABLE B-2
ALPHA-EMITTING TRU ISOTOPES
WITH HALF-LIFE GREATER THAN 100 DAYS²⁸

Isotope	$T_{1/2}$ (yr)	$T(SF)_{1/2}$ (yr)	Thermal Neutron Cross Sections (barn)	Specific Activity (g/Ci)	(SF) Activity (fission/ sec-g)	ν (SF)	Parent	Daughter
$^{233}_{92}\text{U}$	1.62×10^5		$\sigma_c = 49$ $\sigma_f = 524$	1.05×10^2	~ 0		^{233}Pa ^{233}Np	^{229}Th
$^{237}_{93}\text{Np}$	2.14×10^6	10^{18}	$\sigma_c = 170$ $\sigma_f = 0.019$	1.42×10^3	5.6×10^{-5}		^{237}U ^{241}Am ^{237}Pu ^{240}Cm ^{236}Np	^{233}Pa
$^{236}_{92}\text{Pu}$	2.85×10^7	3.5×10^9	$\sigma_f = 170$	1.88×10^{-3}	1.6×10^4			^{232}Th
$^{236}_{94}\text{Pu}$	8.6×10^7	4.9×10^{10}	$\sigma_c = 500$ $\sigma_f = 16.8$	5.75×10^{-2}	1.13×10^3	2.26	^{236}Np ^{238}Am ^{242}Cm	^{232}Th
$^{239}_{94}\text{Pu}$	2.4×10^4	5.5×10^{15}	$\sigma_c = 274$ $\sigma_f = 741$	1.63×10^1	1.00×10^{-2}	2.2	^{239}Np ^{243}Cm ^{239}Am	^{235}U
$^{240}_{94}\text{Pu}$	6.56×10^3	1.3×10^{17}	$\sigma_c = 286$ $\sigma_f = 0.08$	1.4×10^0	4.1×10^2	2.1	^{240}Cm ^{244}Pu ^{240}Am	^{236}U
$^{242}_{94}\text{Pu}$	3.79×10^5	7.1×10^{10}	$\sigma_c = 19$ $\sigma_f = 0.2$	2.57×10^2	1.7×10^2	2.16	^{242}Cm ^{246}Am	^{238}U
$^{244}_{94}\text{Pu}$	7.6×10^7	2.5×10^{10}	$\sigma_c = 1.8$	5.2×10^{-4}	2.7×10^3		^{248}Cm	^{240}Pu F.P.
^{243}Am	4.58×10^2	2×10^{14}	$\sigma_c = 700$ ^{242}Am $\sigma_c = 100$ ^{242m}Am $\sigma_f = 3.0$	1.06×10^{-3}	2.1×10^{-1}	2.3	^{243}Pu ^{243}Bk ^{247}Cm	^{239}Np
$^{242m}_{95}\text{Am}$	1.52×10^2		$\sigma_c = 2000$ $\sigma_f = 6000$	1.03×10^{-1}				^{238}U ^{234}Th
$^{243}_{95}\text{Am}$	7.95×10^3		$\sigma_c = 74$ $\sigma_f = 0.07$	5.40×10^0			^{243}Pu ^{247}Bk	^{239}Np
$^{242}_{96}\text{Cm}$	4.45×10^{-1}	7.2×10^6	$\sigma_c = 20$ $\sigma_f < 5$	3.01×10^{-4}	7.58×10^6		^{242}Am ^{246}Cf	^{238}Pu F.P.
$^{243}_{96}\text{Cm}$	3.2×10^1		$\sigma_c = 250$ $\sigma_f = 660$	2.2×10^{-2}			^{243}Bk	^{239}Pu

TABLE B-2

(CONT.)

Isotope	$T_{1/2}$ (yr)	$T(SF)_{1/2}$ (yr)	Thermal Neutron Cross Sections (barn)	Specific Activity (g/Ci)	(SF) Activity (fission/ sec-g)	ν (SF)	Parent Daughter
$^{244}_{96}\text{Cm}$	1.76×10^1	1.11×10^7	$\sigma_c = 15$	1.2×10^{-2}	4.34×10^6		$^{240}_{96}\text{Pu}$ $^{244}_{96}\text{Am}$ F.P. $^{244}_{96}\text{Bk}$
$^{245}_{96}\text{Cm}$	9.3×10^3		$\sigma_c = 200$ $\sigma_f = 1920$	6.4×10^0			$^{241}_{96}\text{Pu}$ $^{245}_{96}\text{Cf}$ $^{245}_{96}\text{Am}$ $^{245}_{96}\text{Bk}$
$^{246}_{96}\text{Cm}$	5.5×10^3	1.7×10^7	$\sigma_c = 15$	3.8×10^0	3.2×10^6		$^{242}_{96}\text{Pu}$ $^{246}_{96}\text{Cf}$ $^{246}_{96}\text{Am}$ F.P. $^{246}_{96}\text{Bk}$
$^{247}_{96}\text{Cm}$	1.6×10^7		$\sigma_c = 180$	1.1×10^{-2}			$^{243}_{96}\text{Pu}$ $^{247}_{96}\text{Cf}$
$^{248}_{96}\text{Cm}$	4.7×10^5	4.6×10^6	$\sigma_c = 6$	3.27×10^2	1.16×10^7		$^{244}_{96}\text{Pu}$ $^{252}_{96}\text{Cf}$ $^{248}_{96}\text{Bk}$ F.P.
$^{250}_{96}\text{Cm}$		1.7×10^4			3.1×10^9		$^{242}_{96}\text{Pu}$ $^{250}_{96}\text{Cf}$ F.P.
$^{247}_{97}\text{Bk}$	1.4×10^3			9.66×10^{-1}			$^{243}_{97}\text{Bk}$ $^{247}_{97}\text{Cf}$
$^{248}_{98}\text{Cf}$	9.59×10^{-2}	$\geq 1.5 \times 10^4$		6.64×10^{-4}	$\geq 3.6 \times 10^9$		$^{244}_{98}\text{Cf}$ $^{248}_{98}\text{Bk}$ F.P. $^{248}_{98}\text{Bk}$
$^{249}_{98}\text{Cf}$	3.60×10^2	1.5×10^9	$\sigma_c = 270$ $\sigma_f = 1735$	2.51×10^{-1}	2.5×10^4		$^{245}_{98}\text{Cf}$ $^{249}_{98}\text{Es}$ F.P.
$^{250}_{98}\text{Cf}$	1.32×10^1	1.7×10^4	$\sigma_c = 1500$ $\sigma_f = 350$	9.24×10^{-3}	3.1×10^9		$^{246}_{98}\text{Cf}$ $^{250}_{98}\text{Bk}$ F.P. $^{250}_{98}\text{Es}$
$^{251}_{98}\text{Cf}$	8.8×10^2		$\sigma_c = 3000$ $\sigma_f = 3000$	4.6×10^{-1}			$^{247}_{98}\text{Cf}$ $^{251}_{98}\text{Es}$
$^{252}_{98}\text{Cf}$	2.65×10^0	8.5×10^1	$\sigma_c = 30$	1.86×10^{-3}	6.2×10^{11}	3.8	$^{248}_{98}\text{Cf}$ $^{252}_{98}\text{Fm}$ F.P. $^{252}_{98}\text{Es}$
$^{252}_{99}\text{Es}$	3.84×10^{-1}			12.7×10^{-6}			$^{248}_{99}\text{Es}$ $^{252}_{99}\text{Md}$
$^{254}_{99}\text{Es}$	7.56×10^{-1}	7×10^5	$\sigma_c < 40$	5.4×10^{-4}	7.4×10^7		$^{250}_{99}\text{Es}$ F.P.

^a Has no parent isotope but is produced by a (n,γ) reaction.

The transuranium elements are normally produced by neutron irradiation of uranium or higher atomic number elements in a nuclear reactor. The transuranium radionuclides are also produced in thermonuclear explosions and by accelerators. Reactor irradiation is usually the preferred method of production when more than trace quantities of a nuclide are desired.

The major differences between the transuranium radionuclides and uranium are the greater specific activity, shorter spontaneous fission half-lives, and the larger number of X-ray and gamma-ray photons emitted per disintegration of the transuranium radionuclides. In handling or processing these nuclides one is confronted not only with dose rates that may require extensive shielding, but also with the problem of ensuring that radionuclides do not escape into the work environment and enter the body by inhalation, ingestion or other methods. Table B-3 presents the maximum permissible concentration (MPC) values for the major TRU isotopes.

The high specific activities require an exceptional degree of diligence in all manipulations to assure control and absolute containment. The loss of control over even the slightest amount, by any normal standards of measurement, of a transuranium radionuclide can lead to serious hazards and timely and costly decontamination requirements. The potential for difficulties in handling and working with transuranium radionuclides depends upon the type of operation to be performed as well as the physical properties of the particular nuclide being used.

PHYSICAL PROPERTIES

Modes of Decay

Most transuranium radionuclides emit alpha particles. Many also emit neutrons and some emit beta particles. Most particle emissions are accompanied by X-ray or gamma-ray emissions of relatively low energy, usually a few tens of kilovolts.

TABLE B-3
CONCENTRATIONS IN AIR AND WATER ABOVE NATURAL BACKGROUND

TRU ELEMENT	ISOTOPE ¹		RESTRICTED AREAS ²		UNRESTRICTED AREAS ³	
			AIR ($\mu\text{Ci/ml}$)	WATER ($\mu\text{Ci/ml}$)	AIR ($\mu\text{Ci/ml}$)	WATER ($\mu\text{Ci/ml}$)
Americium	Am 241	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Am 242m	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	3×10^{-10}	3×10^{-3}	9×10^{-12}	9×10^{-5}
	Am 242	S	4×10^{-8}	4×10^{-3}	1×10^{-9}	1×10^{-4}
		I	5×10^{-8}	4×10^{-3}	1×10^{-9}	1×10^{-4}
	Am 243	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Am 244	S	4×10^{-6}	1×10^{-1}	1×10^{-7}	5×10^{-3}
		I	2×10^{-5}	1×10^{-1}	8×10^{-7}	5×10^{-3}
Berkelium	Bk 249	S	9×10^{-10}	2×10^{-2}	3×10^{-11}	6×10^{-4}
		I	1×10^{-7}	2×10^{-2}	4×10^{-9}	6×10^{-4}
	Bk 250	S	1×10^{-7}	6×10^{-3}	5×10^{-9}	2×10^{-4}
		I	1×10^{-6}	6×10^{-3}	4×10^{-8}	2×10^{-4}
Californium	Cf 249	S	2×10^{-12}	1×10^{-4}	5×10^{-14}	4×10^{-6}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
	Cf 250	S	5×10^{-12}	4×10^{-4}	2×10^{-13}	1×10^{-5}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	3×10^{-5}
	Cf 251	S	2×10^{-12}	1×10^{-4}	6×10^{-14}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}

TABLE B-3

(CONT.)

		RESTRICTED AREAS ²		UNRESTRICTED AREAS ³		
TRU ELEMENT	ISOTOPE ¹	AIR (μ Ci/ml)	WATER (μ Ci/ml)	AIR (μ Ci/ml)	WATER (μ Ci/ml)	
Curium	Cf 252	S	6×10^{-12}	2×10^{-4}	2×10^{-13}	7×10^{-6}
		I	3×10^{-11}	2×10^{-4}	1×10^{-12}	7×10^{-6}
	Cf 253	S	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
		I	8×10^{-10}	4×10^{-3}	3×10^{-11}	1×10^{-4}
	Cf 254	S	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
		I	5×10^{-12}	4×10^{-6}	2×10^{-13}	1×10^{-7}
	Cm 242	S	1×10^{-10}	7×10^{-4}	4×10^{-12}	2×10^{-5}
		I	2×10^{-10}	7×10^{-4}	6×10^{-12}	2×10^{-5}
	Cm 243	S	6×10^{-12}	1×10^{-4}	2×10^{-13}	5×10^{-6}
		I	1×10^{-10}	7×10^{-4}	3×10^{-12}	2×10^{-5}
	Cm 244	S	9×10^{-12}	2×10^{-4}	3×10^{-13}	7×10^{-6}
		I	1×10^{-10}	8×10^{-4}	3×10^{-12}	3×10^{-5}
	Cm 245	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Cm 246	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	8×10^{-4}	4×10^{-12}	3×10^{-5}
	Cm 247	S	5×10^{-12}	1×10^{-4}	2×10^{-13}	4×10^{-6}
		I	1×10^{-10}	6×10^{-4}	4×10^{-12}	2×10^{-5}
	Cm 248	S	6×10^{-13}	1×10^{-5}	2×10^{-14}	4×10^{-7}
		I	1×10^{-11}	4×10^{-5}	4×10^{-13}	1×10^{-6}
	Cm 249	S	1×10^{-5}	6×10^{-2}	4×10^{-7}	2×10^{-3}
		I	1×10^{-5}	6×10^{-2}	4×10^{-7}	2×10^{-3}

TABLE B-3

(CONT.)

TRUE ELEMENT	ISOTOPE ¹		RESTRICTED AREAS ²		UNRESTRICTED AREAS ³	
			AIR ($\mu\text{Ci/ml}$)	WATER ($\mu\text{Ci/ml}$)	AIR ($\mu\text{Ci/ml}$)	WATER ($\mu\text{Ci/ml}$)
Einsteinium	Es 253	S	8×10^{-10}	7×10^{-4}	3×10^{-11}	2×10^{-5}
		I	6×10^{-10}	7×10^{-4}	2×10^{-11}	2×10^{-5}
	Es 254m	S	5×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
		I	6×10^{-9}	5×10^{-4}	2×10^{-10}	2×10^{-5}
	Es 254	S	2×10^{-11}	4×10^{-4}	6×10^{-13}	1×10^{-5}
		I	1×10^{-10}	4×10^{-4}	4×10^{-12}	1×10^{-5}
	Es 255	S	5×10^{-10}	8×10^{-4}	2×10^{-11}	3×10^{-5}
		I	4×10^{-10}	8×10^{-4}	1×10^{-11}	3×10^{-5}
Fermium	Fm 254	S	6×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
		I	7×10^{-8}	4×10^{-3}	2×10^{-9}	1×10^{-4}
	Fm 255	S	2×10^{-8}	1×10^{-3}	6×10^{-10}	3×10^{-5}
		I	1×10^{-8}	1×10^{-3}	4×10^{-10}	3×10^{-5}
	Fm 256	S	3×10^{-9}	3×10^{-5}	1×10^{-10}	9×10^{-7}
		I	2×10^{-9}	3×10^{-5}	6×10^{-11}	9×10^{-7}
Neptunium	Np 237	S	4×10^{-12}	9×10^{-5}	1×10^{-13}	3×10^{-6}
		I	1×10^{-10}	9×10^{-4}	4×10^{-12}	3×10^{-5}
	Np 239	S	8×10^{-7}	4×10^{-3}	3×10^{-8}	1×10^{-4}
		I	7×10^{-7}	4×10^{-3}	2×10^{-8}	1×10^{-4}

TABLE B-3

(CONT.)

TRUE ELEMENT	ISOTOPE ¹	RESTRICTED AREAS ²		UNRESTRICTED AREAS ³	
		AIR ($\mu\text{Ci/ml}$)	WATER ($\mu\text{Ci/ml}$)	AIR ($\mu\text{Ci/ml}$)	WATER ($\mu\text{Ci/ml}$)
Plutonium	Pu 238	S	2×10^{-12}	7×10^{-14}	5×10^{-6}
		I	3×10^{-11}	1×10^{-12}	3×10^{-5}
	Pu 239	S	2×10^{-12}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	1×10^{-12}	3×10^{-5}
	Pu 240	S	2×10^{-12}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	1×10^{-12}	3×10^{-5}
	Pu 241	S	9×10^{-11}	3×10^{-12}	2×10^{-4}
		I	4×10^{-8}	1×10^{-9}	1×10^{-3}
	Pu 242	S	2×10^{-12}	6×10^{-14}	5×10^{-6}
		I	4×10^{-11}	1×10^{-12}	3×10^{-5}
	Pu 243	S	2×10^{-6}	6×10^{-8}	3×10^{-4}
		I	2×10^{-6}	8×10^{-8}	3×10^{-4}
	Pu 244	S	2×10^{-12}	6×10^{-14}	4×10^{-6}
		I	3×10^{-11}	1×10^{-12}	1×10^{-5}

¹ S = Soluble; I = Insoluble

² Restricted area is discussed in 10CFR Part 20.103

³ Unrestricted area is discussed in 10CFR Part 20.105

Of the 46 known isotopes of elements 93 (Np) through 99 (Es) with half-lives greater than one day, 30 are alpha emitters, nine decay by beta emission, five capture orbital electrons, and two undergo spontaneous fission as the primary modes of decay. Several of these radionuclides, however, are parents of decay chains whose various decay products have half-lives which cause a complicated pattern of growth and decay of daughter products after the initial preparation of an isolated and essentially pure radionuclide. The precursors from irradiation may also exist in a given sample, as a result of incomplete separation, and add to the complex decay chain.

If the half-life of the "parent" transuranic radionuclide is much longer than that of the "daughter" (e.g. Am-243-Np-239, Cm-245-Pu-241, or Es-254-Bk-250), the activity of the material handled will approach the state of secular equilibrium in which the number of daughter atoms formed per unit time is equal to the number which decay. The rate of approach to this steady state is governed by the half-life of the daughter as $(1 - e^{-\lambda t})$. No equilibrium exists if the daughter is longer lived than the parent (e.g. Am-241-Np-237, Cm-244-Pu-240 or Cf-252-Cm-248). These two conditions--secular and no equilibrium--prevail; transient equilibrium, which requires the parent and daughter to have approximately equal half-lives, seldom occurs in the transuranium decay schemes.

Specific Activity

The relatively short half-lives of many of these radionuclides leads to a high specific activity, i.e. a large number of curies per gram of nuclide. The half-life, specific activity and an activity comparison normalized to Pu-239 are shown in Table B-4.

When one recalls that Pu-239 has a specific activity some 10^5 times that of natural uranium, the rapid decay rates for the transuranium radionuclides are clear. The radius of spherical particles of the more common transuranium radionuclides that would decay at the rate of 1000 d/m--a contamination level at which it is necessary to consider the need for decontamination, ranges from about 10^{-4} to about 10^{-5} cm. It is necessary to consider the degree of control

TABLE B-4
RADIOACTIVE DECAY
AND SPONTANEOUS FISSION HALF-LIVES
OF SELECTED NUCLIDES

Isotope	Radioactive half-life (yr)	Specific activity (Ci/g)	Normalized to Pu-239	Spontaneous fission half-life (yr)	Calculated neutron emission rate (n/sec/g)
Pu-239	24,400	0.062	1	5.5×10^{15}	1.03
Am-243	7,950	0.185	3	--	--
Am-241	458	3.24	52	2×10^{14}	0.06
Pu-238	86.4	17.6	280	4.9×10^{10}	3.4×10^3
Cm-244	17.6	83.3	1,300	1.3×10^7	1.2×10^7
Pu-241*	13.2	113	1,800	--	--
Cf-252	2.646	536	8,600	85	2.3×10^{12}
Bk-249*	0.86	1,680	27,000	6×10^8	2.7×10^5
Es-254	0.756	1,870	30,000	7×10^5	2.9×10^6

*Beta emitters; all others decay by alpha emission.

required to avoid the loss of control of even one particle of these small sizes and the containment problems for handling these radionuclides.

The high specific activities also lead to higher dose rates for small quantities of materials if one is measuring quantity in units of weight. For all of these nuclides it is necessary to be alert to the potential for X-ray and gamma-ray radiations as well as neutron and beta radiations, even though the primary mode of decay may be by alpha particle emission.

Spontaneous Fission

Spontaneous fission of transuranium radionuclides results in the emission of many neutrons in addition to the build-up of fission products that emit both beta and gamma rays. Some of the transuranium radionuclides have relatively short spontaneous fission half-lives corresponding to high spontaneous fission decay rates. In fact, the very high spontaneous fission rate of Cf-252 has provided a most useful application of this radionuclide as a neutron source. It has unique potentials because it provides a small source of an intense neutron flux. Table B-2 summarizes the spontaneous fission half-lives and neutron emission rates for some of these radionuclides.

CHEMICAL PROPERTIES

The chemical reactions of the transuranium radionuclides are similar to and may be expected to generally parallel the reactions of plutonium; there are, of course, many specific exceptions to their plutonium-like chemical behavior. The chemical reactions of plutonium have been extensively studied and several good reviews and handbooks summarizing these properties are readily available. For example, the two volume Plutonium Handbook edited by Wick details the chemical as well as physical properties of plutonium.

The transuranium radionuclides are metals, although with the exception of plutonium, americium and neptunium, they are seldom available in quantities sufficient for visual recognition as metals. These radionuclides, as finely divided

metals, are pyrophoric—reacting rapidly with oxygen to form oxides. They are best stored and handled in inert atmospheres until reacted to form some stable compound. In many of their chemical compounds they may be in a powder form and easily dispersed. The minute particles that may break or be rubbed off of even large crystal forms lead to challenging handling and containment programs if rigid control over these materials is to be established and maintained.

The high specific activity of the transuranium radionuclides leads to handling difficulties other than contamination control. Self-heating, radiolytic decomposition actions in aqueous solution, and spontaneous ignition when exposed to air (oxygen) constitute the bulk of these problems. A kilogram of plutonium metal is warm to the touch, emitting ~ 2 watts of thermal energy, while Cm-244 and Cm-242, potential thermoelectric power sources, emit approximately 3 and 120 watts/g of thermal energy, respectively. In most applications only Pu-238, Cm-242, Cm-244 and Cf-252 produce sufficient heat to be of concern. Table B-5 summarizes the decay heat production of some transuranium radionuclides.

The chemical form of a transuranium radionuclide can change the magnitude and nature of the radiation hazards encountered. The high energy alpha emissions can result in an alpha particle nuclear reaction which can result in significant neutron emissions (e.g. beryllium, oxygen, chlorine, fluorine and nitrogen chemical reactions involving transuranium radionuclides may transform a safely shielded facility designed for alpha and low-energy photon radiations into a poorly shielded facility where neutron radiation becomes of primary concern).

CRITICALITY CONSIDERATIONS

The potential for a self-sustaining nuclear chain reaction of criticality accident is currently not as significant for the transplutonium elements as for Pu-239 or U-235 because of the limited (generally less than 100 g) amounts of these materials available at any given facility. If quantities of any transuranium radionuclide greater than 300 grams are available, a careful and detailed review of criticality control provision is mandatory. Only a few transplutonic nuclides appear to be capable of supporting a chain reaction with thermal neutrons; the

TABLE B-5
DECAY HEAT PRODUCTION

<u>Nuclide</u>	<u>Decay Heat, Watts/g</u>
Pu-238	0.57
Pu-239	0.0019
Pu-240	0.014
Pu-241	0.00011
Am-241	0.11
Am-243	0.0061
Cm-242	120
Cm-244	2.8
Bk-249	0.4
Cf-252	39

remainder are fissionable by fast neutrons only (as metals or alloys). The fissile nuclides in solution present the most serious criticality control problems since the amount of material in solution is difficult to inventory and the critical mass is usually lower for solutions than for metals. Further, many solvents such as H_2O are good moderators.

Table B-6 compares the neutron cross-sections and the number of neutrons emitted per fission event of several transplutonic nuclides with those of the more familiar fissile materials, U-233, U-235, and Pu-239.

TABLE B-6
NEUTRON CRITICALITY PARAMETERS

<u>Nuclide</u>	<u>σ_c (barn)</u>	<u>σ_f (barn)</u>	<u>neutrons/ fission</u>	<u>Minimum critical mass* (g)</u>
U-233	49	524	2.50	590
U-235	99	579	2.43	820
Pu-239	254	754	2.89	510
Am-242m	5500	6390	~3.0	1000
Cm-243	250	590	~3.2	500-1000
Cm-245	335	1900	~3.2	500
Cm-247	180	1026	~3.3	500
Cf-249	270	630	~3.7	500-1000
Cf-251	1100	3000	~3.7	500-1000
Es-254	40	2000	~3.9	500

* Minimum critical mass for fully reflected solutions.

APPENDIX C

BITUMINIZATION^{74,83,84}

TERA CORPORATION

1.0 PROPERTIES OF BITUMEN⁷⁴

The term bitumen covers mixtures of high molecular weight related in their nature to the aliphatic and aromatic hydrocarbons. Bitumen can be separated by a solvent into two components: the asphaltenes, which precipitate out as a black tar, and the malthenes, which remain in the solution in the form of a dark viscous oil. The properties of the fractions obtained depend only on the solvent chosen (usually this is normal heptane). There is no sharp boundary between the two phases. The malthenes have the properties of a viscous liquid. The presence of asphaltenes in bitumens gives them colloid properties. Asphaltenes show a tendency, in combination with the aromatic fraction of the malthenes, to form complex micelles suspended in the malthene phase. If there are sufficient aromatic substances to saturate the absorption capacities of the asphaltenes, the micelles are mobile in the malthene phase and the colloidal solution is in the sol state. If there are insufficient aromatic substances, the micelles are attracted to each other and form a network, giving the bitumen elastic properties (the solution is then in the gel state).

Mechanically, bitumens behave either as elastic solids or as viscous liquids, and there are also intermediate states.

Bitumens can be divided into the following categories according to method of production:

- Bitumens obtained by straight-run distillation--the residue after distillation of heavy petroleum

Softening point	from 34 to 65°C
Penetration	from 22 to 2 mm
Density at 25°C	from 1.00 to 1.1 g/cm ³
Loss of weight on heating	< 2%
Flash point	>230-250°C
Elasticity at 25°C	from <100 to >25 mm

- Pitches - Bitumens liquefied by addition of a solvent; characterized by a viscosity S.T.V. at 25°C

Density	from 0.92 to 1.04 g/cm ³
Kinematic viscosity at 25°C	10-15 s (for liquid bitumens)
(opening diameter 10 mm)	400-600 s (for viscous bitumens)

- Oxidized bitumens - Highly colloidal bitumen products formed by blowing air through certain petroleums. Because of their specific properties they are used in industry in static conditions. Temperature fluctuations usually have little effect on them.

Penetration at 25°C	
ranges from	0.7 to 4.5 mm
Density at 25°C	1.02 - 1.04
Softening point	70 - 140°C

- Cracked bitumens - Obtained by pyrogenic breakdown of heavy molecules. Temperature fluctuations have a considerable effect on these. They are used mainly in cases where good flow at high temperatures and subsequent rapid hardening on cooling is required.

Softening point	77 - 85°C
Penetration at 25°C	<0.5 mm
Flow temperature	150°C
Minimum spray temperature	200°C

- Bitumen emulsions - Formed by the emulsification of bitumens in soapy water. There are two types of emulsions: anionic (emulsifier: alkaline soap) and cationic (emulsifier: amine salt). Bitumen emulsions can be used without previous heating. On contact with a prepared surface, the emulsion spreads over it, the water evaporates and a rigid coating is left.

A particular interest was taken in bitumen for the incorporation of solid radioactive contaminated particles, as bitumen may be compared to a strong cement, readily adhesive, highly waterproof and durable. It is a plastic substance which imparts controllable flexibility to mixtures of mineral aggregates with which it usually combines. Moreover, bitumen resists very efficiently

the action of most acids, alkalis and salts. Although a solid or a semi-solid substance at normal ambient temperature, bitumen can be liquefied without difficulty by heat application.

The viscosity of bitumen decreases when the temperature applied increases. The relationship between temperature and viscosity appears to be not identical when considering different sources of types and grades of bitumen. The application temperature is normally specified for various uses. However, because of viscosity variations, the specification of temperature alone does not suffice for a most efficient use of bitumen.

The most suitable application viscosity will be determined by such factors as:

- Type of mixer
- Characteristics of the radioactive concentrates (solution, sludge, etc.)
- Storage conditions, etc.

Two shipments of any given bitumen are likely to fluctuate widely in their physical properties and composition, even when processed from an identical source. Each bituminous substance has a predetermined consistency and some fixed well-defined physical properties over which one has no control.

2.0 TYPES OF WASTE SUITABLE FOR INCORPORATION INTO BITUMEN

Bitumen has been extensively studied in Europe and numerous references are available (see References 77, 83, and 84).

Bituminization has handled a variety of TRU waste including:

- Chemical sludges from various chemical precipitation techniques.
- Ion exchange material including synthetic organic, synthetic inorganic and naturally occurring exchange materials.
- Regenerants and concentrated salt solutions including boric acid and aqueous raffinate from the second Purex solvent extractor cycle (oxidants such as nitrates and nitrites including quantities of sulfates, chlorides, fluorides and phosphates).
- Organic solvents with the addition of thickeners (organics include tributyl phosphate, n-dodecane, carbon tetrachloride, tetrachloroethylene and mixtures of two or more).
- Incinerator ash consisting of silicates, oxides, carbonates and phosphates of sodium, calcium, aluminum, magnesium, iron, etc.
- Plastic wastes (if shredded).

3.0 LIMITATIONS⁸⁴ AND CHARACTERISTICS²⁹

The radioactivity level of the mixed solutions might limit their bituminization, especially in view of a safe long-term storage of the obtained bitumen blocks. The safe limits regarding the thermal and radiation stability of industrial scale bitumen waste blocks as function of their specific activity and composition are still subject to further investigations and discussions. It is obvious that the degree of hazards involved depends mainly on the storage conditions, composition, amounts and dimensions of the highly active bitumen blocks.

Tables C-1 through C-4 present certain experimental data which provide the reader with a better understanding of bitumen features.

Leachability of bitumen products is dependent upon the type of waste incorporated and its content in the waste form. A typical asphalt containing 20 percent soluble salts has a leachability of approximately 5×10^{-4} g/(cm²-day); those containing more soluble salts are more leachable. Leach rates for alkaline and alkaline earth elements can be as low as 3×10^{-7} g/(cm²-day), although initial leach rates as high as 10^{-2} g/(cm²-day) have been reported.²⁰ Leach rates for rare-earth and actinide elements of 1×10^{-7} , 8×10^{-7} g/(cm²-day) are reported. There appears to be little difference in the leach rate for the incorporation of a given waste in various types of bitumen. Experiments indicate slightly lower leach rates using straight distillation bitumen. The amount of residual water remaining in the bitumen waste form affects the leach rate. In water, certain bitumen products have shown large volume increases as a result of swelling; this would result in increasing leach rates.²⁰

The flash point of bitumens is generally $\sim 280^{\circ}\text{C}$. The thermal stability of bitumen is related to its volatiles content. Bitumen burns more vigorously if appreciable quantities of oxidizing compounds which decompose at the temperatures encountered during combustion are present, although the temperature at which burning begins may not be much, if any lower, than for bitumens alone. The presence of some oxidizing compounds may lower the spontaneous ignition temperature dangerously. No work has proved conclusively that bitumen

TABLE C-1
RESULTS OF LEACHING TEST OF BITUMEN MIXES IN WATER

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No.	Asphalt		Sludge		MEN's ratio w/o dry solid	Tissue		Test Piece				Leached activity μCi/ml	Leaching rate g/cu dry		Vol ml	St-ric dynamic
	Make	Weight g-dry	Make	Weight g-dry		NaClide	VC/g-dry solid VC/plate	Weight g	Vol ml	Sp. Wt.	Surface cm ²					
1	Emulsified ^{HA}	200	R-Cu	86	30	¹³⁷ Cs	6.9 227	110	89	1.29	123	25	8.6×10^{-7}	Sea	900	S
2	Emulsified ^{HA}	100	R-Cu	66	40	¹³⁷ Cs	6.9 300	127	94	1.35	140	55	1.29×10^{-6}	Sea	900	S
3	Emulsified ^{HA}	100	R-Cu	100	50	¹³⁷ Cs	6.9 480	144	94	1.52	137	80	1.27×10^{-6}	Sea	900	S
4	Straight	100	R-Cu	43	30	¹³⁷ Cs	6.9 161	70	53	1.32	78	30	1.8×10^{-6}	Sea	1000	S
5	Straight	100	R-Cu	67	40	¹³⁷ Cs	6.9 187	66	52	1.20	73	270	1.35×10^{-5}	Sea	1000	S
6	Straight	100	R-Cu	67	40	¹³⁷ Cs	10.3 268	65	50	1.30	76	108	3.47×10^{-6}	Sea	1000	S
7	Straight	200	R-Cu	123	40	¹³⁷ Cs	3.5 87	62	47	1.32	77	120	1.20×10^{-5}	Sea	1000	S
8	Straight	100	R-Cu	100	50	¹³⁷ Cs	6.9 268	72	56	1.29	77	215	8.0×10^{-6}	Sea	1000	S
9	Straight	100	Pe	67	40	¹³⁷ Cs	10.3 457	110	78	1.42	88	1500	3.76×10^{-5}	Sea	900	S
10	Straight	250	R-Cu	64	30	¹³⁷ Cs	6.9 169	72	56	1.28	81	25	1.43×10^{-7}	dist	1000	S
11	Straight	100	R-Cu	67	40	¹³⁷ Cs	10.3 280	68	54	1.26	78	70	2.16×10^{-6}	dist	1000	S
12	Straight	100	R-Cu	100	50	¹³⁷ Cs	6.9 276	80	56	1.43	60	205	1.46×10^{-6}	dist	1000	S
13	Straight	200	R-Cu	133	40	¹³⁷ Cs	10.3 306	123	92	1.34	730	170	5.75×10^{-7}	dist	1000	D
14	Straight	100	R-Cu	43	30	¹³⁷ Cs	3.5 127	116	93	1.25	220	30	2.70×10^{-7}	dist	1000	D
15	Emulsified ^{HA}	100	Al	43	30	⁸⁹ Sr	7.0 69	115	90	1.27	136	7	1.26×10^{-5}	Sea	900	S
16	Emulsified ^{HA}	200	Al	87	40	⁸⁹ Sr	7.0 86	108	82	1.32	135	16	1.98×10^{-5}	Sea	900	S

Note: 1) Emulsified asphalt-HA: asphalts containing 40% water

2) R-Cu Sludge: $\text{CuO} \cdot x\text{H}_2\text{O} \cdot \text{Cu}(\text{OH})_2$

3) Al sludge: $\text{Na}_2\text{PO}_4 \cdot x\text{H}_2\text{O} + (\text{SO}_4)_2 \cdot \text{Ca}(\text{OH})_2 + \text{NaOH}$

4) Pe sludge: $\text{Na}_2\text{PO}_4 \cdot x\text{H}_2\text{O} + \text{Ca}(\text{OH})_2 + \text{NaOH}$

TABLE C-1

(CONT.)

No.	Asphalt:		Sludge		MLSS ratio w/o dry solid	Trace:		Test Piece				Leached activity cpm/ml	Leaching rate g/cm ² day	Status of drum:		
	Make	Weight g-dry	Make	Weight g-dry		Mucilage	MLSS/g solid ml/plate	Weight g	Vol cm ³	Sp. Wt.	Surface cm ²			Vol ml		
17	Emulsified ^{MA}	200	Al	200	50	MS _F	2.0 g	68	48	1.42	74	26	5.27×10^{-5}	Sea	1000	5
18	Emulsified ^{MA}	200	Al	200	50	MS _F	2.0 g	78	53	1.47	77	48	3.55×10^{-5}	den	1000	5
19	Straight	100	Al	43	30	MS _F	2.0 g	41	30	1.22	74	20	6.86×10^{-5}	Sea	1000	5
20	Straight	100	Al	43	30	MS _F	2.0 g	80	48	1.25	75	13	4.33×10^{-5}	den	1000	5
21	Straight	100	Al	67	40	MS _F	2.0 g	69	52	1.32	77	26	6.22×10^{-5}	Sea	1000	5
22	Straight	100	Al	67	40	MS _F	2.0 g	68	51	1.33	74	26	5.11×10^{-5}	den	1000	5
23	Straight	100	Al	100	50	MS _F	2.0 g	75	52	1.44	78	185	3.5×10^{-5}	Sea	1000	5
24	Straight	100	Al	100	50	MS _F	2.0 g	75	53	1.43	76	290	3.1×10^{-5}	den	1000	5
25	Straight	100	Al	67	40	MS _F	3.03 g	69	53	1.30	78	40	8.36×10^{-5}	Sea	1000	5
26	Straight	100	Al	67	40	MS _F	3.03 g	70	52	1.34	78	50	7.67×10^{-5}	den	1000	5
27	Straight	100	Al	67	40	MS _F	2.0 g	72	56	1.28	79	25	5.93×10^{-5}	den	1000	5
28	Straight	100	Pc	67	40	MS _F	2.0 g	115	87	1.40	80	40	9.26×10^{-5}	Sea	900	5
29	Straight	200	Al	66	30	MS _F	2.0 g	96	87	1.37	104	32	7.23×10^{-5}	Sea	900	5
30	Emulsified ^{MA}	100	Al	100	50	MS _F	2.0 g	128	90	1.42	500	220	6.36×10^{-5}	Sea	1000	5
31	Straight	100	Al	67	40	MS _F	2.0 g	62	59	1.27	190	160	1.57×10^{-4}	den	1000	5

Note: 1) Emulsified asphalt-MA: amionic containing 40% water

2) E-Ru sludge: CuSO₄·4H₂O/CH₃3) Al sludge: MgSO₄·9H₂O/150, 74Ca(OH)₂/94OH4) Pc sludge: BaSO₄/94SO₄/Ca(OH)₂/94OH

TABLE C-2

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SLUDGE INCORPORATION IN ASPHALT WITH SURFACTANT

Nuclide: Ce-144, Cs-137, Sr-90

Sludge: Mixture of ferric hydroxide and calcium phosphate

Surfactant: Cationic diamine

Nuclide	Ferric hydroxide sludge					Pure asphalt		Neutralizing asphalt		Separated water by surfactant						Final product
	Water % wt	Dry wt g	Solid cont't % wt	Total activity d	Specific activity d/g	Type	wt g	Type	wt g	Water Vol ml	Eff. %	pH	Total activity d	Specific activity d/g	Weight g	
¹⁴⁴ Ce	65	2440	30	27912	9.8	Straight	1700	anion equal'd	300	950	53.5	8.9	37 (0.072)	1.6x10 ⁻⁴	350	
	65	1600	40	15680	9.8	"	670	"	250	740	62.2	7.8	3.5 (0.022)	4.8x10 ⁻⁵	190	
	47	1850	50	16355	6.3	"	600	"	450	870	55.0	6.0	71.5 (0.02)	8.2x10 ⁻⁵	212	
	65	3000	36	6000	1.1	"	2110	"	370	1150	48.0	9.3	320 (0.02)	2.8x10 ⁻⁴	420	
¹³⁷ Cs	65	4000	40	8000	1.2	"	1680	"	625	1950	67.0	8.4	370 (0.02)	2.7x10 ⁻⁴	405	
	65	3700	50	7400	1.3	"	1180	"	400	1650	63.0	8.2	440 (0.02)	2.7x10 ⁻⁴	330	
⁹⁰ Sr	65	1600	40	10720	6.7	"	670	"	250	570	52.5	7.8	108 (0.02)	1.6x10 ⁻⁴	230	

* Copper ferrocyanide sludge was used.

