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Vol. 2

Data Base for Radioactive Waste Management

Waste Source Options Report

Prepared by R. E. Wild, O. I. Oztunali, J. J. Clancy, C. J. Pitt, E. D. Picazo

Dames and Moore, Inc.

Prepared for
U.S. Nuclear Regulatory
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ACKNOWLEDGMENTS

The authors wish to acknowledge the invaluable contribution of G. Roles of the U.S. Nuclear Regulatory Commission in accumulation of the available information and his generous assistance in bringing this report into a final form. The assistance of C. Bishop and T. Johnson of the Low-Level Waste Licencing Branch, U.S. Nuclear Regulatory Commission during the definition of the data base and the editing of the report is also gratefully acknowleged.

1.0 INTRODUCTION

This report presents a data base on low-level radioactive waste (LLW) sources, as well as options for processing this waste. The data base includes estimates of:

- the physical, chemical, and radiological characteristics of LLW projected to be routinely generated during the period from 1980 to the year 2000,
- the changes in these characteristics under a number of viable waste treatment technology options;
- the costs for these waste treatment options based upon currently available technology, and
- data on occupational exposures and environmental releases from the waste treatment options.

These characteristics may be utilized to determine performance objectives and technical requirements for acceptable disposal of the wastes, and to determine the environmental impacts of selected disposal alternatives.

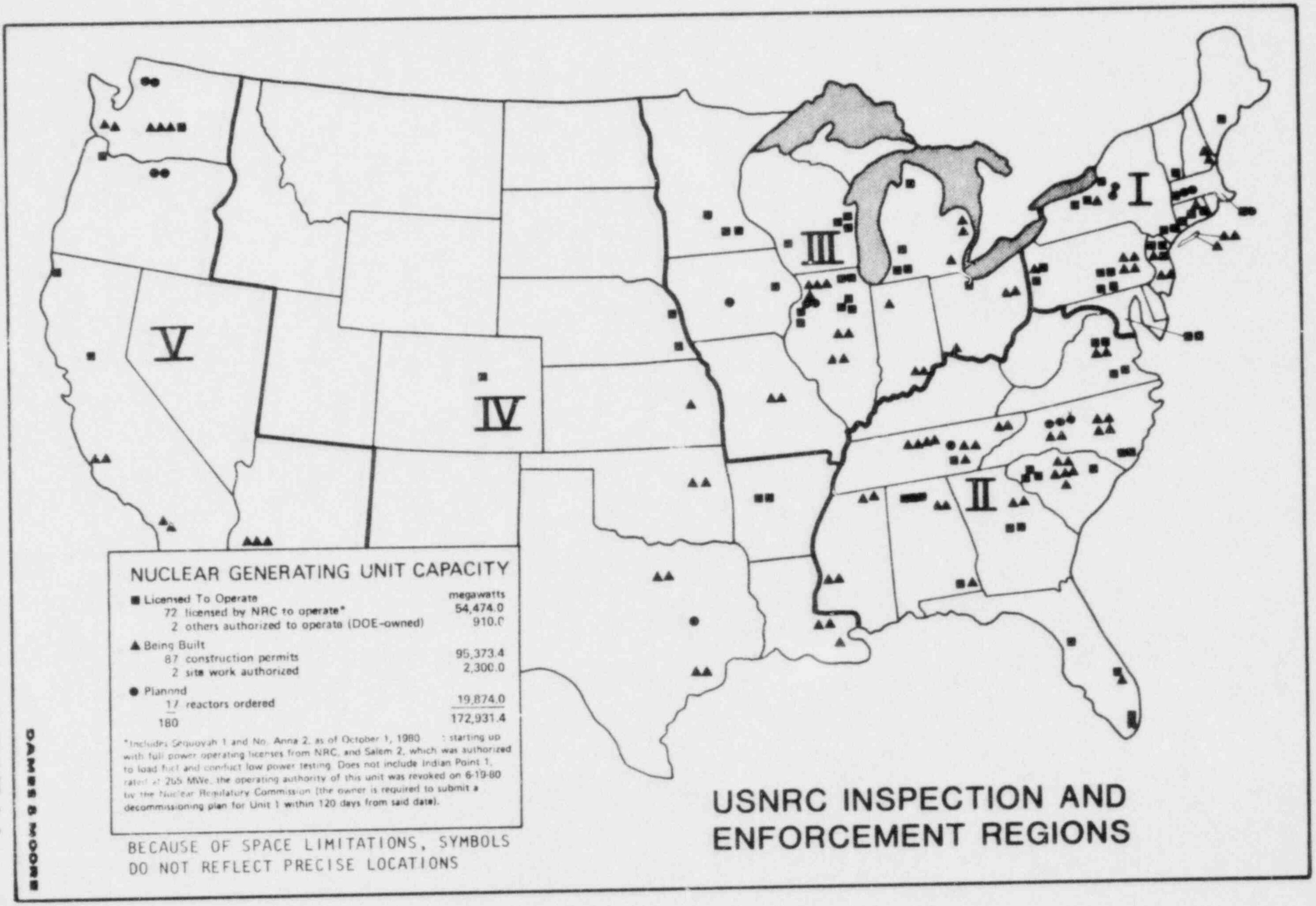
There are many facilities and diverse processes that generate radioactive waste, ranging from nuclear fuel cycle facilities to medical institutions and industrial facilities. To determine the environmental impacts of disposing of these wastes, their physical, chemical and radiological characteristics are estimated and projected on a regional basis over a time period from 1980 to the year 2000. Radioactive wastes with these projected characteristics are then assumed to be disposed into a reference near-surface disposal facility which is typical of existing disposal facilities. This provides a base case against which potential alternatives for waste form and disposal facility design and operation can be analyzed.

Several studies have been performed in the past on projected LLW characteristics and/or generation rates.(1-7) These studies have been limited in scope, and have concentrated on a specific portion of the subject that is considered in this report. They have provided general information and guidance, however, on specific generators and/or waste properties and have contributed significantly in the integration of the information into a flexible and comprehensive data base.

The regions considered as part of developing the waste projections are shown in Figure 1-1. The five regions range in number from 7 to 14 states each, and correspond to the five NRC Inspection and Enforcement (I&E) regions. Each region could represent a large multi-state compact formed for waste disposal.

Projecting regional waste generation to the year 2000 results in an upper-bound volume of waste produced during this period of about one million cubic meters (about 35 million ft^3) of waste per region, sufficient to fill a single disposal facility of up to a few hundred acres in size using existing disposal practices. Existing commercial disposal facilities range from twenty to a few hundred acres in size. A million cubic meters of waste corresponds to an average of 50,000 m^3 (1.77 million ft^3) of waste disposed per year over a period of 20 years, or about 4167 m^3 (147,000 ft^3) of waste per month. By comparison, the current limitation on monthly receipt at the Barnwell, S.C. disposal site is 200,000 ft^3 per month and this limit will be reduced to 100,000 ft^3/month by October, 1981.(8)

Within the last few years, a considerable amount of data has been generated on the characteristics of radioactive waste streams. Even so, in some cases the data is rather limited and simplifying assumptions are made as a result. The waste projections are also limited by the inherent variable nature of waste generation. Facilities producing waste may expand, reduce or otherwise modify operations,



NUCLEAR GENERATING UNIT CAPACITY

		megawatts
■ Licensed To Operate		
72	licensed by NRC to operate*	54,474.0
2	others authorized to operate (DOE-owned)	910.0
▲ Being Built		
87	construction permits	95,373.4
2	site work authorized	2,300.0
● Planned		
17	reactors ordered	19,874.0
180		172,931.4

*Includes Sequoyah 1 and No. Anna 2, as of October 1, 1980 : starting up with full power operating licenses from NRC, and Salem 2, which was authorized to load fuel and conduct low power testing. Does not include Indian Point 1, rated at 265 MWe, the operating authority of this unit was revoked on 6-19-80 by the Nuclear Regulatory Commission (the owner is required to submit a decommissioning plan for Unit 1 within 120 days from said date).

BECAUSE OF SPACE LIMITATIONS, SYMBOLS DO NOT REFLECT PRECISE LOCATIONS

USNRC INSPECTION AND ENFORCEMENT REGIONS

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FIGURE 1.1

depending upon governmental, social, or economic influences which are not readily predictable. Future development in waste treatment processes is also expected to alter the characteristics of the waste streams that are produced, as are regulatory requirements and actions.

Given the inherent uncertainties in waste projections over the next twenty years and beyond, this report has concentrated on wastes which are either presently being routinely generated, or are expected within the next few years to be produced in significant quantities. These include wastes from the present once-through uranium fuel cycle, institutional wastes, and radioactive industrial wastes. There are also a number of other waste streams which may be produced in the future -- e.g., wastes produced from recycle of uranium fuel -- but the timing for their generation, their production rates, and their characteristics are speculative at this time. These streams are discussed in a separate chapter in lesser detail. In any case, new waste streams will be continuously generated as processes change, new facilities are built, and so forth.

Development of the data base has been divided into three components: (1) the characteristics of untreated LLW, (2) the waste treatment systems which can be utilized and their potential effect on LLW, and (3) alternative LLW characteristics under several of these waste treatment options. The waste sources have been subdivided into a number of individual streams, each of which differ significantly in characteristics and generation sources. The individual waste streams are then regrouped into macro-streams which are distinguished by the macroscopic properties of the wastes. All of these streams are presently being generated and shipped to waste disposal facilities or have a reasonably high possibility of being generated by the year 2000. The detailed breakdown enables (1) rapid and flexible calculation of impacts, (2) incorporation of future waste treatment technologies, (3) a rapid increase in the number of waste streams considered, and (4) improvements in the accuracy of information in a given stream.

It is expected that much additional data on waste characteristics will be acquired over the next few years. Additional waste streams may also be identified. Therefore, the structure of the data base on waste characteristics has been designed to be flexible to incorporate new data in a straightforward manner.

Chapter 2.0 presents an overview of the waste generators, describes the waste sources (streams) that will be considered, presents a brief description of the processes that lead to the generation of these wastes and provides physical and chemical descriptions. Chapter 3.0 presents the characteristics (including volumes and radioactivity concentrations) of these waste streams prior to waste treatment. The waste processing and treatment options that can be applied to these streams are grouped according to their effect on waste volume -- i.e., volume reduction by compaction, evaporation and incineration, and volume increase by solidification, use of absorbents, and packaging -- and are presented in Chapter 4.0. Several impact measures (occupational exposures, population exposures, costs, and energy use) associated with selected waste processing options are also presented in Chapter 4.0. Chapter 5.0 presents the volumes and characteristics of alternative waste spectra (all the waste streams that are projected to be generated by the year 2000) after application of selected waste treatment options. Chapter 6.0 describes some of the potential waste streams which may be generated in the future, but for which projections of waste quantities potentially produced to the year 2000 are considered to be speculative.

Detailed calculations and an information base for Chapter 3.0 are presented in Appendices A and B, and a more extensive discussion of the information given in Chapter 4.0 is presented in Appendices C and D.

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2.0 WASTE STREAM DESCRIPTIONS

This chapter provides a description of the waste streams which are presently being routinely generated or are expected to be routinely generated in significant quantity in the near future. Section 2.1 is an overview of current waste generators, which comprise nuclear fuel cycle waste generators and non-nuclear fuel cycle waste generators. Sections 2.2 through 2.5 provide a more detailed discussion of the waste streams produced by these waste generators.

This section presents a brief description of the waste stream sources as well as some of the physical and chemical characteristics of the waste streams. The information on the volumes and radiological characteristics is presented in Chapter 3.0, and information on the waste processing technologies (including packaging) that are currently applied to these streams and that may be applied in the future can be found in Chapter 4.0.

2.1 Overview of Waste Generators

In this report, 25 distinct waste streams have been considered in detail and are summarized in Table 2-1. As shown in the table, the 25 waste streams may be grouped into the following five major waste sources, which include three generic fuel cycle sources and two generic non-fuel cycle sources:

- Nuclear fuel cycle
  - Central station nuclear power plants
  - Fuel fabrication plants
  - Uranium hexafluoride (UF_6) conversion plants

- Non-nuclear fuel cycle
  - Institutional facilities
  - Industrial facilities

TABLE 2-1

Waste Sources and Streams

Nuclear Fuel Cycle	Abbreviation
Central Station Nuclear Power Plants	
Ion Exchange Resins	IXRESIN
Concentrated Liquids	CONCLIQ
Filter Sludges	FSLUDGE
Cartridge Filters	FCARTRG
Compactible Trash	COTRASH
Noncompactible Trash	NCTRASH
Nonfuel Reactor Components	NFRCOMP
Decontamination Resins	DECONRS
Fuel Fabrication Facilities	
Process Wastes	PROCESS
Compactible Trash	COTRASH
Noncompactible Trash	NCTRASH
Uranium Hexafluoride Plants	
Process Wastes	PROCESS
Non-fuel Cycle	
Institutional Facilities	
Liquid Scintillation Vials	LIQSCVL
Absorbed Liquid Waste	ABSLIQD
Biowaste	BIOWAST
Trash	COTRASH
Industrial Facilities	
Waste from Isotope Production Facilities	ISOPROD
High Activity Waste	HIGHACT
Tritium Manufacturing Waste	TRITIUM
Sealed Sources	SOURCES
Accelerator Targets	TARGETS
Source and Special Nuclear Material Waste	SSWASTE
Source and Special Nuclear Material Trash	SSTRASH
Low Activity Waste from Various Sources	LOWASTE
Low Activity Trash from Various Sources	LOTRASH

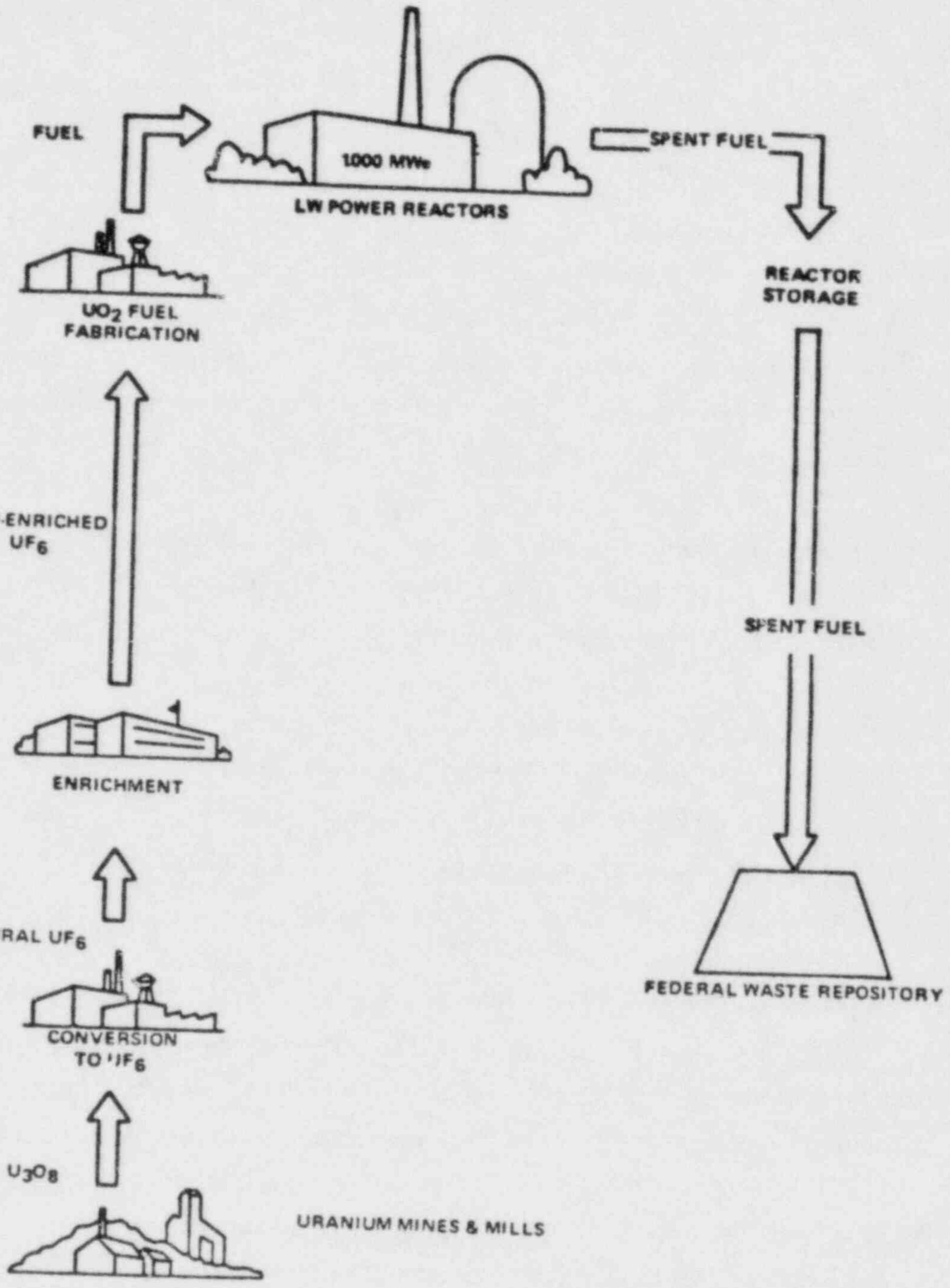
A brief overview of waste generation by nuclear and non-nuclear fuel cycle facilities is presented below.

2.1.1 Nuclear Fuel Cycle

Nuclear fuel cycle waste generators include facilities involved in the commercial generation of electrical power through the use of nuclear energy. The current fuel cycle is based upon once-through use of uranium fuel as shown in Figure 2-1.(1)

The nuclear fuel cycle begins with mining and milling of uranium ore. Uranium ore is generally obtained from either open pit or underground mines and is usually shipped to a centralized mill for processing. Uranium mills convert uranium ore - which usually contains between 0.1 to 2 weight percent uranium - to yellowcake, which consists primarily of U_3O_8. Disposal of liquid and solid wastes generated as part of milling operations has been already addressed in an NRC rulemaking action and is not considered further in this report. Additional information can be located in NUREG-0706.(2)

Yellowcake produced from milling operations is then shipped to conversion plants which convert U_3O_8 to uranium hexafluoride (UF_6). This conversion is utilized since: (1) current central station nuclear power plants are designed for operation with uranium enriched in the fissile isotope U-235, (2) the major enrichment technology currently utilized is based on the gaseous diffusion process, and (3) UF_6 is an easily volatilized compound of uranium suitable for the gaseous diffusion process. The conversion process generates liquid and solid waste streams, most of which are recycled to recover uranium prior to storage in on-site ponds or reuse within the plant. On-site storage at conversion facilities is presently regulated by NRC under 10 CFR Part 40. Small quantities of low-activity wastes contaminated with natural uranium are shipped off-site to near-surface disposal facilities. These wastes are discussed further in this report. Currently,



ONCE-THROUGH URANIUM FUEL CYCLE

DAMES & MOORE

FIGURE 2-1

there are two UF_6 conversion facilities in operation in the United States, one plant is located in Region III and one in Region IV.

Following conversion, natural UF_6 is shipped to federally owned facilities for enrichment in the fissile isotope U-235. In this process, the U-235 content of the uranium is raised from natural concentrations (about 0.7 weight percent) to 2 to 4 weight percent. Currently, three enrichment plants using the gaseous diffusion process are in operation at Portsmouth, Ohio; Paducah, Kentucky; and Oak Ridge, Tennessee. These plants are owned and operated by the Federal government and wastes produced from plant operation are not sent to commercial disposal facilities. Hence, waste streams produced from uranium enrichment operations are not considered further in this report.

Enriched UF_6 is then shipped to commercial fabrication plants which convert the enriched UF_6 to uranium dioxide (UO_2) powder, produce UO_2 pellets, fabricate fuel rods containing the UO_2 pellets, and combine the fuel rods into fuel assemblies for use in light water reactors. Most of the liquids, sludges, and other wastes produced during the UF_6-to-UO_2 conversion process are presently being stored at the fabrication plants, although some wastes in the form of dry solids (principally CaF_2) contaminated with low levels of enriched uranium are being shipped off-site for disposal. Low-activity waste, principally trash, is also generated during the pelletizing and subsequent fabrication processes, and is also shipped off-site for disposal. Table 2-2 provides a summary of the current LWR fuel fabrication industry.

Fuel assemblies are then shipped to central station nuclear power plants, utilizing light water power reactors (LWR) for production of electrical power through use of the energy released during fission of the uranium fuel. During operations, waste is generated in a number of forms having specific activities ranging from low to moderately

TABLE 2-2

Current LWR Fuel Fabrication Industry

Licencee and Plant Location	Plant Feed Material	Plant Product	Plant Capacity (MTU/yr) Current	Estimated 1985
Babcock & Wilcox Lynchburg, VA (2)[a]	UO_2 pellets UF_6	Fuel assys	230	830[b]
Babcock & Wilcox Apollo, PA (1)	UF_6			c
Combustion Engineering Hematite, MO (3)	UF_6	UO_2 powder or pellets		d
Combustion Engineering Windsor, CT (1)	UO_2 powder	Fuel assys	150	150
Exxon Nuclear Richland, WA (5)	UF_6	Fuel assys	665	1,030[e]
General Electric Wilmington, NC (2)	UF_6	Fuel assys	1,500	1,500
Westinghouse Electric Columbia, SC (2)	UF_6	Fuel assys	750	1,600

---

a Region number.

b Babcock and Wilcox (B&W) plans to expand operations to increase capacity to 1,200 MTU/yr by the early 1990's. The capacity listed in the table for 1985 is an interpolation of present and future capacity. In addition, a UF_6 to UO_2 conversion operation will be added as well as a UO_2 pelletizing operation.

c Currently, the B&W Apollo plant converts UF_6 to UO_2 powder and ships the UO_2 to its Lynchburg plant for fabrication into fuel assemblies.

d The Combustion Engineering (CE) Hematite plant produces UO_2 pellets or powder which are then transferred to the CE Windsor plant for fabrication into fuel assemblies.

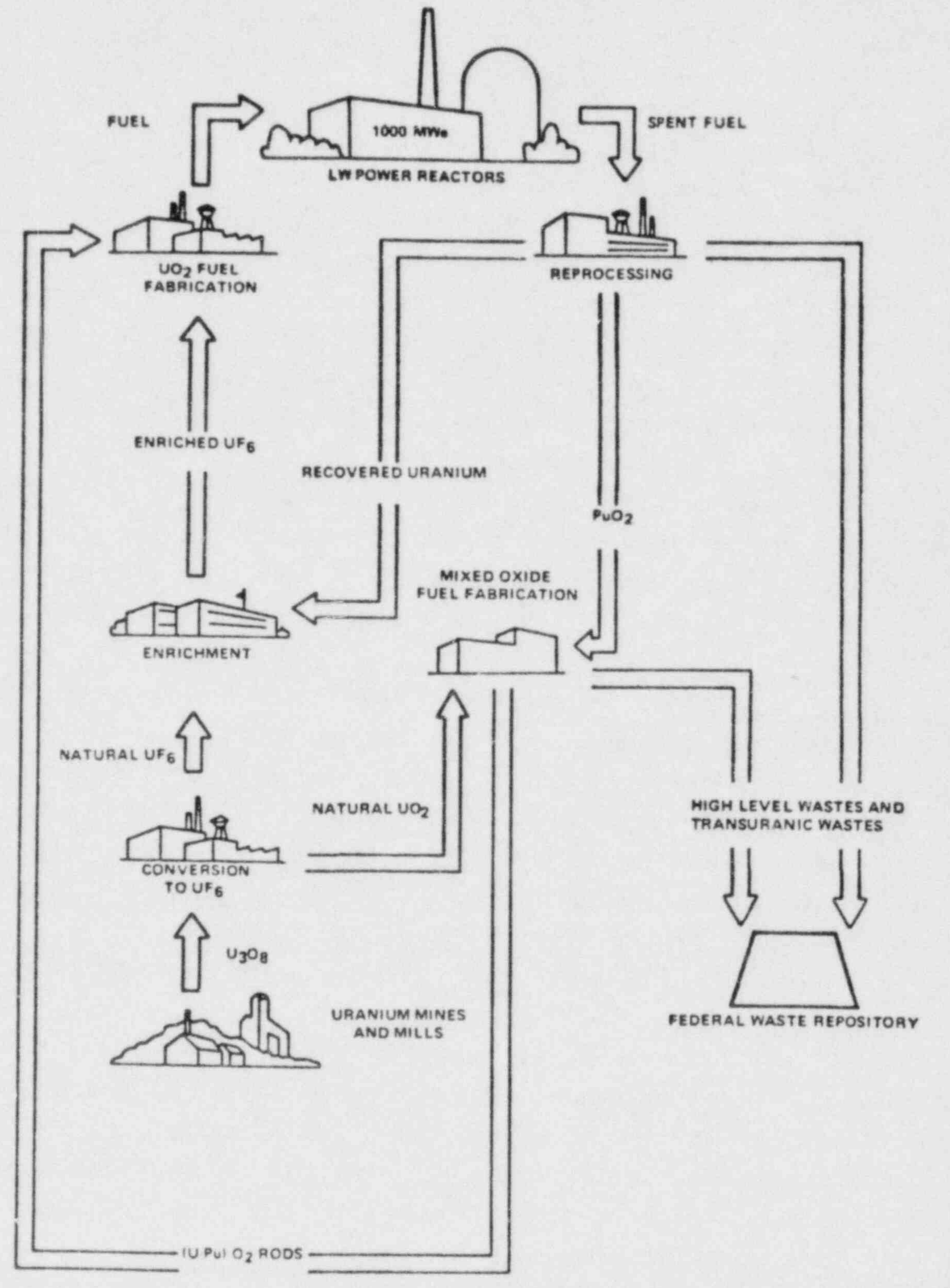
e Expanded to 1,030 MTU/yr in 1980.

Source : NRC Data

high levels. Much, if not most, of the waste is generated as a result of operating and maintaining plant processes which maintain concentrations of radiocontaminants in the reactor coolant and other process systems to low levels and reduce effluent releases from the plant to acceptable levels. The presence of such radiocontaminants in reactor coolant systems can result from activation of corrosion products or from leakage of fission products out of the fuel rods. The treatment and maintenance operations result in wet wastes such as filter sludge, spent resins, evaporator bottoms, as well as compactible and non-compactible dry wastes. Liquids such as evaporator bottoms are solidified while other wet wastes such as ion-exchange resins are generally dewatered and packaged into containers for shipment. Some compaction is usually performed on compactible dry wastes. The wastes are then generally shipped off-site for disposal in commercial facilities.

Currently, there are 74 light water power reactors in operation in the United States, of which 48 are pressurized water reactors (PWR), and 26 are boiling water reactors (BWR). There is also one operating high-temperature gas-cooled reactor (HTGR). The locations of these operating reactors, as well as the locations of the reactors under construction, are shown in Figure 1-1.

The fuel used in the reactors must be periodically replaced. Generally, about one-third of the fuel in the reactor core is replaced approximately every 12 to 18 months. Most of this spent fuel is stored at the power stations within large spent fuel holding pools. A small fraction of this fuel, however, is presently stored off-site in fuel pools located within two facilities originally designed to reprocess the fuel. However, one facility (in New York) suspended reprocessing operations in 1971, and the other (in Illinois) never became operational. Additional facilities specifically constructed for storage of spent fuel may be constructed in the future, these may be located either at the operating reactor sites or at away-from-reactor (regional) sites.



URANIUM AND PLUTONIUM
RECYCLE FUEL CYCLE

FIGURE 2-2

The ultimate disposition of the spent fuel is uncertain at this time. One option is to treat the spent fuel as high level waste and dispose of the spent fuel in a Federal repository to be constructed and operated by the Department of Energy (DOE). Another option is to recycle the spent fuel as shown in Figure 2-2.

In this option, spent fuel would be shipped to a reprocessing plant which, using chemical separation processes, would recover residual uranium and plutonium for reuse in reactors. Recovered uranium would be shipped as UF_6 to an enrichment plant for enrichment in U-235. Recovered plutonium would be shipped as plutonium dioxide (PuO_2) powder to a mixed oxide (MOX) fuel fabrication plant where it would be combined with natural UO_2 and fabricated into MOX fuel rods. The mixed oxide fuel rods would then be shipped to a fabrication plant where the MOX fuel rods would be combined with natural uranium fuel rods and assembled into fuel assemblies for reinstallation into LWR's. High level and transuranic wastes generated during reprocessing and MOX fuel fabrication operations would be shipped to a Federal repository for disposal.

For the last four years, the policy of the United States as announced by former President Carter has been to defer the uranium recycle option. There are no reprocessing or MOX fuel fabrication facilities operating in the country and spent fuel removed from nuclear power reactors is currently being stored pending operation of a Federal repository. It is possible that in the future, the country's policy on uranium fuel recycling may change. However, at present the timing and extent of future fuel reprocessing and MOX fuel fabrication operations are speculative, as is the quantity of waste to be generated through such operations.

2.1.2 Non-Fuel Cycle Waste Generators

Non-fuel cycle waste generators include the approximately 20,000 facilities licensed by NRC or Agreement State agencies to use radioactive materials. Non-fuel cycle waste generators may be classified as either institutional or industrial.

Institutional waste generators include hospitals, medical schools and research facilities, colleges, and universities. Waste generation rates and waste characteristics vary significantly between institutional waste generators and it is therefore difficult to consider each type of institution as a separate waste generator. Therefore, all institutional facilities are considered as a single waste source in this report.

Industrial waste generators are also considered as a single waste source for the same reason, and include industries which produce and distribute radionuclides, manufacture materials containing radioisotopes for industrial uses, and use radioisotopes in laboratory studies, instruments, devices, and manufacturing processes. Industrial waste generators have not been surveyed to as great an extent as other types of waste generators.

2.2 Central Station Nuclear Power Plants

Central station nuclear power plants presently in operation in the United States include 74 light water reactors (LWR's) and a single high temperature gas cooled reactor (HTGR). The waste generated by the single HTGR is volumetrically and radiologically negligible compared with the wastes generated by LWR's,(3) and is therefore not considered further in this report.

Electricity for commercial use is also generated as a by-product of the Hanford "N" plutonium production reactor and the Shippingport light water breeder reactor. Wastes generated by these facilities are disposed in facilities operated by the Department of Energy (DOE) and not in commercial disposal facilities.

Two types of LWR's are in operation today: pressurized water reactors (PWR's) and boiling water reactors (BWR's). The majority of the LWR waste streams are generated by operation of in-plant liquid radwaste processing systems. An overview of these systems excerpted from reference 4 is presented in Section 2.2.1. Five waste streams are common to PWR's and BWR's: spent ion exchange resins which result from the use of deep bed ion exchangers and/or demineralizers, concentrated liquids (evaporator bottoms) which result from the use of evaporators, filter sludges which result from the use of pre-coat filters, and trash (compactible and non-compactible) which results from many functions performed at LWR's. Cartridge filters are another form of waste but are used much more extensively in PWR's than in BWR's. These waste streams are considered in detail in Sections 2.2.2 through 2.2.6. Other waste streams that are expected to be generated during LWR operations are discussed in Section 2.2.7.

2.2.1 Overview of Liquid Radwaste Processing Systems

The basic functions of liquid radioactive waste (radwaste) processing systems are to reduce the accumulation of radioactive contaminants within the plant and to reduce the amount of these contaminants released from the plant. In so doing, radioactive contaminants are concentrated within the processing systems in several forms. These processing systems are typically somewhat different for BWR's and for PWR's and are considered below.

2.2.1.1 BWR Systems

Boiling water reactors route steam generated in the primary coolant as it circulates through the core directly to the turbines. The steam is then condensed, treated, and pumped back to the reactor core. The systems used to treat primary coolant and liquid radwaste are briefly described in this section. Typical BWR radwaste treatment systems are shown in Figures 2-3 and 2-4. The main difference between the two systems is the type of condensate polishing performed.

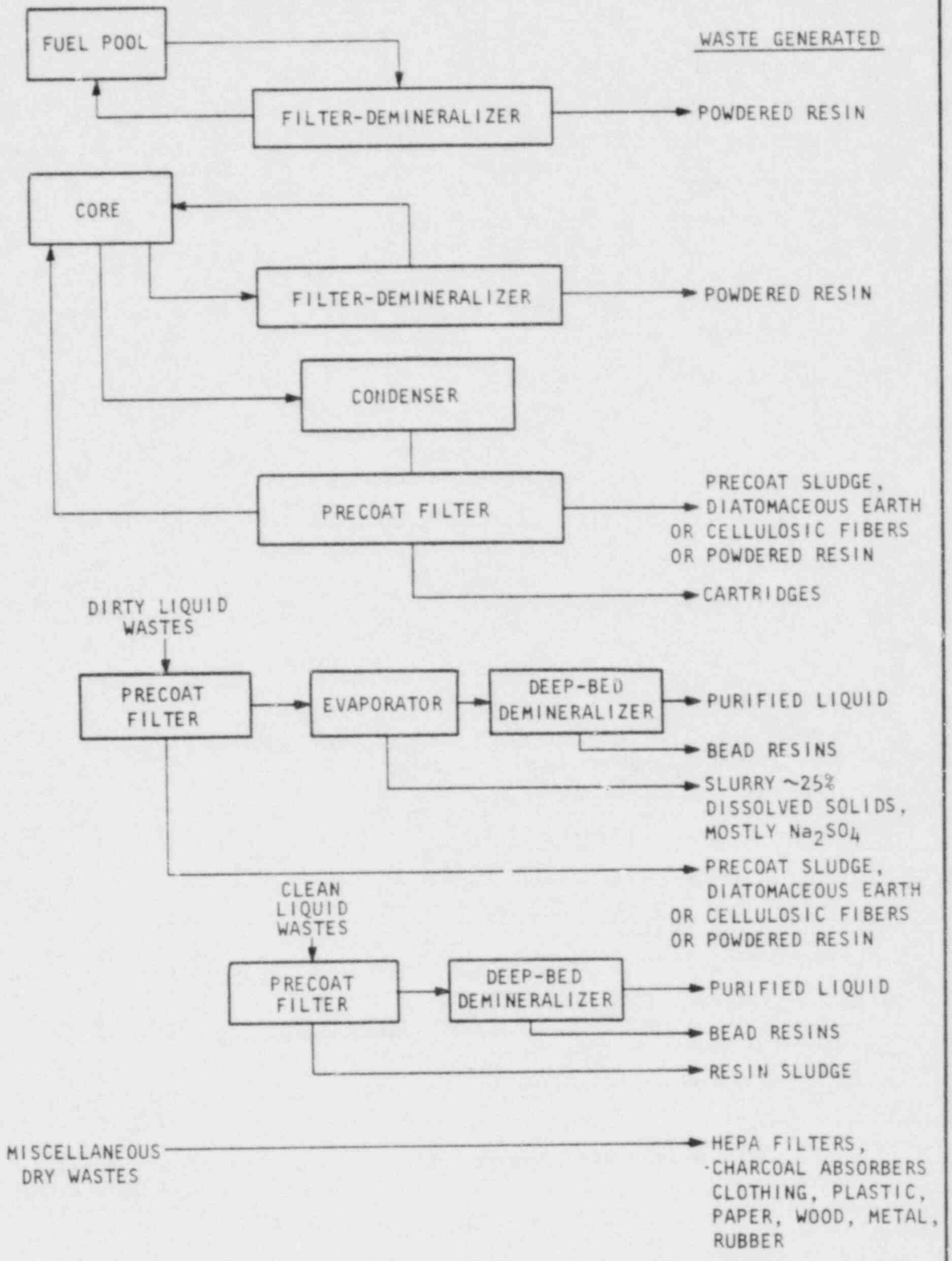
Under operation, a fractional volume of the coolant is bled off and routed to the reactor water cleanup system where the bled coolant is treated to remove suspended and dissolved solids. Pre-coat filter/dimeralizers are used alone or in combination with cartridge filters, pre-coat filters, or deep bed ion exchangers.

The condensate polishing system processes coolant after it has been routed to the turbines as steam and condensed. Suspended and dissolved solids are removed by deep bed exchangers (Figure 2-3) or pre-coat filter/demineralizers (Figure 2-4). Both treatment methods are sometimes used in conjunction with cartridge filters. Pre-coat filters are often used with deep-bed ion exchangers.

The spent fuel pool cleanup system uses pre-coat filter/demineralizers, pre-coat filters, cartridge filters, or cartridge filters and deep-bed ion exchangers in series to remove dirt and radioactive contaminants from fuel pool water.

The clean radwaste system collects and processes liquids expected to have conductivities of less than about 10 µmho/cm. Such liquids usually consist of leaking water collected by the equipment drains. Pre-coat filters and filter/demineralizers, deep bed ion exchangers, and cartridge filters are used for treatment. The effluent is either recycled or discharged.

The dirty radwaste system collects and treats liquids expected to have conductivities between about 10 and 200 µmho/cm. These liquids are collected by floor drains. The liquids may be processed by the



TYPICAL BWR RADWASTE PROCESSING SYSTEM WITH PRE-COAT FILTER CONDENSATE POLISHING

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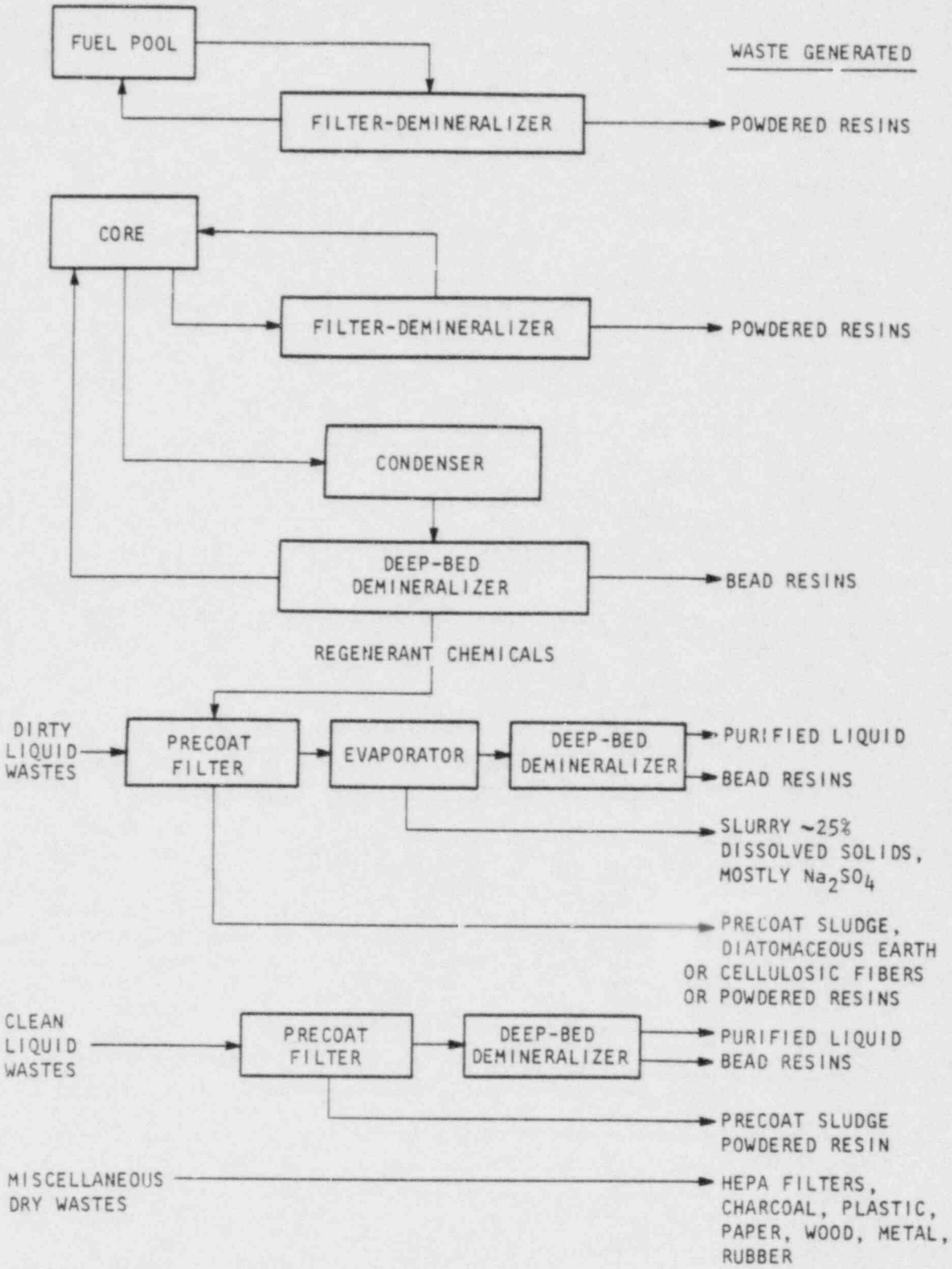
FIGURE 2-3



TYPICAL BWR RADWASTE PROCESSING SYSTEM WITH DEEP BED CONDENSATE POLISHING same methods in the clean radwaste system, or evaporated and solidified for disposal.

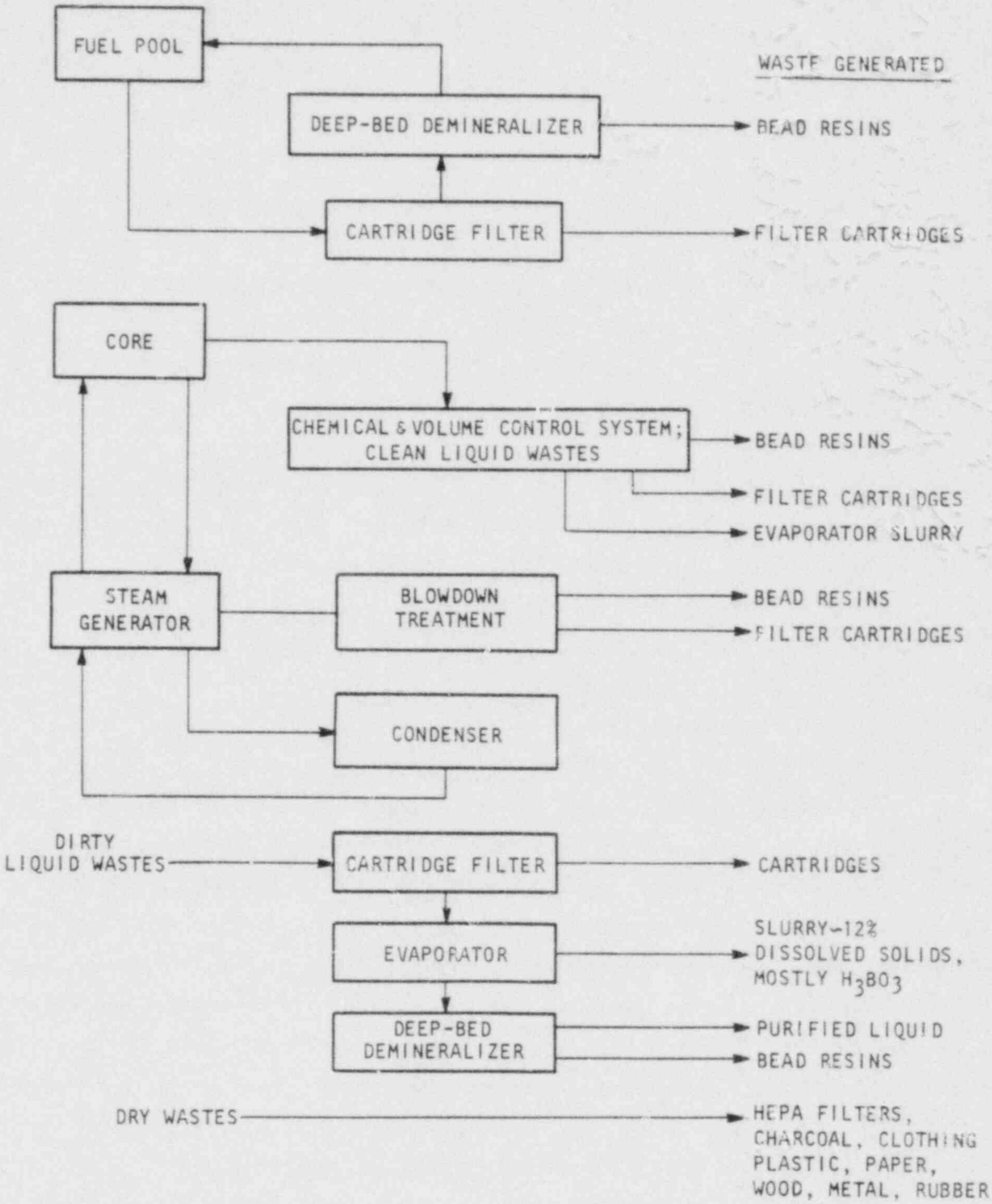
The chemical radwaste system collects and processes other high conductivity liquids. These include liquids from resin regeneration, equipment decontamination, and laboratory drains. Many older plants process these liquids through the same equipment used for dirty radwaste. Newer plants use a separate system which often includes an evaporator for processing.

The laundry waste system usually includes cartridge filters, pre-coat filters, or pre-coat filter/demineralizers. Laundry wastes are processed through these systems and discharged. Several future plants include evaporators in the system design. Laundry waste treatment systems are not shown in Figures 2-3 and 2-4.

2.2.1.2 PWR Systems

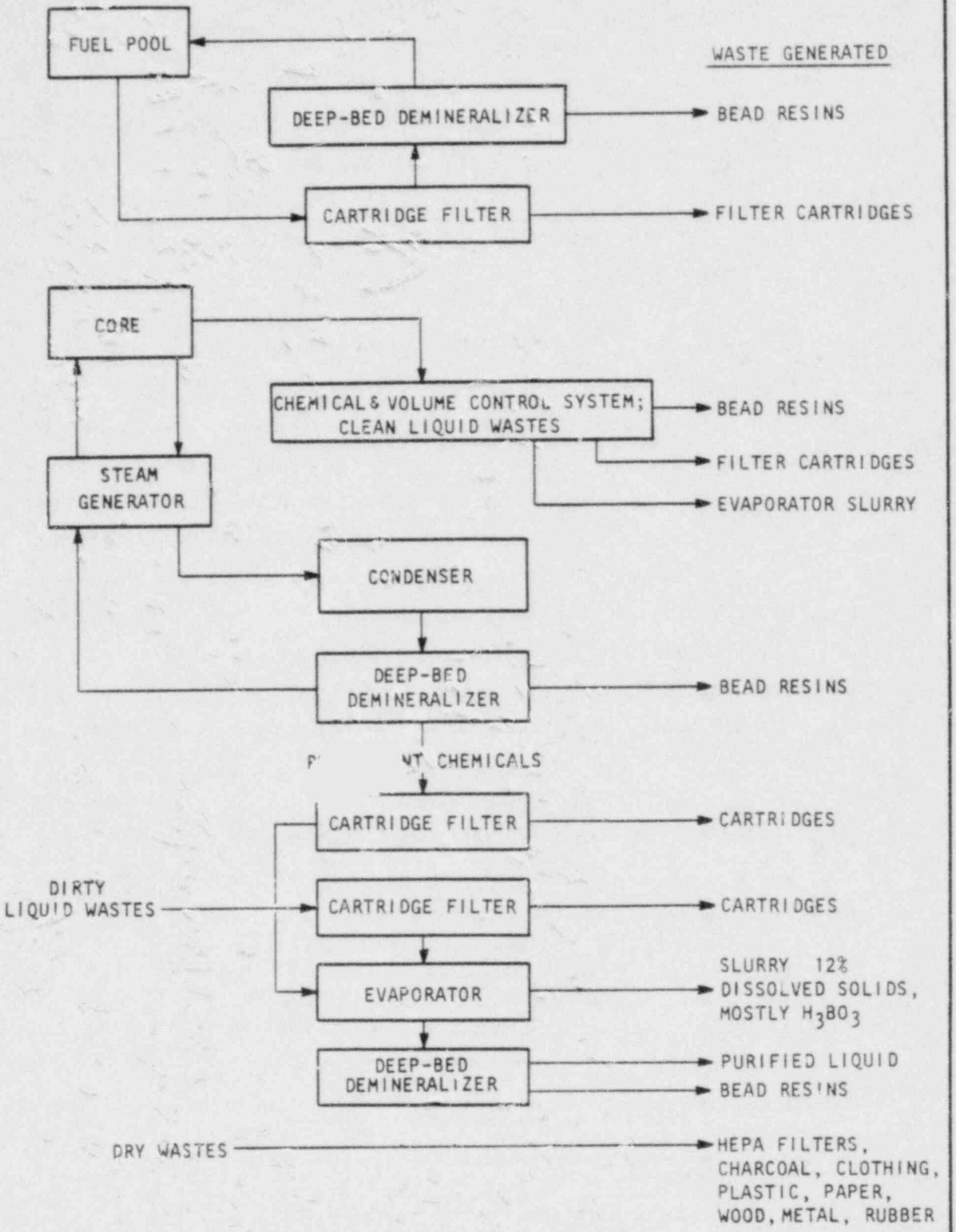
Pressurized water reactors use indirect means to drive the turbines. The primary coolant is kept under high pressure to prevent boiling and is recirculated through the core to the steam generators to induce boiling in the secondary coolant. Steam from the secondary coolant drives the turbines, is condensed, usually treated, and returned to the steam generators. The systems used to treat primary and secondary coolant and liquid radwaste are described in this section. Typical PWR radwaste treatment systems are shown in Figures 2-5 and 2-6.

The function of the chemical and volume control system is to maintain the quantity and quality of the primary reactor coolant. The primary coolant contains fission and corrosion products and small amounts of transactinide elements which escape from the reactor fuel elements. The water quality of the coolant is maintained by bleeding a small fraction of the coolant from the discharge of the reactor circulation pumps and removing contamination by filtration and/or ion-exchange.



TYPICAL PWR RADWASTE SYSTEM WITH PARTIAL FLOW CONDENSATE POLISHING

FIGURE 2-5



TYPICAL PWR RADWASTE PROCESSING SYSTEM
WITH FULL FLOW CONDENSATE POLISHING

DAMES & MOORE

FIGURE 2-6

The boron control system adjusts the boron concentration of the primary coolant as required by reactor operating conditions. This is accomplished by the use of special ion exchange resins which retain certain boron compounds at low temperatures and release them at higher temperatures. The boron control system is connected to the chemical and volume control system.

The steam generator blowdown system condenses the steam generated in the secondary cooling system, and returns it to the steam generator. Radioactive and non-radioactive contamination can enter the secondary coolant from the primary coolant through leaks in steam generator tubing. In older plants, this contamination is controlled by partial replacement (rather than treatment) of the secondary coolant or, more commonly, by treating a portion of the condensate by filtration and/or by ion exchange (Figure 2-5). Newer plants use a full flow condensate polishing system (Figure 2-6) to treat all of the condensed steam by filtration and/or ion exchange before returning it to the steam generator.

The spent fuel pool cleanup system removes activated corrosion products which break free of the fuel rods, fission products which leak from the fuel, and dust which falls into the pool. Removal is accomplished by filtration and ion exchange.

The chemical waste system collects and treats liquid waste from regeneration of ion exchange resins, equipment decontamination, and from in-plant chemistry laboratories. These wastes are collected and sometimes processed separately from other liquid wastes. These wastes are normally processed by evaporation.

The laundry waste system collects and filters laundry waste liquids. The filtrate is usually discharged or recycled, however, a few plants evaporate the filtrate. Some plants use reverse osmosis to remove contaminants.

The miscellaneous liquid waste system collects and treats liquids from floor drains and leaking equipment by filtration or by ion exchange or both. Treated liquids are either discharged or recycled. These miscellaneous liquids may be collected and processed by the same equipment as chemical wastes or by a separate system.

2.2.2 Spent Ion Exchange Resins

Ion exchange resins are used extensively in both BWR's and PWR's as indicated in Table 2-3. The resins, which are made from organic polymers, are used in the form of small (about 1 mm diameter) beads or granules and are commonly referred to as bead resins. Use of powdered ion exchange resins is discussed in Section 2.2.4.

Some resins are specifically designed to remove cations and others to remove anions from liquids. Cation and anion resins may be used alone, in sequence, or simultaneously as a mixture of the two types. The resins are usually packed in cylindrical tanks and the contaminated liquid is passed through the tank. These tanks have heights greater than their diameters to increase the contact time of the contaminated liquid with the resin. The unit is called a deep bed demineralizer if it contains both cation and anion resins and a deep bed ion exchanger if it contains only one type of resin. As the waste liquid flows through the resin bed, dissolved radiocontaminants chemically replace (exchange with) ions in the resins.

In general, for contaminants present in roughly equal concentrations more highly charged ions are more strongly bound to the resin than those with lower charge. For ions of the same charge and roughly equal concentrations, those with the larger (less hydrated) radius are more strongly bound. This process continues until the ion exchange capacity of the resin is exhausted. At this point, the spent resin is either replaced or regenerated. Resins are regenerated by washing them with water containing a high concentration of the ion originally

TABLE 2-3

Application of Ion Exchange Resins in LWR's

BWR's	PWR's
Condensate polishing system	Condensate polishing system
Reactor water cleanup	Chemical and volume control system
Clean radwaste system	Boron control system
Dirty radwaste system	Spent fuel pool cleanup
Chemical waste system	Steam generator blowdown system
Spent fuel pool cleanup	Miscellaneous waste system
	Chemical waste system

bound to the resin, sulfuric acid (H_2SO_4) is commonly used for cation resins and sodium hydroxide (NaOH) for anion resins. Because of the high regenerant concentration, radiocontaminants are displaced from the resin. The regeneration solutions are sent to the chemical radwaste system (see Section 2.2.3). Regeneration can be repeated several times before resin replacement becomes necessary.

Spent resins are transferred out of the tanks and into shipping containers as a slurry. The excess water is removed before shipment to a disposal site. Removal of the free water is called dewatering. Dewatered resins typically contain 42 to 55 percent water.(5) Most of this water is absorbed into the resin and is not mobile, however, some exists as interstitial water. Spent resin waste in shipping containers is generally transported in casks which provide radiation shielding.

Although there is little data available on the physical properties of PWR spent resins, they are expected to be similar to those of BWR spent resins and unused resins. An average density of 0.91 g/cm^3 has been reported in one survey.(5) This value is slightly above the range of typical fresh resin densities (0.67 to 0.85 g/cm^3).(5) The higher average density may be due to the presence of additional absorbed water and/or the decrease in the volume of cation resin which occurs as exchange sites are occupied by ions other than hydrogen.

Bead resins consisting of cross-linked styrene-divinylbenzene polymer are the most common type of ion exchange resin used in LWR's. These resins contain functional groups which bind exchangeable cations or anions. Cation resins containing highly acidic sulfonic acid functional groups ($-SO_3^-$) and anion resins containing quaternary ammonium functional groups ($-NH_3^+$) are best suited for most applications. Cation exchange sites are normally occupied by hydrogen ions (H^+) although sodium (Na^+) and lithium-7 (Li^+) forms are also used. Anion exchange sites are normally occupied by hydroxide ions (OH^-)

with chloride (Cl^-) and carbonate (CO_3^{-2}) forms being less common. The nature and amounts of other chemical species in spent resin waste are dependent on the type of liquids processed by the resins and the liquids used for regeneration. These spent resins are expected to contain ionic species bound to the ion exchange sites that are not removed during regeneration and the particulate matter not removed prior to treatment.