TABLE C-3

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RESULT OF TEST OF THE SALTING-OUT EFFECT OF SURFACTANTS

Test No.	Base material	Type	Surfactant	Type	Water (g)	M ₂ added (g)	Neutralizing asphalt	Weight (g)	Asphalt (Total) (g)	Basic (g)	pH	Water (ml) (vol)	(g)	Preparation	S. P. (g)
Base material: emulsified	Cationic emulsified	B-1	NaOH/Ca ₃ (PO ₄) ₂	65	122	180	Alkaline emulsified	81.5	100	X	7.6	73	35.0	P	3.0
					180										
					240										
		B-2	Ca ₃ (PO ₄) ₂	71	168	230	- ditto -	83.5	100	X	5.6	135	64.0	P	5.0
					230										
					360										
	Anionic emulsified	B-3	AlON ₃ /Ca ₃ (PO ₄) ₂	73	158	260	- ditto -	83.5	100	X	8.6	150	45.0	P	8.0
					260										
					370										
		B-4	Per(OH) ₃ /Ca ₃ (PO ₄) ₂	65	100	200	- ditto -	83.5	100	X	11.9	80	30.0	P	11.0
					200										
					300										
Base material: emulsified	Cationic emulsified	B-1	NaOH/Ca ₃ (PO ₄) ₂	65	122	180	Alkaline emulsified	81.5	100	X	7.6	73	35.0	P	3.0
					180										
					240										
		B-2	Ca ₃ (PO ₄) ₂	71	168	230	- ditto -	83.5	100	X	5.6	135	64.0	P	5.0
					230										
					360										
	Anionic emulsified	B-3	AlON ₃ /Ca ₃ (PO ₄) ₂	73	158	260	- ditto -	83.5	100	X	8.6	150	45.0	P	8.0
					260										
					370										
		B-4	Per(OH) ₃ /Ca ₃ (PO ₄) ₂	65	100	200	- ditto -	83.5	100	X	11.9	80	30.0	P	11.0
					200										
					300										
Base material: emulsified	Cationic emulsified	B-1	NaOH/Ca ₃ (PO ₄) ₂	65	122	180	Alkaline emulsified	81.5	100	X	7.6	73	35.0	P	3.0
					180										
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					230										
					360										
	Anionic emulsified	B-3	AlON ₃ /Ca ₃ (PO ₄) ₂	73	158	260	- ditto -	83.5	100	X	8.6	150	45.0	P	8.0
					260										
					370										
		B-4	Per(OH) ₃ /Ca ₃ (PO ₄) ₂	65	100	200	- ditto -	83.5	100	X	11.9	80	30.0	P	11.0
					200										
					300										
Base material: emulsified	Cationic emulsified	B-1	NaOH/Ca ₃ (PO ₄) ₂	65	122	180	Alkaline emulsified	81.5	100	X	7.6	73	35.0	P	3.0
					180										
					240										
		B-2	Ca ₃ (PO ₄) ₂	71	168	230	- ditto -	83.5	100	X	5.6	135	64.0	P	5.0
					230										
					360										
	Anionic emulsified	B-3	AlON ₃ /Ca ₃ (PO ₄) ₂	73	158	260	- ditto -	83.5	100	X	8.6	150	45.0	P	8.0
					260										
					370										
		B-4	Per(OH) ₃ /Ca ₃ (PO ₄) ₂	65	100	200	- ditto -	83.5	100	X	11.9	80	30.0	P	11.0
					200										
					300

TABLE C-4
OPERATIONAL RESULTS FOR BITUMEN SYSTEM TESTING⁸⁴

Sludge Run (Total: 48 Runs)	Used Bitumen	Straight Asphalt 60/80 or 80/100
	Sludge Component	$\text{Fe}(\text{OH})_3 / (\text{Fe}(\text{OH})_3 + \text{Ca}_3(\text{PO}_4)_2) = 20-45$
	Treated Volume	Av. $1.2\text{m}^3/\text{Run}$, Total Volume: 59.2m^3 , Solid Contents of Sludge: About 5w/o
	Operational Condition	Thermo-oil Temperature: Max. 300°C , Operational Time: $3.5-7.0\text{ Hr/Run}$ Mixing Temperature: $200-230^\circ\text{C}$, Evaporation Rates of Water: Max. 31 kg/H
	Product	Specific Gravity: Av. 1.43 , Treated Volume: About 100 l/Run , Radioactivity: Av. 2.0 mCi/Run Mix. Ratio; Bitumen: Solid = $57:43(\text{Av.})$ Surface Dose Rates on Product: Av. 13 mR/H
Evaporator Concentrate Run (Total: 2 Runs)	Used Bitumen	Straight Asphalt 60/80
	Concentrate Component	Na^+ : 0.52 , Cl^- : 0.31 , SO_4^{2-} : 1.62 , Fe: 1600ppm
	Treated Volume	Av. $0.18\text{m}^3/\text{Run}$, Total Volume: 0.36m^3 , Solid Contents of Concentrate: About 8w/o
	Operational Condition	Thermo-oil Temperature: Max. 300°C , Operational Time: $2.0-3.5\text{ Hr/Run}$ Mixing Temperature: $130-150^\circ\text{C}$ (Only Evaporator Concentrate) Evaporation Rates of Water: Max. 62.5 kg/H
	Product	Specific Gravity: Av. 1.34 , Treated Volume: About 100 l/Run , Radioactivity: Av. 7.0 mCi/Run , Mix. Ratio; Bitumen: Solid= $65.35(\text{Av.})$ Surface Dose Rates on Product: Av. 20mR/H (10 parts of 35 is solid of concentrate. 25 parts of 35 is solid of sludge.)

containing significant amounts of oxidizing compounds will not detonate at the temperatures and pressures generated under conceivable storage and transport conditions. Other workers report no detonation problems. Phase separation may occur in bitumen waste forms with even relatively low waste form temperatures (60°C) encountered in storage and transport. The thermal conductivity of typical bitumen waste forms is 0.0-0.3W/m/°C.²⁰

The mechanical strength of bitumen waste forms will be highly dependent upon the properties of the bitumen used. The type and amount of waste contained will also affect the strength of the waste form. Temperature will have an effect on the mechanical strength of the waste form as bitumen softens with increasing temperature. The incorporation of salts which tend to oxidize bitumen may lead to an undesirable hardness and brittleness in the final product.

The irradiation of bitumen can result in gaseous evolution, chemical reactions between bitumen and incorporated solids (resulting in radiation-produced radicals), and oxidation of the bitumen. Radiation stability of the waste form has been shown to be dependent upon the type of bitumen used, since the radiolytic yield for different hydrocarbons differs. It has been suggested that, when hard bitumen is irradiated, porosity develops and the gaseous radiolysis products escape. In more elastic bitumens, the gas tends to remain in the sample, though considerable swelling occurs. Work at Oak Ridge has shown the opposite result. Radiolytic gas evolution is noted at integrated doses as low as 10^6 rads. The gases evolved are potentially explosive. Doses of 10^7 rads and above may produce swelling with an increase in sample volume of as much as 160 percent. Radiolytic oxidation of bitumen increases its hardness and brittleness.²⁰

The radiation stability of the solids to be incorporated in bitumen and the possible chemical reaction of radiolysis-produced radicals must also be considered. Generally, the solids have greater radiation stability than the bitumen itself with the result that the radiation stability of the waste form is higher than the bitumen alone. In some cases, irradiation was found to have no effect on leach rates, while in others, the leach rates were significantly increased.²⁰

Bitumen waste forms are normally non-corrosive, although this is partly a function of the type of waste incorporated. Asphalts efficiently resist the action of most acids, alkalis and salts. Bacteriological attack on bitumen and bitumen compounds has been noted. Such attack has been regarded as generally insignificant, but the need for the study of long-term effects is noted.²⁰

APPENDIX D
UREA-FORMALDEHYDE POLYMER^{53,70,76}

TERA CORPORATION

1.0 UREA-FORMALDEHYDE RESINS⁷⁶

The urea-formaldehyde (UF) resins used in radwaste solidification systems are viscous, syrupy, milky-colored materials which are commercially available from a number of suppliers, among them the Borden Chemical Company (Casca-Resin) and the American Cyanamid Company (Cyanaloc). The products have a limited shelf life depending upon the temperature. Upon prolonged exposure to air or with addition of an acid catalyst, cross-linking polymerization occurs and a solid is obtained.

When used as a solidification agent for radwaste, the mixture of UF resin and radwaste is adjusted to pH 1-2 by addition of a weak acid or acid salt catalyst such as phosphoric acid (H_3PO_4) or sodium bisulfate ($NaHSO_4$). To minimize the amount of catalyst needed to adjust the pH for highly buffered solutions (e.g., partially neutralized boric acid wastes), dilute solutions of strong acids (sulfuric acid) may be used. Upon addition of the catalyst, a condensation-polymerization reaction occurs which is similar to the reaction that took place during the partial polymerization used in the preparation of the resin. The reaction is slightly exothermic and is both temperature and pH dependent. The amount of catalyst added controls the setting time. All of the several possible mechanisms for forming the cross-linked UF polymer produce water as an end product of the reaction. The UF polymer formed varies according to reaction conditions but typically has a low molecular weight and an average degree of polymerization of 6-7. The time required for the polymer to reach full strength is shortened by increasing the temperature or increasing the catalyst concentration.

In actual operating practice, the volume ratio of evaporator concentrates to UF resin ranges from 1 to 3. The amount of catalyst required to reach the desired pH must be determined for each waste and normally comprises about 2 to 3 volume percent. After the initial setting, curing to a reasonably hard solid takes place over several hours and sometimes free, slightly acidic water, which can be mildly corrosive to the container, is released in the process. Portland cement or sorbent materials have been added to the waste containers to eliminate this residual water after the mixture has set. Urea-formaldehyde resins are also used to encapsulate radwastes such as dewatered filter sludges and spent demineralizer resins slurried with small quantities of liquid waste. Certain wastes

(e.g., soap solutions and concentrated Na_2SO_4) are difficult to incorporate into UF resin. Acceptable sodium sulfate products can be obtained with fresh UF resin by diluting the Na_2SO_4 to less than 10 wt percent prior to resin addition or by precipitating the excess sulfate with calcium chloride.

Paper pulp or wood flour imparts strength when added to UF resin formulations, and many of the same substances that are added to cement (e.g., the various clays) can also be added to UF resin to make the products less leachable. In general, however, product strength decreases and leachability increases as the ratio of liquid to UF resin increases. On the other hand, UF products exposed to air lose water by evaporation and may become friable if completely dehydrated.

Each of the UF systems now being marketed uses an in-line mixing procedure for combining the UF resin with the radwaste. All are essentially batch processes since the catalyst is added to the resin and waste mixture either in the fillport to the product vessel or in the vessel itself. A stirred-vessel process for incorporating radwaste in UF resin is depicted in D-1. Waste and resin are pumped to the product container through an in-line mixer, while the catalyst solution is concurrently pumped to the product vessel in which the waste, resin, and catalyst are stirred.

FIGURE D-1

STIRRED-VESSEL PROCESS FOR INCORPORATING RADWASTE IN
UREA-FORMALDEHYDE RESIN

2.0 UF CHARACTERISTICS^{70,20}

Urea-formaldehyde is ordinarily used as an aqueous emulsion containing approximately 40 percent water, with an inert material such as cellulose added to improve the adhesive properties of the liquid. The properties of urea-formaldehyde polymer are well characterized. The properties of urea-formaldehyde base TRU radioactive waste forms, as currently generated or proposed, are not known. The types of waste incorporated and their content in the final waste form are expected to have a large effect on their properties.

Some work has been done to characterize leaching of simulated urea-formaldehyde waste forms for liquid wastes. The rate of release of radionuclides depends upon the chemical content of the waste form (no chemicals versus boric acid, sodium sulphate or other exped chemical content), the UF liquid emulsion to waste volume ratio (1:1 to 1:3), and the commercial sources of the urea-formaldehyde. Bulk leach rates were not reported, but the fractional cesium releases per day varied from approximately 3×10^{-2} to 2×10^{-1} .²⁰ Boric acid and sodium sulfate addition, such as is found in various radioactive wastes, was found to enhance cesium leachability over that for the pure system (more so than similar addition for a cement system). Increasing the waste content increases the leachability of UF. Variability in the leach rates of UF systems from different suppliers differed by roughly 40 percent. Chemical additives to reduce leachability (e.g., Tiger-Lock) did lead to some improvement, and varied directly with the amount of additive used. Cast UF may ultimately crack and craze which could seriously diminish the leach resistance of a UF waste form.

Urea-formaldehyde resins and waste forms will burn when exposed to a flame but are also self-extinguishing, typically within 1-2 seconds. Differential Thermal Analyses also confirmed that the UF solids are non-flammable. The DTA analyses gave no evidence of exothermic reactions when the coupons were heated to $\sim 500^{\circ}\text{C}$ in air. Evaporation of water occurs up to about 130°C , and decomposition of the polymer occurs between 250 and 350°C . Decomposition is essentially complete at 350°C . Prolonged exposure to flame will reduce UF to an ash. UF (33 percent UF - 67 percent H_2O) was heated to 650°C in a furnace

in order to determine the change in volume and weight with increasing temperature. Significant volume changes began to occur at 150°C. Carbonization took place between 150° and 250°C, and partial carbonization and partial decomposition took place between 250 and 400°C. Decomposition was complete above 450°C. The residue fraction amounted to one percent of the original volume and 0.3 percent of the original weight after six hours at temperature.⁷⁰

UF has poor freeze-thaw resistance due to the large quantity of contained water. The thermal conductivity of urea-formaldehyde and UF waste forms is low (7.0 - 10.0×10^{-4} cal/sec/cm/°C for UF molding compound) minimizing the risk of radionuclide dispersion in a thermal environment. The toxicity of gases generated during burning is very low. UF product stability is good in a sealed system, but exposed to air, the polymer loses water by evaporation. Measurements were made of the density of the UF polymer as a function of time when the test specimens were exposed to open air. The density of the UF solid changed rapidly. The density of a 50 percent UF - 50 percent H₂O test coupon decreased nearly 50 percent after 10 days in dry air while the density of a 33 percent UF - 67 percent H₂O sample decreased about 70 percent after 15 days curing in dry air. A test coupon 25 percent UF - 75 percent H₂O decreased in density by nearly 80 percent after 10 days curing in dry air, though a test specimen that was sealed in plastic showed only an 11 percent decrease in density after 12 days of curing, and a test specimen sealed in plastic received from a commercial supplier more than six months ago is still intact without appreciable change. Thus, UF polymer will retain its integrity over a long period of time if it is not exposed to dry air, and prolonged drying leads to product degradation, decreasing mechanical strength, and increased leachability.

Water retention or binding by the UF polymer was also tested.⁷⁰ All tests showed some free water release (as liquid) during the first 48 hours of setting. A 50 percent UF solid block (~7 liter) exhibited release of five percent of the water. A 33 percent UF block exhibited 0.7 percent water release initially and then 1-2 percent later on, and a 25 percent UF block exhibited a 1.0 percent water release during the 48-hour test. The liquid released in all instances was acidic, having a pH of about 1-2.

Urea-formaldehyde polymer with a 1:1 UF emulsion to liquid waste ratio exhibits a compressive strength of about 50 kg/cm^2 . However, in reactor waste applications, a 1:2 UF emulsion to liquid waste ratio is common, and can be as low as 1:3. These products should have significantly lower compressive strengths, as shown in Table D-1. The mechanical properties are also dependent upon both the type of waste solidified and its content in the waste form. Prolonged drying will degrade the properties of these waste forms. The possibility of the cast UF ultimately cracking and crazing can be minimized by additives such as cellulose. Table D-2 presents some limited data on immobilization of resin with UF, as well as bitumen and Portland Cement.

The threshold dose for mild to moderate radiation damage in urea-formaldehyde is approximately 3×10^6 rads, and for moderate to severe damage, 2×10^7 rads. A G value of 0.5 molecules/100 eV was obtained for gas evolution by radiolysis for urea-formaldehyde with a cellulosic filler. Gases evolved are primarily hydrogen. The radiolysis properties of a UF waste form will be greatly dependent on the waste type and content. Urea-formaldehyde with cellulosic fillers is more susceptible to radiation damage and will be brittle, and will blister, swell, and crumble upon exposure to gamma radiation.²⁰

The resistance of urea-formaldehyde polymer to chemical attack is good. However, while it is resistant to chemical attack by oils, solvents, and greases, urea-formaldehyde is decomposed by strong acid and alkali, and attacked by weak acid and alkali. The effect of acid and alkali on UF waste forms will depend strongly on the type of waste incorporated and its content in the waste form.²⁰

Urea-formaldehyde is polymerized by acid catalysts, which requires a pH~2 in the emulsion-waste mix. The monomer polymerizes by a condensation reaction which liberates water, some of which is present as free water in the waste package. The volume of free liquid released in UF solidification is about 0.5-1.0 percent by volume. This liquid is acidic with a pH~2 and may also contain free formaldehyde which leads to severe container corrosion problems. Because of this, steel containers for UF waste forms are ordinarily asphalt coated, though the container lifetimes are still considered to be short.²⁰

TABLE D-1
RELATIVE STRENGTHS OF UREA-FORMALDEHYDE POLYMER
AND CONCRETE BLOCKS⁷⁰

<u>Block Composition</u> ^(a)	<u>Density (g/cm³)</u> ^(b)	<u>Compression Strength (psi)</u>	<u>Comments</u>
50% U-F-water	0.95	707	good set
33% U-F-water	0.86	177	some inner liquid present
25% U-F-water	0.79	389	good set
27% Cement (III) 54% Sand	2.09	2940	good set
27% Cement (III) 54% Sand	2.10	2830	good set

(a) Volume percent

(b) The densities were determined after a six-day cure

TABLE D-2
IMMOBILIZATION OF ION EXCHANGE RESIN ^{(a)53}

Sample Number	Sample Description [Vol Agent/Vol Resin]	Resin Content, wt%	Vol. Product Vol Resin	Compressive Strength	Abrasion Loss (g/rev) ^(b)
IE-0	Urea-Formaldehyde Control [1 UF/1 water with NaHSO ₄ ^(c)]	0	--	30	0.5
IE-1	Urea-Formaldehyde [1 UF/2 resin with NaHSO ₄ ^(c)]	53	1.0	50	0.2
IE-2	Urea-Formaldehyde [1 UF/2 resin with Fe ₂ (SO ₄) ₃ ^(d)]	55	1.0	46	0.2
IE-3	Urea-Formaldehyde [1 UF/3 resin with Fe ₂ (SO ₄) ₃ ^(e)]	68	1.0	25	1.3
IE-4	Urea-Formaldehyde (1 UF/4 resin with NaHSO ₄ ^(c)]	75	1.0	5	Not Available
B-1	Bitumen	60	0.86	^(f)	0.005
IR-8	Type II Portland Cement ^(g)	24	1.66	94	1.1

- (a) Powdered Dowex 1x8, 200-400 mesh in chloride form; it was moist to the touch and contained about 60% water
 (b) Using Los Angeles abrasion machine.
 (c) About 2% by weight added as catalyst.
 (d) About 4% by weight added as catalyst.
 (e) About 6% by weight added as catalyst.
 (f) Sample flows at pressures greater than 5 kg/cm².
 (g) See also Table 8 for additional results with portland cement.

Urea-formaldehyde is susceptible to biodegradation, although very little is known in this respect. Biodegradability is increased by the inclusion of cellulose filler and may be affected by the incorporation of biodegradable components in the waste. Free formaldehyde, if present in the waste form, should minimize biodegradability.²⁰

The following conclusions were made from this preliminary evaluation of urea-formaldehyde solidified products:

- The solid is non-flammable but is combustible.
- Water is not completely bound in urea-formaldehyde sets, and a small quantity of an absorbent and possibly neutralizing agent was usually required in the drum after the polymerization was complete to assist in retaining the water.
- The solid does have reasonable mechanical strength.
- The solid loses mechanical strength if allowed to completely dry out.

APPENDIX E
CEMENT IMMOBILIZATION^{20,76,53,34,114}

TERA CORPORATION

1.0 INTRODUCTION

The chemical and physical properties of cement are well known, particularly because of its use as a construction material over the years. It is also readily available and inexpensive. However, knowledge of the effect of mixing cement with radioactive wastes is somewhat limited.

Information on the properties of the cement waste forms currently being generated at nuclear power industries is limited. The property information which is known for simulated cement waste forms is significant information regarding the pure cement system itself.

However, property evaluations made on such cement waste forms can only serve as indications of waste form properties, since, in general, they do not adequately simulate power reactor wastes. Also many differing radioactive wastes are often mixed and solidified together, with the result that the properties of the waste relating to the ultimate properties of the solidified form, are not known. As mentioned previously, the chemical and physical properties of even a given waste type within a facility vary over wide ranges of specific activity, radionuclide content, pH, salt content, chemical composition, viscosity, etc., making simulations difficult.¹¹³

The two methods used for mixing rad-waste with cement are in-drum (or in-container) and in-line mixing. These are described following a discussion of the physical and chemical properties of cement.

Portland cement is often used in radwaste solidification processes. The major constituents of portland cement are tricalcium silicate (3CaO SiO_2), dicalcium silicate (2CaO SiO_2), tricalcium aluminate ($3\text{CaO Al}_2\text{O}_3$), and tetracalcium alumina ferrite ($4\text{CaO Al}_2\text{O}_3 \text{Fe}_2\text{O}_3$).⁷⁶ Portland cement is classified as Type I, II, III, IV, or V, depending upon the proportions of these compounds in the mixture. Minor constituents, such as lime (CaO), magnesia (MgO), or gypsum (MgSO_4), can have a significant effect of the swelling and setting properties of

the cement. Gypsum is added to prevent flash setting. The American Society for Testing and Materials (ASTM) has defined the restrictions on the chemical composition of portland cements as those imposed in all national standard specifications.⁷⁶

Studies on the fundamental chemistry of the hardening and setting of portland cement show that upon addition of pure water, both the dicalcium and tricalcium silicates react to form an amorphous, high-strength "rigid gel" or "mineral glue" composed of colloidal tricalcium disilicate hydrate ($3\text{CaO} \cdot 2\text{SiO}_2 \cdot 3\text{H}_2\text{O}$) in a nearly homogenous mass. The tricalcium aluminate and tetracalcium alumina ferrite apparently form the crystalline hexahydrates ($3\text{CaO} \cdot \text{Al}_2\text{O}_3 \cdot 6\text{H}_2\text{O}$ and $3\text{CaO} \cdot \text{Fe}_2\text{O}_3 \cdot 6\text{H}_2\text{O}$) indicating a capacity for holding water that is nearly double that of the silicate compounds. The total amount of water chemically tied in hardened portland cement paste corresponds to approximately 25 percent by weight. In the initial setting process, a coagulation structure is formed by individual crystallites in the amorphous gel; subsequent hardening proceeds as a fine crystalline network builds within the coagulated framework.²⁵

In actual nuclear waste disposal, where radwaste solutions and slurries (instead of pure water) are combined with the portland cement, the chemistry of solidification becomes far more complicated and undefinable. Practical experience, however, has shown that, although Type I portland cement is the one most commonly available, Type II is preferable in most radwaste applications because it is more resistant to sulfate deterioration. Boric acid wastes are known to retard the setting of portland cement, and some investigations of boric acid and cement mixtures are being carried out at Brookhaven National Laboratory²⁰ and Hanford Engineering Development Laboratory.^{53,34} In general, waste-cement products with satisfactorily high mechanical strength and low leach rate contain about 5 to 10 percent by weight waste solids with a rapid decrease in strength reported when the waste solids exceed 10 to 15 percent by weight. Marked decreases in mechanical strength may also occur when dewatered resins and/or filter sludges are incorporated in cement. Using cement as the solidification agent for liquid radwaste always results in a net volume increase. The final

solidified waste volume for waste-cement products can be as much as a factor of 2 (or more) greater than the volume of the incorporated liquid. The use of additives such as clays, shales, fly ash, or sodium silicate can either enhance or mitigate certain chemical or physical properties of the cementing process. The liquid tolerance of portland cement is increased by sodium silicate addition, thus permitting greater shipping efficiencies (i.e., volume of waste per unit volume shipped).

In the early nuclear facilities, in-drum mixing was accomplished by electrically driven paddle-type mixers or drum rolling, and by filling the drums with dry cement intimately mixed with vermiculite prior to introducing the liquid through a header at the top center of the capped drum. The vermiculite served as a medium to disperse the liquid evenly throughout the mix. More recently, a fully automatic in-drum mixing scheme has been developed in which preweighed dry cement and a metal mixing bar are placed in a closed-top 208-liter drum. The drum is then placed behind shielding and remotely filled with waste liquid which has been adjusted to the proper pH, concentration, etc. After being capped remotely, the drum is next placed on an end-over-end tumbler for thorough mixing. A line diagram of the system is given in Figure E-1. Figure E-2 presents a similar schematic for an in-line system.

FIGURE E-1

IN-DRUM MIXING PROCESS FOR INCORPORATING RADWASTE IN CEMENT⁷⁶

FIGURE E-2

IN-LINE MIXING PROCESS FOR INCORPORATING
RADWASTE IN CEMENT²⁶

2.0 CEMENT CHARACTERISTICS^{20, 34, 114, 116}

Leach rate testing of cements has used a wide variety of testing methods with little attempt at standardization. Among other factors, values determined depend upon the radionuclide measured, leachant composition, frequency of leachant changing, pH, temperature, and duration of testing. These tests generally do not attempt to predict the rate of radionuclide release, but rather to serve as a comparison for different solidification methods. Because different test procedures are used, leachability data found in the literature cannot normally be correlated. Leach rates are reported in many different units. The most common appears to be as a fraction release of the species of interest, or as a bulk leach rate, $\text{g}/(\text{cm}^2\text{-day})$, which relates the mass of the sample, its external surface area, and the duration of leach testing to the cumulative fraction release of the species of interest.²⁰

Bulk leach rates of 10^{-1} - 10^{-9} $\text{g}/(\text{cm}^2\text{-day})$ for cement monoliths have been reported in the literature.²⁰ Values for the leaching of soluble salts, such as sodium nitrate in cement, are generally 10^{-1} - 10^{-2} $\text{g}/(\text{cm}^2\text{-day})$, with the higher values associated with high salt content. Cesium and strontium exhibit lower bulk leach rates when adsorbed on sludges, clays, ion-exchange resins and zeolites. Leach rates of 10^{-4} - 10^{-6} $\text{g}/(\text{cm}^2\text{-day})$ of strontium from hydroxide sludges are generally reported. Cesium and strontium leach rates of 10^{-2} - 10^{-5} $\text{g}/(\text{cm}^2\text{-day})$ from clays and zeolites are found in the literature. Mixed ion-exchange resins in cement leached in sea water released 3.15 percent of contained Cs-137 in 2,540 hours.²⁰ Leachability of organic ion-exchange resins increases with total radiation dose. Zeolites in cement exhibit cesium and strontium release rates of generally 10^{-4} - 10^{-5} $\text{g}/(\text{cm}^2\text{-day})$, but the rate of cesium release is increased by the addition of ferric hydroxide sludge to the concrete. Plutonium and americium bulk leach rates in cement, of 10^{-7} - 10^{-9} $\text{g}/(\text{cm}^2\text{-day})$, have been reported. The leach rates of cesium and strontium are also a function of the type of cement used (portland, high alumina, or pozzolanic). The relatively high leach rates for many concrete waste forms is ascribed to the inherent porosity of the material, with an actual surface area exposed to the leachant of up to 8,000 times that of the external surface.

The thermal stability of concrete waste forms under normal conditions for the incorporation of Savannah River Plant high level waste has been classified as adequate.¹¹⁶ That concrete itself is thermally stable under normal conditions is evidenced by the extent to which concrete is used as a construction material. Concrete is incombustible and has good fire resistant properties. In fire environments it tends to lose hydration water above 121°C but with only relatively small decreases in the compressive strength under 538°C. If held at this or higher temperatures for appreciable periods of time, considerable strength losses and failure by cracking or spallation may occur. The effect of high temperatures on concrete is highly dependent on the aggregate, which is radioactive waste in this case. No information on the thermal stability concrete waste forms has been found. Concrete waste forms may fail due either to gas generation as a result of waste decomposition or differences in thermal expansion of the concrete and the waste. This behavior is aggravated in most radioactive waste forms since the waste content of the concrete is high, resulting in low mechanical strengths. In a fire, waste form damage is dependent upon the rate at which heat is absorbed by the concrete. The thermal conductivity of concrete is highly dependent upon the aggregate, density, and water content, with values of $3.4\text{--}8.6 \times 10^{-3}$ cal/sec/cm/°C reported for commercial construction concretes.

The mechanical strengths of concrete waste forms currently generated at operating facilities are not known. However, low strengths are probable due to the high waste loadings used. Some concretes may have essentially no mechanical strength either as a result of the cement not curing or the waste loading being so high that sufficient cement is not present to bind it together. The compressive, tensile, and impact strengths of the waste are directly related to failure under load and resultant dispersion of the waste. The effect of hydroxide sludge content on compressive strength has been measured and is found to decrease rapidly with sludge loading. However, these waste forms are not directly indicative of most waste forms. Evaluation of the likelihood of cracking, fracturing, and resultant dispersion of the waste form requires a knowledge of its mechanical properties which is currently not available. Concrete waste forms are also susceptible to degradation as a result of freeze-

thaw cycling. This may be of particular importance since some wastes contain substantial quantities of absorbed water.

Considering the activity contents of solidified low-level TRU waste forms, it is unlikely that radiation stability is of major importance. Concrete itself has good radiation stability as shown by its use and long term exposure in nuclear reactor applications. However, the presence of water, nitrates, and other components susceptible to radiolysis could result in gas generation and pressurization problems which are potentially hazardous. The radiation stability of reactor wastes has not been evaluated in the context of incorporation into concrete forms.

Concrete is susceptible to corrosion in burial environments, particularly by acid and sulfate waters. Sulfate and boric acid containing wastes may contribute significantly to such corrosion. Chemical reactions of a waste constituent may result in degradation of the waste form. Corrosion of the concrete waste form would be expected to increase leachability, decrease mechanical strength, and promote dispersibility. The integrity at long times and in burial environments of power reactor waste forms as currently generated is not known. Concrete placed in a moist environment is alkaline due to dissolution of hydroxides and hydration of free lime. This should have little effect on the container since at ordinary temperatures steel is rather stable in an alkali environment.

Concrete itself is not biodegradable although it may be affected by the action of chemicals generated by organisms in the soil. However, if the constituents of the waste incorporated in the waste form are biodegradable, radionuclide dispersibility may result.

3.0 FIXATION OF WASTES BY CEMENT

3.1 FIXATION OF LIQUID WASTES USING PORTLAND CEMENT

Experiments were performed with portland cement (type I - II) to determine what quantity of liquid waste could be incorporated while still obtaining a quality cement product, i.e., one with no free solution and with good strength. For water, formulations containing up to 36 wt percent water were generally acceptable. Up to 14 volume percent NPH* was successfully fixed in concrete by first sorbing the NPH on vermiculite, then fixing the vermiculite in the cement. Up to 15 volume percent toluene was also successfully fixed in cement after first mixing it with an organic dispersing agent.⁷⁶

The results of cement fixation of water are summarized in Table E-1. Water contents less than 26 wt percent exhibited free liquid when allowed to set in a sealed system. At least 46 percent water could be fixed in concrete by use of an accelerator such as calcium sulfate (plaster of paris), sodium silicate, etc., but this normally results in a soft set (see sample 11, Table E-1). The results generally agree with the ASTM specifications for mixing water with portland cement. The specifications call for mixing 485 milliliters of water per kilogram of portland cement, which is equivalent to 33 wt percent water.

Results of cement fixation of NPH are summarized in Table E-2. The tests were generally performed by sorbing the NPH on a sorbent, such as vermiculite, then fixing the sorbent into cement. Up to 14 volume percent of NPH on vermiculite was successfully fixed; however, higher volume fractions of fixed organic resulted in free liquid being present. The free liquid was generally a result of the water in the cement mixture displacing the NPH from the sorbent. Although some reabsorption of the NPH by the cement occurred, some free organic was normally present.

* NPH - Normal paraffin hydrocarbon

TABLE E-1
FIXATION OF LIQUID WASTES USING PORTLAND CEMENT⁷⁶

Sample Number	Water Volume, ml	Cement Volume, ml	Volume Water, % Added	Weight% Water,	Initial Density, g/ml	Free ^(a) Liquid Volume, ml	Compression Strength, psi ^(b)	Comments
1	3500	8000	30	27	1.98	0	1590	very thick mix
2	3500	8000	30	26	2.07	0	1770	hard to mix
3	4000	7000	36	30	1.91	0	2190	good mix
4	4000	7000	36	32	1.86	0	3010	good mix
5	4000	6500	38	32	1.80	<10	1880	good mix
6	4000	6000	40	34	1.94	0	1240	thin mix
7	4000	6000	40	36	1.98	55	820	thin mix
8	4500	6000	43	37	1.79	330	710	thin mix
9	5000	6000	45	39	1.75	>300	710	thin mix
10	6000	6000	50	44	1.73	>500	1060	thin mix
11	5000	4000	50	46	1.60	0	440	soft set
		+ 1000 plaster of paris						
12	4000	2000	68	60	1.54	0	<170	very soft set
		+ 335 cc sodium silicate liquid						

(a) The determination of free water was made after a 24 hour set time.

(b) Compressive strength was determined after a seven day set time.

TABLE E-2

FIXATION OF ORGANIC LIQUID WASTES USING⁷⁶
PORTLAND CEMENT AND SORBENT MATERIALS

Sample Number	Water Volume, ml	Cement Volume, ml	Sorbent Material	Sorbent Volume, ml	Amount NPH Added, Volume, ml	% of Product	Initial Density, g/ml	Free Liquid Volume, ml	Compression Strength, psi
1	4,000	7,000	None	0	800	12	1.90	>500	2226
2	4,000	6,600	Zorb-A11	2,400	800	10	1.83	Trace dry after 4 days	1678
3	4,000	6,600	Vermiculite	2,400	800	10	1.63	Trace dry after 4 days	830
4	12,000	19,800	Vermiculite	7,200	2,400	11	1.59	Trace dry after 3 days	---
5	12,000	19,800	Zorb-A11	14,000	3,500	15	1.58	<100 ml dry after 3 days	---
6	12,000	19,800	Vermiculite	14,000	3,500	15	1.52	< 50 ml dry after 2 days	---
7	1,500	2,500	Vermiculite	1,350	450	14	1.49	0	---
8	1,500	2,500	Vermiculite	1,800	600	18	---	~ 25 ml	---
9	1,500	2,500	Vermiculite	2,250	750	22	---	~ 50 ml	---

E-11

Incorporation of immiscible liquids into cement using organic dispersants was also reviewed. By addition of equal volumes of toluene and organic dispersant (e.g., Tergitol NP-27* or Tween 85**), up to 15 volume percent of toluene could be fixed in cement without accumulation of free liquid.

3.2 IMMOBILIZATION OF TREATED WASTE RESIDUE

General studies on direct fixation of waste residues from simulated TRU wastes have been performed.³⁴ In particular, waste residues resulting from the acid digestion process have been incorporated into concrete and glass as possible solid waste forms for long-term waste storage.

Two sets of successful waste solidification runs were made with Portland type II cement mixes. Three types of waste material were incorporated into the cement: dehydrated (at 350°C) ADTU residue, HEPA filter residue, and ion exchange resin. Compositions of the test mixtures are shown in Table E-3 along with the compression strength data after a six-day cure in water. The data show that some of the six-day water cured material had compression strengths greater than 4,000 psi.^{***} Test specimens containing 13 and 9 volume percent dehydrated ADTU residue (samples 10E and 10F) produced cement sets with compression strengths of 5,320 and 4,420 psi, respectively. Test specimens containing 20 and 11 volume percent of residue from HEPA filter tests (samples 11B and 11C) produced cement sets with compression strengths of 4,700 and 3,800 psi, respectively. The two specimens containing 19 and 20 volume percent ion exchange resin (samples 11E and 11F) had compression strengths of 2,210 and 3,570 psi, respectively. However, the resin was not wetted by the cement which resulted in the resin not being set in the cement, particularly on the exposed surface of the cement.

* Carbide and Carbon Chemicals Co., New York, N. Y.

** Atlas Powder Company, Wilmington, Delaware

*** Concrete testing 2,000 to 4,000 psi after a 28-day cure is used on many construction projects.

TABLE E-3
STABILIZATION OF ADTU RESIDUE IN PORTLAND CEMENT MIXES³⁴
(6-DAY WATER CURE SPECIMENS)

Sample No.	Composition (Weight %)				Composition (Volume %)				Residue Description	Compression Strength (psi)	Density (g/cc)
	Cement	Aggregate	Water	Waste Residue (Dehydrated)	Cement	Aggregate	Water	Waste Residue (Dehydrated)			
10A Concrete Reference	18	69	13	0	20	61	19	0	Sand	1470	2.2
10B Cement Reference	77	0	23	0	70	0	30	0	None	4240	2.1
10D	20	67	13	0	21	61	18	0	Ground Glass	1770	2.0
10E	67	0	25	8	57	0	30	13	ADTU Evaporated at 350°C	5320	2.0
10F	68	0	25	7	61	0	30	9	ADTU Neutralized and dehydrated	4420	2.0
11A Concrete Reference	18	69	13	0	20	60	20	0	Sand	1240	2.1
11B	69	0	29	2	51	0	29	20	Large HEPA	4700	2.0
11C	68	0	28	4	57	0	32	11	Small HEPA	3860	2.0
11D	75	0	24	1	67	0	30	3	Small HEPA ash	2160	2.1
11E	69	0	21	10	57	0	24	19	Dry Anion exchange resin	2210	1.9
11F	66	0	15	19	62	0	18	20	Wet Cation exchange resin	3570	1.9

E-13

3.3 CALCIUM SULFATE-CEMENT SYSTEM⁵³

The residue that results from acid digestion of combustible wastes is primarily composed of acidic sulfates. Neutralizing the sulfate residues with lime yields primarily calcium sulfate. Calcium sulfate as a dry powder is a desiccating agent and accelerates the setting time of portland cement. Plaster of paris ($\text{CaSO}_4 \cdot 1/2 \text{H}_2\text{O}$) was used as the test reagent in these immobilization studies. The results, listed in Table E-4, were corrected for the additional half-mole of water. The resulting phase diagram for the three-component system of calcium sulfate-cement-water is shown in Figure E-3. The outer parameters were defined by the minimum water that could be added and still allow the mixture to be mechanically mixed and poured without the cement setting; and, the maximum water that can be added without any free liquid remaining after a 24-hour cure in a sealed metal container (simulated waste drum package). The minimum water compositions depend primarily on the mixer used and are not sharply defined; the maximum water compositions are sharply defined by the presence of free liquid. Maximum and minimum water lines define the range of workable compositions of cement and water for the salt tested.

The resulting products within the range of workable compositions had a wide range of characteristics. For example, the solidified products with a high water content exhibited low compressive strengths and also high abrasion losses. These products would be less desirable due to the ease with which they could be mechanically dispersed. Samples with low water content were generally stronger and had much lower abrasion losses. In general, these solidified products would be more desirable. Compressive strength for calcium sulfate mixes can be shown to be directly related to density, water content, and abrasion loss, as shown in Figure E-4. Thus, within the workable composition of the cement-water mixture, the compositions that produce the most desirable products can be further defined as shown in Figure E-3.

TABLE F-4
IMMOBILIZATION OF CALCIUM SULFATE IN CEMENT⁵³

Sample Number	Boundary ^(a)	Composition, wt%			Solid Density, kg/m ³	Compressive ^(b) Strength, kg/cm ²	Abrasion ^(c) Loss, g/rev
		Calcium Sulfate	Water	Cement			
1	Max.	0	35.7	64.3	1780	78	0.5
2	Min.	0	25.6	74.4	2050	115	0.6
3	--	2.3	36.2	61.5	1760	75	0.7
4		4.8	36.7	58.5	1760	58	1.9
5	Max.	3.9	45.2	50.9	1600	25	3.2
6	Min.	5.4	26.3	68.3	1950	164	0.4
7	--	8.2	47.1	44.7	1550	25	2.8
8	--	6.2	58.8	35.0	1470	<3	5.3
9	Max.	5.4	64.1	30.5	1340	<2	15.5
10	Min.	28.0	33.8	38.2	1840	110	0.5
11	--	14.0	66.1	19.9	1300	<1	15.5
12	--	51.1	48.9	0	1570	32	N.A.
13	Max.	10.6	74.6	14.8	1260	<1	32.7
14	--	48.6	51.4	0	1370	18	0.7
15	--	41.9	58.1	0	1380	10	1.2
16	Max.	35.7	64.3	0	1290	3	8.6
17	Min.	61.0	39.0	0	1650	62	0.4

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Determined after seven days curing.

(c) Determined using the Los Angeles Abrasion Machine.

FIGURE E-3
COMPOSITION RANGES FOR IMMOBILIZATION OF CALCIUM
SULFATE RESIDUES WITH PORTLAND CEMENT⁵³

HEDL 7704-175.1

FIGURE E-4

COMPRESSION STRENGTH VS. WATER CONTENT, DENSITY, AND
ABRASION LOSS FOR PORTLAND CEMENT-CALCIUM SULFATE PRODUCTS⁵³

3.4 FERRIC SULFATE-CEMENT SYSTEM

Ferric sulfate salts affect both the acidity and the setting properties of portland cement mixes. Preliminary immobilization studies showed that cement products containing ferric sulfate were of very low quality due to the acidic nature of the ferric sulfate, but that neutralizing ferric sulfate using lime gave higher quality. Test data from the lime-neutralized ferric sulfate are shown in Table E-5 and summarized in Figure E-5.

The results indicate that the water content required in the mixes increased very sharply with increasing ferric sulfate content. However, the higher water content resulted in lower-strength mixes, similar to the strength exhibited by calcium sulfate mixes.

3.5 SODIUM SULFATE-CEMENT SYSTEM

Results of sodium sulfate immobilization tests in cement are shown in Figure E-6; test data are listed in Table E-6. Unlike calcium sulfate, sodium sulfate does not accelerate the setting of the cement, but it does tend to increase the fluidity of the mix. Thus, sodium sulfate-cement products exhibit good strength over a wide range of compositions, up to 36 percent sodium sulfate.

3.6 SODIUM NITRATE-CEMENT SYSTEM

Results of sodium nitrate immobilization tests in cement are shown in Figure E-7 and summarized in Table E-7. Sodium nitrate tends to increase the fluidity of the mix and has little effect on the setting properties of the cement and on the workable water content of the mix. The resulting products had good strengths over a wide range of compositions, up to 44 wt percent sodium nitrate.

3.7 CALCIUM CHLORIDE-CEMENT SYSTEMS

Calcium chloride is commonly used in the concrete industry as an accelerator for portland cement mixes. The workable compositions for calcium chloride in

TABLE E-5
IMMOBILIZATION OF FERRIC SULFATE IN CEMENT⁵³

Sample Number	Boundary (a)	Composition, wt%			Solid Density, kg/m ³	Compressive Strength ^(c) kg/cm ²
		Ferric Sulfate ^(b)	Water	Cement		
1	Min.	3	26	71	1890	219
2	Min.	0	26	74	2020	164
3	Max.	3	53	44	1460	26
4	--	9	33	58	1770	164
5	Min.	5	32	63	1810	160
6	Max.	6	53	42	1450	16
7	Max.	0	36	64	1750	148
8	Max.	7	57	36	1500	3
9	Min.	7	38	55	1670	37
10	Max.	9	56	35	1390	1
11	Min.	8	40	52	1690	77
12	Min.	12	43	45	1610	62
13	Min.	18	46	35	1470	26
14	Max.	13	57	30	1380	1
15	Max.	16	56	28	1430	5
16	Min.	16	60	24	1540	30

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Ferric sulfate stoichiometrically neutralized with calcium hydroxide.

(c) Determined after seven days curing.

HEDL 705-145.1

FIGURE E-5

COMPOSITION RANGES FOR IMMOBILIZATION OF
NEUTRALIZED FERRIC SULFATE WITH PORTLAND CEMENT⁵³

HEDL 7705-145.2

FIGURE E-6

COMPOSITION RANGES FOR IMMOBILIZATION OF SODIUM SULFATE IN PORTLAND CEMENT⁵³

TABLE E-6
IMMOBILIZATION OF SODIUM SULFATE IN CEMENT⁵³

Sample Number	Boundary ^(a)	Composition, wt%			Solid Density, kg/m ³	Compressive ^(b) Strength, kg/cm ²	Abrasion ^(c) Loss, g/rev
		Sodium Sulfate	Water	Cement			
1	Min.	0	25.7	74.3	2020	261	0.83
2	Max.	0	37.6	62.4	1670	104	0.76
3	Min.	4.1	23.5	72.4	2070	341	1.09
4	Max.	3.8	39.2	57.0	1690	77	0.59
5	Min.	7.8	22.5	69.7	2160	338	0.84
6	Max.	7.9	36.5	55.6	1790	67	0.56
7	Min.	11.7	23.9	64.3	2120	254	0.84
8	Max.	11.0	34.4	54.6	1750	71	0.65
9	Min.	19.2	21.5	59.3	2160	187	0.77
10	Max.	17.5	36.7	45.8	1780	76	0.61
11	Min.	32.3	16.5	51.2	2220	131	0.54
12	--	36.8	19.6	43.6	2040	172	0.29

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Determined after seven days curing.

(c) Determined using the Los Angeles Abrasion Machine.

HEDL 7705-145.6

FIGURE E-7
COMPOSITION RANGES FOR IMMOBILIZATION OF SODIUM
NITRATE WITH PORTLAND CEMENT⁵³

TABLE E-7
IMMOBILIZATION OF SODIUM NITRATE IN CEMENT⁵³

Sample Number	Boundary ^(a)	Composition, wt%			Solid Density, kg/m ³	Compressive ^(b) Strength, kg/cm ²	Abrasion ^(c) Loss, g/rev
		Sodium Nitrate	Water	Cement			
1	Min.	3	25	72	2290	252	--
2	Min.	6	24	70	2060	137	0.9
3	Max.	2	34.5	63.5	1800	67	1.2
4	Min.	8	19	73	2090	177	1.0
5	--	--	27	67	2050	70	0.8
6	Max.	5	33	62	1850	59	1.0
7	Min.	11	19	70	2160	187	0.9
8	Max.	10	30	60	1910	47	1.5
9	Min.	16	19	65	2230	258	0.6
10	Max.	13	28	59	1910	40	1.8
11	Min.	22	15	63	2220	276	0.5
12	Max.	20	23	57	1970	46	1.8
13	Min.	31	14	55	2200	214	0.3
14	Max.	28	21	51	1980	89	1.0
15	Min.	44	11	45	2200	285	0.7
16	Max.	44	19	37	1960	40	1.8
17	Max.	46	18	36	1880	24	2.9
18	Min.	58	9	33	2000	179	0.4

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Determined after seven days curing.

(c) Determined using the Los Angeles Abrasion Machine.

cement were determined as shown in Figure E-8 and summarized in Table E-8. Like calcium sulfate, calcium chloride tends to dry out the cement so that above about 15 wt percent calcium chloride the cement will no longer set and the compressive strength is minimal.

3.8 SODIUM CHLORIDE-CEMENT SYSTEMS

The workable compositions of sodium chloride in cement were determined as shown in Figure E-9 and summarized in Table E-9. Like sodium nitrate, sodium chloride increases the fluidity of the mix and is workable over a wide range of compositions. Products of acceptable strength were obtained for cement mixtures containing as much as 51 wt percent sodium chloride.

3.9 ION-EXCHANGE RESIN-CEMENT SYSTEMS

Ion-exchange resin wastes are generated in various nuclear fuel cycle facilities. The largest fraction of resin waste comes from reactors, which use the resin to remove radionuclides from core cooling water and steam condensates. Resins used in this manner become highly radioactive from loaded or absorbed radionuclides prior to discharge, and therefore are classified as intermediate-level wastes.

Resin wastes are generally dewatered by draining prior to solidification and are then slurried into the cement mixture with just enough water to produce the desired product. The cement-water slurry fills the void space between resin beads; this factor, together with apparent resin shrinkage on addition of the cement slurry, allows packaging efficiencies as high as 100 percent, as shown in Tables E-10 and E-11. The presence of excess water during (or after) setting can cause the resin to swell. Total disintegration of cement-resin samples caused by swelling has also been observed when water was added to the solidified material.

Figures E-10 and E-11 show the workable cement compositions that can be used with a selected anion and cation resin, respectively.

HEDL 7705-145.3

FIGURE E-8

COMPOSITION RANGES FOR IMMOBILIZATION OF CALCIUM CHLORIDE RESIDUES WITH PORTLAND CEMENT⁵³

TABLE E-8

IMMOBILIZATION OF CALCIUM CHLORIDE IN CEMENT⁵³

Sample Number	Boundary ^(a)	Composition, Wt%			Solid Density, kg/m ³	Compressive ^(b) Strength, kg/cm ²	Abrasion ^(c) Loss, g/rev
		Calcium Chloride	Water	Cement			
1	Min.	0	25.5	74.5	2000	296	0.80
2	Max.	0	33.9	66.1	1890	14	0.73
3	Min.	2.9	28.2	68.9	2010	278	1.00
4R	Max.	3.2	39.1	57.7	1700	55	0.90
5	Min.	5.3	27.0	67.7	1920	139	0.75
6	Max.	4.8	41.3	53.9	1610	31	0.92
7	Min.	9.9	34.2	55.9	1830	46	0.90
8R	Max.	9.0	49.2	41.8	1540	14	2.93
9	Min.	18.6	33.5	47.8	1770	<1	18.0
10	Max.	14.4	44.8	40.8	1730	<1	25.0

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Determined after seven days curing.

(c) Determined using the Los Angeles Abrasion Machine.

HEDL 7705-145.5

FIGURE E-9

COMPOSITION RANGES FOR IMMOBILIZATION OF
SODIUM CHLORIDE RESIDUES WITH PORTLAND CEMENT⁵³

TABLE E-9

IMMOBILIZATION OF SODIUM CHLORIDE IN CEMENT⁵³

Sample Number	Boundary ^(a)	Composition, Wt%			Solid Density, kg/m ³	Compressive ^(b) Strength, kg/cm ²	Abrasion ^(c) Loss, g/rev
		Sodium Chloride	Water	Cement			
1	Min.	0	25.6	74.4	2390	180	0.76
2R	Max.	0	40.3	59.7	1750	65	0.75
3	Min.	3.6	23.9	72.5	2050	188	1.25
4R	Max.	3.6	38.5	57.9	1700	37	0.72
5	Min.	8.1	24.4	67.5	2100	165	1.07
6	Max.	6.4	37.3	56.3	1820	44	1.10
7	Min.	10.4	24.4	65.2	2050	201	0.85
8	Max.	10.4	35.0	54.6	1810	29	0.67
9R	Min.	16.6	17.9	65.5	2180	270	0.83
10	Max.	17.1	21.9	53.8	1830	34	1.29
11	Max.	34.2	19.3	46.5	1920	42	0.72
12R	Min.	33.0	14.9	52.1	2140	224	1.00
13	Max.	51.1	14.5	35.4	1904	99	1.10
14	Min.	45.9	12.8	41.3	2090	367	0.85
15R	Max.	51.0	19.7	29.3	1830	30	4.0
16	Min.	51.4	12.3	36.4	1980	226	0.89
17	Min.	58.0	12.5	29.5	1850	85	0.95
18	Max.	58.1	19.1	22.8	1880	19	6.5
19	Min.	67.4	13.9	18.6	1730	30	5.0
20	Max.	66.0	16.3	17.7	1810	12	15.0
21	Min.	70.1	13.4	16.5	--	--	--

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Determined after seven days curing.

(c) Determined using the Los Angeles Abrasion Machine.

TABLE E-10
IMMOBILIZATION OF ANION EXCHANGE RESIN IN CEMENT⁵³

Sample Number	Water Boundary ^(a)	Composition, Wt%			Produce Volume (ml)	Initial Resin Volume (ml)	Packing Efficiency % ^(c)	Initial Density (g/cc)	Comprehensive Strength (kg/cm ²)	Abrasion Loss ^(d) (g/rev)
		Cement	Water	Resin ^(b)						
1(e)	Max.	62	38	0	9350	0	0	1.75	1270	0.7
2(e)	Min.	75	25	0	8780	0	0	2.00	1700	0.9
3	Max.	58	39	3	9990	1000	10	1.67	940	1.1
4	Min.	71	26	3	8350	1000	12	1.92	920	1.0
5	Max.	52	41	7	12180	3000	25	1.51	570	1.2
6	Min.	67	25	8	9590	3000	31	1.81	1370	1.6
7	Max.	45	37	18	15310	7000	46	1.42	50	2.6
8	Min.	57	25	18	11620	7000	60	1.77	1310	1.1
9	Min.	54	21	25	4280	3000	70	1.66	1420	0.6
10	Max.	40	30	30	16230	10000	62	1.39	160	2.7
11	Max.	37	25	38	14360	12000	84	1.32	180	1.7
12	Min.	46	15	39	12870	12000	93	1.44	530	0.9
13	Min.	33	15	52	11360	12000	106	1.31	320	3.4
14	Max.	23	29	48	12920	12000	93	1.21	<20	30
15	Min.	43	15	42	16220	15000	92	1.39	460	1.6

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Dowex 1 x 8,200-400 mesh, chloride form. Apparent bulk density of resin = 625 kg/m³ (~50% water)

(c) Ratio of initial volume to final volume x 100. Apparent density of resin in cement = 1025 kg/m³.

(d) As determined using the Los Angeles Abrasion Machine.

(e) Control.

TABLE E-11

IMMOBILIZATION OF CATION EXCHANGE RESIN IN CEMENT⁵³

Number	Boundary ^(a)	Composition, Wt%			Product Volume (ml)	Initial Resin Volume (ml)	Packing ^(c) Efficiency %	Initial Density (g/cc)	Compressive Strength (kg/cm ²)	Abrasion Loss ^(d) (g/rev)
		Cement	Water	Resin ^(b)						
1 ^(e)	Max.	62	38	0	9,350	0	0	1.75	1,270	0.7
2 ^(e)	Min.	75	25	0	9,780	0	0	2.00	1,700	0.9
3	Max.	57	38	5	10,110	1,000	10	1.68	670	1.3
4	Min.	71	23	6	8,580	1,000	12	1.97	1,490	1.2
5	Max.	51	37	12	12,740	3,000	24	1.66	620	1.2
6	Min.	65	23	12	10,090	3,000	30	1.88	1,260	1.0
7	Max.	42	32	26	12,120	6,000	50	1.48	120	~4
8	Min.	55	20	25	11,090	6,000	54	1.73	1,490	1.3
9	Max.	35	30	35	15,220	10,000	66	1.43	<300	70
10	Min.	48	17	35	13,110	10,000	76	1.66	880	1.8
11	Min.	33	15	52	16,790	15,000	89	1.33	190	15
12	Max.	27	31	43	19,160	15,000	78	1.39	<500	75
13	Min.	26	17	57	15,560	15,000	96	1.28	50	30

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composite.

(b) Amberlite-200, 40-60 mesh, sodium form. Apparent bulk density ~ 760 kg/m³. (~50% water.)

(c) Ratio of initial volume to final volume x 100. Apparent density of resin in cement = 1200 kg/m³.

(d) As determined using the Los Angeles Abrasion Machine.

(e) Control.

HEDL 7704-175.3

FIGURE E-10

COMPOSITION RANGES FOR IMMOBILIZATION OF DOWEX
1x8 ANION EXCHANGE RESIN IN PORTLAND CEMENT

FIGURE E-II

COMPOSITION RANGES FOR IMMOBILIZATION
OF AMBERLITE 200 CATION EXCHANGE RESIN IN CEMENT⁵³

3.10 SODIUM SILICATE-CEMENT SYSTEM

Sodium silicate is sometimes used to accelerate setting of cement systems containing liquid wastes. Such setting can be very rapid (one minute), but the initial product is generally more like a gel than a cement product. The sodium silicate/portland cement system was studied to characterize workable compositions. Data for the system are summarized in Table E-12. Cement mixes containing sodium silicate and similar additives permit a high liquid content and generally higher waste content than regular cement. The curves shown in Figure E-12 exhibit this. These curves are similar to those for calcium sulfate and calcium chloride, but vastly different than those for most sodium salts such as sodium sulfate. Three regions are shown in Figure E-12. The main region (between the dotted and solid lines) defines compositions that can be made by most cement mixers. The second region (cross-hatched lines) includes cement/water compositions that are compatible but that set too quickly for ordinary mixers to handle upon addition of sodium silicate. A mixer that adds the sodium silicate just as the cement is poured is required for these compositions. This latter type of mixer yields the best products even for the slower setting mixes.

Product strengths at comparatively high cement contents are enhanced by the addition of sodium silicate. Low cement content and correspondingly high water content products are soft, of low strength and dehydrate rapidly in air.

TABLE E-12

CHARACTERISTICS OF SODIUM SILICATE-CEMENT COMPOSITIONS⁵³

Sample Number	Water Boundary (a)	Composition, Wt%			Initial Density g/cc	Compressive Strength (kg/cm ²)	Abrasion Losses (g/rev)
		Cement	Water	Sodium Silicate			
1(c)	Max.	62.6	37.4	-0-	1.74	1190	1.2
2	Max.	36	62	2	1.38	145	1.8
3(c)	Min.	73.3	26.7	-0-	1.94	1570	1.2
4	Min.	72.6	26.5	0.8	1.97	1570	1.1
5	Max.	30.9	64.8	4.3	1.28	20	14.5
6	Min.	70.5	27.2	2.3	1.92	1740	0.7
7	Max.	21.7	69.1	9.2	1.17	20	35
8	Min.	65.5	30.2	4.3	1.78	910	0.8
9	Max.	20.3	68.0	11.7	1.23	20	11
10	Min.	59.2	32.2	8.6	1.71	620	0.8
11	Max.	16.6	67.4	16.0	1.16	20	11
12	Min.	45.5	42.2	12.4	1.47	210	27
13	Max.	10.2	68.0	21.8	1.18	20	11
14	Min.	37.3	50.6	12.1	1.36	90	7.4
15	Min.	25.8	57.8	16.5	1.19	22	11
16	-- (instant gel)	51.7	30.1	18.2	1.65	815	--
17	-- (instant gel)	16.3	55.4	28.3	1.23	25	--

(a) Maximum water above which cement will produce residual free liquid; minimum water necessary for a mixable composition.

(b) As determined using Los Angeles Abrasion Machine.

(c) Control.

FIGURE E-12
COMPOSITION RANGES FOR SODIUM SILICATE-PORTLAND CEMENT SYSTEM⁵³

APPENDIX F
DEPARTMENT OF ENERGY FACILITY
TRU WASTE CHARACTERISTICS

TERA CORPORATION

1.0 IDAHO NATIONAL ENGINEERING LABORATORY

The Idaho National Engineering Laboratory (INEL), formerly named the National Reactor Testing Station, was established in 1949 as an area where nuclear reactors, support plants, and equipment could be constructed, tested, and operated. The INEL covers an area of about 2,300 km² on the central edge of the semiarid Snake River Plain. Since 1949, 51 nuclear facilities, mostly reactors, have been built, or are under construction or design at the INEL. These facilities include the Idaho Chemical Processing Plant (ICPP), which reprocesses irradiated spent nuclear fuel.⁵

TRU waste is generated at various on-site facilities as well as from numerous off-site DOE facilities. The TRU waste is composed of several categories: evaporator sludge; filter sludge; decontamination and decommissioning (D&D) materials (lumber, concrete, etc.); laboratory equipment (fume hoods, glassware, etc.); process equipment (tanks, pumps, etc.); protective clothing; and maintenance tools. The waste may be hazardous chemically as well as radiologically. The buried waste, in particular, is suspected of containing toxic chemicals, pyrophoric materials, and other materials that present nonradiological hazards.⁷⁷

Solid contaminated waste, except for the calcined solids from ICPP, is transported to the INEL Radioactive Waste Management Complex (RWMC); uncontaminated solid waste is deposited in a sanitary land fill.

1.1 DESCRIPTION OF FACILITIES⁵

INEL is comprised of the following:

1. Argonne National Laboratory - West (ANL-W)
2. Naval Reactors Facility (NRF)
3. Idaho Chemical Processing Plant (ICPP)
4. Test Area North Support Facility (TAN)

5. Loss of Fluid Test Facility (LOFT)
6. Power Burst Facility (PBF)
7. Test Reactor Area (TRA)
8. Central Facilities Area (CFA)
9. INEL Solid Radioactive Waste Management Areas
10. Auxiliary Reactor Area
11. Low Power Test and Experimental Beryllium Oxide Reactor Facilities
12. Idle Facilities

1.2 ON-SITE TRU WASTE GENERATORS⁶⁹

1.2.1 ARGONNINE NATIONAL LABORATORY-WEST (ANL-W)

ANL-W performs irradiation tests of reactor fuels and structural materials for DOE's fast breeder reactor program. These tests are subjected to thorough examination in the Hot Fuel Examination Facility adjoining the Experimental Breeder Reactor-II at ANL-W. In addition, the HEEF supports the Sodium Loop Safety Facility program at the Engineering Test Reactor, located at the Test Reactor Area. Intermediate-level TRU waste (see Table F-1A) resulting from these examinations is loaded into a waste container, which consists of inner and outer waste can assemblies, weighing a total of 270 kg (600 lb) loaded.

The waste container is described in Reference 11.

1.2.2 NAVAL REACTORS FACILITY (NRF)⁶⁹

Intermediate-level TRU waste (described in Table F-1B) is generated at the NRF Expanded Core Facility, where various post-irradiation examination programs involving light water, reactor breeder and naval reactor fuels are performed. Combustible waste material is segregated from noncombustible waste material and packaged separately. Noncombustible intermediate-level TRU waste consists of fuel cutting chips and contaminated equipment.

TABLE F-1A
CHARACTERISTICS OF WASTE FROM
ARGONNE NATIONAL LABORATORY-WEST⁶⁹

PHYSICAL

(a) Combustible TRU Waste

Bottles	polyethylene
Tubing	polyvinylchloride, tygon, rubber
O-rings	neoprene, silicone, teflon
Paper products	tissues, cardboard, towels, blotting paper, cellulose filters
Rags	cotton, mixed synthetics
Sheeting	polyethylene, polyvinylchloride
Wood products	plywood
Plastics	Lexan, nylon, electrical insulation

(b) Noncombustible TRU Waste

Ferrous metals	carbon steel, stainless steel, cast iron
Nonferrous metals	aluminum, brass, bronze, copper, lead, sodium oxide
Glass	bottles, fiberglass, tubing, mirrors, magnifying glass
Ceramics	porcelain insulators, quartz, micarta
Passivated chemicals	granular aluminum silicate used to solidify dissolved analytical samples of EBR-II and mixed oxide experimental fuels. The solvent for these samples is aqua regia.

NUCLEAR

- (a) Consists of 25 percent Co-58, 25 percent Cr-51, 25 percent Mn-54, 10 percent Co-60, 10 percent Cs-137, and the balance is other mixed fission products. Transuramics amount to less than 1 g Pu-239 per package.
- (b) For the purpose of criticality analyses, the fissile material limit for the container is 200 g of Pu. The major nuclide constituent will be fission products and induced activity contaminated with TRU elements.

RADIATION LIMITS

Less than or equal to 1,000 R/h at the container surface. Acceptance of higher radiation levels is under consideration.

TABLE F-1B

CHARACTERISTICS OF WASTE FROM NAVAL REACTORS FACILITY⁶⁹

PHYSICAL

(a) Combustible TRU Waste

Bottle	polyethylene, polypropylene
Tubing	rubber
Slave boots	polyvinylchloride, polyurethane, polyurethan with hypalon
Bags	polyvinylchloride, polyurethane, polyethylene, glassine
Rags	diaper cloth
Sheets	polyethylene, polyvinylchloride
Wood	G-tips, ruler, file handles, paint brush handles
Plastics	pipe end caps
Miscellaneous	RTV material, brush bristles, Fiberglass filter, Buna-N Slave Finger Tips

(b) Noncombustible TRU Waste

Equipment	tools and machinery (e.g., vacuum cleaners)
Fuel cutting chips	includes UO_2 , ThO_2 and Zircaloy

NUCLEAR

- (a) For criticality analysis purposes, combustible TRU waste contains assorted fission products and a maximum of 5 g of fissile material.
- (b) Combustible TRU waste generated through 1977 has contained no account-able fuel.
- (c) Noncombustible TRU waste will contain a maximum of 5 g fissile material.

RADIATION LIMITS

- (a) Combustible TRU waste containers have radiation levels less than or equal to 30 R/h at the surface of the container.
- (b) Noncombustible TRU waste will have radiation levels of 150 R/h or less at the surface of the container.

Besides these wastes, there are also the usual spectrum of low-level TRU waste (low-level of beta-gamma radiation) which are also generated and disposed at INEL.

1.2.3 TEST REACTOR AREA⁶⁹

Intermediate-level TRU waste (described in Table F-1C) from various irradiation experiments originated by Oak Ridge National Laboratory and Battelle Memorial Institute is being stored at INEL. The general purpose of these experiments was to study the behavior of various fuel rods in varying flux fields as burn-up increases. The BMI-50-12 experiment (see Table F-1C) contains an appreciable amount of sodium (approximately 63 g) which was used as a heat transfer medium and is contained in the experiment capsule in three layers, ranging in thickness from 0.6 mm (0.025 in.) to 1.5 mm (0.060 in.). Permanent disposition at INEL cannot occur with the sodium present. The sodium will be removed from the experiment or rendered nonreactive prior to shipment to such disposition.

1.2.4 CHEMICAL PROCESSING PLANT (CPP)⁶⁹

The CPP Fluorinel and Storage Facility will generate low-level and intermediate-level TRU waste after becoming operational in 1981. This facility employs the Fluorinel Fuel Dissolution Process as a "head-end" feed preparation process for the CPP main solvent extraction system. Fuels from reactors such as the Power Burst Facility, Loss of Fluid Test (LOFT) Facility, Shippingport Pressurized Water Reactor, and Training-Research-Isotope Production Reactor-General Atomics (TRIGA), will be processed. The waste will consist of various solid constituents of fuel and cladding (ZrO_2 , B_4C) removed from the process waste stream by a photo-etched disc filter system as well as the usual array of low-level TRU contaminated wastes. No further information is available on these waste characteristics.

TABLE F-1C

CHARACTERISTICS OF WASTE FROM THE TEST REACTOR AREA⁶⁹PhysicalTRU Waste

The material results from Oak Ridge National Laboratory and Battelle Memorial Institute (Ohio) experiments. ORNL experiment numbers are: ORNL 43-85 thru 88 and 106 thru 111 (excluding 109). The BMI designation is BMI-50-12. These experimental materials are currently being stored in the ETR canal in "X-baskets." (The dimensions of an X-basket are 0.95 m (37-9/16 in.) long and 35 mm (1-3/8 in.) in diameter).

Nuclear

Materials from the above listed experiments contain a total of 39 g Pu; 36 g U-233 and Th; deleted U and normal U in amounts less than normally reportable.

<u>Experiment No.</u>	<u>Quantity Pu (g)</u>	<u>Quantity U-233 (g)</u>
BMI-50-12	7	---
ORNL 43-85	---	9
ORNL 43-86	--	9
ORNL 43-87	---	9
ORNL 43-88	--	9
ORNL 43-106	7	---
ORNL 43-107	7	---
ORNL 43-108	6	---
ORNL 43-110	6	---
ORNL 43-111	6	---

Radiation Limits

Estimated to be up to 25,000 R/h at surface of "X-baskets."

1.3 SOLID RADIOACTIVE WASTE FACILITIES^{5,90}

The solid radioactive waste facilities at the INEL are the Radioactive Waste Management Complex (RWMC), SL-1 Burial Ground, ANL Solid Waste Storage Area, and the ICPP Calcined Waste Storage Area.

At the RWMC and SL-1 Burial Grounds, fission and activation product wastes are buried directly in the soil below ground level. Uranic and transuranic wastes are currently stored above the land surface in fire-resistant and watertight containers at the Transuranic Storage Area (TSA). These containers are placed on an asphalt pavement and covered with plywood-plastic and soil. At the ANL Solid Waste Storage Area and the ICPP Calcined Waste Storage Area, radioactive wastes are stored in watertight steel containers and concrete vaults placed below the ground.

1.3.1 INEL RADIOACTIVE WASTE MANAGEMENT COMPLEX (RWMC)⁵

The RWMC was established in 1952 to accommodate the radioactive wastes generated by INEL operations. In addition to wastes generated at INEL, the RWMC has also received wastes from the DOE's Rocky Flats Plant since 1954. Most of these wastes are contaminated with uranic and transuranic nuclides. Lesser quantities of waste have also been received over the years from other DOE-associated facilities.

In terms of area, RWMC constitutes 70 percent of all INEL waste facilities, including seepage ponds and storage areas for high-level waste. Four major on-site contractors ship their waste to RWMC. Also, other DOE facilities throughout the country send TRU waste to RWMC for storage.

Waste received for storage or disposal must comply with DOT shipping requirements, DOE-ID Manual 0529, and RWMC documents such as waste packaging guidelines, standard practices, and operating procedures.

RWMC is divided in two fenced sections: the subsurface disposal area (SDA), which is 35.6 ha, and the transuranic storage area (TSA), which is 22.3 ha. Approximately 8.1 ha remains for future disposal at SDA and 17.8 ha remains for future disposal at TSA. These are described below.

SUBSURFACE DISPOSAL AREA (SDA)

Underground and above-ground disposal activities are carried out in the 35.6 ha SDA. Facilities include trenches, pits and soil vaults or shafts.

Facilities for the TRU waste retrieval projects (Initial Drum Retrieval and Early Waste Retrieval (EWR) projects) are located within the SDA. These retrieval activities are contained within two structures known as the air support weather shields. The EWR is double contained, with the operating area enclosed in a second structure within the weather shield. The waste retrieved by these projects is repackaged and placed in a 20-year retrieval storage on the TSA-R pad.

TRANSURANIC DISPOSAL AREA (TDA)

Waste containing less than 10 n Ci/g of TRU elements is disposed of above ground on the Transuranic Disposal Area. The TDA consists of asphalt pads having level surfaces of about 1,900 m², enclosed on three sides by an earth berm. One such pad is presently in use (which is designated as Transuranic Disposal Area-TDA) at the RWMC. Drums and crates of uranic and transuranic waste containing less than 10 nCi of transuranic activity per gram of waste, are stacked on the pad in a horizontal geometric configuration to a height of about 5.5 m. Once a section of the pad is filled with containers, earth is moved over the top and sides of the stack to provide a moisture barrier over the waste. This storage concept allows for greater container integrity, an all-weather working surface, and maximum utilization of surface area within the RWMC.

TRANSURANIC STORAGE AREA (TSA)

The construction of the TSA is similar to that of the TDA and affords the same advantages. Transuranic wastes containing more than 10 nCi/g of TRU elements are stored on an interim basis on this pad, which measures 45.7 m x 228.6 m. The waste is packaged in containers designed for 20-year integrity. The bulk of the wastes stored on the TSA pad arrives at the INEL by rail from the Rocky Flats Plant. Some U-233 wastes from Bettis Atomic Power Laboratory (BAPL), which is associated with the DOE's Pittsburgh Naval Reactor Office, and other potential sources, such as the DOE's Mound Facility at Miamisburg, Ohio, are also received --along with any INEL-generated transuranic wastes. The containers must meet 20-year integrity requirements for storage on the TSA pad, along with other safety and administrative imposed requirements.

The TSA is filled in 25 x 46-m sections called "cells." The drums are stacked five drums high in a close-pitched triangular array with a layer of fire-retardant plywood and plastic sheeting separating each level. The sides of the cells are lined with 1.2 x 1.2 x 2.1-m fiberglass-coated wooden boxes that contain waste too bulky for the drums. The boxes define the cells and provide end support for the drum array. Each cell in the 228.6 m length of the pad is separated by a 0.9 m-thick soil firewall as the cell is completed. The entire array is covered with fire-retardant plywood and nylon-reinforced polyvinyl before the final 0.6 to 0.9 m-thick soil cover is placed.

Waste containing more than 10 nCi/g of TRU elements, plus activation and/or fission products in concentrations requiring special handling, is stored in the Intermediate-Level Transuranic Storage Facility (ILTSF). This facility consists of 26 concrete-filled carbon steel storage vaults constructed from 7.9 m vertical pipes, 61 and 40.1-cm in diameter. The pipes are enclosed in a 1.5-m-high earth embankment. A removable concrete plug in each vault provides shielding.

1.4 SOLID RADIOACTIVE WASTE DATA^{77,5}

Table F-1D presents a summary of approximate volumes of TRU wastes at INEL. Before 1970, TRU waste, totaling about $62,300 \text{ m}^3$ was buried in the SDA. Included were approximately 195,000 drums, 6,000 boxes, and 13,000 cartons described below. Approximately $4,670 \text{ m}^3$ of the total buried waste has been retrieved in experimental projects.

Of the buried waste, an estimated 9 percent is actually uranium waste -- essentially free from transuranics. About $2 \times 10^5 \text{ kg}$ of U-238 and $1.3 \times 10^2 \text{ kg}$ of U-235 are included in the buried waste. However, the activity is only about 67 Ci, or about 0.03 percent of the total activity of the buried TRU waste. In the stored waste, U-235 and U-238 comprise a fraction of less than 10^{-8} of the total activity. Twenty-six percent of the buried waste is nitrate salts of low TRU content. Before 1970, an estimated $14,200 \text{ m}^3$ of beta-gamma waste was buried in the SDA, along with the TRU waste.

Most of the cardboard cartons, wooden boxes, and metal drums originally containing the buried waste have deteriorated. Furthermore, some of the waste, such as large pieces of equipment, was buried without containers. As a result, some of the soil adjacent to the waste has become contaminated. If the buried TRU waste were retrieved, it was estimated that $106,200 \text{ m}^3$ of contaminated soil would require retrieval.* The retrieval operations themselves would contribute to the increased volume of soil that would become contaminated.

* The $106,200 \text{ m}^3$ of contaminated soil is based on experience from the Early Waste Retrieval project. Experimental results indicated an approximate ratio of contaminated soil to waste of 1.5:1. Thus, $1.5 \times (62,300 \text{ m}^3 \text{ of TRU waste} + 14,200 \text{ m}^3 \text{ of beta-gamma waste}) = 106,200 \text{ m}^3$ of contaminated soil.

TABLE F-ID
SUMMARY OF APPROXIMATE VOLUMES
OF TRU WASTE AND ASSOCIATED MATERIALS⁷⁷

<u>Waste</u>	<u>Total Emplaced (m³)</u>	<u>Inventory Projected to Jan. 1, 1985 (ft³)</u>
Buried TRU Waste	62,300	56,600
Buried Beta-Gamma Waste (Intermixed with TRU Waste)	14,200	14,200
Contaminated Soil (Associated with Buried TRU Waste)	0	106,200
Stored TRU Waste	34,000*	56,600

* Through December 31, 1977.

The TRU waste received since 1970 (approximately $31,150 \text{ m}^3$ through the end of 1977) has been stored aboveground on the TSA. Through the end of 1977, an additional $2,850 \text{ m}^3$ of TRU waste had been stored on the TSA as a result of experimental projects for retrieving the buried waste. The $34,000 \text{ m}^3$ of waste is stored in approximately 83,000 drums, 5,000 boxes, and 170 bins. It is estimated that, by the beginning of 1985, about $56,600 \text{ m}^3$ of TRU waste will be in aboveground storage. Included in this total is the additional $1,840 \text{ m}^3$ from 1978 experimental retrieval projects.

Beta-gamma waste, as such, is not mixed with the stored TRU waste. However, there are small quantities of beta-gamma emitting radionuclides in the stored waste. The radiation levels produced at the surface of the steel bins average 50 mR/hr. For drums, the average is 3 mR/hr; for boxes, about 1 mR/hr.

Solid radioactive waste enters the SDA and TSA in the following containers:

- (1) Special shielded cask inserts (SDA)
- (2) 208- and 114-liter steel drums mostly from off-site shippers (TSA)
- (3) $1.2 \times 1.2 \times 2.1 \text{ m}$ wooden boxes (TSA)
- (4) Compacted 0.4 m^3 bales (SDA)
- (5) Fiber barrels (SDA)
- (6) Other special containers, such as the 0.02 mm yellow plastic bags used for compactible waste (SDA)

Special administrative forms contain information on waste volume and nuclide content and activity which is used as input to the DOE quarterly computerized radioactive waste reporting system (called SWIMS). Waste currently placed in the SDA is considered disposed of, whereas that stacked in the TSA is considered stored. From 1954 through November 1970, waste containing approximately $2 \times 10^5 \text{ Ci}$ of plutonium-contaminated waste was disposed of at the INEL Subsurface Disposal Area. Pu-241 comprised about 87 percent of this activity with lesser

amounts of Pu-238, -239, -240, and -242. The transuranic wastes that were buried at the Radioactive Waste Management Complex prior to November 1970 are presently being retrieved. An air-support building, is being used to avoid potential contamination of the environment during the retrieval program. The air-support building is positioned over the work site, the soil overburden is removed, the disposed 208-liter drums are recovered and then repackaged in 314-liter steel drums or wood and fiberglass boxes. After the drums have been removed from the area under the air-support building, the excavated area is backfilled with soil, smoothed, and graded to comply with drainage requirements. The air-support building is then moved to another location and the work sequence repeated. The repackaged drums from this retrieval program are being stored on the TSA. Ultimate disposal will be at a federal waste repository.⁵

While much of the data following are somewhat dated, they do provide the reader with an appreciation of the variety of radionuclide and other waste characteristics expected in the TRU waste. This is especially important when considering waste retrieval and processing.

Table F-IE is a list of the radioactive nuclides disposed of at the INEL in 1974. The radioactive nuclides stored at the TSA during 1974 are listed in Table F-IF. Table F-IG shows a 21-year summary of wastes disposed of at the SDA or stored at the TSA.

Table F-IH lists the TRU wastes at INEL as of December 31, 1977. The waste is categorized into Retrievably Stored - Contact Handled, Retrievably Stored - Remote Handled and Subsurface Disposal.

Table F-II categorizes the totals from Table F-IH with projections through the year 1984.

Table F-IJ gives the isotopic composition of TRU nuclides in TSA waste and Table F-IK gives the number, weight and average weight of TRU waste containers through 1977. Table F-IL records the combustibility and compactibility of TSA waste for the period from September 1971 through December 1977.

Table F-IM provides the TRU storage area data for the period September 1971 through December 1976.

TABLE F-1E
SOLID RADIONUCLIDES DISPOSED OF AT INEL SDA DURING 1974⁵

<u>Nuclide</u>	<u>Curies</u>
Antimony-125	107
Cerium-141	2
Cerium-144	805
Cesium-134	66
Cesium-137	1,424
Chromium-51	82
Cobalt-58	2
Cobalt-60	7,662
Europium-154	42
Europium-155	24
Iron-59	1,440
Manganese-54	72
Mixed activation products	92
Mixed fission products	761
Nickel-59	3,200
Plutonium-238	1
Plutonium-239	1
Plutonium-240	1
Plutonium-241	5
Plutonium-242	1
Praseodymium-144	324
Radium-226	1
Rhodium-106	48
Rubidium-86	63
Ruthenium-106	338
Strontium-90	1,579
Thorium-232	1
Uranium-233	1
Uranium-235	1
Uranium-238	5
Unidentified alpha	1
Unidentified beta and gamma	29
Yttrium-90	1,081
Zinc-65	361
Zirconium/niobium-95	154
Total	19,770

TABLE F-IF
TSA NUCLIDE SUMMARY IN Curies FOR 1974⁵

<u>Nuclide</u>	<u>Curies</u>
Americium-241	4,453
Mixed fission products	22
Plutonium-238	58
Plutonium-239	1,921
Plutonium-240	472
Plutonium-241	16,390
Plutonium-242	1
Uranium-232	3
Uranium-233	204
Uranium-238	<u>1</u>
Total	23,530

TABLE F-1G
SOLID RADIOACTIVE WASTE DISPOSED OF
OR STORED AT RWMC 1952-1974⁵

<u>Year(s)</u>	<u>Disposed</u>		<u>Stored</u>	
	<u>Volume (m³)</u>	<u>Curies</u>	<u>Volume (m³)</u>	<u>Curies</u>
1952-1964	61,738	771,300		
1965	7,533	702,900		
1966	9,487	924,300		
1967	9,657	877,800		
1968	13,735	301,500		
1969	11,498	971,000		
1970	12,517	504,400	1,416	4,225
1971	4,022	350,900	7,108	12,670
1972	3,540	214,700	5,919	28,500
1973	3,880	339,800	5,891	24,600
1974	<u>3,718</u>	<u>19,700</u>	<u>4,135</u>	<u>23,530</u>
Total	141,325	5,978,000	24,469	93,525

TABLE F-1H

TRU WASTE AT INEL
AS OF DECEMBER 31, 1977⁸²

Storage Area	V o l u m e		Curies	Box	Barrel	Bin	G r a m s		
	m ³	ft ³					PU	Am-241	U-233
<u>Retrievably Stored - Contact-Handled TRU</u>									
TSA #1 ¹ 10/70-10/75	27,450	969,260	120,900	4,241	64,519	83	148,400	11,290	40,590
TSA #2 10/75-12/77	4,583	161,825	44,390	787	8,728	78	49,480	2,000	15,040
TSAR1 ² 1/77-12/77	2,006	70,832	9,469	--	9,378	11	12,140	946	--
TOTALS	34,039	1,201,917	174,759	5,028	82,625	172	210,020	14,466	55,630
<u>Retrievably Stored - Remote-Handled TRU</u>									
11 TSF ³ 11/76-12/77	9	304	54	--	76	--	19	--	--
<u>Subsurface Disposal TRU</u>									
SDA ⁴ 1954-10/70	59,522	2,102,000	191,000	6,042	182,250	12,783 ⁶	344,000	--	--
TDA ⁵	7,190	253,800	3,494	1,243	15,000	--	11	--	--

¹ Transuranic Storage Area - 20 year retrievable storage² Transuranic Storage Area - Retrieved waste from subsurface disposal³ Intermediate Level Transuranic Storage Area - Intermediate gamma TRU waste⁴ Subsurface Disposal Area - Shallow land burial TRU wastes⁵ Transuranic Disposal Area - Pad disposal of 10nCi/gm TRU⁶ Cartons

TABLE F-II
TRU WASTE AT INEL
AS OF OCTOBER 1, 1984⁸²

	<u>Cubic Meters</u>		<u>Cubic Feet</u>
<u>Retrievably Stored - Contact-Handled TRU</u>			
As of 12/31/77	34,039	or	1,201,917
Projection thru 1984	<u>23,641</u>	or	<u>834,766</u>
Totals	57,680	or	2,036,682

<u>Retrievably Stored - Remote-Handled TRU</u>			
As of 12/31/77	9	or	318
Projection thru 1984	<u>400</u>	or	<u>14,124</u>
Totals	409	or	14,442

<u>Subsurface Disposal TRU</u>			
As of 12/31/77	59,522	or	2,102,000
Projection thru 1984	<u>0</u>		<u>0</u>
Totals	59,522	or	2,102,000

TABLE F-1J
ISOTOPIC COMPOSITION
OF TRU NUCLIDES IN TSA WASTE⁸²

<u>Nuclide</u>	<u>Weight (%)</u>
Am-241	5.15
Pu-238	0.34
Pu-239	69.57
Pu-240	4.36
Pu-241	0.30
Pu-242	0.01
U-233	20.27

TABLE F-1K
AVERAGE WEIGHT OF TRU WASTE CONTAINERS⁸²

	D R U M S			B O X E S		
	<u>No. of Drums</u>	<u>Weight (lbs)</u>	<u>Average Weight</u>	<u>No. of Boxes</u>	<u>Weight (lbs)</u>	<u>Average Weight</u>
1970*	9,378	1,787,825	190.6			
1971*	2,726	871,646	319.6	562	1,205,060	2,183.1
1972	15,690	5,641,154	363.2	975	3,063,110	3,141.7
1973	9,097	3,000,723	329.9	944	2,813,612	2,980.0
1974	6,860	2,444,782	356.9	774	2,006,220	2,692.0
1975	8,782	3,261,068	371.3	613	1,316,289	2,566.9
1976	4,279	1,596,536	372.4	492	1,359,950	2,764.1
1977	<u>3,464</u>	<u>1,471,801</u>	<u>424.9</u>	<u>514</u>	<u>1,415,634</u>	<u>2,754.2</u>
	60,266	19,975,565	331.5 (45 lbs/ft ³)	4,764	13,179,875	2,766.6, (25 lbs/ft ²)

* Partial year's data.