Gases such as carbon dioxide (CO_2) and oxides of nitrogen (NO_X) and sulfur (SO_X) can be produced in spent resins as a result of the combined effects of chemical and radiolytic decomposition.(6) These processes can be augmented by biological decomposition in the disposal environment. Inorganic ion exchangers made from natural and synthetic zeolites have a greater resistance to decompositon but are rarely used in LWR's due to their lower ion exchange capacities.

2.2.3 Concentrated Liquids

Concentrated liquid waste is produced by the evaporation of a wide variety of LWR liquid streams. The concentrated waste consists of liquids with an elevated suspended and dissolved solids content and of sludge which results from supersaturation during evaporation. The sources of these liquid streams, many of which are interrelated, are listed in Table 2-4.

Concentrated liquids from BWR's have a higher average total solids content than those from PWR's. This difference is probably due to more extensive resin regeneration in BWR's as compared to PWR's. Sulfuric acid used to regenerate cation resins forms sodium sulfate when mixed with sodium hydroxide used to regenerate anion exchange resins. Sodium sulfate is the primary chemical constituent in BWR concentrated liquids and is more soluble than boric acid, the primary chemical constituent in PWR concentrated liquids. Table 2-5 lists chemical species commonly found in concentrated liquid waste.

TABLE 2-4

Sources of Liquids Concentrated by Evaporation in LWR's

BWR's	PWR's
Regeneration of resins	Regeneration of resins
General decontamination waste liquids	General decontamination waste liquids
System effluents from :	System Effluents from :
Clean radwaste	Liquid radwaste
Dirty radwaste	Chemical radwaste
Chemical radwaste	Laundry waste
Laundry waste	Steam generator blowdown

TABLE 2-5

Chemical Species Found in LWR Concentrated Liquids

BWR	PWR
Anti-foaming agent	Ammonia
Calcium	Boric acid
Carbonate	Boron
Chloride	Calcium
Citric acid	Chloride
Ethylenediamine tetracetic acid (EDTA)	Chromate
Fluorides	Citric and oxalic acids
Iron	EDTA
Magnesium	Fluoride
Miscellaneous organics and Oils	Iron
Oxides	Magnesium
Permanganate	Miscelleneous organics and Oils
Phosphate	Nitrate
Potassium	Permanganate
Silica	Potassium
Sodium	Silica
Sulfate	Sodium
	Sulfate
	Thiosulfate

Source : References 5,7.

Concentrated liquids from BWR's have an average pH of 9 (range 4.5 to 12), an average solids content of 25 percent by weight (range 7 to 50), and an average density of 1.2 g/cm^3.(5) Concentrated liquids from PWR's have an average pH of 6.5 (range 4 to 9), an average solids content of 11.4 percent by weight (range 2 to 20), and an average density of 1.00 g/cm^3.(5) These concentrated liquids are currently solidified in various matrix materials including urea-formaldehyde and cement prior to transfer to a disposal site.

2.2.4 Filter Sludge

Filter sludge is waste produced by pre-coat filters and consists of filter aid and waste solids retained by the filter aid. Diatomaceous earth, powdered mixtures of cation and anion exchange resins, and high purity cellulose fibers are common filter aids.(5,8) These materials are slurried and deposited (pre-coated) as a thin cake on the initial filter medium (wire mesh, cloth, etc.). The filter cake removes suspended solids from liquid streams. Pre-coat filters using powdered resins also remove dissolved solids but are not as effective as deep bed demineralizers (mixed bed ion exchange columns) due to the shorter contact time of the liquid with the resin.

The application of pre-coat filters in LWR's is summarized in Table 2-6. Although pre-coat filtration is applied to similar functional systems in PWR's and in BWR's, the extent of application is much greater in BWR's. Condensate polishing generates the largest volume of sludge in both PWR's and BWR's. Pre-coat filtration may be used in conjunction with ion exchange columns and evaporation, or it may be the only form of treatment removing suspended solids from a particular liquid stream.

The bulk properties of PWR and BWR filter sludge are similar since both consist mainly of the same pre-coat materials. The average density of unsolidified filter sludge is 0.86 g/cm^3.(5) Small amounts

TABLE 2-6

Application of Pre-Coat Filters
and Cartridge Filters in LWR's

BWR's	PWR's
Condensate polishing system	Steam generator blowdown
Reactor water cleanup	Condensate polishing system
Spent fuel pool cleanup	Boron control system
Equipment and floor drains	Spent fuel pool cleanup
Chemical waste system	Laundry waste system
Laundry waste system	

of carbon dioxide and other gases can be generated from powdered resin and cellulose sludge due to chemical and biological attack and/or by radiation damage. Diatomaceous earth is composed of silica (SiO_2) which is more resistant to these types of attack. Crud (metal oxides) and dirt are the predominant types of filtered solids. Sludges from filter/demineralizers also contain species removed from liquid wastes by ion exchange. Currently most LWR's dewater but do not solidify filter sludges before shipment for disposal.

2.2.5 Cartridge Filters

Cartridge filters contain one or more disposable filter elements. These elements may be typically constructed of woven fabric, wound fabric, or pleated paper supported internally by a stainless steel mesh, as well as pleated or matted paper supported by an external stainless steel basket.(8) Paper filter elements are often impregnated with epoxy. Woven fabric filters are typically constructed of cotton and nylon. Cartridge filters are effective in removing suspended solids, but do not have the ion exchange capacity of filter/demineralizers.

Cartridge filters are used to treat the same streams which are processed through pre-coat filters (see Table 2-6), and are used much more extensively in PWR's than BWR's. Many plants use cartridge filters in conjunction with ion exchange columns, evaporators, and pre-coat filters.

The physical and chemical characteristics of waste cartridge filters are primarily those of the filter elements since their volume is large compared to the crud and dirt they contain. Filter elements containing natural fibers are subject to decomposition and oxidation which are induced by chemical attack, bacterial action, and radiation damage. A density of 0.6 g/cm^3 is taken as being representative of unpackaged filter cartridges.(5) Currently, cartridge filters are usually packed into 55-gallon drums (between 3 to 12 per drum) prior to transfer to a disposal site.(5).

2.2.6 Trash

Trash is the most varied waste stream generated by LWR's and can contain everything from paper towels to irradiated reactor internals. Some of the materials which have been identified as being shipped as trash are listed in Table 2-7.

A recent survey(5) found that compactible and non-compactible items are frequently shipped in the same container and that packaging small pieces of activated metal with relatively innocuous materials is common. Another plant was reported to cut up its non-compactible waste and ship it with compactible waste. Such factors make characterization of trash difficult.

The pysical and chemical characteristics of LWR trash can be discussed only in qualitative terms. In general, compactible trash contains more combustible material (e.g., paper, plastic), and non-compactible trash contains more metallic components (e.g., pipes) and failed equipment. It is usually assumed that the volume percentages of compactible trash and combustible trash are the same. Similarly, the volume percentages of non-compactible trash and non-combustible trash are assumed to be the same. Trash containing cellulose is subject to chemical attack by acids and oxidizers, and to degredation by bacterial action. The density of as-generated trash cannot be accurately estimated due to its highly variable composition and because trash is often compacted before shipment.

2.2.7 Other LWR Waste Streams

Other LWR waste streams considered in this report are waste nonfuel reactor components and waste from routine decontamination of reactors during their operating life.

TABLE 2-7 . Material Shipped as LWR Trash

Material	BWR's		PWR's	
	C*	N*	C	N
Anti-contaminant clothing	X		X	
Cloth (rags, mops, gloves)	X		X	
Conduit				X
Contaminated dirt	X	X		
Contaminated tools and equipment				
Hand tools	X	X		
Eddy current equipment				X
Vessel inspection equipment				X
Ladders		X		X
Lighting fixtures				X
Spent fuel racks				X
Scaffolding		X	X	
Laboratory equipment			X	X
Filters				
Filter cartridges	X	X		
HEPA filters		X	X	X
Respirator cartridges	X			
Glass	X	X	X	
Irradiated Metals				X
Flux wires				X
Flow channels		X		
Fuel channels		X		X
In-core instrumentation				X
Poison channels		X		
Shim rods				X
High density concrete block				X
Miscellaneous metal	X	X		X
Aerosol cans		X		
Buckets	X			
Crushed 55 gal drums		X		
Fittings		X		
Pipes and valves		X		X
Miscellaneous wood	X	X	X	X
Paper	X		X	
Plastic				
Bags, gloves, shoe covers	X	X	X	
Sample bottles	X			
Rubber	X			X
Sweeping compounds		X		

---

* C : compactible, combustible; N : noncompactible, noncombustible.

Source: Reference 5.

Nonfuel reactor components consist of fuel channels, control rods, control rod channels, shim rods, in-core instrumentation, and flux wires. These components are usually manufactured with corrosion-resistant alloys which may contain boron, cadmium, or hafnium as the neutron absorber. Many of these components are exposed to the primary reactor coolant and all are exposed to the in-core neutron flux. The physical and chemical characteristics of this waste stream is expected to resemble that of activated stainless steel and boron steel.

LWR decontamination waste is expected to be produced in the future from routine full-scale decontamination of LWR primary coolant systems. The components included in these systems include the reactor core, the reactor pressure vessel, coolant system piping, various pumps, and turbines. The purpose of decontamination is to reduce in-plant occupational radiation exposures by removing crud accumulated on surfaces which are in contact with the primary coolant. It is assumed (see Appendix A) that the principal waste stream generated during these routine decontamination operations will be ion-exchange resins used to process the decontamination solutions. Evaporator bottoms may also be produced during these activities; however, the characterization of evaporator bottoms for this report appears to be too speculative at this time.

The physical characteristics of LWR decontamination resin waste are expected to be similar to the currently used ion exchange resins (see Section 2.2.2). However, they are likely to contain higher concentrations of multivalent cations of iron, nickel, chromium, manganese, cobalt, copper, zinc, and other transition elements found in reactor grade steel used in reactor components and in fuel crud. The waste resins are also projected to contain large quantities of chelating agents.(9)

2.3 Other Nuclear Fuel Cycle Facilities

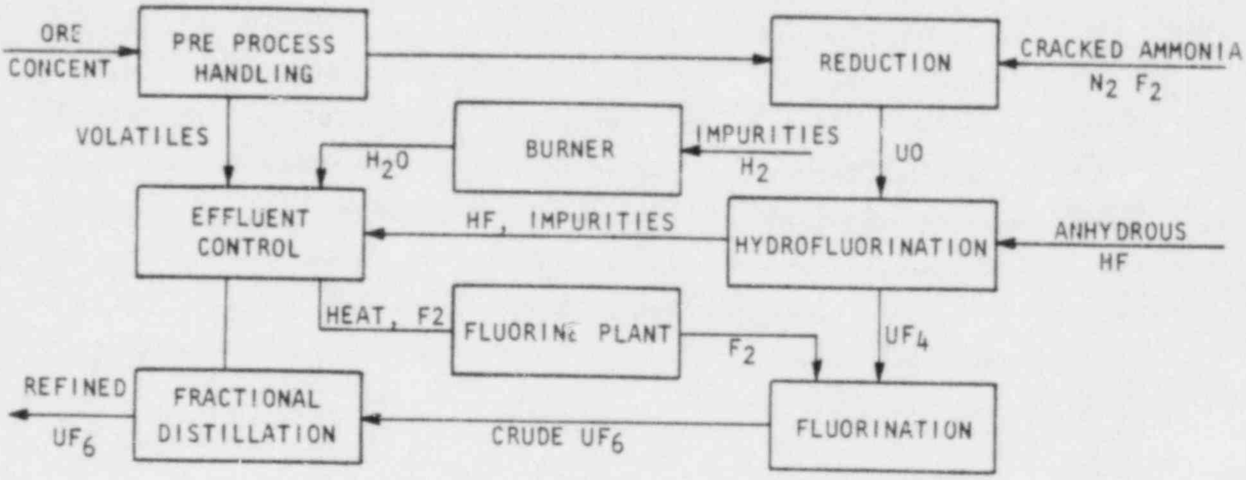
Other nuclear fuel cycle waste streams considered in this appendix include process wastes from uranium hexafluoride (UF_6) conversion plants and fuel fabrication facilities, and trash from fuel fabrication facilities. These wastes are generally not well characterized. Process wastes are dewatered before shipment for disposal but rarely solidified. No data could be found for trash from UF_6 conversion facilities.

2.3.1 Uranium Hexafluoride Conversion Wastes

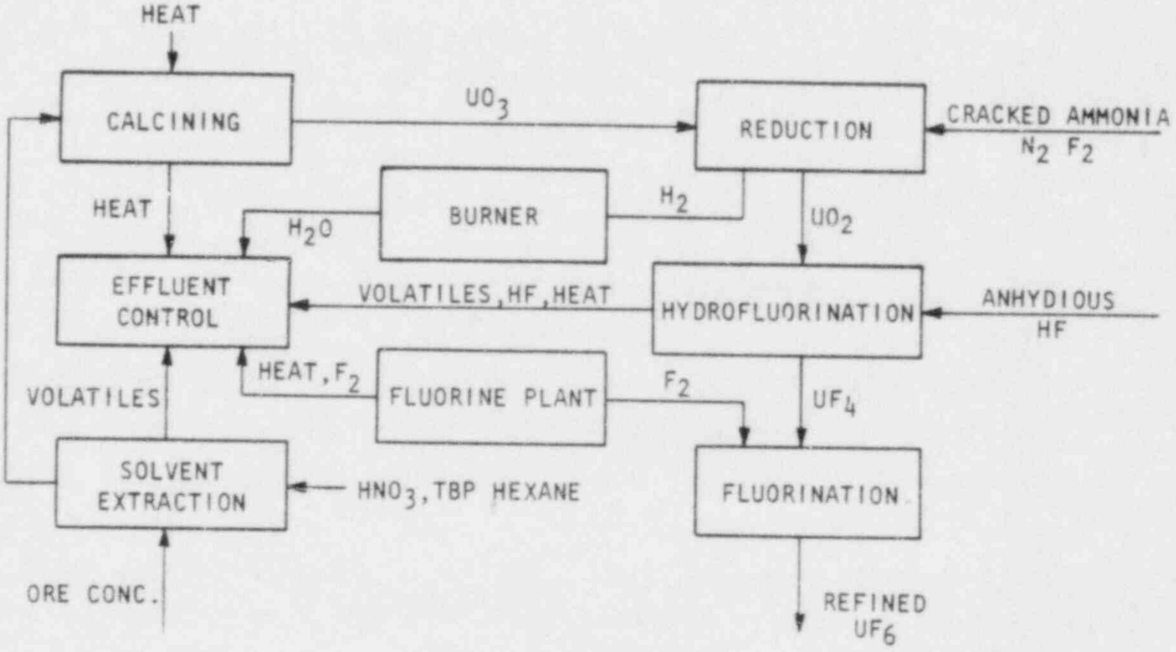
Processed uranium ore or yellowcake containing about 0.71 percent fissile U-235, must be enriched in U-235 prior to utilization as fuel in LWR's. The gaseous diffusion process (the major technology currently used for enrichment) requires that the uranium be converted to UF_6 which is an easily-volatized compound suitable for this process.

There are two commercially operated uranium hexafluoride (UF_6) conversion facilities in the United States. One facility uses the solvent extraction-fluorination process (wet process) and the other uses the fluorination-fractionation process (dry process). These processes are illustrated in Figure 2.7.(10)

The flourination-fractionation process produces UF_6 from yellowcake (U_3O_8) by successive reduction, hydrofluorination, and fluorination steps carried out in fluidized bed reactors. The crude UF_6 is subsequently purified by fractional distillation. The solvent extraction-fluorination process uses the same steps; however, purification of crude UF_6 by fractionation at the end of the process is replaced by purification of U_3O_8 prior to the reduction step. The fluorination-fractionation process produces more solid waste and the solvent extraction-fluorination process more liquid waste.(11)



(A) DRY PROCESS



(B) WET PROCESS

UF_6 CONVERSION PROCESSES

DAMES & MOORE

FIGURE 2-7

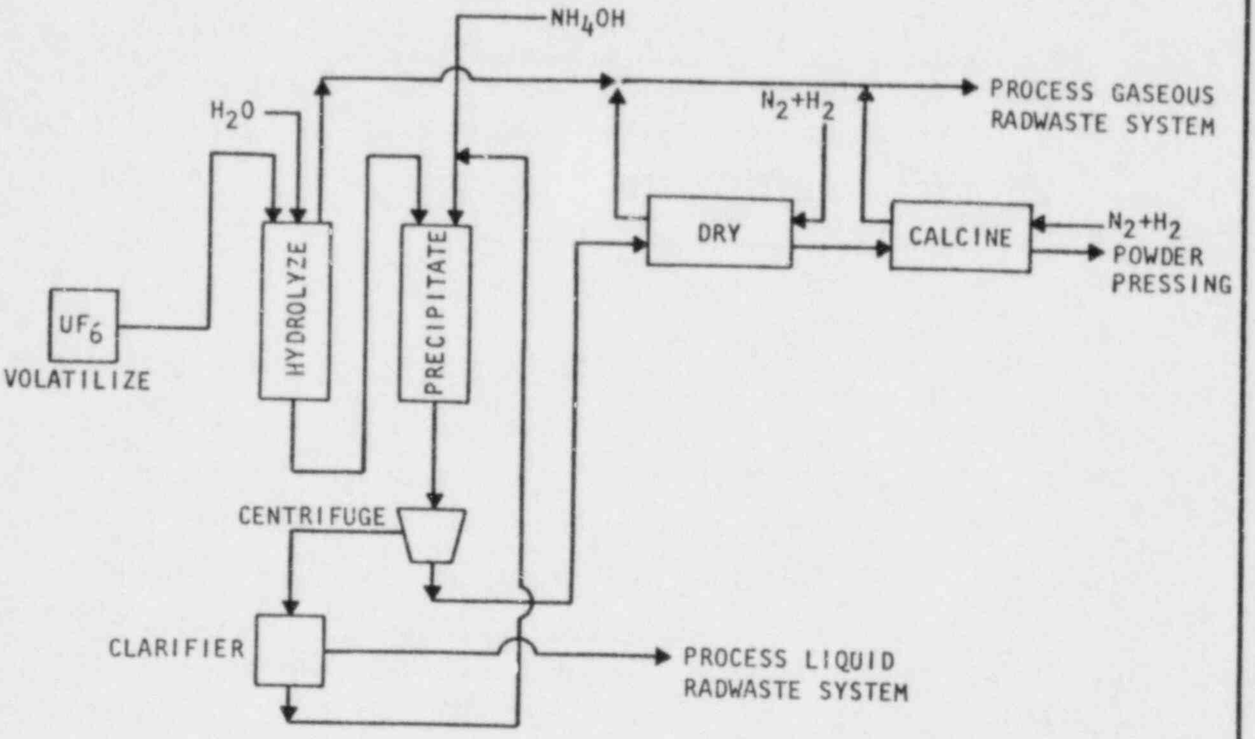
Many of the waste streams generated during the conversion process are recycled in the plant to recover uranium. Some process wastes, however, are shipped for disposal. These wastes consist primarily of calcium fluoride generated in hydrogen fluoride gas scrubbers, bed materials from fluidized bed reactors, and lime from treatment of liquid effluents.

2.3.2 Fuel Fabrication Wastes

Fuel fabrication is the final step before uranium fuel is utilized in LWR's. Currently operating fuel fabrication facilities are presented in Table 2-2. In fuel fabrication facilities, enriched UF_6 from gaseous diffusion plants is converted into a solid form (usually uranium dioxide) and then into fuel pellets, fuel rods, and finally fuel assemblies. A large portion of the wastes generated during this production are recycled to recover uranium.(12)

Fuel fabrication facilities use either an ammonium diuranate (ADU) process or a dry direct conversion (DDC) process to convert UF_6 to UO_2. The ADU process, which is illustrated in Figure 2.8, hydrolyzes UF_6 with water followed by neutralization with ammonium hydroxide to produce a slurry of ADU. The material is recovered by centrifugation or filtration and calcined to form UO_2. The DDC process, which is illustrated in Figure 2.9, routes vaporized UF_6 through a series of UO_2F_2 beds fluidized by either steam alone or by steam and cracked ammonia. The product UO_2 accumulates in the final bed and is removed for fabrication.

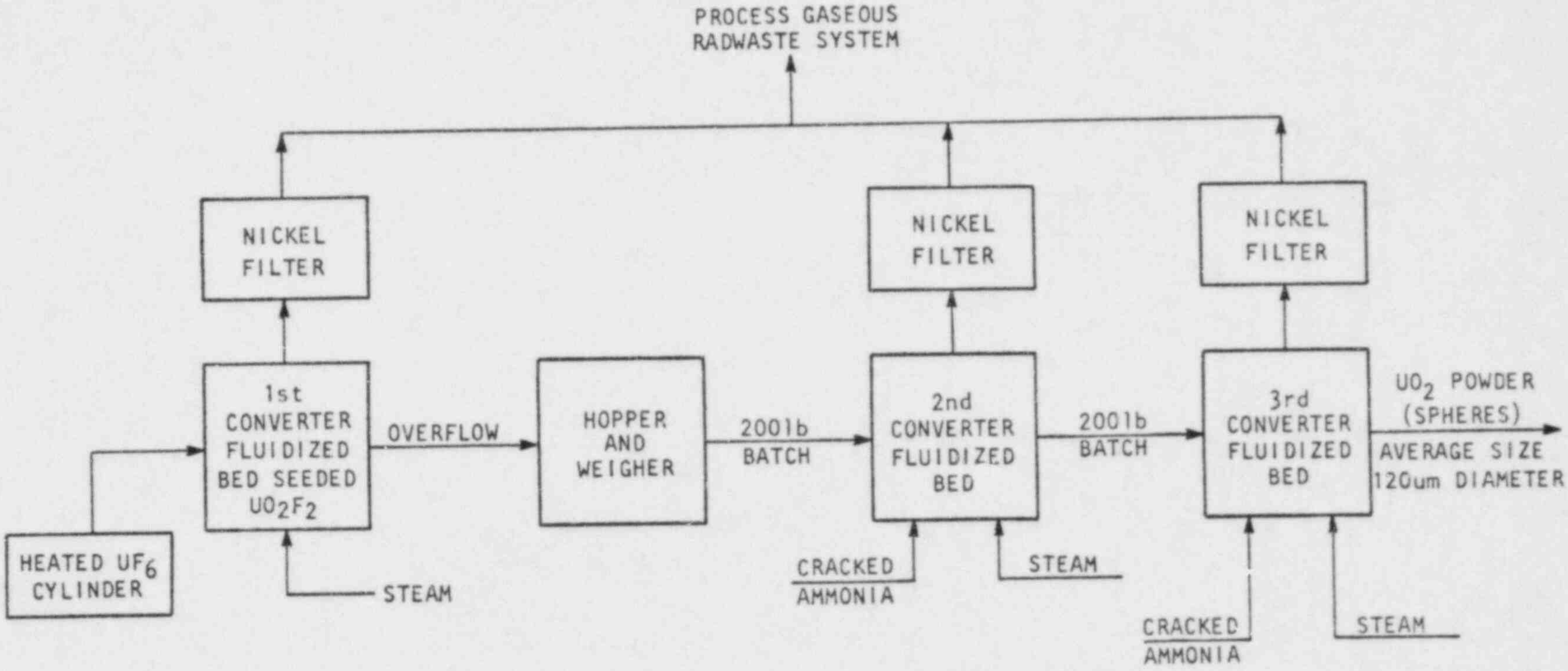
Process wastes shipped for disposal include limestone used in calcium fluoride scrubbers, oxides from calciners, filter sludges, and small amounts of oils. Trash shipped for disposal includes paper, plastic, equipment, and miscellaneous combustible materials. These wastes generally contain only U-235 and U-238 as their radioactive components.



AMMONIUM DIURANATE (ADU) PROCESS

DAMES & MOORE

FIGURE 2-8



DIRECT DRY CONVERSION (DDC) PROCESS SYSTEM

FIGURE 2-9

DAMES & MOORE

2.4 Institutional Facilities

Institutional waste generators include colleges and universities, medical schools, research facilities (e.g., the National Institute of Health), and hospitals. These institutions use radioactive materials in many diverse applications. Sealed sources and foils are widely used as an integral part of analytical instruments and irradiators. Labelled pharmaceuticals and biochemicals are used in nuclear medicine for therapy and diagnosis, and in biological research to study the physiology of humans, animals, and plants. Radioactive materials are also used by many other academic disciplines such as chemistry, physics, and engineering. Radioactive waste streams are also produced by institutions as a by-product of research using neutron activation analysis, particle accelerators, and research reactors.

Based upon information received from surveys(13-14), institutional wastes may be classified into four volumetrically significant groups: liquid scintillation vials containing scintillation fluid (shipped with absorbent materials), other liquids (solidified or shipped with absorbent materials), biological wastes (shipped with absorbent materials and lime), and trash. In addition to these streams, institutional facilities generate two volumetrically smaller waste streams, accelerator targets and sealed sources, that have been included under the next section on industrial wastes.

2.4.1 Liquid Scintillation Wastes

Liquid scintillation counting techniques are used to some extent by nearly all fuel cycle and non-fuel cycle waste generators; however, applications in biological research produce the only significant volume of waste scintillation vials and fluids. The vials are made of glass and occasionally polyethylene, and are usually about half full of counting fluid. Table 2-8 lists the common constituents of these fluids. These vials are normally packed in twice their volume of an

TABLE 2-8

Common Constituents of Liquid Scintillation Fluids

Usage	Common Name	Description
Solvents		Toluene
		Xylene
		1,4-Dioxane
Emulsifiers	Triton-X-100	Mixture of polyethoxy alkylphenols
		Methanol
		Ethanol
Solubizers	Hyamine hydroxide	Benzyldimethyl (2-(2-((4-(1,1,3,3-tetramethylbutyl) -m- tolyl) oxy) ethoxy) ethyl)-ammonium hydroxide
	Protosol NCS Soluene Bio Solv	Other high molecular weight quarternary ammonium bases
Primary Scintillators		Naphthalene
	PPO	2,5-Diphenyl-oxazole
	PBD	2-phenyl-5-(4- biphenylyl)-1,3,4 - Oxadizole
	Butyl-PBD	2-(4-t-Butylphenyl)-5(4-biphenylyl)-1,3,4-Oxadiazole
Secondary Scintillators	POPOP	1,4-Bis-2-(5-phenyloxazolyl)-benzene
	DMPOPOP	1,4,Bis-2-(4-methyl-5-phenyloxazolyl)-benzene
	Bis-MSB	p-Bis-(o-methylstyryl)-benzene
	PBBO	2-(4-Biphenylyl)-6-phenylbenzoxazole

absorbant such as diatomaceous earth or vermiculite prior to shipment for disposal.(13,14)

Liquid scintillation counters are normally used to detect beta emitting radionuclides and less frequently to detect alpha-emitting radionuclides. This is accomplished by converting the kinetic energy of the emitted particles into flashes of light which can be detected by photomultiplier tubes. Chemicals which perform this conversion (scintillators) are dissolved in a solvent which also contains the radionuclide to be measured. The wave length of the emitted light is usually in the blue or near ultraviolet regions.

Flammable organic solvents comprise the major constituents of scintillation fluids. The most common solvent is toluene although xylene and 1,4-dioxane are also used.(14-15) These compounds are toxic and 1,4-dioxane is a known carcinogen.(16) The toxicity of these and other components of LLW are discussed in Section 2.6. Introduction of aqueous samples into toluene and xylene requires the use of chemicals such as alcohols to increase the miscibility. High molecular weight quarternary ammonium bases are often used to dissolve tissues for counting. These bases are also used to absorb CO_2 produced by oxidation of tissues and other organic samples labelled with C-14. Scintillation fluids may also contain one or more primary and secondary scintillators. Typical concentrations of primary scintillators or fluors range from 4 to 9 grams per liter with secondary scintillators concentrations of approximately 1 gram per liter.(17)

2.4.2 Absorbed Liquids

Absorbed liquids have not been as well characterized as liquid scintillation vials, in part because the composition of absorbed liquids is not constrained by the requirements of liquid scintillation counting techniques. Approximately 50 percent of these absorbed liquids are scintillation fluids;(13) the remaining liquids are aqueous and organic liquids generated by diverse preparatory and analytical procedures such as wastes from elution of Tc-99m generators, radioimmunoassay procedures, and tracer studies.

Typical components of the scintillation fluids shipped as absorbed liquids are given in Table 2-8. The remaining liquids are a mixture of aqueous and organic liquids. The relative volumes of aqueous and organic liquids are not known. However, the available data indicates that about 79 percent of surveyed institutional facilities ship aqueous liquids for disposal, and 47 percent ship organic liquids other than scintillation fluids.(14)

2.4.3 Biological Wastes

Biological wastes are generated primarily through research programs at universities and at medical schools. The waste consists of animal carcasses, tissues, animal bedding, and excreta, as well as vegetation and culture media. Radioactive excreta from humans undergoing diagnostic or therapeutic procedures which use radioactive materials are not included since virtually all such materials are discharged to sewers.(13)

Volumetrically, the most significant component of biowaste is animal carcasses.(18) Mice, rats, rabbits, cats, dogs and similar culture media are commonly used for experimental purposes. The carcasses, tissues, and excreta have some pathologenic potential and may possibly contain carcinogenic compounds labelled with suitable radionuclides. However, the radionuclide concentrations of such compounds are expected to be extremely low due to the very high sensitivity of radiation detecting instruments. Carcasses are normally shipped for disposal packed with absorbent material and lime inside a 30 gallon drum which is placed inside a 55 gallon drum. The space between the drums is filled with an absorbent material. This procedure roughly doubles the as-generated waste volume.(18)

2.4.4 Trash

Institutional trash consists almost entirely of materials which are both compactible and combustible.(14) It generally consists of paper, rubber and plastic gloves, disposable and broken labware, and disposable syringes. The physical properties of institutional trash are estimated to resemble those of paper and plastic.

2.5 Industrial Facilities

Wastes from industrial facilities may be grouped into five streams which are relatively small in volume but high in activity: medical isotope production wastes, highly activated wastes, tritium manufacturing wastes, sealed sources, and accelerator targets.

In addition, there are two groups of industrial facilities that generate four volumetrically significant waste streams containing relatively low levels of radioactivity: (1) facilities using source and special nuclear materials, and (2) facilities that use radioactive material and generate low specific activity wastes containing less than 3.5 Ci/m^3 (0.1 Ci/ft^3). Wastes from each of these groups of facilities may be broken down into trash and other miscellaneous wastes.

2.5.1 Medical Isotope Production Wastes

Medical isotope production wastes result from production of fission isotopes for medical use through irradiation of very highly enriched uranium. Although some institutions using large quantities of radioactive materials in research and medical applications produce their own radioactive isotopes, most of these radionuclides are produced by the industrial isotope generators. The wastes generated consist of paper, plastic, glass, metal, and aqueous solutions of inorganic salts. The aqueous solutions are commonly solidified in small metal containers and packed with low-specific-activity trash in a common container (55-gallon drums) for shipment. This practice precludes an accurate estimate of the volume of trash relative to the volume of solidified aqueous solutions.

2.5.2 High-Specific-Activity Wastes

The high-specific-activity industrial waste stream is a generic stream which includes miscellaneous wastes of relatively high activity, which is arbitrarily defined as an activity which exceeds 3.5 Ci/m^3 or 0.1 Ci/ft^3. The high-specific-activity industrial wastes are expected to include activated metal and equipment produced by accelerators, activated metal and equipment from research reactors and sub-critical assemblies, and activated metal from neutron generators. The properties of these wastes are expected to resemble those of the LWR non-fuel reactor components waste stream.

2.5.3 Tritium Manufacturing Wastes

Tritium is the most widely used of all radioisotopes. In addition to applications in biological research and medicine, it is used in a wide variety of products, most commonly in illuminators. Although tritium is a naturally-occurring isotope, artificial production of tritium is more economical than enrichment of natural tritium. The waste generated during the production of tritium and in the wide range of manufacturing processes which use tritium are considered in this waste stream. The waste generated during tritium production is assumed to consist of lithium fluoride, trash, plastic, and a small quantity of metal. A larger quantity of waste is assumed to contain waste chemicals which are generated by conversion of tritium gas to tritiated water and by incorporation of tritium into chemical compounds. Although these chemicals are not well-characterized, small quantities of a large number of physiologically active and/or toxic compounds are expected to be present.

2.5.4 Sealed Sources

Sealed sources and foils contain radioactive materials which are encapsulated to prevent leakage of the radioactive material. Low-activity sealed sources are used as calibration and reference standards for many types of radiation detectors. They are also used in some gas chromatographs. High-activity sealed sources are used in neutron generators as both generators and targets, and in medical and industrial irradiators. This waste stream includes industrial sealed sources as well as sealed sources from institutions.

2.5.5 Accelerator Targets

Accelerator targets are used to produce radionuclides by direct bombardment with charged particle beams or by indirect reactions of the target fragments with other materials. Accelerator targets are also used to study nuclear reactions and to produce and study the properties of various subatomic particles. Targets from institutional sources are included in this waste stream. Spent targets are most commonly made of titanium foils which contain absorbed tritium.

2.5.6 Source and Special Nuclear Material Wastes

Source and special nuclear material wastes are produced outside the nuclear fuel cycle by industries which process and fabricate depleted uranium and manufacture chemicals containing uranium. Although little information is available, it appears that most of the waste is generated through processing of depleted uranium. These wastes are distinguished from other non-fuel cycle wastes by the almost complete absence of radionuclides other than those included in the definitions of source and special nuclear materials. They are considered as two streams: trash and other miscellaneous wastes. The constituents of wastes received at the Maxey Flats disposal facility during 1977 which contain predominately source and special nuclear materials are given in Table 2-9.(19)

TABLE 2-9

Estimated Constituents of Wastes Containing Source and Special Nuclear Materials

Trash	Other Wastes
Saw Blades	Slag
Brick	Uranium Oxides
Floor Sweepings	Lime Sludge
Graphite	Plutonium Oxides
Limestone	
Mantle Scrap and Trimmings	
Oil	
Filter	
General Combustibles	
Plastic	

2.5.7 Low Specific Activity Wastes

The last group of waste streams are low specific activity wastes containing less than 3.5 Ci/m^3 (0.1 Ci/ft^3). The major contributors to this group of streams are the industrial equivalents of institutions. Such waste is generated by pharmaceutical companies, independent testing laboratories, and analytical laboratories. The characteristics of low specific activity industrial wastes are expected to resemble those of institutional wastes; however, since the limited data available is insufficient to justify separate waste streams for scintillation fluids, adsorbed liquids, and biowastes, they are also considered as two streams: trash and other miscellaneous wastes.

2.6 Toxicity

Most of the untreated waste streams generated by fuel cycle and non-fuel cycle facilities contain toxic chemicals. In most of these cases, the toxic compounds are present in low concentrations and are confined to a few waste packages representing a small fraction of the total waste volume. Other wastes streams, listed in Table 2-10, contain larger quantities of toxic materials. In addition, decomposition of organics in these and other waste streams (i.e., trash) can produce additional toxic compounds.(20)

A recently completed study(21) discusses the chemical toxicity of low-level wastes in depth. On the basis of this study and the data obtained from analysis of trench leachate,(20) it is concluded that low-level radioactive wastes do not represent significant toxicological risk and that no acute or chronic adverse effects are expected to result from current waste processing and disposal practices.

For illustration purposes, the following example may be considered. Toluene and xylene, which are representative of the most toxic compounds present in significant quantities in scintillation fluids, have

TABLE 2-10

Waste Streams Containing Significant Quantities of Toxic Chemicals

Waste Stream	Representative Toxic Components
L-DECONRS	Chelated metal ions and free chelating agents (0.5-1.0 kg/m^3)
I-LIQSCVL	Toluene, benzene, xylene, 1,4-dioxane containing 5-10 g/l of potentially toxic primary and secondary scintillators(17)
I-ABSLIQD	Aqueous solutions of potential toxic salts and chelates, a variety of toxic organic solvents and compounds
I-BIOWAST	Pathological hazards; traces of toxic radio-labelled compounds
N-LOWASTE	Contains materials listed above for I-LIQSCVL, ABSLIQD, and BIOWAST
N-ISOPROD	Suspected presence of chelates and organic solvents
N-TRITIUM	Organic solvents and compounds including poisons and pharmaceuticals

human oral LDL_0 values of about 50 mg/kg. An LDL_0 is defined as the lowest dose, measured in weight of compound per unit body weight, observed to be lethal. The highest concentrations detected in leachate from the Maxey Flats, Kentucky and West Valley, New York disposal facilities (which have unfavorable conditions for leaching and result in higher concentrations of toxic chemicals in the leachate) are 18 mg/l for toluene and 0.5 mg/l for xylene. A typical 70 kg man would have to drink about 200 liters of leachate at one time directly from the burial trench to reach the LDL_0 for toluene and about 7000 liters to reach the LDL_0 for xylene.

The waste processing options discussed in Chapter 4.0 are expected to further reduce the toxicological risks associated with low-level waste by: (1) destroying toxic materials by incineration, (2) reducing the mobility of toxic compounds by use of improved solidification methods, and (3) reducing contact of water with buried waste forms by improving waste form stability through use of incineration and improved solidification processes.

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3.0 WASTE CHARACTERISTICS

This chapter presents information on the volumes and radiological characteristics of the waste streams projected to be generated to the year 2000. The waste streams considered are those discussed in the previous chapter. Information on the packaging characteristics of these waste streams can be found in Chapter 4.0.

The following symbols will be used for the major waste generators for the remaining discussion in this report:

Symbol	Facility
P	PWR's
B	BWR's
L	LWR's
F	Fuel Fabrication Facilities
U	UF_6 Conversion Plants
I	Institutions
N	Industry

The waste streams outlined in the previous chapter will be discussed in four major groups: LWR process wastes, trash, low specific activity wastes, and special wastes. These groups and the waste streams that make up each group are presented in Table 3-1.

These streams are combined into these four groups based upon similarities in their macroscopic characteristics. For example, LWR process wastes are usually wet wastes that have comparatively higher specific activities than either the trash group or the low specific activity group. The trash group is self-evident and contains most of the combustible LLW generated. The low specific activity waste group includes all the streams containing comparatively small activities and which are not included in the LWR process waste group or the trash

TABLE 3-1 . Waste Groups and Streams

Waste Stream	Symbol
Group I : LWR Process Wastes	
PWR Ion Exchange Resins	P-IXRESIN
PWR Concentrated Liquids	P-CONCLIQ
PWR Filter Sludges	P-FSLUDGE
PWR Filter Cartridges	P-FCARTRG
BWR Ion Exchange Resins	B-IXRESIN
BWR Concentrated Liquids	B-CONCLIQ
BWR Filter Sludges	B-FSLUDGE
Group II : Trash	
PWR Compactible Trash	P-COTRASH
PWR Noncompactible Trash	P-NCTRASH
BWR Compactible Trash	B-COTRASH
BWR Noncompactible Trash	B-NCTRASH
Fuel Fabrication Compactible Trash	F-COTRASH
Fuel Fabrication Noncompactible Trash	F-NCTRASH
Institutional Trash (large facilities)	I-COTRASH
Institutional Trash (small facilities)	I+COTRASH
Industrial SS Trash (large facilities)*	N-SSTRASH
Industrial SS Trash (small facilities)	N+SSTRASH
Industrial Low Trash (large facilities)	N-LOTRASH
Industrial Low Trash (small facilities)	N+LOTRASH
Group III : Low Specific Activity Wastes	
Fuel Fabrication Process Wastes	F-PROCESS
UF_6 Process Wastes	U-PROCESS
Institutional LSV Waste (large facilities)*	I-LIQSCVL
Institutional LSV Waste (small facilities)	I+LIQSCVL
Institutional Liquid Waste (large facilities)	I-ABSLIQD
Institutional Liquid Waste (small facilities)	I+ABSLIQD
Institutional Biowaste (large facilities)	I-BIOWAST
Institutional Biowaste (small facilities)	I+BIOWAST
Industrial SS Waste	N-SSWASTE
Industrial Low Activity Waste	N-LOWASTE
Group IV : Special Wastes	
LWR Nonfuel Reactor Components	L-NFRCOMP
LWR Decontamination Resins	L-DECONRS
Waste from Isotope Production Facilities	N-ISOPROD
Tritium Manufacturing Waste	N-TRITIUM
Accelerator Targets	N-TARGETS
Sealed Sources	N-SOURCES
High Activity Waste	N-HIGHACT

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* SS : Source and Special Nuclear Material, LSV : Liquid Scintillation Vials.

group. The "special" waste group contains streams that contain relatively high concentrations of radioactivity and are small in volume when compared with the other three groups.

This grouping of waste streams simplifies the application of generic waste treatment technologies and disposal procedures to general groups, thereby increasing the flexibility of the data base.

As shown in Table 3-1, six of the waste streams have been separated into two components and the additional six streams resulting from this separation have been denoted by a plus sign after the waste generator symbol (I or N) instead of the usual minus sign. These streams are industrial SSTRASH, industrial LOTRASH, institutional COTRASH, institutional LIQSCLV, institutional ABSLIQD, and institutional BIOWAST. The reason for this separation is to identify the volumes of waste from generators that can more easily implement their own waste treatment processes (e.g., comparatively large facilities, denoted by a minus sign), and the waste from those generators that cannot do the same (e.g., comparatively small facilities, denoted by a plus sign).

The waste streams that are not considered in detail in this report (e.g., decommissioning and reprocessing wastes) can be classified as a fifth group of wastes. These streams are briefly discussed in Chapter 6.0.

3.1 Volume Projections

This section discusses estimates of waste volumes expected to be routinely generated on a regional basis and disposed through the year 2000. The waste volumes and total activities presented in sections 3.1.1 and 3.1.2 are those estimated by NRC staff as described in Appendix A.

These estimates were developed considering current waste generation rates as well as projected waste generation growth rates. The regions used in the projections correspond to the five NRC regions as shown in Figure 1.1. In developing these projections, nuclear fuel cycle waste volumes were assumed to be proportional to the nuclear electrical generation capacity. Non-fuel cycle waste volumes were assumed to grow at a linear rate based upon a least-squares fit of existing data on individual waste streams.

The volumes estimated by NRC staff are frequently based on waste volumes as-shipped and therefore may not be directly applicable to estimate as-generated volumes. Section 3.1.3 discusses modification of NRC volume estimates to obtain as-generated waste volumes used in this report to evaluate the effects of the waste processing options described in Chapter 4.0. Table 3-2 summarizes the information sources of the waste volumes and activities incorporated in the data base. Estimation of these activities is described in Section 3.2. NRC estimated regional distributions and waste generation growth rates are used for all waste streams.

3.1.1 Nuclear Fuel Cycle Wastes

Projections of nuclear electrical generation capacity were principally based upon a review of information on nuclear power stations currently built and operable, under construction, or planned or on order.(1-4) Projections made by NRC licensing staff regarding start-up times were also used to supplement the basic information.(5)

Based upon this data, two scenarios were developed for central station nuclear power plant construction - a "low" scenario and a "high" scenario. The low scenario assumes that construction continues on power reactors which are already under construction but that any additional construction of power reactors essentially ceases until at least the late 1980's. The high scenario assumes that construction

TABLE 3-2

Summary of Sources of Volumes and Activities Incorporated in Waste Source Options Data Base

Waste Type	Source of Data	
	Volumes	Activities
Light Water Reactors		
Process Wastes	NRC	D&M
Trash	NRC	NRC[a]
Other Wastes	NRC	NRC[a]
Fuel Fabrication		
Process	NRC[b]	D&M
Trash	NRC[b]	D&M
UF_6 Conversion		
Process	D&M[d]	D&M[d]
Institutional Wastes	NRC[c]	D&M[d]
Industrial Wastes	NRC	NRC[a]

(a) Dames & Moore (D&M) developed scaling factors applied to NRC estimated total activities to calculate radionuclide concentrations.

(b) Total fuel fabrication waste volume estimated by NRC and distributed between waste types by D&M.

(c) As-generated volumes estimated by D&M from NRC as-shipped volume estimates.

(d) Estimated by D&M.

commences on a number of additional plants, including those units planned as of the beginning of 1980, as well as plants for which construction has been deferred indefinitely. The projected regional capacity by the year 2000 for both scenarios is presented in Table 3-3. Also shown, in parantheses, are the number of plants projected to be operating. As shown, the total U.S. capacity by the year 2000 is projected to range between 146,000 and 169,000 MW(e). The high-range scenario is used in this report to determine waste volumes.

It is believed that the projections in Table 3-3 effectively provide a lower and upper bound of the generating capacity which would be available by the year 2000. As of June 30, 1979, 27 units were listed as "planned", representing a capacity of 32,726 MW(e).(1) Of these 27 units, 19 had definite projected start-up dates. Only one year later, 11 of these original 27 units had been canceled (13,202 MW(e)). Out of the remaining 16 units, three had been deferred indefinitely, only five (with a total capacity of 5,910 MW(e)) are listed as having definite start-up dates.(4) Of these five units, applications for construction have been submitted to NRC for only three of them (Allens Creek Unit 1, Pebble Springs Unit 1 and 2), and no construction permits for these three units have to date been issued. It would not be surprising, therefore, if no more than half of the planned units discussed above were actually constructed by the year 2000. The slowdown in construction of and planning for new nuclear generating facilities is probably due to a number of reasons -- e.g., a lessening in the demand for additional electrical generating capacity, the slowdown in the economy coupled with the large costs of constructing a nuclear power station, and public concern over the safety of nuclear power. It is possible that interest in building new nuclear generating units may increase in the future. However, it takes a number of years to construct and license a nuclear power station. Assuming that it requires a conservative minimum of 12 years from the time of initial application to start-up of a single unit, an application would have to be tendered by no later than 1988 in order to be operating by the year 2000.

TABLE 3-3

Projected LWR Capacity by the Year 2000, in MW(e)[a]

	Low Scenairo		High Scenario	
Region	PWR	BWR	PWR	BWR
1	17,691 (20)	12,216 (14)	22,411 (24)	14,516 (16)
2	38,958 (39)	17,239 (16)	44,058 (43)	18,173 (17)
3	18,785 (21)	13,550 (18)	22,295 (24)	13,550 (18)
4	8,901 (8)	3,078 (3)	8,091 (8)	4,228 (4)
5	15,580 (14)	1,165 (2)	18,100 (17)	3,719 (4)
	97,805 (102)	47,248 (53)	114,955 (116)	54,186 (59)
	146,333 (155)		169,141 (175)	

(a) Since the original projections were made, construction of a 907 MW(e) PWR (North Anna Unit 4 in Region II) has been definitely cancelled. Start up of another facility -- Allens Creek, a 1150 Mw(e) BWR located in Region IV -- has been delayed.

Therefore, only those planned units for which an application has already been received by NRC or received within the next few years could realistically contribute to the waste generated by LWR's by the year 2000. Finally, any delays in the start-up times for units currently under construction would act to further reduce the amount of waste produced by LWR's by the year 2000.

A summary of volumes and gross specific activities of LWR waste streams projected to be generated on a "per MW(e)-yr" basis is presented as Table 3-4. The data used to construct this table were principally obtained from ONWI-20,(6) and are averages based upon NRC staff estimates of the use of condensate polishing systems (CPS) as part of water treatment in LWR's. For the tables, 60% of BWR's are assumed to use deep bed CPS and 40% pre-coat CPS, 51% of PWR's were assumed to use CPS and 49% were not. The volumes shown, with the exception of cartridge filters, are for untreated wastes. Concentrated liquids (evaporator bottoms) are reported as-generated prior to solidification. Resins and filter sludges are reported as-dewatered, and the trash streams are reported as-generated prior to such processing options as incineration or compaction. The volumes for cartridge filters are given as-packaged for shipment.

Projected volumes of activated non-fuel LWR components (e.g., poison curtains, flow channels, and control rods) are difficult to characterize. LWR core components are replaced on an infrequent basis and frequently, small components are shipped to disposal facilities by placing the components in the middle of a container of otherwise low activity material such as trash. For this report, LWR's are projected to generate about 1 m^3 (35 ft^3) of core component waste per GW(e)-yr at a gross specific activity of 140,000 Ci/m^3 (4000 Ci/ft^3). This projection is based upon a review of disposal site radioactive shipment records.(7) NRC staff believe that these projections are likely to be conservative (see Appendix A), as the non-compactible trash stream discussed above probably already contains activated core

TABLE 3-4 . Summary of Principal LWR Waste Streams

	Volumes (m^3/MW(e)-yr)		Activity (Ci/MW(e)-yr)	
Waste Type	BWR	PWR	BWR	PWR
Resins	0.081	0.018	1.14	0.40
Concentrated liquids	0.223	0.124	0.20	0.11
Filter Sludge	0.179	0.002	1.40	0.006
Cartridge Filters	-	0.011	-	0.12
Trash				
Total	0.326	0.326	0.402	0.063
Compactible	0.221	0.215	0.005	0.005
Non-Compactible	0.105	0.111	0.397	0.058
Totals:	0.808	0.478	3.29	0.699

components (i.e., core components are to a certain extent counted twice in this report - see Section 2.2.6).

Another waste stream which is difficult to project will be generated by periodic decontamination of the primary coolant systems of LWR's. The purpose of such full-scale primary decontamination operations is to reduce plant personnel exposure by removing crud accumulated on surfaces in contact with the primary coolant. Although full-scale primary collant decontamination operations have not been routinely performed in LWR's in the past, such routine operations are expected in the near future. Some utilities are considering dilute chemical decontamination on an annual basis, and some utilities are considering concentrated chemical decontamination every few years. The types and characteristics of wastes generated from these activities (resins, sludges, solidified liquids, cartridge filters) are expected to vary considerably. Furthermore, considering additional factors such as the operating life of the plant, the history of fuel failures, the chemistry of the coolant he design of the plant, and the range of possible liquid clean-up and waste processing systems which affect the characteristics of the wastes expected to be generated from routine decontamination, it is difficult, if not impossible, to characterize these future wastes accurately. Nevertheless, an estimate of the characteristics of this potentially significant stream is indicated.

For this report, it is assumed that every operating LWR undergoes a full-scale primary coolant decontamination operation every 5 to 10 years using a dilute chemical decontamination process and that the decontamination solution is processed using ion-exchange resins.(8) This results in BWR and PWR resin waste streams of approximately 95 and 47.5 m^3, respectively, per operation. This is based upon an assumption that the volume of contaminated liquid generated per operation is 760 m^3 and 380 m^3, respectively, for a BWR and a PWR, and an assumption that approximately 0.125 m^3 of dewatered resin is required to process 1 m^3 of decontamination solution.(8) Contained in these resins will be significant quantities of chelating agents and other decontamination chemicals. A generation rate of 1666 m^3/yr is estimated on the basis of the high-scenario growth rate of LWR generated capacity and an assumed decontamination frequency of once every seven years.

The projected volume of fuel fabrication wastes was obtained from ONWI-20(6) and is estimated to be 122 m^3 per Gw(e) of installed LWR capacity. The estimated average activity of these wastes is 8.47×10^{-4} Ci/m^3. The volume of process waste was not estimated by NRC; however, disposal site records(7) indicate that the process waste volume amounts to about 15% of the total volume. Of the remaining volume, NRC estimates that 85% is combustible and 15% is noncombustible trash.

The volumes and activities of waste from uranium conversion (UF_6) facilities were estimated from information obtained from References 9 and 10. The resulting volume and activity are 9.64 m^3/GW(e) and 3.80×10^{-4} Ci/m^3, respectively.

3.1.2 Non-Fuel Cycle Wastes

Projections of total activities, volumes, and regional dependency through the year 2000 for non-fuel cycle wastes were developed from a number of sources. Included are medical and bioresearch wastes, wastes from production of medical isotopes, industrial high-activity wastes, industrial tritium wastes, and industrial low activity wastes. Starting with 1980 waste generation rates, non-fuel cycle wastes volumes and activities are assumed to increase at linear rates calculated by assuming a least-square fit to existing data.

Projections of institutional (i.e., academic, medical and bioresearch) wastes, including dry solids, scintillation vials, absorbed liquids, biological wastes (animal carcasses, tissues, etc.), and accelerator targets, were derived principally using NUREG/CR-0028(11) and its follow-up report NUREG/CR-1137.(12) Based upon this data, total volumes of medical and bioresearch wastes in 1980 are estimated to be 19,120 m^3, while the total activity is estimated to be 4412 Ci. Total volumes and activities are estimated to increase at a rate of 1280 m^3 and 295 Ci per year. Dry solids (trash) constitute 42% of the total volume, scintillation vials 39%, absorbed liquids 10%, biological wastes 9% and accelerator targets 0.2%. Fifty-six percent of the activity is projected to be contained in accelerator targets. The regional distribution of medical and bioresearch wastes are assumed to correspond to the institutional population surveyed(12) - i.e., region 1: 31%, region 2: 22%, region 3: 27%, region 4: 8%, and region 5: 12%.

A summary of estimated current and projected future volumes and activities in industrial wastes is provided as Table 3-5. Compared to institutional wastes (academic, medical, and bioresearch wastes) and fuel cycle wastes, less information is available for industrial waste streams. Consequently, industrial waste streams are difficult to characterize.

Estimates of medical isotope production waste are based upon consideration of disposal site radioactive shipment records.(13) Wastes from this source are generated in region 1.

Industrial tritium waste volumes were estimated from a number of sources.(7,13,14) For this report, about three-quarters of the tritium waste is assumed to be generated in region 1, the region with the major manufacturing of tritium products. The remainder is assumed to be divided equally among the other 4 regions.

Industrial high- and low-specific activity wastes are arbitrarily divided at a concentration level of about 0.1 Ci/ft^3 (3.5 Ci/m^3). Estimates of industrial high and low activity wastes are based upon

TABLE 3-5 . Estimated Current and Projected Future Volumes and Activities of Industrial Waste Streams

Waste Streams	Volumes (m^3) Current	 Added per year	Gross Specific Activity (Ci/m^3)
Medical isotope production waste:	192.6	13.8	573
Industrial tritium waste:	99.3	6.7	2326
Industrial high-activity waste (> 3.5 Ci/m^3):			
o Sealed sources	5.3	.36	5700
o Other high activity waste	74.4	5.0	210
Industrial low-activity waste (< 3.5 Ci/m^3):			
o Source and special nuclear material	12,050	807	0.03
o Other low activity waste	4,608	309	0.03

consideration of disposal site radioactive shipment records.(7,13) Sealed sources from institutional facilities are included in the estimates presented in Table 3-5 for the industrial sealed source waste stream. The regional distribution of these wastes is assumed to be the same as that of the institutional waste streams.

3.1.3 Volume Projections to the Year 2000

The total regional untreated waste volumes projected to the year 2000 are summarized in Table 3-6. In generating this table, regions IV and V were combined into one region. These volumes were calculated from the estimated 1980 volume by applying the regional waste distributions and generation growths rates given in Appendix A. The 1980 volumes listed in Table 3-7 were obtained from NRC and were estimated from the following assumptions:

- o The P-IXRESIN, B-IXRESIN, P-FSLUDGE, and B-FSLUDGE waste stream volumes are assumed to be "dewatered" volumes.
- o The P-CONCLIQ and B-CONCLIQ waste streams are assumed to be concentrated to the levels currently practiced in the industry. the solids content (by weight) of these streams range from 2% to 20% in PWR's and 7 to 50% for BWR's with an average of about 11% for PWR's and 25% for BWR's.(6)
- o The P-FCARTRG waste stream volume is assumed to be that of the packaged waste.
- o None of the LWR trash waste streams are assumed to be treated by compaction or by incineration.
- o The I-LIQSCVL, I+LIQSCVL, I-ABSLIQD, and I+ABSLIQD waste stream volumes represent volumes prior to packaging. Estimated shipping volumes include two volume parts absorbent material to one volume part waste.(12)

TABLE 3-6 . UNTREATED WASTE VOLUMES (M**3)

	REGION 1		REGION 2		REGION 3		REGION 4	
	VOL	%	VOL	%	VOL	%	VOL	%
P-IXRESIN	6.93E+03	.79	1.30E+04	1.34	6.59E+03	1.00	8.14E+03	1.25
P-CONCLIQ	4.87E+04	5.54	9.12E+04	9.45	4.63E+04	7.06	5.72E+04	8.79
P-FSLUDGE	8.56E+02	.10	1.60E+03	.17	8.14E+02	.12	1.01E+03	.15
P-FCARTRG	4.35E+03	.50	8.16E+03	.84	4.14E+03	.63	5.12E+03	.79
B-IXRESIN	2.10E+04	2.39	2.51E+04	2.60	2.05E+04	3.12	9.67E+03	1.49
B-CONCLIQ	5.79E+04	6.59	6.93E+04	7.17	5.64E+04	8.60	2.67E+04	4.10
B-FSLUDGE	4.65E+04	5.30	5.57E+04	5.77	4.54E+04	6.92	2.14E+04	3.30
P-COTRASH	8.49E+04	9.66	1.59E+05	16.47	8.07E+04	12.31	9.97E+04	15.33
P-NCTRASH	4.36E+04	4.96	8.16E+04	8.45	4.14E+04	6.32	5.12E+04	7.87
B-COTRASH	5.74E+04	6.54	6.87E+04	7.12	5.60E+04	8.54	2.65E+04	4.07
B-NCTRASH	2.72E+04	3.10	3.26E+04	3.38	2.66E+04	4.05	1.26E+04	1.93
F-COTRASH	4.72E+04	5.37	1.18E+05	12.22	0.	0.	7.08E+04	10.88
F-NCTRASH	8.34E+03	.95	2.09E+04	2.16	0.	0.	1.25E+04	1.92
I-COTRASH	4.36E+04	4.97	3.10E+04	3.21	3.80E+04	5.79	2.81E+04	4.33
I+COTRASH	4.36E+04	4.97	3.10E+04	3.21	3.80E+04	5.79	2.81E+04	4.33
N-SSTRASH	8.98E+04	10.22	1.80E+04	1.86	3.59E+04	5.48	3.59E+04	5.52
N+SSTRASH	8.98E+04	10.22	1.80E+04	1.86	3.59E+04	5.48	3.59E+04	5.52
N-LOTRASH	1.52E+04	1.73	1.01E+04	1.05	1.52E+04	2.32	1.01E+04	1.56
N+LOTRASH	1.52E+04	1.73	1.01E+04	1.05	1.52E+04	2.32	1.01E+04	1.56
F-PROCESS	1.56E+04	1.78	3.91E+04	4.05	0.	0.	2.34E+04	3.61
U-PROCESS	0.	0.	0.	0.	1.41E+04	2.14	1.41E+04	2.16
I-LQSCNVL	1.52E+04	1.73	1.08E+04	1.12	1.33E+04	2.02	9.83E+03	1.51
I+LQSCNVL	1.52E+04	1.73	1.08E+04	1.12	1.33E+04	2.02	9.83E+03	1.51
I-ABSLIQD	1.73E+03	.20	1.23E+03	.13	1.51E+03	.23	1.12E+03	.17
I+ABSLIQD	1.73E+03	.20	1.23E+03	.13	1.51E+03	.23	1.12E+03	.17
I-BIOWAST	4.87E+03	.55	3.46E+03	.36	4.24E+03	.65	3.14E+03	.48
I+BIOWAST	4.87E+03	.55	3.46E+03	.36	4.24E+03	.65	3.14E+03	.48
N-SSWASTE	3.17E+04	3.61	6.34E+03	.66	1.27E+04	1.93	1.27E+04	1.95
N-LOWASTE	1.81E+04	2.06	1.21E+04	1.25	1.81E+04	2.76	1.21E+04	1.85
L-NFRCOMP	6.48E+02	.07	1.04E+03	.11	6.22E+02	.09	5.77E+02	.09
L-DECONRS	7.35E+03	.84	1.22E+04	1.27	8.05E+03	1.23	7.35E+03	1.13
N-ISOPROD	5.20E+03	.59	0.	0.	0.	0.	0.	0.
N-HIGHACT	8.09E+02	.09	5.74E+02	.06	7.04E+02	.11	5.22E+02	.08
N-TRITIUM	2.65E+03	.30	2.09E+02	.02	2.09E+02	.03	4.18E+02	.06
N-SOURCES	5.78E+01	.01	4.10E+01	.00	5.04E+01	.01	3.73E+01	.01
N-TARGETS	4.16E+02	.05	2.95E+02	.03	3.62E+02	.06	2.68E+02	.04
TOTAL	8.78E+05		9.66E+05		6.56E+05		6.50E+05	

TABLE 3-7

Estimated 1980 Untreated Waste Volume Generation Rates

Waste Stream	Basic Volume
Group I : LWR Process Wastes	
P-IXRESIN	17.6 m^3/GW(e)-yr
P-CONCLIQ	123 m^3/GW(e)-yr
P-FSLUDGE	2.2 m^3/GW(e)-yr
P-FCARTRG	11.0 m^3/GW(e)-yr
B-IXRESIN	80.7 m^3/GW(e)-yr
B-CONCLIQ	223 m^3/GW(e)-yr
B-FSLUDGE	179 m^3/GW(e)-yr
Group II : Trash	
P-COTRASH	215 m^3/GW(e)-yr
P-NCTRASH	110 m^3/GW(e)-yr
B-COTRASH	221 m^3/GW(e)-yr
B-NCTRASH	105 m^3/GW(e)-yr
F-COTRASH	80.9 m^3/GW(e)-yr
F-NCTRASH	14.3 m^3/GW(e)-yr
I-COTRASH	4014 m^3/yr
I+COTRASH	4014 m^3/yr
N-SSTRASH	5122 m^3/yr
N+SSTRASH	5122 m^3/yr
N-LOTRASH	1445 m^3/yr
N+LOTRASH	1445 m^3/yr
Group III : Low Specific Activity Wastes	
F-PROCESS	26.8 m^3/GW(e)-yr
U-PROCESS	9.6 m^3/GW(e)-yr
I-LIQSCVL	1402 m^3/yr
I+LIQSCVL	1402 m^3/yr
I-ABSLIQD	159 m^3/yr
I+ABSLIQD	159 m^3/yr
I-BIOWAST	448 m^3/yr
I+BIOWAST	448 m^3/yr
N-SSWASTE	1808 m^3/yr
N-LOWASTE	1719 m^3/yr
Group IV : Special Wastes	
L-NFRCOMP	0.99 m^3/GW(e)-yr
L-DECONRS	1666 m^3/yr
N-ISOPROD	148 m^3/yr
N-TRITIUM	99.3 m^3/yr
N-TARGETS	38.2 m^3/yr
N-SOURCES	5.3 m^3/yr
N-HIGHACT	74.4 m^3/yr

- For calculational convenience, the fraction of the liquid scintillation vial fluid volume currently estimated to be shipped as part of the ABSLIQD waste stream (50% by volume) has been included in the LIQSCVL waste streams. The volume of the LIQSCVL stream represents the volume of the vials containing the scintillation fluid; the actual fluid volume is assumed to be one-half of the vial volume.(13,14)

- The I-BIOWAST and I+BIOWAST stream volumes represent volumes prior to packaging for shipment. Estimated shipping volumes are 0.92 volume parts lime and/or absorbent material to one volume part waste.(15)

- The N-SSWASTE and N-LOWASTE waste stream volumes represent volumes shipped for disposal.

- The L-DECONRS stream volume is composed of "dewatered" ion exchange resins which are projected to be generated during postulated future routine LWR decontamination activities.

- The N-ISOPROD stream volume represents the waste volume as packaged for shipment. Each package is assumed to contain a small volume of liquid solidified in cement within a metal cannister which is then packaged with trash in a 55-gallon drum.

- All other industrial waste stream volumes are assumed to be as shipped for disposal.

3.2 Radionuclide Concentrations

This section briefly summarizes the available information and the procedures used in estimating the radionuclide concentrations of the waste streams projected to be generated between the years 1980 and 2000 for the waste streams presented in Table 3-1. Detailed calculations and additional information can be found in Appendix B.

Low-level radioactive wastes contain a large number of naturally occurring and man-made radionuclides at the time they are produced. Many of these radionuclides are very short-lived and are not of long-term radiological concern. Other isotopes with half lives up to a few years may reach the disposal site but decay to insignificant levels shortly thereafter.

Two criteria were used in selecting the radionuclides considered: (1) its half life must be more than a few years (five years was used as a general guide); and (2) it must be present in comparatively significant quantities in LLW. The biological toxicities of radionuclides were also considered. Radionuclides that will be considered in this report are presented in Table 3-8.

The sources of data on the concentrations of the radionuclides listed in Table 3-8 include:

- o computer-assisted calculations;(16-18)
- o surveys of waste generators;(6,11,12,19)
- o disposal site records;(7,13,20) and
- o radiochemical analyses.(21-25)

Data from these sources suffer several limitations. Nevertheless, cumulative information from all of these sources are sufficient for estimating waste characteristics for purposes of analyzing generic disposal impacts. However, it is essential to consider the limitations of data from each individual source in order to utilize the information from that source properly.

For example, computer calculations, which are often employed in predicting the radioactivity of wastes generated by "burn-up" of nuclear fuels, are based on fuel compositions, consumption (burn-up) rates, and elemental compositions of neutron-irradiated materials. While such calculations can be reasonably accurate, they are not as

TABLE 3-8

Radionuclides Considered in Waste Source Options

Isotope	Half Life (years)	Principal Means of Production
H-3	12.3	Fission; Li-6 (n, α)
C-14	5730	N-14 (n, p)
Fe-55	2.60	Fe-54 (n, γ)
Co-60	5.26	Co-59 (n, γ)
Ni-59	80,000	Ni-58 (n, γ)
Ni-63	92	Ni-62 (n, γ)
Sr-90	28.1	Fission
Nb-94	20,000	Nb-93 (n, γ)
Tc-99	2.12×10^5	Fission; Mo-98 (n, γ), Mo-99 (β^-)
I-129	1.17×10^7	Fission
Cs-135	3.0×10^6	Fission; daughter Xe-135
Cs-137	30.0	Fission
U-235	7.1×10^8	Natural
U-238	4.51×10^9	Natural
Np-237	2.14×10^6	U-238 $(n, 2n)$, U-237 (β^-)
Pu-238	86.4	Np-237 (n, γ), Np-238 (β^-); daughterCm-242
Pu-239	24,400	U-238 (n, γ), U-239 (β^-), Np-239 (β^-)
Pu-240	6,580	Multiple n-capture
Pu-241	13.2	Multiple n-capture
Pu-242	2.79×10^5	Multiple n-capture; daughter Am-242
Am-241	458	Daughter Pu-241
Am-243	7950	Multiple n-capture
Cm-243	32	Multiple n-capture
Cm-244	17.6	Multiple n-capture

well-suited to determining the range of radionuclide concentrations produced by variations of operating conditions at a given reactor nor to representing wastes generated by typical reactors for purposes of analyzing generic disposal impacts.

A common limitation of obtaining concentrations of individual radionuclides from surveys and from disposal site records is that they are frequently derived by application of pre-determined distributions to total gross beta/gamma activities obtained during screening measurements made at the time the wastes are shipped for disposal. These measurements are usually made with relatively unsophisticated instruments and are generally conservative since they include activities contributed by short-lived radionuclides.

The concentrations of several of the radionuclides listed in Table 3-8 have been measured in samples of LWR process wastes.(21-24) These samples include those taken from smaller and older reactors as well as those taken from reactors with a history of fuel failure problems, and are thus believed to be conservative with respect to future LWR wastes. Since radioactive concentrations vary with a reactor's operational cycle (fluctuation in power level, shutdowns and refueling), a larger number of samples is needed to more accurately determine average concentrations.

Furthermore, the sensitivities (minimum detection limits) of the analytical procedures for the radionuclides of interest are not identical but vary with the type and energy of the radiation and with the presence of chemical and radiochemical interferences. Thus, while a few data points may be available for an isotope, they may not be any more accurate than those obtained from screening measurements.

An additional point to be considered in using currently available radionuclide concentrations in the various waste streams is that the processes generating these wastes and the controls on these processes are likely to change. It is anticipated that this change may be away from fission products (e.g., Cs-137 and Sr-90) and toward corrosion products (e.g., Co-60) due to better fuel cladding properties. It is also probable that radionuclide concentrations may increase as the older plants with less sophisticated waste treatment systems are phased out of operation.

The approach developed to estimate radionuclide concentrations in LLW to the year 2000 seeks to minimize the limitations of the available data through use of averaging procedures which reflect the quantity and quality of the available data. A brief discussion of the methodologies used to arrive at these estimates is presented in the following sections. The details of the calculations as well as a complete data compilation are contained in Appendix B. The estimated radioactive concentrations for the untreated waste streams given in Table 3-1 are presented in Tables 3-9 through 3-12.

3.2.1 Central Station Nuclear Power Plants

The LWR process waste streams (all waste streams except trash) are the best characterized of all the LLW streams. This situation allows the 23 radionuclides (Pu-239 and Pu-240 cannot be distinguished by radiochemical methods and are considered here as a single isotope) listed in Table 3-8 to be divided into three groups: (1) radionuclides for which the number of measurements is sufficient to allow averaging; (2) radionuclides for which the number of measurements is insufficient to allow direct averaging; and (3) radionuclides which have not been measured or for which measured concentrations are considered unrepresentative.

Radionuclides in the first group include Co-60, Cs-137, U-238, Pu-238, Pu-239/240, Am-241 and Cm-244. These radionuclides are hereafter referred to as the "basic" isotopes. The estimated concentrations of these basic isotopes are calculated as the geometric means of the

TABLE 3-9

GROUP 1

ISOTOPIC CONCENTRATIONS (CI/M**3)

	P-IXRESIN	P-CONCLIQ	P-FSLUDGE	P-FCARTRG	B-IXRESIN	B-CONCLIQ	B-FSLUDGE
TOTAL	3.36E-02	1.09E-01	1.06E+00	1.86E+00	4.63E+00	2.77E-01	5.24E+00
H-3	2.66E-03	3.45E-03	2.59E-03	1.15E-03	1.92E-02	6.24E-04	1.26E-02
C-14	9.74E-05	1.27E-04	9.55E-05	4.25E-05	1.19E-03	3.89E-05	7.78E-04
FE-55	2.34E-03	2.27E-02	3.10E-01	5.55E-01	9.48E-01	7.60E-02	1.44E+00
NI-59	2.79E-06	2.71E-05	3.71E-04	6.60E-04	9.80E-04	7.85E-05	1.49E-03
CO-60	4.53E-03	4.40E-02	6.00E-01	1.07E+00	1.59E+00	1.27E-01	2.41E+00
NI-63	8.61E-04	8.36E-03	1.14E-01	2.04E-01	2.15E-02	1.72E-03	3.25E-02
NB-94	8.84E-08	8.58E-07	1.17E-05	2.09E-05	3.09E-05	2.48E-06	4.70E-05
SR-90	1.94E-04	2.52E-04	1.89E-04	8.40E-05	3.64E-03	1.18E-04	2.37E-03
TC-99	8.23E-07	1.07E-06	8.03E-07	3.58E-07	7.65E-05	2.50E-06	5.00E-05
I-129	2.44E-06	3.16E-06	2.37E-06	1.06E-06	2.04E-04	6.65E-06	1.33E-04
CS-135	8.23E-07	1.07E-06	8.03E-07	3.58E-07	7.65E-05	2.50E-06	5.00E-05
CS-137	2.19E-02	2.85E-02	2.14E-02	9.54E-03	2.04E+00	6.65E-02	1.33E+00
U-235	4.71E-08	6.15E-08	1.46E-07	3.64E-07	5.33E-08	3.44E-08	3.32E-07
U-238	3.71E-07	4.84E-07	1.15E-06	2.87E-06	4.20E-07	2.71E-07	2.61E-06
NP-237	9.06E-12	1.18E-11	2.81E-11	7.02E-11	1.02E-11	6.61E-12	6.38E-11
PU-238	2.60E-05	5.12E-05	4.76E-05	2.51E-04	8.34E-05	1.99E-04	4.66E-04
PU-239/240	1.82E-05	3.31E-05	1.55E-04	3.80E-04	5.34E-05	9.43E-05	2.36E-04
PU-241	7.94E-04	1.44E-03	6.75E-03	1.66E-02	2.60E-03	4.60E-03	1.15E-02
PU-242	3.99E-08	7.25E-08	3.39E-07	8.34E-07	1.17E-07	2.06E-07	5.18E-07
AM-241	1.87E-05	2.99E-05	2.64E-04	1.64E-04	2.32E-05	1.20E-04	1.56E-04
AM-243	1.26E-06	2.02E-06	1.78E-05	1.10E-05	1.57E-06	8.10E-06	1.05E-05
CM-243	9.92E-09	1.17E-08	3.10E-07	1.93E-07	2.70E-08	2.59E-07	2.97E-07
CM-244	1.38E-05	1.92E-05	1.77E-04	1.10E-04	1.82E-05	2.05E-04	2.24E-04

TABLE 3-10

GROUP 2

ISOTOPIC CONCENTRATIONS (CI/M**3)

	P-COTRASH	P-NCTRASH	B-COTRASH	B-NCTRASH	F-COTRASH	F-NCTRASH	I-COTRASH	N-SSTRASH	N-LOTRASH
TOTAL	2.28E-02	5.25E-01	2.35E-02	3.79E+00	5.58E-06	5.33E-06	1.13E-01	1.12E-05	3.53E-02
H-3	3.04E-04	6.99E-03	6.75E-05	1.09E-02	0.	0.	9.13E-02	0.	2.85E-02
C-14	1.12E-05	2.57E-04	4.17E-06	6.73E-04	0.	0.	5.26E-03	0.	1.64E-03
FE-55	5.97E-03	1.37E-01	6.01E-03	9.69E-01	0.	0.	0.	0.	0.
NI-59	7.11E-06	1.64E-04	6.21E-06	1.00E-03	0.	0.	0.	0.	0.
CO-60	1.15E-02	2.65E-01	1.01E-02	1.62E+00	0.	0.	1.04E-02	0.	3.25E-03
NI-63	2.19E-03	5.05E-02	1.36E-04	2.19E-02	0.	0.	0.	0.	0.
NB-94	2.25E-07	5.18E-06	1.96E-07	3.16E-05	0.	0.	0.	0.	0.
SR-90	2.22E-05	5.11E-04	1.27E-05	2.05E-03	0.	0.	1.45E-03	0.	4.53E-04
TC-99	9.42E-08	2.17E-06	2.68E-07	4.33E-05	0.	0.	3.39E-09	0.	1.06E-09
I-129	2.78E-07	6.41E-06	7.14E-07	1.15E-04	0.	0.	0.	0.	0.
CS-135	9.42E-08	2.17E-06	2.68E-07	4.33E-05	0.	0.	0.	0.	0.
CS-137	2.51E-03	5.78E-02	7.14E-03	1.15E+00	0.	0.	4.56E-03	0.	1.42E-03
U-235	7.89E-09	1.82E-07	1.22E-09	1.97E-07	1.18E-06	1.13E-06	0.	2.36E-06	0.
U-238	6.22E-08	1.43E-06	9.60E-09	1.55E-06	4.40E-06	4.20E-06	0.	8.80E-06	0.
NP-237	1.52E-12	3.49E-11	2.35E-13	3.78E-11	0.	0.	0.	0.	0.
PU-238	5.97E-06	1.38E-04	2.30E-06	3.71E-04	0.	0.	0.	0.	0.
PU-239/240	5.53E-06	1.27E-04	1.16E-06	1.86E-04	0.	0.	0.	0.	0.
PU-241	2.41E-04	5.55E-03	5.63E-05	9.08E-03	0.	0.	0.	0.	0.
PU-242	1.21E-08	2.79E-07	2.53E-09	4.08E-07	0.	0.	0.	0.	0.
AM-241	3.96E-06	9.12E-05	9.67E-07	1.56E-04	0.	0.	4.82E-06	0.	1.51E-06
AM-243	2.67E-07	6.15E-06	6.52E-08	1.05E-05	0.	0.	0.	0.	0.
CM-243	2.74E-09	6.30E-08	1.93E-09	3.12E-07	0.	0.	0.	0.	0.
CM-244	2.61E-06	6.00E-05	1.49E-06	2.41E-04	0.	0.	0.	0.	0.

TABLE 3-11

GROUP 3

ISOTOPIC CONCENTRATIONS (CI/M**3)

	F-PROCESS	U-PROCESS	I-LQSCNVL	I-ABSLIQD	I-BIOWAST	N-SSWASTE	N-LOWASTE
TOTAL	1.08E-04	3.80E-04	9.60E-03	1.99E-01	2.06E-01	2.17E-04	2.11E-02
H-3	0.	0.	5.01E-03	1.42E-01	1.75E-01	0.	1.63E-02
C-14	0.	0.	2.51E-04	8.16E-03	1.01E-02	0.	9.36E-04
FE-55	0.	0.	0.	0.	0.	0.	0.
NI-59	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	0.	0.	3.12E-02	3.99E-03	0.	1.47E-03
NI-63	0.	0.	0.	0.	0.	0.	0
NB-94	0.	0.	0.	0.	0.	0.	0.
SR-90	0.	0.	4.34E-03	4.34E-03	8.33E-03	0.	1.31E-03
TC-99	0.	0.	0.	1.02E-08	6.51E-09	0.	7.76E-10
I-129	0.	0.	0.	0.	0.	0.	0.
CS-135	0.	0.	0.	0.	0.	0.	0.
CS-137	0.	0.	0.	1.37E-02	8.76E-03	0.	1.04E-03
U-235	2.30E-05	1.65E-05	0.	0.	0.	4.60E-05	0.
U-238	8.54E-05	3.64E-04	0.	0.	0.	1.71E-04	0.
NP-237	0.	0.	0.	0.	0.	0.	0.
PU-238	0.	0.	0.	0.	0.	0.	0.
PU-239/240	0.	0.	0.	0.	0.	0.	0.
PU-241	0.	0.	0.	0.	0.	0.	0.
PU-242	0.	0.	0.	0.	0.	0.	0.
AM-241	0.	0.	0.	0.	0.	0.	0.
AM-243	0.	0.	0.	0.	0.	0	0.
CM-243	0.	0.	0.	0.	0.	0.	0.
CM-244	0.	0.	0.	0.	0.	0.	0.

TABLE 3-12

GROUP 4

ISOTOPIC CONCENTRATIONS (CI/M**3)

	L-NFRCOMP	L-DECONRS	N-ISOPROD	N-HIGHACT	N-TRITIUM	N-SOURCES	N-TARGETS
TOTAL	4.04E+03	1.56E+02	1.50E+01	2.10E+02	2.33E+03	5.76E+03	8.04E+01
H-3	0.	1.08E-02	4.20E-02	0.	2.33E+03	8.63E+02	8.04E+01
C-14	2.59E-01	5.13E-04	4.51E-05	1.32E-02	0.	5.76E+01	0.
FE-55	2.23E+03	4.05E+01	0.	1.15E+02	0.	0.	0.
NI-59	1.40E+00	4.49E-02	0.	6.56E-02	0.	0.	0.
CO-60	1.60E+03	7.28E+01	0.	8.48E+01	0.	1.73E+03	0.
NI-63	2.09E+02	3.69E+00	0.	1.06E+01	0.	2.30E+02	0.
NB-94	8.19E-03	1.42E-03	0.	4.47E-04	0.	0.	0.
SR-90	0.	4.28E-02	6.27E+00	0.	0.	1.15E+03	0.
TC-99	0.	1.20E-05	3.27E-04	0.	0.	0.	0.
I-129	0.	3.34E-05	2.72E-06	0.	0.	0.	0.
CS-135	0.	1.20E-05	3.27E-04	0.	0.	0.	0.
CS-137	0.	3.18E-01	8.73E+00	0.	0.	1.15E+03	0.
U-235	0.	6.84E-05	1.02E-05	0.	0.	0.	0.
U-238	0.	5.40E-04	3.81E-05	0.	0.	0.	0.
NP-237	0.	1.32E-08	5.33E-13	0.	0.	0.	0.
PU-238	0.	1.34E+00	1.97E-04	0.	0.	0.	0.
PU-239/240	0.	1.77E+00	5.55E-05	0.	0.	0.	0.
PU-241	0.	3.55E+01	7.10E-03	0.	0.	0.	0.
PU-242	0.	3.37E-03	9.57E-08	0.	0.	0.	0.
AM-241	0.	5.29E-03	1.10E-05	0.	0.	5.76E+02	0.
AM-243	0.	3.59E-04	1.25E-06	0.	0.	0.	0.
CM-243	0.	3.46E-04	1.65E-04	0.	0.	0.	0.
CM-244	0.	3.27E-03	2.88E-07	0.	0.	0.	0.

measured concentrations in each waste stream. The comparatively short-lived isotope Cm-242 is included as a basic isotope for one specific case; it is used to estimate the concentrations of other curium isotopes, Cm-243 and Cm-244, in PWR filter sludge (see below).

Geometric averaging is equivalent to arithmetic averaging of the logarithm of the data values; it is calculated as the (n)th root of the product of the (n) data values. The geometric average corresponds to the use of a log-normal distribution rather than a standard gaussian distribution to represent the variation of the measured value due to independent uncontrollable parameters. This type of averaging has already been recognized by several investigators as being more suitable for environmental data when the applicable statistical distribution is not known.(26-28) The use of geometric means rather than arithmetic means allows representative estimates to be made from sets of data that contain a few concentrations which are several orders of magnitude greater than the majority in the set and which would dominate the average, resulting in unrepresentative values, if arithmetic means were used.

The difference in means is readily illustrated by considering a set of data consisting of 20 values of 1 and one value of 1000. The arithmetic average of these 21 values is 48.6 and the geometric average is 1.39. The geometric average is clearly more representative of the typical value. Variations of this magnitude have been observed in radionuclide concentration of waste streams at several LWR's.(21-23) Geometric averaging is therefore a scientifically accepted compromise between the impracticality of investigating the conditions under which each sample was collected and the use of uncharacteristically high arithmetic means.

The second and third group of radionuclides were "scaled" to the above list of basic radionuclides. The scaled radionuclides and the basic radionuclides are given in Table 3-13.

TABLE 3-13

Basic and Scaled Radionuclides for LWR Process Waste Streams

Basic Isotope	Scaled Isotopes
Co-60	Fe-55, Ni-59, Ni-63, Nb-94
Cs-137	H-3, C-14, Sr-90 Tc-99, I-129, Cs-135
U-238	U-235, Np-237
Pu-238	--
Pu-239/240	Pu-241, Pu-242
Am-241	Am-243, Cm-242*
Cm-242	Cm-243, Cm-244*
Cm-244	--

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* Only for the P-FSLUDGE waste stream.

The second group of radionuclides -- those for which the number of measurements is insufficient to allow direct geometric averaging -- consists of H-3, C-14, Fe-55, Ni-63, Sr-90, I-129, Pu-241, and Pu-242. The concentrations of these radionuclides are calculated by "scaling" to the concentration of an appropriate basic isotope. These radionuclides are paired on the basis of a common source and/or method of production. For example, activated corrosion products (Fe-55 and Ni-63) are scaled to Co-60 which is also an activated corrosion product; fission products (Sr-90, I-129, and H-3, which is also produced by activation) are scaled to Cs-137 which is also a fission product; and Pu-241 and Pu-242 are scaled to Pu-239/240, the nuclides they originate from through multiple neutron capture. Carbon-14 is rather difficult to categorize; it is arbitrarily scaled to Cs-137.

Scaling was accomplished using data for samples which were analyzed for both the radionuclide to be scaled and the appropriate basic isotope. The ratio of the concentration of the radionuclide to be scaled to that of the basic isotope was calculated for each data pair. A "scaling factor" for each of the radionuclides in this second group was then calculated as the geometric average of each set of ratios. (The scaling factors were calculated by reactor type only (BWR's and PWR's) rather than by reactor type and by waste stream like the basic radionuclides.) The computed scaling factors were then applied to the geometric averages of the basic radionuclides to obtain the estimated concentrations of the scaled radionuclides given in Table 3-13. An additional scaling factor was calculated by this procedure for Cm-242 in PWR filter sludge using Cm-242/Am-241 data pairs for PWR cartridge filters.

The third group of radionuclides consists of Ni-59, Nb-94, Tc-99, Cs-135, U-235, Np-237, Am-243, and Cm-243. For these radionuclides, concentrations obtained from computer calculations(29) (Ni-59 and Nb-94) or from disposal site records(30) were ratioed to the mean concentrations of the basic isotopes to obtain scaling factors.

In the case of U-235, an average enrichment of 2% (to account for burn-up) was assumed, and was then used as described above to estimate concentrations from U-238 concentrations in each stream.

The radioactive concentrations of BWR and PWR trash were estimated by assuming that the radioactivity of the trash is proportional to the total activity of the BWR and PWR process waste streams, respectively. Accordingly, the estimated concentrations (Table 3-9) and the as-generated volumes of LWR process wastes were used to calculate normalized isotopic distributions from the volume-weighted average concentration of each radionuclide in BWR and PWR process wastes. These distributions were then applied to the average gross activities estimated to be contained in PWR compactible and non-compactible trash (0.0228 Ci/m^3 and 0.525 Ci/m^3, respectively), and BWR compactible and non-compactible trash (0.0235 Ci/m^3 and 3.79 Ci/m^3, respectively).(6,20) The resultant concentrations, presented in Table 3-10, are conservative since they are bas on total activities which include the contributions of short-lived radionuclides.

The radionuclide concentrations given in Table 3-12 for LWR non-fuel reactor components (L-NFRCOMP) were estimated by assuming that the total activity is due to neutron activation of steel components. A normalized distribution calculated from ORIGEN calculations of the radioactivity of highly activated metals(29) was applied to a total estimated gross activity of 4040 Ci/m^3.

As noted previously (see Section 3.1.1), the radionuclide concentrations of future LWR decontamination wastes are rather difficult to estimate considering the many factors affecting the concentrations. The distribution of the gross activity between the radionuclides, however, may be expected to resemble the distribution among the radionuclides in crud deposits (metal oxides) in LWR cooling systems.(21-23,25) Therefore, the radionuclide concentrations given in Table 3-12 for LWR decontamination resins were calculated from the available data on crud deposits. Scaling procedures similar to those used for LWR process wastes were used, although no differentiation of nuclide concentrations was made between future BWR and PWR wastes.

The basic crud isotopes are Co-60, Cs-137, U-238, Pu-238, Pu-239/240, Am-241, Cm-242 and Cm-244. Sufficient data is available for Sr-90 and Pu-241 in LWR crud to allow calculations of scaling factors as geometric means of ratios as described for LWR process wastes. Results of the analysis of a single sample(25) were used to scale Fe-55 and Ni-63 to Co-60. Scaling factors for the remaining radionuclides were calculated as geometric means of the corresponding scaling factors for BWR and PWR process wastes. After applying these scaling factors to the concentrations of the basic crud isotopes, the concentrations of all 23 radionuclides were normalized and applied to a total estimated activity of 156 Ci/m^3 to obtain the concentrations given in Table 3-12.

The details of these calculations and the basic data utilized can be found in Appendix B.

3.2.2 Other Nuclear Fuel Cycle Facilities

These waste streams consist of process wastes and trash from fuel fabrication plants and process wastes from uranium conversion facilities. Little data is available on the radionuclide concentrations of these streams, although U-235 and U-238 were the only radionuclides identified as being included in these waste streams.

Radionuclide concentrations in fuel fabrication wastes were determined based on data obtained from radioactive shipment records (RSR's) of waste delivered to the Maxey Flats Disposal Facility. The masses of special nuclear materials reported in the RSR's were used to calculate concentrations of U-235 in each waste stream. Concentrations of U-238 were then calculated by assuming that the uranium in these wastes contained 4 weight percent U-235. The estimated concentration of fuel fabrication wastes are given in Tables 3-10 and 3-11.

The concentrations of U-235 and U-238 in the process waste from uranium conversion facilities were calculated from data given in reference 10. It was assumed that the uranium was unenriched (0.711 percent U-235 by weight). Estimated concentrations are given in Table 3-11.

3.2.3 Institutional Facilities

The most complete set of data available for institutional waste volumes and radionuclides were obtained during surveys of these generators conducted by the University of Maryland. However, in the published form,(11-12) the data is not suitable for estimating the radionuclide concentrations in each waste stream. For the purposes of this report, the survey data was reformatted and additional analysis performed.(15,19) The results of this analysis, presented in Table 3-14, combined with the volumes of each waste stream,(11-12) were used to estimate the radionuclide concentrations in the institutional waste streams given in Tables 3-10 and 3-11. The methodology employed is briefly described below.

The data presented in Table 3-14 was compiled from the survey data base by first summing the total reported activity of each radionuclide shipped to disposal sites, as well as the total volume of all wastes reported to contain each radionuclide. The form of the data did not allow these summations to be made for individual waste streams, but did allow determination of whether a radionuclide was present in a given stream. In Table 3-14, an "X" indicates that an isotope was reported in the stream indicated. The total activity of each radionuclide was then divided by the total volume of waste reported to contain that radionuclide to obtain initial radionuclide concentrations.

TABLE 3-14

Radionuclide Distribution in Institutional Wastes in 1977

Nuclide	Waste Fraction* (ft^3)	Dry Solids	Liquid Scint. Vials	Absorbed Liquids	Biological Wastes	Total Activity Shipped (mCi)
H-3	159,697	X	X	X	X	236,151
C-14	158,060	X	X	X	X	13,488
Na-22	96,539	X		X	X	207
P-32	148,684	X	X	X	X	24,729
P-33	15,020	X	X	X	X	18
S-35	140,729	X	X	X	X	12,649
Cl-36	45,974	X	X	X	X	14
Ca-45	135,238	X	X	X	X	2,041
Sc-46	26,962	X		X	X	128
Cr-51	146,634	X	X	X	X	9,918
Mn-54	14,903	X		X	X	8
Fe-59	37,958	X		X	X	268
Co-57	37,600	X		X	X	212
Co-60	22,979	X		X	X	3,341
Ga-67	34,730	X		X	X	2,319
Se-75	79,046	X	X	X	X	948
Rb-86	64,239	X	X	X	X	226
Sr-85	42,931	X		X	X	309
Sr-90	13,997	X	X	X	X	573
Nb-95	10,976	X		X	X	136
Mo-99	13,674	X				15,080
Tc-99^{m}	38,348	X		X	X	19,903
In-111	15,175	X		X	X	179
Sn-113	15,175	X		X	X	194
I-125	148,442	X	X	X	X	47,882
I-131	69,693	X		X	X	6,620
Xe-133	6,234	X				1,356
Cs-137	15,086	X		X	X	1,101
Ce-141	32,856	X		X	X	175
Yb-169	8,490	X		X	X	315
Tl-201	15,667	X		X	X	565
Others	116,895	X	X	X	X	3,760

Source: Reference 12

* Total volume of shipped waste reported to contain a given isotope. Total volume of shipped waste was 185,160 ft^3.

Radionuclide concentrations in each institutional waste stream were derived from the initial concentrations by consideration of: the as-shipped volume of the waste stream relative to the total volume of all four streams (42.3% trash, 38.5% liquid scintillation vials, 10% absorbed liquids and 9% biowaste); the presence or absence of a radionuclide in the waste stream; and the fraction of the as-shipped volume which consists of waste. The following assumptions were then applied.

- One-half the volume of liquid scintillation vials is occupied by scintillation fluids; one-half the volume of absorbed liquids is scintillation fluids and one-half is aqueous liquids.(11)
- The tritium and C-14 activities of liquid scintillation fluids are 10 nCi/cm^3 and 5 nCi/cm^3, respectively.(11-12)
- All Mo-99 and Tc-99m have decayed to Tc-99 prior to shipment.
- The activity of Co-60 in biowaste is one-fifth its activity in the other waste streams.(15)
- Institutions shipped 6230 m^3 of trash containing 30 mCi of Am-241.(15)

The radionuclide concentrations in institutional wastes estimated by this procedure are given in Tables 3-10 and 3-11. Further details of the calculations and the equations utilized to estimate the concentrations can be found in Appendix B.

3.2.4 Industrial Facilities

The radionuclide concentrations of industrial wastes were estimated based upon a number of information sources as summarized in Appendix A. Radionuclide concentrations are presented in Table 3-10, 3-11, and 3-12. The details of the calculations can be found in Appendix B.

Medical isotope production (N-ISOPROD) wastes, which consist of trash and solidified aqueous liquids, were considered as a single waste stream (see Section 2.5). The radionuclide concentrations of this waste stream are not well characterized. Data obtained from available Maxey Flats Disposal Facility RSR's for the radionuclides of interest are limited to the combined Sr-90/Cs-137 radioactivity, grams of U-235, and waste volumes.

In order to estimate the concentrations of the remaining radionuclides, the waste density was assumed to be 1.6 g/cm^3 and the total activity of alpha-emitting transuranic radionuclides was assumed to be 1 nCi/g. The radionuclides were then divided into three groups: (1) activation and fission products, (2) uranium, and (3) transuranium radionuclides. The concentration of U-238 was then calculated by assuming 4% by weight U-235 enrichment. Information regarding the radionuclide distribution in spent fuel(31) was used to obtain normalized distributions of activation and fission products and of transuranics. These distribution were used with the combined activities of Sr-90 and Cs-137 obtained from the Maxey Flats RSR's and the assumed activity of the alpha-emitting transuranics to calculate the radionuclide concentrations given in Table 3-12.

Industrial high activity (N-HIGHACT) wastes consist of neutron irradiation capsules, activated components from research reactors, and other activated waste materials. The radionuclide concentrations of these wastes given in Table 3-12 were calculated using scaling factors developed for highly activated metals from decommissioning activities.(29)

The total radioactivity of industrial tritium manufacturing wastes, 2330 Ci/m^3, is assumed to be due to tritium alone.

Estimation of the activity of sealed sources (N-SOURCES) and the isotopic distribution of this activity is rather difficult since they are shipped for disposal infrequently and at irregular intervals. Scaling factors were assumed and applied based on several sources of information (see Appendix B).

Accelerator targets (N-TARGETS stream) consist of tritium absorbed on titanium foils. Since there is no indication that induced activities are present,(12) the activity of this waste stream, 80.4 Ci/m^3 is assumed to be due to tritium alone.

The only radionuclides identified in source and special nuclear material wastes are U-235 and U-238. The wastes are generated primarily during processing of metals and compounds containing depleted uranium. The uranium isotopes are conservatively assumed to be present in the same ratio as in natural uranium; thus, 4.3% of the total activity is assumed to be due to U-235 and 95.6% due to U-238.

The types of materials comprising the industrial low activity waste stream are the industrial equivalents of institutional wastes - i.e., trash, liquid scintillation vials, absorbed liquids, and biowastes. As discussed in Section 2.5, these types of wastes are not sufficiently well-characterized to be considered as separate streams. It was therefore assumed that these industrial wastes have the same distribution of radionuclide concentrations as institutional wastes. Concentrations of individual radionuclides were then estimated using a volume-weighted averaging technique analogous to that used for LWR trash.

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4.0 WASTE PROCESSING OPTIONS

There are many processing technologies currently available that can be utilized to alter and/or improve the performance characteristics of radioactive waste forms. This section briefly considers several of these technologies and presents their estimated impacts on waste generators and/or disposal site operators. Some additional information is provided in Appendices C and D.

In order to assess the comparative effects of the waste processing options in this report, four impact measures are considered in this section. These impact measures include occupational exposures, population exposures, costs, and energy use. Only incineration is assumed to result in potential significant population exposures as a result of processing. Other processes, including evaporation, compaction, solidification, and packaging, are assumed not to result in significant additional population exposures.

Waste processing options are considered in three sections in this chapter. Section 4.1 addresses processes that result in a reduced volume of waste after processing. Section 4.2 addresses processes that result in an increased volume of waste after processing. Section 4.3 briefly discusses the possible use of high-integrity packages for containment of radionuclides during transportation and after disposal.

4.1 Volume Reduction

There are three basic processes that can be applied to waste streams which result in overall waste volume reduction: (1) physical processes such as compaction, (2) thermal processes such as evaporation, and (3) incineration and other related combustion processes.

Each of these processes produces a concentrate stream and an effluent stream. The respective concentrate streams are compressed wastes, liquids or crystals, and ash. The respective effluents displaced are air, vapor, and gas and vapor. The activity per unit volume of the concentrate stream is usually higher than that of the untreated waste with the possible exception of volatile and semi-volatile nuclides such as tritium, carbon, and iodine which may be entrained as vapor and/or combustion products in the effluent stream.

The volume reduction factor (VRF) is defined in this report as the ratio of the waste volume that is input to the process (untreated volume) to that of the concentrated output (rather than effluent) waste volume (treated volume).

4.1.1 Compaction

Compaction is an often-used method -- particularly at nuclear fuel cycle facilities -- of reducing the volume of waste streams containing compressible material such as paper, plastic, glass, wood, and light-gauge metal. Most of the volume reduction is attained by compressing the waste to reduce its void volume. The term compactor is usually applied to hydraulic or mechanical rams which compress wastes into 55 gallon steel drums. The drums are then used as disposal containers. Typical hydraulic rams generate 20,000 to 30,000 pounds of force, and are fitted with shrouds and simple air filtration systems to minimize release of airborne radioactivity.

Most compactors now in use can achieve average volume reduction factors of about two, while newer compactors which place a metal inner sleeve inside the drum during compaction, which is subsequently withdrawn, are capable of a volume reduction factor of about four.(1) Industrial hydraulic presses similar to those used to crush automobiles may be useful for compacting heavier-gauge metal items such as pipes, tools, cans, drums, and scaffolding.

In this section, three types of compactors are considered: compactors that can be utilized to achieve volume reduction factors of around 1.5 to 2; improved compactors that can achieve volume reduction factors of about 3 to 4; and industrial hydraulic presses which are assumed to be capable of achieving volume reduction factors of about 6. The compactors and improved compactors can be utilized by any facility capable of implementing its own processing system; however, industrial hydraulic presses are assumed to be operable only at a centralized waste processing facility.

The waste streams to which these compaction techniques are applied, and their unit impact measures are summarized in Table 4-1.(2)

4.1.2 Evaporation

Evaporators concentrate liquid wastes by heating them to vaporize the volatile components. The vaporized water generally contains greatly reduced quantities of dissolved solids, suspended solids, and radioactivity relative to those found in the input waste. In the nuclear industry the vaporized water is normally condensed and collected, and then either discharged or recycled after testing to determine whether the condensate requires additional treatment. The concentrated solution (bottoms) left in the evaporator retains virtually all of the solids and radioactivity and is solidified and shipped to a disposal site.

Evaporators can be categorized according to their methods of heat transfer.(3) Natural circulation evaporators use convection as the means of heat transfer. Forced circulation evaporators use mechanical devices such as pumps to improve the flow of liquid over the heating surfaces. Fluidized-bed dryers produce dry salts by injecting atomized waste liquids into a hot bed of inert granules which is suspended (fluidized) in a stream of hot air. The inert carrier process uses a hot bath of inert fluid recirculating at

TABLE 4-1

Compaction Techniques and Impacts

Compaction Technique	Cost* per m^3	Man-Hours* per m^3	Fuel Use* gallons per m^3	Waste Streams	Volume Reduction Factor
Compactor/Shredder	$ 335	15	4.6	P-COTRASH	2.0
				B-COTRASH	2.0
				F-COTRASH	1.5
				I-COTRASH	2.0
				N-SSTRASH	1.5
				N-LOTRASH	2.0
				I-LOSCNVL	1.28
Improved Compactor/ Shredder	$ 503	15	4.6	I+COTRASH	4.0
				N+SSTRASH	3.0
				N+LOTRASH	4.0
Industrial Hydraulic Press	$1006	15	4.6	P-NCTRASH	6.0
				B-NCTRASH	6.0
				F-NCTRASH	6.0

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* Cost and man-hours are given in unit volume of input volume (untreated) waste. Impact measures were obtained from Reference 2.

high velocities as the heat exchanger. Solidification in bitumen can also be considered to be evaporation. The ideal evaporator produces a condensate that is free of radioactivity while attaining the maximum concentration or volume reduction.

In this work, evaporator/crystallizers, a type of forced circulation evaporator, are assumed to be utilized as an option to further concentrate the already concentrated liquid waste streams of LWRs. The assumed volume reduction factors for evaporator/crystallizers are 6.0 and 2.4 for P-CONCLIQ and B-CONCLIQ streams, respectively, and the impact measures are $690, 4.42 manhours, and 56.3 gallons of fuel per m^3 of untreated input waste.(2)

4.1.3 Incineration

Incinerators and related devices decompose combustible waste materials by thermal oxidation. Combustion or incineration involves complete oxidation of wastes by burning in an excess of oxygen (air). Pyrolysis involves partial oxidation in an oxygen-deficient atmosphere. Oxidation can also be accomplished by introducing combustible wastes and air into a bath of molten salt. Alternatively, acid digesters oxidize wastes in a hot mixture of concentrated nitric and sulfuric acids.

The various types of incinerators, pyrolyzers, and other such devices currently used or being developed for volume reduction of radioactive waste are too numerous to be discussed here individually. Two representative types of incinerators have been selected for discussion in this report: pathological incinerators and fluidized bed incinerators. The waste streams treated with these two types of incinerators and the resultant unit impact measures are presented in Table 4-2.(2,4)

Pathological incinerators are typically multiple-chamber, hot refractory hearth incinerators and are normally operated with little or no

TABLE 4-2

Incineration Techniques and Impacts

Incineration Technique	Cost* per m^3	Man-Hours* per m^3	Fuel Use gallons per m^3	Waste Streams	Volume Reduction Factor
Pathological Incinerator	\$2060	8	116	I-COTRASH	20.0
				N-SSTRASH	10.0
				N-LOTRASH	20.0
				I-LQSCNVL	4.52
				I-ABSLIQD	100.0
				I-BIOWAST	15.0
Fluidized Bed Incinerator (at generators)	\$1938	6.12	129	P-IXRESIN	18.0
				P-CONCLIQ	8.0
				P-FSLUDGE	5.0
				B-IXRESIN	18.0
				B-CONCLIQ	6.4
				B-FSLUDGE	5.0
				P-COTRASH	80.0
				B-COTRASH	80.0
				F-COTRASH	40.0
				L-DECONRS	18.0
Fluidized Bed Incinerator (at regional facility)	\$1039	5.35	72	I+COTRASH	80.0
				N+SSTRASH	40.0
				N+LOTRASH	80.0

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* Cost and man-hours are given in unit volume of untreated waste. Impact measures were obtained from Reference 2,4.

off-gas treatment. They are designed primarily for the incineration of animal carcasses and operate at approximately 900 to 1000°C. Pathological incinerators may also be used by institutional waste generators for volume reduction of other biowastes, scintillation fluids, organic liquids, and trash. Aqueous liquids can also be evaporated on the refractory hearth.

Fluidized bed incinerators operate by injecting combustible wastes into a hot bed of inert granules fluidized by a stream of hot gas. Typical fluidized bed incinerators can burn trash, organic solvents, and ion exchange resins. Wastes are normally screened to remove metal objects and shredded before entering the process vessel. The process vessel is maintained at 800 to 1000°C. The ash produced is carried out of the process vessel, separated from the hot effluent stream, and collected for subsequent solidification.

Recent investigations(5) indicate that thermal combustion is apparently the most effective way of removing chelating agents (chemicals that increase radionuclide mobility from the waste and during migration in groundwater) from the wastes. This requires the use of incinerators for improving the waste form by eliminating the presence of chelating agents.

4.2 Volume Increase

There are three basic processes that can be applied to waste streams which result in an overall increase in waste volume: solidification, addition of absorbent materials, and packaging. The activity per unit volume of the product stream is lower than that of the input waste.

The volume increase factor (VIF) is defined in this report as the ratio of the volume of the treated waste product to the volume of the input untreated waste.

4.2.1 Solidification

This section considers a number of solidification processes that can be applied to waste streams such as LWR process wastes (concentrated liquids, resins, filter sludges and cartridge filters) or dry salts and ashes produced by calciners and incinerators. Cartridge filters are assumed to be solidified by pouring the solidification agent into the spaces between the currently utilized shipping containers and the cartridges. This results in no change to the currently shipped volume of the waste stream.

The solidification agents or techniques considered in this report are selected from those which are currently in use or are being actively marketed. These include cement, synthetic polymer, and urea-formaldehyde systems (see Appendix D).

Although urea-formaldehyde is a synthetic polymer, its properties are sufficiently different from those of more recently introduced polymers (vinyl ester styrene, epoxy, polyester) to justify separate consideration. Absorbents such as vermiculite and diatomaceous earth are not considered to be solidification agents since they do not chemically or physically bind the wastes.

Both cement and urea-formaldehyde solidification systems are currently used by LWR's although the use of urea-formaldehyde is decreasing. Bitumen (another agent) and vinyl ester-styrene (a synthetic polymer) are being actively marketed. Several bitumen solidification systems (which are widely used in Europe) have been sold but are not yet operational in this country. Synthetic polymer systems are being used in LWR's, including the Dresden-Unit 1 nuclear power plant where decontamination solutions are to be solidified. Polyester (another synthetic polymer) has been evaluated in laboratory and pilot plant studies using simulated LWR liquid wastes and may be routinely used in the future.

In the analyses to determine the performance and technical requirements for disposal of LLW, three solidification scenarios are postulated:

- o Solidification scenario A assumes continuation of existing practices resulting in waste performance characteristics which are comparatively less desirable than the following two solidification scenarios. This is simulated by assuming that 50 percent of the waste stream is solidified using urea-formaldehyde systems and the other 50 percent using cement systems.

- o Solidification scenario B assumes improved waste performance characteristics over the previous case. This is simulated by assuming that 50 percent of the waste stream is solidified using cement systems and the other 50 percent using synthetic polymer systems.

- o Solidification scenario C assumes further improved waste performance characteristics achievable with the currently available technology. This is simulated by assuming that all the waste is solidified using synthetic polymer systems.

These solidification processes, volume increase factors, and the impact measures associated with these processes are summarized in Table 4-3.

4.2.2 Absorbent Materials

Absorbent materials are currently added to several institutional waste streams to minimize potential transportation impacts. These streams include liquid scintillation vial (LSV) waste, absorbed liquid waste, and biowaste. Existing commercial disposal facility operators require that these wastes be packaged with specified proportions of waste to absorbent material before they are accepted for disposal.(7,8) For

TABLE 4-3

Solidification Techniques and Impacts

Solidification Technique	Cost* per m^3	Man-Hours* per m^3	Fuel Use gallons per m^3	Waste Streams	Volume Increase Factor
Scenario A	$1282	24	40	P-CONCLIQ	1.4
				B-CONCLIQ	1.4
Scenario B	$1873	24	40	P-IXRESIN	1.65
				P-CONCLIQ	1.82
				P-FSLUDGE	1.65
				B-IXRESIN	1.65
				B-CONCLIQ	1.56
				B-FSLUDGE	1.65
				I-ABSLIQD	1.65
Scenario C	$2445	24	40	P-IXRESIN	2.00
				P-CONCLIQ	2.00
				P-FSLUDGE	2.00
				B-IXRESIN	2.00
				B-CONCLIQ	2.00
				B-FSLUDGE	2.00
				I-ABSLIQD	2.00
				All Ash	2.00

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* Cost and man-hours are given in unit volume of treated waste. Impact measures are detailed in References 2 and 6.

example, LSV waste is required to be packaged using enough absorbent material to absorb twice the total volume of the liquid that is in the package.(8)

The absorbent materials used include vermiculite and diatomaceous earth. Lime is frequently added to the biowaste stream. Double-packaging of these waste streams is also used for additional safety. For the liquid scintillation vial and the absorbed liquid waste streams, a volume increase factor of 3.0 is assumed. For the biowaste stream, a volume increase factor of 1.92 is assumed.

Waste packages containing absorbent material cannot be processed by compaction or incineration at a centralized processing facility with currently available methods. This is because many of the common absorbent materials, an integral part of the waste stream when the package leaves the waste generator, are not compactible or incinerable; absorbents that are incinerable are either not cost-effective or not compatible with the waste streams. Other processing techniques are either not compatible with the waste streams (e.g., cement solidification of liquid scintillation vials) or would result in an increase of the volume of the waste, and as a consequence would not be cost-effective. Therefore, these wastes would have to be processed by the waste generator. While many waste generators are capable of implementing their own waste processing alternatives such as solidification instead of use of absorbent material, there is no alternative cost-effective treatment method (other than the use of absorbents) for small waste generators such as individual physicians, small medical groups, and small colleges for several waste streams. Therefore, it is assumed in this report that no processing takes place after the waste leaves the generator for the following waste streams: I+LIQSCVL, I+ABSLIQD, and I+BIOWAST.

4.2.3 Packaging

Waste packaging also results in an overall increase in waste volume where the entire container volume is not utilized. Generally the waste generator attempts to minimize the void volume within the containers. For purposes of determining the performance and technical requirements for disposal, the waste volume increase due to packaging is conservatively neglected -- i.e., volume increase reduces radionuclide concentrations. Moreover, there is very little applicable data available on the packaging efficiency of waste streams. The uncertainties in other estimates in this report partially compensate for exclusion of packaging efficiency from volume calculations. The effect of packaging on transportation and occupational exposures are considered in Volume 3 of this series of reports.

4.3 High Integrity Containers

It has been standard practice in the past to assume no confinement capability following disposal for the containers in which the wastes are shipped. There is little data available, but the data that does exist indicates great variability in the length of time in which the containers retain their form and/or integrity after disposal.