TABLE F-1L
COMBUSTIBILITY AND COMPACTIBILITY - TSA WASTE⁸²
(Does not include retrieved wastes)

	<u>9/71 - 12/77</u>				
	<u>Total</u>	<u>Comp Comb</u>	<u>Comp NComb</u>	<u>NComp Comb</u>	<u>NComp NComb</u>
<u>Unit Count</u>					
Drums	48,917	15,677	1,190	408	31,842
Boxes	4,766	404	423	205	3,734
Bins	161	160	1	--	--
<u>Volume (m²)</u>					
Drums	10,374	3,325	252	87	6,710
Boxes	14,849	1,259	1,318	639	11,633
Bins	547	544	3	--	--
TOTAL	25,770	5,128	1,573	726	18,343
Percent		19.9	6.1	2.8	71.2

TABLE F-1M

TRANSURANIC STORAGE AREA DATA
 SEPTEMBER 1971, THROUGH DECEMBER 1976
 TABULATED BY CONTENT CODE⁸²

Content Description	Drums	Volume Cu. Ft.	Boxes	Volume Cu. Ft.	Weight lbs.	PU Grams	AM Grams
Not Recorded - Unknown	1,903	21,088			392,805	3,241	32
First Stage Sludge	4,957	37,821			2,537,489	26,224	10,249
Second Stage Sludge	6,472	48,842			3,469,429	1,523	16
Organic Set Ups (Oil Solids)	3,366	27,581			1,784,055	1,837	0
Special Set Ups (Cement)	851	6,812			508,472	910	7
Evaporated Salts	12	107	1	112	10,692	6	7
Combustibles (Rags, Gloves, Poly)	865	6,623			164,845	-0-	-0-
Non-compressible, Non-combust.	777	5,762			184,474	-0-	-0-
Solidified Grinding Sludge, Etc.	41	305			9,880	-0-	-0-
Solid Binary Scrap Powder, Etc.	12	88			2,950	-0-	-0-
Dirt	135	993			83,535	-0-	-0-
Sludge	23	169			6,800	-0-	-0-
Alpha Hot Cell Waste	40	160			3,674	16	-0-
American Process Residue	120	897			43,997	150	-0-
Sludge, Filter	1	7			145	14	-0-
Cemented Sludge	53	537			19,072	1,061	-0-
Graphite	758	5,619			197,179	6,274	-0-
Graphite Cores	32	235			8,327	405	-0-
Benelex and Plexiglas	16	118	16	1,792	63,728	67	-0-
Graphite Scarfings	16	118			3,827	81	-0-
Graphite Heels	4	41	1	112	3,500	783	-0-
Tantalum	192	1,412			48,365	2,372	18
Paper and Rags - Dry	4,945	36,835	123	16,176	1,576,644	2,662	91
Filters, Absolute (8x8)	110	809			16,912	215	7
Paper and Rags - Moist	7,293	53,738	8	896	1,455,248	2,212	11
Plastics, Teflon, Wash, Pvc	1,832	13,625	9	1,000	333,056	1,145	39
Insulation & CMS Filter Media	251	1,860	78	8,736	195,778	6,562	0
Leaded Rubber Gloves & Aprons	509	3,743			172,042	14,025	16
Insulation	219	1,761	1	112	36,138	217	0
Insulation Heel	1	11			411	199	0
Crucible, Lead	30	221			11,448	91	0
Brick, Fire	886	6,519	24	2,688	387,140	2,789	0
Grit	5	17			2,220	21	0
Blacktop Concrete Dirt and Sand	917	6,890	81	1,072	669,417	647	0

TABLE F-1M

(CONT.)

<u>Content Description</u>	<u>Drums</u>	<u>Volume Cu. Ft.</u>	<u>Boxes</u>	<u>Volume Cu. Ft.</u>	<u>Weight lbs.</u>	<u>PU Grams</u>	<u>AN Grams</u>
Oil Dirt Residues From Incinerator	11	81			4,209	10	0
Cement Insul. & Filter Media	206	1,515	2	224	56,971	5,253	17
Crucible and Sand	1	7			282	35	0
Sand, Slag and Crucibles	6	67			2,700	1,164	0
Sand, Slag, and Crucible Neels	8	59			1,707	1,468	0
Electrorefining Salt	2	15			476	24	0
Ash, Incinerated (Virgin)	8	59			3,212	359	0
Soot	13	96			2,826	702	0
Resin, *On Column Unleached	29	266			11,528	2,716	0
Resin, Leached	6	59			2,389	263	0
Resin, Leached and Cemented	139	1,022			40,500	2,964	21
Glass	761	5,881	1	112	190,594	3,841	16
Raschig Rings, Unleached	1,096	8,060			215,924	11,562	0
Raschig Ring, Leached	22	166			6,545	46	0
Washables, Rubber, Plasticu	6	67			2,813	81	0
Gloves, Drybox	93	510			19,533	759	0
Plexiglass and Benelox	48	364			12,971	90	0
Metal Scrap (Non SS)	1,669	12,718	2,589	289,702	7,981,075	27,319	43
Metal, Leached (Non SS)	457	3,361	1	112	141,841	13,531	5
Filters CWS	58	460	466	52,192	886,546	5,548	13
Equipment Boxes			12	1,344	12,687	88	8
High Level Acid	235	1,728			75,815	17	0
High Level Caustic	691	5,081			229,878	20	0
High Level Sludge/Cement	1,998	14,692			1,260,952	7	0
16 nCi/gm Non-Combustible	1	7			335	0	0
Contaminated Soil			36	4,032	160,002	1	0
LSA 100 nCi/gm Combustible	103	757			23,168	0	0
LSA 100 nCi/gm Non-Combustible	110	609			22,782	0	0
LSA Paper, plastics, Etc.	352	2,611	6	672	82,244	1	0
LSA Metal, Glass, Etc.	110	809	334	37,492	918,936	68	0
Concrete, Asphalt, Etc.	704	5,233	171	10,426	1,022,373	326	0
Wood	24	176	54	6,055	123,892	467	0
Bldg. 776 Process Sludge	5	37	19	2,128	89,887	23	0
Laundry Sludge			11	1,232	46,980	43	0
Equipment	1	7			178	11	0
Dirt	470	3,456			255,463	0	0
Sludge	296	2,177	8	896	176,751	64	0
TOTALS	47,404	363,658	4,252	467,323	28,492,732	154,559	10,606

Tables F-1N, F-1O and F-1P present the radionuclide distribution for the various stored and buried wastes over the time period 1971 through 1977 while Table F-1Q presents radionuclide content of the waste over the period 1978 through 1984. For buried waste, the data are listed both at the time of emplacement (1954 through 1970) and as decayed to 1985. For stored waste, values at the time of receipt are given for waste placed on the TSA through 1977 and for the total inventory projected to 1985. The estimates and calculations for the projected inventory take into account both radioactive decay and an assumption of increased waste compaction in the future by the waste generators.⁷⁷ No detailed listing is available for the quantities of radionuclides present in the beta-gamma waste that is intermixed with the buried TRU waste. Table F-1R gives the distribution of beta-gamma radionuclides in the beta-gamma waste currently being disposed in the SDA. This distribution is presented as an example of what the distribution might have been in the pre-1970 buried beta-gamma waste at the time of disposal. Decay adjustment of the distribution to 1985 is also given in Table F-1R.

Table F-1S is a summary of the inventory data available on the INEL complex. The data for TRU waste presently in retrievable storage are the container volumes and are considered to be accurate within $\pm 10\%$. The subsurface-disposed TRU quantities are container volumes, based on tabulations of containers shipped, and do not reflect a review of waste shipment records. The disposed volume probably is accurate within $\pm 30\%$. However, due to container degradation, the mixing of waste with soil, and the TRU waste generation associated with retrieval operations, the total TRU retrieved volume may be a factor of 2 to 3 larger than the original waste volume (expected to be a factor of 1.5 or greater).

TABLE F-1N
NUCLIDE DISTRIBUTION - TSA STORED WASTE (CURIES)⁸²

Nuclide	1971	1972	1973	1974	1975	1976	1977	Total	Fraction	$\Sigma > 1$
Ac-227	-0-	-0-	-0-	-0-	7.290E-02	-0-	-0-	7.290E-02	4.273E-07	-
Am-241	6.630E+03	1.193E+04	7.351E+03	4.574E+03	5.600E+03	3.722E+03	5.936E+03	4.574E+04	2.601E-01	26.0
Am-243	-0-	1.000E-04	-0-	-0-	9.250E-04	6.117E-05	-0-	1.006E-03	6.366E-09	-
Ce-144	-0-	-0-	-0-	-0-	-0-	4.200E-01	-0-	4.200E-01	2.462E-06	-
Cf-252	-0-	-0-	-0-	-0-	1.342E-04	-0-	-0-	1.342E-04	7.066E-10	-
Cm-244	-0-	6.000E-01	1.112E+03	-0-	4.054E-01	2.640E-02	-0-	1.112E+03	6.574E-03	.6
Co-58	-0-	-0-	8.300E-01	-0-	-0-	-0-	-0-	8.300E-01	4.065E-06	-
Co-60	-0-	-0-	6.200E+00	-0-	-0-	-0-	-0-	6.200E+00	3.634E-05	-
Cr-51	-0-	-0-	1.450E+00	-0-	-0-	-0-	-0-	1.450E+00	8.499E-06	-
Cs-137	2.468E+00	-0-	3.000E+00	-0-	1.007E-01	4.200E-01	-0-	6.076E+00	3.562E-05	-
Eu-152	-0-	-0-	-0-	-0-	1.690E-01	-0-	-0-	1.690E-01	9.966E-07	-
H-3	-0-	-0-	-0-	-0-	-0-	0.541E-06	-0-	0.541E-06	5.006E-11	-
HAP	-0-	-0-	0.495E-02	-0-	-0-	-0-	-0-	0.495E-02	4.979E-07	-
Ir-192	2.743E-01	1.000E-01	3.750E-01	2.360E-01	1.717E-01	1.897E-01	1.626E-01	7.475E-01	4.307E-04	-
Mn-54	-0-	-0-	1.400E-01	-0-	-0-	-0-	-0-	1.400E-01	0.206E-07	-
Mn-237	-0-	6.445E-04	-0-	-0-	1.210E-04	-0-	7.050E-06	7.732E-04	4.532E-07	-
Pm-147	-0-	-0-	-0-	-0-	4.640E-02	-0-	-0-	4.640E-02	2.720E-03	-
Pu-238	2.314E+01	4.219E+01	5.030E+01	5.797E+01	2.462E+01	4.732E+02	3.301E+04	3.632E+04	2.409E-02	2.4
Pu-239	7.747E+02	1.641E+03	1.726E+03	1.921E+03	2.514E+03	0.995E+02	2.220E+03	1.177E+04	6.070E-02	6.9

TABLE F-IN

(CONT.)

Isotope	1971	1972	1973	1974	1975	1976	1977	Total	Fraction	% > 1
Pu-240	1.902E+02	4.029E+02	4.252E+02	4.718E+02	6.185E+02	2.144E+02	5.331E+02	2.056E+03	1.674E-02	1.6
Pu-241	6.294E+03	1.367E+04	1.390E+04	1.639E+04	2.060E+04	5.611E+03	1.539E+04	9.194E+04	5.389E-01	53.9
Pu-242	1.075E-02	2.213E-02	2.341E-02	3.859E-02	4.702E-02	1.743E-02	4.643E-02	2.859E-01	1.286E-06	-
Pu-238	-0-	-0-	-0-	-0-	2.750E-01	-0-	-0-	2.750E-01	1.617E-06	-
Pu-239	-0-	-0-	-0-	8.000E-01	-0-	-0-	-0-	8.000E-01	4.649E-06	-
Sr-90	-0-	-0-	-0-	-0-	-0-	3.000E-01	-0-	3.000E-01	1.748E-06	-
Tc-99	-0-	-0-	-0-	-0-	1.399E-03	-0-	-0-	1.399E-03	8.140E-09	-
U-232	-0-	-0-	1.202E-03	5.665E-02	6.123E-02	4.701E-02	5.806E-03	1.670E-01	9.769E-07	-
U-233	-0-	-0-	-0-	3.463E+00	3.517E+00	1.999E+00	2.644E-01	9.279E+00	5.439E-05	-
U-235	-0-	1.000E-02	2.567E+00	2.000E+02	1.959E+02	1.172E+02	1.475E+01	5.267E+02	1.837E-03	1
U-238	-0-	-0-	1.099E-04	1.658E+00	1.624E+00	-0-	-0-	3.321E+00	1.953E-05	-
U-236	1.778E-05	-0-	2.399E-04	1.141E-04	1.876E-04	1.446E-04	5.902E-04	1.293E-03	7.519E-09	-
U-234	-0-	-0-	1.750E-06	2.639E-04	2.666E-04	-0-	-0-	5.373E-04	3.120E-09	-
U-230	1.312E-06	1.465E-05	4.040E-05	1.633E-05	1.363E-01	2.740E-04	2.473E-04	1.970E-01	1.155E-08	-
Th-230 & 232	-0-	-0-	1.408E+00	-0-	-0-	-0-	-0-	1.408E+00	8.286E-06	-
TOTAL	1.392E+04	2.769E+04	2.660E+04	2.365E+04	3.250E+04	1.105E+04	3.712E+04	1.706E+05		

TABLE F-10
NUCLIDE DISTRIBUTION - DISPOSED WASTE (CURIES)⁸²

nuclide	1971	1972	1973	1974	1975	1976	1977	Total	Fraction	$\Sigma > 1$
Ag-110M	2.330E-02	-0-	-0-	-0-	-0-	2.100E-01	5.960E-02	3.010E-01	1.520E-02	-
Am-241	-0-	1.000E-05	-0-	-0-	1.050E-07	-0-	3.240E-07	1.051E-05	5.300E-12	-
Ba-133	-0-	-0-	-0-	-0-	-0-	4.000E-07	3.400E-08	4.340E-07	2.197E-13	-
Ce-144	2.022E+00	8.629E+00	-0-	-0-	1.430E+00	4.360E+01	-0-	5.574E+01	2.014E-05	-
Ce-139	4.290E+00	-0-	1.000E+01	-0-	-0-	-0-	-0-	1.429E+01	7.217E-06	-
Cl-36	-0-	-0-	-0-	-0-	-0-	3.930E-08	-0-	3.930E-08	1.975E-14	-
Co-60	-0-	-0-	-0-	-0-	-0-	2.076E-07	-0-	2.076E-07	1.449E-13	-
Co-109	-0-	-0-	-0-	-0-	-0-	-0-	1.100E-06	1.100E-06	5.960E-13	-
Ce-141	2.944E+01	2.789E+04	2.953E+00	1.217E+00	1.095E+00	2.002E+01	1.316E+01	2.797E+04	1.417E-02	1.4
Ce-144	5.465E+01	1.093E+01	4.130E+02	8.047E+02	9.517E+02	3.923E+02	2.022E+03	4.725E+03	2.304E-03	0.2
Cr-51	-0-	-0-	-0-	-0-	-0-	3.970E-08	-0-	3.970E-08	2.005E-14	-
Cr-57	-0-	-0-	-0-	-0-	-0-	-0-	1.110E-06	1.110E-06	5.555E-13	-
Cr-58	8.477E+00	6.108E+00	3.900E+03	2.079E+00	3.154E+01	3.000E+01	1.061E+05	1.100E+05	5.554E-02	5.5
Cr-60	4.610E+01	1.434E+02	2.002E+05	7.462E+03	7.205E+03	4.610E+03	6.717E+04	3.828E+05	1.994E-02	19.8
Cr-51	2.359E+05	5.152E+03	8.003E+04	8.170E+01	2.234E+02	1.517E+05	5.347E+05	9.973E+05	4.904E-01	49.9
Cs-134	2.119E+00	1.288E+00	5.241E+01	6.036E+01	4.612E+01	5.713E+00	1.706E+01	1.937E+02	9.759E-05	-
Cs-137	4.005E+02	2.700E+02	8.947E+02	1.424E+03	5.697E+02	4.724E+02	6.683E+02	4.760E+03	2.174E-01	.2
Eu-152	-0-	-0-	-0-	-0-	1.060E+00	4.30E-01	1.630E+02	1.645E+02	8.300E-05	-
Eu-154	-0-	-0-	-0-	4.192E+01	3.523E+01	5.000E-01	7.656E+01	1.543E+02	7.793E-05	-

TABLE F-10
(CONT.)

	1971	1972	1973	1974	1975	1976	1977	Total	Fraction	$\Sigma > 1$
W-190										
W-240	9.000E-04	1.775E-03	6.975E-02	1.312E-01	9.430E-01	1.360E-01	1.026E-02	2.247E-01	1.140E-07	-
W-241	1.600E-04	6.062E-02	3.190E-01	4.606E+00	3.227E-01	3.568E-02	2.005E-02	5.414E+00	2.730E-06	-
W-242	-0-	9.777E-08	5.557E-02	1.000E-05	7.709E-07	1.065E-07	6.979E-08	1.240E-05	6.300E-12	-
W-225	5.576E+00	1.250E+00	2.293E-01	1.000E+00	3.302E-04	2.021E-01	-0-	8.250E+00	4.120E-06	-
W-06	-0-	4.210E+00	4.150E+00	5.322E+01	5.853E+00	-0-	-0-	7.743E+01	3.931E-05	-
W-101	2.570E-02	9.550E+03	1.076E+00	9.640E-02	2.033E+00	0.020E-02	2.900E-02	9.561E+03	4.029E-03	.5
W-106	8.307E+00	1.093E+01	1.999E+02	3.279E+02	2.614E+02	1.350E+02	1.530E+02	1.106E+03	5.506E-04	-
W-125	1.921E+01	1.915E+00	8.912E+01	1.074E+02	7.327E+03	6.260E+01	6.030E+01	4.200E+02	2.164E-04	-
W-46	4.940E+00	3.121E+00	-0-	-0-	-0-	-0-	-0-	8.060E+00	4.075E-06	-
W-151	-0-	-0-	-0-	-0-	-0-	3.302E+00	-0-	3.302E+00	1.677E-06	-
W-09	-0-	-0-	-0-	-0-	-0-	1.000E-01	-0-	1.000E-01	5.001E-08	-
W-90	1.670E+01	2.103E+01	1.072E+02	1.570E+01	1.764E+01	2.017E+02	2.546E+02	4.104E+01	2.070E-03	.2
W-102	-0-	-0-	-0-	-0-	-0-	7.310E-02	-0-	7.310E-02	3.602E-08	-
W-29	-0-	-0-	-0-	-0-	-0-	3.961E-02	3.200E-02	3.590E-02	2.017E-11	-
W-270	-0-	-0-	-0-	-0-	5.403E-02	-0-	-0-	5.403E-02	2.200E-15	-
W-212	2.100E-04	1.070E-07	5.450E-05	4.605E-05	1.001E-02	9.610E-08	3.664E-04	1.150E-02	5.005E-09	-
W-232	-0-	-0-	-0-	-0-	8.100E+00	-0-	-0-	8.100E+00	4.272E-06	-
W-233	-0-	6.000E-06	-0-	1.010E+00	-0-	9.527E-01	-0-	9.531E-01	4.015E-19	-
W-234	3.210E-04	-0-	1.057E-03	1.174E-05	1.760E-04	6.170E-04	2.070E-02	2.001E-02	1.465E-08	-

TABLE F-10
(CONT.)

Module	1971	1972	1973	1974	1975	1976	1977	Total	Fraction	X > 1
J-235	1.010E-01	1.072E-01	6.430E-02	6.401E-02	7.012E-02	2.74E-02	4.500E-02	4.020E-01	2.490E-07	-
J-236	1.564E-04	-0-	-0-	-0-	2.200E-06	3.170E-05	2.049E-04	2.405E-04	1.215E-10	-
J-238	7.735E+00	0.274E+02	4.723E+03	4.075E+00	5.211E+00	1.710E+00	3.147E+00	3.575E+01	1.006E-05	-
M-14-Alpha	-0-	3.820E-01	4.360E-01	1.604E-02	1.701E-03	-0-	-0-	7.560E-01	2.021E-07	-
M-14-B & G	2.060E+03	1.702E+04	0.000E+01	2.002E+01	1.070E+01	1.310E+02	0.995E+00	2.094E+04	1.020E-02	1.0
P-1M7	5.290E+00	-0-	-0-	-0-	-0-	-0-	-0-	5.290E+00	2.670E-06	-
P-0-65	4.272E+00	-0-	-0-	3.655E+02	4.000E-01	1.701E+00	5.960E-02	3.669E+02	1.053E-04	-
P-0-55	5.130E-02	1.374E+05	1.450E+01	-0-	2.310E+00	3.199E+00	5.295E+01	1.375E+05	6.944E-02	6.9
P-0-MU-95	3.554E+04	2.736E+00	1.237E+02	1.516E+02	1.142E+02	0.912E+01	1.000E+01	3.603E+04	1.020E-02	1.0
TOTAL CITIES	3.509E+05	2.147E+05	3.399E+05	1.032E+04	1.319E+04	2.100E+05	0.241E+05	1.900E+05		

TABLE F-1P
NUCLIDE DISTRIBUTION - ILTSF (CURIES)⁸²

Nuclide	1971	1972	1973	1974	1975	1976	1977	Total	Fraction	$\Sigma > 1$
Eu-155	-0-	-0-	-0-	2.422E+01	2.154E+01	9.000E-02	2.001E+01	6.586E+01	3.326E-05	-
Fe-57	2.935E+04	6.701E+02	5.918E+02	1.440E+03	5.630E+02	1.865E+04	6.319E+04	1.145E+05	5.703E-02	5.0
H-3	5.000E-01	-0-	2.360E-01	-0-	-0-	-0-	-0-	7.300E-01	3.687E-07	-
Hf-161	2.805E-01	-0-	-0-	-0-	4.002E-02	7.310E-02	5.960E-02	4.620E-01	2.333E-07	-
I-131	1.569E+00	-0-	-0-	-0-	8.167E-03	8.800E-01	5.500E+00	7.957E+00	4.019E-06	-
I-130	3.533E-01	3.775E-01	-0-	-0-	2.053E+00	5.890E+01	-0-	6.148E+01	3.115E-05	-
MAP	2.219E+00	1.003E+01	2.309E+04	9.224E+01	1.367E+01	3.074E+01	1.009E+02	2.339E+04	1.179E-02	1.2
MFP	6.769E+01	3.037E+01	1.292E+02	7.614E+02	2.866E+02	2.133E+02	2.613E+02	1.750E+03	6.939E-04	-
Ni-55	1.241E+00	4.370E+00	2.081E+04	7.201E+01	3.669E+02	2.973E+01	7.393E+04	9.519E+04	4.809E-02	4.0
Sn-55	1.485E+06	2.600E+01	-0-	-0-	5.538E+02	-0-	-0-	5.813E+02	2.936E-04	-
Na-22	-0-	-0-	-0-	-0-	-0-	-0-	1.160E-06	1.160E-06	5.659E-13	-
Na-23	3.537E+00	-0-	-0-	-0-	-0-	2.499E+01	-0-	2.853E+01	1.441E-05	-
Si-95	3.724E+00	-0-	-0-	3.196E-01	3.096E-01	2.671E+00	5.503E+01	6.205E+01	3.134E-05	-
Bi-55	5.000E+02	1.299E+01	3.931E+02	1.200E+03	-0-	-0-	-0-	5.992E+03	3.026E-03	1.1
Np-237	-0-	-0-	6.345E-07	-0-	-0-	4.700E-06	-0-	4.835E-06	2.462E-12	-
Pm-147	-0-	-0-	7.400E-01	-0-	-0-	-0-	-0-	7.400E-01	3.737E-07	-
Po-210	1.100E-01	-0-	-0-	-0-	-0-	-0-	-0-	1.100E-01	5.564E-08	-
Po-238	-0-	2.101E-04	1.231E-03	1.679E-02	1.183E-01	1.052E-03	2.287E-03	2.492E-03	1.259E-07	-
Pu-239	1.842E-01	8.807E-03	2.467E-01	5.355E-01	1.275E-01	8.660E-03	1.555E-01	1.267E+00	6.474E-07	-

TABLE F-1P
(CONT.)

Isotope	1971	1972	1973	1974	1975	1976	1977	Total	Fraction	$\Sigma > 3$
W-P	-0-	-0-	-0-	-0-	-0-	1.89E+01	2.84E+01	4.73E+01	9.44E-01	94.4
Pu-238	-0-	-0-	-0-	-0-	-0-	-0-	5.19E-02	5.19E-02	9.03E-03	.1
Pu-239	-0-	-0-	-0-	-0-	-0-	2.52E+00	1.47E-01	2.67E+00	5.33E-02	5.3
Pu-240	-0-	-0-	-0-	-0-	-0-	-0-	6.13E-04	6.13E-04	1.22E-05	-
Pu-241	-0-	-0-	-0-	-0-	-0-	-0-	5.97E-02	5.97E-02	1.14E-03	.1
Pu-242	-0-	-0-	-0-	-0-	-0-	-0-	4.47E-06	4.47E-06	8.91E-08	-
U-233	-0-	-0-	-0-	-0-	-0-	-0-	1.73E-04	1.73E-04	3.45E-06	-
U-235	-0-	-0-	-0-	-0-	-0-	-0-	4.24E-05	4.24E-05	8.47E-07	-
TOTAL	-0-	-0-	-0-	-0-	-0-	2.14E+01	2.86E+01	5.01E+01		

TABLE F-10

RADIONUCLIDE CONTENT OF BURIED AND STORED TRU WASTE⁷⁷

Transuranic Radionuclide	Buried Waste*				Stored Waste			
	Emplaced		As of January 1, 1985		Emplaced (through Dec. 31, 1977)		As of January 1, 1985	
	Curies	Percent	Curies	Percent	Curies	Percent	Curies	Percent
Americium-241	48,007	18.98	46,548	37.06	45,740	26.90	133,284	31.91
Plutonium-238	571.4	0.22	462	0.37	16,120	9.48	44,223	10.59
Plutonium-239	21,043	8.32	19,789	15.76	11,720	6.89	34,230	8.19
Plutonium-240	4,860	1.92	4,564	3.63	2,856	1.68	8,071	1.93
Plutonium-241	178,425	70.55	54,234	43.18	91,940	54.08	194,092	46.46
Plutonium-242	0.2	0.01	0.2	0.01	0.2	0.01	0.2	0.01
Uranium-233	0.5	0.01	0.5	0.01	527	0.31	1,486	0.36
Curium-244	0	0	0	0	1,113	0.65	2,343	0.56
Total Radioactivity	252,907	100	125,598	100	170,016	100	417,729	100
Total Weight of TRU, kg	381		356.9		280.116		787.140	
Total Waste Volume, m ³	62,300		56,600		34,000		56,600	
Transuranic Activity, Ci/m ³	4.06		2.28		4.9		7.4	

*Does not include the beta-gamma waste present.

TABLE F-1R
ESTIMATED DISTRIBUTION OF RADIONUCLIDES
IN THE BETA-GAMMA WASTE⁷⁷

<u>Radionuclide</u>	<u>Half-Life</u>	<u>Percent at Time of Disposal</u>	<u>Percent at Beginning of 1985</u>
Chromium-51	27.7 d	49.9	0
Cobalt-60	5.27 yr	19.8	50
Zirconium-95	64 d	6.9	0
Iron-59	44.6 d	5.8	0
Cobalt-58	70.8 d	5.6	0
Manganese-54	312.5 d	4.8	0
Zirconium-Niobium-95	64 d-3.5 d	1.8	0
Cerium-141	32.5 d	1.4	0
Ruthenium-103	39.4 d	0.5	0
Nickel-59	8×10^4 yr	0.3	15
Cerium-144	284.4 d	0.2	0
Cesium-137	30.17 yr	0.2	10
Strontium-90	29 yr	0.2	10
Ruthenium-106	368 d	0.1	0
Others Identified	3 yr	0.1	0
Mixed Activation Products	5 yr	1.2	3
Mixed Fission Products	30 yr	0.1	10
Unidentified Beta-Gamma	3 yr	<u>1.1</u>	<u>2</u>
		100.0	100.0

TABLE F-15
SUMMARY OF TRU WASTE
BURIED/STORED AT INEL⁵

	<u>Previous Years</u>	<u>1977</u>	<u>1978</u>	<u>Total</u>
BURIED TRU WASTE				
Volume (m ³)	59,530 ± 30%	0	0	59,530 ± 30%
Kg TRU	361	0	0	361
STORED TRU WASTE				
Volume (m ³)	31,130 ± 10%	2,921	1,607*	35,658 ± 10%*
Kg Pu	167	42	23.9*	296.3*
Kg other TRU	64			

* These are values reflecting as generated quantities of TRU wastes in FY 1978. Due to inventory accumulation/reduction in FY 1978, 4,496 m³ (44.2 kg) of TRU waste were buried/stored resulting in an apparent total of 38,546 m³ (317.2 kg).

2.0 LOS ALAMOS SCIENTIFIC LABORATORY (LASL)

2.1 INTRODUCTION^{5,90}

The Los Alamos Scientific Laboratory is located in north-central New Mexico on the Pajarito Plateau, west of the Rio Grande, on the eastern slopes of the Jemez Mountains. LASL has used shallow land burial for disposal of solid radioactive waste since the mid-1940s. The bulk of the material consists of room-generated trash. A variety of special disposal operations has been performed, ranging from the shaft burial of a cylinder containing a few grams of tritium-contaminated solution to the demolition and burial of entire buildings. Currently, there are three active locations (areas A, G, and T) for disposal at LASL.⁹⁰ All solids contaminated with radioactivity are buried on-site at LASL. Contaminated aqueous liquids are treated to produce solids for on-site storage. Contaminated organic liquids are either burned (low-activity level) or jellied (high-activity level) to form a "solid" for on-site storage.

Waste generated at LASL is packaged to provide safe handling and transport to the disposal site.* With the exception of tritium wastes, the packaging is not intended to provide containment after burial. Most of the low-activity trash-type waste is compacted and baled prior to burial.

"Room-trash"-type wastes generated at the LASL DP-West plutonium facility are tested for radioactivity and properly segregated for burial or storage. This same practice is followed for similar types of waste generated at the LASL R&D facilities where transuranic material is worked with. Distinctively hazardous and radioactive wastes, such as contaminated laboratory chemicals and reactive metals, are properly packaged and disposed of in shafts and are immediately covered.

* It should be noted that DOT criteria for waste packaging are not applicable to on-site transport.

2.2 PLUTONIUM METAL FABRICATION WASTE^{26,27,39,71-73,79}

Plutonium-contaminated liquid wastes are processed at two widely separated facilities. One of these, Technical Area 50 (TA-50), receives wastes from several technical areas on the south side of Los Alamos Canyon. The wastes are effluents from radiochemistry laboratories, general chemistry laboratories, accelerator facilities, research reactors, shops and plating rooms. Gross alpha activity level of the waste is in the range of 1×10^{-5} to 1×10^{-3} $\mu\text{Ci/ml}$. Almost all of the alpha activity is plutonium with 80 to 90 percent being the Pu-238 isotope. The high concentration of Pu-238 is a development of recent origin.⁷⁹

The wastes arrive at the treatment facility through a special sewer system installed for the purpose. Where necessary, the wastes are neutralized at the source to permit use of standard materials of construction, cast iron for force mains and clay tile for gravity lines.

The other waste management plant is located at a technical area, TA-21, where research and development programs involving plutonium are a major part of the work. Plutonium fuel materials are synthesized and their properties are studied; plutonium metal, alloys and compounds are produced and fabricated; plutonium-239 is recovered and recycled; and refined plutonium-238 is prepared and evaluated for its use as a power source.

Radioactivity levels and types of waste solutions at the TA-21 plant necessitate use of a variety of management methods. The main waste stream, largely cleanup and decontamination solutions, is delivered to the treatment plant via a separate sewer line of cast-iron construction. Plutonium activity level is usually less than 1×10^{-3} $\mu\text{Ci/ml}$ with the Pu-239 isotope representing about 80 percent of the total.

Other waste streams at TA-21 are delivered through stainless steel lines, in stainless steel tank trailers or in smaller plastic or metal containers. The wastes are usually of a much higher level of plutonium activity than the sewered wastes

and include solutions of a chemical nature not compatible with the chemical treatment methods employed for the main waste stream, e.g., alcohol and vacuum pump oil. The waste delivered in the tank trailers is a highly acid solution containing milligram per liter quantities of americium-241 and plutonium-239.

Approximate annual volumes of the wastes processed are shown in Table F-2A. These and other wastes are generated as a result of work performed at LASL as briefly described above. The following discussion will identify the waste forms and characteristics of these wastes.

2.2.1 PLUTONIUM CHEMISTRY AND METALLURGY WASTES²⁶

The Plutonium Chemistry and Metallurgy Group operations in Technical Area (TA)-21 at LASL include all unit operations involved in Pu-239 metal handling and a complete scrap recovery system. Figure F-1 describes a typical plutonium metal cycle and some of the major process residues handled by scrap recovery. Other operations include basic plutonium chemistry research, development work in Liquid Metal Fast Breeder Reactor (LMFBR) fuels, and development work using 80 percent Pu-238 as an energy source for Space Nuclear Systems and pacemakers thus providing a very broad spectrum of wastes.

The radioactive waste generated consisted of two major streams: residues generated by process operations in the glovebox or hood enclosures and residues generated in the operating room or area containing the transuranic process facilities. Packaged residues are normally sent to scrap recovery for measurement of the plutonium content or transuranic activity by nondestructive assay (NDA) techniques.

Residues are considered recoverable if the plutonium content is sufficient to warrant reclaiming and reusing, based on local criteria. Such packages of measured residues can then be defined as scrap or feed material for a scrap recovery process. Packaged residues not considered recoverable, but which are

TABLE F-2A
ANNUAL QUANTITIES OF PU-CONTAMINATED LIQUID
WASTES, LOS ALAMOS⁷⁹

TA-50 Liters of Waste	TA-21 Liters of Waste			
	<u>Main Stream</u>	<u>Am-PU</u>	<u>Alcohol</u>	<u>Oil</u>
56.5×10^6	11.8×10^6	76,000	340	120

FIGURE F-1
PLUTONIUM METAL FABRICATION CYCLE²⁶

above 10 nCi/g in transuranic activity, are defined as retrievable waste and are logged into a "20-yr-retrievable waste drum." Disposable residues from an area containing transuranic processing facilities are those items containing less than 10 nCi/g of transuranic activity.

Process-generated residues are usually classified as recoverable (scrap) or retrievable (waste), with a low probability of finding packages with transuranic activity less than 10 nCi/g. Room-generated residues are assumed to contain at least trace quantities of transuranic activity simply from having been in the process area, but are normally less than 10 nCi/g. A small portion could be more than 10 nCi/g, but recoverable levels would not be expected.

2.2.1.1 PROCESS-GENERATED RESIDUES FROM A TYPICAL GENERATION^{2a,2j}

Processes generating residues in the plutonium metal fabrication area include research and development work in casting, machining, welding, assembly, and disassembly, plus a variety of other experimental operations in fabrication, preparation of test specimens, and metal handling.

Process-generated residues from all operations are transferred through a conveyor system to one glovebox line for disposal. Material is removed from the glovebox by standard bag-out procedures and transferred to the scrap recovery area for assay (normally by use of a neutron coincidence counter). Packages with recoverable quantities of plutonium are transferred into scrap recovery and discardable items are logged into a retrievable waste drum.

During a typical six month operational period, and using a discard limit of 0.5 g of Pu/kg of waste (0.5 g/kg), 59 weight percent of the material (containing less than 10 percent of the plutonium) was put into retrievable waste drums. Material types, composition, and plutonium content of each stream are shown in Tables F-2B and F-2C. More detailed description of waste items observed is given in Table F-2D.

TABLE F-23

PROCESS-GENERATED RESIDUES FROM FABRICATION AREA²⁶

Material	Unsorted Waste Input			Sorted Waste Output			
	Pu g	Net Wt. kg	Wt. % of Total Waste	Retrievable Waste		Recoverable Scrap	
				(<0.5 g Pu/kg waste)		(<0.5 g Pu/kg waste)	
	Pu g	Net Wt. kg	Wt. % of Total Waste	Pu g	Net Wt. kg	Pu g	Net Wt. kg
Metal	127	166	34	29	123	98	43
Plastic	36	107	22	17	82	19	18
Rubber	23	39	8	3	17	19	23
Cellulosic	260	79	6			260	29
Glass & Ceramic	4	57	12	4	55	1	2
						99	82
Graphite	99	83	17			33	2
Floor Sweeping	33	2	1			33	2
TOTAL	582	483	100	53	284	529	199
% of TOTAL	100	100	100	9	59	91	41

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TABLE F-2C
Pu METAL FABRICATION GLOVEBOX LINE WASTE AND SCRAP²⁷

<u>Material Type</u>	<u>Composition, Wt. % of Total</u>	<u>Average Activity, nCi/g</u>
Metal (non-Pu)	35	3.1×10^4
Plastic and Rubber	19	3.5×10^4
Cellulosics	5	$63. \times 10^4$
Glass and Ceramics	15	0.8×10^4
Insulation	< 1	$< 0.5 \times 10^4$
Graphite	25	7.1×10^4
Sweepings	< 1	$108. \times 10^4$

TABLE F-2D

RESIDUES GENERATED BY PLUTONIUM METAL FABRICATION²⁶

1. (Non-Pu Scrap) Metal
Aluminum foil, tin cans, used scrap pipe (stainless steel, mild steel, and aluminum), small obsolete equipment, etc.
2. Plastic
Primarily polyvinyl chloride (PVC) from bag-in and bag-out operations, and some polyethylene and polypropylene
3. Rubber
Mostly drybox gloves
4. (Combustibles) Cellulosic
Almost totally cheesecloth
5. Glass
Broken laboratory-type-equipment such as beakers, graduated cylinders, and Vycor castings sleeves
6. Ceramics
Magnesium oxide liners and other refractories
7. Insulation
Transite board, asbestos pipe insulation
8. Graphite
Primarily molds and crucibles from casting operations

2.2.1.2 ROOM-GENERATED RESIDUES^{26,27}

Residues generated in an operation area or a room containing transuranic process facilities are usually of considerably lower contamination levels than the process-generated residues. Since the radioactivity content is not typically expected to be at the recoverable level, these residues are more commonly referred to as room-generated waste or room trash. Much of the trash is not really contaminated at all, but some waste generated during maintenance work and cleaning work around process equipment such as transfer lines might contain some contamination. Tables F-2E and F-2F present composition and TRU (primarily plutonium) activity levels.

These wastes are collected in polyethylene bags placed inside 0.06 -m³ cardboard boxes which are inside 114-l drums covered with flame-retardant lids.

Room-trash boxes are not compacted except for limited manual compression as they are being filled. Density data collected during a 4-month period showed that the boxes had an average density of 86 kg/m³.²⁷ Approximately 8 percent of the room-trash boxes contained sufficient TRU contamination to require retrievable storage.

2.2.1.3 RETRIEVABLE WASTE^{26, 27, 71-23}

The retrievable waste stream from all the LASL Plutonium Chemistry and Metallurgy Group operations, including scrap recovery, consists of all process-generated residues that are below established recoverable limits combined with any room-generated residues that assay greater than 10 nCi/g.

Certain operations, such as maintenance on process equipment--even though performed in open room areas under controlled conditions--result in TRU contamination to activity levels above 10 nCi/g.