There are many variables that may affect the integrity of currently used waste containers after disposal. These variables include the stability of the waste form (compactibility, resistance to biologic attack, etc.), the void volume of the container (packaging efficiency), the characteristics of the disposal site (natural elements such as precipitation and humidity), the depth of disposal (static soil pressures), and the chemical characteristics of the surrounding soils and wastes (corrosiveness). Because of the many unquantifiable and site specific variables, no attempt has been made in this report to estimate and incorporate a confinement capability for typical containers.

However, the concept of a high-integrity container (HIC) may be considered as an alternative to waste processing as a means of improving the waste form. In this case, the container would be constructed in a much more robust manner than the containers generally used to transport wastes to disposal facilities. The HIC would be designed to resist crushing from static loads and corrosion from the contained wastes as well as the surrounding soils. The HIC would therefore provide the needed support to disposal cell covers to minimize subsidence and to reduce infiltration. In addition, since the wastes would be contained inside the HIC, leaching of radionuclides from the HIC would be negligible as long as the HIC retained its integrity. (Note that corrosion through of a portion of an HIC, which could compromise its ability to withstand leaching, would not be expected to generally reduce its ability to provide structural support for the disposal cell covers). Another advantage to use of an HIC is that, compared with solidification, it would be easier to assure quality control over the final waste product.

Since HIC's have not been extensively used for packaging wastes for disposal there is less data with which to compare other impact measures such as costs or occupational exposures. These, however, may be discussed in a qualitative manner using solidification of LWR ion-exchange resins and filter media as an example. Use of an HIC would be expected to be more expensive than merely dewatering the resins and filter media but less expensive than solidification. This is because no new equipment would need to be installed at the waste generator's facility. Additional expenses would involve construction and certification of HIC's since unlike solidification, there would be no increase in waste volume using HIC's. Transportation costs and disposal costs would therefore be lower than the solidified case. Occupational exposures from waste processing operations at the waste generator would not be expected to vary significantly from those received during management of LWR process wastes under existing practices. The same types of waste handling, processing, transport and disposal operations would be carried out; one is merely substituting one container design for another. Finally, unlike solidification, there would be no decrease in disposal facility land use efficiency compared with the dewatered case. The energy use would also probably be lower than for the solidified case.

Use of HIC's, as an alternative to solidification of ion-exchange resins and filter media, is allowed by the South Carolina Department of Health and Environmental Control, the State agency regulating disposal waste at the Barnwell, S.C. disposal facility. Performance criteria for HIC's for the Barnwell facility have been drafted by South Carolina and these are listed in Table 4-4.

One HIC design which has been recently approved by the South Carolina Department of Health and Environmental Control is currently being marketed. The HIC is constructed principally of polyethylene and is currrently available in designs ranging from 2.4 m^3 (84 ft^3) to 9 m^3 (316 ft^3). Given adequate lead time for fabricating, special designs are advertised as being available upon request. Costs for a HIC are company proprietary information, but are estimated to run approximately 75% to 85% higher than an equivalently sized carbon steel liner.(9)

TABLE 4-4 . State of South Carolina Criteria for High Integrity Containers

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The general criteria for high integrity containers to be used for high concentration waste forms is as follows:

(1) The container must be capable of maintaining its contents until the radionuclides have decayed approximately 300 years, since two of the major isotopes of concern in this respect are Strontium-90 and Cesium 137 with half-lives of 28 and 30 years, respectively.

(2) The structural characteristics of the container with its contents must be adequate to withstand all the pressure and stress it will encounter during all handling, lifting, loading, offloading, backfilling, and burial.

(3) The container must not be susceptible to chemical, galvanic or other reactions from its contents or from the burial environment.

(4) The container must not deteriorate when subjected to the elevated temperatures of the waste streams themselves, from processing materials inside the container, or during storage, transportation and burial.

(5) The container must not be degraded or its characteristics diminished by radiation emitted from its contents, the burial trench or the sun during storage.

(6) All lids, fittings and closures must be of equivalent materials and construction to meet all of the above requirements and must be completely sealed to prevent any loss of the container contents.

Source: Chem-Nuclear Systems, Inc., "High Integrity Container Systems," November 17, 1980.

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5.0 ALTERNATIVE WASTE SPECTRA

This section describes the four waste spectra that may be utilized to help determine the technical requirements for acceptable disposal of LLW. The concept "spectrum" as used here denotes the total volume and properties of waste streams (36 streams given in Table 3-1) generated between the years 1980 and 2000 after they have been processed by a set of selected waste treatment options. Each spectrum corresponds to a general level of waste performance in terms of waste stability, resistance to wind mobilization, resistance to leaching, and physical, chemical, and radiological properties that can be achieved by establishing operational and/or administrative requirements. The spectra differ significantly in waste volumes, radioactive concentrations, and performance.

General descriptions of the four waste spectra and corresponding waste processing options are presented in Section 5.1, and the data file components are discussed in Section 5.2. The treated waste volumes for these spectra are detailed in Section 5.3.

5.1 Waste Spectra Descriptions

The radioactive concentrations of each waste stream for each spectrum depends on the change in the volume of the stream during processing. Whenever a process is applied to a waste stream that results in a volume reduction, its concentrations are increased accordingly. Similarly, whenever a process is applied that results in a volume increase, the concentrations are decreased accordingly. The minute quantities of radionuclides that are lost during these processes (e.g., the radionuclides may become attached to the processing vessel walls) have been conservatively neglected.

As stated, the four waste spectra are used to consider the range in waste performance which can be achieved through alternative operational and/or administrative requirements. The general assumptions made in these spectra are presented below.

5.1.1 Waste Spectrum 1

This spectrum assumes a continuation of past or existing waste management practices. Some of the LWR wastes -- namely the P-CONCLIQ, B-CONCLIQ, and L-DECONRS waste streams -- are solidified . However, no processing is performed for combustible wastes or streams containing chelating agents or organic chemicals. The following general assumptions are made:

- LWR resins and filter sludges are assumed to be shipped to disposal facilities in a dewatered form.
- LWR concentrated liquids are assumed to be concentrated in accordance with current practices, and are solidified in accordance with solidification scenario A.
- No special effort is made to compact trash.
- Institutional waste streams are shipped to disposal sites after they are packaged in currently utilized absorbent materials.
- Resins from LWR decontamination operations (L-DECONRS stream) are solidified in a synthetic polymer (solidification scenario C).

5.1.2 Waste Spectrum 2

This spectrum assumes that LWR process wastes are solidified using improved solidification techniques (solidification scenario B). LWR concentrated liquids are additionally reduced in volume through an evaporator/crystallizer. Routine compaction is performed on all compactible trash. For certain streams (see below), half of the trash volume is compacted at the facility generating the waste and the other half at a centralized processing facility. The following general assumptions are made:

- o All LWR concentrated liquids are evaporated to 50 weight percent solids, and all LWR process wastes are solidified using solidification scenario B. In the case of cartridge filters, the solidification agent fills the voids in the waste packaged but does not increase the volume.
- o Liquid scintillation vials are crushed at large facilities and packed in absorbent material.
- o All compactible trash streams are compacted; P-COTRASH, B-COTRASH, F-COTRASH, I-COTRASH, N-SSTRASH, and N-LOTRASH are compacted at the source of generation; and I+COTRASH, N+SSTRASH, and N+LOTRASH are compacted at a centralized regional processing facility.
- o Liquids from medical isotope production are solidified using solidification scenario C and stabilized using improved packaging techniques.
- o Waste streams having large amounts of activated metal (P-NCTRASH, B-NCTRASH, N-HIGHACT, and L-NFRCOMP) are stabilized using improved packaging techniques.

5.1.3 Waste Spectrum 3

In this spectrum, LWR process wastes are solidified assuming that further improved waste solidification agents are used (solidification scenario C). LWR concentrated liquids are first evaporated to 50 weight percent solids. All possible incineration of combustible material (except LWR process wastes) is performed. Some incineration is done at the source of generation (fuel cycle trash, LWR decontamination resins, institutional wastes from large facilities and industrial trash from large facilities), and some at a centralized regional processing facility (institutional and industrial trash from small facilities). All incineration ash is solidified using solidification scenario C. Medical isotope production wastes and activated metal wastes are again stabilized.

5.1.4 Waste Spectrum 4

This spectrum assumes extreme volume reduction. All wastes amenable to evaporation or incineration with fluidized bed technology are calcined and solidified using solidification scenario C; LWR process wastes, except cartridge filters, are calcined in addition to the streams incinerated in Spectrum 3. All noncompactible wastes are reduced in volume at a central regional processing facility using a large hydraulic press. This spectrum represents the maximum volume reduction that can currently be practically achieved.

5.1.5 Decayed Waste Concentrations

For the analysis required to determine the performance and technical requirements for acceptable disposal of the wastes, and to determine the environmental impacts of selected alternatives, two different sets of radioisotopic concentrations are utilized: (1) undecayed waste concentrations -- presented in Chapter 3.0 (see Tables 3-9 through 3-12) -- which are applicable for determining operational impacts associated with disposal and inadvertant intruder impacts after the closure of the facility, and (2) the decayed isotopic concentrations -- considered in this section -- which are more appropriate for determining the impacts resulting from cases involving interaction of the entire disposed waste with the environment -- e.g., groundwater migration and exposed waste scenarios. (See Volume 3 of this series of reports.)

In these cases, when the entire activity disposed at the facility interacts with the environment, the wastes disposed throughout the facility operational life must be considered. That is, the above spectra include wastes generated over a period of 21 years, and at the end of this period the concentrations of shorter half-life isotopes in wastes generated during the year 1980 will be significantly reduced from as the concentrations of the same isotopes in the wastes generated during the year 2000. One convenient way to incorporate the effects resulting from the different generation times of wastes is to calculate average concentrations at year 2000 which consider the decay of the isotopes from the time of their generation to the year 2000. This is accomplished by the following procedure:

(1) Calculating the projected untreated volumes generated during each year for each waste stream utilizing the information presented in Chapter 3.0 and Appendix A,

(2) Obtaining the total activity of each radionuclide by multiplying these volumes with the untreated waste concentrations presented in Tables 3-9 through 3-12,

(3) Multiplying this total activity with an appropriate (radionuclide-specific) decay factor to yield the total activity in year 2000,

(4) Summing these modified total waste stream activities to obtain the total activity in year 2000 for each stream and radionuclide, and

(5) Dividing this sum by the total untreated waste volumes to obtain average decayed radionuclide concentrations in year 2000.

These modified concentrations are presented in Tables 5-1 through 5-4.

5.2 Waste Spectrum Data File Components

For each of the four waste spectra, a data file was constructed consisting of four major groups of waste form and packaging parameters:

- Volume reduction and volume increase factors;
- Waste form behavior indices (six indices total),
- Waste processing procedures; and
- Waste packaging and transportation indices.

TABLE 5-1

GROUP 1

ISOTOPIC CONCENTRATIONS (CI/M**3)

	P-IXRESIN	P-CONCLIQ	P-FSLUDGE	P-FCARTRG	B-IXRESIN	B-CONCLIQ	B-FSLUDGE
H-3	1.84E-03	2.39E-03	1.79E-03	7.97E-04	1.34E-02	4.35E-04	8.78E-03
C-14	9.73E-05	1.27E-04	9.54E-05	4.25E-05	1.19E-03	3.89E-05	7.77E-04
FE-55	7.30E-04	7.08E-03	9.67E-02	1.73E-01	2.99E-01	2.39E-02	4.54E-01
NI-59	2.79E-06	2.71E-05	3.71E-04	6.60E-04	9.80E-04	7.85E-05	1.49E-03
CO-60	2.17E-03	2.11E-02	2.88E-01	5.14E-01	7.70E-01	6.15E-02	1.17E+00
NI-63	8.15E-04	7.92E-03	1.08E-01	1.93E-01	2.04E-02	1.63E-03	3.08E-02
NB-94	8.84E-08	8.58E-07	1.17E-05	2.09E-05	3.09E-05	2.48E-06	4.70E-05
SR-90	1.63E-04	2.12E-04	1.59E-04	7.07E-05	3.08E-03	9.97E-05	2.00E-03
TC-99	8.23E-07	1.07E-06	8.03E-07	3.58E-07	7.65E-05	2.50E-06	5.00E-05
I-129	2.44E-06	3.16E-06	2.37E-06	1.06E-06	2.04E-04	6.65E-06	1.33E-04
CS-135	8.23E-07	1.07E-06	8.03E-07	3.58E-07	7.65E-05	2.50E-06	5.00E-05
CS-137	1.86E-02	2.43E-02	1.82E-02	8.12E-03	1.74E+00	5.67E-02	1.13E+00
U-235	4.71E-08	6.15E-08	1.46E-07	3.64E-07	5.33E-08	3.44E-08	3.32E-07
U-238	3.71E-07	4.84E-07	1.15E-06	2.87E-06	4.20E-07	2.71E-07	2.61E-06
NP-237	9.06E-12	1.18E-11	2.81E-11	7.02E-11	1.02E-11	6.61E-12	6.38E-11
PU-238	2.45E-05	4.83E-05	4.49E-05	2.37E-04	7.88E-05	1.39E-04	4.40E-04
PU-239/240	1.82E-05	3.31E-05	1.55E-04	3.80E-04	5.34E-05	9.43E-05	2.36E-04
PU-241	5.63E-04	1.02E-03	4.79E-03	1.18E-02	1.85E-03	3.28E-03	8.20E-03
PU-242	3.99E-08	7.25E-08	3.39E-07	8.34E-07	1.17E-07	2.06E-07	5.18E-07
AM-241	1.85E-05	2.96E-05	2.61E-04	1.62E-04	2.29E-05	1.19E-04	1.54E-04
AM-243	1.26E-06	2.02E-06	1.78E-05	1.10E-05	1.57E-06	8.09E-06	1.05E-05
CM-243	8.52E-09	1.01E-08	2.66E-07	1.66E-07	2.33E-08	2.23E-07	2.56E-07
CM-244	1.06E-05	1.47E-05	1.36E-04	8.44E-05	1.40E-05	1.58E-04	1.72E-04

TABLE 5-2

GROUP 2

ISOTOPIC CONCENTRATIONS (CI/M**3)

	P-COTRASH	P-NCTRASH	B-COTRASH	B-NCTRASH	F-COTRASH	F-NCTRASH	I-COTRASH	N-SSTRASH	N-LOTRASH
H-3	2.11E-04	4.84E-03	4.70E-05	7.60E-03	0.	0.	5.95E-02	0.	1.86E-02
C-14	1.12E-05	2.57E-04	4.17E-06	6.72E-04	0.	0.	5.25E-03	0.	1.64E-03
FE-55	1.36E-03	4.27E-02	1.89E-03	3.05E-01	0.	0.	0.	0.	0.
NI-59	7.11E-06	1.64E-04	6.21E-06	1.00E-03	0.	0.	0.	0.	0.
CO-60	5.52E-03	1.27E-01	4.89E-03	7.84E-01	0.	0.	4.41E-03	0.	1.38E-03
NI-63	2.07E-03	4.7[illegible]-02	1.29E-04	2.08E-02	0.	0.	0.	0.	0.
NB-94	2.25E-07	5.18E-06	1.96E-07	3.16E-05	0.	0.	0.	0.	0.
SR-90	1.87E-05	4.30E-04	1.07E-05	1.73E-03	0.	0.	1.19E-03	0.	3.71E-04
TC-99	9.42E-08	2.17E-06	2.68E-07	4.33E-05	0.	0.	3.39E-09	0.	1.06E-09
I-129	2.78E-07	6.41E-06	7.14E-07	1.[illegible]5E-04	0.	0.	0.	0.	0.
CS-135	9.42E-08	2.17E-06	2.68E-07	4.33E-05	0.	0.	0.	0.	0.
CS-137	2.14E-03	4.92E-02	6.09E-03	9.81E-01	0.	0.	3.78E-03	0.	1.18E-03
U-235	7.89E-09	1.82E-07	1.22E-09	1.97E-07	1.18E-06	1.13E-06	0.	2.3[illegible]-06	0.
U-238	6.22E-08	1.43E-06	9.60E-09	1.55E-06	4.40E-06	4.20E-06	0.	8.[illegible]-06	0.
NP-237	1.52E-12	3.49E-11	2.35E-13	3.78E-11	0.	0.	0.	0.	0.
PU-238	5.64E-06	1.30E-04	2.17E-06	3.51E-04	0.	0.	0.	0.	0.
PU-239/240	5.53E-06	1.27E-04	1.16E-06	1.[illegible]6E-04	0.	0.	0.	0.	0.
PU-241	1.71E-04	3.93E-03	4.01E-05	6.47E-03	0.	0.	0.	0.	0.
PU-242	1.21E-08	[illegible].7[illegible]E-0[illegible]	2.53E-09	4.08E-07	0.	0.	0.	0.	0.
AM-241	3.92E-06	9.02E-05	9.56E-07	1.54E-04	0.	0.	4.76E-06	0.	1.49E-06
AM-243	2.67E-07	6.14E-06	6.51E-08	1.05E-05	0.	0.	0.	0.	0.
CM-243	2.35E-09	5.41E-08	1.66E-09	2.69E-07	0.	0.	0.	0.	0.
CM-244	2.00E-06	4.60E-05	1.15E-06	1.86E-04	0.	0.	0.	0.	0.

TABLE 5-3

```
GROUP  3

ISOTOPIC CONCENTRATIONS  (CI/M**3)
            F-PROCESS U PROCESS I-LQSCNVL I-ABSLIQD I-BIOWAST N-SSWASTE N-LGWASTE
H-3            0.         0.    3.27E-03  9.26E-02  1.14E-01     0.     1.06E-02
C-14           0.         0.    2.51E-04  8.15E-03  1.01E-02     0.     9.35E-04
FE-55          0.         0.       0.        0.        0.        0.        0.
NI-59          0.         0.       0.        0.        0.        0.        0.
CO-60          0.         0.       0.     1.32E-02  1.69E-0[illegible]     0.     6 23E-04
NI-63          0.         0.       0.        0.        0.        0.        0.
NB-94          0.         0.       0.        0.        0.        0.        0
SR-90          0.         0.    3.55E-03  3.55E-03  6.82[illegible]-03     0.     1.[illegible]7 03
TC-99          0.         0.       0.     1.02E-08  6.51[illegible]-09     0.     7.7[illegible]-1
I-129          0.         0.       0.        0.        0.        0.        0.
CS-135         0.         0.       0.        0.        0.        0.        0.
CS-137         0.         0.       0.     1.14E-02  7.26E-03     0.     8.62E-04
U-235       2.30E-05  1.65E-05     0.        0.        0.     [illegible].60E-05     0.
U-238       8.54E-05  3.64E-04     0.        0.        0.     1.71E-04     0.
NP-237         0.         0.       0.        0.        0.        0.        0.
PU-238         0.         0.       0.        0.        0.        0.        0.
PU-239/240     0.         0.       0.        0.        0.        0.        0.
PU-241         0.         0.       0.        0.        0.        0.        0.
PU-242         0.         0.       0.        0.        0.        0.        0.
AM-241         0.         0.       0.        0.        0.        0.        0.
AM-243         0.         0.       0.        0.        0.        0.        0.
CM-243         0.         0.       0.        0.        0.        0.        0.
CM-244         0.         0.       0.        0.        0.        0.        0.
```

TABLE 5-4

GROUP 4

ISOTOPIC CONCENTRATIONS (CI/M**3)

	L-NFRCOMP	L-DECONRS	N-ISOPROD	N-HIGHACT	N-TRITIUM	N-SOURCES	N-TARGETS
H-3	0.	7.51E-03	2.74E-02	0.	1.52E+03	5.63E+02	5.24E+01
C-14	2.59E-01	5.12E-04	4.51E-05	1.32E-02	0.	5.75E+01	0.
FE-55	6.98E+02	1.27E+01	0.	2.97E+01	0.	0.	0.
NI-59	1.40E+00	4.49E-02	0.	6.56E-02	0.	0.	0.
CO-60	7.70E+02	3.50E+01	0.	3.60E+01	0.	7.34E+02	0.
NI-63	1.98E+02	3.49E+00	0.	9.95E+00	0.	2.16E+02	0.
NB-94	8.19E-03	1.42E-03	0.	4.47E-04	0.	0.	0.
SR-90	0.	3.61E-02	5.14E+00	0.	0.	9.42E+02	0.
TC-99	0.	1.20E-05	3.27E-04	0.	0.	0.	0.
I-129	0.	3.34E-05	2.72E-06	0.	0.	0.	0.
CS-135	0.	1.20E-05	3.27E-04	0.	0.	0.	0.
CS-137	0.	2.71E-01	7.24E+00	0.	0.	9.53E+02	0.
U-235	0.	6.84E-05	1.02E-05	0.	0.	0.	0.
U-238	0.	5.40E-04	3.81E-05	0.	0.	0.	0.
NP-237	0.	1.32E-08	5.33E-13	0.	0.	0.	0.
PU-238	0.	1.26E+00	1.84E-04	0.	0.	0.	0.
PU-239/240	0.	1.77E+00	5.55E-05	0.	0.	0.	0.
PU-241	0.	2.52E+01	4.75E-03	0.	0.	0.	0.
PU-242	0.	3.87E-03	9.57E-08	0.	0.	0.	0.
AM-241	0.	5.23E-03	1.09E-05	0.	0.	5.69E+02	0.
AM-243	0.	3.59E-04	1.25E-06	0.	0.	0.	0.
CM-243	0.	2.98E-04	1.38E-04	0.	0.	0.	0.
CM-244	0.	2.51E-03	2.11E-07	0.	0.	0.	0.

The first three groups of parameters are discussed in this section. The last group of indices are described in Volume 3 of this series of reports.

5.2.1 Volume Reduction and Volume Increase Factors

These factors were previously introduced in Chapter 4.0. The volume reduction factor (VRF) is the ratio of the volume of the untreated input waste to the volume of the treated waste product. It is used in describing the effects of the volume reduction processes discussed in Section 4.1.

The volume increase factor (VIF) is defined as the ratio of the volume of the product waste stream to the volume of the input waste stream. It is used in describing the effects of the volume increase processes discussed in Section 4.2.

Additional information concerning the volume reduction and volume increase factors of the waste processes selected for the alternative waste spectra are provided in Appendices C and D. The volume reduction and the volume increase factors assumed for the waste streams for each waste spectra are presented in Table 5-5. The volume increase factors are derived from waste/binder weight ratios given in Appendix D. Waste/binder weight ratios were selected to maximize waste form performance rather than to minimize volume increase. Volume increase factors for solidification scenarios A and B are averages of those of the individual binders used.

5.2.2 Waste Form Behavior Indices

The characteristics important in determining the impacts resulting from management and disposal of waste include the flammability of the waste form at the time of disposal, the dispersibility of the waste form several decades after disposal, the structural stability of the

TABLE 5-5 . VOLUME REDUCTION AND INCREASE FACTORS

STREAM	SPECTRUM VRF	 VIF	SPECTRUM 2 VRF	 VIF	SPECTRUM 3 VRF	 VIF	SPECTRUM 4 VRF	 VIF
P-IXRESIN	1.00	1.00	1.00	1.65	1.00	2.00	18.00	2.00
P-CONCLIQ	1.00	1.40	6.00	1.82	6.00	2.00	8.00	2.00
P-FSLUDGE	1.00	1.00	1.00	1.65	1.00	2.00	5.00	2.00
P-FCARTRG	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
B-IXRESIN	1.00	1.00	1.00	1.65	1.00	2.00	18.00	2.00
B-CONCLIQ	1.00	1.40	2.40	1.56	2.40	2.00	6.40	2.00
B-FSLUDGE	1.00	1.00	1.00	1.65	1.00	2.00	5.00	2.00
P-COTRASH	1.00	1.00	2.00	1.00	80.00	2.00	80.00	2.00
P-NCTRASH	1.00	1.00	1.00	1.00	1.00	1.00	6.00	1.00
B-COTRASH	1.00	1.00	2.00	1.00	80.00	2.00	80.00	2.00
B-NCTRASH	1.00	1.00	1.00	1.00	1.00	1.00	6.00	1.00
F-COTRASH	1.00	1.00	1.50	1.00	40.00	2.00	40.00	2.00
F-NCTRASH	1.00	1.00	1.00	1.00	1.00	1.00	6.00	1.00
I-COTRASH	1.00	1.00	2.00	1.00	20.00	2.00	20.00	2.00
I+COTRASH	1.00	1.00	4.00	1.00	80.00	2.00	80.00	2.00
N-SSTRASH	1.00	1.00	1.50	1.00	10.00	2.00	10.00	2.00
N+SSTRASH	1.00	1.00	3.00	1.00	40.00	2.00	40.00	2.00
N-LOTRASH	1.00	1.00	2.00	1.00	20.00	2.00	20.00	2.00
N+LOTRASH	1.00	1.00	4.00	1.00	80.00	2.00	80.00	2.00
F-PROCESS	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
U-PROCESS	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
I-LQSCNVL	1.00	3.00	1.28	3.00	4.52	2.00	4.52	2.00
I+LQSCNVL	1.00	3.00	1.00	3.00	1.00	3.00	1.00	3.00
I-ABSLIQD	1.00	3.00	1.00	1.65	1.00	2.00	100.00	2.00
I+ABSLIQD	1.00	3.00	1.00	3.00	1.00	3.00	1.00	3.00
I-BIOWAST	1.00	1.92	1.00	1.92	15.00	2.00	15.00	2.00
I+BIOWAST	1.00	1.92	1.00	1.92	1.00	1.92	1.00	1.92
N-SSWASTE	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
N-LOWASTE	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
L-NFRCOMP	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
L-DECONRS	1.00	2.00	1.00	2.00	18.00	2.00	18.00	2.00
N-ISOPROD	1.00	1.30	1.00	2.00	1.00	2.00	1.00	2.00
N-HIGHACT	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
N-TRITIUM	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
N-SOURCES	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
N-TARGETS	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

waste form, the resistance of the waste form to leaching, the chemical content of the waste, and the accessibility of the radionuclides in the waste to transfer agents such as wind or water. These six properties were quantified through six waste form behavior indices defined in Table 5-6 and discussed below.

The flammability index ranks waste forms according to their flammability prior to disposal. Waste forms which will not burn even on prolonged exposure to open flame and moderately intense heat are assigned an index of (0). Those waste forms that will sustain combustion are assigned an index of (3). Between these extremes are two additional flamability categories. Waste forms which will ignite but will not sustain burning under these conditions are assigned an index of (2). Waste forms consisting of a mixture of materials with flammability indices (0) and (2) (e.g., solidification scenarios A and B) are assigned an index of (1).

The dispersibility index is a qualitative measure of the potential for suspension of radioactivity should the waste form be exposed to wind or to human activity after a significant period (on the order of 100 years). Waste forms which are estimated to have a low probability of becoming suspended are assigned an index of (0). Those waste forms which have a high potential of becoming suspended are assigned an index of (3). Waste forms which tend to crumble or fracture extensively and those that are subject to relatively rapid (within about 100 years) decomposition are assigned an index of (2). Waste forms consisting of a mixture of materials with dispersibility indices of (0) and (2) are assigned an index of (1).

The leachability index is a qualitative measure of the waste form's resistance to leaching and is determined by the solidification procedures used. Unsolidified waste forms, which are assumed to be readily leached, are assigned an index of (1). Solidification scenarios A, B, and C (discussed in the previous section) are assigned an index of 2, 3, and 4, respectively.

TABLE 5-6 . Waste Form Behavior Indices

Parameter and Symbol	Indices
Flammability (F)	0 = non-flammable
	1 = low flammability (mixture of material with indices of 0 and 2)
	2 = burns if heat supplied (does not support burning)
	3 = flammable (supports burning)
Dispersibility (D)	0 = near zero
	1 = slight to moderate
	2 = moderate
	3 = severe
Leachability (L)	1 = unsolidified waste form
	2 = solidification scenario A
	3 = solidification scenario B
	4 = solidification scenario C
Chemical Content (C)	0 = no chelating agents or organic chemicals
	1 = chelating agents or organic chemicals are likely to be present in the waste form
Stability (S)	0 = structurally unstable waste form
	1 = structurally stable waste form
Accessibility (A)	1 = readily accessible
	2 = moderately accessible
	3 = accessible with difficulty

The chemical content index denotes whether the waste form may contain chelating agents or organic chemicals that increase the mobility of radionuclides during and/or after leaching. An index value of (0) indicates a likelihood that these chemicals or agents are absent, and an index value of (1) indicates a likelihood of their presence.

The stability index denotes whether the waste form is likely to reduce in volume after disposal due to compressibility, large internal void volumes, and/or chemical and biological attack (no credit is taken for the waste containers). An index value of (0) indicates a likelihood of structural instability, whereas a value of (1) indicates a structurally stable waste form.

The last index, the accessibility index, is a correction factor for contaminated metals, and ranks the waste forms according to the accessibility of the radionuclides to transfer agents such as wind and water. Surface contaminated wastes and waste containing radioactivity in readily soluble forms are assigned an index of (1). The waste forms that are almost exclusively activated metals with imbedded radioactivity not readily accessible to the elements are assigned an index of (3). Other waste forms (e.g., non-compactible trash which contains a lot of equipment) are assigned an index of (2).

A single waste property may determine the value of more than one index and a single performance characteristic may be described by more than one index. For example, in Spectra 1 and 2, the tendency of combustible materials in the trash waste streams to decompose contributes to both the dispersibility and the instability of these streams. On the other hand, the ability of a waste form to retain the radioactivity it contains is described by both its leachability and its accessibility index. In this case, leachability is based on the properties of the waste binder (solidification agent) while accessibility is based on the properties of the waste itself. Waste behavior indices that have been assumed for the four waste spectra considered are presented in Table 5-7.

TABLE 5-7 . Waste Form Behavior Index Values

	Waste Spectrum 1						Waste Spectrum 2						Waste Spectrum 3						Waste Spectrum 4					
	F	D	L	C	S	A	F	D	L	C	S	A	F	D	L	C	S	A	F	D	L	C	S	A
P-IXRESIN	2	1	1	0	0	1	1	1	3	0	1	1	2	0	4	0	1	1	1	0	4	0	1	1
P-CONCLIQ	1	1	2	0	1	1	1	1	3	0	1	1	2	0	4	0	1	1	1	0	4	0	1	1
P-FSLUDGE	1	3	1	0	0	1	1	1	3	0	1	1	2	0	4	0	1	1	1	0	4	0	1	1
P-FCARTRG	2	2	1	0	0	1	1	1	3	0	1	1	2	0	4	0	1	1	2	0	4	0	1	1
B-IXRESIN	2	1	1	0	0	1	1	1	3	0	1	1	2	0	4	0	1	1	1	0	4	0	1	1
B-CONCLIQ	1	1	2	0	1	1	1	1	3	0	1	1	2	0	4	0	1	1	1	0	4	0	1	1
B-FSLUDGE	1	3	1	0	0	1	1	1	3	0	1	1	2	0	4	0	1	1	1	0	4	0	1	1
P-COTRASH	3	2	1	0	0	1	3	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
P-NCTRASH	0	0	1	0	0	2	0	0	1	0	1	2	0	0	1	0	1	2	0	0	1	0	1	2
B-COTRASH	3	2	1	0	0	1	3	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
B-NCTRASH	0	0	1	0	0	2	0	0	1	0	1	2	0	0	1	0	1	2	0	0	1	0	1	2
F-COTRASH	3	2	1	0	0	1	3	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
F-NCTRASH	0	0	1	0	0	2	0	0	1	0	0	2	0	0	1	0	0	2	0	0	1	0	0	2
I-COTRASH	3	2	1	0	0	1	3	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
I+COTRASH	3	2	1	0	0	1	3	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
N-SSTRASH	2	2	1	0	0	1	2	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
N+SSTRASH	2	2	1	0	0	1	2	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
N-LOTRASH	3	2	1	0	0	1	3	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
N+LOTRASH	3	2	1	0	0	1	3	2	1	0	0	1	1	0	4	0	1	1	1	0	4	0	1	1
F-PROCESS	0	3	1	0	1	1	0	3	1	0	1	1	0	3	1	0	1	1	0	3	1	0	1	1
U-PROCESS	0	3	1	0	1	1	0	3	1	0	1	1	0	3	1	0	1	1	0	3	1	0	1	1
I-LIQSCVL	3	3	1	1	0	1	3	3	1	1	0	1	1	0	4	0	1	1	1	0	4	0	1	1
I+LIQSCVL	3	3	1	1	0	1	3	3	1	1	0	1	3	3	1	1	0	1	3	3	1	1	0	1
I-ABSLIQD	3	3	1	1	1	1	3	1	3	1	1	1	1	0	4	0	1	1	1	0	4	0	1	1
I+ABSLIQD	3	3	1	1	1	1	3	3	1	1	1	1	3	3	1	1	1	1	3	3	1	1	1	1
I-BIOWAST	2	3	1	1	0	1	2	3	1	1	0	1	1	0	4	0	1	1	1	0	4	0	1	1
I+BIOWAST	2	3	1	1	0	1	2	3	1	1	0	1	2	3	1	1	0	1	2	3	1	1	0	1
N-SSWASTE	0	3	1	0	1	1	0	3	1	0	1	1	0	3	1	0	1	1	0	3	1	0	1	1
N-LOWASTE	3	3	1	1	0	1	3	3	1	1	0	1	3	3	1	1	0	1	3	3	1	1	0	1
L-NFRCOMP	0	0	1	0	0	2	0	0	1	0	1	2	0	0	1	0	1	2	0	0	1	0	1	2
L-DECONRS	2	0	4	1	1	1	2	0	4	1	1	1	1	0	4	0	1	1	1	0	4	0	1	1
N-ISOPROD	1	1	3	1	0	1	1	0	4	1	1	1	1	0	4	1	1	1	1	0	4	1	1	1
N-HIGHACT	0	0	1	0	0	3	0	0	1	0	1	3	0	0	1	0	1	3	0	0	1	0	1	3
N-TRITIUM	3	3	1	1	1	1	3	3	1	1	1	1	3	3	1	1	1	1	3	3	1	1	1	1
N-SOURCES	0	0	1	0	1	2	0	0	1	0	1	2	0	0	1	0	1	2	0	0	1	0	1	2
N-TARGETS	0	0	1	0	1	1	0	0	1	0	1	1	0	0	1	0	1	1	0	0	1	0	1	1

5.2.3 Processing Indices

Processing impacts in addition to those associated with treatment operations performed in Spectrum 1 include occupational and population exposures, costs, and energy use. Population impacts from processing depend primarily on the radioactive contents of the waste streams and secondarily on the location at which the processing takes place. Only incineration (pathological incinerators and incinerator/calciners) is assumed to result in a release of radioactivity which could result in significant additional population exposures. Occupational exposures depend on the environment in which the waste processing is being performed in addition to the waste activity. The costs of waste processing also depends on the size of the facility as well as the specific process being utilized.

In order to account for these variations, four indices have been assigned to each waste stream in each spectrum and are utilized in the calculation of waste processing impacts. These indices are summarized in Table 5-8, and the values assigned for these indices for all the waste streams and the waste spectra being considered in this report are presented in Table 5-9. More information on the calculation of the waste processing impacts can be found in Volume 3 of this series of reports.

5.3 Treated Waste Volumes

The total waste volumes after processing for each of the waste spectra for the entire United States between the years 1980 and 2000 are presented in Table 5-10. After the computation of these volumes, Spectrum 1 is normalized to 1 million cubic meters for purposes of determining performance and technical requirements for acceptable disposal of LLW. This analysis allows consideration of a generic, nationwide source term based on normalizing the total U.S. volume to one million cubic meters. The subsequent waste spectra volumes are computed and are presented in Table 5-11.

TABLE 5-8 . Waste Processing Indices

	Value	Meaning
First Digit - IPR	0	No volume Reduction
	1	Regular Compaction
	2	Improved Compaction
	3	Hydraulic Press
	4	Evaporation
	5	Pathological Incineration
	6	Small Calciner
	7	Large Calciner
Second Digit - ISL	0	No Solidification
	1	Solidification Scenario A
	2	Solidification Scenario B
	3	Solidification Scenario C
Third Digit - ILC	0	No Processing
	1	Processing at the Generator
	2	Processing at the Disposal Site
Fourth Digit - IEN	0	No Incineration
	1	Urban Environment
	2	Rural Environment

TABLE 5-9 . Waste Processing Indices

	Waste Spectrum 1				Waste Spectrum 2				Waste Spectrum 3				Waste Spectrum 4			
	IPR	ISL	ILC	IEN	IPR	ISL	ILC	IEN	IPR	ISL	ILC	IEN	IPR	ISL	ILC	IEN
P-IXRESIN	0	0	1	0	0	2	1	0	0	3	1	0	6	3	1	2
P-CONCLIQ	0	1	1	0	4	2	1	0	4	3	1	0	6	3	1	2
P-FSLUDGE	0	0	1	0	0	2	1	0	0	3	1	0	6	3	1	2
P-FCARTRG	0	1	1	0	0	2	1	0	0	3	1	0	0	3	1	0
B-IXRESIN	0	0	1	0	0	2	1	0	0	3	1	0	6	3	1	2
B-CONCLIQ	0	1	1	0	4	2	1	0	4	3	1	0	6	3	1	2
B-FSLUDGE	0	0	1	0	0	2	1	0	0	3	1	0	6	3	1	2
P-COTRASH	0	0	0	0	1	0	1	0	6	3	1	2	6	3	1	2
P-NCTRASH	0	0	0	0	0	0	0	0	0	0	0	0	3	0	1	0
B-COTRASH	0	0	0	0	1	0	1	0	6	3	1	2	6	3	1	2
B-NCTRASH	0	0	0	0	0	0	0	0	0	0	0	0	3	0	1	0
F-COTRASH	0	0	0	0	1	0	1	0	6	3	1	2	6	3	1	2
F-NCTRASH	0	0	0	0	0	0	0	0	0	0	0	0	3	0	2	0
I-COTRASH	0	0	0	0	1	0	1	0	5	3	1	1	5	3	1	1
I+COTRASH	0	0	0	0	2	0	2	0	7	3	2	2	7	3	2	2
N-SSTRASH	0	0	0	0	1	0	1	0	5	3	1	1	5	3	1	1
N+SSTRASH	0	0	0	0	2	0	2	0	7	3	2	2	7	3	2	2
N-LOTRASH	0	0	0	0	1	0	1	0	5	3	1	1	5	3	1	1
N+LOTRASH	0	0	0	0	2	0	2	0	7	3	2	2	7	3	2	2
F-PROCESS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
U-PROCESS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
I-LIQSCVL	0	0	1	0	1	0	1	0	5	3	1	1	5	3	1	1
I+LIQSCVL	0	0	1	0	0	0	1	0	0	0	1	0	0	0	1	0
I-ABSLIQD	0	0	1	0	0	2	1	0	0	3	1	0	5	3	1	1
I+ABSLIQD	0	0	1	0	0	0	1	0	0	0	1	0	0	0	1	0
I-BIOWAST	0	0	1	0	0	0	1	0	5	3	1	1	5	3	1	1
I+BIOWAST	0	0	1	0	0	0	1	0	0	0	1	0	0	0	1	0
N-SSWASTE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
N-LOWASTE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
L-NFRCOMP	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
L-DECONRS	0	3	1	0	0	3	1	0	6	3	1	2	6	3	1	2
N-ISOPROD	0	2	1	0	0	3	1	0	0	3	1	0	0	3	1	0
N-HIGHACT	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
N-TRITIUM	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
N-SOURCES	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
N-TARGETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

TABLE 5-10. CUMULATIVE VOLUMES (M**3)

STREAM	SPECTRUM 1 VOLUME	SPECTRUM 1 % VOL	SPECTRUM 2 VOLUME	SPECTRUM 2 % VOL	SPECTRUM 3 VOLUME	SPECTRUM 3 % VOL	SPECTRUM 4 VOLUME	SPECTRUM 4 % VOL
P-IXRESIN	3.46E+04	.96	5.71E+04	2.26	6.93E+04	3.88	3.85E+03	.45
P-CONCLIQ	3.41E+05	9.43	7.38E+04	2.92	8.12E+04	4.55	6.09E+04	7.08
P-FSLUDGE	4.28E+03	.12	7.06E+03	.28	8.56E+03	.48	1.71E+03	.20
P-FCARTRG	2.18E+04	.60	2.18E+04	.86	2.18E+04	1.22	2.18E+04	2.53
B-IXRESIN	7.62E+04	2.11	1.26E+05	4.98	1.52E+05	8.54	8.47E+03	.99
B-CONCLIQ	2.94E+05	8.14	1.37E+05	5.41	1.75E+05	9.81	6.57E+04	7.65
B-FSLUDGE	1.69E+05	4.67	2.79E+05	11.04	3.38E+05	18.94	6.76E+04	7.87
P-COTRASH	4.24E+05	11.74	2.12E+05	8.40	1.06E+04	.59	1.06E+04	1.23
P-NCTRASH	2.18E+05	6.02	2.18E+05	8.62	2.18E+05	12.20	3.63E+04	4.22
B-COTRASH	2.09E+05	5.77	1.04E+05	4.13	5.22E+03	.29	5.22E+03	.61
B-NCTRASH	9.90E+04	2.74	9.90E+04	3.92	9.90E+04	5.54	1.65E+04	1.92
F-COTRASH	2.36E+05	6.52	1.57E+05	6.23	1.18E+04	.66	1.18E+04	1.37
F-NCTRASH	4.17E+04	1.15	4.17E+04	1.65	4.17E+04	2.34	6.95E+03	.81
I-COTRASH	1.41E+05	3.89	7.04E+04	2.79	1.41E+04	.79	1.41E+04	1.64
I+COTRASH	1.41E+05	3.89	3.52E+04	1.39	3.52E+03	.20	3.52E+03	.41
N-SSTRASH	1.80E+05	4.97	1.20E+05	4.74	3.59E+04	2.01	3.59E+04	4.18
N+SSTRASH	1.80E+05	4.97	5.99E+04	2.37	8.98E+03	.50	8.98E+03	1.04
N-LOTRASH	5.06E+04	1.40	2.53E+04	1.00	5.06E+03	.28	5.06E+03	.59
N+LOTRASH	5.06E+04	1.40	1.27E+04	.50	1.27E+03	.07	1.27E+03	.15
F-PROCESS	7.82E+04	2.16	7.82E+04	3.09	7.82E+04	4.38	7.82E+04	9.10
U-PROCESS	2.81E+04	.78	2.81E+04	1.11	2.81E+04	1.57	2.81E+04	3.27
I-LQSCNVL	1.47E+05	4.08	1.15E+05	4.56	2.17E+04	1.22	2.17E+04	2.53
I+LQSCNVL	1.47E+05	4.08	1.47E+05	5.84	1.47E+05	8.26	1.47E+05	17.16
I-ABSLIQD	1.68E+04	.46	9.22E+03	.36	1.12E+04	.63	1.12E+02	.01
I+ABSLIQD	1.68E+04	.46	1.68E+04	.66	1.68E+04	.94	1.68E+04	1.95
I-BIOWAST	3.02E+04	.83	3.02E+04	1.19	2.09E+03	.12	2.09E+03	.24
I+BIOWAST	3.02E+04	.83	3.02E+04	1.19	3.02E+04	1.69	3.02E+04	3.51
N-SSWASTE	6.34E+04	1.75	6.34E+04	2.51	6.34E+04	3.55	6.34E+04	7.38
N-LOWASTE	6.03E+04	1.67	6.03E+04	2.39	6.03E+04	3.38	6.03E+04	7.01
L-NFRCOMP	2.89E+03	.08	2.89E+03	.11	2.89E+03	.16	2.89E+03	.34
L-DECONRS	7.00E+04	1.93	7.00E+04	2.77	3.89E+03	.22	3.89E+03	.45
N-ISOPROD	6.75E+03	.19	1.04E+04	.41	1.04E+04	.58	1.04E+04	1.21
N-HIGHACT	2.61E+03	.07	2.61E+03	.10	2.61E+03	.15	2.61E+03	.30
N-TRITIUM	3.48E+03	.10	3.48E+03	.14	3.48E+03	.19	3.48E+03	.41
N-SOURCES	1.87E+02	.01	1.87E+02	.01	1.87E+02	.01	1.87E+02	.02
N-TARGETS	1.34E+03	.04	1.34E+03	.05	1.34E+03	.08	1.34E+03	.16
TOTALS	3.62E+06		2.53E+06		1.79E+06		8.59E+05	

TABLE 5-11 . Normalized Volumes

	SPECTRUM 1		SPECTRUM 2		SPECTRUM 3		SPECTRUM 4	
	VOL	%	VOL	%	VOL	%	VOL	%
P-IXRESIN	9.58E+03	.96	1.58E+04	2.26	1.92E+04	3.88	1.06E+03	.45
P-CONCLIQ	9.43E+04	9.43	2.04E+04	2.92	2.24E+04	4.55	1.68E+04	7.08
P-FSLUDGE	1.18E+03	.12	1.95E+03	.28	2.37E+03	.48	4.73E+02	.20
P-FCARTRG	6.02E+03	.60	6.02E+03	.86	6.02E+03	1.22	6.02E+03	2.53
B-IXRESIN	2.11E+04	2.11	3.48E+04	4.98	4.22E+04	8.54	2.34E+03	.99
B-CONCLIQ	8.14E+04	8.14	3.78E+04	5.41	4.84E+04	9.81	1.82E+04	7.65
B-FSLUDGE	4.67E+04	4.67	7.71E+04	11.04	9.35E+04	18.94	1.87E+04	7.87
P-COTRASH	1.17E+05	11.74	5.87E+04	8.40	2.93E+03	.59	2.93E+03	1.23
P-NCTRASH	6.02E+04	6.02	6.02E+04	8.62	6.02E+04	12.20	1.00E+04	4.22
B-COTRASH	5.77E+04	5.77	2.88E+04	4.13	1.44E+03	.29	1.44E+03	.61
B-NCTRASH	2.74E+04	2.74	2.74E+04	3.92	2.74E+04	5.54	4.56E+03	1.92
F-COTRASH	6.52E+04	6.52	4.35E+04	6.23	3.26E+03	.66	3.26E+03	1.37
F-NCTRASH	1.15E+04	1.15	1.15E+04	1.65	1.15E+04	2.34	1.92E+03	.81
I-COTRASH	3.89E+04	3.89	1.95E+04	2.79	3.89E+03	.79	3.89E+03	1.64
I+COTRASH	3.89E+04	3.89	9.73E+03	1.39	9.73E+02	.20	9.73E+02	.41
N-SSTRASH	4.97E+04	4.97	3.31E+04	4.74	9.93E+03	2.01	9.93E+03	4.18
N+SSTRASH	4.97E+04	4.97	1.66E+04	2.37	2.48E+03	.50	2.48E+03	1.04
N-LOTRASH	1.40E+04	1.40	7.00E+03	1.00	1.40E+03	.28	1.40E+03	.59
N+LOTRASH	1.40E+04	1.40	3.50E+03	.50	3.50E+02	.07	3.50E+02	.15
F-PROCESS	2.16E+04	2.16	2.16E+04	3.09	2.16E+04	4.38	2.16E+04	9.10
U-PROCESS	7.77E+03	.78	7.77E+03	1.11	7.77E+03	1.57	7.77E+03	3.27
I-LQSCNVL	4.08E+04	4.08	3.19E+04	4.56	6.01E+03	1.22	6.01E+03	2.53
I+LQSCNVL	4.08E+04	4.08	4.08E+04	5.84	4.08E+04	8.26	4.08E+04	17.16
I-ABSLIQD	4.63E+03	.46	2.55E+03	.36	3.09E+03	.63	3.09E+01	.01
I+ABSLIQD	4.63E+03	.46	4.63E+03	.66	4.63E+03	.94	4.63E+03	1.95
I-BIOWAST	8.34E+03	.83	8.34E+03	1.19	5.79E+02	.12	5.79E+02	.24
I+BIOWAST	8.34E+03	.83	8.34E+03	1.19	8.34E+03	1.69	8.34E+03	3.51
N-SSWASTE	1.75E+04	1.75	1.75E+04	2.51	1.75E+04	3.55	1.75E+04	7.38
N-LOWASTE	1.67E+04	1.67	1.67E+04	2.39	1.67E+04	3.38	1.67E+04	7.01
L-NFRCOMP	7.98E+02	.08	7.98E+02	.11	7.98E+02	.16	7.98E+02	.34
L-DECONRS	1.93E+04	1.93	1.93E+04	2.77	1.07E+03	.22	1.07E+03	.45
N-ISOPROD	1.87E+03	.19	2.87E+03	.41	2.87E+03	.58	2.87E+03	1.21
N-HIGHACT	7.21E+02	.07	7.21E+02	.10	7.21E+02	.15	7.21E+02	.30
N-TRITIUM	9.63E+02	.10	9.63E+02	.14	9.63E+02	.19	9.63E+02	.41
N-SOURCES	5.16E+01	.01	5.16E+01	.01	5.16E+01	.01	5.16E+01	.02
N-TARGETS	3.71E+02	.04	3.71E+02	.05	3.71E+02	.08	3.71E+02	.16
	1.00E+06		6.99E+05		4.94E+05		2.38E+05	

6.0 OTHER POTENTIAL WASTE STREAMS

This section contains a discussion of waste streams other than the basic streams discussed in Chapters 2.0 and 3.0 and which: (1) are not currently being sent to LLW disposal facilities, (2) are non-routine, or (3) are very speculative in terms of timing or waste generation rates. Wastes which fall into this category include those from:

- Decommissioning of nuclear fuel cycle facilities.
- U.S. Government operations:
- Decontamination of the Three Mile Island Unit 2 nuclear generating station;
- Transuranic-contaminated wastes, including wastes from potential recycle of nuclear fuel;
- Operations at independent spent fuel storage installations;
- Low-level waste resulting from the implementation of the "West Valley Demonstration Project".

These potential waste streams are discussed in the following subsections. Additional information is given in Appendix A.

6.1 Decommissioning of Nuclear Fuel-Cycle Facilities

Nuclear fuel cycle facilities will eventually reach the end of their useful lives and would then be considered candidates for decontamination and decommissioning. In some cases, decontamination and decommissioning activities may merely involve removing enough residual contamination to allow safe modification and reuse as a nuclear facility. In other cases, the facility may be decontaminated to the point that it can be released for unrestricted use.

The timing and extent of potential decontamination and decommissioning activities at a nuclear installation are very speculative at this time. The timing and extent of decommissioning activities may depend upon other factors than the useful life of a nuclear facility -- e.g., upon economic decisions or regulatory requirements. It is considered unlikely that significant volumes of wastes from decommissioning nuclear fuel cycle facilities will be produced prior to the year 2000. Nonetheless, NRC staff (see Appendix A) has investigated the potential volumes, activities, and other characteristics of wastes generated from decommissioning of a number of different types of nuclear fuel cycle facilities, and these volumes and activities can be briefly investigated to help gauge the potential impacts of future waste streams. Waste streams considered include those generated from decommissioning: (1) light water reactors, (2) uranium fuel fabrication plants, (3) uranium fuel recycle facilities.

6.1.1 Decommissioning of Light Water Reactors

A significant source of waste to be generated in the future will be from decommissioning light water power reactors. The volumes and activities which will be produced are speculative to a high degree, and depend upon such factors as the length of service life of a plant prior to decommissioning, the size and design of a plant, the operating history of the facility, and the decommissioning mode undertaken (e.g., immediate dismantlement after shut down vs. deferring dismantlement for up to several years following shut-down).

Pacific Northwest Laboratories (PNL) has recently completed a pair of studies on the technology, safety, and costs of decommissioning a large reference PWR(1) and a large reference BWR.(2) The model for the reference PWR is the Portland General Electric Company Trojan nuclear plant having a generating capacity of 1175 MW(e) (3500 MW(t)), and using a Westinghouse four-loop nuclear steam supply system. The model for the reference BWR is the Washington Public Power System's

Nuclear Project No. 2 (WPPSS-2) at Hanford, Washington. This 1155 MW(e) unit (3320 MW(t)), which is expected to start operation in 1982, uses a General Electric BWR-5 nuclear steam supply system. The plant uses a Mark-II containment.

A summary of the waste volumes and activities estimated by PNL for the two reference LWR's is provided in Table 6-1. The volumes and activities are projected from an assumption of immediate dismantlement following 40 calendar years of operation at 75% of full power, or 30 effective full power years (EFPY). Dismantlement of the reference PWR is projected to require 4 years, while dismantlement of the reference BWR is projected to require 3 1/2 years.

The volumes and activities summarized in Table 6-1 are based upon paper studies rather than actual data and should be interpreted with some care. Actual volumes and activities from decommissioning a given LWR may be highly site specific and a function of such factors as the size and design of the unit, the rated power level, the amount of time spent at full power, and the time between shutdown and dismantlement. However, it is apparent that on the order of 99% of the activity from decommissioning wastes will be contained in activated metal. Relative volumes and activities for various activated metal components are shown in Table 6-2. As shown, specific activities of BWR activated components are estimated to vary by four orders of magnitude, while PWR components vary by six orders of magnitude. Of special interest for disposal purposes are the BWR core shroud and the PWR core shroud and lower grid plate.

Potential volumes of decommissioning wastes generated to the year 2000 are speculative; however, it is not expected that volumes and activities generated (if generated) during this time period will be significant compared to other routinely generated LWR waste streams. In any case, the characteristics of actual waste generated from a particular LWR would be analyzed as part of a decommissioning

TABLE 6-1

Summary of Wastes From Decommissioning
a Reference PWR and a Reference BWR

Waste Stream	Volume (m^3)	Activity (Ci)
Reference 1155 MW(e) BWR:		
Activated metal	138	6,552,310
Activated concrete	90	170
Contaminated metal	15,543	8,574
Contaminated concrete	1,676	55
Dry solid waste (trash)[a]	3,386	--
Spent resins[b]	42	228
Filter cartridges[c]	--	--
Evaporator bottoms[d]	519	43,753
Reference 1175 MW(e) PWR:		
Activated metal	418	4,841,320
Activated concrete	707	2,000
Contaminated metal	5,465	900
Contaminated concrete	10,613	100
Dry solid waste (trash)[a]	1,418	--
Spent resins[b]	30	42,000
Filter cartridges[c]	8.9	5,000
Evaporator bottoms[d]	133	--

(a) Volumes shown are as-generated and prior to additional treatment such as compaction or incineration. Most of the trash is considered to be combustible.

(b) BWR spent resins actually include spent resins and filter sludge. Volumes shown are dewatered volumes.

(c) PWR filter cartridge volumes are as-solidified in concrete in 55-gallon drums. Filter cartridges are assumed not to be used in the BWR wet waste treatment system.

(d) PWR and BWR evaporator bottom volumes are as-generated prior to solidification.

TABLE 6-2

Volumes and Activities of Decommissioned LWR Activated Metals

Component	Disposal Volume $(m^3)^a$	Activity (Ci)	Specific Activity (Ci/m^3)
Reference BWR:			
Steam separator assembly	10	9,600	960
Fuel support pieces	5	700	140
Control rods and in-core instruments	15	189,000	12,600
Control rod guide tubes	4	100	25
Jet pump assemblies	14	20,000	1,429
Top fuel guide	24	30,100	1,254
Core support plate	11	650	59
Core shroud	47	6,300,000	134,043
Reactor vessel wall	8	2,160	46
Total	138	6,552,310	
Reference PWR:			
Pressure vessel cylindrical wall	108	19,170	178
Vessel head	57	<10	.18
Vessel bottom	57	<10	.18
Upper core Support assembly	11	<10	.91
Upper support columns	11	<100	9.1
Upper core barrel	6	<1,000	167
Upper core grid plate	14	24,310	1,736
Guide tubes	17	<100	6
Lower core barrel	91	651,000	7,154
Thermal shields	17	146,100	8,594
Core shroud	11	3,431,100	311,909
Lower grid plate	14	553,400	39,529
Lower support columns	3	10,000	333
Lower core forging	31	2,500	81
Miscellaneous internals	23	2,000	87
Reactor cavity liner	15	<10	.7
Total	485	4,841,320	

(a) Disposal volumes include the disposal container after the activated metal components have been cut into managable pieces.

environmental impact statement prepared for that facility. The volumes and activities estimated by PNL are for large modern units and such units are not expected to undergo decommissioning until well after the year 2000. Reactors potentially dismantled prior to the year 2000 are expected to be considerably smaller in capacity, have shorter operating lives than the reactors used as models for the PNL studies, and are expected to generate considerably lower waste volumes and/or activites.

There are a number of early low power units generally constructed as demonstration projects forerunning larger, more economical to operate units with capacities on the order of several hundred to a thousand MW(e). Although utilities would generally prefer to keep the older units operable for as long as they are cost-effective, costs of upgrading the older units to meet new NRC safety requirements may result in some of the older plants being decommissioned prior to the year 2000, and prior to the end of their otherwise servicable lives.

A specific example is the Indian Point Unit 1 Plant located near Buchanan, New York. This 175 MW(e) (600 MW(t)) PWR was shut down in October 1974 by its utility, Consolidated Edison, due to inability to meet new NRC requirements on emergency core cooling systems (ECCS). Consolidated Edison has recently determined that the cost of upgrading the plant to meet the new ECCS and other requirements would be greatly in excess of the possible economic gain, and have announced their intention of decommissioning the unit. The proposed timing and mode of decommissioing (safe storage, immediate dismantlement, or deferred dismantlement) however, has not yet been determined.

6.1.2 Decommissioning of Uranium Fuel Fabrication Plants

A relatively minor source of decommissioning wastes, compared to decommissioning light water reactors, will be wastes from decommissioning uranium fuel fabrication facilities. Potential waste volumes from decommissioning a relatively large fuel fabrication facility plant have been estimated by Pacific Northwest Laboratories (PNL),(3) and estimates based upon this study are summarized in Table 6-3. In the PNL study, a model plant is assumed which is based upon an existing facility operated by the General Electric Company in Wilmington, North Carolina. The plant is assumed to be operated for 40 years at a production rate of 1000 metric tons of uranium oxide fuel per year. Feed to the plant is enriched UF_6. All of the calcium fluoride (CaF_2) wastes and other conversion process sludges which are generated during the process converting UF_6 to UO_2 are assumed to be stored on-site in large lagoons until decommissioning.

As shown in Table 6-3, the calculated volumes of wastes generated from decommissioning the plant include trash and other miscellaneous material from decontaminating buildings and other facilities, as well as several thousand cubic meters of low activity bulk material such as CaF_2. The total quantity of uranium contained in the 1091 m^3 of miscellaneous trash is projected by PNL to be approximately 270 kg. The concentration of uranium in the 27,000 m^3 of low activity material is expected to be low.

These estimated quantities should be used with some care. For example, the timing of future fabrication plant decommissioning activities is very speculative, and would probably depend more on economic than safety considerations. Although the amount of fuel fabrication capacity would naturally be a function of nuclear power plant capacity, the total potential decommissioning volume would not be expected to show a strong dependence on capacity. Rather, total volumes of waste material obtained from decommissioning fuel fabrication plants would be a function of the number of plants operating and the design of individual plants rather than a function of the total throughput of uranium feed through the plants.

Projected volumes of CaF_2 and other chemical sludges produced from

TABLE 6-3

Waste Volumes Generated From Decommissioning a Model 1000 MT/yr UO_2 Fuel Fabrication Plant

Wastes from decommissioning buildings and other site structures:

Waste Category	Volume (m^3)
Hoods, equipment and components	764.4
Pipe, conduit, duct, trays, fixtures, etc.	118.52
HEPA and roughing filters	51.66
Concrete rubble	39.66
Contaminated liner and soil materials	91.0
Miscellaneous	25
Total	1,091

Low-activity bulk solids:

Waste Category	Volume (m^3)
Chemical sludge	1,282
Contaminated CaF_2	25,296
Other miscellaneous contaminated material	3,206
Total	29,784

UF_6 conversion are also speculative. The generation rate of UF_6 conversion sludges at a particular facility is strongly dependent on the design of the conversion process used at the facility. Space limitations at an individual plant may result in process sludges being transferred to LLW disposal sites during plant operation rather than being left on-site in lagoons for later consideration. Existing and future sludge lagoons at fabrication facilities may, rather than being collected and delivered to a LLW disposal site during decommissioning, be disposed in-place or treated to recover the contained uranium.

6.1.3 Decommissioning Uranium Fuel Recycle Facilities

Should uranium recycling be eventually adopted as a national policy, then uranium recycle facilities which would be constructed would eventually require decommissioning. Such decommissioning activities would occur relatively remote from today--at least beyond the year 2000. Volumes and activities of wastes that would result in decommissioning some reference uranium fuel recycle facilities have been estimated by PNL. In NUREG-0278,(4) the technology, safety, and costs of decommissioning a 1500 MTHM/year fuel reprocessing plant are assessed, using the uncompleted Barnwell, South Carolina reprocessing plant owned by Allied-General Nuclear Services as a model. In NUREG/CR-0129,(5) the technology, safety and costs of decommissioning a small mixed oxide fuel fabrication plant are assessed.

A potential source of wastes which may be generated in the next few years would be from decommissioning the Nuclear Fuel Services (NFS) reprocessing plant located in West Valley, New York. The reprocessing plant has not operated since 1972 and NFS announced in 1976 their intention to withdraw from the nuclear fuel reprocessing business. The eventual disposition of the facility, which includes a fuel reprocessing plant, 600,000 gallons of liquid high level waste stored in a tank (see Section 6.6), and a waste disposal area, is being addressed at this time. Fairly recently, DOE published a report which addressses alternatives for eventual disposition of the site, including full or partial decommissioning or continued use as some manner of nuclear production or research facility.[6] After completion of this study of alternatives, which was mandated by Congress, legislation was passed in 1980 (the West Valley Demonstration Project Act) that charges DOE with the responsibility to develop, construct, and operate a high-level liquid waste solidification project at the West Valley plant. This project will solidify the 600,000 gallons of liquid high-level waste presently stored in underground tanks to a final form acceptable for disposal into a Federal repository. Decontamination of existing facilities to prepare for the project, activities during the waste solidification project, and final decontamination of facilities at the end of the project will generate substantial volumes of low-level waste. Some of this waste is expected to be contaminated with transuranic radionuclides. The estimated volumes of these wastes are discussed in Section 6.6. DOE has not yet determined where these wastes will be disposed, but it appears that most of it will be consigned to Federal (DOE) disposal areas.

6.2 U.S Government Operations

Since the first commercial LLW disposal facilities were opened in 1962 (at Beatty, Nevada and Maxey Flats, Kentucky), considerable volumes of wastes generated by U.S. government agencies have been shipped to commercial sites for disposal. Most of this waste was produced by laboratories operated by or under contract to the Atomic Energy Commission (AEC). One of the original intents of this practice was to help provide some initial business to the then fledgling commercial disposal industry. This practice was continued by the AEC's successors, the Energy Research and Development Administration (ERDA) and the Department of Energy (DOE), until October 1979, when it was discontinued by DOE to help alleviate the shortage in commercial LLW disposal capacity.[7] Currently, all wastes generated by DOE facilities are disposed in DOE disposal sites. Small quantities of wastes produced by other government agencies such as the Department of Defense (non-classified waste only) or the U.S. Department of Agriculture, however, are still occasionally shipped to commercial LLW disposal facilities.

6.3 Three Mile Island Unit 2 Decontamination[8]

The March 28, 1979 accident at the Three Mile Island (TMI) Unit 2 nuclear power station has resulted in extensive damage to the reactor core as well as generation of significant quantities of contaminated water. Removal of damaged core components and other plant equipment, processing of the contaminated water, and decontamination of contaminated plant equipment and surfaces is projected to take about 5 to 9 years. Over this time period, radioactive wastes in various solid forms will be generated. NRC has prepared and published a programmatic environmental impact statement (PEIS) related to decontamination and disposal of radioactive wastes resulting from the accident (NUREG-0683).[8] In this document, NRC staff investigated a wide variety of decontamination and waste processing alternatives. Bounding (probable minimum and probable maximum) volumes of wastes projected to be delivered to LLW disposal facilities as a result of these decontamination and waste processing alternatives have been set out in the PEIS, and a summary of these projections is presented in Table 6-4.

The range in projected volumes reflects the fact that the actual volumes of waste generated will depend upon decisions regarding which decontamination and waste treatment alternatives will be implemented. In many cases, such decisions will be made as the decontamination operations progress. The decontamination and waste treatment operations will also generate some volumes of waste that will not be disposed of at near-surface disposal facilities. These include fuel or pieces of fuel removed from the reactor, other transuranic contaminated wastes, and some very high specific activity ion-exchange resin wastes generated as a result of treating contaminated reactor building water.

TABLE 6-4. Volumes of TMI-2 Packaged Solid Waste to Be Disposed of at a Commercial Low-Level Waste Disposal Site

Type of Package	Best-Case Conditions			Worst-Case Conditions	
	Package Volume (ft^3)	Number of Packages	Shipped Volume (ft^3)	Number of Packages	Shipped Volume (ft^3)
55-Gallon Drums					
Low activity	7.5	3,200	24,000	15,400	115,500
Intermediate activity	7.5	502	3,765	1,707	12,800
LSA Boxes[a]					
Low activity	80	1,042	83,360	2,128	170,240
Contaminated Equipment	70	86	6,020	293	20,510
and Hardware, Mirror Insulation	80	53	4,240	-	-
EPICOR II Resins[b]					
1st stage	50	49	2,450	49	2,450
2nd stage	50	14	700	14	700
3rd stage	175	6	1,050	6	1,050
Reactor Building Sump Cleanup					
Filters[c]	10	11	110	11	110
2nd stage	50	2	100	4	200
3rd stage	190	1	190	2	380
Primary System Cleanup[c]					
Filters	10/7.5/150[d]	16	990	57	1,340
2nd stage	50	4	200	44	2,200
3rd stage	190	3	570	12	2,280
		Totals	128,260		329,760

(a) Low specific activity.

(b) Will require special disposal procedures (e.g., deeper burial) if disposed of at a commercial disposal site.

(c) If any of these wastes contain fuel debris or greater than 10 nCi/gm transuranic materials, they would not be accepted at a commercial LLW facility.

(d) Primary system cleanup generates 3 filter types.

6.4 Generation of Transuranic Contaminated Waste

This section discusses the past and potential future generation and disposal of waste containing or contaminated with transuranic radioisotopes (isotopes having atomic numbers greater than that of uranium, which has an atomic number of 92). To put this discussion into perspective, however, a brief background is needed regarding past and probable future government disposal policies toward TRU waste.

Background

At one time, transuranic waste was disposed at near-surface disposal facilities operated by the AEC in addition to 5 of the 6 commerical disposal facilities. However, in 1970, the AEC initiated a policy whereby most government-produced wastes containing TRU isotopes in concentrations greater than 10 nanocuries per gram of waste material were placed into retrievable storage pending transfer to a repository for ultimate disposal. The 10 nanocurie per gram limit was based upon rough comparison with the potential hazards of upper concentration levels of naturally occurring radium in the earth's crust. However, TRU waste generated as a result of AEC (and later DOE) contracts with private contractors (and some DOE contractors) was still sent to commercial disposal facilities in addition to TRU wastes from commercial mixed oxide fuel fabricators and source manufacturers.

Retrievable storage of commercially-generated TRU waste (pending development of an ultimate repository of the waste) by the Federal government was the intent of a rule proposed in 1974.(9) Under this rule, commercial TRU waste would have been consigned to retrievable storage facilities operated by the Federal government pending the development of a facility for the ultimate disposition of the waste. A sensitivity level of 10 nanocuries per gram was proposed for measurements to determine the presence or absence of TRU contamination. At the time of the proposed rule, it was expected that commercial recycle of plutonium fuel for use in breeder reactors and in light water reactors as a mixed oxide would greatly increase in the near future. It was expected that significant additional volumes and quantities of TRU waste material would therefore soon be generated.

This rule, however, has never been finalized. The draft environmental impact statement published(10) in support of the proposed rule was withdrawn by the Energy Research and Development Administration (ERDA) when the AEC was reorganized to form ERDA and NRC. The Department of Energy (DOE), ERDA's successor, is continuing the policy of retrievable storage of government produced TRU waste but has stated that it does not have legal authority to accept commercial TRU waste for retrievable storage.

In the meantime, individual state initiatives have resulted in a 10 nanocurie per gram disposal limit for TRU waste at all operating commercial low-level waste disposal facilities. Although at one time five of the six commercial LLW disposal sites accepted TRU waste for disposal (the Barnwell, South Carolina facility has never accepted TRU waste for disposal), this practice has been discontinued. The last commercial facility to accept TRU waste for disposal was the site located in the center of the Hanford Reservation near Richland, Washington and operated by the Nuclear Engineering Company (NECO). From 1976 to 1979, the NECO-Richland facility was the only commercial disposal facility accepting TRU waste for disposal. TRU waste acceptance at the NECO-Richland facility in concentrations exceeding 10 nCi/gm was prohibited by the State of Washington in November 1979.(11)

TRU Waste Generation

Compared to operations conducted by the Department of Energy (DOE), there has been only relatively small quantities of transuranic (TRU) waste generated by the commercial sector. Major sources of transuranic wastes which have been delivered in the past to commercial disposal sites have included:

- DOE and its successors, the Energy Research and Development Administration (ERDA) and the Atomic Energy Commission (AEC);
- DOE, ERDA, and AEC contractors;
- Reprocessing of spent uranium fuel at the West Valley, New York commercial fuel reprocessing plant.
- Research and development of plutonium fuels, including fabrication of small quantities of mixed-oxide (MOX) fuels for test purposes in light water reactors; and
- Research studies of irradiated reactor fuel.

Within the last few years, the amount of transuranic waste delivered to commercial disposal facilities has been further reduced to even lower levels and has been finally discontinued. This has been caused by a number of factors. One factor was the policy announced by AEC in 1970 whereby AEC-produced TRU waste in concentrations greater than 10 nCi/gm were consigned to retrievable storage at AEC facilities pending the availability of a repository for the ultimate disposition of the waste. TRU waste generated as a result of AEC (and later DOE) contracts with private companies, however, was still sent to commercial disposal sites. The only commercial reprocessing facility ever to operate in the United States was the facility operated by Nuclear Fuel Services (NFS) near West Valley, New York. In 1972, this facility was shut down and has not operated since. In 1976, President Carter announced a national policy of deferment of commercial fuel reprocessing. This policy of deferring fuel reprocessing has halted most of the mixed oxide fuel research and development work in the commercial sector. Prior to the cutoff of TRU disposal at the NECO-Richland site, most commercial mixed oxide fuel fabrication test facilities had an active program underway for facility clean-up and decontamination.

Table 6-5 is a summary of the quantities of plutonium delivered to the NECO-Richland site during the years 1976 through 1978, and the

TABLE 6-5

Grams of Plutonium Delivered to NFCO-Richland Disposal Facility Between 1/1/76 and 5/24/79

	1979	1978	1977	1976
Babcock and Wilcox Lynchburg, VA	52 (J)	270 (J)	35 (J)	--
Babcock and Wilcox Leechburg, PA	--	27 (G)	414 (J)	7074 (B) 630 (G) 945 (J)
Westinghouse Cheswick, PA	--	152 (G) 148 (J)	222 (G) 120 (J)	273 (G) 856 (J)
General Electric Vallecitos, CA	350 (G)	1006 (G) 2268 (J)	469 (G) 810 (J)	65 (G) 117 (J)
Battelle Columbus, OH	29 (G) 98 (H)	22 (G) 18 (H) 268 (J)	--	--
Battelle (PNL) Richland, WA	--	--	10 (G) 113 (J)	21 (J)
Kerr-McGee Cimmaron, OK	--	77 (J)	49 (J)	1798 (B) 474 (J)
Nuclear Fuel Services Erwin, TN	--	594 (J)	--	76 (J)
Allied General Nuclear Services Barnwell, SC	--	20 (J)	--	--
US Army Material Command	--	--	--	1 (B)
Lovelace Foundation, Albq.NM	--	--	*	*
LFE Environmental, Rich., CA	--	*	*	--
General Atomic Company San Diego, CA	--	--	--	*
Total	529	4870	2242	12330
(B)	--	--	--	--
(G)	379	1207	701	8873
(H)	98	18	--	968
(J)	52	3645	1541	2489
% of Total: (B) + (G)	90%	25%	31%	75%
% of Total: (H) + (J)	10%	75%	69%	75%

(B) DOE-Owned, Lease Agreement - Non-Waiver of Use Charge
(G) DOE-Owned Production and Research Programs
(H) Owned by Other U.S. Government Agencies
(J) Privately Owned (Domestic)
* Less than 1 gram
** To 5/24/79

year 1979 to May 24.(12) Most of the TRU waste generated was from clean-up and decontamination of former plutonium research laboratories and small-scale MOX fuel fabrication facilities. Small quantities of waste were also generated from burn-up studies of LWR fuel (e.g., Battelle Columbus Laboratory). Not shown on this table are some very small quantities of wastes contaminated with Pu-238 (estimated at less than 5.7 m^3/year) and produced from the manufacture of radioactive power sources. Significant quantities of TRU waste shipped to the NECO-Richland site during this time period were owned by DOE -- i.e., 75% in 1976, 31% in 1977, 25% in 1978, and 69% in 1979 up to May 24. Much of the other plutonium contaminated wastes were generated as a result of DOE-contracted work.

Future generation of TRU waste is speculative but may arise from three basic sources: decontamination of existing small scale plutonium research and fuel fabrication facilities, studies of irradiated LWR fuel, and recycle of spent uranium fuel. Based on information received by NRC staff from industry and DOE, it appears that decontamination of existing plutonium fuel fabrication facilities would generate approximately 4956 m^3 of waste over an approximate 3-year time period. These wastes are expected to have low radiation levels permitting contact handling of waste packages. Following these decontamination and decommissioning activities, potential TRU waste volumes are projected to drop to low levels (approximately 75 m^3/yr) and would result from destructive examination of reactor fuels.(13) These wastes are expected to have high surface radiation levels and would require remote handling. Plutonium-238 contaminated waste from manufacture of heat sources would also be expected to continue at a rate of about 5.7 m^3 per year. Of course, the current lack of commercial storage capacity combined with DOE's position on TRU waste acceptance has a great effect on the timing of the generation of such waste. Any waste generated would have to be stored on-site.

Finally, significant quantities of TRU waste could be generated in the future through implementation of a plutonium-based nuclear fuel cycle -- that is, through reprocessing of irradiated LWR fuel to extract residual fissile uranium and plutonium and through fabricating the received uranium and plutonium into mixed oxide fuel for reuse in LWR's. Potential volumes and activities of wastes that would be generated by uranium recycle operations have been estimated by a number of groups, including NRC(14), DOE(15), and the national laboratories.(16,17) Most of the waste thus generated would be contaminated with (or suspected of being contaminated with) transuranic isotopes and would not be acceptable at current disposal facilities.

In any case, the timing of the generation of such waste is very speculative. The current policy of the United States is to defer processing of spent light water reactor fuel. Spent uranium fuel removed from nuclear power reactors is presently stored without attempting to extract the residual fissile uranium and plutonium for reuse. Even if the national policy regarding recyle of uranium fuel were to change within a short time period, it would still be several years before significant quantities of wastes would be produced. Of the three commercial reprocessing plants that have been constructed in the United States -- at West Valley, New York; Morris, Illinois; and Barnwell, South Carolina -- only the West Valley plant has ever operated. This plant, however, has not operated since 1972. None of the three facilities could operate today without extensive modification. Of the three, the Barnwell facility would require the least construction--principally construction of a waste solidification facility, a facility for conversion of liquid plutonium nitrate to solid plutonium oxide, and probable installation of additional airborne effluent treatment systems. The Morris facility would require major changes in the design of the processing operations. The West Valley plant would require considerable modification to meet seismic and radiation shielding requirements. In addition, the operator of the West Valley plant -- Nuclear Fuel Services, Inc.-- has previously (1976) expressed a desire not to continue in the reprocessing business.(6)

There are currently no large scale commercial facilities for fabrication of mixed oxide fuel, although a number of small scale commercial laboratories and research facilities are in existence that have in the past fabricated small batches of MOX fuel for experimental use in LWR's. Such large scale facilities would have to be constructed.

Finally, there are a number of institutional considerations. Licensing requirements for construction of new uranium recycle facilities or modification of old ones would tend to delay operation of such facilities. Such licensing requirements would include regulatory review, publication of environmental impact statements and other environmental assessments, and probable hearings. DOE would have to finalize and implement plans for acceptance of TRU and high-level waste for retrievable storage pending disposal into a repository. The costs for such retrievable storage have not been finalized by DOE and, as discussed earlier, DOE has taken the position that it does not have legal authority to accept commerical TRU waste for storage. In addition, no decision has been made regarding criteria for high-level and TRU waste form characteristics for disposal. Such criteria would probably have to be finalized prior to construction of high-level waste solidification facilities at reprocessing plants.

6.5 Independent Spent Fuel Storage Facilities

As there is at this time neither an ongoing nuclear fuel reprocessing industry or a federal high-level waste repository, spent nuclear fuel removed from nuclear power plants must be [illegible] stored. This spent fuel is currently being stored in fuel pools located within nuclear power stations as well as within two facilities originally designed to process the spent fuel: the General Electric (GE) reprocessing plant located near Morris, Illinois, and the Nuclear Fuel Services (NFS) reprocessing plant located near West Valley, New York. The GE facility never became operational and the NFS facility suspended reprocessing operations in 1971. As of the end of 1979, the total amount of spent fuel stored in the Morris and West Valley plants corresponded to about 9 percent of the total U.S. commercial inventory of stored LWR fuel.(18)

The existing storage capacity for spent LWR fuel is not likely to be adequate until a repository or an ongoing fuel reprocessing industry is developed. Additional storage capacity has been provided through fuel storage densification in existing fuel storage pools. Alternatives that may be used to provide needed additional storage capacity in the future include construction of new pools at power plants, expansion of storage capacity in the West Valley and Morris facilities, use of the fuel storage capacity of the uncompleted Barnwell, South Carolina reprocessing plant, or construction of new independent spent fuel storage facilities. Dry storage concepts for aged spent fuel are also being developed and are of high interest for use at either reactor sites or away-from-reactor sites. Recently, NRC published a new set of regulations, 10 CFR Part 72, which establish rules for licensing independent spent fuel storage facilities, if and when they are constructed.(19)

Wastes from storing spent LWR fuel would primarily arise from treatment of the storage basin water, receiving and unloading spent fuel, and plant ventilation systems. These wastes include spent resins, filter sludges and miscellaneous trash, and are similar in composition to wastes produced from other light water reactor operations.

Waste volumes generated to the year 2000 from LWR fuel storage are expected to be relatively small. Most of the waste volumes generated would continue to be included with other wastes shipped from power plants. Only small quantities of wastes are produced by the current two facilities practicing away-from-reactor storage. LLW generated at the West Valley plant is disposed on-site at the co-located LLW disposal site. At the Morris plant, low specific activity trash is currently shipped to a LLW disposal site. Liquid wastes and filter sludges generated from backflushing and regenerating the fuel pool water filter system are stored in a large (2.6 million liter capacity) low activity waste (LAW) tank. The LAW tank was originally constructed and intended to store low level liquids generated during the operation of the reprocessing plant. Eventually, General Electric plans to install a solidification system to solidify the liquids and other wet wastes and send the solidified waste material to a LLW disposal site.(20)

DOE has estimated the annual volumes of waste that could be generated from a large (3000 MTHM) independent spent fuel storage installation, assuming that one is constructed.(15) These volumes are listed in Table 6-6 and are based upon a conservative (in terms of waste generation) assumption of an operating mode in which one-sixth of the storage capacity is replaced each year. The total volume of waste produced from such a large facility is comparable to the annual generation rate of a single 1000 MW(e) light water reactor.

At this time, NRC has not received any application for construction and operation of an independent spent fuel storage facility. The timing for future construction of a storage facility (and associated waste volume generation) is somewhat speculative.

6.6 Low-Level Waste from West Valley Demonstration Project

The solidification of the commercial liquid high-level waste currently stored in tanks at Western New York Nuclear Service Center (WNYNSC) and the decontamination of the reprocessing cells and equipment for functional use or decommissioning are expected to result in generation of low-level wastes. This waste will be generated in both liquid and solid forms. All liquid wastes are expected to be solidified prior to eventual disposition.

There are several studies currently ongoing to determine preferred

TABLE 6-6

Estimated Annual Waste Volumes Generated
From Assumed Operation of a 3,000 MTHM
Spent Fuel Storage Facility

Waste Category	Volume (m^3)
Compactable and Combustible Wastes	
Combustible trash	630
Ventilation filters	23
Liquids and Other Wet Wastes	
Bead resins	2
Filter precoat sludge	8
Sulfate concentrate	7
Miscellaneous solution concentrates	10
Non-combustible material	
Non-combustible trash	51
Failed equipment	19
Total	750

alternative actions in accordance with the NEPA process. Several alternatives for solidification and decontamination are under consideration.(21) A preliminary study(22) has indicated extreme variability of the expected amount of LLW expected from decontamination operations.

There are four major alternatives with minor variations to be considered. These alternatives are briefly discussed below.

The alternative called "sludge/salt separated" involves removal and processing of the HLW from the tanks with sludge and salt fractions separated (salt containing minute amounts of residual radioactivity), and decommissioning of the facility and the HLW tanks. Two major options are: (a) protective storage, and (b) dismantlement. Both options envision the use of the old facility for HLW processing. The second alternative is called "sludge/salt unseparated" and differs from the above only in the HLW processing techniques; all the HLW are processed together.

The alternative called "interim form" envisions an interim form for the HLW which can be fused salt or agglomerated calcine. The same two major options in addition to these two waste forms (protective storage or dismantlement) yield four subalternatives. The final alternative is called "in-tank solidification." In this alternative HLW liquid wastes are solidified in the tanks, no HLW tank decommissioning is necessary, and no new equipment installation is required.

The expected low-level wastes from these alternatives are summarized in Table 6-7. The characteristics of these wastes cannot be accurately estimated at the present time.

TABLE 6-7 . Low-Level Waste* Packages From West Valley Demonstration Project

	55-gallon Drums				4x4x8 Boxes		
Alternative and Option	Trash	Salt Cake	Decon	Resin	LLW	TRU	4x4x4
Salt/Sludge Separated							
(a) Protective Storage							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	5100	680	290	70	24	---
Final Decon	920	---	6800	110	---	---	---
(b) Dismantlement							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	5100	680	290	70	24	---
Final Dismantlmt	1100	---	6800	110	710	171	360
Salt/Sludge Unseparated							
(a) Prot.Storage							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	---	680	290	70	24	---
Final Decon	920	---	6800	110	---	---	---
(b) Dismantlement							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	---	680	290	70	24	---
Final Dismantlmt	1100	---	6800	110	710	171	360
Interim Waste Form							
(a) Prot Strg, FuSalt**							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	---	680	290	70	24	---
Final Decon	920	---	6800	110	---	---	---
(b) Prot Strg, AggCal**							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	5100	680	290	70	24	---
Final Decon	920	---	6800	110	---	---	---
(c) Dismantlmt, FuSalt							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	---	680	290	70	24	---
Final Dismantlm	1100	---	6800	110	710	171	360
(d) Dismantlmt, AggCal							
Initial Decon	540	---	3400	110	337	113	---
HLW Operations	1500	5100	680	290	70	24	---
Final Dismantlm	1100	---	6800	110	710	171	360
In-tank Solidification							
(a) Protective Storage	920	---	6800	110	---	---	---
(b) Dismantlement	1100	---	6800	110	710	41	---

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* Estimated TRU fractions of the packages are : Trash and 4x4x4 Boxes = 50%; Salt Cake = 0%; Decon and Resin = 25%; 4x4x8 Boxes cannot be given as a percentage, therefore they have been specified.

** FuSalt is the Fused Salt Option; AggCal is the Agglomerated Calcine Option.

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APPENDIX A

PROJECTIONS OF ROUTINELY GENERATED LOW LEVEL RADIOACTIVE WASTE THROUGH THE YEAR 2000

PROJECTIONS OF ROUTINELY GENERATED LOW LEVEL RADIOACTIVE WASTE THROUGH THE YEAR 2000

U.S. Nuclear Regulatory Commission

Low-Level Waste Licensing Branch
Division of Waste Management

1980

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INTRODUCTION

This document contains NRC staff projections of the volumes, activities, and regional distributions of different types and forms of low-level radioactive wastes (LLW) expected to be routinely generated and shipped to LLW disposal facilities through the year 2000. The regions used in this analysis are the existing five NRC regions.

The projections contained in this document are divided into two main sections: (1) projections of fuel cycle wastes, and (2) projections of non-fuel cycle wastes. The projections include wastes which are currently being generated and shipped to LLW disposal sites or are expected to be routinely generated in the near future. Wastes from the nuclear fuel cycle (Section 1) include those from uranium conversion plants, uranium fuel fabrication plants, and light water power reactors, while those from non-fuel cycle sources (Section 2) include wastes from a number of sources including hospitals, universities, and industrial concerns

The projections will be used to help assess the potential regional impacts of LLW disposal for use in developing a regulation, 10 CFR Part 61, for near-surface disposal of radioactive waste. In making the projections, emphasis was placed upon major waste streams which are being produced today or are expected in the near future. Although an effort was made to use the best available data in making the projections, the projections should be used with some care. In some cases, the available data--particularly for non-fuel cycle industrial waste streams--is limited and an effort is currently being made to acquire additional data with the aim of reducing uncertainties.

The projections are also limited by the inherent variable nature of LLW generation. Facilities producing waste may open, close, or otherwise modify operations, depending upon economic or other influences which are not readily predictable. Regulatory actions may also have a significant impact on waste volumes and activities.

A third category of wastes is also included in this document in Section 3. These include wastes which (1) are not currently being sent to LLW disposal facilities but may have in the past, (2) are non-routine, or (3) are considered at this time to be speculative in regard to the volumes which may be generated as well as the timing of generation.

1.0 PROJECTIONS OF LLW GENERATED BY THE NUCLEAR FUEL CYCLE

Table 1 contains a summary of volumes and gross specific activities of wastes generated from the nuclear fuel cycle and includes wastes from light water reactor (LWR) operations and from uranium fuel fabrication plants. The volumes and activities from LWR operations and uranium fuel fabrication plants are listed on a "per MW(e)-yr" basis--that is, the volumes and total activities of the wastes annually produced are assumed to be multiples of the electrical generation capacity. The volumes and specific activities shown are taken from ONWI-20 (Ref. 1). In Reference 1, the basic waste generation data for BWRs using either a precoat or a deep bed condensate polishing system (CPS) was averaged from the output of 14 units over several years time, while PWR waste generation data for units with or without a CPS was averaged from the output of 23 units. Projected averages were determined from data obtained from Appendix C of Reference 1.

The volumes shown, with the exception of filter cartridges, are for untreated wastes. Concentrated liquids (evaporator bottoms) are reported as-generated prior to solidification. Resins and filter sludges are reported as-dewatered, and the trash streams are reported as-generated prior to such processing operations as incineration or compaction. The volumes for cartridge filters are given as-packaged for shipment. Additional information can be obtained from Reference 1.

Also shown are estimated volumes and gross specific activities for nonfuel core components such as poison curtains, flow channels, and control rods. The high specific activity of these core components is due to neutron activation, which results in a waste form having a relatively low leaching rate. Core components from LWRs are replaced on an infrequent basis, making projections of this waste stream difficult. In addition, nonfuel core components are frequently shipped to disposal facilities by placing the components in the middle of a container of otherwise low activity material such as trash. The surrounding trash acts as shielding for handling and transport.

Projections of activated core components are approximated based upon unpublished 1977 radioactive shipment records from the Maxey Flats, Kentucky disposal facility. Raw data was doubled (Maxey Flats had received approximately half the waste activity in the country in 1977) and divided by the existing LWR plant capacity during that year (Ref. 2).

Another waste stream which is difficult to project will be generated by periodic decontamination of the primary coolant systems of light water reactors. The purpose of such full-scale primary coolant decontamination operations is to reduce plant personnel exposures by removing crud accumulated on surfaces in contact with the primary coolant. Although full-scale primary coolant decontamination operations have not been routinely performed in LWRs in the past, NRC has published an environmental statement regarding such an operation being performed at the Dresden Unit 1 nuclear power station. In the decontamination process for Dresden Unit 1, a decontamination solution is circulated and flushed through the coolant system, which dissolves the crud deposits. The decontamination solution is then removed from the coolant system and processed

Table 1 Summary of Principal Nuclear Fuel Cycle Waste Streams

	Untreated Waste Volumes (ft^3/MWe-yr)*					
	Boiling Water Reactors			Pressurized Water Reactors		
Waste Type	Deep Bed CPS**	Precoat CPS	Projected Ave.†	Without CPS	With CPS	Projected Ave.†
Resins††	4.6	0.23	2.85	0.94	0.32	0.62
Concentrated liquids#	12.7	0.6	7.86	3.9	4.8	4.36
Filter sludge††	5.4	7.7	6.32	-	0.15	0.0765
Cartridge filters##	-	-	-	0.39	0.39	0.39
Trash¶						
Total	11.5	11.5	11.5	11.5	11.5	11.5
Compactible	7.8	7.8	7.8	7.6	7.6	7.6
Noncompactible	3.7	3.7	3.7	3.9	3.9	3.9
Total	34.2	20.0	28.53	16.7	17.2	16.88
	Untreated Waste Activity (Ci/MWe-yr)					
Resins	1.9	.0014	1.14	0.61	0.2	0.40
Concentrated liquids	0.58	0.016	0.35	0.20	0.024	0.11
Filter sludge	2.0	0.5	1.40	-	0.012	.00612
Cartridge filters	-	-	-	0.12	0.12	0.12
Trash						
Total	0.402	0.402	0.402	0.063	0.063	0.063
Compactible	0.0052	0.0052	0.0052	0.0049	0.0049	0.0049
Noncompactible	0.397	0.397	0.397	0.058	0.058	0.058
Total	4.88	0.92	3.29	1.00	0.42	0.699

Light Water Reactor Nonfuel Core Components (per GWe)-Yr)¶¶

4,000 Ci and 35 cf

Light Water Reactor Primary Coolant Decontamination Waste (per reactor)§

Reactor Type	Resins (m^3)
PWR	95
BWR	47.5

Generated at 5-10 year intervals

Table 1 (Cont'd)

Fuel Fabrication Wastes§§

48 MTU/GW(e) of domestic reactor capacity
90 cf of waste/MTU waste produced

4,320 cf of waste/ GW(e) of reactors
24 uCi/ cf for all wastes (average)
104 mCi/ GW(e) (all uranium)
146 kgU/ GW(e)

Types of Wastes:
- Trash (85% combustible, 15% noncombustible)
- Filter Sludges
- Pre and HEPA filters
- Oil
- Process Sludges (insufficient data for breakdown)

Regional Distribution: I: 20% II: 50% III: 0% IV: 0% V: 30%

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[*] Waste streams for LWRs were based upon projections in Reference 1 (ONWI-20).

[**] Condensate polishing system.

[†] Projected average PWR and BWR waste volumes and activities were determined based upon data obtained from Appendix C of ONWI-20 (Ref. 1). Of 58 BWRs either in operation or under construction representing an electrical generating capacity of 52, 531 MW(e), units using precoat condensate polishing systems (CPS) accounted for 40% of this capacity (21,175 MW(e), 23 units), while units using deep bed CPS accounted for 60% (31,356 MW(e), 35 units). Of 41 PWRs either in operation or under construction representing an electrical generating capacity of 37,292 MW(e), units with CPS accounted for 51% of the capacity (19,081 MW(e), 20 units), while units with no CPS accounted for 49% (18,211 MW(e), 21 units).

[††] Dewatered volumes.

[#] As-generated volumes prior to solidification.

[##] Volumes as-packaged for shipment.

[¶] As-generated volumes prior to possible further processing by techniques such as incineration or compaction.

[¶¶] Volumes and activities of LWR poison curtains, flow channels, control rods, and other miscellaneous nonfuel core components were estimated based upon 1977 data (Ref. 2) from the Maxey Flats, Kentucky disposal site. Raw data was doubled (Maxey Flats had received approximately half the waste activity in the country in 1977), and divided by the existing plant capacity during that year.

[§] Based upon Reference 3. Resin volumes are given as de-watered.

[§§] Wastes from fuel fabrication plants were mainly based on Reference 1. However, the volumes and activity from reference 1 were increased by the inclusion of process sludges.

through an evaporator. The evaporator bottoms are then solidified in vinyl ester styrene (a synthetic polymer) for shipment to an offsite disposal facility.

Although the Dresden-1 decontamination operation can be considered in many respects a prototype of future primary coolant decontamination processes at other nuclear power plants, it is still difficult to project future volumes and other characteristics of decontamination wastes. There may be a number at possible decontamination processes utilized--e.g., from dilute chemical processes on an annual basis to more concentrated processes at intervals of several years--and the waste streams generated may vary in kind (e.g., resins, solidified liquids) and in volume from operation to operation and plant to plant. Other plant-specific factors which would influence the volumes, radioactivity content, and other characteristics of the wastes generated would include the operating history of the plant (e.g., history of fuel failures), the design of the plant and liquid clean-up and processing systems, the chemistry of the primary coolant, and the length of time between decontamination operations. Institutional matters such as the policies of a specific utility could also be a consideration.

Notwithstanding this uncertainty, NRC staff believe that wastes generated from routine full-scale decontamination of reactor primary coolant systems should be represented in the low-level waste source data base. As shown in Table 1, it is assumed that every operating LWR undergoes a full-scale primary coolant decontamination operation every 5 to 10 years using a dilute chemical decontamination process (Ref. 4). This results in BWR and PWR resin waste streams of approximately 95 and 47.5 m^3, respectively, per operation. This assumes that the volume of contaminated liquid generated per operation are 760 m^3 and 380 m^3, respectively, and assumed that approximately 0.125 m^3 of dewatered resin is required to process 1 m^3 of contaminated liquid. Contained in these resins will be significant quantities of chelating agents and other decontamination chemicals.

Projections for fuel fabrication wastes were assumed to be proportional to power plant capacity and were obtained from Reference 1. However, volumes and activities listed in Reference 1 were increased by inclusion of process sludges.

Tables 2 and 3 list the projected nuclear power generation rates (for purposes of waste disposal) for each of the 5 NRC regions through the year 2000. Also shown is the projected number of operating power reactors operating per year by region. The projections were principally based upon a review of nuclear power stations currently built and operable, under construction, or planned or on order. Such information is available from DOE (Refs. 5, 6) or from Nuclear News (Refs. 7, 8) on a bi-annual basis. Projections regarding startup times made by NRC licensing staff were also used to supplement the basic information (Ref. 9).

Two scenarios are assumed for nuclear power station construction:

(1) A "low scenario," Table 2, which assumes tht construction continues on power reactors which are already under construction but that any

Table 2 Projected Regional U.S. Nuclear Power Capacity - Low Scenario

YEAR OPERABLE *a	REGION										TOTAL		
	1		2		3		4		5				
	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	TOTAL
	10,070	6,050	9,754	6,407	8,534	4,571	2,219	778	2,484	65			
	(13)*b	(8)	(12)	(7)	(11)	(9)	(3)	(1)	(3)	(1)	33,061	17,871	50,932
1980	1,115*c		4,064						1,084				
	11,185*d	6,050	13,818	6,407	8,534	4,571	2,219	778	3,568	65	39,324	17,871	57,195
	(14)	(8)	(16)	(7)	(11)	(9)	(3)	(1)	(4)	(1)	(47)	(26)	(73)
1981	1,050		3,225	1,250		2,981			1,106				
	12,235	6,050	17,043	7,657	8,534	7,552	2,219	778	4,674	65	44,705	22,102	66,807
	(15)	(8)	(19)	(8)	(11)	(12)	(3)	(1)	(5)	(1)	(53)	(30)	(83)
1982		819	4,683		1,120	1,078	1,111		2,370	1,100			
	12,235	6,869	21,726	7,657	9,654	8,630	3,330	778	7,044	1,165	53,989	25,099	79,088
	(15)	(9)	(23)	(8)	(12)	(13)	(4)	(1)	(7)	(2)	(61)	(33)	(92)
1983		2,115	3,168		2,391	2,138	2,400		1,100				
	12,235	8,984	24,894	7,657	12,045	10,768	5,730	778	8,144	1,165	63,048	29,352	92,400
	(15)	(11)	(26)	(8)	(15)	(15)	(6)	(1)	(8)	(2)	(70)	(37)	(107)
1884	1,200		900		1,120		1,111		3,706				
	13,435	8,984	25,794	7,657	13,165	10,768	6,841	778	11,850	1,165	71,085	29,352	100,437
	(16)	(11)	(27)	(8)	(16)	(15)	(7)	(1)	(11)	(2)	(77)	(37)	(114)
1985	1,989	2,132	3,675	3,417	2,250		1,250	1,150	1,218				
	15,424	11,116	29,469	11,074	15,415	10,768	8,091	1,928	13,068	1,165	81,467	36,051	117,518
	(18)	(13)	(30)	(11)	(18)	(15)	(8)	(2)	(12)	(2)	(86)	(43)	(129)
1986	1,200	1,100	2,017	2,466	2,250				2,512				
	16,624	12,216	31,486	13,540	17,665	10,768	8,091	1,928	15,580	1,165	89,446	39,617	129,063
	(19)	(14)	(32)	(13)	(20)	(15)	(8)	(2)	(14)	(2)	(93)	(46)	(139)
1987			1,807			2,782		1,150					
	16,624	12,216	33,293	13,540	17,665	13,550	8,091	3,078	15,580	1,165	91,253	43,549	134,802
	(19)	(14)	(34)	(13)	(20)	(18)	(8)	(3)	(14)	(2)	(95)	(50)	(149)

Table 2 (Cont'd)

YEAR OPERABLE	REGION 1		REGION 2		REGION 3		REGION 4		REGION 5		TOTAL		
	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	TOTAL
1988	1,067		2,185	2,466									
	17,691	12,216	35,478	16,006	17,665	13,550	8,091	3,078	15,580	1,165	94,505	46,015	140,520
	(20)	(14)	(36)	(15)	(20)	(18)	(8)	(3)	(14)	(2)	(98)	(52)	(150)
1989				1,233									
	17,691	12,216	35,478	17,239	17,665	13,550	8,091	3,078	15,580	1,165	94,505	47,248	141,753
	(20)	(14)	(36)	(16)	(20)	(18)	(8)	(3)	(14)	(2)	(98)	(53)	(151)
1990			2,180		1,120								
	17,691	12,216	37,658	17,239	18,785	13,550	8,091	3,078	15,580	1,165	97,805	47,248	145,053
	(20)	(14)	(38)	(16)	(21)	(18)	(8)	(3)	(14)	(2)	(101)	(53)	(154)
1991													
	17,691	12,216	37,658	17,239	18,785	13,550	8,091	3,078	15,580	1,165	97,805	47,248	145,053
	(20)	(14)	(38)	(16)	(21)	(18)	(8)	(3)	(14)	(2)	(101)	(53)	(154)
1992			1,280										
	17,691	12,216	38,938	17,239	18,785	13,550	8,091	3,078	15,580	1,165	97,805	47,248	146,333
	(20)	(14)	(39)	(16)	(21)	(18)	(8)	(3)	(14)	(2)	(102)	(53)	(155)

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*a - Operable prior to 1980.
*b - Number of reactors.
*c - Electrical energy capacity added in year.
*d - Total capacity available in year.

Table 3 Projected Regional U.S. Nuclear Power Capacity - High Scenario

	REGION										TOTAL		
	1		2		3		4		5				
YEAR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	TOTAL
OPERABLE	10,070	6,050	9,754	6,407	8,534	4,571	2,219	778	2,484	65			
*a	(13)*b	(8)	(12)	(7)	(11)	(9)	(3)	(1)	(3)	(1)	33,061	17,871	50,932
1980	1,115*c		4,064						1,084				
	11,185*d	6,050	13,818	6,407	8,534	4,571	2,219	778	3,568	65	39,324	17,871	57,195
	(14)	(8)	(16)	(7)	(11)	(9)	(3)	(1)	(4)	(1)	(47)	(26)	(73)
1981	1,050		3,225	1,250		2,981			1,106				
	12,135	6,050	17,043	7,657	8,534	7,552	2,219	778	4,674	65	44,705	22,102	66,807
	(15)	(8)	(19)	(8)	(11)	(12)	(3)	(1)	(5)	(1)	(53)	(30)	(83)
1982		819	4,683		1,120	1,078	1,111		2,370	1,100			
	12,235	6,869	21,726	7,657	9,654	8,630	3,330	778	7,044	1,165	53,989	25,099	79,088
	(15)	(9)	(23)	(8)	(12)	(13)	(4)	(1)	(7)	(2)	(61)	(33)	(94)
1983		2,115	3,168		2,391	2,138	2,400		1,100				
	12,235	8,984	24,894	7,657	12,045	10,768	5,730	778	8,144	1,165	63,048	29,352	92,400
	(15)	(11)	(26)	(8)	(15)	(15)	(6)	(1)	(8)	(2)	(70)	(37)	(107)
1984	1,200		900		1,120		1,111		3,706				
	13,435	8,984	25,794	7,657	13,165	10,768	6,841	778	11,850	1,165	71,085	29,352	100,437
	(16)	(11)	(27)	(8)	(16)	(15)	(7)	(1)	(11)	(2)	(77)	(37)	(114)
1985	3,139	2,132	3,675	3,417	2,250		1,250	1,150	1,218				
	16,574	11,116	29,469	11,074	15,415	10,768	8,091	1,928	13,068	1,165	82,617	36,051	118,668
	(19)	(13)	(30)	(11)	(18)	(15)	(8)	(2)	(12)	(2)	(87)	(43)	(130)
1986	2,270	1,100	2,017	2,466	2,250				2,512				
	18,844	12,216	31,486	13,540	17,665	10,768	8,091	1,928	15,580	1,165	91,666	39,617	131,283
	(21)	(14)	(32)	(13)	(20)	(15)	(8)	(2)	(14)	(2)	(95)	(46)	(141)
1987			1,807	934		2,782		2,300					
	18,844	12,216	33,293	14,474	17,665	13,550	8,091	4,228	15,580	1,165	93,473	45,633	139,106
	(21)	(14)	(34)	(14)	(20)	(18)	(8)	(4)	(14)	(2)	(97)	(52)	(149)

Table 3 (Cont'd)

	REGION										TOTAL		
	1		2		3		4		5				
	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR	TOTAL
1988	1,067	1,150	2,185	2,466					1,260	1,277			
	19,911	13,366	35,478	16,940	17,665	13,550	8,091	4,228	16,840	2,442	97,985	50,526	148,511
	(22)	(15)	(36)	(16)	(20)	(18)	(8)	(4)	(16)	(3)	(102)	(56)	(158)
1989		1,150		1,233					1,260				
	19,911	14,516	35,478	18,173	17,665	13,550	8,091	4,228	18,100	2,442	99,245	52,909	152,154
	(22)	(16)	(36)	(17)	(20)	(18)	(8)	(4)	(17)	(3)	(103)	(58)	(161)
1990			3,460		1,120					1,277			
	19,911	14,516	38,938	18,173	18,785	13,550	8,091	4,228	18,100	3,719	103,825	54,186	158,011
	(22)	(16)	(39)	(17)	(21)	(18)	(8)	(4)	(17)	(4)	(107)	(59)	(166)
1991													
	19,911	14,516	38,938	18,173	18,785	13,550	8,091	4,228	18,100	3,719	103,825	54,186	158,011
	(22)	(16)	(39)	(17)	(21)	(18)	(8)	(4)	(17)	(4)	(107)	(59)	(166)
1992	1,250		2,560		2,390								
	21,161	14,516	41,498	18,173	21,175	13,550	8,091	4,228	18,100	3,719	110,025	54,186	164,211
	(23)	(16)	(41)	(17)	(23)	(18)	(8)	(4)	(17)	(4)	(112)	(59)	(171)
1993			1,280		1,120								
	21,161	14,516	42,778	18,173	22,295	13,550	8,091	4,228	18,100	3,719	112,425	54,186	166,611
	(23)	(16)	(42)	(17)	(24)	(18)	(8)	(4)	(17)	(4)	(114)	(59)	(173)
1994	1,250		1,280										
	22,411	14,516	44,058	18,173	22,295	13,550	8,091	4,228	18,100	3,719	114,955	54,186	169,141
	(24)	(16)	(43)	(17)	(24)	(18)	(8)	(4)	(17)	(4)	(116)	(59)	(175)

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*a - Operable prior to 1980.

*b - Number of reactors.

*c - Electrical energy capacity added in year.

*d - Total capacity available in year.

additional construction of power reactors essentially ceases until at least the late 1980's; and

(2) A "high scenario", Table 3, which assumes that construction commences on a number of additional plants, including those units planned as of December 31, 1979 as well as plants for which construction has been deferred indefinitely.

Table 4 lists by region the reactors assumed to be operable in 1980. This represents the base upon which the two scenarios are built. For purposes of calculating impacts from LLW disposal, the electrical capacity of the Three Mile Island Unit 2, and the Humbolt Bay Unit 3 is conservatively included in the total assumed 1980 LWR capacity. The contributions of the Shippingport light water breeder reactor (Ref. 10) and the Fort St. Vrain high temperature graphite reactor, (Ref. 11) however, are discounted as neither ship LLW to commercial disposal facilities. The contribution of Indian Point Unit 1 has also been discounted. The reactor has been idle since late 1974 and in February 1980, the utility (Consolidated Edison) decided to decommission it rather than upgrade it to meet the latest NRC requirements on emergency core cooling systems (Ref. 12). Not shown on Table 4 is the Hanford N reactor, which is a DOE plutonium production reactor that generates electrical energy as a byproduct activity. Waste produced by this reactor is disposed by DOE and not in commercial disposal sites.

Table 5 is a listing of reactors currently under construction which, when added to those in Table 4, combine to form the low scenario. The list of reactors under construction was basically obtained from Reference 6, although the projected start-up dates were updated by more recent projections by NRC licensing staff (Ref. 9). Excluded from this list are a number of reactors listed in Reference 6 as being under construction, but have been either canceled or deferred indefinitely. These canceled and deferred units are listed in Table 6.

Table 7 is a listing of the additional nuclear generating units which could potentially be constructed by the year 2000, and which when added to those in Tables 4 and 5, forms the high scenario. Included in Table 7 are those reactors listed as "deferred indefinitely" in Table 6, as well as those reactors listed as "planned" in Reference 6. (Excluded from Table 7 are a number of reactors listed as "planned" in Reference 6, but which have been recently canceled. See Table 8) The rationale for the assumed start-up times for these units is contained in the footnotes to Table 7. Generally, the dates given were those provided in Reference 8, although in some cases, the times were so indefinite that projected start-up dates had to be conservatively postulated.

It is believed that Tables 2 and 3 effectively provide a lower and upper bound of the generating capacity which would be available by the year 2000. Of the 19 units listed in Table 7, 3 have actually been deferred indefinitely and 11 are listed in Reference 8 as having indefinite start-up dates. It would not be surprising, therefore, if no more than half of the units listed in Table 7 were actually constructed by the year 2000. The slowdown in construction of

Table 4 Nuclear Power Reactors Assumed to be in Operation by 1980

Reactor	State Located	Type	Capacity MW(e)	Start-up
Region 1:				
Beaver Valley 1	Pa	PWR	852	1976
Calvert Cliffs 1	Md	PWR	845	1974
Calvert Cliffs 2	Md	PWR	845	1974
Indian Point 1	NY	PWR	265	1962*
Indian Point 2	NY	PWR	873	1973
Haddam Neck (Conn. Yankee)	Conn	PWR	575	1967
Fitzpatrick	NY	BWR	821	1974
Indian Point 3	NY	PWR	965	1976
Maine Yankee	Maine	PWR	825	1972
Millstone 1	Conn	BWR	660	1970
Millstone 2	Conn	PWR	830	1975
Nine Mile Point 1	NY	BWR	620	1969
Oyster Creek 1	NJ	BWR	650	1969
Peach Bottom 2	Pa	BWR	1065	1973
Peach Bottom 3	Pa	BWR	1065	1974
Pilgrim 1	Ma	BWR	655	1972
R. E. Ginna	NY	PWR	470	1969
Salem 1	NJ	PWR	1090	1976
Salem 2	NJ	PWR	1115	1980
Shippingport	Pa	LWBR	60	1957**
Three Mile Island 1	Pa	PWR	819	1974
Three Mile Island 2	Pa	PWR	906	1979†
Vermont Yankee	Vt	BWR	514	1972
Yankee-Rowe	Ma	PWR	175	1960
Region 2:				
Browns Ferry 1	Al	BWR	1065	1973
Browns Ferry 2	Al	BWR	1065	1974
Browns Ferry 3	Al	BWR	1065	1976
Brunswick 1	NC	BWR	821	1976
Brunswick 2	NC	BWR	821	1975
Crystal River 3	Fla	PWR	825	1977
E. I. Hatch 1	Ga	BWR	786	1974
E. I. Hatch 2	Ga	BWR	784	1978
H. B. Robinson	SC	PWR	700	1970
J. M. Farley 1	Ala	PWR	829	1977
J. M. Farley 2	Ala	PWR	829	1980
North Anna 1	Va	PWR	907	1978
North Anna 2	Va	PWR	907	1980
Oconee 1	SC	PWR	887	1973
Oconee 2	SC	PWR	887	1973

Table 4 (Continued)

Reactor	State Located	Type	Capacity MW(e)	Start-up
Region 2: (Cont.)				
Oconee 3	SC	PWR	887	1974
St. Lucie 1	Fla	PWR	802	1976
Sequoyah 1	Tn	PWR	1148	1980
Surry 1	Va	PWR	822	1972
Surry 2	Va	PWR	822	1973
Turkey Point 3	Fla	PWR	693	1972
Turkey Point 4	Fla	PWR	693	1973
W. B. McGuire 1	NC	PWR	1180	1980
Region 3:				
Big Rock Point	Mich	BWR	72	1962
Davis-Besse 1	Ohio	PWR	906	1977
D. C. Cook 1	Mich	PWR	1054	1975
D. C. Cook 2	Mich	PWR	1100	1978
Dresden 1	Ill	BWR	200	1959
Dresden 2	Ill	BWR	794	1970
Dresden 3	Ill	BWR	794	1971
Duane Arnold 1	Iowa	BWR	538	1974
Kewannee	Wis	PWR	535	1974
La Crosse (Genoa)	Wis	BWR	50	1967
Monticello	Minn	BWR	545	1970
Palisades	Mich	PWR	805	1971
Point Beach 1	Wis	PWR	497	1970
Point Beach 2	Wis	PWR	497	1972
Prairie Island 1	Minn	PWR	530	1973
Prairie Island 2	Minn	PWR	530	1974
Quad-Cities 1	Ill	BWR	789	1972
Quad-Cities 2	Ill	BWR	789	1972
Zion 1	Ill	PWR	1040	1973
Zion 2	Ill	PWR	1040	1973
Region 4:				
Arkansas 1	Ark	PWR	850	1974
Arkansas 2	Ark	PWR	912	1978
Cooper	Nebr	BWR	778	1974
Ft. Calhoun	Nebr	PWR	457	1973
Ft. St. Vrain	Colo	HTGR	330	1974††

Table 4 (Continued)

Reactor	State Located	Type	Capacity MW(e)	Start-up
Region 5:				
Diablo Canyon 1	Ca	PWR	1084	1980
Humboldt Bay 3	Ca	BWR	65	1963#
Rancho Seco 1	Ca	PWR	918	1974
San Onotre 1	Ca	PWR	436	1967
Trojan 1	Oreg.	PWR	1130	1975
	48	PWR	57,195##	
	26	BWR		
	1	LWBR		
	1	HTGR		

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*The reactor was shutdown in October 1974 due to the inability of the plant (an early design) to meet new requirements on emergency core cooling systems (ECCS). The plant operator (Consolidated Edison) has recently decided to decommission the plant rather than upgrade the ECCS and restart the plant (Ref. 12).