It is generally concluded that materials from inside gloveboxes and process equipment will be above the 10 nCi/g level.²⁶

TABLE F-2E
COMPOSITION OF ROOM TRASH^{27,26}

<u>Material Type</u>		<u>Composition (Wt.%)</u>	<u>Total</u>
A.	Cellulosics		82
1.	Chart paper, computer paper, other office type paper waste, surgeons' gloveboxes and tissue	47	
2.	Brown "Kraft" paper, masking tape	5	
3.	Miscellaneous Paper	10	
4.	Cheesecloth Mostly associated with decon- tamination work and wetted with cleaning solution.	6	
5.	Clothing Booties, coveralls, caps, under- shirts, shorts, and shower room cleanup including bars of soap and floor sweepings	15	
B.	Surgeons' Gloves (both rubber and plastic)		4
C.	Plastic Polyethylene bags, reagent bottles, bags from unpacking items such as face masks.		4
D.	Styrofoam (Packing materials and coffee cups)		3
E.	Glass (Glass wool used for prefilter- ing room air exhaust, sample bottles)		1
F.	Metal (Flashlight batteries, wire, conduit, metal shavings, tin cans, knives, aerosol cans and aluminum foil)		5
TOTAL			100

TABLE F-2F
TRU ACTIVITY DISTRIBUTION
IN ROOM-GENERATED RESIDUES²⁶

<u>Activity Level</u> <u>nCi/g</u>	<u>Percent of Waste</u>
0-1	~ 79
1-10	~ 8
10-100	~ 6
100-5,000	~ 5
> 5,000	~ 1

The results obtained from the characterized retrievable waste are shown in Table F-2G, with a detailed discussion of each material type presented below. The distribution of plutonium contamination on the various waste matrices is given in Tables F-2H and F-2I.

An inventory of the chemicals used in the plutonium processing areas during one calendar year was compiled with more than 40 reagents identified. Estimates reveal that only minute quantities of these reagents enter the solid waste stream as contamination.²⁶ None of the dry bulk chemicals are pyrophoric in nature. All organic liquids which are utilized have high vapor pressures. Consequently, the probability of these compounds entering the solid waste stream during normal use and operation is remote. The inorganic bulk liquids (acids, hydroxides, caustics, etc.) are used in plutonium processing steps and are transferred to the liquid waste treatment facility. Only trace contaminant levels are expected in the solid wastes. A complete listing of chemicals used, and items which become contaminated with these chemicals, is given in Table F-2J.

Retrievable waste have been generally divided into six classes.²⁶ These are briefly discussed below.

Cellulosics

All of the cheesecloth used in glovebox operations for cleanup work is reused as long as possible before it is transferred to recovery. In many cases, the cheesecloth has been exposed to nitric acid or oil. This material is quite high in plutonium concentration and is routinely counted and transferred to the incinerator. For this reason, none of the process-generated cellulosics are in the retrievable waste category. Cheesecloth used outside the gloveboxes during maintenance work and cleanup operations contributes a significant portion of the less than 0.05 g Pu/kg waste stream.

TABLE F-2G
COMPOSITION OF RETRIEVABLE WASTE
FROM PLUTONIUM CHEMISTRY AND METALLURGY GROUP²⁶

<u>Material</u>	<u>Percent of total Weight</u>	<u>Percent of total Volume</u>	<u>g Pu/kg waste</u>	<u>Activity (nCi/g)</u>
Cellulosic	5	9	0.066	4,700
Plastic	20	39	0.157	11,200
Process solids	32	18	2.271	161,900
Metal	22	23	0.191	13,600
Glass	16	8	0.134	9,600
Rubber	5	3	0.208	14,800

TABLE F-2H
TRU ACTIVITY DISTRIBUTION IN
RETRIEVABLY GENERATED AND STORED WASTE²⁶

<u>Waste Type</u>	Percent of Waste Containing g Pu per Kg Waste								
	<u><0.05</u>	<u>0.05-0.1</u>	<u>0.1-0.15</u>	<u>0.15-0.2</u>	<u>0.2-0.25</u>	<u>0.25-0.3</u>	<u>0.3-0.35</u>	<u>0.35-0.4</u>	<u>>0.4</u>
Cellulosics	48	22	8	9	0	11	0	0	2
Plastics	16	18	21	10	7	6	5	3	4
Metal	28	20	6	10	12	18	8	0	8
Glass	28	32	17	8	1	10	2	0	2
Rubber	11	46	10	6	13	0	2	0	12

TABLE F-21

TRU ACTIVITY DISTRIBUTION
IN PROCESS SOLID WASTE STREAMS²⁶

<u>Activity Level</u> <u>(g Pu per Kg Waste)</u>	<u>Percent of Waste</u>
<0.5	8
0.51-1.0	4
1.01-1.5	20
1.51-2.0	16
2.01-2.5	32
2.51-3.0	13
3.01-3.5	4
3.51-4.0	3
>4.01	0

TABLE F-2J

PROCESS CHEMICALS CONTAMINATION OF TRU WASTE GENERATED
IN PLUTONIUM PROCESSING AREA 26

<u>Chemical</u>	<u>Area</u>	<u>Annual Issue (kg)</u>	<u>Use Description</u>	<u>Waste Items Contaminated</u>
Aluminum Nitrate	Scrap Recovery	3,182	Used to complex fluoride ions in nitrate solutions from reduction residue dissolvers, bulk fusion solutions, and in ion exchange feed adjustment.	Trace quantities may be found in exhaust filters, on dry box gloves, and on rubber window gaskets. If spilled, may be found in larger quantities on wet cheesecloth. Small amounts may also be found on cardboard transfer containers.
Ammonium Bifluoride	Metallography	0.5	May be used in combination with acids for metal etching processes. Used in quantities of 1 g or less.	Found in very low quantities in cheesecloth when spilled. Disposal may cause contamination on PVC plastic bags or on polyethylene containers.
Calcium Fluoride	Scrap Recovery	93	Used with nitric acid for the leaching of incinerator ash.	Trace quantities may be found in exhaust filters, on dry box gloves, and rubber window gaskets. Larger quantities are found on heating mantles when process solutions boil over. Residual amounts are found in empty reagent bottles.
Carbon Tetrachloride	Fabrication	47	Used with dry ice in a dew point chamber. This product is no longer used.	Evaporated in glovebox exhaust system.

TABLE F-2J

(CONT.)

<u>Chemical</u>	<u>Area</u>	<u>Annual Issue (kg)</u>	<u>Use Description</u>	<u>Waste Items Contaminated</u>
Cerous Nitrate	Metallography	0.5	Used in quantities of 1 g or less in combination with several types of acids for metal etching.	Not usually found in waste but could be found in cheesecloth if spilled and in polyethylene jar used for liquid disposal.
Chlorothene	Metallography	416 <i>ℓ</i>	Used as a lubricant for sample polishing of fuel pellets in metallography processes.	Evaporated in glovebox exhaust system.
Copper Granules	Fabrication	0.5	Used for compacting and compression testing.	Small amounts may have been found in glovebox floor sweepings if spilled.
Copper Shot	Fabrication	0.9	Used in compression and compacting testing.	Not normally found as waste contaminant, but could have been spilled and consequently be present in floor sweepings.
Diethylene triamine	Metallography	1.1	Standard epoxy catalyst used for setting up fuel pellets for metallographic processes.	Usually found in floor sweepings from grinding operation and may be found in glovebox exhaust HEPA filters.
Epoxy Cement	Metallography	6 <i>ℓ</i>	Epoxy cement used to mount samples for etching.	Found in floor sweepings and on grit paper as well as exhaust filters and in cheesecloth.
Ethanol	Metal Production	416 <i>ℓ</i>	Used to dehydrate plutonium peroxide cake prior to hydrofluorination step.	Filtered into a 50- <i>ℓ</i> glass jar contained inside a 114- <i>ℓ</i> metal drum and transported for liquid waste processing.

TABLE F-23
(CONT.)

<u>Chemical</u>	<u>Area</u>	<u>Annual Issue (kg)</u>	<u>Use Description</u>	<u>Waste Items Contaminated</u>
Ferric Nitrate	Scrap Recovery	9.1	Used in hydroxide precipitations in combination with aluminum nitrate to act as a carrier precipitator.	Residual amounts found in cardboard transfer containers and if spilled could be found in cheesecloth and floor sweepings. Particles may be found adhering to exhaust filters and dry box gloves.
Ferrous Ammonium Sulfate	Scrap Recovery	14	Used in combination with urea and hydroxylamine nitrate in the reduction of plutonium to the trivalent state prior to the cation exchange process.	Resulting solution from ion exchange is transferred for liquid waste treatment. In a case of a leaky pipe or valve it could be found in cheesecloth in retrievable waste.
Iodine Crystals	Metal Production	11	Used with calcium metal as a booster for reduction of the plutonium fluoride to plutonium metal during the bomb reduction procedure.	Found in the reduction slag which is sent to recovery in stainless steel cans. May also be found on PVC plastic bags resulting from transfer. Can be found in small amounts in exhaust filters.
Lithium Fluoride	Metallurgy	0.5	Used in etching fuel pellets in metallurgy.	Found in exhaust filters, dry box gloves, and PVC plastics.
Magnesium Oxide	Metal Production	2,455	Used as a crucible material and as a packing sand during bomb reduction of plutonium fluoride.	May be found on stainless steel cans and PVC bags used in transfer to recovery section.

TABLE F-2.1

(CONT.)

<u>Chemical</u>	<u>Area</u>	<u>Annual Issue (kg)</u>	<u>Use Description</u>	<u>Waste Items Contaminated</u>
Nitric Acid (Bulk) (Technical Grade)	Scrap Recovery	52,300 ℓ	Used in all leaching processes: ash, graphite, surface con- taminated materials and in bulk fusion and dissolution of sand slag and crucible.	Found in residual amounts on all leached and pickled materials such as dry box gloves, plastic, glass and ceramics, metals and graphite. Also found in all process residues from bulk fusion and ash leaching.
Nitric Acid (Analytical Reagent)	Scrap Recovery	5,290	Used in ion exchange column regeneration, washing, and if needed, during column elution.	Found in spent ion exchange resin. (nitric acid is so commonly used it may be found on most items; filter- aid, line filters, dry box gloves and gaskets, and on cheesecloth used in cleaning.)
Nitric Acid (Analytical Reagent)	Metal Production	399	Used to adjust nitrate concen- tration prior to peroxide pre- cipitation and to treat filtrate after precipitation.	Found in exhaust filters, dry box gloves, and PVC plastics.
Nitric Acid (Technical Grade)	Metallography	258	Used in etching fuel pellets in metallography.	Disposed of in a self- contained acid drain in dry box line and therefore may be a contaminant on poly- ethylene bottle and PVC bag. It may also be found on exhaust filters and dry box gloves.

TABLE F-2J

(CONT.)

<u>Chemical</u>	<u>Area</u>	<u>Annual Issue (kg)</u>	<u>Use Description</u>	<u>Waste Items Contaminated</u>
Oil, Heat Treating	Fabrication	19 ℓ	Use in hot bath for heat treating metal parts	Found in cheesecloth when spilled and in exhaust filters from vaporization and on dry box gloves and window gaskets. Also found on PVC plastic bags.
Oil, Hydraulic	Fabrication	853 ℓ	Hydraulic fluid used to operate NC (numerically controlled) machine.	May be found on cheesecloth when leakage in system occurs. Transferred for liquid waste disposal.
Oil, Lubricating	Metallography	2	Machining oil and a grinding and polishing lubricant.	Transferred to recovery in glass jars. May be found in cheesecloth when spilled, and on PVC plastic when bagged out of line.
Oil, Machining	Pu-238 Research	7	Machining oil.	Transferred to recovery in glass jars.
Oil, Vacuum Pump	Fabrication	19 ℓ	Operational fluid for diffusion pump.	Drained into tin can when changed and transferred for liquid waste treatment.
Oil, Vacuum Pump	Fuels Research	38 ℓ	Operational fluid for vacuum pump.	Transferred to recovery in glass jars or sent for liquid waste disposal.
Oxalic Acid	Scrap Recovery	421	Used to precipitate the eluate from the anion-exchange system.	Residual amounts found in polyethylene transfer bags and may be found on cheesecloth if spilled. Filtrate is recycled to ion exchange feed adjustment.

TABLE F-2J

(CONT.)

<u>Chemical</u>	<u>Area</u>	<u>Annual Issue (kg)</u>	<u>Use Description</u>	<u>Waste Items Contaminated</u>
Paint, Krylon Clear	Fabrication	1.5 <i>ℓ</i>	Used in changing dry box windows.	Found on rubber window gaskets and occasionally in exhaust filters.
Potassium Chloride	Metal Production	32 <i>ℓ</i>	Used in combination with sodium chloride for electrorefining plutonium.	Packaged in tin can and bagged out of line in PVC plastic. Trace amounts may be found in exhaust filters.
Potassium Hydroxide	Scrap Recovery	9,318	Used to manufacture scrub solution for off-gases from the hydrofluorination system and those from the SCC dissolvers. It is used for treating electrorefining salt residues.	Scrub solutions are transferred to liquid waste treatment.
Potassium Pyrosulfate	Scrap Recovery	982	Used in bulk fusion in combination with sodium fluoride to treat insoluble plutonium-bearing solids.	Residual amounts found in cardboard transfer containers and in exhaust filters.
Silicon Metal	Fabrication	0.028	Used in experimental casting work.	Not found in waste stream.
Sodium Chloride	Electrorefining	41	Mixed with potassium chloride and used in the electrorefining process of plutonium.	Packaged in tin cans and bagged out of line in PVC plastic. Trace amounts may be found in exhaust filters.
Sodium Fluoride	Scrap Recovery	184	Used in bulk fusion in combination with potassium pyrosulfate to treat insoluble plutonium-bearing solids.	Residual amounts found in transfer containers and trace amounts may be found in exhaust filters.
Sodium Hydroxide	Scrap Recovery	1,888	Used to prepare scrub solution for incinerator off-gases. Also used to clean oxalate storage tank.	Solutions transferred for liquid waste treatment.

TABLE F-2J

(CONT.)

<u>Chemical</u>	<u>Area</u>	<u>Annual Issue (kg)</u>	<u>Use Description</u>	<u>Waste Items Contaminated</u>
Sodium Nitrite	Scrap Recovery	318	Used for the oxidation of plutonium from the trivalent state to the tetravalent state.	Small quantities may be found in cardboard transfer containers, in exhaust filters, and in cheesecloth if spilled.
Sulfuric Acid	Metallography	8	Used in metal sample etching.	Disposal in self-contained acid drain may result in contamination of PVC plastic and polyethylene plastic jar.
Sulfuric Acid	Metal Production	25	Used in adjustment of nitrate feed prior to peroxide precipitation.	Found in resulting solution filtrate which is transferred to recovery. May be found as a contaminant on some glassware, plastics, and dry box gloves.
Trichloroethylene	Scrap Recovery	10	Used to reduce viscosity of oils during filtration step in recovery process.	Residue may be found in filter-aid.
Trichloroethylene	Fabrication Metal Production	632 30	Used in ultrasonic baths for cleaning and with cheesecloth for cleaning metal parts.	Evaporated in line.
Triethylene Tetramine	Metallography	0.5	Epoxy catalyst used in metallography.	Usually found in floor sweepings and may be found in exhaust filters.
Urea	Scrap Recovery	9	Used in ion exchange feed solutions to remove nitrite ions.	Not usually found in solid waste.

Plastic

Retrievable plastic wastes consist primarily of PVC bags and bag stubs. Some sheet material used as temporary floor covering, and laboratory wares such as funnels, petri dishes, graduated cylinders, wash bottles, tubing, and gaskets also appear in this waste stream. PVC bags are generally contaminated to less than 0.1 g Pu/kg waste.

Process Solids

Incinerator ash, after being leached with nitric acid and calcium fluoride, is the main solid residue from scrap recovery operations. Any other solid material collected from glovebox cleaning is also leached in the same fashion. Other terms used on a local basis include sweepings, heels, ash heels, and leached residues. Present discard limit for this material based on the economics of recovery is 4 g Pu/kg waste.²⁶

Metal

Retrievable metal wastes are represented by a diverse stream of nails, nuts, bolts, wiring, conduit, tin cans, stainless steel dressing jars, aluminum foil, lathe turnings, hacksaw blades, screw drivers, tweezers, hammers, hair dryers, hot plates, heating coils, vacuum cleaners, and furnaces. Most of the tin cans in which off-site ash (from Hanford) had been received were less than 0.1 g Pu/kg waste. The contamination present on tools and small equipment can vary greatly depending on usage, time in the glovebox, and cleaning effort.

Glass and Ceramic

Retrievable glass and ceramic materials primarily include normal laboratory glassware such as beakers, cylinders, graduated cylinders, and 1-liter to 9-liter bottles. Heating mantles used in batch leaching operations are occasionally discarded, but are normally reduced to a small residue by volatilizing the silicon in a hydrofluorination treatment.

Rubber

Almost all of this waste stream was glovebox gloves. When room trash (over 10 nCi/g) is merged with the process generated waste, some surgeon's gloves used in trash collection contribute to this waste stream. Occasionally items such as rubber stoppers and tubing would appear.

Tables F-2K through F-2M present additional information on the retrievable waste streams.

2.3 PLUTONIUM RECOVERY^{72, 71}

Besides wastes generated in various process activities extensive waste may be generated in plutonium recovery operations. Table F-2N presents the TRU waste form generated at LASL by plutonium recovery operations via incineration and subsequent ash leaching. Tables F-2O through F-2R present additional information on this TRU waste source.

2.4 BURIED AND STORED TRU WASTE³⁹

Besides various generated waste, previous waste management program activities may require that the buried and stored TRU waste be retrieved. The potential for waste container failure and waste form leaching may result in a substantial waste source.

2.4.1 STORAGE FACILITIES⁹⁰

Located in area G, the retrievable storage trench is 9.1 m wide, 122 m long, and 7.6 m deep with a 1:4 slope entrance ramp and an asphalt-lined floor. Sumps at the low point in the trench collect all water entering the trench as a result of rain or snowmelt. Waste packages are stacked in the trench to within 1-2 m from the surface, and the stack is then covered with 1.9 cm plywood and completely encased with a reinforced vinyl material. Finally, the waste stack is covered with 1.2 m of backfill.

TABLE F-2K
MATERIAL SENT TO 20-YEAR RETRIEVABLE STORAGE²⁷

<u>Material Type</u>	<u>Average Activity nCi/g</u>
Metal	13×10^3
Plastic and Rubber	17×10^3
Glass and Ceramics	8×10^3
Insulation	5×10^3

TABLE F-2L
COMPOSITION OF 20-YEAR RETRIEVABLE WASTE
FROM ROUTINE OPERATIONS²⁷

<u>Material Type</u>	<i>Composition (Wt. %)</i>			
	<u>October</u>	<u>November</u>	<u>December</u>	<u>Average</u>
Metal	25	22	32	28
Plastic	36	30	23	29
Rubber	5	8	10	8
Cellulosics	12	11	14	12
Glass	18	29	21	22
Process Solids	4	<1	<1	1

TABLE F-2M

TYPICAL WEIGHT-VOLUME RELATIONSHIP
OF 20-YEAR RETRIEVABLE WASTE,
NON-COMPACTED,
PACKAGED IN 114-LITER DRUMS²⁷

<u>Material Type</u>	<u>Weight Percent</u>	<u>Volume Percent</u>
Metal	32	22
Plastic	23	32
Rubber	10	10
Cellulosics	14	26
Glass	21	10

TABLE F-2N
PLUTONIUM RECOVERY (BY INCINERATION)
WASTE GENERATION⁷²

1. Incineration

Wastes

Replaced burning chamber, lid, hydraulic cylinder lid closer, valves and piping on oxygen supply system
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Glovebox windows and gaskets
HEPA filters
Tygon tubing
Anti-contamination clothing
Booties
Masking tape

2. Screen Incinerator Ash

Waste

Damaged screen
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Glovebox windows and gaskets
HEPA filters
Tramp metal
Anti-contamination clothing
Booties
Masking tape

3. Ash Leach

Waste

Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Heating mantles
Broken glass pots and condensers
Cans or containers used to supply ash feed
Anti-contamination clothing
Booties
Masking tape

4. Pickle Scrap Metal

Waste

Hot plates
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Heating mantles
Broken glass pots and condensers
Anti-contamination clothing
Booties
Masking tape

5. Filter Ash Leach Slurry

Wastes

Replaced tanks
Replaced valves and fittings on transfer lines and sample taps
Replaced parts on Nash vacuum pump
Surgeons' gloves
Glovebox gloves
PVC bags & bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Used freezerette containers
Damaged plastic bottles
Tin cans
Glassware
Broken glass jugs
Pipettes and syringes
Broken flasks and Buchner filters
Damaged stainless-steel fan filters and screens
Carbon filter elements
Anti-contamination clothing
Booties
Masking tape

6. Filter Pickling Solutions

Wastes

Replaced tanks
Replaced valves and fittings on transfer lines, vacuum beakers, and sample taps
Replaced parts on Nash vacuum pump
Surgeons' gloves
Glovebox gloves
PVC bags & bag stubs
Cheese cloth
Glovebox windows & gaskets
Paper & Plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Tin cans
Broken jars
Pipettes and syringes
Broken frit filters and flasks
Filter paper
Anti-contamination clothing
Booties
Masking tape

7. Nondestructive Assay (Leached Ash and Non-plutonium Scrap Metal)

Wastes

Surgeons' gloves
17H drums
Polyethylene drum liners
Anti-contamination clothing
Booties
Masking tape

8. Storage of Retrievable Waste

Wastes

Surgeons' gloves
17H drums
Booties
Masking tape

9. Ion-Exchange Feed PreparationChemicals

Nitric Acid
Sodium Nitrate
Hydroxylamine nitrate
Urea

Wastes

Replaced tanks
Replaced pump or pump parts
Replaced piping, fittings, and valves
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Pipettes and syringes
Anti-contamination clothing
Booties
Masking tape

10. Filter Ion-ExchangeChemicals

Filter aid - Celite

Wastes

Filter case replacement
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Hand tools
Broken stainless-steel filter elements
Replaced piping & valves on filter system
Filter solids in freezerette containers
Anti-contamination clothing
Booties
Masking tape

11. Nondestructive Assay
(Ion-Exchange Filter Solids)

Waste items from all unit operations go to nondestructive assay and are then either recycled to appropriate scrap recovery processes or sent to 20-year retrievable storage.

12. Ion ExchangeChemicals

Nitric acid
Hydroxylamine nitrate

Wastes

Broken ion-exchange columns
Ion-exchange resin
Replaced tanks
Replaced pumps or pump parts
Replaced piping, fittings and valves
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Anti-contamination clothing
Booties
Masking tape

13. Storage of Retrievable Wastes

See description of 8 above.

14. Collection of $Pu(NO_3)_4$ Product
(for Transfer to Metal Preparation)Wastes

Surgeons' gloves
Polyethylene bags
Polyethylene bottles
Anti-contamination clothing
Booties
Masking tape

15. Oxalate PrecipitationChemicals

Oxalic acid

Wastes

Replaced glass tanks (feed tanks)
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Pipettes and syringes
Anti-contamination clothing
Booties
Masking tape

16. Plutonium Nitrate Product
(Transfer to Metal Preparation)Wastes

Part of metal preparation flowsheet

17. Filtration of Oxalate SlurryWastes

Replaced glass holding tanks (filtrate receiver)
Surgeons' gloves
Glovebox gloves
PVC bags and bag stubs
Cheese cloth
Glovebox windows and gaskets
Paper and plastic from window changes
HEPA filters
Tygon tubing
Corroded hand tools
Broken glass frit filters and Littenmeyer flasks
Anti-contamination clothing
Booties
Masking tape

TABLE F-2IN

(CON1.)

18. Calcine Oxalate Cake to PuO₂Wastes

Replaced furnace or parts such as electric elements or ceramic liners
 Surgeons' gloves
 Glovebox gloves
 PVC bags and bag stubs
 Cheesecloth
 Glovebox windows and gaskets
 Paper and plastic from window changes
 HEPA filters
 Corroded hand tools
 Anti-contamination clothing
 Booties
 Masking tape

19. Pure PuO₂ (Sampling and Storage -
Some Off-Site Shipments)Wastes

Surgeons' gloves
 Glovebox gloves
 PVC bags and bag stubs
 Cheesecloth
 Glovebox windows and gaskets
 Paper and plastic from window changes
 HEPA filters
 Hand tools
 Polyethylene bags
 Spatulas
 Anti-contamination clothing
 Booties
 Masking tape

20. PuO₂ Product (When Transferred
Off-Site)Wastes

Damaged 6M containers
 Replaced balance
 Replaced forklift truck and/or parts
 Surgeons' gloves
 Leather gloves
 Hand tools
 Vermiculite
 Special tape and labels
 Security seals
 Shipping documents
 Anti-contamination clothing
 Booties
 Masking tape

21. Incinerator Liquid Scrubber
for Combustion GasesChemicals

Sodium hydroxide

Wastes

Replaced scrub pot
 Replaced heat exchanger
 Surgeons' gloves
 Cheesecloth
 Tygon tubing
 Anti-contamination clothing
 Booties
 Masking tape

22. Filter Caustic Scrub SolutionWastes

Replaced tank
 Replaced piping, fittings and valves
 Surgeons' gloves
 Glovebox gloves
 PVC bags and bag stubs
 Cheesecloth
 Glovebox windows and gaskets
 Paper and plastic from window changes
 HEPA filters
 Tygon tubing
 Corroded hand tools
 Plastic bottles
 Broken Erlenmeyer flasks and frit filters
 Pipettes and syringes
 Filter solids (returned to incinerator)
 Anti-contamination clothing
 Booties
 Masking tape

23. Filter Noncondensable GasesWastes

Replaced equipment listed and/or parts for pumps and piping
 Surgeons' gloves
 Cheesecloth
 HEPA filters
 Tygon tubing
 Corroded or contaminated hand tools
 Polyethylene bags
 Anti-contamination clothing
 Booties
 Masking tape

24. Sample Caustic Solution (From
Scrub Pot and Seal Liquid
Used in Nash Vacuum Pump)Wastes

Pump or replacement parts for pump
 Surgeons' gloves
 Polyethylene bags
 Plastic bottles
 Pipettes and syringes
 Anti-contamination clothing
 Booties
 Masking tape

TABLE F-2N

(CONT.)

25. Discard of Waste Caustic Solution
(Transferred to Liquid Waste Treatment)Wastes

Replaced tank
 Replaced pump or pump parts
 Replaced piping, fittings and valves
 Surgeons' gloves
 Anti-contamination clothing
 Booties
 Masking tape

26. Sample Ion-Exchange EffluentWastes

Replaced tanks
 Replaced pump or pump parts
 Replaced piping, fittings and valves
 Surgeons' gloves
 Glovebox gloves
 Polyethylene bags
 Plastic bottles
 Pipettes and syringes
 Anti-contamination clothing
 Booties
 Masking tape

27. Discard of Ion-Exchange Effluent
(Transferred to Liquid Waste Treatment)Chemicals

Sodium hydroxide

Wastes

Replaced pump or pump parts
 Replaced trailer tank
 Replaced piping, fittings and valves
 Replaced tractor
 Surgeons' gloves
 Hand tools
 Anti-contamination clothing
 Booties
 Masking tape

28. Solidify Neutralized Liquid
Waste ChemicalsChemicals

Portland cement

Wastes

Replaced parts for pug mill
 Replaced tanks
 Replaced pump or pump parts
 Replaced piping, fittings and valves
 Surgeons' gloves
 Rags
 Contaminated tools
 Damaged fire hose
 Anti-contamination clothing
 Booties
 Masking tape

29. Storage of Retrieval Wastes

No waste

TABLE F-20
RETRIEVABLE WASTES
SORTED ACCORDING TO COMBUSTIBILITY

	<u>Combustible</u>	<u>Noncombustible</u>
Waste Density (kg/m ³)	132	194
Weight Percent	45	55
Volume Percent	54	46
g Pu/kg waste	0.186	0.190
nCiPu g waste	13,262	13,547

TABLE F-2P
RETRIEVABLE WASTE GENERATED
PER GRAM OF PLUTONIUM PROCESSED

<u>Unit Operation</u>	<u>kg Waste/g Pu</u>
Ash Leaching	0.045
Ion Exchange	0.005
Alloy Processing	0.003

TABLE F-20
RETRIEVABLE WASTE GENERATED
PER MAN-HOUR

<u>Unit Operation</u>	<u>kg Waste/Man-Hour</u>
Ash Leaching	0.522
Ion Exchange	0.125
Alloy Processing	0.046

TABLE F-2R
COMPOSITION OF RETRIEVABLE WASTE
FROM UNIT OPERATIONS^{72,27,71}

<u>Material Type</u>	Composition (wt%)		
	<u>Ash Leaching</u>	<u>Ion Exchange</u>	<u>Alloy Processing</u>
Metal	23	0	22
Plastic	14	26	17
Rubber	7	12	0
Cellulosics	11	2	9
Glass	14	8	38
Process Solids	31	52	14

Wastes placed in the trench are limited to 1 g Pu-238 per 208-liter drum, 100 g of other TRU per 114-liter drum, 200 g of other TRU per 208-liter drum, or 175 g per m^3 to a maximum of 350 g per crate.

Wastes contaminated with Pu-238 in excess of 1 g per drum or with U-233 with high gamma radiation are stored in concrete casks and placed into shallow trenches (40-80 m long by 2.8 m deep). The outside of each cask is coated by the manufacturer with a bitumin epoxy material and is coated a second time with an asphalt roofing compound. The casks are placed vertically in a trench, and the trench is backfilled almost to the top of the cask.

Two 114-liter drums of waste containing up to a maximum of 40 g each of Pu-238 or 200 g U-233 are placed inside each cask. The cask lids are sealed with roofing tar. When a portion of a trench is filled, corrugated galvanized sheet metal is used to cover the trench over the tops of the casks, and the trench is backfilled to ground level and mounded 0.5 to 1.0 m above ground level.

2.4.2 SOLIDS III WASTE TRENCHES⁵

Most small solid articles contaminated with radioactivity are put into 0.03 m^3 steel foot-operated rubbish bins with polyethylene bag liners. If a slightly larger volume is needed 0.06 m^3 , corrugated cardboard cartons lined with plastic bags are used. When these plastic liners are full, they are sealed and collected either in 0.34 m^3 plastic bags or in the cardboard boxes which are also sealed. Other containers, such as 3.8-liter, 19-liter, 114-liter, and 208-liter steel cans or drums, are used for special situations. The large bags, boxes, and other containers are placed in steel transportable waste bins. The bins are picked up by special trucks and taken to the contaminated waste disposal area for discharge. Large contaminated solid items are covered as necessary to prevent spread of contamination during shipment and transported separately to the disposal area. This miscellaneous bagged and boxed waste currently makes up about two-thirds of the waste buried at LASL. Over 90 percent is contaminated with plutonium. Dewatered sludge from the low-level liquid waste treatment plant is put into 208-liter steel drums and hauled to the disposal area.

Disposal trenches are used for the disposal of low-activity nonretrievable radioactive wastes. These wastes include contaminated trash from laboratory areas, failed equipment, and solid residues from liquid waste treatment. Wastes are packaged in plastic bags, cardboard boxes, or wooden crates.

2.4.3 SOLIDS IN WASTE SHAFTS⁵

Deep shafts, augered into the volcanic tuff, are used for the disposal of high beta-gamma waste, tritium waste, contaminated animal tissue, cement paste, and classified contaminated waste.

Small items that are known to contain somewhat larger amounts of radioactivity, particularly beta-gamma radiation, are put into vertical shafts or holes in the ground and covered at 0.9 m intervals. These shafts are 0.9 m in diameter and 9 m deep. In some cases the shaft is lined with 0.15 m of concrete and the waste is covered with concrete or asphalt. Since 1956, any waste containing a significant quantity (milligram) of plutonium has been buried separately in special shafts or holes.

Since mid-1968, the dewatered sludge from one liquid waste disposal facility and an americium raffinate (approximately 6,050 liter/month containing about 1.06 mg Am-241/liter and 2.12 mg Pu-239/liter) are fixed in cement in a continuous process and pumped through an underground pipeline to deep holes located in an abandoned waste disposal area about 46 m away from the cementing facility. These holes or wells are 1.8 to 2.4 m in diameter and 7.6 to 20 m deep at both areas G and T. They are filled with waste up to 1 m from ground surface or up to where the allowable level is reached. No more than 500 g of fissile material may be disposed of in a shaft, and radiation doses at the shaft surface cannot exceed 100 mR/hr. Whenever doses are exceeded, backfill is added to reduce the dose. When the shaft is filled with waste, excavated turf is added to fill void spaces, a final 1 m concrete plug is poured, and the concrete is mounded above the shaft surface to provide a final seal with proper drainage. Before mid-1968, the americium raffinate was fixed in cement by a batch process in 208-liter drums which were hauled by truck to the waste pits. Most of the americium discarded is in this cemented product.

2.4.4 AREA G PIT TRU WASTE DISPOSAL³⁹

Of the waste buried in Area G at LASL approximately 50 percent is considered combustible, consisting mainly of cellulosic materials, rubber, and plastic. A typical composition of the previous wastes is somewhat speculative but an estimated composition is provided in Table F-25.

The recorded transuranic contamination occurs in several types of waste. Prior to 1968, drums of sludge from the liquid treatment plant were delivered to Area G for burial. These drums contained isotopes of plutonium and americium in significant quantities. After that date, the sludge was mixed with a cement paste and disposed of in another area. A considerable portion of the plutonium contamination in the disposal pits is present on routine laboratory waste. On an individual package basis, the contamination was generally too low to be assayed by equipment existing at the time of burial.³⁹ The primary source of data on contamination levels in this routine waste is the tabulation of write-offs from the Accountability Group A-3, at LASL.³⁹ An undefined portion of the contamination occurs on nonroutine waste; items of equipment, gloveboxes, filter plenums, etc. Again, the only source of data concerning possible levels of contamination is the accountability records.

From the available information on pit operation dates and quantities of material disposed of, an estimate has been made of the total transuranic content of the pits in Area G that were used through 1972. The contaminants are uniformly distributed throughout the waste pits even though the assumption of homogeneity ignores the existence of "hot-spots" which probably exist within the burial pits. There were not estimates for these hot-spots.

The contamination levels in the disposal pits in Area G that were used through the end of 1972 are summarized in Table F-2T. The significant drop in contamination levels occurring in Pit 6, compared with the previous pits, was a result of a change in disposal practices at LASL. The segregation of "transuranic retrievable waste" and the discontinuation of deliveries of significantly contaminated sludge were the two main contributors to this decrease.³⁹

TABLE F-25

TYPICAL COMPOSITION OF TRANSURANIC
CONTAMINATED WASTE AT AREA G, TA-54³⁹

<u>Category</u>	<u>(Retrievable) Process Wastes (% volume)</u>	<u>(Nonretrievable) Room Wastes (% volume)</u>
Metal	19.3	5
Glass	10.5	1
Cellulosics	22.3	83
Rubber	8.0	4
Plastic	40.3	7

TABLE F-2T
ESTIMATED RADIONUCLIDE CONTENT OF LASL AREA G PITS³⁹

Pit No.	Volume (m ³)	Pu-239 (Ci)		Pu-240 (Ci)		Am-241(Ci)	Total TRU(Ci)	Est. Conc. of TRU(nCi/g)
		<u>Sludge</u>	<u>Routine</u>	<u>Sludge</u>	<u>Routine</u>	<u>Sludge</u>		
1	3.4×10^4	45	12	7	10	605	679	12.5
2	3.4×10^4	23	77	7	17	352	476	8.8
3	3.4×10^4	34	25	3	2	586	650	12.0
4	3.4×10^4	14	13	3		219	279	4.6
5	3.4×10^4	21	37	18	10	315	401	7.4
6	3.4×10^4	1	17	5	3		26	0.5

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3.0 HANFORD RESERVATION

The semiarid Hanford Reservation site occupies about 1,480 square kilometers of the southeastern part of the State of Washington. The reactor facilities are located along the Columbia River in what is known as the 100 Area. The reactor fuel processing and waste management facilities are in the 200 Area which is on a plateau about 11 kilometers from the river. The 300 Area, just north of Richland, contains the reactor fuel manufacturing facilities and the research and development laboratories.

In early 1943, the United States Army Corps of Engineers selected the Hanford site as the location for reactor and chemical separation facilities for the production and purification of plutonium for possible use in nuclear weapons (Manhattan Project). A total of eight graphite-moderated reactors using the Columbia River water for once-through cooling, and a new type of dual purpose reactor (N-Reactor) using a recirculating water coolant and producing both plutonium and steam for electricity, were eventually built along the Columbia River. Today, only the N-Reactor remains in operation.⁵

3.1 DESCRIPTION OF FACILITIES AND RADIOACTIVE WASTE INVENTORIES⁵

The DOE facilities at the Hanford Reservation include nuclear reactors for plutonium production (100 Area); fuels and plutonium processing and high- and low-activity radioactive waste management (200 Area); fuel fabrication (300 Area); laboratory complex (300 Area); the Fast Flux Test Facility (400 Area); central service facility (1100 Area); on-site construction services (3000 Area); and administrative services (700 Area).

Solid wastes contaminated with mixed fission and activation products have been disposed of by shallow land burial at the Hanford site since the beginning of operations in 1944. Indeed, over 95 percent of all TRU waste generated and disposed at Hanford has been buried in a non-retrievable mode. A total of 26 sites in the 100 Area (26 ha), 28 sites in the 200 Area (68 ha), and 11 sites in the 300 and 600 Area (26 ha) have been used for the storage and disposal of solid radioactive waste.⁹⁰ Various facilities are discussed below.

3.1.1 100 AREA - REACTOR FACILITIES⁴⁾

The 100 Area contains:

- One operating reactor (N-Reactor)
- Eight retired or standby reactors
- Cribs and burial grounds

Solid waste from 100-N Area was buried in trenches in other 100 Areas in the past. The waste contained dried salts and oxides of the activation and fission products found in solution in the primary coolant water. A few metallic components, taken from the active reactor zone, containing activated zirconium, iron, and steel were also buried in these facilities. Waste containing trans-uranium elements and/or high concentration (10 mCi/g) of long-lived radio-nuclides (>5-year half-life) was shipped to the 200 Areas for disposal with the following exceptions:

- Contaminated Equipment Storage - An area was set aside for temporarily storing contaminated equipment removed from radiation zones prior to reuse or disposal.
- Dummy Storage Silos - Silos are in active use for storage of irradiated process spacers, called dummies because they resemble fuel elements but contain no uranium and are used as spacers. No contamination exists on the ground surface. Dummies are periodically removed.
- 100-K Area Silos - Two silos in the 100-K Area (each 2.4 m in diameter and 7.2 m deep) are used for disposal of metallic scrap from N Reactor's fuel storage basin. This waste consists of irradiated fuel element support hardware, bent spacers and other metallic refuse found in the storage basin. Ruptured fuel element fragments are not included in this waste.

Radioactive waste management facilities in the five 100 Areas in which the reactors were retired are largely the terminated cribs, trenches, burial areas and the contaminated facilities such as the reactors, spent fuel basins and effluent

holdup basins. The status of these facilities is constantly monitored and maintained in their terminated state pending permanent disposition. The 183-H basin is currently used for receiving the neutralized chemical waste from the 300 Area fuels fabrication facility for solar evaporation and eventual recovery of copper and uranium.

All current wastes are shipped to the 200 Area for disposal. However, 1.6×10^4 m³ of solid waste was buried in the past as well as current N-Reactor primary coolant release to the 100-N crib.

Detailed records were not kept in the past. However, total estimated input inventories at the reactor facilities through 1972 are:

<u>Radionuclide</u>	<u>Ci</u>
Sr-90, Cs-137 and Pu-239	160
Eu-152 and Pu-239	650
Co-60	76,100
Fe-59	150,000

Total estimated input inventories for crib and burial grounds through 1972 (except for 100-N Crib) are:

<u>Radionuclide</u>	<u>Ci</u>
Co-60	25,900
Sr-90	34
Pu-239	0.38
Zn-65	1,000
Mn-54 and Zr-95	3,500

Table F-3A presented the estimated inventory in various reactor facilities (through 1972) which contain TRU isotopes. This waste is potentially subject to retrieval and subsequent treatment and disposal.