**The Shippingport light water breeder reactor is operated by the Department of the Navy and does not transport low-level waste generated during operations to commercial disposal sites (Ref. 10).

†This reactor is currently closed due to an accident in March 1979. Decontamination of the plant is proceeding.

††The Fort St. Vrain high temperature graphite reactor generates, compared to light water reactors, a negligible quantity of low-level waste. What small quantities of low activity waste that have been generated are being stored onsite (Ref. 11).

#This plant was shut down by the plant operator in July 1976 for refueling, maintenance, modification of the plant to meet seismic criteria, and geologic studies of the area. These geologic studies are currently continuing (Ref. 7).

##The total includes Three Mile Island 2 and Humboldt Bay 3 but deletes Shippingport, Fort St. Vrain, and Indian Point 1.

Table 5 Nuclear Power Generating Units Under Construction in 1980

Reactor	State Located	Type	Capacity MW(e)	Assumed Start-up*
Region 1:				
Beaver Valley 2	Pa	PWR	833	1985
Hope Creek 1	NJ	BWR	1067	1985
Hope Creek 2	NJ	BWR	1067	1988
Limerick 1	Pa	BWR	1065	1983
Limerick 2	Pa	BWR	0	1985
Millstone 3	Conn	PWR	1156	1985
Nine Mile Point 2	NY	BWR	1100	1986
Seabrook 1	NH	PWR	1200	1984
Seabrook 2	NH	PWR	1200	1986
Shoreham	NY	BWR	819	1982
Susquehanna 1	Pa	BWR	1050	1981
Susquehanna 2	Pa	BWR	1050	1983
Region 2:				
A. W. Vogtle 1	Ga	PWR	1110	1985
A. W. Vogtle 2	Ga	PWR	1100	1986
BelleFonte 1	Al	PWR	1213	1982
BelleFonte 2	Al	PWR	1213	1983
Catawba 1	SC	PWR	1145	1983
Catawba 2	SC	PWR	1145	1985
Cherokee 1	SC	PWR	1280	1990
Cherokee 2	SC	PWR	1280	1992
Grand Gulf 1	Miss	BWR	1250	1981
Grand Gulf 2	Miss	BWR	1250	1985
Hartsville A1	Tn	BWR	1233	1985
Hartsville A2	Tn	BWR	1233	1986
Hartsville B1	Tn	BWR	1233	1988
Hartsville B2	Tn	BWR	1233	1989
North Anna 3	Va	PWR	907	1986
North Anna 4	Va	PWR	907	1987
Phipps Bend 1	Tn	BWR	1233	1986
Phipps Bend 2	Tn	BWR	1233	1988
River Bend 1	La	BWR	934	1985
Sequoyah 2	Tn	PWR	1148	1981
Shearon Harris 1	NC	PWR	900	1984
Shearon Harris 2	NC	PWR	900	1987
Shearon Harris 3	NC	PWR	900	1990
Shearon Harris 4	NC	PWR	900	1988
V. C. Summer 1	SC	PWR	900	1981

Table 5 (Continued)

Reactor	State Located	Type	Capacity MW(e)	Assumed Start-up*
Region 2: (Cont.)				
St. Lucie 2	Fla	PWR	810	1983
Waterford 3	La	PWR	1113	1982
Watts Bar 1	Tn	PWR	1177	1981
Watts Bar 2	Tn	PWR	1177	1982
W. B. McGuire 2	NC	PWR	1180	1982
Yellow Creek 1	Miss	PWR	1285	1985
Yellow Creek 2	Miss	PWR	1285	1988
Region 3:				
Bailly	Ind	BWR	644	1987
Braidwood 1	Ill	PWR	1120	1985
Braidwood 2	Ill	PWR	1120	1986
Byron 1	Ill	PWR	1120	1983
Byron 2	Ill	PWR	1120	1984
Callaway 1	Mo	PWR	1120	1982
Callaway 2	Mo	PWR	1120	1990
Clinton 1	Ill	BWR	933	1983
Clinton 2	Ill	BWR	933	1987
E. Fermi 2	Mich	BWR	1093	1981
La Salle 1	Ill	BWR	1078	1981
La Salle 2	Ill	BWR	1078	1982
Marble Hill 1	Ind	PWR	1130	1985
Marble Hill 2	Ind	PWR	1130	1986
Midland 1	Mich	PWR	460	1983
Midland 2	Mich	PWR	811	1983
Perry 1	Ohio	BWR	1205	1983
Perry 2	Ohio	BWR	1205	1987
W. H. Zimmer 1	Ohio	BWR	810	1981
Region 4:				
Black Fox 1	Okla	BWR	1150	1985
Black Fox 2	Okla	BWR	1150	1987
Comanche Peak 1	Tx	PWR	1111	1982
Comanche Peak 2	Tx	PWR	1111	1984
South Texas 1	Tx	PWR	1250	1983
South Texas 2	Tx	PWR	1250	1985
Wolf Creek	Kans	PWR	1150	1983

Table 5 (Continued)

Reactor	State Located	Type	Capacity MW(e)	Assumed Start-up*
Region 5:				
Diablo Canyon 2	Ca	PWR	1106	1981
Palo Verde 1	Az	PWR	1270	1982
Palo Verde 2	Az	PWR	1270	1984
Palo Verde 3	Az	PWR	1270	1986
San Onofre 2	Ca	PWR	1100	1982
San Onofre 3	Ca	PWR	1100	1983
WPPSS 1	Wash	PWR	1218	1984
WPPSS 2	Wash	BWR	1100	1982
WPPSS 3	Wash	PWR	1242	1984
WPPSS 4	Wash	PWR	1218	1985
WPPSS 5	Wash	PWR	1242	1986
	54	PWR	89,138	
	27	BWR		

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*Start-up dates are based upon projections of NRC licensing staff.

Table 6 Nuclear Power Generating Units Under Construction But Recently Canceled or Deferred Indefinitely

Reactor	State Located	Region	Type	Capacity MW(e)	Percent Constructed*
CANCELLED:					
Davis-Besse 2	Ohio	3	PWR	906	0
Davis-Besse 3	Ohio	3	PWR	906	0
Jamesport 1	NY	1	PWR	1150	0
Jamesport 2	NY	1	PWR	1150	0
Sterling 1	NY	1	PWR	1150	0
Tyrone 1	Wis	3	PWR	1100	0
				6,362	
DEFERRED INDEFINITELY:					
Forked River 1	NJ	1	PWR	1070	5.6
Cherokee 3	SC	2	PWR	1280	0
River Bend 2	La	2	BWR	934	0
				3,284	

*References 7 and 8

Table 7 Projected Start-Up Schedules for "Planned" or "Deferred" Nuclear Generating Units

Reactor	State Located	Type	Capacity MW(e)	Projected Start-Up, As of:		Assumed Start-Up
				6/30/79(Ref. 4)	6/30/80(Ref. 7)	
Region 1:						
Forked River 1	NJ	PWR	1070	1983	1986	1986*
Haven 1	NY	PWR	1250	1992	Indef.	1992**
Haven 2	NY	PWR	1250	1994	Indef.	1994**
Montague	Mass	BWR	1150	1988	Indef.	1988**
Montague	Mass	BWR	1150	1989	Indef.	1989**
Pilgrim 2	Mass	PWR	1150	1985	Indef.	1985**
Region 2:						
Cherokee	SC	PWR	1280	1990	Indef.	1990†
River Bend 2	La	BWR	934	Indef.	Indef.	1987††
T. L. Perkins 1	NC	PWR	1280	Indef.	Indef.	1992#
T. L. Perkins 2	NC	PWR	1280	Indef.	Indef.	1993#
T. L. Perkins 3	NC	PWR	1280	Indef.	Indef.	1994#
Region 3:						
Carroll County 1	Ill	PWR	1120	1990	1992	1992##
Carroll County 2	Ill	PWR	1120	1990	1993	1993##
Vandalia	Iowa	PWR	1270	Indef.	Indef.	1992¶
Region 4:						
Allens Creek 1	Texas	BWR	1150	1985	1987	1987##
Region 5:						
Pebble Springs 1	Oreg	PWR	1260	1986	1988	1988##
Pebble Springs 2	Oreg	PWR	1260	1988	1989	1989##
Skagit 1	Wash	BWR	1277	1986	Indef.	1988¶¶
Skagit 2	Wash	BWR	1277	1988	Indef.	1990¶¶
		13	PWR	15,870		
		6	BWR	6,938		
				22,808		

Footnotes: See next page.

Table 7 (Cont'd)

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*Actually deferred indefinitely. Assumed date based on Ref. 8.

**Ref. 5.

†Actually deferred indefinitely. Assumed date based on Ref. 5.

††Actually deferred indefinitely. Ref. 8 characterizes construction as 5% complete. Date given is approximately the earliest date that start-up could occur if construction were to be resumed within the next year.

#Refs. 5-8 characterize start-up dates as indefinite. Construction has not yet commenced. The dates assume that the interest in the project resumes shortly and that it requires a minimum of 12 years to receive a construction permit, build the first unit, receive an operating license, and come to initial criticality. Units 2 and 3 are assumed to follow at yearly intervals, corresponding to the utility original schedule at the time the reactors were ordered (Ref. 8).

##Ref. 8.

¶Indefinite, according to References 5-8. The percent constructed is zero (Ref. 8). The date given assumes a renewed interest within a short time period, and a length of 12 years to start-up, as in footnote #.

¶¶Original dates were 1986 and 1988, respectively, for the 2 units (Ref. 5) but the dates are currently indefinite. The utility is currently looking for an alternative site for the 2 units on the Hanford Reservation. As the hydrology, geology, etc., of the Hanford Reservation are well characterized, and a skilled labor force already exists in the area, the start-up dates are assumed to occur in a relatively short time period. The originally projected two-year stagger between the two units is retained.

Table 8 Recent Cancellations of Planned Nuclear Power Reactors

Reactor	Type	Capacity MW(e)	Projected Start-up, As of: 6/30/79 (Ref.5)	Projected Start-up, As of: 6/30/80 (Ref. 8)
Region 1:				
NEP 1	PWR	1150	1987	Canceled
NEP 2	PWR	1150	1989	Canceled
Region 3:				
Erie 1	PWR	1267	1986	Canceled
Erie 2	PWR	1267	1986	Canceled
Greenwood 2	PWR	1264	Indef.	Canceled
Greenwood 3	PWR	1264	Indef.	Canceled
Haven 1	PWR	900	1989	Canceled
Region 5:				
Palo Verde 4	PWR	1270	1987*	Canceled
Palo Verde 5	PWR	1270	1989*	Canceled
Sundesert 1(PG&E)	BWR	1200	Indef.**	Canceled
Sundesert 2(PG&E)	BWR	1200	Indef.**	Canceled
		13,202		

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*Canceled July 1979.

**Removed from "Planned" as of August 1, 1979.

and planning for new nuclear generating facilities is probably due to a number of reasons--a lessening in the demand for additional electrical generating capacity, the slowdown in the economy coupled with the large costs of constructing a nuclear power station, and public concern over the safety of nuclear power (heightened by the accident at the Three Mile Island station).

Tables 7 and 8 illustrate this slowdown in a graphic manner. As of June 30, 1979, 27 units were listed in Reference 5 as "planned," representing 32,726 MW(e) capacity. Of these 27 units, 19 had definite projected start-up dates. Only one year later, 11 of these original 27 units had been canceled (13,202 MW(e)). Out of the remaining 16 units, 3 have been deferred indefinitely; only 5 (5,910 MW(e)) are listed in Reference 8 as having definite start-up dates. Of these 5 units, applications from construction have been submitted to NRC for only 3 of them (Allens Creek Unit 1, Pebble Springs Units 1 and 2), and no construction permits for these three units have to date been issued.

It is possible that after a few years, interest in building new nuclear generating units may increase. However, it takes a number of years to construct and license a nuclear power station. Assuming that it requires a conservative minimum of 12 years from the time of initial application to start-up of a single unit, an application would have to be tendered by no later than 1988 in order to be operating by the year 2000. Therefore, only those planned units for which an application is received by NRC within the next few years could realistically contribute to the amount of LLW generated by LWRs by the year 2000. (NRC's current case load forecast is that no more than one application for a 2-unit plant--specifically the Carroll County Units--will probably be received by NRC within the next few years (Ref. 9)). Finally, any delays in the start-up times for the reactors listed in Tables 5 and 7 would act to further reduce the amount of LLW produced by LWRs by the year 2000.

2. PROJECTIONS OF NONFUEL CYCLE WASTES

Shown in Table 9 are projections of total activity, volume, and regional dependency through the year 2000 of non-fuel cycle wastes. Included are medical and bioresearch wastes, wastes from the production of medical isotopes, industrial high-activity wastes, industrial tritium wastes, and industrial low activity wastes. The projected increases in total volume and activity were generated by assuming least squares linear fits to existing data. The rationale for the volumes, activities, and regional dependence shown are listed as footnotes to the table.

3. OTHER POTENTIAL LOW LEVEL WASTE STREAMS

This section contains a discussion of waste streams which are outside of the basic streams listed in Sections 1 and 2, and which (1) are not currently being sent to LLW disposal facilities, (2) are nonroutine, or (3) are considered at this time to be too speculative. Wastes which fall into this category include those from:

Table 9 Summary of Principal Nonfuel Cycle Wastes

Medical and Bioresearch Wastes*

	Dry Solids	Scin Vials	Absorb Liq.	Biological	Acc Targets
Volume %	42%	39%	10%	9%	0.2%
Volume (in 1977)	220,000 cf	200,000 cf	52,000 cf	47,000 cf	830 cf
Activity					56%

$A_{ttl}(\text{year}) = 295\,x + 3{,}527$ Ci $x = (\text{year} - 1977)$

$V_{ttl}(\text{year}) = 45{,}184\,x + 539{,}462$ cf $x = (\text{year} - 1977)$

Regional Distribution I: 31% II: 22% III: 27% IV: 8% V: 12%

Medical Isotope Production Waste**

$A_{ttl}(\text{year}) = 7914\,y + 94484$ Ci $y = (\text{year} - 1978)$

$V_{ttl}(\text{year}) = 488\,y + 5825$ cf $y = (\text{year} - 1978)$

(Wastes are exclusively from Region I)

Industrial High-Activity Wastes (greater than 0.1 Ci/cf)†

Sealed Sources: $A_{ttl}(\text{year}) = 2{,}052\,x + 24{,}500$ Ci $x = (\text{year} - 1977)$

Other High Activity Wastes: $A_{ttl}(\text{year}) = 1{,}047\,x + 12{,}500$ Ci $x = (\text{year} - 1977)$

$V_{ttl}(\text{year}) = 176\,x + 2{,}100$ cf $x = (\text{year} - 1977)$

(Regional distribution is assumed to be the same as for Medical and Bioresearch Wastes)

Industrial Tritium Wastes

$A_{ttl}(\text{year}) = 15{,}500\,x + 184{,}500$ Ci $x = (\text{year} - 1977)$

$V_{ttl}(\text{year}) = 235\,x + 2{,}800$ cf $x = (\text{year} - 1977)$

Regional Distribution: I: 76% II: 6% III: 6% IV: 6% V: 6%

Table 9 (Cont'd)

Industrial Low-Activity Wastes (less than 0.1 Ci/cf)

Source and Special Nuclear Material:[**]

$A_{ttl}(\text{year}) = 23\,x + 280$ Ci $x = (\text{year} - 1977)$

$V_{ttl}(\text{year}) = 28{,}500\,x + 340{,}000$ cf $x = (\text{year} - 1977)$

Regional Distribution: I: 50% II: 10% III: 20% IV: 10% V: 10%

Other Low-Activity Wastes:[**]

$A_{ttl}(\text{year}) = 9.2\,x + 110$ Ci $x = (\text{year} - 1977)$

$V_{ttl}(\text{year}) = 10{,}900\,x + 130{,}000$ cf $x = (\text{year} - 1977)$

Regional Distribution: I: 30% II: 20% III: 30% IV: 10% V: 10%
(Similar to Medical and Bioresearch waste Distribution)

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[*] Medical and bioresearch wastes were derived based upon NUREG/CR-0028 (Ref. 13) and upon its follow-up report, NUREG/CR-1137 (Ref. 14). The volume and activity of the waste sampled in NUREG/CR-1137 represented about half of the institutional waste sent to disposal sites in 1977. The regional distribution of medical and bioresearch wastes were assumed to correspond to the NUREG/CR-1137 survey population.

[**] Medical isotope production waste was based on 1977 burial records of the Maxey Flats, Kentucky disposal site (Ref. 2). Wastes from this source are from Region 1.

[†] Estimates of industrial high activity wastes and sealed sources were based on 1977 records from the Maxey Flats disposal facility, (Ref. 2) and doubled. The regional distribution was assumed to be the same as that of the medical waste stream.

[††] Industrial tritium wastes were estimated using the responses from I&E Bulletin 79-19 as a guide (Ref. 15). Using this data, it was determined that approximately 140,000 curies of tritium was reported in 1978-generated wastes and that one shipper disposed of nearly all of the tritium. Of this shipper's waste volume, 10% was assumed to contain the tritium. However, owing to the limited extent of the I&E survey, waste quantities were extrapolated to account for wastes which may have been disposed of by facilities which are licensed strictly by Agreement States and would not be included in the I&E bulletin survey population. Since the major shipper of tritium waste was also identified as the major purchaser of tritium in the U.S., it was assumed that a total of 200,000 curies was disposed of in 1978. Three-quarters of the tritium (76%) was assumed to be generated in Region I, the region with the major user of tritium, and assumed that the remainder was divided equally among the other four regions. In this way, the waste stream is represented in each region.

Table 9 (Cont'd)

NMIS records (Ref. 16) were also analyzed to refine the breakdown of high activity tritium waste disposals in the recent past. This review indicates that the projections do not overlook any major generators of tritium wastes. However, these waste generators are likely to operate at nonuniform rates and potential shifts in operations (plants opening in new region, plants closing, etc.) may result in wide fluctuations in future waste generation rates. Therefore, the conservative approach for the projection--i.e., to link most of the waste with the Region I and to include a smaller fraction from each of the other regions as a representative distribution so that this potentially important waste stream is not omitted entirely--is believed to be practical for the purposes of this document.

#Estimates of industrial low activity wastes were based upon October 1979 burial records at the Barnwell, South Carolina disposal facility (Ref. 17). Raw data for source and special nuclear material wastes was multiplied by a factor of 0.5 (fraction of 1979 source and special nuclear material waste volume accepted by Barnwell), multiplied by 12 (number of months in a year), and multiplied by a factor of 0.86 to convert 1979 numbers to 1977 rates. Raw data for other low-activity waste was multiplied by 0.7 (fraction of 1979 source and special nuclear material waste volume accepted by Barnwell), multiplied by 12, and multiplied by a factor of 0.86.

- U.S. Government operations;
- Decontamination of the Three Mile Island Unit 2 nuclear operating station;
- Wastes from recycle of nuclear fuel, including operations at a commercial fuel reprocessing plant as well as operations at a mixed oxide fuel fabrication plant;
- Operations at an independent spent fuel storage installation;
- Decommissioning of uranium fuel cycle facilities;
- Transuranic-contaminated wastes.

These potential waste streams are discussed in the following subsections.

3.1 U.S. Government Operations

Since the first commercial LLW disposal facilities were opened in 1962 (at Beatty, Nevada and Maxey Flats, Kentucky), considerable volumes of wastes generated by U.S. government agencies have been shipped to commercial sites for disposal. Most of this waste was produced by laboratories operated by or under contract to the Atomic Energy Commission (AEC). One of the intents of this practice was to help provide some initial business to the then fledgling commercial disposal industry. This practice was continued by the AEC's successors, the Energy Research and Development Administration (ERDA) and the Department of Energy (DOE), until 1979, when it was discontinued by DOE to help alleviate the shortage in commercial LLW disposal capacity (Ref. 18). Currently, all wastes generated by DOE facilities are disposed in DOE disposal sites. Small quantities of wastes produced by other government agencies such as the Department of Defense (unclassified waste only) or the U.S. Department of Agriculture, however, are still occasionally shipped to commercial LLW disposal facilities.

3.2 Three Mile Island Unit 2 Decontamination

The March 28, 1979 accident at the Three Mile Island (TMI) Unit 2 nuclear power station has resulted in damage to the reactor core as well as generation of significant quantities of contaminated water. Removal of damaged core components and other plant equipment, processing of the contaminated water, and decontamination of contaminated plant equipment and surfaces is projected to take about 5 to 9 years. Over this time period, radioactive wastes in various solid forms will be generated. NRC has prepared and published a programmatic environmental impact statement (PEIS) related to decontamination and disposal of radioactive wastes resulting from the accident (Ref. 19). In this document, NRC staff investigated a wide variety of decontamination and waste processing alternatives. Bounding (probable minimum and probable maximum) volumes of wastes projected to be deliverd to LLW disposal facilities as a result of these decontamination operations and waste processing alternatives have been set out in the PEIS. A summary of these projections excerpted from the PEIS is included in this appendix as Table 10.

The range in projected volumes reflects the fact that the actual volumes of waste generated will depend upon decisions regarding which decontamination and waste treatment alternatives are implemented. In many cases, such decisions will be made as the decontamination operations progress.

The decontamination and waste treatment operations will also generate some volumes of waste that will not be disposed at near-surface disposal facilities. These include fuel or pieces of fuel removed from the reactor, other transuranic-contaminated wastes (if generated), and some very high specific activity ion-exchange media wastes generated as a result of treating contaminated reactor building water.

3.3 Uranium Fuel Recycle Wastes

The current policy of the United States is to defer processing of spent light water reactor fuel. Spent uranium fuel removed from nuclear power reactors is presently stored without attempting to extract the residual fissile uranium and plutonium for reuse. If the national policy were to change, however, and recycle operations were implemented, then additional waste streams would result from reprocessing operations as well as from fabrication of mixed oxide (MOX, a blend of UO_2 and PuO_2) fuel for use in light water reactors.

Potential volumes and activities of waste streams which would be generated from recycle of uranium fuel are speculative at this time. Such waste streams are not being produced today and even if the national policy regarding recycle of uranium fuel were to change within a short time period, it would still be several years before significant quantities of wastes would be produced. Of the three commercial reprocessing plants that have been constructed in the United States--that is, at West Valley, New York, Morris, Illinois, and Barnwell, South Carolina--only the West Valley plant has ever operated. This plant, however, has not operated since 1972. None of the three facilities could operate today without extensive modification. Of the three, the Barnwell facility would require the least construction--principally construction of a waste solidification facility, a facility for conversion of liquid plutonium nitrate to solid plutonium oxide, and probable addition of additional airborne effluent treatment systems. The Morris facility would require major changes in the design of the processing operations. The West Valley plant would require considerable modification to meet seismic and radiation shielding requirements. In addition, the operator of the West Valley plant--Nuclear Fuel Services, Inc.--has previously (1976) expressed a desire not to continue in the reprocessing business.

There are currently no large scale commerical facilities for fabrication of mixed oxide fuel, although a number of small scale commercial laboratories and research facilities are in existence that have in the past fabricated small batches of MOX fuel for experimental use in LWRs. Such large scale facilities would have to be constructed.

Table 10 Volumes of TMI-2 Packaged Solid Waste to be Disposed of at a Commercial Low-Level Waste Disposal Site

		Best-Case Conditions		Worst-Case Conditions	
Type of Package	Package Volume (ft^3)	Number of Packages	Shipped Volume (ft^3)	Number of Packages	Shipped Volume (ft^3)
55-Gallon Drums					
Low activity	7.5	3,200	24,000	15,400	115,500
Intermediate activity	7.5	502	3,765	1,707	12,800
LSA Boxes*					
Low activity	80	1,042	83,360	2,128	170,240
Contaminated Equipment	70	86	6,020	293	20,510
and Hardware, Mirror Insulation	80	53	4,240	-	-
EPICOR II Resins					
1st stage**	50	49	2,450	49	2,450
2nd stage	50	14	700	14	700
3rd stage	175	6	1,050	6	1,050
Reactor Building Cleanup					
Filters†	10	11	110	11	110
2nd stage	50	2	100	4	200
3rd stage	190	1	190	2	380
Primary System Cleanup†					
Filters††	10/7.5/150	16	990	57	1,340
2nd stage	50	4	200	44	2,200
3rd stage	190	3	570	12	2,280
Total		128,260			329,760

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*Low specific activity.

**Will require special disposal procedures (e.g., deeper burial) if disposed of at a commercial disposal site.

†If any of these wastes contain fuel debris or greater than 10 nCi/gm transuranic materials, they would not be accepted at a commercial LLW facility.

††Primary system cleanup generated 3 filter types.

Licensing requirements for construction of new uranium recycle facilities or modification of old ones would also delay operation of such facilities. Such licensing requirements would include regulatory review, publication of environmental impact statements and other environmental assessments, and probable hearings.

Potential volumes and activities of wastes that would be generated by uranium recycle operations have been estimated by a number of groups, including NRC (Ref. 20), DOE (Ref. 21), and the national laboratories (Refs. 22, 23). However, as stated previously, the timing of the generation of such wastes is very speculative. In any case, much of the waste which would be generated by a reprocessing plant or a MOX fabrication plant would be contaminated or suspected of being contaminated with transuranic radionuclides in excess of 10 nCi/gm and would not be acceptable at existing commercial disposal facilities. (Also see Section 3.6.)

3.4 Independent Spent Fuel Storage Facilities

As there is at this time neither an ongoing nuclear fuel reprocessing industry or a federal high-level waste repository, spent nuclear fuel removed from nuclear power plants must be safely stored. This spent fuel is currently being stored in fuel pools located within nuclear power stations as well as within two unused reprocessing plants: the General Electric (GE) reprocessing plant located near Morris, Illinois, and the Nuclear Fuel Services (NFS) reprocessing plant located near West Valley, New York. As of the end of 1979, the total amount of spent fuel stored in the Morris and West Valley plants corresponded to about 9 percent of the total U.S. commercial inventory of stored LWR fuel (Ref. 24).

The existing storage capacity for spent LWR fuel is not likely to be adequate until a repository or ongoing fuel reprocessing industry is developed. Additional storage capacity can be developed through densification of existing fuel storage capacity or construction of new pools at power plants, expansion of storage capacity at the West Valley and Morris facilities, use of the fuel storage capacity at the uncompleted Barnwell, South Carolina reprocessing plant, or construction of new independent spent fuel storage facilities. Recently, NRC published a new set of regulations, 10 CFR Part 72, which establish rules for licensing independent spent fuel storage facilities, if and when they are constructed (Ref. 25).

Wastes from storing spent LWR fuel would primarily arise from treatment of the storage basin water, receiving and unloading spent fuel, and maintenance of plant ventilation systems. These wastes include spent resins, filter sludges, and miscellaneous trash, and are similar in composition to wastes produced from other light water reactor operations.

Waste volumes generated to the year 2000 from LWR fuel storage are expected to be relatively small. Most of the waste volumes generated would continue to be included with other wastes shipped from power plants. Only small quantities of wastes are produced by the current two facilities practicing away-from-reactor storage. LLW generated at the West Valley plant is disposed onsite at the colocated LLW disposal site. At the Morris plant, low specific activity trash is currently shipped to a LLW disposal site. Liquid wastes and filter sludges generated from backflushing and regenerating the fuel pool water filter system are stored in a large (680,000 gallon capacity) low activity waste (LAW) tank. The LAW tank was originally constructed and intended to store low-level liquids generated during the operation of the reprocessing plant. Eventually, General Electric plans to install a solidification system to solidify the liquids and other wet wastes and send the solidified waste material to a LLW disposal facility (Ref. 26).

DOE has estimated the annual volumes of waste that could be generated from a large (3000 MTHM) independent spent fuel storage installation, assuming one is constructed (Ref. 21). These volumes are listed in Table 11 and are based upon a conservative (in terms of waste generation) assumption of an operating mode in which one-sixth of the storage capacity is replaced each year. The total volume of waste produced from such a large facility is comparable to the annual generation rate of a single 1000 MW(e) light water reactor.

At this time, NRC has not received any application for construction and operation of an independent spent fuel storage facility. The timing for future construction of a storage facility (and associated waste volume generation) is speculative.

3.5 Decommissioning of Nuclear Fuel-Cycle Facilities

Nuclear fuel cycle facilities will eventually reach the end of their useful lives and would then be considered candidates for decontamination and decommissioning. In some cases, decontamination and decommissioning activities may merely involve removing enough residual contamination to allow safe modification and reuse as a nuclear facility. In other cases, the facility may be decontaminated to the point that it can be released for unrestricted use.

The timing and extent of potential decontamination and decommissioning activities at a nuclear installation are believed to be speculative. The timing and extent of decommissioning activities may depend upon factors other than the useful life of a nuclear facility--i.e., upon economic decisions or regulatory requirements. It is considered unlikely that significant volumes of wastes from decommissioning nuclear fuel cycle facilities will be produced prior to the year 2000.

3.5.1 Decommissioning of Light Water Reactors

A large source of waste to be generated in the future will be from decommissioning light water power reactors. The volumes and activities which will be produced are uncertain, and depend upon such factors as the length of service life of a plant prior to decommissioning, the size and design of a plant, and the decommissioning mode undertaken (e.g., immediate dismantlement after shutdown vs. deferring dismantlement for up to several years following shutdown).

Table 11 Estimated Annual Waste Volumes Generated From Assumed Operation of a 3,000 MTHM Spent Fuel Storage Facility

	Volumes	
Waste Category	(m^3)	(ft^3)
Compactible and Combustible Wastes:		
. Combustible trash	630	22,245
. Ventilation Filters	23	812
Liquids and other wet wastes:		
. Bead resins	2	71
. Filter precoat sludge	8	282
. Sulfate concentrate	7	247
. Miscellaneous solution concentrates	10	353
Noncombustible material:		
. Noncombustible trash	51	1,800
. Failed equipment	19	671
Total:	750	26,481

Forty calendar years of operating life is generally considered an appropriate assumption for the length of service life of a large modern LWR prior to decommissioning. Based upon this assumption, Table 12 was generated, illustrating a number of reactors which can be postulated to be candidates for decommissioning in the general neighborhood of the year 2000. Using this criteria, only two reactors--Shippingport and Dresden 1--would be projected for decommissioning prior to the year 2000. However, as discussed below, such projections are uncertain and may depend upon factors other than the assumed 40 year operating life of the units.

The first 6 plants listed (plus the La Crosse unit) are early low power units generally constructed as demonstration projects forerunning larger, more economical to operate units with capacities on the order of several hundred to a thousand MW(e). Although utilities would generally prefer to keep the older units operable for as long as they are cost-effective, costs of upgrading the older units to meet new NRC safety requirements may result in some of the older plants being decommissioned prior to the year 2000, and prior to the end of their otherwise servicable lives. Short discussions of the present and possible future status of these early units follow:

- Shippingport, the first nuclear power reactor constructed, is presently operated by the Department of Navy for research into the possible utilization of a thorium fuel cycle. The reactor is expected to operate for as long as it is useful as a research and test vehicle, and as its operation is not related to its cost-effectiveness as a power generator.

- Dresden 1, was the first BWR built for commercial use. The operating utility, Commonwealth Edison, is currently putting the plant through a full scale primary coolant system decontamination procedure, with the intention of continued operation of the unit.

- Yankee-Rowe. This unit continues to generate electricity with no major apparent problems. In 1979, its capacity factor was 81%.

- Indian Point 1. This unit was shut down in October 1974 by its utility, Consolidated Edison, due to inability to meet new NRC requirements on emergency core cooling systems (ECCS). Consolidated Edison has recently determined that the cost of upgrading the plant to meet the new ECCS and other requirements would be greatly in excess of the possible economic gain, and have announced their intention of decommissioning the unit. The proposed timing and mode of decommissioning (safe storage, immediate dismantlement, or deferred dismantlement) however, has not yet been determined.

- Big Rock Point. This BWR is presently in operation, although its status is somewhat uncertain. The utility, Consumers Power Company, is currently evaluating the costs of recent modifications requested by NRC.

Table 12 Power Reactors Assumed to Be Eligible for Decommissioning After Forty Years Operation

Name	Located	Type	Power		Start-up	Postulated Shutdown
			WW(e)	MW(t)		
Shippingport	Shippingport, Pa (I)*	PWR	60	236	1957	1997
Dresden 1	Morris, Ill. (III)	BWR	200	700	1959	1999
Yankee Rowe	Rowe, Mass. (I)	PWR	175	600	1960	2000
Indian Point 1	Buchanan, NY (I)	PWR	265	615	1962	2002
Big Rock Point	Big Rock Point, Mich. (III)	BWR	72	240	1962	2002
Humboldt Bay	Eureka, Calif. (V)	BWR	65	242	1963	2003
Haddam Neck	Haddam Neck, Conn. (I)	PWR	575	1825	1967	2007
LaCrosse	LaCrosse, Wis. (III)	BWR	50	165	1967	2007
San Onofre	San Clemente, Calif. (V)	PWR	436	1347	1967	2007
Oyster Creek	Toms River, NJ (I)	BWR	650	1930	1969	2009
Nile Mile Point 1	Scriba, NY (I)	BWR	620	1850	1969	2009
R. E. Ginna 1	Ontario, NY (I)	PWR	470	1520	1969	2009
Millstone 1	Water Ford, Conn. (I)	BWR	660	2011	1970	2010
H. B. Robinson	Hartsville, SC (II)	PWR	700	2200	1970	2010
Dresden 2	Morris, Ill. (III)	BWR	794	2527	1970	2010
Monticello	Monticello, Minn (III)	BWR	545	1670	1970	2010
Point Beach 1	Two Creeks, Wis. (III)	PWR	497	1518	1970	2010

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*Region.

- Humboldt Bay. This unit has been shut down since July 1976 for refueling, maintenance, seismic modifications, and area geologic studies requested by NRC. Geologic studies are continuing and the future of the unit is uncertain.

- La Crosse (Genoa) This unit is currently operating although the utility, Dairyland Power Cooperative, was issued a show cause order in February 1980 by NRC regarding installation of a site dewatering system. The purpose of the system would be to preclude potential liquefaction of site soil during a design basis earthquake. No information is currently available regarding the effect of the potential costs of this new requirement on the continued operation of the unit.

NRC has recently completed a pair of studies on the technology, safety, and costs of decommissioning a large 1175 MW(e) PWR (NUREG/CR-0130, Ref. 27) and a large 1155 MW(e) BWR (NUREG/CR-0672, Ref. 28). Projected volumes and activities of waste produced by these operations are provided in Tables 13 and 14. Additional data regarding the assumptions used in the projections are also provided as footnotes to the tables.

It is believed that the projected volumes and activities in Tables 12 and 13 conservatively bound the potential impacts from decommissioning LWR's through the year 2000. Reactors potentially dismantled during this period are expected to be considerably smaller in capacity and length of operation than the reactors used as models for the decommissioning studies.

3.5.2 Decommissioning of Uranium Fuel Fabrication Plants

A relatively minor source of decommissioning waste, compared to decommissioning light water reactors, will be wastes from decommissioning uranium fuel fabrication facilities. In the current uranium fuel cycle, yellowcake produced from uranium milling operations is shipped to a conversion plant where the yellowcake is converted to UF_6 and shipped to a gaseous diffusion plant for enrichment in U-235. Following enrichment, the UF_6 is then reconverted to UO_2 and fabricated into fuel assemblies at a uranium fuel fabrication plant. A list of currently operating uranium fuel fabrication plants is included as Table 15.

Decommissioning a fuel fabrication facility is not expected to generate significant (compared with decommissioning a light water reactor) volumes or activities of waste. Potential waste volumes from decommissioning a relatively large fuel fabriciation facility plant have been estimated by Pacific Northwest Laboratories (PNL), and estimates based upon this study are summarized in Table 16 (Ref. 29). In the PNL study, a model plant is assumed which is based upon an existing facility operated by the General Electric Company in Wilmington, North Carolina. The plant is assumed to be operated for 40 years at a production rate of 1000 metric tons of uranium oxide fuel per year. Feed to the plant is UF_6. All of the calcium flouride wastes and other conversion process sludges which are generated during the process converting UF_6 to UO_2 are assumed to be stored on-site in large lagoons until decommissioning.

Table 13 Summary of Wastes From Decommissioning
a 1175 MW(e) Reference PWR*

Waste Stream	Volume (m^3)	ft^3	Activity(Ci)
Activated metal**	(484)	17,085	4,841,320
Activated concrete**	(707)	24,957	2,000
Contaminated metal†	(5,465)	192,915	900
Contaminated concrete**	(10,613)	374,745	100
Dry solid waste (trash)††	(1,418)¶¶	50,625¶¶	-
Spent resins#	(30)¶¶	1,060¶¶	42,000
Filter cartridges##	(8.9)§	315§	5,000
Evaporator bottoms¶	(133)¶¶	4,696¶¶	-

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*The model for the reference facility is the Portland General Electric Company's Trojan nuclear plant (1175 MW(e), 3500 MW(t)), which uses a Westinghouse four-loop nuclear steam supply system. The waste volumes and activities are projected from an assumption of immediate dismantlement following 40 calendar years at 75% of full power operation, or 30 effective full power years (EFPY). The dismantlement is projected to require 4 years of effort, in addition to two years of planning (Ref. 27).

**Activities and volumes for activated metals and concrete are provided in Table G.4-2 and Table G.4-3 of Ref. 27. These are the following:

Component	Volume ft^3	(m^3)	Activity (Ci)	Specific Activity Ci/ft^3	(Ci/m^3)
Activated Metals:					
. Pressure vessel cylindrical wall	3,800	(108)	19,170	5	(178)
. Vessel head	2,000	(57)	<10	.005	(.18)
. Vessel bottom	2,000	(57)	<10	.005	(.18)
. Upper core support assembly	400	(11)	<10	.025	(.91)
. Upper support columns	400	(11)	<100	.25	(9.1)
. Upper core barrel	200	(6)	<1,000	5	(167)
. Upper core grid plate	500	(14)	24,310	49	(1,736)
. Guide tubes	600	(17)	<100	.17	(6)
. Lower core barrel	3,200	(91)	651,000	203	(7,154)

Table 13 (Cont'd)

Component	Volume		Activity (Ci)	Specific Activity	
	ft^3	(m^3)		Ci/ft^3	(Ci/m^3)
Activated Metals (Cont'd)					
. Thermal shields	600	(17)	146,100	244	(8,594)
. Core shroud	400	(11)	3,431,100	8,578	(311,909)
. Lower grid plate	500	(14)	553,400	1,107	(39,529)
. Lower support columns	100	(3)	10,000	10	(333)
. Lower core forging	1,100	(31)	2,500	2.3	(81)
. Misc. internals	800	(23)	2,000	2.5	(87)
. Reactor cavity liner	512	(15)	<10	.02	(.7)
Subtotal:	17,112	(485)	4,841,320		
Activated concrete:					
. Bio. shield concrete	24,960	(707)	<2,000		
Subtotal	24,960	(707)	<2,000		
Total	42,072	(1191)	4,843,320		

Note that the concentrations range from .18 Ci/m^3 to 311,909 Ci/m^3 (six orders of magnitude). Burial volumes include the disposal container after the activated metal components and concrete have been cut into managable pieces.

†Volumes for contaminated material are obtained from Tables G.4-2, G.4-4, and G.4-5. The activity contained in the contaminated metal and concrete is postulated in Ref. 27 to be less than 1,000 curies, total. The majority of the contamination is contained in the metal components (about 99%). In addition, there is approximately 10 times as much contaminated metal (by volume) than contaminated concrete from decommissioning a BWR. (See NUREG/CR-0672). Assuming the ratio of the specific activities of metal and concrete is the same for a BWR as for a PWR, one obtains:

$$\frac{\frac{8574}{549,249}}{\frac{55}{59,187}} = \frac{\frac{x}{192,915}}{\frac{y}{374,745}}$$

x = Ci of contaminated PWR metal

;

y = Ci of contaminated BWR concrete

and $x + y = 1000$ Ci

Solving this, one obtains

$x = 899$ Ci.

$y = 101$ Ci.

Table 13 (Cont'd)

Therefore, 10% of the total contaminated component is assumed to be assigned to concrete, the rest to metal.

††Volumes shown are "as generated" (unprocessed) and are obtained from p. G-33. (1,350 drums x 5 x 0.21m^3/drum = 1418 m^3, or 50,625 ft^3). The waste is characterized in the report as compactable and combustible. In the report, the waste is assured to be compacted 5-fold and shipped in 1,350 drums; 420 drums (31%) are assumed in the report to require shipment in shielded casks due to radiation readings. Otherwise, no activity levels are given.

#Volumes of activities of spent resins from decommissioning are obtained from Table G.4-6 and p. G-35--i.e., 30 m^3 (1,060 ft^3) having an activity of 42,000 Ci.

##Information for cartridge filters is obtained from Table G.4-6 and p. G-35. Forty-two cartridge filters are assumed to be generated, at 120 Ci contained activity per filters (5,000 Ci total). The volumes given in the report (315 ft^3) are generated assuming that each filter is solidified in concrete in a 55-gallon drum. No information is given regarding the volumes of the cartridge filters themselves. Therefore, the volumes in the table are for solidified volumes. (The dimensions of a cartridge filter are variable, in any case, depending upon the design.)

¶Information regarding the volumes of evaporator bottoms generated are on p. G-35. Volumes and gross chemical characteristics of the evaporator bottoms are:

decontamination chemicals	57 m^3
borated water	38 m^3
rinsing, flushing, & washing water	38 m^3
	133 m^3 (4696 ft^3)

The report also assumes that the bottoms are solidified to 9400 ft^3 of solidified volume (VIF = 2.0). No information is given regarding the activity, although the following is asumed in the report for surface radiation readings of the solidified waster containers:

22	containers @	150 R/hr
45	" "	0.21 R/hr
27	" "	<0.2 R/hr
94	containers total	

¶¶As-generated volumes, prior to further processing or solidification.

§Volumes as-solidified in 55-gallon drums.

Table 14 Summary of Wastes from Decommissioning a 1155 MW(e) Reference BWR[(a)]

Waste Stream	Volume (m^3)	ft^3	Activity (Ci)
Activated metal (b)	(138)	4,873	6,552,310
Activated concerete (c)	(90)	3,178	170
Contaminated metal (d)	(15,543)	549,249	8,574
Contaminated concrete (d)	(1,676)	59,187	55
Dry soild waste (trash) (e)	(3,386) (i)	119,526 (i)	-
Spent resins (f)	(42) (i)	1,483 (i)	228
Filter cartridges (g)	-	-	-
Evaporator bottoms (h)	(519) (i)	18,321 (i)	43,753

(a) The model for the reference facility is the Washington Public Power Supply Systems Nuclear Project No. 2 (WPPSS-2) at Hanford, Washington. The 1155 MW(e) unit (3320 MW(t)) uses a General Electric BWR/5 nuclear steam supply system and the plant uses a Mark-II containment. The unit is expected to start operation in 1982. The volumes and activities are projected from an assumption of immediate dismantlement following 40 calendar years of operation at 75% of full power, or 30 effective full power years (EFPY). The dismantlement is projected to require 3 1/2 years to complete. (Ref. 28).

(b) Volumes for activated metal are taken from Table I.3-2. Total activity for activated metal is taken from Table I.3-3. Relative volumes and activities for various activated metal components include:

Component	Burial Volume (m^3)	Activity (Ci)	Concentration (Ci/m^3)
Steam separator assembly	10	9,600	960
Fuel support pieces	5	700	140
Control rods and in-core instruments	15	189,000	12,600
Control rod guide tubes	4	100	25
Jet pump assemblies	14	20,000	1,429
Top fuel guide	24	30,100	1,254
Core support plate	11	650	59
Core shroud	47	6,300,000	134,043
Reactor vessel wall	8	2,160	46
	138	6,552,310	

Note that the concentrations range from 25 Ci/m^3 to 134,043 Ci/m^3 (Four orders of magnitude). Burial volumes include the disposal container after the activated metal components have been cut into managable pieces.

For the most part, activated metal is assumed in the study to be packaged in specially constructed steel boxes.

(c)Volumes and activity for activated concrete are taken from Tables I.3-2 and I.3-3. In Table I.3-3, the activated concrete is referred to as the sacrificial shield. The volume and activity shown for the sacrificial shield include only the neutron activated portion of the shield. The remainder is shipped as contaminated material.

(d)Total volumes for contaminated metal and contaminated concrete are taken from Table I.3-2 and Table I.3-4. Total activity for contaminated metal and contaminated concrete is obtained by summing the values in Tables E.2-5, E.2-7, and E.2-10, which adds to 8629 Ci. Of this, 8515 Ci is definitely attributed to contaminated metal. Of the remaining 114 Ci listed in Table E.2-10, at least 59 Ci is attributed to contaminated metal. The remaining 55 Ci (listed under "Primary Containment" in Table E.2-10) is assumed to be consigned to the contaminated concrete.

(e)Volumes for dry solid wastes are taken from p. I-41. These wastes are characterized as including discarded plastic sheeting, rags, and anti-contamination clothing, and total 3,386 m^3 uncompacted (119,526 ft^3). The dry wastes are characterized in the report as being compacted in a five-fold manner prior to shipment and are therefore assumed to be combustible. There is no data on contained activity, although it can be noted that the report assumes that 84% of the compacted waste requires shielding.

(f)The category "spent resins" actually includes filter sludges and resins. Volumes and activities are taken from p. H-23 and Table H.5-10. A total volume of 42 m^3 of filter sludges and resins is assumed to be generated, solidified in concrete to 54 m^3, and packaged in 19 steel cask liners, each of which is assumed to have an average radioactivity content of 12 Ci. This leads to an assumed solidification volume increase factor of 1.3 and a total activity of 12 x 19 = 228 Ci. Table H.5-10 also assumes about 80% use of space in the disposal liners.

(g)Filter cartridges are not used in the BWR wet waste treatment system.

(h)Evaporator bottoms volumes and activities are determined from the information obtained from p. H-23 and Table H.5-10. Four groups of evaporator bottoms can be postulated from the data, each group having differing volumes and activities:

Group #	Volume (m^3)	Activity (Ci)	Remarks
1	$\frac{52}{225}$ (438) = 101[(C)]	31,200	(A)
2	$\frac{108}{225}$ (438) = 210	1,296	(A)
3	$\frac{65}{225}$ (438) = 127	156	(A)
4	$\frac{95}{1.18}$(438) = 81[(D)]	101	(B)
Total	519	32,753	

(A) Chemical decontamination solutions containing EDTA/citrox.

(B) Chemical decontamination solutions containing phosphonic acid which are assumed to be neutralized with sodium hydroxide. The neturalizing process results in a solution with an estimated 12-wt% solids concentration.

(C) Volumes are estimated by the ratio of the number of casks per group to the total number of casks, times the total as-generated volume.

(D) As-generated volume calculated from the volume of solidified solution divided by the same volume increase factor used in the report for groups 1, 2, and 3.

(i) Unprocessed volumes. See footnotes (e) through (h).

Table 15 Current LWR Fuel Fabrication Industry*

Licensee	Plant Location (NRC Region)	Plant Feed Material	Plant Product	Plant Capacity (MTU/yr)	
				Current	Estimated 1985
Babcock & Wilcox	Lynchburg, VA (2)	UO_2 pellets	Fuel assys	230	830**
Babcock & Wilcox	Apollo, PA (1)	UF_6			†
Combustion Engineering	Hematite, MC (3)	UF_6	UO_2 powder or pellets		††
Combustion Engineering	Windsor, CT (1)	UO_2 powder	Fuel assys	150	150
Exxon Nuclear	Richland, WA (5)	UF_6	Fuel assys	665	1,030#
General Electric	Wilmington, NC (2)	UF_6	Fuel assys	1,500	1,500
Westinghouse Electric	Columbia, SC (2)	UF_6	Fuel assys	750	1,600

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*In addition to the fuel fabrication plants listed in the table, there are two existing conversion facilities which convert yellowcake produced during uranium milling operations into UF_6 for enrichment. These two facilities are located at Metropolis, Illinois and Sequoyah, Oklahoma.

**Babcock and Wilcox (B&W) plans to expand operations to increase capacity to 1,200 MTU/yr by the early 1990s. The capacity listed in the table is an interpolation of present and future capacity. In addition, a UF_6 to UO_2 conversion operation will be added as well as a UO_2 pelletizing operation.

†Currently, the B&W Apollo plant converts UF_6 to UO_2 powder and ships the UO_2 to its Lynchburg plant for fabrication into fuel assemblies.

††The Combustion Engineering (CE) Hematite plant produces UO_2 pellets or powder which are then transferred to the CE Windsor plant for fabrication into fuel assemblies.

#Expanded to 1,030 MTU/yr in 1980.

As shown in Table 16, the calculated volumes of wastes generated from decommissioning the plant include trash and other miscellaneous material from decontaminating buildings and other facilities, as well as several thousand cubic meters of low activity bulk material such as CaF_2. The total quantity of uranium contained in the 1091 m^3 of miscellaneous trash is projected by PNL to be approximately 270 kg. The total quantity of uranium contained in the 27,000 m^3 of low activity material is also expected to be low. This corresponds to a generation rate of approximately 0.63 m^3 of dry CaF_2 generated per MT of UO_2.

These estimated quantities should be used with some care. For example, the timing of future fabrication plant decommissioning activities is very speculative, and would probably depend more on economic than safety considerations. Although the amount of fuel fabrication capacity would naturally be a function of nuclear power plant capacity, the total potential decommissioning volume would not be expected to be strong function of capacity. Rather, total volumes of waste material obtained from decommissioning fuel fabrication plants would be a function of the number of plants operating and the design of individual plants rather than a function of the total throughput of uranium feed through the plants.

Projected volumes of CaF_2 and other chemical sludges produced from UF_6 conversion are also considered to be speculative. The rate of production of UF_6 conversion sludges at a facility is a strong function of the design of the conversion process used at the facility. Space limitations at an individual plant may result in process sludges being transferred to LLW disposal sites during plant operation rather than being left onsite in lagoons for later consideration. Existing and future sludge lagoons at fabrication facilities may, rather than being collected and delivered to a LLW disposal site during decommissioning, be disposed in-place or treated to recover the contained uranium.

3.5.3 Decommissioning Uranium Fuel Recycle Facilities

Should uranium recycling be eventually adopted as a national policy, then uranium recycle facilities which would be constructed would eventually require decommissioning. Such decommissioning activities would occur relatively remote from today--at least beyond the year 2000. Volumes and activities of wastes that would result in decommissioning some reference uranium fuel recycle facilities have been estimated by PNL. In NUREG-0278, the technology, safety, and costs of decommissioning a 1500 MTHM/year fuel reprocessing plant are assessed, using the uncompleted Barnwell, South Carolina reprocessing plant owned by Allied-General Nuclear Services as a model (Ref. 30). In NUREG/CR-0129, the technology, safety and costs of decommissioning a small mixed oxide fuel fabrication plant are assessed (Ref. 34).

A potential source of wastes which may be generated in the next few years would be from decommissioning the Nuclear Fuel Services (NFS) reprocessing plant located in West Valley, New York. The reprocessing plant has not operated since 1972 and NFS announced in 1976 their intention to withdraw from the nuclear fuel reprocessing business. The eventual disposition of the

Table 16 Waste Volumes Generated From Decommissioning a Model 1000 MT UO_2/yr Fuel Fabrication Plant

Wastes from decommissioning buildings and other site structures:

Waste Category	Volume	
	m³	ft³
. Hoods, Equipment and Components	764.4	26,991
. Pipe, Conduit, Duct, Trays, Fixtures, etc.	118.52	4,185
. HEPA and Roughing Filters	51.66	1,824
. Concrete Rubble	39.66	1,400
. Contaminated Liner and Soil Materials	91.0	3,213
. Miscellaneous	25	883
Total:	1,091	38,496

Low-activity bulk solids:

Waste Category	Volume*	
	m³	ft³
. Chemical Sludge	1,282	45,283
. Contaminated CaF²	25,296	893,208
. Other Miscellaneous Contaminated Material	3,206	13,208
Total:	29,784	1,051,699

*Calculated from data in Ref. 29.

facility which includes a fuel reprocessing plant, 600,000 gallons of liquid high level waste stored in a tank, and a waste disposal area, is being addressed at this time. Fairly recently, DOE published a report which addresses alternatives for eventual disposition of the site, including full or partial decommissioning or continued use as some manner of nuclear production or research facility (Ref. 32). After completion of this study of alternatives, which was mandated by Congress, legislation was passed in 1980 (The West Valley Demonstration Project Act) that charges DOE with the responsibility to develop, construct, and operate a high-level liquid waste solidification project at the West Valley plant. This project will solidify the 600,000 gallons of liquid high-level waste presently stored in underground tanks into a final form acceptable for disposal into a federal repository. Decontamination of existing facilities to prepare for the project, as well as activities during the waste solidification project and final decontamination of facilities at the end of the project, will generate substantial volumes of low-level waste. Much if not most of this waste is expected to be contaminated with transuranic radionuclides. DOE has not yet determined where these wastes will be disposed, but it appears that most will be consigned to federal (DOE) disposal areas.

3.6 Generation of Transuranic Waste

Compared to operations conducted by the Department of Energy (DOE), there has been only relatively small quantities of transuranic (TRU) waste generated by the commercial sector. Major sources of transuranic wastes which have been delivered in the past to commercial disposal sites have included wastes from:

- DOE and its successors, the Energy Research and Development Administration (ERDA) and the Atomic Energy Commission (AEC);
- DOE, ERDA, and AEC contractors;
- The West Valley, New York commercial fuel reprocessing plant;
- Research and development on plutonium fuels, including fabrication of small quantities of mixed-oxide (MOX) fuels for test purposes in light water reactors; and
- Research studies of irradiated reactor fuel.

Within the last few years, the amount of TRU waste delivered to the commercial sites has been further reduced to even lower levels and has been finally discontinued. In 1970, the AEC initiated a policy in which AEC and AEC-contractor produced TRU waste in concentrations greater than 10 nCi/gm were consigned to retrievable storage at AEC facilities pending the availability of a repository for ultimate disposition of the w te. This policy is being continued today by DOE. In 1972, the operator of the West Valley reprocessing plant--Nuclear Fuel Services (NFS)--shut down operations at the facility and it has never operated since. In 1976, NFS announced their intention of withdrawing from the reprocessing business. Also in 1976, President Carter announced a national policy of deferment of commercial fuel reprocessing. This policy of deferring fuel reprocessing has halted most of the mixed oxide fuel research and development work in the commercial sector.

Although at one time five of the six commercial LLW disposal facilities accepted TRU waste for disposal (the Barnwell, South Carolina facility has never accepted TRU waste for disposal), this practice has been discontinued. The last commercial facility to accept TRU waste for disposal was the site located in the center of the Hanford Reservation near Richland, Washington and operated by the Nuclear Engineering Company (NECO). From 1976 to 1979, the NECO-Richland facility was the only commercial disposal facility accepting TRU waste for disposal. TRU waste acceptance at the NECO-Richland facility in concentrations exceeding 10 nCi/gm was prohibited by the State of Washington in November 1979.

Table 17 is a summary of the quantities of plutonium delivered to the NECO-Richland site during the years 1976 through 1978, and the year 1979 to May 24 (Ref. 33). Most of the TRU waste generated was from clean-up and decontamination of former plutonium research laboratories and small-scale MOX fuel fabrication facilities. Small quantities of waste (e.g., Battelle Columbus) were also generated from burn-up studies of LWR fuel. Not shown on this table is some very small quantities of wastes contaminated with Pu-238 (estimated at less than 200 ft^3/year) and produced from the manufacture of radioactive power sources. It is interesting to note that significant quantities of TRU waste shipped to the NECO-Richland site were owned by DOE--i.e., 75% in 1976, 31% in 1977, 25% in 1978, and 69% in 1979 up to May 24. Much of the other plutonium contaminated wastes--even if not directly owned by DOE--were generated as a direct result of DOE-contracted work.

Upon learning of Washington State's prohibition of TRU disposal at the NECO-Richland site, NRC requested DOE to "finalize and implement its plans for routine acceptance of commercial TRU waste for retrievable storage." (Ref. 34) Potential DOE acceptance of Commercial TRU waste for retrievable storage was subsequently discussed in a January 29, 1980 meeting in Denver, Colorado between NRC, DOE and industry representatives. At this meeting, information on projected future TRU waste volumes was provided by industry and is summarized as Table 18 (Ref. 35). As shown, the total volume of TRU wastes to be considered over the next 4-5 years is less than 200,000 ft^3. By far the major portion of this would be TRU wastes generated on a one-time only basis as a result of decontamination and decommissioning activities at previously used plutonium research laboratories. These facilities include those operated by Westinghouse, Kerr-McGee, NFS-Erwin, GE-Vallecitos, Exxon Nuclear, possibly Rockwell, and B&W in Leechburg, PA and Lynchburg, VA. These wastes would have low radiation levels permitting contract handling of waste packages. Following these decontamination and decommissioning activities, potential TRU waste volumes are projected to drop to low levels and would result from destructive examination of reactor fuels at the following facilities: B&W Lynchburg, GE-Vallecitos, Battelle Memorial Institute, GA, and Rockwell Santa Susana. These wastes are expected to have high surface radiation levels and would require remote handling. The annual volume of these wastes is projected to be about 2,700 ft^3/yr. Plutonium-238 contaminated waste from manufacture of heat sources should also continue at a few hundred cubic feet per year.

At the January 1980 meeting, DOE took the position that DOE did not have the authority to accept commercial TRU waste. DOE has continued to hold to this position to this day. This obviously affects the potential generation of waste generated from decontamination and decommissioning operations at plutonium reseach laboratories. No solution has been reached to this date.

Table 17 Grams of Plutonium Delivered to NECO-Richland Disposal Facility Between 1/1/76 and 5/24/79

	1979	1978	1977	1976
Babcock and Wilcox Lynchburg, VA	52 (J)	270 (J)	35 (J)	--
Babcock and Wilcox Leechburg, PA	--	27 (G)	414 (J)	7074 (B) 630 (G) 945 (J)
Westinghouse Cheswick, PA	--	152 (G)	222 (G)	273 (G)
General Electric Vallecitos, CA	350 (G)	1006 (G) 2268 (J)	120 (J) 810 (J)	117 (J)
Battelle Columbus, OH	29 (G) 98 (H)	22 (G) 18 (H) 268 (J)	--	--
Battelle (PNL) Richland, WA	--	--	10 (G) 113 (J)	21 (J)
Kerr-McGee Cimmaron, OK	--	77 (J)	49 (J)	1798 (B) 474 (J)
Nuclear Fuel Services Erwin, TN	--	594 (J)	--	76 (J)
Allied General Nuclear Services Barnwell, SC	--	20 (J)	--	--
US Army Material Command	--	--	--	1 (B)
Lovelace Foundation, Albq. NM	--	--	*	*
LFE Environmental, Rich., CA	--	*	*	--
General Atomic Company San Diego, CA	--	--	--	*
Total	529	4870	2242	12330
(B)	-	--	--	--
(G)	379	1207	701	8873
(H)	98	18	--	988
(J)	52	3645	1541	2489
% of Total: (B) + (G)	90%	25%	31%	75%
% of Total: (H) + (J)	10%	75%	69%	75%

(B) DOE-Owned, Lease Agreement - Nonwaiver of Use Charge.
(G) DOE-Owned Production and Research Programs.
(H) Owned by Other U.S. Government Agencies.
(J) Privately Owned (Domestic).
*Less than 1 gram.
**To 5/24/79.

Table 18 Summary of Current and Projected TRU Waste Volumes*

Facility	On Hand**	1980	1981	1982
Babcock and Wilcox Leechburg, PA	--	7560	15,500†	15,500†
Babcock and Wilcox†† Lynchburg, VA	7600	260	155	155
Battelle Memorial Institute†† Columbus, OH	--	1050	220	220
Exxon Nuclear Richland, WA	--	1900	--	--
General Electric Vallecitos, CA	--	8225	8,630	130
Kerr-McGee Cimarrou, OK	--	8000	12,000	10,000
Monsanto Research Corporation# Dayton, OH	375	200	1,400	200
Nuclear Fuel Services Erwin, TN	--	8000	21,000	25,000
Rockwell International Canoga Park	300	--	--	--
Westinghouse†† Cheswick, PA	1000	8000	10,000	2,000
Total:	9,275	43,195	68,905	53,205
Combined total:	174,580			

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*These volumes do not include TRU wastes which are generated by DOE and DOE-contractor operations and are currently consigned to retrievable storage by DOE.

**Boxed and ready for shipment.

†These volumes are quite uncertain.

††Waste volumes include high activity (require shielding) process residues generated as a result of irradiated fuel studies.

#Waste is produced from manufacture of radiation sources, using the isotopes Am-241, Pu-238, and Cf-252.

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APPENDIX B

RADIONUCLIDE CONCENTRATION PROJECTIONS

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APPENDIX B : RADIONUCLIDE CONCENTRATION PROJECTIONS

This appendix contains a compilation of available information on radioisotopic concentrations in low-level radioactive waste (LLW), and presents methodologies utilized in estimating the radionuclide concentrations of the untreated LLW projected to be generated between the years 1980 and 2000. A listing of the waste streams considered in this report is presented in Table B-1.

B.1 INTRODUCTION

This section briefly discusses the background for estimating radioisotopic concentrations in LLW, presents the radionuclides considered in this appendix, and discusses averaging techniques utilized in processing the available information.

B.1.1 Background

Most previous generic attempts to quantify the activities of individual radionuclides in LLW have been based on indirect methods. These methods include use of sophisticated computer codes and use of radiation exposure rates. The ORIGEN code(1) gives good results for neutron activated wastes such as spent fuel, fuel cladding, and reactor components but is not designed to calculate activities in wastes such as spent resins, filter sludges, and liquid wastes. The GALE codes(2,3) are better suited to calculating activities in the latter types of wastes.

A more empirical approach is to calculate total activites by applying the "6CEn formula" to measured exposure rates of the waste and then distributing the activity among isotopes expected to be present according to predetermined factors. This is the most widely used method. The 6CEn formula is applicable to gamma emitting point sources and is expressed as:

TABLE B-1 : Waste Groups and Streams

Waste Stream	Symbol
Group I : LWR Process Wastes	
PWR Ion Exchange Resins	P-IXRESIN
PWR Concentrated Liquids	P-CONCLIQ
PWR Filter Sludges	P-FSLUDGE
PWR Filter Cartridges	P-FCARTRG
BWR Ion Exchange Resins	B-IXRESIN
BWR Concentrated Liquids	B-CONCLIQ
BWR Filter Sludges	B-FSLUDGE
Group II : Trash	
PWR Compactible Trash	P-COTRASH
PWR Noncompactible Trash	P-NCTRASH
BWR Compactible Trash	B-COTRASH
BWR Noncompactible Trash	B-NCTRASH
Fuel Fabrication Compactible Trash	F-COTRASH
Fuel Fabrication Noncompactible Trash	F-NCTRASH
Institutional Trash	I-COTRASH
Industrial Source and Special Nuclear Material Trash	N-SSTRASH
Industrial Low Trash	N-LOTRASH
Group III : Low Specific Activity Wastes	
Fuel Fabrication Process Wastes	F-PROCESS
UF_6 Process Wastes	U-PROCESS
Institutional Liquid Scintillation Vial Waste	I-LIQSCVL
Institutional Liquid Waste	I-ABSLIQD
Institutional Biowaste	I-BIOWAST
Industrial Source and Special Nuclear Material Waste	N-SSWASTE
Industrial Low Activity Waste	N-LOWASTE
Group IV : Special Wastes	
LWR Nonfuel Reactor Components	L-NFRCOMP
LWR Decontamination Resins	L-DECONRS
Waste from Isotope Production Facilities	N-ISOPROD
Tritium Production and Manufacturing Waste	N-TRITIUM
Accelerator Targets	N-TARGETS
Sealed Sources	N-SOURCES
High Activity Waste	N-HIGHACT

$$I = 6CEn \qquad (B\text{-}1)$$

where

I = Radiation Intensity (R/hr) at 1 foot from the Source

C = number of curies

E = average gamma enery in MeV

n = Total number of gammas per disintegration

Exposure rates are usually measured on packaged wastes rather than point sources and the factors used to distribute the activity are frequently based on limited information.

Only recently have LLW waste activites been determined by direct surveys of LLW generators and radiochemical measurements become available.(4-13) These data plus additional information from disposal site radioactive waste shipment records (RSR's) (14-16) form the basis of the estimated activites of the majority of nuclear fuel cycle LWR wastes. For institutional and industrial wastes, representative specific radioisotope concentration data obtained by direct measurements are not yet available. However, a number of surveys of these waste generators have been performed and provide guidance in estimating radioisotope concentrations of such waste.(5-8)

B.1.2 Radionuclides in Low-Level Waste

Low-level radioactive wastes contain a large number of naturally occurring and man-made radionuclides at the time they are produced. Many of these radionuclides are very short-lived and are not of long-term radiological concern. Other isotopes with half lives up to a few years may reach the disposal site but decay to insignificant levels shortly thereafter.

Two criteria were used in selecting the radionuclides considered: (1) its half life must be more than a few years (five years was used as a general guide); and (2) it must be present in comparatively significant quantities in LLW. In this selection process, the biological toxicities of radionuclides were also considered. The radionuclides that will be considered in this appendix are presented in Table B-2.

Average energy per gamma-ray and total gamma energies emitted per disintegration for the isotopes given in Table B-2 are presented in Table B-3.(17) The total energies presented in the first column of this table correspond to the term (En) utilized in Equation B-1 -- i.e., total gamma energy emitted per disintegration.

In addition to these radionuclides, several other isotopes have been observed in LLW received at the commercial disposal sites.(15) These radionuclides, which are listed in Table B-4 together with the average energy per gamma-ray and total gamma energies emitted per disintegration, have not been considered in this appendix in developing the specific isotopic concentration data.

The radionuclides considered in this appendix that are presented in Table B-2 are briefly discussed below.

Tritium (H-3) is one of the most commonly found isotopes in LLW. In the nuclear fuel cycle, H-3 is produced by fission and by neutron capture from lithium-6 -- i.e., the reaction [Li-6 (n,alpha) H-3]. Naturally occurring H-3 is produced by cosmic radiation in the upper atmosphere. Tritium decays by beta emission to He-3 with a half-life of 12.4 years. The maximum energy of the emitted beta particle is 0.0186 MeV. No other radiations are produced.

Carbon-14 is produced by neutron capture from nitrogen-14 -- i.e., the reaction [N-14 (n, p) C-14]. Naturally occurring C-14 is produced by cosmic radiation in the upper atmosphere. Carbon-14 decays by beta emission to N-14 with a half-life of 5730 years. The maximum energy

TABLE B-2 . Radionuclides Considered in Analyses

Isotope	Half Life (years)	Radiation Emitted	Principal Means of Production
H-3	12.3	β	Fission; Li-6 (n, α)
C-14	5730	β	N-14 (n, p)
Fe-55	2.60	X-rays	Fe-54 (n, γ)
Co-60	5.26	β, γ	Co-59 (n, γ)
Ni-59	80,000	X-rays	Ni-58 (n, γ)
Ni-63	92	β	Ni-62 (n, γ)
Sr-90	28.1	β	Fission
Nb-94	20,000	β, γ	Nb-93 (n, γ)
Tc-99	$2.12\text{x}10^5$	β	Fission; Mo-98 (n, γ) Mo-99 (β^-)
I-129	$1.17\text{x}10^7$	β, γ	Fission
Cs-135	$3.0\text{x}10^6$	β	Fission; daughter Xe-135
Cs-137	30.0	β, γ	Fission
U-235	$7.1\text{x}10^8$	α, γ	Natural
U-238	$4.51\text{x}10^9$	α, γ	Natural
Np-237	$2.14\text{x}10^6$	α, γ	U-238 $(n, 2n)$ U-237 (β^-)
Pu-238	86.4	α, γ	Np-237 (n, γ) Np-238 (β^-); daughter Cm-242
Pu-239	24,400	α, γ	U-238 (n, γ) U-239 (β^-) Np-239 (β^-)
Pu-240(a)	6,580	α, γ	Multiple n-capture
Pu-241	13.2	α, β, γ	Multiple n-capture
Pu-242	$2.79\text{x}10^5$	α	Multiple n-capture; daughter Am-242
Am-241	458	α, γ	Daughter Pu-241
Am-243	7950	α, γ	Multiple n-capture
Cm-243	32	α, γ	Multiple n-capture
Cm-244	17.6	α, γ	Multiple n-capture

(a) Pu-239 and Pu-240 are considered as a single radionuclide in the impact analyses since they generally cannot be radiochemically distinguished. The activity of Pu-240 is added to that of Pu-239.

TABLE B-3

Gamma Radiation Characteristics of Selected Radionuclides

Nuclide	Total[a] Energy (MeV/Dis.)	Average[b] Energy (MeV/gamma)
Co-60	2.50E+00	1.25E+00
Nb-94	1.57E+00	7.87E-01
Cs-137	5.63E-01	6.62E-01
Np-237	1.54E-01	2.11E-01
U-235	1.26E-01	1.80E-01
Cm-243	7.46E-02	2.49E-01
Am-243	3.63E-02	7.30E-02
Am-241	2.17E-02	6.00E-02
Sr-90	5.02E-03	7.60E-01
I-129	3.60E-03	4.00E-02
Ni-59	1.09E-03	3.50E-01
Pu-239	1.66E-05	2.21E-01
Cm-244	1.45E-05	6.20E-02
Pu-238	9.81E-06	1.08E-01
Pu-241	2.32E-07	1.45E-01
H-3, C-14	0	0
Fe-55, Tc-99	0	0
Cs-135, U-238	0	0
Pu-242	0	0

(a) Total energy per disintegration is the sum over all gammas emitted of the individual gamma energy times its emission frequency.

(b) Average energy per gamma is the total energy per disintegration divided by the sum of the emission frequencies (which may be less than unity).

TABLE B-4

Gamma Radiation Characteristics of Several Other Radionuclides Occasionally Observed in LLW

Nuclide	Total[a] Energy (MeV/Dis.)	Average[b] Energy (MeV/gamma)
Th-232+D	4.03E+00	1.15E+00
Bi-207	1.54E+00	8.37E-01
Eu-154	1.33E+00	8.50E-01
Ra-226+D	1.18E+00	9.58E-01
Eu-152	1.01E+00	6.76E-01
Kr-85	2.11E-01	5.14E-01
K-40	1.60E-01	1.46E+00
Ti-44	1.38E-01	7.34E-02
Pb-210+D	1.89E-02	4.70E-02
Th-230	5.76E-04	8.22E-02
U-234	1.06E-04	5.32E-02
Cl-36	1.53E-05	5.10E-01
Be-10, Ar-39	0	0
Ca-41, Nb-93m	0	0
Mo-93, Re-187	0	0
U-233, U-236	0	0

(a) Total energy per disintegration is the sum over all gammas emitted of the individual gamma energy times its emission frequency.

(b) Average energy per gamma is the total energy per disintegration divided by the sum of the emission frequencies (which may be less than unity).

of the emitted beta particles is 0.156 MeV. No other radiations are produced.

Iron-55 is produced by neutron capture from iron-54 -- i.e., the reaction [Fe-54 (n,gamma) Fe-55]. It decays by electron capture to magnesium-55 with a half-life of 2.60 years. The decay is accompanied by the emission of manganese x-rays, and of continuous internal bremsstrahlung with energies up to 0.23 MeV.

Cobalt-60 is produced by neutron capture from cobalt-59 -- i.e., the reaction [Co-59 (n,gamma) Co-60]. It decays by beta emission to nickel-60. More than 99 percent of the emitted beta particles have a maximum energy of 0.314 MeV. Approximately 0.12 percent are emitted with a maximum energy of 1.48 MeV. Each Co-60 decay is accompanied by the emission of two gamma rays, one with an energy of 1.173 MeV and the other with 1.332 MeV.

Nickel-59 is produced by neutron capture from nickel-58 -- i.e., the reaction [Ni-58 (n,gamma) Ni-59]. It decays by electron capture to cobalt-59 with a half-life of approximately 80,000 years. The decay is accompanied by the emission of cobalt x-rays and continuous internal bremsstrahlung with energies up to 1.06 MeV.

Nickel-63 is produced by neutron capture from nickel-62 -- i.e., the reaction [Ni-62 (n,gamma) Ni-63]. It decays by beta emission to copper-63 with a half-life of 92 years. The emitted betas have a maximum energy of 0.067 MeV. No other radiations are emitted.

Stronium-90 is produced by fission. It decays by beta emission to yttrium-90 with a half-life of 28.1 years. The emitted beta particles have a maximum energy of 0.546 MeV. Yttrium-90 is also radioactive and decays to zirconium-90 with a half-life of 64 hours. The beta particles emitted by Y-90 have a maximum energy of 2.27 MeV. No other radiations are associated with either beta decays.

Niobium-94 is produced by neutron capture from niobium-93 -- i.e., the reaction [Nb-93 (n,gamma) Nb-94]. It decays by beta emision to molybdenum-94. The emitted beta particles have a maximum energy of 0.49 MeV. Each beta decay is accompanied by the emission of two gamma photons, one with an energy of 0.702 MeV and the other with an energy of 0.871 MeV.

Technicium-99 is produced by fission, and by beta decay of molybdenum-99 produced by neutron capture from molybdenum-98 -- i.e., the reaction [Mo-98 (n,gamma) Mo-99 ($beta^-$) Tc-99]. The half-life of molybdenum-99 is 66 hours. Technicium-99 decays by beta emission to ruthenium-99 with a half-life of 2.12×10^5 years. The emitted beta particles have a maximum energy of 0.292 MeV. No other radiations are associated with the decay.

Iodine-129 is produced by fission. It decays by beta emission to xenon-129 with a half-life of 1.7×10^7 years. The emitted beta particles have a maximum energy of 0.150 MeV. The beta decay is accompanied by the emission of xenon x-rays and conversion electrons with energies of 0.005 MeV and 0.034 MeV, respectively.

Cesium-135 is produced by fission and by decay of xenon-135. It decays by beta emission to barium-135 with a half-life of 3.0×10^6 years. The emitted beta particles have a maximum energy of 0.21 MeV. No other radiations are associated with the decay.

Cesium-137 is produced by fission. It decays with a half life of about 30 years by beta emission to barium-137. Approximately 7 percent of the emitted beta particles have a maximum energy of 1.176 MeV. The remaining beta particles are emitted with a maximum energy of 0.514 MeV. Approximately 85 percent of the beta decays are accompanied by the emission of a 0.662 MeV gamma photon. Barium x-rays and conversion electrons with energies of 0.624 MeV and 0.656 MeV are also emitted.

Uranium-235 is a naturally occurring primordial isotope. It is also produced by the decay of plutonium-239. Uranium-235 decays by alpha emission with a half-life of 7.1×10^{8} years to thorium-231. Approximately 57 percent of the emitted alpha particles have an energy of 4.40 MeV, 18 percent have an energy of 4.37 MeV, and 8 percent have an energy of 4.58 MeV. The alpha decays are accompanied by the emission of thorium x-rays.

Uranium-235 is the first naturally occurring member of the actinium (or 4n+3) decay series. The principal members of this series are listed in Table B-5. The presence and amounts of the other members of this series depend on the history of the sample. All members are expected to be present in old undisturbed ore bodies, but only U-235 and Th-231 are present in appreciable quantities in freshly fabricated fuel.

Uranium-238 is another naturally occuring primordial isotope. It decays by alpha emission to thorium-234 with a half-life of 4.51×10^{9} years. Approximately 75 percent of the emitted alpha particles have an energy of 4.20 MeV and 25 percent have an energy of 4.15 MeV. These decays are accompanied by the emission of thorium x-rays and conversion electrons with energies of 0.030 MeV and 0.043 MeV.

Uranium-238 is the first naturally occurring member of the uranium (or 4n+2) decay series. The principal members of this series are listed in Table B-6. A man-made branch of this series consisting of curium-242 and plutonium-238 joins the main series at U-234. As with the actinium series, the presence and amounts of the members of the uranium series depend on the history of the sample. The only U-238 daughters expected in appreciable quantities in freshly fabricated fuel are Th-234 and Pa-234m.

Neptunium-237 is produced by beta decay of uranium-237 which is produced by neutron capture from uranium-238 -- i.e., the reaction

TABLE B-5

Uranium-235 (4n+3) Decay Series Radionuclides*

Symbol/Element	Isotope	Half-Life	Type of Disintegration[a] and Particle Energy	Daughter
92 (U) Uranium	235	7.1E8 years[b]	a : 4.559 MeV	Th-231
90 (Th) Thorium	231	25.6 hours	b : 0.30 MeV	Pa-231
91 (Pa) Protoactinium	231	3.43E4 years	a : 5.02 MeV	Ac-227
89 (Ac) Actinium	227	21.6 years	a : 4.94 MeV	Fr-223
			b : 46 KeV	Th-227
90 (Th) Thorium	227	18.2 minutes	a : 6.03 MeV	Ra-223
87 (Fr) Francium	223	22 minutes	a : 5.34 MeV	At-219
			b : 1.2 MeV	Ra-223
88 (Ra) Radium	223	11.7 days	a : 5.864 MeV	Rn-219
85 (At) Astatine	219	0.9 minutes	a : 6.27 MeV	Bi-215
			b : ?	Rn-219
86 (Rn) Radon	219	3.92 seconds	a : 6.81 MeV	Po-215
83 (Bi) Bismuth	215	8 minutes	b : ?	Po-215
84 (Po) Polonium	215	1.83E-3 seconds	a : 7.37 MeV	Pb-211
			b : 0.74 MeV	At-215
82 (Pb) Lead	211	36.1 minutes	b : 1.39 MeV	Bi-211
85 (At) Astatine	215	1.0E-4 seconds	a : 8.0 MeV	Bi-211
83 (Bi) Bismuth	211	2.11 minutes	a : 6.617 MeV	Tl-207
			b : ?	Po-211
84 (Po) Polonium	211	0.52 seconds	a : 7.442 MeV	Pb-207
81 (Tl) Thallium	207	4.79 minutes	b : 1.44 MeV	Pb-207
82 (Pb) Lead	207	Stable		

(a) (a) denotes alpha-particle, (b) denotes beta-particle, gamma radiation not included. Particle energy is the maximum for the type of disintegration.

(b) Exponential notation, i.e., 7.1E8 = 7.1×10^8.

Source : Reference 17, 18.

TABLE B-6

Uranium-238 (4n+2) Decay Series Radionuclides

Symbol/Element	Isotope	Half-Life	Type of Disintegration[a] and Particle Energy	Daughter
92 (U) Uranium	238	4.5E9 years[b]	a : 4.20 MeV	Th-234
90 (Th) Thorium	234	24.1 days	b : 0.19 MeV	Pa-234
91 (Pa) Protoactinium	234	1.18 minutes	b : 2.32 MeV	U -234
		6.7 hours	b : 1.13 MeV	U -234
92 (U) Uranium	234	2.5E5 years	a : 4.768 MeV	Th-230
90 (Th) Thorium	230	8.0E4 years	a : 4.68 MeV	Ra-226
88 (Ra) Radium	226	1620 years	a : 4.777 MeV	Rn-222
86 (Rn) Radon	222	3.82 days	a : 5.486 MeV	Po-218
84 (Po) Polonium	218	3.05 minutes	a : 5.998 MeV	Pb-214
			b : 0.33 MeV	At-218
82 (Pb) Lead	214	26.8 minutes	b : 6.70 MeV	Bi-214
85 (At) Astatine	218	1.5-2 seconds	a : 6.63 MeV	Bi-214
83 (Bi) Bismuth	214	19.7 minutes	a : 5.51 MeV	Tl-210
			b : 3.17 MeV	Po-214
84 (Po) Polonium	214	1.64E-4 seconds	a : 7.683 MeV	Pb-210
81 (Tl) Thallium	210	1.32 minutes	b : 2.3 MeV	Pb-210
82 (Pb) Lead	210	19.4 years	b : 61 KeV	Bi-210
			a : 3.72 MeV	Hg-206
83 (Bi) Bismuth	210	5.0 days	a : 4.69 MeV	Tl-206
			b : 1.155 MeV	Po-210
84 (Po) Polonium	210	138.3 days	a : 5.3 MeV	Pb-206
80 (Hg) Mercury	206	8.1 minutes	b : 1.3 MeV	Tl-206
81 (Tl) Thallium	206	4.2 minutes	b : 1.51 MeV	Pb-206
82 (Pb) Lead	206	Stable		

(a) (a) denotes alpha-particle, (b) denotes beta-particle, gamma radiation not included. Particle energy is the maximum for the type of disintegration.

(b) Exponential notation, i.e., 4.5E9 = 4.5×10^9.

Source : Reference 17, 18.

[U-238 (n,gamma) U-237 (beta$^-$) Np-237]. The half-life of U-237 is 6.75 days. Neptunium-237 decays by alpha emission to Pa-233 with a half-life of 2.14x10^6 years. Approximately 75 percent of the alpha particles are emitted with an energy of 4.78 MeV and 12 percent with an energy of 4.65 MeV. A fraction of these decays are accompanied by emission of protactinium x-rays, and by conversion electrons with energies of up to 0.082 MeV. Neptunium-237 is a member of the man-made neptunium (or 4n+1) decay series. The principal members of this series are listed in Table B-7.

Plutonium-238 is produced by beta decay of neptunium-238 and by decay of curium-242 whose respective half-lives are 2.1 days and 163 days. These parent isotopes are produced by decay of isotopes produced by neutron capture of uranium-238. Plutonium-238 decays by alpha emission to U-234 with a half-life of 86.4 years, and is considered a member of the uranium series. Approximately 72 percent of the alpha particles are emitted with an energy of 5.50 MeV, and 28 percent with an energy of 5.46 MeV. Uranium x-rays and conversion electrons with energies of 0.02 MeV and 0.039 MeV are also emitted.

Plutonium-239 is produced by beta decay of neptunium-239 which is the daughter of another beta emitter, U-239, which itself is produced by neutron capture from U-238 -- i.e., the reaction [U-238 (n,gamma) U-239 (beta$^-$) Np-239 (beta$^-$) Pu-239]. The half-lives of U-239 and Np-239 are 23.5 minutes and 2.35 days, respectively. Plutonium-239 decays by alpha emission to U-235 with a half-life of 24,400 years and is considered a member of the actinium series. Approximately 88 percent of the alpha particles are emitted with an energy of 5.16 MeV, and 11 percent with an energy of 5.11 MeV. Uranium x-rays and conversion elections with energies up to 0.047 MeV are also emitted.

Plutonium-240 is produced by multiple neutron capture from U-238 and Pu-239. It decays by alpha particle emission to U-236 with a half-life of 6580 years. Approximately 76 percent of the emitted alpha

TABLE B-7

Neptunium-237 (4n+1) Decay Series Radionuclides

Symbol/Element	Isotope	Half-Life	Type of Disintegration[a] and Particle Energy	Daughter
94 (Pu) Plutonium	241	13.2 years	a : 4.9 MeV	U-237
			b : 21 KeV	Am-241
95 (Am) Americium	241	458 years	a : 5.49 MeV	Np-237
92 (U) Uranium	237	6.75 days	b : 0.248 MeV	Np-237
93 (Np) Neptunium	237	2.14E6 years[b]	a : 4.78 MeV	Pa-233
91 (Pa) Protoactinium	233	27.0 days	b : 0.568 MeV	U-233
92 (U) Uranium	233	1.62E5 years	a : 4.82 MeV	Th-229
90 (Th) Thorium	229	7.34E3 years	a : 5.05 MeV	Ra-225
88 (Ra) Radium	225	14.8 days	b : 0.32 MeV	Ac-225
89 (Ac) Actinium	225	10.0 days	a : 5.83 MeV	Fr-221
87 (Fr) Francium	221	4.8 minutes	a : 6.34 MeV	At-217
85 (At) Astatine	217	3.2E-2 seconds	a : 7.07 MeV	Bi-213
83 (Bi) Bismuth	213	47 minutes	a : 5.87 MeV	Tl-209
			b : 1.39 MeV	Po-213
84 (Po) Polonium	213	4.2E-6 seconds	a : 8.38 MeV	Pb-209
81 (Tl) Thallium	209	2.2 minutes	b : 1.99 MeV	Pb-209
82 (Pb) Lead	209	3.30 hours	b : 0.637 MeV	Bi-209
83 (Bi) Bismuth	209	Stable		

(a) (a) denotes alpha-particle, (b) denotes beta-particle, gamma radiation not included. Particle energy is the maximum for the type of disintegration.

(b) Exponential notation, i.e., 2.14E6 = 2.14×10^{6}.

Source : Reference 17.

particles have an energy of 5.17 MeV, and 24 percent have an energy of 5.12 MeV. Uranium x-rays and conversion electrons with energies of 0.026 MeV and 0.040 MeV are also emitted. Since U-236 decays by alpha emission to Th-232 with a half-life of 2.4×10^7 years, Pu-240 is considered to be a member of the thorium (or 4n) decay series. The principal members of this series are listed in Table B-8.

Plutonium-241 is produced by multiple neutron capture from U-238, Pu-239, and related isotopes. It decays primarily by beta emission to americium-241 with a half-life of 13.2 years. About 0.0023 percent of the decays are by alpha particle emission to U-237. The emitted beta particles have a maximum energy of 0.021 MeV. The low intensity alpha particles have energies of 4.90 MeV and 4.85 MeV. X-rays are emitted in both decay modes. Plutonium-241 is a member of the neptunium decay series.

Plutonium-242 is produced by multiple neutron capture from U-238, Pu-239, and related isotopes. It decays by alpha emission to U-238 with a half-life of 2.79×10^5 years. Approximately 76 percent of the emitted alpha particles have an energy of 4.90 MeV and 24 percent have an energy of 4.86 MeV. During the decay, uranium x-rays are also emitted. Plutonium-242 is a member of the uranium decay series.

Americium-241 is produced by the beta decay of Pu-241. It decays by alpha emission to Np-237 with a half-life of 458 years. Approximately 85 percent of the emitted alpha particles have an energy of 5.49 MeV, and 13 percent have an energy of 5.44 MeV. Neptunium x-rays and conversion electrons with energies of 0.022, 0.038 and 0.054 MeV are also emitted. Americium-241 is a member of the neptunium decay series.

Americium-243 is produced by multiple neutron capture from U-238, Pu-239, and related isotopes, and from the beta decay of Pu-243. It decays by alpha emission to Np-239 with a half-life of 7950 years.

TABLE B-8

Thorium-232 (4n) Decay Series Radionuclides

Symbol/Element	Isotope	Half-Life	Type of Disintegration[a] and Particle Energy	Daughter
90 (Th) Thorium	232	1.39E10 years[b]	a : 4.007 MeV	Ra-228
88 (Ra) Radium	228	6.7 years	b : 0.06 MeV	Ac-228
89 (Ac) Actinium	228	6.13 hours	b : 2.09 MeV	Th-228
90 (Th) Thorium	228	1.91 years	a : 5.423 MeV	Ra-224
88 (Ra) Radium	224	3.64 days	a : 5.681 MeV	Rn-220
86 (Rn) Radon	220	54.0 seconds	a : 6.280 MeV	Po-216
84 (Po) Polonium	216	0.16 seconds	a : 6.774 MeV	Pb-212
			b : ?	At-216
82 (Pb) Lead	212	10.6 hours	b : 0.58 MeV	Bi-212
85 (At) Astatine	216	3.0E-4 seconds	a : 7.79 MeV	Bi-212
83 (Bi) Bismuth	212	60.5 minutes	a : 6.086 MeV	Tl-208
			b : 2.25 MeV	Po-212
84 (Po) Polonium	212	3.0E-9 seconds	a : 8.78 MeV	Pb-208
81 (Tl) Thallium	208	3.10 minutes	b : 1.79 MeV	Pb-208
82 (Pb) Lead	208	Stable		

(a) (a) denotes alpha-particle, (b) denotes beta-particle, gamma radiation not included. Particle energy is the maximum for the type of disintegration.

(b) Exponential notation, i.e., 1.39E10 = $1.39x10^{10}$.

Source : Reference 17, 18.

Approximately 87 percent of the emitted alpha particles have an energy of 5.23 MeV. Neptunium x-rays are also emitted. Americium-243 is a member of the actinium decay series.

Curium-242 is not listed in Table B-2 as an isotope to be considered in this appendix (primarily because of short half-life); however, it is utilized as an intermediate isotope in several concentration calculations.

It is produced by beta decay of Am-242 and by multiple neutron capture from U-238, Pu-239, and related isotopes. It decays by alpha emission to Pu-238 with a half life of 163 days. Approximately 74 percent of the alpha particles are emitted with an energy of 6.12 MeV, and 26 percent with an energy of 6.07 MeV. Plutonium x-rays and conversion electrons with energies of 0.022 and 0.039 MeV are also emitted. Curium-242 is a member of the uranium decay series.

Curium-243 is produced by multiple neutron capture from U-238, Pu-239, and related isotopes. It decays with a half-life of 32 years to Pu-239 by alpha decay (99.7%) and to Am-243 by electron capture (0.3%). Approximately 73 percent of the emitted alpha particles have an energy of 5.79 MeV, 11.5 percent have an energy of 5.74 MeV, 6 percent have an energy of 6.06 MeV, and 6 percent have an energy of 5.99 MeV. Plutonium x-rays and conversion electrons with energies of 0.04, 0.048, 0.088, 0.106, and 0.156 MeV are also emitted. Curium-243 is a member of the actinium decay series.

Curium-244 is produced by multiple neutron capture from U-238, Pu-239, Am-243 and related isotopes. It decays by emission to Pu-240 with a half-life of 17.6 years. Approximately 77 percent of the emitted alpha particles have an energy of 5.81 MeV and 23 percent have an energy of 5.77 MeV. Plutonium x-rays and conversion electrons with energies of 0.022 and 0.038 MeV are also emitted. Curium-244 is a member of the thorium decay series.

B.1.3 Discussion

Available radioisotopic concentration data suffer several limitations. For example, computer calculations, which are often employed in predicting the radioactivity of wastes generated by "burn-up" of nuclear fuels, are based on fuel compositions, consumption (burn-up) rates, and elemental compositions of neutron-irradiated materials. While such calculations can be reasonably accurate, they are not well-suited for determining the range of radioactivity concentrations produced by variations of operating conditions at a given reactor nor for representing a typical reactor.

A common limitation of concentration data of individual radionuclides obtained in surveys and from disposal site records is that they are frequently derived by application of pre-determined distributions to the total gross beta/gamma activities obtained during screening measurements made at the time the wastes are shipped for disposal. These measurements are usually made with relatively unsophisticated instruments and are extremely conservative since they include activities for several very short-lived radionuclides.

Although the concentrations of some of radionuclides listed in Table B-2 have been measured in samples of LWR process wastes,(9-12) these samples include those taken from smaller and older reactors as well as those taken from reactors with a history of fuel failure problems, and thus may not be representative of future LWR wastes. Since radioactive concentrations vary with a reactor's operational cycle (fluctuation in power level, shutdowns and refueling), a larger number of samples is needed to more accurately determine average concentrations.

Furthermore, the sensitivities (minimum detection limits) of the analytical procedures for the radionuclides of interest are not identical but vary with the type and energy of the radiation and with the presence of chemical and radiochemical interferences. Thus, while a few data points may be available for an isotope, they may not be any more accurate than those obtained from screening measurements.

An additional point to be considered in using currently available data on radionuclide concentrations in the various waste streams is that the processes generating these wastes and the controls on these processes are likely to change. For example, radionuclide concentrations in ion-exchange resins could increase significantly as newer better-shielded remotely-operated waste handling systems come on-line. Similarly, increased use of etched-disk filters in place of precoat filters could result in higher concentrations in filter sludges. On the other hand, radionuclide concentrations in some streams could decrease as the older plants with less sophisticated waste treatment systems are phased out of operation. The approach developed to estimate radionuclide concentrations in LLW to the year 2000 seeks to minimize the limitations of the available data through use of averaging procedures which reflect the quantity and quality of the available data.

Extensive use is made of geometric averaging of the data points where applicable. The geometric averaging is equivalent to the arithmetic averaging of the logarithm of the data values; it is calculated as the (n)th root of the product of the (n) data values. The geometric average corresponds to the use of a log-normal distribution rather than a standard gaussian distribution to represent the variation of the measured value due to independent uncontrollable parameters. This type of averaging has already been recognized by several investigators as being more suitable for environmental data when the applicable statistical distribution is not known.(19-21) The use of geometric means rather than arithmetic means allows representative estimates to be made from sets of data that contain a few data points which are several orders of magnitude greater than the majority in the set and which would dominate the average if arithmetic means were used.

The difference in geometric and arithmetic means is readily illustrated by considering a set of data consisting of 20 values of 1 and one value of 1000. The arithmetic average of these 21 values is 48.6 and the geometric average is 1.39. The geometric average is clearly more representative of the typical value. Variations of this magnitude have been observed in radionuclide concentrations of waste streams at several LWR's.(9-12) Geometric averaging is therefore a scientifically accepted compromise between the impracticality of investigating the conditions under which each sample was collected and the use of uncharacteristically high arithmetic means.

B.2 CENTRAL STATION NUCLEAR POWER PLANTS

The waste streams generated by central station nuclear power plants are discussed in this section in three distinct groups: (1) LWR process wastes, (2) LWR trash, and (3) other LWR streams.

B.2.1 LWR Process Wastes

The LWR process waste streams (all waste streams except trash and non-fuel core components) are the best characterized of all the LLW streams. This situation allows the 23 radionuclides (Pu-239 and Pu-240 cannot be distinguished by radiochemical methods and are considered here as a single isotope) listed in Table B-2 to be divided into three groups: (1) radionuclides for which the number of measurements is sufficient to allow averaging; (2) radionuclides for which several representative measurements exist, however, the number of measurements is insufficient to allow direct averaging; and (3) radionuclides which have not been measured or for which measured concentrations are considered unrepresentative of the waste stream.

Radionuclides in the first group include Co-60, Cs-137, U-238, Pu-238, Pu-239/240, Am-241, Cm-242 and Cm-244. These radionuclides are hereafter referred to as the "basic" isotopes. The comparatively short-lived isotope Cm-242, although not included in Table B-2 and not considered in the analysis, is included as a basic isotope and used to estimate the concentrations of other curium isotopes as described below. The estimated concentrations of these basic isotopes are calculated as the geometric means of the measured concentrations in each waste stream. (Exceptions, which are discussed below, are Cm-243 and Cm-244 in PWR filter sludge.) The second and third group of radionuclides are "scaled" to the above listed basic radionuclides, and their concentrations are calculated as a fraction of the concentration of an appropriate basic radionuclide. The scaled radionuclides and the basic radionuclides are given in Table B-9.

TABLE B-9

Basic and Scaled Radionuclides for LWR Process Waste Streams

Basic Isotope	Scaled Isotopes
Co-60	Fe-55, Ni-59, Ni-63, Nb-94
Cs-137	H-3, C-14, Sr-90 Tc-99, I-129, Cs-135
U-238	U-235, Np-237
Pu-238	--
Pu-239/240	Pu-241, Pu-242
Am-241	Am-243, Cm-242*
Cm-242	Cm-243, Cm-244*
Cm-244	--

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* Only for P-FSLUDGE waste stream.

The second group of radionuclides, those for which the number of measurements is insufficient to allow direct geometric averaging, consists of H-3, C-14, Fe-55, Ni-63, Sr-90, I-129, and Pu-241. The concentrations of these radionuclides are calculated by "scaling" to the concentration of an appropriate basic isotope. These radionuclides are paired on the basis of a common source and/or method of production. For example, activated corrosion products (Fe-55 and Ni-63) are scaled to Co-60 which is also an activated corrosion product; fission products Sr-90, I-129, and H-3 (H-3 is also produced by activation) are scaled to Cs-137 which is also a fission product; and Pu-241 is scaled to Pu-239/240, the nuclide it originates from through multiple neutron capture. Carbon-14 is rather difficult to categorize; it is scaled to Cs-137.

Scaling is accomplished using data for samples which have been analyzed for both the radionuclide to be scaled and the appropriate basic isotope. The ratio of the concentration of the radionuclide to be scaled to that of the basic isotope is calculated for each data pair. A "scaling factor" for each of the radionuclides in this second group is then calculated as the geometric average of each set of ratios, i.e., this calculational procedure is given by the equation:

$$\log(SF_X) = (1/N) \sum_{i=1}^{N} \log(X_i/B_i)$$

where

SF_X = Scaling factor for isotope X

X_i = activity of isotope X in the i^{th} sample

B_i = activity of the basic isotope B in the i^{th} sample

N = number of samples for which measurements of both X and B exist.

Due to the limited amount of data, the scaling factors are calculated by reactor type only (BWR's and PWR's) rather than by reactor type and by waste stream like the basic radionuclides. The activity of isotope

X in a given stream is then estimated by multiplying the activity of the corresponding basic isotope B in that stream by the scaling factor for X. A special scaling factor is calculated by this procedure for Cm-242 in PWR filter sludge using Cm-242/Am-241 data pairs for PWR cartridge filters.

The third group of radionuclides consists of Ni-59, Nb-94, Tc-99, Cs-135, U-235, Np-237, Pu-242, Am-243, and Cm-243. For these radionuclides scaling factors obtained from information other than direct radiochemical measurements are used.(22) Nickel-59 is scaled to Co-60 using respective activities of 0.5 and 810 Ci/reference reactor year (RRY); Niobium-94 is scaled to Co-60 by taking the geometric mean of the average of Nb-94 scaling factors for activated metals and the Nb-94/Co-60 activity ratio of 0.140/1.30; and the Cs-135 scaling factor is set equal to that for Tc-99.(22)

Although data is available for U-235, most of the activities are reported as being less than the limits of detection. Activities of U-238 are low and frequently near the lower limit of detection. Use of the U-235 data would effectively equate U-235 and U-238 activities. This is not reasonable since virtually the entire commercial fuel cycle is based on fuel enriched to about 4 weight percent U-235. Accordingly, U-235 is scaled to U-238 for LWR wastes by assuming an average 2% enrichment to account for burn-up. Neptunium-237 activity is scaled to that of its precursor, U-238, using activites of 0.011 and 450 Ci, respectively.(23) The activity of Pu-242 is scaled to the combined average basic activity of its Pu-239/240 precursors; Am-243 is scaled to the average basic Am-241 activity; and Cm-243 is scaled to the Cm-242 average basic activity by using the geometric mean of the two ratios for each pair.(22)

The available radiochemical measurements for the basic radionuclides utilized in the concentration projections have been obtained from several reactors. These reactors and their abbreviations are presented in Table B-10.

TABLE B-10 . Plant Abbreviations

	Number	Net MW(e)
BWRs with Deep Bed CPS*		
Big Rock Point	B-1	63
James A. Fitzpatrick	B-2	821
Millstone Point, Unit 1	B-3	660
Nine Mile Point, Unit 1	B-4	610
Oyster Creek	B-5	620
Pilgrim, Unit 1	B-6	670
BWRs with Pre-Coat CPS		
Browns Ferry, Units 1, 2, and 3	B-7	1067 each
Monticello	B-8	536
Quad Cities, Units 1, 2	B-9	800 each
PWRs with CPS		
Calvert Cliffs, 1, 2	P-1	850 each
Robert E. Ginna	P-2	490
Indian Point 1, 2, and 3	P-3	285, 873, 965
Oconee 1, 2, 3	P-4	860 each
Surry 1, 2	P-5	775 each
Zion 1, 2	P-6	1100 each
PWRs without CPS		
Connecticut Yankee (Haddam Neck)	P 7	575
Robert E. Ginna (prior to 11/78)	P-2	490
Point Beach	P-8	497
Yankee Rowe	P-9	175

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* CPS : condensate polishing system.

The results of the radiochemical measurements and the geometric averages calculated from these results are presented in Tables B-11 through B-17.(9-12)

Designation of BWR's as having deep bed or pre-coat CPS refers to use of deep bed demineralizers or pre-coat filters in their condensate polishing systems (CPS). PWR's are grouped according to whether or not they use a CPS. PWR's with partial flow systems are considered to have CPS. The average isotopic activities given in Tables B-11 through B-17 show no significant dependence on the type of CPS used. Differences of approximately a factor of two are insignificant in view of the small number of samples in some CPS groups and of the large variations in activity of a given isotope within each group. Wastes from BWR's and PWR's are distinguishable on the basis of their activities and their chemistries. The "grand average" isotopic activities given in Tables B-11 through B-17 are considered to be representative of the LWR process waste streams. To convert measured concentrations from a mass basis to a volume basis, these activities are used in conjunction with the estimated densities of each stream presented below:(4)

Stream	Estimated Density (g/ml)	
	BWR	PWR
Resins	0.81	0.91
Concentrated Liquids	1.20	1.00
Filter Sludges	0.86	0.86
Cartridge Filters	--	0.60

The lack of Cm-242, Cm-243, and Cm-244 data requires special scaling factors for PWR filter sludge. Curium-242 is scaled to Am-241 and Cm-244 is scaled to Cm-242 as the geometric means of the ratios calculated from the PWR filter cartridge data given in Table B-17. The Cm-242 factor is then applied to the average basic Am-241 activity in PWR filter sludge (Table B-16) to obtain the Cm-242 activity in the sludge. The normal Cm-243 scaling factor and the special Cm-244 scaling factor (see below) are then applied to the Cm-242 filter sludge activity to obtain their respective activities in filter sludge.

TABLE B-11 . Activities (μCi/g dry) Measured in BWR Spent Resins

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
B-1	6.30E+00	3.20E+00	<4.00E-07[a]	1.80E-04	1.10E-04	6.20E-05	<1.00E-05	<1.00E-05
	3.20E+01	6.30E+01	1.80E-06	1.30E-03	1.70E-03	3.70E-04	2.20E-04	3.60E-04
B-3	2.48E+00	6.17E-01	--	9.09E-03	5.32E-03	2.78E-03	1.26E-02	2.18E-02
B-4	6.24E+00	3.17E+01	<3.00E-06	4.00E-05	3.00E-05	6.00E-05	NA[b]	NA
B-5	9.00E+00	3.80E+00	<1.00E-06	4.10E-04	2.40E-04	7.10E-05	2.10E-04	2.20E-05
	9.06E+01	6.59E+01	--	3.01E-04	3.64E-04	2.61E-04	7.63E-04	1.34E-04
	7.00E-01	5.10E-02	<3.00E-07	7.10E-06	3.20E-06	2.00E-06	5.00E-06	5.00E-07
B-6	1.55E+00	5.00E+00	<2.00E-07	5.40E-05	5.30E-05	1.10E-05	8.90E-05	8.80E-06
	1.20E-01	1.89E+00	<2.00E-07	5.70E-06	5.40E-06	7.40E-06	1.80E-05	5.00E-07
	2.90E-02	5.30E-02	<2.00E-07	5.00E-06	3.10E-06	2.10E-06	1.80E-05	1.10E-06
Average of Deep Bed CPS BWR's	2.50E+00	2.76E+00	5.18E-07	1.01E-04	7.71E-05	4.35E-05	9.52E-05	2.02E-05
B-9	3.01E+00	5.19E+00	--	2.09E-04	9.18E-05	3.80E-05	2.10E-03	1.80E-04
	5.48E-01	1.20E+00	--	1.54E-04	5.14E-05	1.30E-05	1.96E-04	2.32E-05
	3.49E+00	5.14E+00	NA	5.00E-05	2.50E-05	4.40E-06	3.30E-05	5.30E-06
	9.14E-01	1.97E+00	--	8.75E-05	6.86E-05	2.83E-05	5.47E-04	4.16E-05
	3.37E+00	4.98E+00	NA	9.70E-05	3.20E-05	4.90E-06	4.00E-05	6.00E-06
	2.82E-01	3.32E-01	NA	--	--	--	5.20E-04	6.03E-05
Average of Pre-Coat CPS BWR's	1.31E+00	2.17E+00	NA	1.06E-04	4.82E-05	1.25E-05	2.32E-04	2.63E-05
Grand Average	1.96E+00	2.52E+00	5.18E-07	1.03E-04	6.59E-05	2.87E-05	1.36E-04	2.25E-05

(a) Indicates lower limit of detection; included in averages.

(b) NA = Not Analyzed.

TABLE B-12 . Activities (µCi/g) Measured in BWR Concentrated Liquids

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
B-2[a]	8.90E-03	4.00E-04	7.00E-09	1.60E-08	5.50E-08	NA[b]	NA	NA
B-3	1.85E+00	2.12E-01	--	2.06E-04	1.10E-04	5.21E-05	1.15E-03	1.02E-04
	6.20E-01	3.78E-01	--	8.61E-05	2.87E-05	1.18E-04	7.69E-04	9.33E-05
	2.27E-01	3.50E-01	--	7.26E-04	2.96E-04	4.17E-04	1.15E-03	1.26E-04
B-4	9.60E-02	2.29E-01	1.50E-06	1.30E-05	8.00E-06	5.00E-06	NA	NA
B-5	3.90E+01	2.10E+00	<3.00E-07[c]	2.80E-03	1.40E-03	1.00E-04	2.80E-03	2.00E-04
	3.76E-06	1.47E-05	--	8.21E-02	1.20E-02	1.26E-02	5.09E-02	1.04E-02
	2.60E-02	2.20E-03	<4.00E-10	2.60E-05	1.10E-05	9.00E-06	1.30E-04	2.30E-05
	1.40E+00	7.20E-01	<3.00E-07	1.70E-04	5.90E-05	1.50E-05	2.50E-04	2.60E-05
	2.50E-02	2.60E-02	<4.00E-06	1.40E-04	6.40E-05	6.20E-05	5.80E-04	2.60E-04
	1.02E+01	2.38E-01	--	3.24E-05	1.76E-05	9.59E-05	1.89E-05	1.00E-05
	1.67E+01	4.60E-01	<2.00E-05	1.14E-03	7.20E-04	4.10E-04	2.90E-03	1.40E-03
	4.51E-05	2.59E-02	NA	1.32E-04	1.02E-04	4.08E-05	1.73E-04	4.55E-05
	1.80E-01	2.79E-01	NA	1.91E-04	9.33E-04	4.77E-04	8.69E-03	2.44E-03
Average of Deep Bed CPS BWR's	1.11E-01	5.54E-02	2.26E-07	1.66E-04	7.86E-05	1.00E-04	8.83E-04	1.71E-04
No data for Pre-Coat CPS BWR's								

(a) Activities in (µCi/ml)
(b) NA = Not Analyzed.
(c) Indicates lower limit of detection; included in averages.