TABLE F-3A
ESTIMATED INVENTORIES IN REACTOR FACILITIES
THROUGH 1972⁴¹

	<u>Inventory (Curies)</u>
<u>100-B</u>	
2 Reactor Blocks	13,000 (⁶⁰ Co)
2 Metal Storage Basins	20 (⁹⁰ Sr, ¹³⁷ Cs, ²³⁹ Pu)
3 107 Basins and Soil	150 (¹⁵² Eu, ²³⁹ Pu)
<u>100-D</u>	
2 Reactor Blocks	8,500 (⁶⁰ Co)
2 Metal Storage Basins	20 (⁹⁰ Sr, ¹³⁷ Cs, ²³⁹ Pu)
2 107 Basins and Soil	100 (¹⁵² Eu, ²³⁹ Pu)
<u>100-F*</u>	
1 Reactor Block	4,300 (⁶⁰ Co)
1 Metal Storage Basin	10 (⁹⁰ Sr, ¹³⁷ Cs, ²³⁹ Pu)
1 107 Basin and Soil	50 (¹⁵² Eu, ²³⁹ Pu)
<u>100-H</u>	
1 Reactor Block	4,300 (⁶⁰ Co)
1 Metal Storage Basin	10 (⁹⁰ Sr, ¹³⁷ Cs, ²³⁹ Pu)
1 107 Basin and Soil	50 (¹⁵² Eu, ²³⁹ Pu)
<u>100-K</u>	
2 Reactor Blocks	26,000 (⁶⁰ Co)
2 Metal Storage Basins	100 (⁹⁰ Sr, ¹³⁷ Cs, ²³⁹ Pu)
6 107 Basins and Soil	300 (¹⁵² Eu, ²³⁹ Pu)
TOTAL (Ci)	160 (⁹⁰ Sr, ¹³⁷ Cs, ²³⁹ Pu) 650 (¹⁵² Eu, ²³⁹ Pu)

* Area 100-F includes the following:

Sawdust Repository	15 (⁹⁰ Sr)
(Research)	0.3 (²³⁹ Pu)
Animal Waste	
Leaching Trench	4 (⁹⁰ Sr)
(Research)	0.08 (²³⁹ Pu)

TERA CORPORATION

Various unplanned releases have occurred in the 100 Areas. These include:⁴⁾

<u>DATE</u>	<u>BLOC.</u>	<u>DESCRIPTION</u>	<u>NUCLIDE & AMOUNT</u>
3-26-54	105-H	Ground level contamination occurred while 102-C Basin was empty for repair. Particulate contamination up to 50 mrad/hr covered 20% of 100-B Area. No particles above 10 mrad/hr were found outside 100-B Area. Roadways were water flushed and the concentrated activity was removed to the burial ground.	50% Rare Earths 5% Ru-103, -106 Traces of Sr, U, Pu
12-4-51	P-11	Fire occurred in contaminated waste storage area of a remotely located research facility. Fire spread to a chemical laboratory and exhaust filters. Contamination was confined to immediate vicinity of laboratory. Contaminated soil surfaces were covered with sand.	0.7 Ci Pu
9-17-56	105-C	A fuel element burned for 6 minutes at the reactor discharge face resulting in stack emission of particulates. Contamination levels on ground were 5000 to 20,000 c/m. Maximum particle 350 mrad/hr. Area involved was decontaminated.	Np-239 and mixed F.P.

3.1.2 200-AREA REACTOR FUEL PROCESSING AND WASTE MANAGEMENT FACILITIES⁴⁾

In addition to the fuel processing facilities the following waste disposal facilities receive waste from all other Hanford areas, as well as outside agencies. Table F-3B indicates the sources and products of waste output from the 200 Areas operations.

3.1.2.1 CRIBS (FOR INTERMEDIATE LEVEL WASTES)¹⁵⁾

- 144 deactivated
- 8 not used
- 10 standby
- 15 currently in use

177 Total

Table F-3C gives the estimated inventories discharged to 200 Area cribs and ponds while Table F-3D gives the estimated inventory of plutonium at these sites.

TABLE F-3B
RADIOACTIVE WASTE SOURCES⁴¹

Source	Type of Waste		Solid
	Liquid		
	Inter- mediate Level	Low Level	
<u>200 EAST AREA</u>			
Purex Plant	X	X	X
B Plant	X	X	X
244-AR Vault	X	X	X
241-A, AX, AY Tank Farms	X	X	X
Critical Mass Laboratory	--	X	X
Semiworks	--	--	--
CR Vault	--	--	--
ITS-1 and 2 Concentrators	X	X	X
241-BX, BY and C Tank Farms	--	--	--
<u>200 WEST AREA</u>			
T Plant	X	X	X
242-T Concentrator	X	X	X
Laundry	--	X	X
231-Z	--	X	X
U Plant	--	X	--
Z Plant	X	X	X
UO ₂ Plant	X	X	X
222-S Laboratory	X	X	X
241-SX Tank Farm	--	X	--
Redox Plant	--	--	--
241-TX Tank Farm	--	--	--
242S Evaporator	--	X	X
<u>FROM OTHER CONTRACTORS</u>			
Pacific Northwest Laboratory	X	X	X
United Nuclear Industries, Inc.	X	X	X
Westinghouse Hanford Company	X	X	X
J. A. Jones Construction Company	--	--	X

TABLE F-3C
ESTIMATED INVENTORIES^(a) RADIOACTIVE LIQUID WASTE
TO GROUND 200 AREAS⁴¹

	<u>Units</u>	<u>To Ponds and Ditches</u>	<u>To Cribs</u>	<u>To Specific Retention Sites</u>	<u>Total</u> ^(b)
Decay Through 1972 (From Startup) 200 East					
Volume	(l)	3.0×10^{11}	2.0×10^{10}	1.0×10^8	3.0×10^{11}
Pu	(g)	$<7.0 \times 10^2$	1.0×10^4	4.0×10^2	1.0×10^4
Beta	(Ci)	6.0×10^3	6.0×10^4	4.0×10^4	1.0×10^5
Sr-90	(Ci)	1.0×10^3	1.0×10^4	8.0×10^3	2.0×10^4
Ru-106	(Ci)	$<3.0 \times 10^1$	2.0×10^3	3.0×10^2	3.0×10^3
Cs-137	(Ci)	4.0×10^2	3.0×10^3	1.0×10^4	2.0×10^4
Co-60	(Ci)	$<3.0 \times 10^1$	1.0×10^2	2.0×10^1	$<2.0 \times 10^2$
U	(kg)	$<1.0 \times 10^4$	3.0×10^4	5.0×10^4	8.0×10^4
U-233	(g)	$<5.0 \times 10^2$	5.0×10^2		$<1.0 \times 10^4$
Decayed Through 1972 (From Startup) 200 West					
Volume	(l)	2.0×10^{11}	1.0×10^{10}	8.0×10^7	2.0×10^{11}
Pu	(g)	8.0×10^3	8.0×10^4	9.0×10^4	2.0×10^5
Beta	(Ci)	4.0×10^2	3.0×10^4	2.0×10^4	5.0×10^4
Sr-90	(Ci)	$<1.0 \times 10^2$	6.0×10^3	2.0×10^2	7.0×10^3
Ru-106	(Ci)	$<2.0 \times 10^0$	9.0×10^1	2.0×10^0	9.0×10^1
Cs-137	(Ci)	$<9.0 \times 10^1$	5.0×10^3	1.0×10^4	2.0×10^4
Co-60	(Ci)	$<9.0 \times 10^0$	3.0×10^1	3.0×10^0	$<4.0 \times 10^1$
U	(kg)	$<6.0 \times 10^3$	4.0×10^4	1.0×10^3	4.0×10^4

TABLE F-3C

(CONT.)

	<u>Units</u>	<u>To Ponds and Ditches</u>	<u>To Cribs</u>	<u>To Specific Retention Sites</u>	<u>Total^(b)</u>
Decayed Through 1972 (From Startup)			Total 200-E + 200-W		
Volume	(l)	5.0×10^{11}	3.0×10^{11}	2.0×10^8	5.0×10^{11}
Pu	(g)	9.0×10^4	9.0×10^4	9.0×10^4	2.0×10^5
Beta	(Ci)	7.0×10^3	9.0×10^4	6.0×10^4	2.0×10^5
Sr-90	(Ci)	2.0×10^3	2.0×10^4	8.0×10^3	3.0×10^4
Ru-106	(Ci)	$< 3.0 \times 10^1$	2.0×10^3	3.0×10^2	3.0×10^3
Cs-137	(Ci)	5.0×10^2	1.0×10^4	2.0×10^4	4.0×10^4
Co-60	(Ci)	$< 4.0 \times 10^1$	$< 2.0 \times 10^2$	2.0×10^1	$< 2.0 \times 10^2$
U	(kg)	$< 7.0 \times 10^3$	6.0×10^4	5.0×10^4	1.0×10^5
U-233	(g)	$< 5.0 \times 10^2$	5.0×10^2		$< 1.0 \times 10^3$

- (a) The estimated inventories (or concentrations) of radionuclides are reported to one significant figure to reflect the uncertainty in the data. Where the estimated inventories are noted as less than (<), the recorded inventory is based on the analytical limit of detection.
- (b) Totals do not necessarily equal the sum of individual contributions because of rounding of the data.

TABLE F-3D
ESTIMATED INVENTORIES OF PLUTONIUM IN DISPOSAL SITES⁴¹

<u>Disposal Site</u>	<u>Plutonium (kg)</u>	<u>Disposal Site</u>	<u>Plutonium (kg)</u>
216-U-10		216-Z-3	5.7
U Pond and Ditches	8.3	216-Z-4	0.002
216-B-7		216-Z-5	0.34
224-B, 221-B, 5-6 Cell and Construction Waste	4.3	216-Z-6	0.005
216-S-1 and -2		216-Z-7	2.0
Redox Process Condensate	1.2	216-Z-8	0.05
216-T-32		216-Z-9 ^(a)	38. ^(b)
224-T Waste	3.2		
216-Z-1AA	30.	216-Z-10	0.05
216-Z-1AB	17.	216-Z-12	25.
216-Z-1AC	11.	216-Z-16	0.07
216-Z-1 and 2	7.0	216-Z-17	0.05
		216-Z-18	22.

(a) "Z-9" Crib.

(b) Other estimates indicate that the system contains between 25 and 70 kilograms of plutonium.

Silicate glasses have been proposed for the treatment of TRU contaminated soil, plutonium-contaminated incinerator ash, and pyrolysis process residues. Preparation and properties of these glasses, and glasses made of HEDL acid digestion process residues are described in Reference 50.

A plutonium-contaminated soil was obtained from the 216-Z-9 Trench site at Hanford near the Plutonium Reclamation Facility (PRF) (see Table F-3E). The sandy, moist sample contained some organic sludge consisting primarily of degraded PRF solvents. The chemical composition of the soil is shown in Table F-3F. The Pu-239 concentration was 680 mg/liter of soil. Table F-3G presents the composition of Hanford basalt.

Melt charges were prepared by thoroughly mixing the soil with appropriate amounts of B_2O_3 and Na_2CO_3 which served as fluxes to reduce the temperature necessary to melt the mixture. Although not used, anhydrous borax ($Na_2O \cdot 2B_2O_3$) could be used in place of the Na_2CO_3 and B_2O_3 .

Plutonium-contaminated ash resulting from incineration of combustible organic material was obtained from the Hanford Plutonium Reclamation Facility. The concentration of Pu-239 was 8mg/g. Normally incinerator ash generated at Hanford is sent from the PRF to the Los Alamos Scientific Laboratory where it is contacted with an HNO_3 -HF solution for recovery of the plutonium. A small amount of plutonium, typically about 2 mg/g of residue, remains after processing. Both leached and nonleached ash could serve as feed to the immobilization process.

Residues were obtained from the Battelle-Pacific Northwest Laboratories laboratory-scale pyrolysis-gasification unit. The residues resulted from processing of nonradioactive charge and therefore contained no radioactive component. The as-received ash (chemical composition shown in Table F-3H) was a grayish-tan, relatively soft material, consisting of particles ranging from about 1 cm in diameter down to fluffy fines (estimated less than 100 mesh). The bulk density was:

TABLE F-3E
COMPOSITION OF Z-9 TRENCH SOIL⁵⁰

<u>Component</u>	<u>wt%</u>
SiO ₂	55.4
Al ₂ O ₃	11.2
FeO	6.1
MgO	2.4
CaO	2.5
Na ₂ O	2.4
K ₂ O	1.1
TiO ₂	1.7
Volatiles *	13.0
Pu-239 (mg/liter)	680

* Loss on ignition at 700°C. Volatiles are primarily water and degraded organic solvents from the PRP.

TABLE F-3F
COMPOSITION OF HANFORD BASALT⁵⁰

<u>Component</u>	<u>Concentration (wt%)</u>
SiO ₂	52
Al ₂ O ₃	13
FeO	14
CaO	8
Na ₂ O	3
MgO	4
K ₂ O	1.5
TiO ₂	2.5

TABLE F-3G

CHEMICAL ANALYSIS OF ASH
FROM PYROLYSIS-GASIFICATION PROCESS⁵⁰

<u>Ash Constituents^a</u>	<u>wt%</u>
H ₂ O ^b	4.6
SiO ₂	34.2
TiO ₂	16.6
Al ₂ O ₃	17.0
CaO	5.5
CaCl ₂ ^c	3.6
CaSO ₄ ^c	1.6
NaCl ^c	1.8
KCl ^c	1.3

^a The ash was made by pyrolysis of a feed stock containing 15 wt% rubber tubing, 15 wt% Neoprene, 20 wt% PVC, 10 wt% Tygon, 10 wt% polyethylene, 15 wt% tissue paper, and 15 wt% rags.

^b Adsorbed water released at 110°C.

^c Arbitrary assignments of 4.0% Cl and 0.4% S found by analysis.

TABLE F-3H
COMPOSITION OF ZINC BOROSILICATE FRIT⁵⁰

<u>Component</u>	<u>wt%</u>
SiO ₂	37
B ₂ O ₃	15.1
Na ₂ O	5.5
K ₂ O	5.5
ZnO	28.9
CaO	2.0
MgO	2.0
SrO	2.0
BaO	2.0

	<u>g/cm³</u>
As-received	0.211
After pulverizing to pass 6 mesh screen	0.272
Tap density of pulverized material	0.409

Table F-3f presents the waste formulations for these materials using the zinc borosilicate glass frit described in Table F-3h.

3.1.2.2 SOLID RADIOACTIVE WASTE BURIAL GROUNDS^{4,5}

About $1.90 \times 10^5 \text{ m}^3$ of contaminated solids have been buried on the 200 Area plateau since the start of chemical processing operations. These wastes consist of so-called "dry waste" (soiled clothing, laboratory supplies, tools, etc., packed in cardboard, wood or metal containers) and industrial waste (primarily items of failed process equipment packaged in heavy metal or concrete boxes).

Typical waste packages and waste content:

Mixed Fission Products

- Cardboard boxes - Slightly or potentially contaminated wastes such as wiping tissue and laboratory refuse.
- Plastic shrouds - Unrepairable equipment is wrapped in sheet plastic.
- Steel drums - Grossly contaminated wastes such as rags and small hardware.
- Wooden, concrete or steel boxes - Large equipment depending on size, weight, and radioactivity. No curie limit.
- Flat cars in rail tunnels - Major equipment too large for transport. No curie limit.

TABLE F-31
SUMMARY OF CONDITIONS
FOR MAKING GLASSES FROM TRU-BEARING WASTES⁵⁰

<u>Waste Type</u>	<u>Waste</u>	Melt Composition, wt %			
		<u>Basalt</u>	<u>Frit^a</u>	<u>B₂O₃</u>	<u>Na₂CO₃</u>
Z-9 Trench soil	20	60	--	10	10
Z-9 Trench soil	80	--	--	10	10
Z-9 Trench soil	70	--	--	10	20
Incinerator ash	2.2	82.8	--	5	10
Pyrolysis residue	20	--	80	--	--
Pyrolysis residue	25	--	75	--	--
Pyrolysis residue	30	--	70	--	--
Pyrolysis residue	35	--	65	--	--
Pyrolysis residue	40	--	60	--	--

^a Zinc borosilicate glass frit.

Transuranic Wastes in Retrievable Storage:

- Steel drums - Miscellaneous small items containing 10 nCi/g or more.
- Steel boxes - Equipment contaminated above 10 nCi/g that is too large to place in drums.
- Fiberglass, plastic-covered, double-walled boxes - Special boxes used for large-equipment items being removed from the shut-down plutonium fabrication facility.

Burial trenches are marked with permanent monuments which give location and designation. Trenches are covered with enough soil to reduce surface readings to less than 1 millirad per hour. A chemical operator and radiation monitor are present during dumping and burial operations.

Transuranic-bearing waste containing more than 10 nCi/g has been segregated in specially constructed burial trenches since May 1, 1970; before then the transuranic wastes were buried with the mixed-fission-product wastes. Contaminated solid-waste is buried or stored in 28 sites in the 200-Area plateau using 68 hectares (167 acres). Waste from reactor operations has been buried in 26 sites (26 ha) in the 100 Areas, and waste from laboratory operation has been buried in 11 sites (26 ha) in the 300 and 600 Areas.

During initial operations at Hanford, wastes that were contaminated or suspected of being contaminated with radioactivity were packaged and buried with no attempt made to segregate them as to type or level of radioactivity. Segregation was a matter of size. Large equipment was packaged in large wooden boxes; small miscellaneous materials were collected in cardboard cartons. Methods for estimating the quantity of plutonium contained in wastes were conceived in the late 1950's and measurement capability was developed in the 1960's.

Prior to 1968, essentially all of the waste buried was generated by the 200 Area operating facilities. Since 1968, waste generated by 300 Area operations also has been buried in the 200 Area sites. Beginning in December 1973, waste from reactor operations in the 100 Area was being sent to the 200 Area for burial. Small volumes of solid waste, generated by offsite DOE operations, have been and will continue to be buried here. During 1973, approximately 60 m³ containing less than 4 Ci of mixed radioactive materials were received from offsite.

Tables F-3J and F-3K give the estimated inventories of the burial grounds and Purex storage tunnels, respectively.

The dry wastes listed in Table F-3J have been characterized and are presented in Table F-3L.

These earlier detailed tables are being included to provide a clearer breakdown of waste buried in the 200 Areas. The following data are the latest available to the author at this time.

Besides normal operations several unplanned releases have occurred in the 200 Areas. These include:⁴¹

<u>DATE</u>	<u>LOCATION</u>	<u>DESCRIPTION</u>	<u>NUCLIDE & AMOUNT</u>
2-72	224-T Bldg. southeast side	During remodeling of 224-T Building for Pu storage, gross alpha contamination was found in soil on back side of building. The 224-T Building was originally constructed with vent lines from process tanks entering tile piping at ground level to rear of building. The jointed tile piping went to a common tile header which fed into 221-T Building cells. During the years of process operation, alpha-laden moisture seeped through pipe joints and grossly contaminated the subsoil. Excavations of urea showed soil contamination below the surface of an area 16 m long by 3.7 m wide by 3.7 m deep. A total of 139 drums of soil were removed from the zone containing approximately 70 g of plutonium. The zone is marked for underground contamination which remains below the header pipe. (It is believed a similar contaminated condition exists behind Building 224-B in 200-East Area.)	10 g Pu-239 residual
4-71	216-Z-1B waste line near 236-Z	The 216-Z-1B crib line from 234-S complex broke at location approximately 1.8 m south and 3.7 m west of southwest corner of 236-Z Building. An excavation 7.6 m long by 1.8 m wide by 2 m deep uncovered gross alpha contamination in soil to greater than 6 million d/min/100cm ² of surface. Approximately one hundred 208-liter drums of contaminated soil were removed and buried in 200-W Area Pu "storage for recovery" burial ground. Much contamination still remains under 1.8 m of clean soil.	10 g Pu-239
1-69	233-5	Plutonium-contaminated water backed up in 233-5 Filter House drain line and overflowed into a low spot in the ground directly north of 233-5 Filter House. An area of 125 m ² was affected. 21 m ³ of clean gravel were spread over the spill.	0.1 g Pu-239

TABLE F-3J

ESTIMATED DECAYED STATUS OF SOLID WASTE ^(a) BURIAL GROUNDS IN THE 200 AREAS THROUGH 1972 ⁹⁰

	Volume (m ³)	Uranium (G)	Plutonium (G)	Total (G)	Sr-90 (Ci)	Cs-137 (Ci)	Pu-238 (Ci)	Other Radioactive (Ci)
Burial Ground - 200 East Area								
Dry Waste No. 001	3.1×10^3	4×10^5	9×10^2	5×10^0	1×10^0	1×10^0	9×10^{-7}	
Dry Waste No. 12A	1.5×10^4	1×10^6	9×10^3	7×10^1	1×10^1	2×10^1	3×10^{-1}	
Dry Waste No. 12B	7.4×10^3	7×10^3	1×10^3	2×10^3	2×10^2	2×10^2	6×10^1	
Minor Construction No. 4	1.6×10^3	1×10^3	1×10^1	5×10^{-1}	1×10^{-1}	1×10^{-1}	2×10^{-6}	
Construction (No Number)	2.3×10^3	2×10^3	2×10^1	6×10^{-1}	1×10^{-1}	2×10^{-1}	3×10^{-5}	
222 Vaults	1.7×10^2	1×10^3	1×10^0	3×10^1	7×10^0	8×10^0	6×10^{-6}	
Industrial Waste No. 002	9.1×10^3	3×10^5	8×10^2	1×10^3	3×10^2	3×10^2	6×10^{-4}	
Industrial Waste No. 005	3.1×10^3	1×10^5	4×10^2	4×10^2	1×10^2	1×10^2	2×10^{-3}	
Industrial Waste No. 052	6.2×10^3	1×10^5	1×10^3	1×10^3	2×10^2	2×10^2	3×10^{-2}	
Industrial Waste No. 010	1.5×10^4	8×10^5	5×10^3	5×10^4	4×10^3	5×10^3	2×10^1	
TOTAL 200-E ^(b)	6.2×10^4	3×10^6	2×10^4	6×10^4	5×10^3	6×10^3	2×10^1	
Burial Ground - 200 West Area								
Dry Waste No. 001	7.1×10^3	7×10^5	9×10^4	9×10^0	2×10^0	2×10^0	2×10^{-6}	
Dry Waste No. 002	8.5×10^3	1×10^6	1×10^5	3×10^1	6×10^0	7×10^0	1×10^{-4}	
Dry Waste No. 003	1.1×10^4	7×10^7	7×10^4	5×10^1	1×10^1	1×10^1	3×10^{-3}	
222-T Vaults	6.8×10^1	3×10^2	3×10^{-1}	4×10^1	9×10^0	1×10^1	9×10^{-6}	
222-S Vaults	1.8×10^2	7×10^2	7×10^{-1}	2×10^2	5×10^1	6×10^1	6×10^{-3}	
Industrial Waste No. 001	1.4×10^4	9×10^5	2×10^3	2×10^3	6×10^2	4×10^2	1×10^{-2}	
Industrial Waste No. 002	1.9×10^4	2×10^6	6×10^3	2×10^6	3×10^3	4×10^3	1×10^2	
Dry Waste No. 03A	5.1×10^3	9×10^6	4×10^3	1×10^5	3×10^2	3×10^2	1×10^2	1×10^5
Dry Waste No. 04A	1.8×10^4	5×10^8	4×10^6	3×10^2	5×10^1	5×10^1	2×10^0	
Dry Waste No. 04B	8.5×10^3	4×10^6	2×10^6	2×10^5	3×10^2	3×10^2	9×10^1	2×10^5
Calsson - No. 1	6.2×10^0	2×10^5	5×10^2	1×10^2	1×10^1	2×10^1	2×10^0	
Calsson - No. 2	2.3×10^0		5×10^{-1}	3×10^0	4×10^{-1}	4×10^{-1}	8×10^{-2}	
Calsson - No. 3	4.8×10^0	1×10^5	5×10^2	3×10^3	3×10^2	4×10^2	8×10^1	
Calsson - No. 4	1.5×10^1	5×10^4	7×10^2	1×10^4	1×10^3	1×10^3	4×10^2	
Calsson - Alpha 1	5.7×10^0	4×10^4	1×10^3	4×10^4	7×10^2	7×10^2	4×10^2	1×10^4
TOTAL 200-W ^(b)	9.1×10^5	6×10^9	4×10^5	4×10^5	7×10^3	7×10^3	1×10^3	3×10^5
TOTAL 200 AREAS ^(b)	1.5×10^6	6×10^9	4×10^5	5×10^5	1×10^4	1×10^4	3×10^3	3×10^5

(a) The estimated inventories or concentration of radionuclides are reported to one significant figure to reflect the uncertainty in the data.

(b) Totals do not necessarily equal the sum of individual contributions because of rounding of the data.

TABLE F-3K
PUREX EQUIPMENT STORAGE TUNNELS^(a) INVENTORY^(b)
SEPTEMBER 30, 1973⁹⁰

	<u>Tunnel No. 1</u>	<u>Tunnel No. 2</u>	<u>Total</u> ^(c,d)
Volume, m ³	566	566	1,133
Total MFP, Ci ^(d)	6,000	20,000	30,000
Sr-90, Ci	1,000	40	1,000
Ru-106, Ci	1	7	8
Cs-137, Ci	1,000	40	1,000
Co-60, Ci	--	20,000	20,000
Pu-239, g	--	500	500

-
- (a) Currently the distribution of waste plutonium stored on the Hanford site is 40 percent in tanks, 39 percent in solid disposal sites and 21 percent in liquid disposal sites. The total amount of plutonium in these sites is estimated to be 940 kg $\pm$ 30 percent.
- (b) The estimated inventories (or concentrations) of radionuclides are reported to one significant figure to reflect the uncertainty in the data.
- (c) Totals do not necessarily equal the sum of individual contributions because of rounding of the data.
- (d) Curies shown are decayed to September 30, 1973.

TABLE F-3L
CHARACTERISTICS OF DRY WASTE BURIED
ON AN AVERAGE DAY (MAXIMUM)*⁵

<u>Source</u>	<u>No.</u>	<u>Container</u>	<u>Volume (ft³)</u>	<u>Activity, Ci</u>
222-S Bldg. (Hot cell)	4(5)	30-gal	30(38)	20. (70) F.P. ^(a)
	9(22)	Cardboard boxes	40(100)	0.6(3) F.P.
221-T	14(22)	Cardboard boxes	60(100)	0.1(2) F.P.
Laundry	9(22)	Cardboard boxes	40(100)	
Misc.	5(22)	Cardboard boxes	20(100)	
BNW	70(150)	Cardboard boxes	300(700)	< 5 (< 10)
HEDL	45(75)	Cardboard boxes	200(300)	

(a) Fission Products

* () designate maximum daily volume.

TABLE F-3M

ALPHA WASTE SHIPPED FROM HANFORD 300 AREA TO ARCHO³⁶

WASTE SOURCE	2ND HALF CY-1974			TOTAL CY-1974		
	No. of Drums Shipped	Average Weight, kg	Total Volume, m ³	No. of Drums Shipped	Average Weight, kg	Total Volume, m ³
HEOL DRUMS						
Fuel Fabrication Work	28	71	5.9	115	60	24.4
Fuel Reprocessing Work	50	62	10.6	99	57	21.0
Quality Control (Fuel Fab.)	18	55	3.8	62	53	13.1
Totals (Net Wt. 16,540 kg)	96	63 av	20.3	276	59 av	58.5
BNW DRUMS						
Fuel Fabrication Work	2	N.A. ^(b)	0.4	4	N.A.	0.8
Fuel Reprocessing Work	17	N.A.	3.6	27	N.A.	5.7
Biological Studies	8	N.A.	1.7	14	N.A.	3.0
Nuclear Safety & Safeguards	5	N.A.	1.1	5	N.A.	1.1
Environmental Studies	1	N.A.	0.2	1	N.A.	0.2
Totals (Net Wt. ~2100 kg)	33		7.0	51		10.8
HEDL VAULTS						
Fuel Fab. Glove Box	2	14,000	36	3	18,500 ^(a)	41
HEPA Filters	0	0	0	3	1,400 ^(a)	2.4
BNW VAULTS						
HEPA Filter Housings	0	0	0	3	23,600 ^(a)	8.5

(a) Previously reported data in Table 7 of HEDL-TME 74-61 listed the non-drum packaged waste as weighing 5000, 3000, and 26,000 pounds. These should have been 10,000, 3000, and 52,000 pounds, respectively.

(b) Not available

A total of 2-million curies have been buried in the 200 Areas; this has now decayed to 750-thousand curies. As of December 31, 1977,⁵ the following quantities of plutonium (Pu), uranium (U), and thorium (Th) have been buried with solid wastes:

	<u>Plutonium</u> (kgs)	<u>Depleted</u> <u>Uranium</u> (kgs)	<u>Enriched</u> <u>Uranium</u> (kgs)	<u>Normal</u> <u>Uranium</u> (kgs)	<u>Thorium</u> (kgs)
Stored Retrievable	107.9	19.5	52	70	916
Buried, Not Retrievable	357	589	178	1,030	358

3.1.3 300 AREA - REACTOR FUEL MANUFACTURING FACILITIES AND RESEARCH AND DEVELOPMENT LABORATORIES^{5,3,6}

There is a 2.84×10^5 liter capacity temporary storage facility in the 300 Area; however, all liquid wastes are currently transferred to the 200 Area for ultimate disposal.

Currently generated solid waste is sent to the 200 Area burial ground; however between 1944 and 1967 approximately $2.1 \times 10^4 \text{ m}^3$ were buried in the 300 Area. It is estimated that this buried waste contained 10 Ci of uranium contamination and 0.5 Ci of thorium contamination.

Three contractors are conducting research and development studies in the area: Westinghouse-Hanford, Battelle-Northwest, and United Nuclear Industries. Data on transuranic contaminated waste shipments for the Hanford 300 Area to Atlantic Richfield Hanford Company (ARCHCO) shows that for CY-1974, 327 drums of transuranic contaminated waste were shipped to ARCHCO. Total volume of the drummed waste was 280 kg/m^3 . In addition, 52 m^3 of nondrummed waste (i.e., gloveboxes, equipment, and other bulk items not shipped in drums) weighing 143,000 kg was shipped to ARCHCO for disposal in CY-1974.

The drummed waste shipments from HEDL and Battelle (BNW) during the second half of CY-1974 are summarized in Table F-3M. A total of 129 drums of waste were shipped to ARHCO for retrievable storage. The total weight of the HEDL drums was 6,240 kg and the total volume was 20.4 m³, giving an average density of 306 kg/m³. Data in Table F-3M are categorized according to the fuel cycle activity generating the waste. In addition to the 129 drums of waste, 35 m³ of non-drummed waste was also shipped to ARHCO for retrievable storage.

The TRU waste shipments during the CY-1974, CY-1975 and first half of CY-1976 are presented in Tables F-3N and F-3O. The volumes and their TRU content are reported for combustible and noncombustible wastes. For the first half of CY-1976, 16.1 m³ of combustible waste and 7.4 m³ of noncombustible waste were shipped for disposal. This compares with the first half CY-1975 shipments of 40.7 m³ containing 1,344 grams of plutonium.

The volumes of 300 Area fission product and transuranic alpha waste generated during six-month intervals from the latter part of FY-1971 to FY-1975 are shown in Table F-3P. The percent of the total waste that was classified as alpha waste varied widely from 4 to 30%. The high percentage of alpha waste for 1972 and 1973 included a large quantity of suspect alpha waste generated in biological studies. Improved segregation of this suspect waste was obtained through laboratory analyses, better recordkeeping, and overall improved management by the waste generators. It was determined that much of the suspect waste was non-alpha waste which resulted in the marked reduction in alpha waste generation shown during FY-1973 though other factors were involved. The present ratio of alpha to non-alpha wastes generated in the 300 Area appears to be nearly constant at about 10 percent. Correct segregation of wastes can be an important factor in reducing the amount of waste that is designated as alpha waste.

TABLE F-3N

TRU CONTAMINATED SOLID WASTE GENERATED IN HEDL FACILITIES³¹

<u>Waste Category</u>	<u>First Quarter CY-1975</u>	<u>Second Quarter CY-1975</u>	<u>Third Quarter CY-1975</u>	<u>Fourth Quarter CY-1975</u>	<u>Total CY-1975</u>	<u>Total CY-1974</u>	<u>Percent Change from CY-1974</u>
VOLUME:							
Combustible ^a , m ³	19.3	112.1	6.6	10.0	48.0	58.5	-18
Noncombustible ^b , m ³	<u>0</u>	<u>9.3</u>	<u>0.2</u>	<u>5.3</u>	<u>14.8</u>	<u>43.4</u>	<u>-66</u>
Total	19.3	21.4	6.8	15.3	62.8	101.9	-38
TRU CONTENT:							
Combustible ^a , g Pu	675	655	21	201	1,552	1,300	+19
Noncombustible ^b , g Pu	<u>0</u>	<u>14</u>	<u>12</u>	<u>1</u>	<u>27</u>	<u>4</u>	<u>+525</u>
Total, g Pu	675	669	33	202	1,579	1,304	+21

^a Drum Waste.^b Packaged Waste (such as obsolete equipment, concreted drum waste, etc.).

TABLE F-30
FROM 300 AREA TO 200 AREA (1/76-6/76)

Function	No. of Drums ^(a)	Average Weight kg	Total Volume m ³	Pu Content g
Combustible Waste				
Quality Control	25	59	5.30	203
Fuel Fabrication	22	54	4.66	167
Post Irradiation Testing	27	54	5.73	N.A. ^(c)
Maintenance and Miscellaneous	2	80	0.42	2.6
Subtotal	76	57	16.11	373
Noncombustible Waste ^(b)	2 Steel boxes	N.A. ^(c)	7.36	N.A. ^(c)
Total			23.5	373

(a) 0.2 m³ drums

(b) Packaged waste - obsolete equipment, concreted drum waste, etc.

(c) Not assayed due to high activity.

TABLE F-3P

300 AREA WASTE DISPOSAL SUMMARY (HEDL SHIPMENTS)³⁶

<u>Types of Waste</u>	FY-1971	FY-1972		FY-1973		FY-1974		FY-1975
	<u>Jan.-June</u>	<u>July-Dec.</u>	<u>Jan.-June</u>	<u>July-Dec.</u>	<u>Jan.-June</u>	<u>July-Dec.</u>	<u>Jan.-June</u>	<u>July-Dec.</u>
Transuranic Alpha (Vol. in m ³)	37.9	84.9	143.3	94.0	25.0	30.8	46.3	55.8
Fission-Product (Vol. in m ³)	241.3	228.9	342.2	374.0	586.6	475.2	469.0	525.6
Total Waste (Vol. in m ³)	279.2	313.8	485.5	468.0	611.6	506.0	515.3	581.4
Fraction Alpha Waste (Percent)	14	27	30	20	4	6	9	10

3.1.4 600 AREA¹⁶

The 600 Area is all other Hanford Reservation areas not included in the 100, 200, 300 or 400 Areas. In the past miscellaneous low-level solid radioactive wastes may have been buried in this area; however, accurate records were not kept and volumes and composition are not known.

3.2 SOLID WASTES

3.2.1 OVERVIEW⁵

About $1.90 \times 10^5 \text{ m}^3$ of contaminated solids have been buried on the 200 Area plateau since the start of chemical processing operations. These wastes consist of so-called "dry waste" (soiled clothing, laboratory supplies, tools, etc., packed in cardboard, wood or metal containers) and industrial waste (primarily items of failed process equipment packaged in sealed metal containers) and segregated in the burial trenches since May 1970. Contaminated solid waste is buried in 19 sites on the 200 Area plateau using 63 hectares. Some failed equipment from the Purex Facility is stored in tunnels on flat cars.

A review of the distribution of waste by categories defined based on plutonium concentration and general radiation levels, indicates that the waste in the 200 Area is generally low radiation level boxed dry waste. Tables F-3Q and F-3R present this information based upon Reference 18. It would appear that similar ratios would be appropriate for the present inventory of the 200 Area.

Solid radioactive waste buried in the ground is considered to be in long-term storage. An exception is that waste buried after April 30, 1970, which is suspected of containing transuranium nuclides; this is considered to be in interim storage (20 years). Large items of solid waste, such as failed equipment from locations where the presence of transuranium nuclides can be safely ruled out, are packaged in concrete boxes and buried in industrial waste burial trenches. Small items of failed equipment and trash-type contaminated waste, from locations where the presence of transuranium nuclides can be safely ruled out, are packaged in cardboard cartons, wood boxes, or steel or fiber drums and buried in the so-called "dry waste" trenches.

TABLE F-3Q

ESTIMATED VOLUMES OF BURIED TRANSURANIUM SOLID WASTE
AT HANFORD 200 AREA BURIAL GROUNDS THROUGH 1969⁵

Type	Surface Dose Rate (mrem/hr)	Pu ^a Content (kg)	Beta Activity As Charged (Ci)	Volume As Charged (m ³)
1. Low radiation level, boxed dry waste	< 10	298	2,000	48,100
2. High radiation level, boxed dry waste	> 10	34	8,000	19,800
3. Failed metal equip- ment (boxed or bagged)	> 10	<u>23</u>	<u>620,000</u>	<u>70,750</u>
Total		355	630,000	138,650
Boxed large equipment in Purex rail tunnels	> 1,000	> 1	> 10,000	1,550
Cribs and surrounding soil with Pu $\leq$ 20 Ci/kg	< 10	260	3,100,000	

-
- a Plutonium in use at Hanford typically is 94% Pu-239 and 6% Pu-240 and has specific alpha activity of 0.07 Ci/g of the element.

TABLE F-3R

ESTIMATED SOLID WASTE BURIALS IN HANFORD 200 AREAS DURING CY 1969

	<u>Percent Burnable</u>	<u>Surface Dose Rate (mrem/hr)</u>	<u>Total Pu (g)</u>	<u>Non-Pu Beta- Gamma Activity (Ci)</u>	<u>Total Volume (m³)</u>
1. Pu contaminated, low radiation level, boxed dry waste ^a	75	< 2	4,000	-	790 ^c
2. Pu contaminated, intermediate radiation level, boxed dry waste ^a	75	10-100	500	130	630
3. Pu contaminated, high radiation level, boxed dry waste ^a	50	>100	400	28,000	1,130
4. Failed metal equipment, high radiation level (boxed or bagged) ^b	10	>10	200	65,000	540
5. Low transuranic, low radiation level, boxed dry waste ^a	50	<10	-	34,000	680 ^d
6. Low transuranic, high radiation level, boxed dry waste ^a	10	>10	-	40,000	1,700
Total			5,000	168,000	5,470 ^e

^a Waste in cardboard boxes, wood boxes, and metal drums consists of cloth, paper, plastic, tools, glass filters, dirt, and miscellaneous small hardware.

^b Consists primarily of failed metal equipment in wood or concrete boxes.

^c Includes 70 m³ from Rocky Flats.

^d Includes 42 m³ from Lawrence Radiation Laboratory.

^e Total volumes were 4,300 m³ in 1967 and 4,950 m³ in 1968.

The various waste containers used provide containment of nontransuranic radioactive contamination and minimize radiation exposure to personnel during temporary storage, handling, shipment, and burial operations. Once buried, no reliability is placed on the containers for confinement or retrievability of these materials. Although burial of the waste containers by backfilling the trench is normally done at the close of the day's receipt of solid waste, waste trenches are immediately backfilled whenever the dose rate at the edge of the trench reaches 100 mR/hr. Solid waste is normally covered with three to six meters of earth to prevent uptake of radionuclides by plant life or disturbance by burrowing animals. An exception, waste contained in concrete boxes or small drums, may have a minimum dirt cover of four feet provided that radiation levels at grade are less than 1 mR/hr. Periodic routine surveillance of filled burial trenches is provided to assure that the burial grounds are maintained to meet existing standards.

Solid waste, containing or suspected of containing transuranium nuclides above 10nCi/g buried after April 30, 1970, is packaged and buried in compliance with DOE-M-0511. This directive states that "such wastes shall be segregated from other solid wastes and shall be packaged and buried so that they can be readily retrievable, as contamination-free packages, within an interim period of 20 years."

Formerly, the waste was packaged in steel drums and steel or concrete boxes and buried in special trenches. Subsequent evaluation of steel drums directly buried in Hanford soils indicated that failures could occur in less than 20 years and retrieval, as contamination-free packages, might not be possible. Two alternatives to direct burial were implemented on a test basis, either of which will protect the containers from direct contact with the soil and will permit ease of retrieval. One method consists of stacking steel drums on a pad and covering with plywood and nylon sheeting prior to covering with four feet of earth. The other method is placing the drums in a concrete "vee" trench, placing a metal cover over the top and covering with earth.

Small items of transuranic-bearing solid waste, containing fission products capable of generating high dose rate gamma emissions, are packaged and shipped in shielded casks. This waste is discharged from the casks to coissons, which can be retrieved.

Some of the Purex processing equipment is so large and becomes so contaminated in service, that its transport to a burial ground would require abnormally high exposure of operating personnel. Such equipment is placed on flatcars and pushed into the solid waste storage tunnels appended to the Purex Plant.

3.2.2 TRU BURIAL FACILITIES⁴¹

Asphalt pads are currently being used for the retrievable storage of low-activity TRU wastes. The wastes are packaged in 208-l drums or in steel boxes and stacked on the pad in 7.3-m wide x 7.3 m-high modules. Flame-retardant plywood is placed between the layers, and plywood is placed on top of the modules. The completed module is covered with PVC laminated nylon sheet and 1.3 m of soil overburden. TRU waste generated between June 1972 and March 1973 was placed in 208-l drums and stored in a concrete "vee" trench. Waste drums were placed at a 45° angle inside the trench. A steel cover was then placed over the trench and covered with 1.2 m of soil as an overburden.

Underground caissons are used for storage of small packages of high-activity TRU waste. The waste is packaged in 3.8-liter or 18.9-liter paint cans with clips or resin to hold the lids in place. The caissons are reinforced concrete cylinders buried 4.0 m below the surface. The caissons have 0.9-m-diameter chutes, offset or convoluted to reduce surface radiation exposure.

Extremely large, heavy, or highly contaminated waste equipment is stored on railroad flat cars in two railroad tunnels. The contamination may be TRU or non-TRU and is usually associated with activation products. One tunnel is filled with eight flat cars and is now inactive. The second tunnel is 515 m long and can store about 42 flat cars. At present (1977), nine cars of waste are stored in this tunnel.⁴¹

TRU waste unsuitable for pad or caisson storage due to size, security requirements, or surface radiation is stored in alpha trenches. This waste is packaged in fiberglass-reinforced polyester-covered plywood boxes, reinforced concrete boxes, metal boxes, or 208-l drums. Once placed in the trench, the waste is covered with 1 meter of soil. These trenches are also used for storage of TRU and non-TRU classified waste in 208-l drums. The drums are stacked vertically and covered with a plywood cover. Completely filled trenches are covered with a PVC laminated nylon sheet and 1.2 m of soil. Classified wastes are covered on the day they are placed in the trenches.

Currently used packaging for radioactive wastes ranges from cardboard boxes sealed with tape to flanged steel containers with reinforced concrete shielding.

The fissile content of each container accepted for burial is measured before or subsequent to packaging, and the amount of fissile material is recorded on the shipping papers. All of the wastes are monitored for surface contamination at their point of origin. Dose rates at the surface of a package are determined and recorded. All packages must be free from external contamination. Generally, radiation rates at the surface of a package are less than 10 mR/hr. High-activity wastes are transported in shielded metal containers.

The accumulated volume of TRU waste at Hanford is approximately 162,360 m³, which contains 638 kg of TRU isotopes.⁵ Table F-3S presents a breakdown of this data.

TABLE F-35
SUMMARY OF TRU WASTE BURIED/STORED AT HANFORD⁵

	<u>Previous Years</u>	<u>1977</u>	<u>1978</u>	<u>Total</u>
Buried TRU Waste				
Volume (m ³)	155,000 +200% -50%	0	0	155,000 +200% -50%
Kg TRU	365	0	0	365
Stored TRU Waste				
Volume (m ³)	6500 ±10%	600 ±10%	700 ±10%	7800 ±10%
Kg TRU	63.4	14.6	5.2	83.2

4.0 SAVANNAH RIVER PLANT

Since 1953, the Savannah River Plant (SRP), near Aiken, South Carolina, has been producing special nuclear materials, primarily plutonium and tritium, for defense purposes. The SRP facilities were constructed and initially operated for the Atomic Energy Commission. Operation since 1975 has continued under the direction of the Energy Research and Development Administration (ERDA) and presently the Department of Energy (DOE).

The SRP operations produce high-level radioactive waste in the chemical processing of fuel and target elements after irradiation in the SRP nuclear reactors. This waste is stored as an alkaline liquid with a precipitated sludge until the decay heating has abated appreciably. Then the supernatant liquid is converted to salt cake to reduce volume and mobility. The storage facilities are large underground tanks that are engineered to provide reliable isolation of waste from the environment.

At SRP, solid wastes contaminated with transuranium nuclides are generated primarily in the final purification areas (B lines) of the two separations plants. Wastes from the B line of the F plant contain principally Pu-239 while those from the B line of the H plant contain principally Pu-238. All wastes are buried in a local on-site burial ground. Through calendar year 1965, most waste was in cardboard cartons with relatively little provision for confinement and retrievability. Since that time, coincident with large scale processing of Pu-238, most wastes containing > 0.1 curie of Pu-239 or Pu-238 per package have been encapsulated in concrete before burial to immobilize the waste and to facilitate possible future retrieval. As of 1970, the mandated 10 nCi/g criteria for retrievable storage of TRU-contaminated waste has been utilized.

4.1 SOLID RADIOACTIVE WASTE STORAGE SITE (BURIAL GROUND)

4.1.1 FACILITY DESCRIPTION^{5, 90}

One centrally located solid radioactive waste storage site is used to store all radioactive solid waste produced at the Savannah River Plant as well as

occasional special DOE shipments from off-site. This storage site occupies 78.9 hectares approximately 10 km from the nearest plant boundary.

The solid waste storage site is divided into sections for accommodating various levels and types of radioactivity in waste materials: transuranium (TRU) alpha waste, low beta-gamma waste, and high beta-gamma waste (high beta-gamma and low beta-gamma solid radioactive wastes are those containing primarily beta-gamma emitters which are segregated according to radiation measurement). Examples of the materials in storage include:

- Contaminated equipment - obsolete or failed tanks, pipes, jumpers, and other process equipment from the fuel separations plants.
- Reactor and reactor fuel hardware - fuel components and housings not containing irradiated fuel and spent deionizer resins.
- Spent lithium-aluminum targets - the waste target alloy after tritium has been extracted.
- Oil from gas displacement pumps in the tritium facilities - before burial, oil is placed in drums containing an absorbent material.
- Mercury from gas pumps in tritium facilities - before initiation of recycling in 1968, deteriorated and contaminated mercury was buried with primary containment by one-liter polyethylene bottles.
- Incidental waste from laboratory and production operations - small equipment, spent air filters, clothes, analytical waste, decontamination residues, plastic sheeting, and gloves.
- Occasional shipments from offsite - tritiated waste from the Mound Facility, Pu-238 process waste from Los Alamos Scientific Laboratory (LASL) and the Mound Facility, and debris from two U.S. military airplane accidents in foreign countries.

4.1.2 STORAGE MODES⁹⁰

There are two types of facility storage modes at SRP: pads and trenches. Waste emplacements are covered promptly with soil to reduce radiation exposures and the potential of fire and wind-blown contamination. When a trench is nearly filled with waste, it is backfilled with soil.

4.2 TRU WASTE

Transuranic waste was originally buried in plastic bags and cardboard boxes in earthen trenches designated specifically for this waste. At Savannah River beginning in 1965, TRU waste was segregated according to TRU content into two categories, retrievable and nonretrievable, and additional containment was added for retrievable waste. Waste containing greater than 0.1 Ci per package was placed in prefabricated concrete containers and then buried. These retrievable containers were 1.8 meters in diameter by 2 meters high. Waste that did not fit into the prefabricated concrete containers was encapsulated in concrete. Transuranium waste from the Savannah River Laboratory was buried in cubical concrete containers. Waste containing less than 0.1 Ci per package was buried unencapsulated in alpha trenches. A total of 242,900 m³ was buried under these practices.⁹⁰

In 1974, the storage procedures were modified to reflect new DOE criteria governing retrievable storage of solid transuranic waste. Transuranium wastes contaminated to greater than 10nCi/g are now stored, protected from contact with water-saturated soil, in containers that can be retrieved intact and free of external contamination for at least twenty years from the time of storage. Combustible and noncombustible wastes are stored in separate containers. Polyethylene-lined galvanized drums are used as the primary container; waste packages containing more than 0.1 Ci are additionally protected by closure in concrete cylinders. In accordance with DOE Manual 0511, canyon equipment and other bulky wastes contaminated with transuranium nuclides to greater than 10 nCi/g and also intensely contaminated with gamma emitters are stored directly in earthen trenches protected from contact with water-saturated soil. Waste is placed in the trench and covered with soil soon after its placement.

4.3 SOLVENT³³

Spent (degraded) solvent from the chemical separations area has been stored at the burial ground since 1955. The burial ground is situated between the two chemical reprocessing areas. There are currently 22 solvent storage tanks, two of which are no longer in service due to past leaks. Before 1972, the solvent inventory was reduced by burning. Burning resulted in very low emission of airborne radioactivity but did release copious quantities of black smoke. Burning operations ceased in 1972 due to environmental restrictions on smoke emissions. During 16 years of solvent incineration, about 1,400,000 liters of spent solvent was burned.³³

The current inventory of solvent is shown in Table F-4A. The 562,000 liters of solvent contains 45.6 Ci of TRU nuclides and about 46.2 Ci of fission products. Pu-238 is the predominant TRU nuclide in the solvent, and the primary fission product is Ru-106.

4.4 MISCELLANEOUS RADIOACTIVE WASTE

In addition to the waste sources discussed above, certain miscellaneous radioactive waste sources exist at SRP. Most of the locations of miscellaneous waste disposal are marked or are within controlled areas. These include locations containing radioactivity from spills, equipment failure, etc., but not locations containing activity from ongoing operations. Table F-4B presents those areas which are TRU-contaminated.

4.5 WASTE CHARACTERISTICS⁵

The solid wastes at SRP can be divided into six categories. These will be individually discussed to provide insight into SRP solid waste characteristics.

4.5.1 CARDBOARD BOXES BURIED IN TRENCHES

These wastes generally originate in glove enclosures and their associated laboratories. They are characterized by relatively high combustible content and

TABLE F-4A
DEGRADED SOLVENT INVENTORY³³

<u>Tank</u>	<u>Solvent Volume, liters</u>	<u>TRU Content, Ci</u>	<u>Major TRU Nuclide (alpha percent)</u>	<u>Fission Product Content, Ci</u>
1	23,060	0.5	Cm-244 (66%)	3.5
2	30,330	27.0	Pu-238 (88%)	5.2
3	31,750	0.02	Pu-238 (70%)	0.013
4	1,040	0.006	Pu-238 (67%)	0.001
5	36,970	17.0	Cm-244 (93%)	11.0
6	73,140	0.06	Cm-244 (80%)	0.26
7	7,840	0.01	Pu-238 (77%)	0.014
8 ^a	Contains no liquid			
9	15,630	0.006	Pu-239 (30%)	2.0
10	5,160	0.003	Pu-238 (50%)	0.05
11	23,40	0.2	Cm-244 (56%)	0.6
12	19,000	0.15	Pu-238 (63%)	0.5
13	43,470	0.08	Pu-238 (97%)	0.2
14	85,800	0.15	Pu-238 (98%)	0.15
15	28,540	0.18	Cm-244 (63%)	1.4
16	7,560	0.003	-	0.0007
17 ^a	380	0.013	Pu-239 (60%)	0.29
18	3,650	0.007	Pu-239 (30%)	3.5
19	73,710	0.1	Cm-244 (65%)	2.6
20	29,860	0.06	Pu-238 (60%)	15.0
21	11,720	0.001	-	0.013
22	10,960	0.0008	Cm-244 (70%)	0.004
Total	56,2970	45.6	Pu-238 (54%)	46.2

^a Tank no longer in service due to past leak.

TABLE F-4B
MISCELLANEOUS RADIOACTIVE WASTE IN SOIL²³

<u>Date</u>	<u>Location & Brief Description</u>	<u>Disposal Area (ft²)</u>	<u>Treatment of Area After Release</u>	Remaining		
				<u>Volume (ft³)</u>	<u>Curie Content</u>	<u>Type Waste</u>
2/54	Plutonium solution released thru the F-Area sanitary sewer system from Building 221-FB Line	2X10 ³ outside F Area fence	Residue from treated sewage (Building 607-IF) deposited in trench and covered	10 ³	0.02	Pu
6/69	Curium-244 aerosol released from Building 773-A cell #12 exhaust stack of the High Level Caves	1X10 ⁵	None	-	0.02	α, primarily Cm-244
12/71	Curium-244 released from the High Level Caves separator pit to the ground area underneath E-Wing Building 773-A.	1X10 ³	None	-	6	α, γ, primarily Cm-244

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relatively low external radiation level. Through 1965 a total of about 2 kg of Pu-239 was buried in this manner. Since that time these wastes have been limited to no more than 0.1 curie per package on the basis of a passive gamma analysis. The most common cardboard box is a 38 cm cube. Other sizes that are used include 61- and 76-cm cubes. Approximately 70 percent of all waste buried through 1970 consisted of cardboard boxes. It is estimated that if this waste were retrieved the volume of the restored waste would be triple the original volume.

4.5.2 BOXED OR DRUMMED WASTE IN CONCRETE CULVERTS

Since 1965 cardboard cartons or metal drums containing < 0.1 curie per package (principally combustible waste having low external radiation level) have been placed in large concrete containers before burial. These containers are made by pouring a concrete bottom in a standard culvert section. After the container is filled the top is sealed in place using epoxy resin. These concrete containers weigh about 7,000 kg. This waste form makes up 2.6 percent of the total waste buried through 1970.

4.5.3 LARGE EQUIPMENT IN MONOLITHIC POURS

Large items of equipment significantly contaminated with transuranium nuclides - encased in wooden or metal boxes to promote safe transport to the burial ground - are further encased in concrete using the earth as a form. After a concrete footing is poured in an earth excavation of suitable size, the boxed waste is placed on the footing and concrete is poured all around and on top. A total of about 200 such burials have been made through 1970 (approximately 3.3 percent of total waste buried). Outer dimensions of the concrete pours have varied from about 1.2 m cubes to about 6-m cubes.

Tables F-4C and F-4D present the Pu-239 contamination level on an exhumed Purex feed adjustment tank. This may provide some guidance on TRU content of similar components.

TABLE F-4C
CONTRIBUTION OF INDIVIDUAL SURFACES TO RESIDUAL
PU-239¹ IN EXHUMED PUREX FEED ADJUSTMENT TANK³³

<u>Vessel Surface</u>	<u>Area ft²</u>	<u>Total Pu-239 on Surface, dis/min</u>	<u>% of Total Pu-239 on Vessel</u>
Wall, inside	201	7.0×10^7	7.4
Wall, outside	201	4.0×10^7	4.2
Bottom, inside	50	1.1×10^7	1.2
Bottom, outside	50	3.8×10^7	4.0
Top, inside	43	5.4×10^6	0.6
Top, outside	56	5.2×10^8	54.7
Top flange	5.5	9.6×10^6	1.0
Annular opening	7.1	3.3×10^6	0.3
Coils	307	6.1×10^7	6.4
Trunion guides and supports	11	1.9×10^6	20.0

¹ Total = 9.5×10^8 dis/min
Total Pu-239 = 7.0 mg
Total Pu-239 Pu = 4.3×10^5 nCi

TABLE F-4D

PU-239 CONTAMINATION IN EXHUMED PUREX FEED
ADJUSTMENT TANK, ON WEIGHT BASIS³³

Total Empty Weight	7,300 lb
Weight of Coils	<u>1,820</u>
TOTAL WEIGHT	9,120
	or 4.14×10^6 g

$$\text{nCi/g} = \frac{4.30 \times 10^5 \text{ nCi}}{4.14 \times 10^6 \text{ g}} = 0.1 \text{ nCi/g}$$

4.5.4 SRL TRANSPLUTONIUM WASTES

Wastes from the Savannah River Laboratory (containing principally americium, curium, and californium) are placed in 0.6 m square by 1 to 1.2 m long concrete boxes before burial. These boxes weigh about 450 kg. The wastes originate primarily in glove boxes and process cells. Most of the waste is combustible and the radiation level at the surface of the concrete box is generally in the range of 10-1000 mrem/hr. This waste makes up about seven percent of the waste material buried.

4.5.5 LARGE EQUIPMENT DIRECTLY BURIED IN TRENCHES

Prior to 1966 large items of process equipment having significant transuranium contamination were enclosed in wooden or metal boxes and buried directly in trenches without concrete encapsulation. At present this practice is permitted only for those items of process equipment that have been substantially decontaminated of transuranium materials before burial. In general, this equipment has a relatively high external radiation level from fission product contamination. This waste makes up about five to six percent of all the waste buried through 1970.

4.5.6 RETRIEVABLE 208-LITER DRUMS

Drummed wastes received from off-site are presently stored in the Savannah River burial ground. These waste containers (approximately 11.2 percent of the waste buried through 1970) are in relatively good physical condition and may be considered retrievable for a period of at least several years.

4.6 QUANTITY OF WASTE^{5, 90}

The estimated volume and curie content of waste buried in 1975 is listed in Table F-4E. The total estimated volume and curies of waste in storage are summarized in Table F-4F. Table F-4G presents the maximum quantities of radionuclides stored at a single location.

TABLE F-4E
SOLID WASTE BURIED IN 1975 FROM SRP OPERATIONS ^(a)

Radio-nuclide	Earthen Trenches		Retrievable Storage		Total	
	Volume, m ³	Curies	Volume, m ³	Curies	Volume, m ³	Curies
Cf-252	4	--	40	1.5	44	1.5
Cm-244	400	--	10	50	410	50
Pu-238 ^(c)	420 ^(b)	(b)	40	4,190	460	4,330
Pu-239 ^(d)	770		80	160	850	160
	--					
Np-237	50	--	10	0.4	60	0.4
U-238	540	1.2	1	--	541	1.2
U-235	220	0.002	3	--	223	0.002
H-3	1,000	59,500	--	--	1,000	59,500
Fission Products	6,470	65,500	2	90	6,472	65,500
Activation Products	320	71,500	10	400	330	71,900
Totals	10,200 ^(b)	197,000 ^(b)	200	4,900	10,400	202,000

(a) Volumes and curies rounded to no more than 3 significant figures.

(b) These totals do not include a process vessel vent filter in a 23 m³ steel tank estimated to contain less than 140 Ci Pu-238 which is buried in an earthen trench. This filter also contains an estimated 6,000 Ci of fission products. Such equipment is excluded from surface storage by DOE Manual Chapter 0511.

(c) The Pu-238 waste typically contains 81 percent Pu-238, 15 percent Pu-239, three percent Pu-240 and one percent Pu-241.

(d) The Pu-239 waste typically contains about 94 percent Pu-239 and six percent Pu-240.

TABLE F-4F

SOLID RADIOACTIVE WASTE STORED IN BURIAL GROUND ^(a)
(THROUGH 1975)

Radio-nuclide	Earthen Trenches		Retrievable Storage		Total		Corrected for Decay, Ci ^(b)
	Volume, m ³	Curies	Volume, m ³	Curies	Volume, m ³	Curies	
Cf-252	400	0	170	12	570	12	7
Cm-244	6,000	1,670	800	40,600	6,800	42,300	33,300
Pu-238	4,600	3,380	560	257,000	5,160	260,000	251,000
Pu-239	12,600	510	600	1,700 ^(d)	13,200	2,210	2,210
Np-237	500	0.2	30	2.8	530	3	3
U-238	25,600	25	1	0	25,600	25	25
U-235	2,600	0.2	10	0	2,600	0.2	0.2
H-3	10,300	3,540,000	10	146,000	10,300	3,690,000	2,550,000
Fission Products	164,000	678,000 ^(f)	50	25,400	164,000	703,000	58,000 ^(f)
Activation Products	22,900	3,250,000 ^(g)	10	450	22,900	3,250,000	970,000
Totals	250,000	7,470,000	2,240	471,000	252,000	7,950,000	3,860,000

(b) Corrected for decay through 1975.

(c) Includes 230,000 curies in DOE waste received from off-site.

(d) Includes 150 curies in DOE waste received from off-site.

(e) Includes 1,470,000 curies in DOE waste received from off-site.

(f) Cs-137 and Sr-90 are estimated at 2% each of the total before decay and 40% each after decay.

(g) Co-60 is estimated to be 98% of this total.

TABLE F-4G
MAXIMUM QUANTITIES OF BURIAL GROUND RADIONUCLIDES
STORED AT SINGLE LOCATIONS

<u>Isotope</u>	<u>Condition</u>	<u>Maximum Quantity, Ci</u>
Pu-238	High-level transuranium cabinet waste in lined, steel drums. Fourteen drums are placed in a concrete container. Container is open prior to sealing.	2000 per concrete concrete container (550 per drum)
Pu-239	Same as above	23 per concrete container (12 per drum)
Cm-244	High-level transuranium waste from SRL in lined steel drums. Fourteen drums are placed in a concrete container. Container is open prior to sealing.	1800 per concrete concrete container (500 per drum)

The present inventory of TRU wastes at SRP as of October 1, 1978 is given in Table F-4H.

The estimated annual generation rate of transuranium solid waste under the present and proposed policies is summarized in Table F-4I. The waste forms generally reflect use of the retrievable criteria for waste with TRU control greater than 10 nCi/g.

4.7 INCIDENT EXPERIENCE

Operating incidents at the Savannah River burial ground from 1955 to 1975 are listed in Table F-4J. Incidents are classified according to whether activity was released to the air, released to surface water, or migrated through the soil.

Incidents in which radioactivity was dispersed by means of airborne material were the most numerous, comprising about two-thirds of the incidents. They involved burying contaminated obsolete equipment, sandblasting to decontaminate reusable equipment, and burning contaminated organic solvents. Burials included jumper pipes, parts of evaporators, and miscellaneous equipment. The equipment usually contained loose particulate material and at times contained small amounts of contaminated liquid.

A typical equipment burial incident would begin with a poorly wrapped or mishandled load of contaminated equipment on its way to a burial trench. Particulate material would escape from the wrapped load, carrying with it a small quantity of radioactivity and depositing it over a small area of burial ground. This would result in a radiation survey and consequential decontamination. The area would be decontaminated, usually by bulldozing the contaminated surface soil into a waste trench.

All eight radioactive waste fires were considered minor incidents. Seven occurred in trenches of exposed, low-level radioactive waste and were of undetermined origin. Spontaneous combustion was considered a probable cause. None burned for longer than one hour, and all were extinguished by smothering

TABLE F-4H
SUMMARY OF TRU WASTE BURIED/STORED AT SRP

	<u>Previous Years</u>	<u>1977</u>	<u>1978</u>	<u>Total</u>
Buried TRU Waste				
Volume (m ³)	30,720 (<u>±5%</u>)	0	0	30,720 (<u>±5%</u>)
Kg TRU	7	0	0	7
Stored TRU Waste				
Volume (m ³)	1400	147	189	1736
Kg Pu	44	{ 3.3	{ 6.3	{ 58.6
Kg Other TRU	5			

TABLE F-4
ESTIMATED ANNUAL GENERATION RATE
OF TRANSURANIUM SOLID WASTE AT SRP

Category	Waste	Radioactivity (Ci)			Volume (m ³)
		Pu-239	Pu-238	Am-241	
1. Cardboard boxes for direct trench burial		10 ⁻⁴	10 ⁻⁴	--	500
2. Burial of drummed waste in concrete culverts		180 (3600 g)	1,400 (85 g)	--	650
3. Large equipment in monolithic concrete pours		40	180	--	65
4. SRL 0.6 x 0.6 x 1 m concrete boxes		20	--	12,000	300
TOTAL		240 (4000 g)	1,600 (100 g)	12,000	1,515

TABLE F-4J
BURIAL GROUND INCIDENTS

<u>Type</u>	<u>Pre- 1960</u>	<u>1960- 1964</u>	<u>1965- 1969</u>	<u>1970- 1975</u>	<u>Total</u>
Airborne contamination					
Equipment burial	14	17	12	0	43
Sandblasting ^a	4	2	0	0	6
Solvent burning ^a	0	14	4	0	18
Fire	2	3	2	1	8
Waterborne contamination					
Flooded trench	0	0	1	1	2
Migration through soil					
Solvent spills	4	3	2	0	9
Water spills	0	1	0	1	2
Vegetation uptake	0	0	4	6	10
Unintentional exhumation	1	2	0	0	3
Unintentional burial	6	1	1	2	10
Overall annual incident rate	5	8	5	2	

^a Discontinued

with backfill soil or by hosing with water. Six fires produced no detectable dispersion of radioactivity beyond the immediate trench. The other two fires produced trace amounts of contamination over small areas inside the burial ground.

Organic solvent spills have accounted for nine minor incidents. All occurred either in transporting the contaminated solvent to the solvent storage area in the burial ground or in transferring the solvent from a tank truck into storage tanks. In all cases, contamination was confined to the immediate area.

Uptake by vegetation is one of the most common routes for dispersal of radioactivity. In each of the ten incidents or observations, radioactive vegetation was removed to a burial trench together with the contaminated soil in which it was growing. Activity levels were of the order of a microcurie per gram of vegetation or less. Essentially all of the observed incidents were due to insufficient soil cover over early trench burials, which allowed deep-rooted vegetation to reach far enough into the soil to pick up buried activity. Present burial standards require at least 1.2 m of soil cover to minimize vegetation uptake.

5.0 MOUND FACILITY

5.1 TRU WASTE CHARACTERISTICS⁵

The polonium (an alpha emitter) and plutonium contaminated solid wastes have been generated at Mound and can be put into five categories: Pu-238 low-level waste (LSA), Pu-238 high-level waste (non-LSA), solidified-clarifloculator-sludge (non-LSA), absorbed Pu-238 acid and caustic wastes (non-LSA), Po-210 low-level waste (LSA), and Po-210 high-level waste (non-LSA). The discard limits for Pu-238 are shown in Table F-5A.

5.1.1 PU-238 LOW-LEVEL WASTE

The Pu-238 low-level waste is composed of contaminated equipment, floor sweepings, used plastic shoe covers, discarded clothing, paper towels, swipes, cleaning rags, blotter paper, etc., from rooms in which glove box and hot cell operations are conducted. Less than 0.35 mCi Pu-238 per m³ is associated with this class of waste. About 60 vol % is collected and shipped in crates and about 40 vol % in drums. Drums are the preferred containers for all waste types. Crates are used only for items that are too large for drums. This waste is considered about 90 volume percent combustible.

5.1.2 PU-238 HIGH-LEVEL WASTE

The Pu-238 high-level waste consists of miscellaneous burnable and nonburnable solids (metal, plastic, glass, etc.) from glove boxes and hot cells used in plutonium production. As shown in Table F-5A, burnables have less than 0.110 g of Pu-238 per 3.78-liter carton, nonburnables less than 0.511 g per 3.78-liter carton, and glass filters less than 0.613 g per one 25.4 cm filter. Generally, this waste is collected in 3.78-liter cardboard cartons with a plastic coating on the inside surface. Three cartons are sealed in a PVC bag as they are removed from the production line. Four of these sealed units, i.e., 12 cartons, are placed in a 114-liter steel drum. However, 208-liter drums and plywood crates are used for

TABLE F-5A

DISCARD LEVELS OF PU-238 AT MOUND FACILITY (1970)

<u>Residue</u>	<u>Pu-238 Discard Level^(a)</u>
Burnables	0.110 g per 3.78 liter can
Nonburnables	0.511 g per 3.78 liter can
Ion exchange raffinate	0.006 g/liter
Evaporator condensate	0.006 g/liter
Caustic bubbler and scrubber solution	0.012 g/liter
Wash solution	0.006 g/liter
Filtrate solution	0.013 g/liter
Spent resin	0.358 g per 3.78-liter can
Glass filters	0.613 g per one 25.4 cm filter
In-line filters	0.461 g per one 25.4 cm filter
Drummed residues	1.85 g per 208-liter drum

a. A typical isotopic composition of the Pu-238 is:

Pu-238	80.94%
Pu-239	15.26%
Pu-240	2.88%
Pu-241	0.79%
Pu-242	0.12%
Pu-236	1.2 ppm

larger items such as contaminated equipment and glove boxes. Large items usually are filled with self-foaming plastic and placed in a crate which is lined with a polyethylene sheet. The open space in the crate is then filled with the self-foaming plastic. This waste is considered to be about 75 volume percent combustible.

5.1.3 SOLIDIFIED-CLARIFLOCCULATOR-SLUDGE

The class of waste called solidified-clariflocculator-sludge is generated when sludge from the contaminated wastewater treatment facility is mixed with cement. The sludge contains the barium, calcium, and iron salts formed as well as the flocculating agent and activated charcoal added during the scavenging-precipitation treatment of the contaminated wastewater from drains, sinks, showers, the laundry, etc. The sludge containing 30 to 35 wt % solids is mixed with 11 to 22 kg of portland cement in a 114-liter steel drum. Less than 3.53 Ci Pu-238 per m³ is associated with this waste.

5.1.4 ABSORBED PU-238 ACID AND CAUSTIC WASTE

Absorbed Pu-238 acid and caustic waste, generally noncombustible, are generated when acidic and caustic solutions from the Pu-238 production lines are reacted with and absorbed by a pulverized clay. In the treatment process, 45 liters of acidic or caustic solution are mixed with 110 liters of ground, dried clay in a 110-liter polyethylene jug (DOT Specification 2S) which is held within a 125-liter steel drum (DOT Specification 17H). The jug is outgassed for 24 hours with acidic solutions and eight hours with the caustic solutions before sealing. The radioactivity associated with this waste is less than 700 mCi per m³.

5.1.5 PO-210 LOW-LEVEL WASTE

The Po-210 low-level waste category consists of contaminated equipment, floor sweepings, used plastic shoe covers, discarded clothing, paper towels, swipes, cleaning rags, etc., from rooms in which glove box and hot cell operations for polonium production are carried out. The Po-210 content of this waste is less

than 0.35 mCi per m³. The waste is collected and shipped in 114-liter steel drums (DOT 17H); 208-liter steel drums (DOT 17C); and plywood crates (DOT 19A or 19B).

5.1.6 PO-210 HIGH-LEVEL WASTE

The Po-210 high-level waste is generated in the glove boxes and hot cells used in polonium production. It consists of the contaminated metal, glass, plastic, cloth, paper, etc., removed from the boxes and cells. The polonium associated with these materials amounts to less than 1,765 Ci/m³. The waste is collected in 114-liter steel drums. This drum is shipped in a 208-liter steel drum used as an overpack.

6.0 OAK RIDGE NATIONAL LABORATORY (ORNL)⁵

Normal operations at ORNL generate up to 7.6 million liters per year of intermediate-level waste (ILW) solution having an average activity concentration of a few hundredths of a curie per liter. This ILW solution is routinely neutralized with sodium hydroxide which precipitates those constituents of the waste solution that are insoluble in alkaline solution. These solids, which contain the bulk of the Sr-90 and the actinides that have been discharged to the waste system, settle and accumulate in the waste storage tanks. The remaining liquid (supernate) is concentrated by evaporation and since 1966 has been periodically disposed of by shale fracturing (a process for the fixation of the waste in an impermeable bedded shale formation at a depth of approximately 300 meters). Approximately 300,000 liters of concentrated waste solution are disposed of annually. The current generation rate for sludge is about 40,000 liters per year. About 1,500,000 liters of sludge have accumulated from the operations of previous years and are presently held in various waste storage tanks.

The concentrated waste solution is alkaline, the major chemical constituent is sodium nitrate and the predominant radionuclide is Cs-137 (about 0.3 Ci/l). This solution is one of three components of the ILW to be processed in the proposed processing facility. The anticipated concentration of radionuclides in each of the three ILW streams is given in Table F-6A. The second ILW component is the sludge which would be diluted to a 12-percent-by-weight slurry to make it pumpable. The predominant radionuclide would be Sr-90. The third component of the ILW to be processed is an assumed waste from future operations at ORNL which is designated as "pilot plant waste."

6.1 LIQUID WASTE STREAMS

Radioactive liquid wastes are produced by a number of different research, development, and production programs at ORNL. These include such things as basic radiochemistry studies, development of reactor fuel reprocessing methods, production of radioisotopes for medical, industrial and research uses, production of transuranium isotopes for research, operation of research reactors, etc. The

TABLE F-6A
DESIGN BASIS WASTE STREAMS FOR ORNL PROCESSING FACILITY

Waste	Nuclide	Specific Activity (Ci/liter)	Estimated Annual Volume (liter/year)	Estimated Annual Curies (Ci/year)	Estimated Total Volume (liters)	Estimated Total Curies (Ci)
Routine Wastes Concentrate	Cs-137	0.26	300,000	8×10^4		
	Ba-137M	0.24		7.4×10^4		
	Sr-90	0.024		8,000		
	Y-90	0.024		8,000		
	Ru-106	0.009		2,800		
	Rh-106	0.009		2,800		
	Pu-239	2.6×10^{-7}		0.08 to 0.8		
		2.6×10^{-6}		80		
		2.6×10^{-4}		7.5×10^4		
		1.3		2.3×10^4		
Pilot-Plant Wastes	Cm-244	1.3	(9,000)	2.5×10^4		
	Cs-137	1.2		2.5×10^4		
	Ba-137M	1.3		2.5×10^4		
	Sr-90	1.3		1.4×10^4		
	Y-90	1.3		1.4×10^4		
	Ce-144	0.7		1.4×10^4		
	Pr-144	0.7		1.4×10^4		
	Pm-147	1.8		3.5×10^4		
	Pu-239	2.6×10^{-5}		0.5		
		0.066		125		
Sludge	Cm-244	0.01	150,000*	1,400		
	Cs-137	0.01		1,400		
	Ba-137M	0.01		16,000		
	Sr-90	0.1		16,000		
	Y-90	0.1		6.4		
	Pu-239	4.2×10^{-5}		240		
		1.6×10^{-3}				
		0.014				
		0.013				
		0.14				
Stored Wastes Sludge	Cs-137	0.014			5.7×10^4	8×10^4
	Ba-137M	0.013				
	Sr-90	0.14				
	Y-90	0.14				
	Pu-239	5.3×10^{-5}				
	Cm-244	2.1×10^{-3}				1.2×10^4

* Volume after dilution for transfer.

liquid radioactive wastes from these and similar activities are collected in underground stainless steel tanks ranging in size from a few hundred liters to several thousand liters and located near the individual facilities producing the waste. The wastes are made alkaline in the collection tanks by the addition of sodium hydroxide to reduce the corrosiveness. The collection tanks are periodically emptied by pumping the contents to one of six 645,000-liter underground concrete central collection tanks. The addition of sodium hydroxide to the raw waste causes some of the dissolved materials to precipitate from the solution and this precipitate settles to the bottom of the gunite tanks as a sludge. The sludge contains essentially all (>99%) of the uranium and transuranium nuclides that might have been in the liquid as well as some of the radioactive fission products such as Sr-90. The sludge is one component of the ILW that is to be processed by the proposed solidification facility. The liquid above the sludge is fed to an evaporator where it is concentrated by a factor of 20 to 30 and the concentrate is temporarily stored in one or more of the remaining gunite tanks. About 300,000 liters per year of concentrated waste solution are generated at ORNL. Periodically, the accumulated concentrate is pumped to the shale fracturing facility in an adjacent valley, mixed with cement, and injected into a shale formation at an approximate depth of 300 meters.

When the construction currently in progress is complete, operation of the ILW collection and concentration system will be essentially the same as it is now except that the six gunite tanks will no longer be used. However, before these tanks can be put into a condition requiring minimal future surveillance, the sludge which has been accumulating since 1943 (~1,500,000 liters) must be removed and immobilized in the proposed new processing facility. This represents a "one time only" task for the new facility since it is planned, once the proposed facility is built, to process sludge routinely rather than let it accumulate as it has in the past.

The proposed processing facility will be designed to handle three types of intermediate level radioactive wastes. This ILW consists of:

1. Currently generated waste solution concentrate, which totals about 300,000 liters per year. This waste will be an alkaline salt solution (1 M NaNO_3 and other minor constituents). Generally, the major radionuclide (about 0.3 Ci/liter) is Cs-137, while only 0.3 Ci/liter or less is Sr-90. The major alpha activity is Cm-244 from the heavy element production program; its concentration is about 0.3 m Ci/liter. The long lived (greater than 30 years) alpha content of the waste concentrate, including the decay products from short lived parent nuclides, is less than 10 n Ci/gram.
2. The sludges produced by the processing and handling operations of the last three decades that are currently stored in the gunite waste tanks. There are about 1,500,000 liters of these sludges. Analyses indicate that these sludges contain between 500,000 and 800,000 Ci of Sr-90, between 60,000 and 90,000 Ci of Cs-137, 12,000 Ci of Cm-244 and 250 to 310 Ci of Pu-239. Current planning anticipates that within the next few years these sludges will be resuspended and pumped to a processing facility. This waste stream, after resuspension and dilution for pumping, will have a specific activity of about 13 Ci/liter of Sr-90. About 40,000 liter/year of currently generated sludge will also be processed. Its composition is expected to be similar to that of the sludge now in storage.
3. Future pilot plant wastes. Several pilot plant programs that have been proposed for ORNL have the potential for generating waste solutions with a higher specific activity than the ORNL intermediate level waste has had in the past. Some of these programs, such as the LMFBR spent fuel reprocessing plant, propose to include processes that will immobilize all waste streams that are generated. If the quantity and concentration of radionuclides are high, however, the wastes would be stored in existing doubly contained waste storage tanks until the isotopes with short half-lives had decayed (at least five years) and the specific activity of the residual isotopes would be low enough so that only moderately thick shielding would be needed in the processing facility. After this time these wastes or some part of them could be processed by the ORNL ILW system.

From 1966 to 1970, a series of seven injections that disposed of 3,750,000 liters of waste grout containing 350,600 Ci of Cs-137 and 26,500 Ci of Sr-90 was made. Some of the parameters of these injections are given in Table F-6B.

TABLE F-6B

INJECTION PARAMETERS FOR ROUTINE ORNL DISPOSAL OF WASTE SOLUTION²⁵

Injection No.	Date	Depth (m)	Waste Volume (liter)	Sr-90 (Ci)	Cs-137 (Ci)	Cm-244 (Ci)	Pu-239 (Ci)
<u>Experimental Injections</u>							
I-7	1964 to 1965	285 to 264		1,436	5,237	--	--
<u>Operational Injections</u>							
ILW1A	1966	264	136,300	3	19,950	NA*	NA
	ILW1B	1966	264	98,400			
	ILW2A	1967	261	325,500	1,050	58,500	NA
	ILW2B	1967	261	234,700			
	ILW3A	1967	261	117,300	9,000	17,000	NA
	ILW3B	1967	261	196,800			
	Water Test		1967	258	--		
	ILW4A	1968	258	90,900	4,300	51,900	NA
ILW4B	1968	258	235,350				
	ILW5	1968	254	306,600	500	69,400	NA
	ILW6	1969	254	300,400	8,900	89,000	NA
	ILW7	1970	254	314,200	2,747	44,833	19.2
	ILW8	1972	251	275,200	45	28,000	0.20
	ILW9	1972	251	258,500	231	23,400	6.51
	ILW10	1972	251	320,800	1,330	18,800	26.67
	ILW11	1972	251	286,750	1,100	23,500	155.74
	ILW12	1975	248	97,300	1,324	12,752	1.02
	ILW13	1975	248	306,600	3,368	35,750	17.83
	ILW14	1975	248	314,000	2,874	30,592	3.58

NA = not analyzed

6.2 TRU WASTE OPERATIONS

Solid radioactive waste generated at this site is collected in suitable containers and transported to one of the burial grounds, where they are stored or disposed of. There are six burial grounds at ORNL. Three of these are inactive, and two (burial grounds 5 and 6) are currently in use. Burial ground 3 is now only used for above-ground storage of contaminated equipment that may be reusable.

Estimated sources of transuranium-solid waste at ORNL are summarized in Table F-6C. Approximately $1.76 \times 10^5 \text{ m}^3$ of radioactive solid waste were buried in ORNL burial grounds through 1972. About 60 percent of this waste was generated at ORNL while the remainder was shipped from off-site (including Argonne National Laboratory). Most of the ORNL waste was transported to the burial ground using only temporary containment such as plastic and sprayed coatings. Wastes from off-site were generally shipped in steel drums and wooden crates. In general, wastes were not segregated with respect to waste type.

At present, ORNL burial grounds are used almost exclusively for ORNL wastes, which are generated at the rate of about $4,250 \text{ m}^3/\text{year}$. Wastes containing negligible quantities of transuranium nuclides continue to be buried in a relatively unretrievable fashion. Low radiation level, transuranium solid waste is packaged in either 208-liter steel (or stainless steel) drums or steel boxes and stored either in a building or a roofed, rock-filled trench to facilitate retrievability. High radiation level transplutonium wastes from the Transuranium Facility are collected and stored in the rock-filled trench as potentially retrievable, sealed concrete containers. Other high radiation level transuranium wastes are packaged in stainless steel containers and stored in marked, steel-lined auger holes.

ORNL classified locally generated radioactive waste intended for burial as follows:¹

- The fissile-alpha waste corresponding to transuranium waste is defined by DOE 0511-21 as the solid waste that exceeds the concentration of 10 microcuries of alpha

TABLE F-6C
ESTIMATED SOURCES OF TRANSURANIUM SOLID WASTE AT ORNL^a

Type	Percent Burnable	Quantity in Storage		Annual Generation Rate	
		Volume (m ³)	Alpha Radioactivity (Ci)	Volume (m ³)	Alpha Radioactivity (Ci)
Retrievable Steel Drums ^(a)					
Low radiation level	50			169 ^(d)	30 (Pu-239)
Retrievable Steel Boxes ^(a)					
Low radiation level	20	0	0	42 ^(d)	6 (Pu-239)
Steel Drums and Boxes ^(a)					
High radiation level	50	0	0	28 ^(d)	5 (Pu-239)
Concrete Casks ^(a)	~50	280	200 (Cm-244)	100 ^(d)	80 (Cm-244)
Solid Waste for Burial					
Low radiation level	~0	113,300 ^(b) (283,300) ^(c)	300 (Pu)	3,335	<1
Solid Waste for Burial					
High radiation level	~0	18,300 ^(b)	100 (Pu)	835 ^(c) (84,500) ^(c)	<0.03
Tank Farm Sludges	0	1,275 ^(c) (7,300) ^(c)	300 (Pu-239)		

- a. Waste from contamination zones or known to contain $> 3531 \mu\text{Ci}/\text{m}^3$ of transuranium nuclides.
b. Quantity as charged.
c. Estimated volume if the waste were prepared for off-site shipment.
d. Since 1970, these packages are stored to facilitate retrievability.

particle activity per kilogram of waste associated with fissionable isotopes. This category includes, primarily, americium, curium, and plutonium, as well as some U-235. This material must be handled so as to permit retrieval within 20 years. These materials originated primarily in the transuranium-processing facilities and in fuel reprocessing research and development operations. Beta-gamma emitting nuclides are often included with these nuclides and in very heavy elements spontaneous fission often occurs with production of neutrons. Certain beryllium isotopes or deuterium will emit photoneutrons. These special situations all create unique problems and demand special methods for handling of the fissile alpha wastes.

- The fissile non-alpha waste is a solid waste that contains one gram or more of non-alpha particle emitting fissionable materials regardless of concentration, or more than 35.7 grams per cubic meter (one gram per cubic foot) of the same material regardless of quantity. Sources of these wastes include various metallurgical operations and other analytical and research and development operations. The objective of handling this waste is to prevent the criticality during or after burial. It is not a direct radiation hazard.

The volume of waste buried at ORNL since the fire of 1961 is about $87,500 \text{ m}^3$ in the area of 7.5 hectares. Transuranium- and uranium-contaminated wastes buried at ORNL before 1974 amount to about 685 m^3 and fissionable material and contaminated waste about 200 m^3 . Not all of the transuranium waste can be considered reasonably retrievable.¹

Fissile alpha-particle emitting wastes (TRU wastes) have an alpha particle radioactivity in excess of 10 microcuries per kilogram (10 nanocuries per gram) originating from fissionable isotopes. This kind of waste, when accompanied by a high level of beta, gamma, or neutron emission, is packaged at the point of origin in reinforced concrete casks. Records are kept of the location and content of each of the trenches containing this material.

6.2.1 STORAGE FACILITIES⁹⁰

TRU wastes are stored in a variety of ORNL facilities ranging from trenches in which material is placed and covered with soil, to stainless-steel-lined shafts, to 85 percent below-ground buildings for 208- or 114-liter metal drums. The selection of the facility depends on the intensity of the radiation involved.

TRU waste containing high levels of beta-gamma or neutron emissions is placed in reinforced concrete casks and stored in unlined trenches. Casks are segregated by trench according to their flammable and non-flammable waste content. Once filled, the trenches are backfilled with soil. TRU waste containing less than 200 mR/hr is stored in stainless steel drums and held in a staging area until a sufficient number of them is available for storage at the Retrievable Waste Storage Facility, Building 7826.

TRU waste with high background readings is stored in stainless-steel-lined shafts of various diameters (ranging from 20.3 to 50.8 cm). The liners are provided with concrete collars and stainless steel caps. Spacing between shafts is not less than 0.9 m, and the transuranic isotope content is limited to 200 g per shaft. The shafts are filled with stainless steel containers of waste and then are capped with a removable concrete plug. The contents of each shaft are recorded.

6.3 DISPOSAL FACILITIES

Non-TRU waste and general radioactive waste are generally disposed of in unlined trenches and/or shafts. Shafts vary in diameter (0.6 to 1.0 m) and depth (less than 6 m). A minimum spacing of 0.9 m in between shafts is maintained, and the radioactive content is limited to 200 g per shaft. Once filled, the shafts are capped with concrete.

Three settling basins have been used in past operations at ORNL. These basins are waste pond no. 1, waste pond no. 2, and the homogeneous reactor test settling basin. Only waste pond no. 1 is still used and serves as an equalization basin for laboratory process waste prior to treatment in the ion-exchange facility.

From 1957 to 1976, waste pond no. 2 served as a settling basin for the effluent from the lime-soda-clay treatment plant. Sediment samples taken after its decommissioning indicated that bottom sediments at this pond contain approximately 5 Ci of Pu-239 and Pu-240 and various fission products. The concentration of radionuclides in the bottom sediments of waste pond no. 1 is expected to be higher than that in waste pond no. 2. However, the bottom sediments in this pond have not been sampled, and consequently no estimate of their radioactive content is available.

The settling basin was used for disposal of low-activity liquid waste generated in the 7500 area. The basin is no longer in service and has been filled and covered with asphalt. No records of its contents are available.

Pits and trenches were used for the disposal of intermediate-level liquid waste. A total of four pits and three trenches was used in this operation until 1965. Radionuclide seepage from one trench and one pit was detected in the earlier days of this operation, and both trench and pit were closed. Approximately 181,680 m³ of liquid waste containing 1 million curies was disposed of in these pits and trenches.

Burial grounds 1 and 2 cover an area of about 2 ha and were closed in 1946. These areas were used primarily for the disposal of contaminated trash, laboratory equipment, and other items that were discarded. Burial ground 3 was opened in 1946 and used until 1951 for underground disposal of waste. Burial ground 4 covers a total area of 9.3 ha and was used until 1959 for the disposal of alpha-and beta-gamma-contaminated waste.

Burial ground 5 has been used for the disposal of all types of wastes including TRU wastes; however, most of the material that has been disposed of in this burial ground is considered as general radioactive waste such as mixed fission products, depleted uranium, filter media, and equipment that could not be economically decontaminated. The 27.5-ha burial ground 6 was opened in 1969 and is used for disposal of waste in trenches and shafts.

6.4 OAK RIDGE Y-12 PLANT⁹⁰

The Oak Ridge Y-12 Plant (427-ha plant area) is located in Bear Creek Valley about 4.8 km from Oak Ridge, Tennessee. The low-activity radioactive waste disposal sites are also located in Bear Creek Valley about 2.9 km west of the main plant site. The classified disposal area is located within the main plant site.

Low-activity radioactive waste is disposed of in trenches (5.5 m deep x 7.9 m wide) at burial grounds 1-A and 2-C. Filled trenches are covered with soil, leveled, and grass-seeded. Burial ground 1-A is used primarily for the disposal of depleted uranium-contaminated materials such as particulate filters, metal drums, and mixed-metal machine turnings. About 9.0×10^6 kg of these materials is disposed of in this burial ground annually. Waste disposal, compaction, and sealing processes presently require the use of land at a rate of 0.04 ha per year.

Materials contaminated with enriched uranium and natural thorium are disposed of in burial ground 2-C at a rate of 4.5×10^5 kg per year. The current land use for this waste disposal operation is 0.04 ha per year. Burial ground 2-B is used for the storage of pure, depleted uranium. About 7.3×10^5 kg of uranium is buried in trenches (4.5 m deep by 4.0 m wide) at this site.

Classified materials such as forms made from uranium and thorium are disposed of in trenches (5.8 m deep x 2.4 m wide) at the classified disposal area. For security reasons, the materials are covered with soil the same day they are placed in the trench. A total of 2.3×10^5 kg of these contaminated materials is buried annually at this site.

Records and descriptions are maintained on each disposal operation; they include the volume, radionuclide content, type of containers, location, and personnel safety measures taken. No further information is presently available concerning these waste materials.

6.5 OAK RIDGE GASEOUS DIFFUSION PLANT (K-25)

The Oak Ridge Gaseous Diffusion Plant is situated on a 259-ha area in eastern Tennessee. Its primary purpose is the enrichment of uranium hexafluoride in the uranium-235 isotope. No high-level radioactive waste is stored or disposed of at K-25. The majority of the low-activity waste buried or stored at this site is material contaminated with uranium and with traces of Np-237, Pu-239, and Tc-99.

6.5.1 BURIAL FACILITIES⁹⁰

There are five low-activity radioactive waste sites at K-25: the old classified burial ground, new classified burial ground, K-33 contaminated waste burial ground, K-722 scrap metal yard, and K-1407-C retention basin. Three of the five sites are land burial grounds, but only one is still active. The fourth site is an active above-ground storage area where noncontaminated and contaminated scrap metal is stored. The fifth site is an active retention basin for radioactive and nonradioactive sludges.

The old classified burial ground covers 1.5 ha and has an average depth of about 9.1 m. Due to the lack of accurate records, no good estimate of the radioactivity of the materials disposed of at this site can be made. However, it can be deduced that most of the material buried is uranium-contaminated with only trace quantities of transuranics and/or fission products.⁹⁰

The new classified burial ground was built in 1975 to dispose of classified radioactive and nonradioactive material and covers an area of 8.9 ha. Waste is buried in trenches 4.5 m deep and 30.5 m long. Each trench is completely filled, covered with soil, and grass-seeded before excavating another trench. At the present time, only 0.1 Ci of uranium is buried at this site, along with trace quantities of TRU and/or fission products.

The K-722 scrap metal yard covers 8.9 ha of land for the storage of scrap metal. The scrap metals consist of steel, stainless steel, aluminum, copper, nickel, and alloys. Two of the 8.9 ha are set aside for the storage of metal that has been exposed to uranium materials or shows alpha or beta-gamma activity on the surface. It is estimated that 2.7×10^6 to 4.5×10^6 kg of contaminated scrap metal are stored in this site. No estimated quantity of radionuclides associated with these materials is available.

6.6 TRU WASTE GENERATION

Table F-6D presents a summary of the TRU waste stored and buried at ORNL.

TABLE F-6D
SUMMARY OF TRU CONTAMINATED WASTE
BURIED/STORED AT ORNL^{79,80,81}

	<u>Previous YEARS</u>	<u>1977</u>	<u>1978</u>	<u>Total</u> [*]
BURIED TRU WASTE				
Volume Waste (m ³)	5,700 $\pm$ 50%	0	0	5,700 $\pm$ 5%
Equipment (m ³)	400 $\pm$ 10%	0	?	400 $\pm$ 10%
Kg TRU	13	0	?	13
STORED TRU WASTE				
Volume (m ³)	1,000 $\pm$ 5% ^{**}	17.8	28.2	1,046 $\pm$ 5%
Kg Pu	1	}1.6	}0.5	}18.1
Kg Other TRU	15			

* Between 1978 and 1984 approximately 200 cubic meters of large equipment will be buried at ORNL.

** 1976 storage of TRU wastes was 51 cubic meters.

7.0 ROCKY FLATS PLANT (RFP)

The Rocky Flats Plant, which is operated for the Department of Energy by Atomics International, is a key producer of plutonium parts for nuclear weapons. It is located approximately 10 miles south of Boulder, Colorado, and 16 miles northwest of Denver. Construction of the present plant complex, consisting of some 75 buildings and structures, was started in 1951, and the plant was put into operation during 1953.⁶⁴

Plutonium-contaminated waste is generated primarily in the plutonium fabrication area and the chemical processing facility. The Research and Development Group, the Analytical laboratories, and other support groups generate lesser quantities of contaminated waste. Only 5% of the contaminated waste is generated in uranium, beryllium, and other nonplutonium operations.

RFP handles many kilograms of plutonium each year. The work, which is performed in glove boxes, currently generates, annually, as much as 40 percent of all DOE waste contaminated with plutonium. Much of the waste generated is expected to be similar in nature to Los Alamos (particularly the plutonium fabrication waste.)

7.1 LIQUID WASTES^{5,148}

Liquids contaminated with plutonium are transferred to a process waste treatment plant. There the plutonium and other chemical contaminants are removed by carrier precipitation as solids for transfer to DOE-approved storage at INEL.

Liquid wastes include the following:

- Plutonium and americium ion column effluents
- Distillate
- Caustic scrubber solution

- HCl effluents
- Miscellaneous solutions
- Laundry water
- Uranium and beryllium process wastewater
- Organic solvents

Most liquids are treated at least three times by neutralization by NaOH and the addition of ferric hydroxide (also $\text{Fe}_2(\text{SO}_4)_3$, CaCl_2 and MgSO_4) to form a carrier precipitate. This precipitate is collected on drum filters, removed as a sludge, mixed with dry cement to absorb free liquids, and placed in 208-liter drums. Liquids containing hydrochloric acid or complexing agents are processed separately by mixing the liquids with portland cement to form a solid in 208-liter drums. Liquids containing acids are neutralized prior to mixing. Organic solutions and oils are also blended with an absorbent (calcium silicate powder, microcel, etc.) to form a solid in the 208-liter drums.

Effluent from treated liquid, which is low in radioactivity but high in soluble chemical contaminants, is evaporated. Salts from the evaporation are packaged in 208-liter drums. The previously released processed aqueous waste is now being retained in a holding pond until construction of a new waste treatment facility is completed.

7.2 RADIOACTIVE SOLID WASTE⁵

In its handling of plutonium and other radionuclides, the Rocky Flats Plant generates radioactive scrap, or residue, and radioactive waste. Scrap is material that can be recycled for productive use; radioactive scrap contains plutonium or enriched uranium that is economically desirable to recover. Waste, on the other hand, has no recoverable value. TRU waste is not economically worth treating to recover the plutonium or uranium in it.

Typical solid wastes are as follows:

- Plastic and paper
- Rubber gloves
- Gloves and Raschig rings
- Non-stainless-steel metal
- HEPA filters
- Cemented resins
- Graphite molds
- Wood
- Sludge
- Equipment

Most of these categories are line-generated wastes, i.e., produced in glove boxes in the chemical processing area or the metal fabrication area.

A total of about 5,230 cubic meters of plutonium contaminated waste has been currently generated annually. Nearly 3,270 cubic meters of this total was packaged and shipped to an off-site retrievable storage location. This is considered TRU waste, which is waste with more than 10 nCi/g. Approximately 1,960 cubic meters has been packaged and shipped to an off-site non-retrievable location; this is non-TRU waste that has less than 10 nanocuries of plutonium per gram.

Recently, the Rocky Flats operation has been generating about 20 kg of waste plutonium annually. The accumulated (through 1978) plutonium in the INEL burial site amounts to about 600 kg of which about 200 kg is retrievable storage.

7.2.1 WASTE COLLECTION

Removing solid waste from glove boxes involves transferring the waste through a glove-box opening into a plastic bag or sleeve clamped to the opening. The bag is then twisted, taped closed, and cut away. Because of its origin, all waste of this type is considered by the Department of Transportation (DOT) to be of NOS (Not Otherwise Specified) activity (formerly HSA or High Specific Activity). LSA (Low Specific Activity) waste has less than 0.1 microcurie of plutonium per gram of waste; NOS waste contains more than 0.1 μ Ci/g.

Other contaminated solid wastes result from materials such as clothing, paper used by employees in process o. controlled areas (but outside glove boxes), and surgeons' gloves; they are designated low specific activity (LSA) if no significant X- or gamma radiation is noted by radiation monitoring surveys and if no measurable plutonium is indicated by drum-counting techniques. These techniques involve detectors that identify and quantify the radioactive elements in each drum containing contaminated waste.

7.2.2 WASTE PACKAGING

Most radioactive wastes are sorted and packaged in 208-liter steel drums manufactured under a quality control program to ensure compliance with DOT specifications. A thick-walled polyethylene drum liner is used to contain TRU wastes inside the drum for retrievable storage. The polyethylene liner provides added resistance to corrosion from the contents, increases resistance to tear or puncture, and is a means of containment even if the steel drum corrodes. The plutonium content of each drum is determined by counting neutrons and gamma radiation.

Some solid wastes also require post-generation treatment. The following techniques are utilized to either reduce the volume of, decontaminate, or lower contamination levels of the waste.

- Large bulky equipment, such as glove boxes, lathes, and duct work, is cut up to reduce size, followed by washing to remove recoverable amounts of plutonium
- Paper, plastic, drums, and HEPA filters are compacted
- Rubber gloves and metal waste are washed in a tumbling mill to reduce contamination and recover plutonium
- Moist waste is dried
- Paper and plastic are baled
- Certain combustibles are incinerated
- Wet wastes are solidified with cement

When these wastes are immobilized, they are packed as efficiently as possible in the appropriate waste container.

Some wastes such as glove boxes, large equipment, and construction materials are too large to fit into a 208-liter drum. These items are disassembled to reduce the volume, cleaned to remove plutonium, and packaged into 1.9 cm thick plywood boxes. For retrievable storage, the boxes are coated with fire-retardant and fiberglass-reinforced polyester. These boxes are manufactured under a quality control program to ensure compliance with DOT specifications. Each item placed into a plywood box is monitored to detect any significant residual plutonium.

7.3 WASTE TYPES, CHARACTERISTICS, AND AMOUNTS

Classified contaminated solid wastes generated by RFP are shipped to Richland, Washington for burial while unclassified (Department of Defense) wastes are shipped to INEL. The activity shipped to INEL in these wastes varies depending upon programs being carried out (Table F-7A). The plutonium (and americium) contaminated solid waste generated at RFP can be put into eight categories: first stage sludge (non-LSA), second stage sludge (non-LSA), evaporator salts (LSA), cemented liquids (non-LSA), grease (non-LSA), line-generated waste (non-LSA), nonline-generated waste (LSA), and crated waste (non-LSA). Sorting or segregation of contaminated waste at RFP is done mainly to facilitate measurement of radioactive content and to aid in the processing of the material.

TABLE F-7A
ESTIMATE OF TRU ACTIVITY SHIPPED BY ROCKY FLATS PLANT
TO INEL FOR STORAGE

Year	Pu-239 (g)	Am-241 (g)
67	50,031	2,129
68	46,486	1,778
69	38,046	4,876

Year	(Ci) ^(b)	(Ci) ^(a)
67	4,622	6,813
68	3,579	5,690
69	2,930	15,603

a. Based on: Am-241 at 3.2 Ci/g

b. Plutonium Isotopic Composition and Alpha Radioactivity

<u>Isotope</u>	<u>Wt. %</u>	<u>Ci/g Pu</u>
Pu-238	0.009	0.0015
Pu-239	93.750	0.0575
Pu-240	5.857	0.0133
Pu-241	0.368	--
Pu-242	<u>0.016</u>	<u>0.0000006</u>
	100.000	0.0723

7.3.1 SOLIDS FROM THE WASTE TREATMENT PLANT

First stage sludge, second stage sludge, evaporator salts, cemented liquids, and grease are generated in the waste treatment plant for radioactive liquids. Aqueous liquids of low radioactivity but high chemical content are sent either to the waste evaporator in the treatment plant or to solar evaporation ponds and then to the evaporator. The evaporator bottoms are sent to a double-drum dryer where the remaining water is removed. Water vapor from the evaporator and drum dryer go up the evaporator stack. The dried salts are drummed for shipment. Aqueous liquids of high radioactivity and high chemical content are processed through two stages of decontamination in the treatment plant using a ferric hydroxide carrier precipitation process. The decontaminated aqueous waste is combined with the low radioactivity--high chemical content waste and is processed in the waste evaporator mentioned above. The precipitate is removed from the supernatant liquid by vacuum filtration. The filter cakes are drummed for shipment as first and second stage sludges. Aqueous liquids of high radioactivity but low chemical content are decontaminated in the second stage of the waste treatment plant. The precipitate is filtered and drummed for shipment as second stage sludge. Aqueous wastes containing complexing agents are not amenable to the above precipitation process. They are discharged into lined drums filled with magnesia and cement to set up before shipment. These solids form the category called cemented liquids. (Ion exchange resins are also cemented.) Organic liquids, primarily cutting oil plus carbon tetrachloride, are shipped to the waste treatment plant where they are mixed with calcium silicate to form a putty-like mass called grease. About 140 liters of organic waste is converted to grease per 208-liter drum used for shipment.

7.3.2 LINE-GENERATED WASTES

Line-generated waste contains the bulk of the plutonium discarded and shipped for burial. This waste category is segregated at the point of generation according to type of waste material (Table F-7B). The residues are segregated by type to aid in assaying for nuclear materials accounting and to facilitate

TABLE F-7B
CATEGORIES AND DISCARD LIMITS AT THE
ROCKY FLATS PLANT LINE-GENERATED WASTE⁵

Category	Discard Limit (g Pu/g)
Sweepings	0.007
Sludge	0.007
MgO Sand	0.007
Ion Exchange Resin	0.007
Incinerator Ash	0.007
Sweepings Heels	0.007
Ash Heels	0.007
Glass and Ceramics	0.0005
Scarfed Molds	0.00035
Graphite Flow Residue	0.002
CWS Filter (0.6 x 0.6 x 0.3 m)	24 g/filter
Dry Box Filters (3.1 x 3.1 x 1.6 cm)	3 g/filter
Washables	0.0006
Combustibles	0.0007
Miscellaneous Scrap Metal	0.0003

processing, including incineration when necessary, to recover plutonium. The type of matrix has to be known to obtain a valid determination of plutonium content by the drum counter. Residues which contain more than the discard concentration of plutonium are processed as scrap to recover the plutonium. Residues which contain less than the discard concentration are discarded as line-generated wastes.

Tables F-7C through F-7E present data on various materials associated with the in-line glove box operations.

7.3.3 NONLINE-GENERATED WASTES

Wastes removed from a radioactive material processing area (other than a processing line) are considered contaminated and are collected in drums as nonlinear-generated solid waste. Drums indicating greater than 0.5 mr/hr at the surface are sent to the drum counter for assaying and possible plutonium recovery. Nonlinear-generated waste which shows less than 0.5 mr/hr at the surface of the drum is considered LSA waste and shipped as such.

7.3.4 CRATED WASTES

Items that are too large to be put into a drum are put into plastic lined plywood boxes or crates. These constitute the category called crated wastes.

Nonlinear-generated plutonium wastes are segregated from nonlinear-generated uranium wastes since they are generated in separate buildings. However, until 1970 they were not kept separate in shipment or burial. Drums of all categories of non-LSA plutonium wastes are mixed in shipment and burial. A comprehensive breakdown of RFP waste discards for first stage sludge, second stage sludge, evaporator salts, and cemented liquids is given in Table F-7F; for grease in Table F-7G; for line-generated waste in Table F-7H; and for crated Pu-contaminated waste in Table F-7I (based on 1967 evaluation).

TABLE F-7C
ORGANIC GLOVEBOX MATERIALS⁷¹¹

<u>Box Segment</u>	<u>Material</u>
Glovebox Gloves	Neoprene/Hypalon
Window Materials	Polycarbonates Polyesters Polymethyl methacrylates
Gasket Materials	Fluorel Viton Silicone Rubber
Shielding Materials	Benelex Polyethylene
Glovebox Filters	HEPA
Miscellaneous	Teflon Polyvinyl Chloride

Fluorel: Minnesota Mining and Mfg. Co.
 Viton: E. I. du Pont de Nemours and Co.
 Benelex: Masonite Corp.
 Teflon: E. I. du Pont de Nemours and Co.

TABLE F-7D

THERMAL PROPERTIES OF VARIOUS POLYMERS COMMON TO GLOVEBOXES

<u>Material</u>	<u>Polyethylene</u>	<u>Teflon</u>	<u>Polyvinyl Chloride</u>	<u>Polymethyl Methacrylate</u>	<u>Poly Carbonate</u>
Decomposition Temperature °C	350	500	200	185	380
Specific Heat cal/g/°C	0.55	0.25	0.25	0.35	0.30
Thermal Conductivity 10^{-4} cal/sec-cm ²	8.0	6.0	~5.0	5.0	4.6
Evolved Vapor Flash Temperature °C	340	-	340	280	420
Autoignition Temperature °C	350	530	454	450	N.A.
Limiting Oxygen Index	0.17	0.95	0.49	0.20	0.26

TABLE F-7E
GLOVEBOX GLOVE MATERIAL

- 30-mil butyl
- 15-mil neoprene
- 30-mil neoprene
- 30-mil leaded neoprene
- 30-mil *Hypalon-neoprene*
- 30-mil leaded Hypalon-neoprene
- 45-mil leaded Hypalon-neoprene
- 80-mil leaded Hypalon-neoprene
multi-layered

TABLE F-7F

FIRST STAGE SLUDGE, SECOND STAGE SLUDGE, CEMENTED LIQUIDS,
AND EVAPORATOR SALTS SHIPPED BY THE ROCKY FLATS PLANT
IN CY 1967^(a), 18

(Quarter	Date CY 67)	First Stage Sludge	Second Stage	
		Volume	Sludge Volume (m ³)	Miscellaneous ^(b) Volume (m ³)
1st		70.6	46	22.9
2nd		48.9	72.4	6.9
3rd		46.8	71	7.5
4th		<u>49.8</u>	<u>62.2</u>	<u>3.7</u>
Total		216.1	251.6	41.0

(Quarter	Date CY67)	Cementing Facility		Evaporator Salts
		Cemented Liquid Volume	Miscellaneous ^(b) Volume (m ³)	Volume (m ³)
1st		33.5	10	1.2
2nd		22.3	6.9	3.3
3rd		5.6	6.2	72.8
4th		<u>16.7</u>	<u>7.7</u>	<u>175.3</u>
Total		78.1	30.8	262.6

- a. Total plutonium charged off in these four waste categories was 8.125 kg.
- b. Contaminated empty bottles used as containers for transporting solutions to the waste treatment building.

TABLE F-7G
GREASE SHIPPED BY THE ROCKY FLATS PLANT IN 1967¹⁸

Date (Quarter 1967)	CCL ₄ -Oil Generated		Shipped ^(a) Volume (m ³)	Empty Contaminated Drums	
	Volume (liter)	Pu (g)		Drums (No.)	Volume (m ³)
1st	19,190	263	20.4	627	130.5
2nd	21,725	199	338.3	440	91.6
3rd	10,600	75	320.1	547	113.8
4th	<u>12,225</u>	<u>292</u>	<u>286.4</u>	<u>334</u>	<u>69.5</u>
Total	63,740	829	565.2	1,948	405.4

- a. The amount of grease shipped is much larger than the CCL₄-oil waste generated because of back-log material.

TABLE F-7H
LINE-GENERATED WASTE SHIPPED BY THE
ROCKY FLATS PLANT IN 1967⁵

Material	Volume (m ³)	Pu (g)
Graphite	88.5	21,211
Sand, Slag, and Crucible	47.7	6,948
Insulation	65.8	13,012
Fire Brick	0.6	355
Resin	5.0	1,207
Scrap Metal	56.8	1,773
Glass	21.0	1,235
Washables	66.2	4,699
Combustibles	23.5	648
Line-Generated	9.0	344
Filters	6.0	592
Tantalum	1.9	169
Miscellaneous	17.1	2,164
Am-241 Waste	8.6	408
Am-244 Waste	<u>1.8</u>	<u>113</u>
Total	376.2	54,878

TABLE F-7I
CRATED PU-CONTAMINATED SOLID WASTE SHIPPED BY THE
ROCKY FLATS PLANT IN 1967⁵

Date (Quarter 1967)	Incinerator Filters		Miscellaneous Pu-Contaminated Wastes		Total Crated Pu-Contaminated Wastes	
	Volume (m ³)	Pu (g)	Volume (m ³)	Pu (g)	Volume (m ³)	Pu (g)
1st	23.4	510	93.5	313	116.9	823
2nd	66.3	482	77.4	264	143.7	746
3rd	62.4	671	81.3	230	143.7	901
4th	<u>38.9</u>	<u>988</u>	<u>175.4</u>	<u>298</u>	<u>214.3</u>	<u>1,286</u>
Total	191.0	2,651	427.6	1,105	618.6	3,756

7.4 RECENT WASTE GENERATION

Table F-7J presents the average plutonium loading for the period 1971-1977 while Table F-7K presents recent waste generation data.

Continuing programs are in effect to reduce contaminated waste by both decreasing the amount generated and reducing post-generation volume. Total contaminated waste shipments from Rocky Flats have significantly decreased from 8,836 m³ in 1971 to 3,943 m³ in 1975, a 55 percent reduction. We anticipate another 50 percent reduction within three years by incorporating in the new chemical processing and liquid waste treatment plants several innovations to reduce the generation of contaminated waste. Transuranium (TRU) waste, that which exceeds 10 nCi/g (370 Bq/g), has shown a similar pattern since we began segregating retrievable from non-retrievable waste. There were 3,794 m³ of TRU waste generated in 1974 and 2,379 m³ in 1975, or a net reduction of 37 percent.

TABLE F-7J
AVERAGE PLUTONIUM LOADING ROCKY FLATS WASTE⁸²

<u>Year</u>	<u>No. of Units</u>	<u>Weight (gms)</u>	<u>Gms Pu/ Container</u>
D r u m s			
1971 *	2,726	2,555	0.94
1972	15,690	27,744	1.76
1973	8,978	12,705	1.42
1974	6,119	28,595	4.67
1975	3,556	30,894	8.69
1976	2,765	15,519	5.61
1977	<u>2,660</u>	<u>27,198</u>	<u>10.2</u>
TOTAL	48,494	145,210	3.42 (Avg.)
B o x e s			
1971 *	552	769	1.39
1972	975	5,383	5.52
1973	944	11,554	12.24
1974	776	6,612	8.39
1975	302	1,047	3.47
1976	492	1,858	3.78
1977	<u>466</u>	<u>4,993</u>	<u>10.71</u>
TOTAL	4,507	32,116	7.13 (Avg.)

* Partial year's data.

TABLE F-7K

SUMMARY OF TRU CONTAMINATED WASTE
GENERATED AND SHIPPED FROM ROCKY FLATS⁵

	<u>1977</u>	<u>1978</u>
Dry Waste Solids (m ³)	3,938	1,001
(kg)	96,180	38,680
Solid Sludge (m ³)	347	270
(kg)	34,110	28,400
Generated TRU Waste		
Volume	2,407	1,280
Kg TRU	33.5	20.7

8.0 PADUCAH GASEOUS DIFFUSION PLANT

The current burial of low-level radioactive waste at Paducah is within the 303 ha main plant site, which is located near Paducah, Kentucky.

8.1 BURIAL FACILITIES^{90,5}

The contaminated aluminum burial site, area J, has an area of about 0.04 ha. A pit was excavated to a depth of about 1.8 m, and the contaminated scrap metal buried in this pit consisted of about 100 to 150 drums of aluminum scrap metal in the form of nuts, bolts, etc. The contamination was primarily natural, depleted, and slightly enriched uranium corrosion products with trace quantities of neptunium and plutonium. The concentration of TRU nuclides is estimated to be orders of magnitude below the level that requires retrievable storage.⁹⁰ The pit was covered with 0.9 m of soil.

8.2 OPERATING PRACTICES

Scrap uranium is generated by precipitation with lime from equipment decontamination and cylinder wash solutions. The filter sludge is collected in plastic-lined drums. When a number of drums are accumulated, personnel load them on a truck and unload them at area C-404. These drums are covered with clay soil up to within a few rows of the end of the current stack. Depleted uranium tetrafluoride is also drummed and transported to area C-404 for burial.

Lime sludges are analyzed for Np-237, Pu-239, Th-230, and Tc-99. Batches containing Np-237 and Pu-239 in excess of 10 nCi/g are stored retrievably in plastic-lined steel drums in warehouse C-7465.

9.0 PORTSMOUTH GASEOUS DIFFUSION PLANT

The Portsmouth Gaseous Diffusion Plant is located on a 1618.8-ha reservation in Pike County, Ohio.

9.1 BURIAL FACILITIES^{90,5}

There are three storage facilities at the Portsmouth site for low-activity radioactive wastes: the X-206-D storage area, the X-744G facility, and the classified waste area.

The X-206-D storage area measures 76.2 m x 41.5 m, and all storage is on the ground. This area is used to store small scrap metal items which have small amounts of radioactive contamination and a high scrap resale value. The metals include copper, monel, stainless steel, nickel, brass, etc. All metal items are decontaminated prior to storage in this area.

The X-744G facility is a warehouse covering 0.8 ha. It is used primarily for storage of UF_6 contained in cylinders and for storage of nonfluorinated materials such as U_3O_8 , UO_2 , UNH_3 , uranium solutions, and radioactive plant waste. The low-activity radioactive wastes stored in this facility include the solid residues from the uranium recovery operation, contaminated alumina, magnesium fluoride, sodium fluoride, and various classified contaminated wastes and materials. Contaminated wastes are stored temporarily until a sufficient amount has been accumulated for burial. The classified waste area is capable of retrievably storing several tons of classified wastes such as barrier tube sheets, steel parts, and certain floor sweepings. All these materials are decontaminated of unclassified contaminated waste before storage. The stored materials are, for the most part, large pieces of metal or equipment having fixed surface contamination which are stored in 208-liter drums. All materials are decontaminated prior to storage.

Buried items in the X-749 trenches fall under two categories. Category 1 (chemical trap waste) includes uranium-bearing solids such as alumina, sodium fluoride, magnesium fluoride, incinerator ash, and filter cake, whose uranium content is not economically recoverable. Category 2 consists of scrap metal that is potentially or slightly contaminated and cannot be sold or placed in a sanitary landfill. Scrap metal is decontaminated before burial.

10.0 PANTEX PLANT⁵

The 1.4-ha burial site is centrally located on the Llano Estacado within the 3683-ha Pantex Plant, which is located 29 km northeast of Amarillo, Texas. All radioactive waste buried at this site is considered retrievable over a period of 20 years.

Radioactive waste is packaged for burial in the generating area in plastic bags and overpacked in 18.9-liter cans or in fibreglassed wood boxes.

Waste is stored in vertical concrete cylinders (6.1 m deep x 1.8 m in diameter) or in trenches (30.5 m long x 4.3 m wide x 4.0 m deep). Once filled with waste, the cylinders are capped with a concrete plug and covered with 1.8 m of compacted soil. Waste stored in trenches is also covered with 1.8 m of compacted soil.

No TRU wastes are routinely generated at Pantex, and the small amount of TRU waste from nonroutine sources is segregated from the other wastes and buried in a separate trench in fibreglassed boxes. The current annual rate of waste burial is $1.4 \text{ m}^3/\text{year}$, which contains 134 kg of uranium, which is the primary component of the wastes.

11.0 SANDIA LABORATORIES (SL)⁵

Sandia Laboratories is located at the foothills of the Manzano Mountains near Albuquerque, New Mexico. The current 0.6-ha radioactive disposal site is located in an environmental test area, Tech Area III, of Sandia Laboratories. An old burial ground (closed in 1960), consisting of an area of 0.11 ha, is located in an explosive test area, Tech Area II.

11.1 BURIAL FACILITIES

The active disposal site is divided into three areas: one for the disposal of classified nuclear weapon components, another for the disposal of bulky debris, and the last for future expansion. Open trenches and burial shafts are the two methods of disposal employed at SL.

The form of the waste determines the packaging used for burial. All liquids are solidified and packaged. All decontamination waste is sealed inside plastic bags. For large equipment, removable contamination is either fixed or contained. Radioactive components are bagged. An instrument survey is performed at this time to determine if radiation levels or removable contamination precludes normal handling.

Open trenches (55 m in length x 11 m in width x 4 m in height) are used for very low activity or suspect radioactive waste. This includes decontamination debris, HEPA filters, surplus contaminated equipment, experimental structures used in contaminated areas, solidified liquids, etc.

Burial shafts are used to dispose of all radioactive devices with security classification, radioactive sources greater than 10 Ci, and uranium machining waste, as well as any other uranium waste that is a potential fire hazard.

Before retrievable storage of TRU waste became a DOE requirement, plutonium-contaminated debris from nuclear weapons tests was placed in one trench in the disposal area. This waste is not considered retrievable. Subsequent TRU waste generated at SL and requiring retrievable storage has been transported to an off-site DOE storage facility. Currently (1978), no TRU wastes are being generated at SL.⁸¹

The current cumulative inventory indicates that a total volume of 1322 m³ of radioactive waste has been disposed of at this site since operations started. This total includes primarily fission product/induced-activation products, 152.8 m³ of uranium-contaminated wastes, 15.7 m³ of TRU-contaminated wastes containing less than 10 nCi/g, and 1.7 m³ of TRU-contaminated wastes containing more than 10 nCi/g (which were buried prior to the retrievable storage requirement). The cumulative decayed activity is 2.4 kCi.

12.0 NEVADA TEST SITE (NTS)⁵

The Nevada Test Site is located in Nye County, Nevada, and comprises an area of 371,158 ha. Of this area, 155,404 ha are used for nuclear testing programs and the remaining area provides a variety of possible radioactive waste management locations. At present, less than 81 ha are being used for waste management operations at NTS. Seven sites are used at NTS for the storage and disposal of radioactive waste: area 5 RWMS, R-MAD RWMS, U3ax crater, U3fi drillhole, Horn Silver mineshaft, U2bu crater, and U8d postshot drillhole.

12.1 STORAGE FACILITIES⁹⁰

Area 5 RWMS is used for both storage and disposal of radioactive waste and comprises an area of 15.7 ha. Pads are used for the storage of tritium-contaminated waste, low-activity TRU waste, and potentially reusable activated or contaminated hardware and equipment.

The R-MAD (reactor-maintenance, assembly, and disassembly) facility is used for the storage of reactor hardware remaining after the preparation of expended fuel elements and covers an area of 22.3 ha. This site was used in the past for both disposal and surface storage of radioactive hardware and waste from the nuclear rocket development program.

Twenty-six additional radioactive waste sites are used for surface storage at NTS. These are considered retired or inactive because no wastes are being added to the sites. Most of the waste stored at these sites consists of contaminated parts from a large exhaust deflector, reactor cores, and nuclear test wastes. Current plans are to remove the waste stored at these sites and consolidate it in a disposal site.

12.2 DISPOSAL FACILITIES

Area 5 RWMS is used for the disposal of classified wastes. The wastes are disposed of in trenches having a width and depth of about 3.2 m and a length of 90-180 m. Excavations are filled with radioactive waste to near surface level. Then a final soil cover at least 0.9-m thick is placed on top of the waste. TRU waste is restricted to area 5 RWMS, where it is stored on the surface in closely spaced, sealed drums. However, a storage pad is being constructed in FY-1978.

The total volume of TRU waste stored is 173.2 m^3 , which contains 3.0 kg of transuranics. No TRU waste was generated or stored in FY 1977 and 1978.⁵

13.0 ARGONNE NATIONAL LABORATORY (ANL)⁵

The Argonne National Laboratory (ANL) is located in Argonne, Illinois, and is operated by the University of Chicago for DOE.

There is no active burial ground at ANL; however, during the decommissioning of the original laboratory (in Chicago proper) building rubble was buried in the Cook County Forest Preserve. Other ANL-generated waste and building rubble were also buried at this surveillance site.

ANL produces a wide variety of solid radioactive waste suitable for land burial. About 40 m³ per year of solid waste results from liquid radwaste treatment. The solid waste produced annually is about 845 m³. In the past all waste was shipped to Sheffield, Illinois, for burial. Present shipping activities are unknown.

ANL also operates one facility at INEL where on-site burial is practiced.

14.0 OTHER DOE SITES⁵

14.1 WASTE QUANTITIES AND PRESENT DISPOSAL METHODS

The quantity of solid waste contaminated with transuranics expected to be generated annually at other DOE sites varies from about one cubic meter at Sandia Laboratory, to about 230 m³ for Argonne National Laboratory (Table F-14). Sites in favorable geographic locations (remote, dry, etc.) bury solid waste on site while those in less favorable locations ship to commercial burial grounds. Based on the experience of the larger national laboratories and production sites, about 10 percent of the total solid waste generated annually by these sites is expected to be non-LSA waste. About 60 percent of this would be expected to be shipped to a TRU waste repository in steel drums and about 40 percent in steel boxes or bins.

TABLE F-14

QUANTITIES OF SOLID WASTE CONTAMINATED WITH TRANSURANICS
EXPECTED TO BE GENERATED AT OTHER DOE SITES⁵

Site	Volume (m ³ /yr)	Burial (for transport and burial)	Containers	Combustible (Volume %)	Alpha Activity ^(a) (as shown) ^(b)
Brookhaven National Laboratory	5.7	off-site	1.2m x 1.8m x 2.4m steel bins	90	range: 1 μ Ci to 1m Ci avg.: 10 μ Ci
Battelle Memorial Institute	15.9	off-site	208-liter steel drums	75	range: unknown avg.: unknown
Lawrence Radiation Laboratory (Berkeley)	32.4	off-site	114- and 208-liter steel drums	90	range: 1 μ Ci to 1 Ci avg.: 100 μ Ci
Lawrence Livermore Laboratory	132.5	off-site	208-liter steel drums	75	range: 1 to 10 Ci avg.: 0.25 Ci

a. Alpha activity includes Pu-238, Pu-239, Am-241 and/or Cm-244.

b. Activity content of packages; not of a unit volume or weight.