TABLE B-13 . Activities (μCi/g dry) Measured in BWR Filter Sludge

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
B-2	1.23E-01	8.00E-04	1.90E-06	1.20E-06	5.00E-07	2.50E-06	NA[a]	NA
B-3	2.32E+01	2.18E+00	--	2.67E-03	1.37E-03	9.55E-04	1.23E-02	2.17E-03
B-4	1.53E+00	1.31E+00	2.00E-05	2.80E-04	1.50E-04	1.80E-05	NA	NA
B-5	1.96E+01	2.90E+00	<3.00E-04[b]	4.80E-04	8.60E-04	5.30E-04	1.80E-03	9.00E-04
	3.44E+00	4.77E+00	NA	8.50E-04	2.30E-04	2.00E-04	1.30E-03	5.50E-04
	2.79E+00	4.89E+00	--	6.12E-04	1.40E-04	--	--	--
	7.58E+00	6.75E+00	NA	7.40E-04	2.70E-04	2.10E-04	7.00E-03	4.10E-04
	2.25E+00	1.34E-01	NA	1.40E-04	1.16E-04	7.40E-04	4.00E-04	2.50E-04
	3.13E+00	6.22E+00	NA	8.00E-04	1.40E-04	2.60E-04	3.20E-03	1.20E-03
	6.50E-02	2.90E-02	<2.00E-07	8.30E-05	3.50E-05	3.80E-05	2.30E-04	2.80E-05
	1.30E-01	2.00E-02	<3.00E-07	1.90E-05	1.30E-05	8.50E-06	1.80E-05	1.50E-06
	2.70E+01	1.93E+00	<4.00E-06	3.00E-03	1.40E-03	4.20E-04	1.60E-02	2.00E-03
	8.30E-01	3.10E+00	<1.00E-06	4.50E-04	7.90E-05	9.60E-05	5.80E-04	1.90E-04
	7.80E+00	4.58E-02	<1.00E-07	1.30E-06	<6.00E-07	<1.00E-07	7.80E-07	2.60E-07
B-6	4.60E-02	3.20E-02	<2.00E-06	5.30E-05	1.80E-05	1.60E-05	8.80E-05	3.50E-06
	4.40E-02	3.10E-02	<2.00E-06	3.60E-05	1.60E-05	2.30E-05	8.20E-05	3.20E-06
Average of Deep Bed CPS BWR's	1.36E-00	3.72E-01	2.01E-06	1.53E-04	6.53E-05	5.37E-05	4.71E-04	8.07E-05
B-7	1.09E+00	4.49E-01	--	7.17E-06	2.53E-06	3.02E-05	<2.42E-05	<3.57E-06
	3.08E-01	4.56E-01	NA	1.60E-06	1.30E-06	4.20E-06	1.90E-06	4.80E-07
	2.52E+01	1.63E+01	--	9.14E-04	3.18E-04	1.44E-03	3.69E-02	1.56E-03
	3.07E+01	<7.33E-02	NA	NA	NA	NA	NA	NA
	5.49E-01	1.83E-01	--	4.54E-05	2.62E-05	1.46E-05	1.65E-04	1.56E-05
	4.45E+00	1.79E-01	NA	NA	NA	NA	NA	NA

TABLE B-13 (Cont'd)

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
B-8	3.10E+00	5.60E+00	<1.00E-05	3.90E-02	2.50E-02	2.00E-03	3.70E-02	4.70E-03
	3.55E+00	8.49E+00	--	1.30E-03	6.46E-04	7.27E-04	1.21E-03	8.99E-04
	1.00E+00	6.30E+00	4.00E-07	5.70E-03	3.80E-03	1.20E-03	1.40E-02	2.00E-03
	5.75E+01	2.39E+00	--	8.25E-03	5.37E-03	2.39E-03	9.79E-03	3.23E-03
	9.50E-01	5.10E+00	<2.00E-06	6.10E-03	3.70E-03	9.70E-04	2.10E-02	2.30E-03
	7.88E+00	4.24E+00	--	1.74E-03	1.24E-03	1.59E-04	6.73E-04	2.15E-04
	1.46E-01	1.80E+00	<6.00E-06	2.70E-03	1.00E-03	4.70E-04	8.00E-03	1.03E-03
	9.97E-01	9.62E+00	--	7.79E-04	3.36E-04	3.60E-04	1.02E-03	4.51E-04
	7.90E+00	4.20E+00	<3.00E-06	2.10E-03	1.10E-03	1.90E-04	6.20E-04	5.20E-04
	1.67E+01	9.33E+00	--	1.72E-03	9.32E-04	7.31E-04	2.61E-03	9.80E-04
	3.10E+01	9.10E+00	NA	3.70E-03	2.10E-03	1.10E-03	7.20E-03	1.90E-03
	6.90E-01	4.30E+00	<2.00E-06	9.50E-04	3.30E-04	2.80E-04	1.50E-03	5.50E-04
	3.70E+00	6.80E+00	NA	1.80E-03	9.40E-04	4.20E-04	1.90E-03	7.40E-04
	3.67E+01	2.58E+00	NA	6.00E-03	3.30E-03	2.00E-03	1.48E-02	3.23E-03
	1.99E+00	3.83E+00	NA	9.60E-04	4.70E-04	4.30E-04	2.20E-03	7.30E-04
	2.12E+02	7.20E+01	<3.00E-06	9.00E-03	2.40E-02	1.56E-03	1.37E-02	3.50E-03
	5.30E+00	2.80E+01	8.00E-05	9.50E-03	7.50E-03	6.60E-04	1.00E-02	4.60E-04
	1.10E+01	3.30E+01	2.00E-05	2.80E-02	2.10E-02	1.70E-03	3.10E-02	3.60E-03
B-9	3.01E+00	5.19E+00	NA	2.10E-04	9.20E-05	3.80E-05	2.10E-03	1.80E-04
	3.80E+00	9.80E+00	NA	1.40E-04	6.40E-05	3.50E-05	2.90E-03	1.50E-04
Average of Pre-Coat CPS BWR's	4.38E+00	3.73E+00	4.81E-06	1.26E-03	7.15E-04	3.49E-04	2.55E-03	4.99E-04
Grand Average	2.80E+00	1.55E+00	3.04E-06	5.42E-04	2.75E-04	1.81E-04	1.41E-03	2.60E-04

(a) NA = Not Analyzed.

(b) Indicates lower limit of detection; included in averages.

TABLE B-14. Activities (μCi/g dry) Measured in PWR Spent Resins

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
P-1	3.34E-04	1.55E-04	<8.00E-07[a]	1.90E-05	1.90E-04	1.20E-05	4.00E-06	<1.00E-06
	1.03E-01	3.67E-01	<1.00E-07	6.20E-06	6.60E-05	1.60E-05	<3.00E-06	<3.00E-06
	1.28E-01	3.06E-01	<6.00E-07	8.30E-06	6.40E-05	8.00E-06	<2.00E-05	<2.00E-06
P-2[b]	7.20E-02	3.19E-03	NA[c]	6.60E-05	7.20E-05	1.50E-05	1.03E-04	1.50E-05
	9.26E-03	1.82E-02	NA	1.65E-05	1.10E-05	1.10E-05	5.00E-05	8.50E-06
	1.56E-02	4.34E-01	NA	7.80E-06	1.24E-05	1.34E-05	9.69E-06	4.60E-06
	1.57E-02	4.26E-01	NA	8.00E-06	9.20E-06	4.20E-06	8.60E-05	1.61E-06
P-4	3.70E+00	<7.90E+01	NA	1.30E-04	7.80E-05	3.00E-05	5.20E-05	2.30E-05
	1.25E-05	5.82E-04	--	2.70E-06	1.03E-06	1.04E-06	4.87E-06	8.78E-07
	3.20E-05	2.95E-03	--	6.83E-06	7.51E-06	2.44E-05	<3.10E-06	1.53E-04
	3.45E-04	7.73E-02	--	1.31E-05	5.28E-06	4.84E-04	3.61E-04	2.45E-04
	3.69E-05	1.54E-02	--	3.67E-05	1.09E-05	2.01E-04	1.58E-05	8.92E-06
	1.86E-05	3.39E-05	--	2.28E-06	1.14E-06	--	--	--
	5.55E-06	3.06E-03	--	5.08E-06	2.05E-06	1.61E-06	1.42E-05	1.72E-06
	3.78E-04	8.81E-04	--	6.21E-04	5.28E-04	6.95E-04	1.67E-03	5.91E-04
	1.96E-04	1.12E-02	--	1.40E-05	8.25E-06	1.17E-05	<3.69E-06	<1.20E-06
	4.81E-04	1.83E-03	--	6.62E-06	1.74E-05	9.57E-06	1.18E-06	<5.63E-07
	2.61E-05	8.30E-04	--	1.04E-05	3.61E-06	4.74E-06	2.87E-05	2.84E-06
	7.86E-05	1.05E-03	--	1.60E-06	6.53E-07	5.65E-07	4.58E-06	4.46E-07
	6.56E-05	1.37E-03	--	2.14E-06	7.87E-07	3.19E-06	1.24E-05	2.96E-06
	2.96E-04	2.56E-04	--	1.65E-06	6.31E-07	1.18E-06	5.42E-06	1.22E-06
	9.59E-06	5.40E-05	--	5.33E-06	1.90E-06	1.57E-06	6.09E-06	1.80E-06
	5.99E-06	2.32E-03	--	3.88E-06	1.19E-06	1.55E-06	7.02E-06	1.37E-06
	4.90E-01	<4.54E-04	--	2.62E-05	2.76E-05	1.42E-05	5.23E-05	4.90E-06
	3.39E-02	<3.20E-04	--	5.64E-05	5.54E-05	--	--	--
	2.69E-05	4.31E-03	--	4.38E-06	2.08E-06	1.09E-06	8.05E-06	6.57E-07
	--	3.04E-03	--	2.46E-06	6.91E-07	6.26E-06	2.79E-06	7.40E-07
	3.30E-06	2.50E-05	--	3.15E-06	9.77E-07	7.45E-07	2.26E-06	5.31E-07
	3.48E-05	1.51E-03	--	6.41E-05	3.45E-05	2.11E-04	2.88E-03	1.45E-04
	4.02E-06	3.16E-05	--	5.13E-06	9.95E-06	--	--	--
	2.41E-04	1.64E-04	--	1.63E-05	8.25E-06	6.04E-05	1.80E-04	2.17E-05

TABLE B-14 (Cont'd)

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
P-6	7.34E+01	2.85E+01	NA	6.20E-05	2.30E-05	1.80E-05	2.60E-05	2.80E-06
	1.57E-02	1.92E-02	NA	1.60E-05	1.20E-05	1.60E-06	<1.00E-06	<1.00E-06
Average of PWR's with CPS	6.88E-04	4.22E-03	3.63E-07	1.09E-05	8.26E-06	9.21E-06	1.69E-05	1.04E-05
P-2	2.06E+00	2.19E+01	4.50E-05	4.00E-04	8.00E-04	7.00E-04	NA	NA
	5.94E+00	4.60E+00	NA	3.70E-03	8.30E-03	3.00E-03	3.40E-03	8.40E-04
	7.80E-01	4.30E-01	<1.00E-07	6.80E-04	2.70E-04	2.70E-05	2.30E-04	3.20E-05
	6.70E-01	2.40E+00	<3.00E-07	1.20E-03	4.70E-04	4.00E-05	1.60E-04	3.90E-05
P-7	9.45E-05	8.80E-07	--	6.88E-08	1.76E-08	--	--	--
	6.01E-02	2.79E+00	--	1.92E-03	6.39E-04	9.21E-04	3.79E-03	8.02E-04
	1.34E+00	1.28E+00	--	2.07E-03	8.80E-04	1.74E-03	4.45E-01	1.93E-02
	<1.31E-02	1.23E+02	--	3.55E-03	1.28E-03	1.84E-03	2.27E-02	2.03E-03
	2.85E-01	1.37E-01	--	2.41E-03	7.15E-04	3.91E-04	2.88E-02	2.05E-03
P-8	5.54E-03	1.87E+00	--	2.93E-06	9.99E-07	5.06E-06	3.07E-06	3.55E-06
	9.20E-06	1.10E-04	NA	NA	NA	NA	NA	NA
	3.57E-03	4.46E-01	--	1.35E-05	6.72E-06	1.16E-04	1.50E-05	4.38E-06
	1.47E-06	2.30E-03	NA	<4.40E-10	<3.90E-10	9.60E-09	7.10E-09	2.40E-09
	1.49E-03	7.43E-01	--	6.30E-04	7.48E-04	6.47E-06	4.41E-06	8.86E-07
	2.70E+00	NA	NA	1.30E-04	1.19E-04	4.21E-05	9.18E-04	1.30E-04
	1.94E-03	8.07E-01	--	1.22E-03	1.84E-03	1.88E-03	2.05E-04	<8.00E-05
	1.90E-05	3.10E-03	NA	NA	NA	NA	NA	NA
	9.13E-04	1.13E+00	--	1.05E-05	3.31E-06	1.75E-06	7.38E-06	1.94E-06
	1.96E+01	4.25E+01	--	8.68E-05	8.12E-05	--	--	--
	2.79E+01	2.54E+00	--	3.22E-03	1.07E-03	2.91E-04	1.65E-03	1.75E-04
	1.37E+02	3.14E+01	--	1.65E-02	2.48E-02	7.55E-03	8.54E-03	3.54E-03
	3.72E+00	<6.51E-03	--	1.42E-02	2.29E-02	1.11E-02	3.75E-02	8.45E-03

TABLE B-14 (Cont'd)

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
P-9	6.58E-03	7.78E-02	--	1.01E-05	6.14E-06	2.14E-05	5.47E-06	1.67E-06
	9.10E-06	1.83E-05	<4.00E-11	5.20E-10	<2.40E-10	<4.00E-11	3.10E-10	1.00E-10
	2.51E-03	4.95E-02	--	1.25E-05	9.04E-06	2.36E-04	1.53E-03	1.08E-05
	1.11E+01	NA	<3.00E-06	5.20E-05	5.30E-05	4.80E-05	6.40E-04	2.10E-05
	3.21E-03	3.32E-03	--	2.83E-04	1.70E-04	8.89E-05	9.02E-04	9.51E-05
	5.70E-01	6.70E-01	NA	2.00E-05	1.50E-05	5.40E-06	NA	2.70E-06
	2.00E+00	1.45E+00	<4.00E-05	1.80E-05	6.60E-06	1.60E-05	1.30E-06	1.00E-06
	2.17E-01	3.96E-01	NA	3.20E-03	2.10E-03	NA	NA	NA
Average of PWR's without CPS	4.10E-02	1.88E-01	4.32E-07	8.89E-05	5.68E-05	5.43E-05	1.56E-04	2.46E-05
Grand Average	4.98E-03	2.41E-02	4.08E-07	2.85E-05	2.06E-05	2.06E-05	4.43E-05	1.52E-05

(a) Lower limit of detection; included in averages.
(b) CPS system at Ginna became operational in January 1978.
(c) NA = Not Analyzed.

TABLE B-15 . Activities (μCi/g) Measured in PWR Concentrated Liquids

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
P-1	1.00E-03	<5.00E-04[a]	<2.00E-04	<3.00E-04	<3.00E-04	<4.00E-04	<4.00E-04	<4.00E-04
P-2[b]	1.05E-01	3.96E-04	NA[c]	8.10E-06	1.33E-06	9.90E-06	6.30E-05	3.40E-06
	6.64E-04	2.87E-02	--	1.83E-04	5.87E-05	7.91E-04	1.41E-03	7.99E-05
P-3[d]	3.50E-02	3.00E-01	1.90E-07	2.00E-07	8.00E-08	3.00E-07	NA	NA
P-4	2.25E-05	1.63E-04	NA	3.90E-06	2.40E-06	NA	NA	NA
	1.38E+00	9.08E-02	--	--	--	2.79E-04	2.35E-04	<4.00E-05
	2.64E-04	4.26E-03	NA	3.13E-06	1.02E-06	1.74E-06	1.43E-06	4.03E-07
	9.71E-02	4.00E-02	--	--	--	1.05E-05	2.92E-05	9.40E-06
	7.70E-01	<2.50E-03	NA	5.70E-05	4.00E-05	6.70E-06	4.20E-05	2.70E-06
	4.37E-02	3.71E-02	--	7.16E-06	7.97E-06	4.50E-06	4.19E-05	4.69E-06
	1.00E-04	4.20E-04	NA	5.20E-06	3.80E-06	6.40E-07	7.00E-08	1.40E-07
	--	<3.29E-03	--	6.25E-04	7.21E-04	2.02E-04	2.11E-03	1.99E-04
	2.55E-04	6.17E-03	NA	1.75E-06	4.97E-07	1.26E-06	9.63E-07	4.30E-07
	1.10E+00	6.59E-01	--	9.77E-04	9.70E-04	3.80E-04	1.86E-03	2.73E-04
	1.01E-01	8.81E-02	NA	6.91E-04	1.05E-03	3.01E-05	8.12E-05	3.67E-05
	6.11E-04	1.98E-02	NA	3.50E-05	1.50E-05	1.10E-05	1.10E-05	1.30E-05
	1.10E-03	3.40E-02	NA	2.10E-06	1.50E-06	1.80E-06	6.40E-07	2.40E-07
	1.26E-05	3.97E-05	NA	2.13E-05	1.23E-05	1.18E-06	6.69E-07	NA
P-6	3.87E+00	2.00E-01	NA	1.40E-04	2.20E-04	1.90E-03	3.10E-05	1.20E-05
	2.60E+01	3.28E+00	NA	5.70E-04	7.80E-04	9.50E-04	9.50E-04	2.10E-04
	5.15E-01	3.03E-01	NA	5.70E-04	5.20E-04	2.70E-04	5.80E-04	6.10E-05
Average of PWR's with CPS	1.40E-02	1.39E-02	6.16E-06	3.39E-05	1.80E-05	2.37E-05	3.81E-05	1.15E-05

TABLE B-15 (Cont'd)

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
P-2[b]	3.67E+00	9.71E-01	NA	2.70E-04	2.20E-04	1.10E-03	1.00E-02	4.10E-04
	2.51E+01	1.22E+00	NA	1.80E-03	3.10E-03	NA	NA	NA
	2.90E-01	3.40E-01	<5.00E-08	1.80E-04	9.20E-05	6.80E-05	2.00E-04	3.10E-05
	1.80E-01	3.56E-01	<7.00E-09	7.00E-07	1.00E-06	2.60E-07	2.00E-08	3.60E-07
	1.89E-02	1.02E-01	1.88E-06	1.00E-06	1.80E-06	NA	NA	NA
P-7	2.37E-01	4.07E+00	--	1.90E-04	7.40E-05	5.37E-05	1.24E-04	2.42E-04
P-8	2.34E+01	<1.35E-02	--	1.03E-02	1.30E-02	2.70E-03	5.38E-03	4.32E-03
	5.60E-01	1.40E-01	NA	4.91E-04	5.58E-04	7.36E-04	3.12E-03	2.68E-04
	1.72E-01	<9.00E-04	NA	1.62E-03	2.32E-03	1.22E-03	1.62E-02	1.40E-03
	1.07E+01	<3.13E-02	--	5.15E-03	5.63E-08	3.29E-03	1.81E-02	3.14E-03
P-9	1.08E+00	1.74E-01	<6.00E-07	1.40E-05	3.00E-06	1.80E-06	1.40E-05	2.60E-06
	9.68E-01	2.54E+00	--	4.18E-05	1.00E-05	1.61E-04	2.98E-05	<5.53E-06
	3.50E-03	NA	<1.00E-06	8.80E-05	2.20E-05	4.70E-05	3.70E-06	4.30E-05
	6.61E-03	4.85E-03	--	4.24E-06	2.47E-06	1.20E-06	8.00E-06	1.33E-06
	2.70E-03	5.50E-03	NA	1.80E-06	7.00E-07	2.70E-07	4.40E-07	1.30E-08
	1.18E-03	NA	<2.00E-07	1.26E-06	2.20E-07	4.30E-06	<6.00E-07	<6.00E-07
	1.98E-03	6.37E-04	NA	1.17E-03	3.60E-02	9.15E-06	1.50E-05	5.40E-06
Average of PWR's without CPS	1.70E-01	7.79E-02	2.07E-07	8.11E-05	6.53E-05	4.08E-05	6.39E-05	3.57E-05
Grand Average	4.40E-02	2.85E-02	4.84E-07	5.12E-05	3.31E-05	2.99E-05	4.78E-05	1.92E-05

(a) Lower limit of detection; included in averages.
(b) CPS at Ginna become operational on Juanuary 1978.
(c) NA = Not Analyzed.
(d) Activities in (μCi/ml).

TABLE B-16 . Activities (μCi/g) Measured in PWR Filter Sludge[a]

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
P-3[b]	1.91E+00	1.51E-01	3.00E-06	1.30E-05	5.50E-05	NA[c]	NA	NA
P-2	2.55E-01	4.10E-03	6.00E-07	2.35E-04	5.90E-04	3.07E-04	NA	NA
Average	6.98E-01	2.49E-02	1.34E-06	5.53E-05	1.80E-04	3.07E-04	NA	NA

(a) Results of analyses of filtered sludge samples reported in units of μCi/filter converted to units of μCi/g by assuming each filtered sample weighed one gram.

(b) P-3 is with CPS, and P-2 is without CPS; P-2 sample collected before installation of CPS in January, 1978.

(c) NA = Not Analyzed

TABLE B-17 . Activities (µCi/g) Measured in PWR Cartridge Filters

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
P-2[a]	5.26E-04	1.17E-03	--	1.07E-04	7.29E-05	1.89E-04	5.51E-04	9.57E-05
	1.82E-02	6.15E-05	--	2.40E-06	3.86E-06	6.18E-06	1.80E-06	2.29E-06
Average of PWR with CPS	3.09E-03	2.68E-04	--	1.60E-05	1.68E-05	3.12E-05	3.15E-05	1.48E-05
P-2[a]	1.19E+02	<5.00E-02[b]	1.00E-05	4.40E-02	6.30E-02	3.00E-02	1.80E-01	4.10E-02
	8.80E-01	2.20E-01	<5.00E-06	7.30E-04	1.40E-03	2.80E-04	4.30E-03	2.40E-04
	2.00E+01	1.23E+01	<5.00E-06	3.70E-03	4.50E-03	2.90E-03	5.50E-03	1.53E-03
	1.01E+01	<2.96E-02	--	1.58E-03	2.43E-03	6.41E-04	1.90E-03	2.73E-04
P-8	7.70E-01	3.00E-02	NA[c]	2.80E-04	4.20E-04	1.20E-04	3.60E-03	2.20E-04
	1.86E+00	<1.05E-02	NA	8.32E-04	1.32E-03	6.87E-05	9.49E-05	1.54E-05
	1.71E+00	<1.80E-03	NA	1.08E-03	1.64E-03	4.60E-04	9.70E-03	8.00E-04
	1.46E+1	<1.29E-02	--	2.80E-02	4.62E-02	1.49E-02	6.63E-02	1.24E-02
9	9.80E+00	<7.30E-03	NA	2.10E-04	4.00E-04	1.60E-04	3.00E-04	4.90E-05
	2.00E+00	<1.50E-03	<5.00E-06	8.70E-05	1.30E-04	5.60E-05	3.60E-04	2.20E-05
	3.69E-01	4.48E-01	--	8.85E-06	1.09E-05	--	--	--
	1.22E+00	NA	<2.00E-06	9.50E-05	1.00E-04	8.60E-06	3.30E-04	2.20E-05
	1.83E+01	<1.42E-02	NA	8.20E-04	2.60E-03	2.80E-04	3.00E-03	2.40E-04
	9.80E+00	<7.30E-03	NA	2.10E-04	4.00E-04	1.60E-04	3.00E-04	4.90E-05
Average of PWR's without CPS	4.45E+00	2.98E-02	4.78E-06	6.68E-04	1.86E-03	3.76E-04	7.32E-03	2.69E-04
Grand Average	1.79E+00	1.59E-02	4.78E-06	4.19E-04	6.33E-04	2.73E-04	1.40E-03	1.83E-04

(a) CPS system became operational January 1978.
(b) Indicates lower limit of detection; included in average.
(c) NA = Not Analyzed.

The radiochemical concentrations used to calculate scaling factors for the second group of isotopes as well as the basic isotope concentrations are presented in Tables B-18 through B-21.

The concentrations in activated metals utilized for calculating the scaling factor for Nb-94 are presented in Table B-22, and the scaling factors for the second and third group of isotopes are summarized in Table B-23.

B.2.2 Trash

The diversity of materials found in LWR trash make direct measurement of isotopic activities impractical. Assuming that LWR trash contains a combined/mixed spectrum of the isotopes found in LWR process wastes, isotopic activities are estimated by calculating volume-weighted average activities of the isotopes listed in Table B-2 for BWR and PWR process waste streams. The volumes utilized in this averaging are given in Table 3-7. Separate averaged activities are calculated for BWR's and PWR's. Each of these averaged activities is then converted to a fraction of the total activity by normalization to the sum of the 23 radionuclide concentrations. The trash scaling factors are listed in Table B-24.

To calculate specific radionuclide concentrations, these scaling factors are used in conjunction with the assumed total activities of 0.0235 Ci/m^3 for BWR compactible trash, 3.79 Ci/m^3 for BWR non-compactible trash, 0.0228 Ci/m^3 for PWR compactible trash, and 0.525 Ci/m^3 for PWR non-compactible trash.(4,16)

B.2.3 Other LWR Wastes

These wastes consist of non-fuel reactor core components and spent ion exchange resins postulated to result from future routine decontamination of central station nuclear power plants.

TABLE B-18

Measured Activities (μCi/g) Used to Calculate Scaling Factors for Activated Corrosion Products in LWR Process Wastes

Waste Stream	Fe-55	Co-60	Ni-63
BWR's			
Resins	1.74E-01	6.24E+00	9.70E-03
Concentrated Liquids	2.90E-01	9.60E-02	2.20E-03
	7.00E-04	8.90E-03	1.00E-04
Filter Sludges	7.70E+00	1.53E+00	2.80E-02
	2.80E-01	1.23E-01	7.40E-03
PWR's			
Resins	1.01E+00	2.06E+00	1.39E+00
Concentrated Liquids	1.28E-01	3.50E-02	1.91E-02
	4.40E-03	1.89E-02	6.10E-03
Filter Sludges	4.38E-02	1.91E+00	1.10E-02
	9.80E-01	2.55E-01	9.30E-02

Source: Reference 12.

TABLE B-19

Measured Activities (μCi/g) Used to Calculate Scaling Factors For Fission Products in LWR Process Wastes

Waste Stream	H-3	C-14	I-129	Cs-137
BWR's				
Resins	3.00E-03	2.08E-04	<2.00E-06[a]	3.17E+01
	NR[b]	4.00E-05	NR	5.10E-02
Concentrated Liquids	2.50E-03	2.70E-06	1.00E-04	2.29E-01
	1.70E-03	7.11E-06	4.00E-07	4.00E-04
	NR	1.06E-03	NR	1.93E+00
Filter Sludges	NR	1.00E-04	3.00E-06	8.00E-04
	2.00E-03	1.07E-03	1.10E-04	1.13E+00
	NR	1.64E-03	NR	5.10E+00
	NR	2.20E-03	NR	6.30E+00
PWR's				
Resins	1.25E-01	5.01E-03	6.00E-04	2.19E+01
	NR	5.47E-04	NR	2.40E+00
Concentrated Liquids	2.72E-02	2.12E-05	2.00E-05	3.00E-01
	1.32E-01	6.04E-05	4.00E-06	1.02E-01
Filter Sludges	NR	2.36E-02	8.00E-05	1.51E-01
	1.30E-03	3.90E-02	1.80E-06	4.10E-03
	NR	7.39E-04	NR	2.20E-01
	NR	1.73E-01	NR	1.23E+01

(a) Indicates lower limit of detection

(b) NR = Not Reported.

Source: References 11, 12.

TABLE B-20

Measured Activities (μCi/g) Used to Scale Sr-90 Activities in LWR Wastes

Sample Number[a]		Sr-90	Cs-137
BWR's :	T-266 (S)	5.33E-03	3.78E-01
	T-295 (S)	1.42E-02	3.50E-01
	T-246 (R)	3.46E-03	6.59E+01
	T-173 (R)	9.51E-03	5.19E+00
	T-379 (R)	1.09E-04	3.32E-01
PWR's :	T-255 (F)	3.12E-07	4.18E-06
	T-205 (S)	4.04E-04	6.37E-04
	T-309 (S)	3.62E-04	2.54E+00
	T-262 (R)	1.70E-03	7.78E-02
	T-308 (R)	7.06E-04	4.95E-02
	Filter C1	5.53E-09	2.38E-09
	T-256 (R)	1.53E-08	8.80E-07
	T-254 (F)	1.02E-07	5.04E-07
	T-223 (S)	5.45E-05	4.26E-03
	T-217 (S)	1.35E-04	6.17E-03
	T-219 (S)	1.04E-05	8.81E-02
	T-221 (S)	2.10E-05	3.97E-05
	T-200 (S)	5.12E-05	1.22E+00
	T-194 (S)	2.35E-03	2.00E-01
	T-192 (S)	8.37E-03	3.28E+00
	T-191 (S)	3.96E-03	3.03E-01
	T-193 (R)	5.88E-06	1.92E-02
	T-142 (F)	6.50E-03	3.00E-02
	T-215 (R)	9.14E-03	2.75E+01
	T-197 (R)	1.59E-05	3.96E-01
	T-202 (R)	1.01E-04	4.34E-01

(a) Source : Reference 11. (S) = Sludge; (R) = Resin; (F) = Filter.

TABLE B-21

Measured Activities (μCi/g) Used to Scale Pu-241 Activities in LWR Wastes

Waste Stream	Pu-239/240	Pu-241
BWR's		
Resins	2.40E-04	1.60E-01
Filter Sludges	2.50E-02	5.20E-02
	3.80E-03	2.60E-02
	3.70E-03	3.50E-01
	3.50E-05	5.10E-01
	2.10E-02	1.80E+00
	1.40E-03	2.50E-01
	2.22E-06	1.91E-06
	2.40E-02	1.30E-01
	7.50E-03	6.20E-01
PWR's		
Resins	2.70E-04	6.00E-03
	4.70E-04	6.80E-03
	6.60E-06	1.60E-01
Concentrated Liquids	1.00E-06	1.50E-04
	9.20E-05	1.00E-03
Cartridge Filters	6.30E-02	3.80E-01
	1.40E-03	6.90E-02
	4.50E-03	1.60E-02

Source : Reference 9,10.

TABLE B-22

Scaling Factors for Activated Metals and Concrete

	Activated Metals		Activated
Isotope	Low Activity	High Activity	Concrete
H-3	NA[a]	NA	7.41E-07
C-14	6.30E-05	6.42E-05	2.25E-05
Fe-55	5.47E-01	5.52E-01	9.74E-01
Ni-59	3.12E-04	3.46E-04	3.85E-05
Co-60	4.04E-01	3.96E-01	2.17E-02
Ni-63	5.06E-02	5.19E-02	4.53E-03
Nb-94	2.27E-06	2.03E-06	1.60E-06

(a) NA = Not Applicable

Source : Reference 24.

TABLE B-23

Scaling Factors Used for Isotopic Activities in LWR Process Wastes

Isotopes		BWR	PWR
H-3	to Cs-137	9.39E-03	1.21E-01
C-14	to Cs-137	5.84E-04	4.45E-03
Fe-55	to Co-60	5.97E-01	5.17E-01
Ni-59	to Co-60	6.17E-04	6.17E-04
Ni-63	to Co-60	1.35E-02	1.90E-01
Sr-90	to Cs-137	1.78E-03	8.83E-03
Nb-94	to Co-60	1.95E-05	1.95E-05
Tc-99	to Cs-137	3.75E-05	3.75E-05
I-129	to Cs-137	1.00E-04	1.11E-04
Cs-135	to Cs-137	3.75E-05	3.75E-05
U-235	to U-238	1.27E-01	1.27E-01
Np-237	to U-238	2.44E-05	2.44E-05
Pu-241	to Pu-239/240	4.87E+01	4.36E+01
Pu-242	to Pu-239/240	2.19E-03	2.19E-03
Am-243	to Am-241	6.75E-02	6.75E-02
Cm-242	to Am-241	NA[a]	4.79E+00[b]
Cm-243	to Cm-242	2.45E-04	2.45E-04
Cm-244	to Cm-242	NA	1.40E-01[b]

(a) Not Applicable

(b) Used only for PWR Filter Sludge

TABLE B-24

Scaling Factors for LWR Trash

Isotope	BWR	PWR
H-3	2.87E-03	1.33E-02
C-14	1.78E-04	4.90E-04
Fe-55	2.56E-01	2.62E-01
Ni-59	2.64E-04	3.12E-04
Co-60	4.28E-01	5.05E-01
Ni-63	5.78E-03	9.63E-02
Sr-90	5.42E-04	9.73E-04
Nb-94	8.34E-06	9.87E-06
Tc-99	1.14E-05	4.13E-06
I-129	3.04E-05	1.22E-05
Cs-135	1.14E-05	4.13E-06
Cs-137	3.04E-01	1.10E-01
U-235	5.20E-08	3.46E-07
U-238	4.09E-07	2.73E-06
Np-237	9.98E-12	6.65E-11
Pu-238	9.79E-05	2.62E-04
Pu-239/240	4.92E-05	2.43E-04
Pu-241	2.40E-03	1.06E-02
Pu-242	1.08E-07	5.32E-07
Am-241	4.11E-05	1.74E-04
Am-243	2.77E-06	1.17E-05
Cm-243	8.23E-08	1.20E-07
Cm-244	6.35E-05	1.14E-04

Non-fuel reactor core components consist of in-core instrumentation and reactor internals other than fuel rods. They are assumed to be decontaminated so that all of their activity is due to neutron activation of the steel components. Scaling factors given in Table B-22 for highly activated metals are used to estimate their activities.

The purpose of the possible future routine decontamination of LWR's is to reduce the radiation exposure of plant personnel by removing radioactive crud accumulated on the inner surfaces of the primary cooling system. A representative process assumed here involves addition of chemicals and strong chelating agents to the primary cooling water to dissolve and remove the crud from piping and component surfaces. Ion exchange resins are then used to remove the radioactive species from the chelates and, in the final stage, to remove the chelating chemicals. The relative activities of the isotopes of interest should therefore be similar in these resins and in the crud although some variations are expected based on the ion selectivity of the resins. This allows estimation of spent ion exchange resin activities by calculation of scaling factors based on available crud data.

The method used to calculate crud scaling factors is similar to that described in Section B.2.1 for LWR process wastes. The data used to calculate the average activities of the basic isotopes are listed in Table B-25. The basic isotopes for crud scaling are Co-60, Cs-137, U-238, Pu-238, Pu-239/240, Am-241, Cm-242, and Cm-244. The activites labelled LWR averages in Table B-25 are the estimated concentrations of these basic isotopes.

Scaling factors for Fe-55, Ni-63, and Sr-90, and Pu-241 are calculated from experimental data. Iron-55 and Ni-63 are scaled to Co-60 using reported areal activities of 1540, 11, and 760 mCi/dm^2, respectively.(13) Plutonium-241 is scaled to Pu-239/240 and Sr-90 to Cs-137 using experimental data (11). The remaining scaling factors

TABLE B-25 . Activities (μCi/g dry) Measured in LWR Crud.

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
B-1	8.90E-02	<1.30E-04[a]	NA[b]	3.00E-05	5.80E-05	2.10E-05	1.50E-06	8.10E-07
	9.10E-03	5.60E-04	NA	2.00E-06	8.30E-06	6.00E-06	1.60E-05	2.00E-06
	4.80E-02	7.00E-02	NA	<6.00E-07	6.30E-06	3.60E-06	<1.20E-06	<6.00E-07
	4.10E-02	5.00E-03	NA	2.70E-05	3.50E-05	1.20E-05	8.10E-07	5.70E-07
	1.00E-02	9.50E-04	NA	3.60E-06	6.40E-06	4.20E-06	<8.00E-06	3.60E-06
	8.90E-03	<3.00E-05	NA	1.30E-05	2.00E-05	1.30E-05	<5.00E-05	<3.00E-05
	3.50E+02	NA	<8.00E-05	2.20E-01	5.10E-01	<3.00E-04	<3.00E-04	1.30E-03
	5.60E+00	5.90E-01	<3.00E-05	2.40E-03	5.00E-03	2.30E-03	<1.00E-04	2.30E-04
	3.20E-01	3.40E-03	<3.00E-04	<3.00E-04	<3.00E-04	<3.00E-05	<3.00E-05	1.20E-04
	4.50E-02	<1.00E-04	NA	1.20E-05	1.20E-05	4.40E-06	7.60E-07	5.70E-07
	6.10E-02	<1.30E-04	NA	6.40E-05	1.10E-04	7.60E-05	8.50E-06	5.00E-06
	3.50E-02	<3.30E-05	NA	1.20E-05	1.50E-06	7.80E-06	1.90E-05	5.60E-06
	1.24E-02	<3.00E-05	NA	8.60E-06	1.20E-05	2.00E-05	3.30E-05	<1.20E-05
	6.10E-02	2.90E-03	NA	5.00E-05	1.00E-04	2.80E-05	1.25E-04	2.00E-05
	6.10E-01	NA	<3.00E-07	3.60E-04	8.20E-04	7.90E-04	2.50E-05	2.00E-05
B-3	3.58E-02	1.68E-03	--	1.75E-06	6.73E-06	4.18E-06	2.50E-04	2.88E-05
	2.74E-02	<6.50E-05	--	1.63E-06	3.67E-05	<7.68E-07	4.46E-06	4.07E-07
	1.20E-02	1.44E-04	--	2.31E-04	7.97E-04	7.18E-06	3.62E-05	3.51E-06
	1.35E-01	<1.09E-04	--	1.48E-05	7.60E-06	2.96E-05	<1.00E-06	1.20E-05
B-5	7.30E-01	2.30E+00	<5.00E-06	1.50E-04	7.00E-05	<2.00E-07	<2.00E-07	2.20E-05
B-7	5.42E+01	<7.26E-02	NA	6.80E-04	3.10E-04	1.70E-04	7.50E-03	1.02E-03
	2.32E+01	<2.18E-02	NA	NA	NA	NA	NA	NA
	1.43E+02	<6.29E-02	NA	1.30E-05	7.10E-06	4.60E-04	1.80E-04	1.90E-04
B-8	1.08E+00	1.27E-01	--	1.16E-04	8.14E-05	3.93E-05	5.73E-05	3.10E-05
	3.80E-02	4.50E-04	<2.00E-06	6.60E-05	4.80E-05	<2.00E-06	1.50E-05	<2.00E-06
	1.71E+00	NA	<5.00E-07	1.10E-03	9.60E-04	2.10E-04	3.90E-03	1.20E-04
	4.81E-02	2.04E-02	--	4.76E-05	2.80E-05	1.16E-05	2.36E-05	1.42E-05

TABLE B-25 (Cont'd)

Plants	Co-60	Cs-137	U-238	Pu-238	Pu-239/240	Am-241	Cm-242	Cm-244
B-9	8.90E-01	1.60E-02	NA	7.40E-05	6.00E-05	1.40E-04	3.30E-05	<3.00E-05
BWR Average	2.39E-01	2.21E-03	7.28E-06	4.88E-05	6.80E-05	2.22E-05	2.25E-05	1.30E-05
P-1	1.72E+00	<1.72E-03	<6.00E-07	7.10E-05	2.00E-04	3.00E-06	2.10E-03	<3.00E-06
	2.84E-01	<2.92E-04	<6.00E-07	1.30E-05	4.80E-06	4.30E-06	6.70E-05	<2.00E-06
	7.26E+00	<7.30E-03	<5.00E-08	4.50E-04	1.70E-03	9.50E-05	3.00E-03	1.30E-04
	1.57E+01	<1.63E-02	NA	2.00E-03	4.70E-03	3.80E-04	3.70E-03	1.10E-04
P-2	1.23E-04	4.37E-06	NA	5.60E-08	6.20E-08	6.60E-08	2.00E-06	1.60E-07
P-5	[illegible].08E-01	5.92E-03	NA	1.50E-04	2.30E-04	8.40E-05	2.30E-03	1.30E-04
	6.42E+00	<4.61E-03	NA	4.20E-05	4.20E-05	5.30E-06	1.19E-04	5.80E-06
	6.36E+01	<5.05E-02	NA	9.20E-04	9.80E-04	8.00E-03	3.60E-03	1.50E-04
P-7	2.28E-04	1.72E-05	--	1.59E-05	4.63E-06	1.67E-06	1.47E-04	9.77E-06
	6.25E-04	<5.47E-07	--	1.12E-05	3.35E-06	8.26E-07	3.63E-05	2.91E-06
P-8	3.53E+00	<2.66E-02	--	4.25E-03	5.72E-03	5.72E-03	1.19E-02	1.80E-03
P-9	1.25E+00	<1.53E-03	--	2.70E-05	3.66E-05	9.94E-06	2.02E-04	8.26E-06
	4.70E+02	NA	<3.00E-06	2.20E-02	4.80E-02	3.30E-03	2.30E-02	6.20E-04
	1.74E+02	NA	<3.00E-06	2.80E-03	5.90E-03	7.70E-04	5.70E-03	2.30E-04
PWR Average	8.32E-01	8.48E-04	6.95E-07	1.29E-04	1.54E-04	3.94E-05	6.87E-04	2.67E-05
LWR Average	3.70E-01	1.62E-03	2.74E-06	6.80E-03	8.99E-03	2.69E-05	7.22E-03	1.66E-05

(a) Indicates lower limit of detection
(b) NA = Not Analyzed

TABLE B-26

Intermediate Scaling Factors Used in Calculation of Crud Scaling Factors

Isotopes	Scaling Factor
H-3 to Cs-137	3.39E-02
C-14 to Cs-137	1.61E-03
Fe-55 to Co-60	5.56E-01
Ni-59 to Co-60	6.17E-04
Ni-63 to Co-60	5.06E-02
Sr-90 to Cs-137	1.35E-01
Nb-94 to Co-60	1.95E-05
Tc-99 to Cs-137	3.75E-05
I-129 to Cs-137	1.05E-04
Cs-135 to Cs-137	3.75E-05
U-235 to U-238	1.27E-01
Np-237 to U-238	2.44E-05
Pu-241 to Pu-239/240	2.00E+01
Pu-242 to Pu-239/240	2.19E-03
Am-243 to Am-241	6.75E-02
Cm-243 to Cm-242	2.45E-02

TABLE B-27

Scaling Factors Used for Decontamination Spent Resin Wastes Based on LWR Crud

Isotope	Scaling Factor
H-3	6.90E-05
C-14	3.29E-06
Fe-55	2.60E-01
Ni-59	2.88E-04
Co-60	4.67E-01
Ni-63	2.36E-02
Sr-90	2.74E-04
Nb-94	9.11E-06
Tc-99	7.67E-08
I-129	2.14E-07
Cs-135	7.67E-08
Cs-137	2.40E-03
U-235	4.39E-07
U-238	3.46E-06
Np-237	8.44E-11
Pu-238	8.58E-03
Pu-239/240	1.13E-02
Pu-241	2.27E-01
Pu-242	2.48E-05
Am-241	3.39E-05
Am-243	2.30E-06
Cm-243	2.22E-06
Cm-244	2.10E-05

given in Table B-26 are calculated as geometric means of the BWR and PWR process waste scaling factors given in Table B-22. These intermediate scalings are used to calculate isotopic activities in units of µCi/g of crud. The scaling factors given in Table B-27 are the fractional activities of each isotope with respect to the total activity of all 23 isotopes in crud and can be applied to the total activity of waste streams expected to have isotopic distribution similar to crud.

B.3 OTHER WASTE STREAMS

This section considers LLW streams originating from sources other than LWR's. These wastes are discussed in three sections: other nuclear fuel cycle wastes, institutional wastes, and industrial wastes.

B.3.1 Other Nuclear Fuel Cycle Wastes

These wastes consist of process wastes from uranium hexafluoride conversion plants and process wastes and trash from fuel fabrication plants. The only isotopes of interest identified in these wastes are U-235 and U-238. Since UF_6 plants process unenriched uranium containing about 0.711 weight percent U-235, 4.3 percent of the total activity is assigned to U-235 and 95.7 percent to U-238.

Fuel fabrication plants process materials enriched to about 4.0% uranium weight percent U-235, so that 21.2 percent of the total activity of their wastes is due to U-235 and 78.8 percent due to U-238. The U-238 activity is calculated from 1977 Maxey Flats disposal records (14) by assigning the total reported weight of special nuclear material in each type of waste to U-235.

B.3.2 Institutional Wastes

Isotopic activites of institutional wastes are rarely determined by direct measurements. The utility of such measurements is questionable due to the diversity of uses of radioactive materials at institutions. This situation necessitates use of data obtained during a 1977 survey of institutional low-level waste generators to estimate activities of institutional waste streams.(7,8) These data, which consists of a reformatted presentation of the information gathered during the survey, have been presented in Table 3-14. The data includes the total activity of each isotope shipped and the total waste volume reported to contain a particular isotope. An "X" indicates waste streams expected to contain a given isotope.

The method (see Section 3.2.3) used to estimate the isotopic activities of institutional waste streams is conservative since they are calculated by assigning the total activity to only a fraction of the total volume of waste shipped and then assigning equal concentrations to that fraction of each waste stream containing a given isotope.

B.3.3 Industrial Wastes

Estimation of the activities of industrial wastes are based primarily upon a number of information sources provided by the NRC.(15,16) These sources include data taken from Maxey Flats and Barnwell disposal site radioactive waste shipment records (RSR's).

Medical Isotope Production Wastes

Medical isotopes are produced by neutron irradiation of highly enriched uranium encased in steel and aluminum capsules. The irradiation capsules are assumed to be included in the high activity industrial waste stream. All other wastes are considered here as a single waste stream. The isotopic composition of these wastes has not been determined directly but is assumed to resemble LWR spent fuel. However, due to the comparatively short irradiation time and the fact that the material is highly enriched, very low concentrations of transuranic isotopes are present in these wastes.

Of the isotopes of concern, only the combined Sr-90/Cs-137 activity and grams of U-235 are quantified in Maxey Flats RSR's. The average Sr-90/Cs-137 activity is 15.1 Ci/m^3 and the average U-235 activity is 1.13×10^{-3} Ci/m^3 (corresponding to a reported U-235 content of 526 g/m^3). The scaling factors listed in Table B-28 have been calculated using this data, the isotopic composition of spent fuel,

TABLE B-28

Scaling Factors for Medical Isotope Production Wastes

Isotope	Activity in Spent Fuel[a] (Ci/metric ton)	Scaling Factor
Group 1		
H-3	5.14E+02	2.80E-03
C-14	5.54E-01	3.00E-06
Fe-55	0	-
Ni-59	0	-
Ni-63	0	-
Co-60	0	-
Sr-90	7.68E+04	4.18E-01
Nb-94	0	-
Tc-99	4.01E+00[b]	2.18E-05
I-129	3.33E-02	1.81E-07
Cs-135	4.01E+00[b]	2.18E-05
Cs-137	1.07E+05	5.82E-01
Subtotal :	1.84E+05	
Group 2		
U-235	1.71E-02	1.02E-05[c]
U-238	3.14E-01	3.81E-05[c]
Group 3		
Np-237	7.66E-06	6.13E-11
Pu-238	2.82E+03	2.26E-02
Pu-239/240	7.98E+02	6.38E-03
Pu-241	1.02E+05	8.16E-01
Pu-242	1.37E+00	1.10E-05
Am-241	1.58E+02	1.26E-03
Am-243	1.80E+01	1.44E-04
Cm-242	1.69E+04	1.35E-01
Cm-243	4.14E+00[b]	3.31E-05
Cm-244	2.38E+03	1.90E-02
Subtotal :	1.25E+05	

(a) Taken from Reference 25.

(b) Calculated from Cs-137 activity using LWR scaling factor.

(c) Based on Maxey Flats RSR data.

and a conservatively assumed transuranic radionuclide activity of 1.60×10^{-3} Ci/m^3 (corresponding to 1.0 nCi/g for a waste density of 1.6 g/cm^3).

Scaling factors are calculated in three groups. Isotopes in Group 1 with non-zero activities are fission products, and their scaling factors are calculated as fractions of the total activity of the Group 1 isotopes in spent fuel. The scaling factors for the Group 1 isotopes are applied to an activity of 15.0 Ci/m^3. The U-235 scaling factor is the ratio of U-235 activity from the RSR's to the total activity of the waste. The U-238 activity is calculated from the above quoted U-235 activity by conservatively assuming 4 percent enrichment. The scaling factors for the transuranics in Group 3 are assumed to be applicable as fractions of the total transuranic activity in spent fuel. The total alpha emitting transuranic activity (all nuclides except Pu-241 in Table B-28) is assumed to be 1.0 nCi/g or 1.6 nCi/cm^3 (one-tenth the applicable limit of 10 nCi/g). These alpha emitting transuranic nuclides are 18.4 percent of the total activity as shown in Table B-28. This yields a total transuranic activity of 8.7 nCi/ml; the isotopic concentrations are calculated from this activity using the scaling factors given in Table B-28.

This method ignores extraction of Tc-99 and other isotopes from the waste stream for sale as medical isotopes. It also ignores the fact that the uranium targets are highly enriched. Radiochemical analysis of these wastes is needed for more accurate characterization.

High Activity Wastes

High activity wastes consist of neutron irradiation capsules, activated components from research reactors, and other activated waste materials. Isotopic activities of these wastes are calculated using the scaling factors for highly activated metals from decomissioning activities given in Table B-22.

Tritium Production and Manufacturing Wastes

A common industrial method of producing tritium is neutron irradiation of lithium targets. Irradiation of these targets does not induce significant quantities of long-lived radioisotopes other than tritium. Thus the total radioactivity of industrial tritium production wastes, 2330 Ci/m^3, is assumed to be due to tritium alone.(16)

Sealed Sources

Estimation of the activity of sealed sources and foils, and the isotopic distribution of this activity, is difficult since they are shipped for disposal infrequently and at irregular intervals. The following radionuclides and scaling factors are assumed for this stream and applied to a total activity of $5.76x10^3$ Ci/m^3:

Nuclide	Activity Fraction
H-3	0.15
C-14	0.01
Ni-63	0.04
Co-60	0.30
Sr-90	0.20
Cs-137	0.20
Am-241	0.10

Accelerator Targets

Accelerator targets consist of tritium absorbed on titanium foils. Since there is no indication that induced activities are present,(6) the total activity of 80.4 Ci/m^3 contained in this waste stream(16) is assumed to be due to tritium.

Source and Special Nuclear Material Wastes

The only radionuclides identified in source and special nuclear material wastes are U-235 and U-238. These wastes are generated primarily during processing of metals and/or compounds containing depleted uranium. The uranium isotopes are conservatively assumed to be present in the same ratio as in natural uranium; therefore, 4.3 percent of the total activity is assumed to be due to U-235 and 95.6 percent due to U-238, and these fractions are applied to total activities of 0.217 mCi/m^3 and 0.0112 mCi/m^3 for waste and trash, respectively.

Low Specific Activity Waste

The types of materials comprising the industrial low activity waste stream are the industrial equivalents of institutional wastes - i.e., trash, liquid scintillation vials, absorbed liquids, and biowastes. These wastes are not sufficiently well-characterized to be considered as separate streams. It is therefore assumed that these industrial wastes have the same distribution of radionuclide concentrations as institutional wastes. The scaling factors estimated for these wastes are presented in Table B-29.

TABLE B-29

Scaling Factors for
Low Activity Industrial Wastes

Isotope	General Wastes	Trash
H-3	7.73E-01	8.07E-01
C-14	4.44E-02	4.65E-02
Co-60	6.97E-02	9.21E-02
Sr-90	6.21E-02	1.28E-02
Te-99	3.68E-08	3.00E-08
Cs-137	4.93E-02	4.02E-02
Am-241	0	1.51E-06

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APPENDIX C

VOLUME REDUCTION TECHNOLOGY

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APPENDIX C : VOLUME REDUCTION TECHNOLOGY

This appendix contains a brief description of the currently available volume reduction processes which may be utilized to process LLW, and information on their effects on LLW.

C.1 INTRODUCTION

The four basic types of volume reduction processes which can be applied to low-level radioactive wastes are compaction, evaporation, calcination, and incineration. Each of these processes generates a concentrate stream and an effluent stream. The concentrate streams are compacted waste for compaction, concentrated liquids and/or solids (crystals) for evaporation and calcination, and ash for incineration. The effluent streams are displaced air for compaction, distillate for evaporation and calcination, and off-gases and vapors for incineration.

The efficiency of a volume reduction process is described by its volume reduction factor (VRF) and its decontamination factor (DF). The volume reduction factor is usually defined as:

$$VRF = V_w/V_c \tag{C.1}$$

where:

V_w = volume of waste treated in time interval t
V_c = volume of concentrate produced in time interval t

This definition assumes that effluent treatment systems generate negligible volumes of secondary wastes (such as HEPA filters, liquid filters, scrubber solutions, etc.).

The effluent stream normally contains trace amounts of contaminants. The degree of decontamination of the effluent is expressed as the decontamination factor (DF):

$$DF = A_w/A_e \qquad (C-2)$$

where:

A_w = amount of a specified component (mass or radioactivity) in the waste treated in time interval t.

A_e = amount of the specified component in the effluent in time interval t.

Decontamination factors can be strongly influenced by the type of effluent treatment used.

This appendix describes the types of volume reduction equipment available and, where information is available, discusses achievable volume reduction and decontamination factors for each type. Compactors are described in Section C.2, evaporators in Section C.3, and incinerators and calciners in Section C.4. Section C.5 describes dual function systems. Some of these systems can both evaporate and calcine wastes while others are capable of evaporation and solidification. Ion exchange and filtration are not considered here as volume reduction techniques.

C.2 COMPACTION

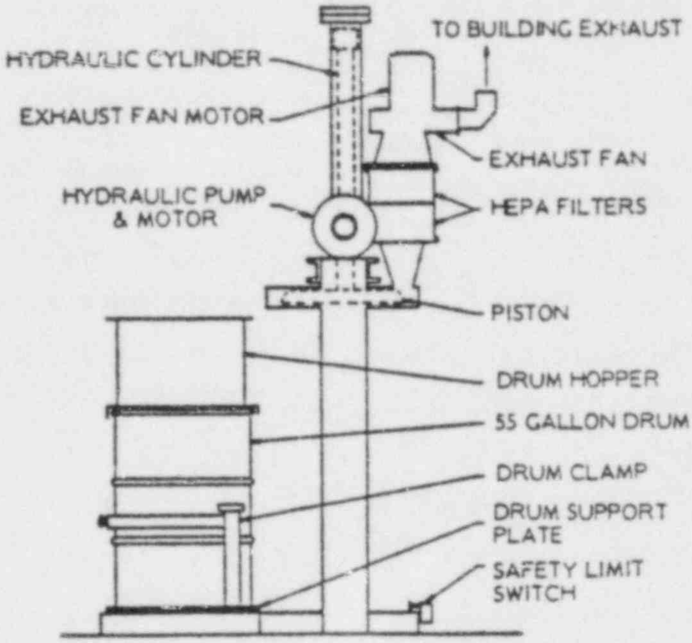
Compactors are frequently used at LWR's to reduce trash volume; trash typically consists of paper, rags, glassware, disposable clothing, etc. Compactors compress these wastes, driving out air as voids are reduced. The amount of void volume and the resiliency of trash materials limit the final volume reduction attained.

Typical trash compactors consist of a vertical mechanical or hydraulic ram, a platen, and a protective shroud and air filtration system, and use a standard 55 gallon drum as the compaction vessel. A hydraulic compactor is shown in Figure C-1. Common compactors generate a compressive force ranging from 30 to 150 kg/cm^2 (430-2100 psi) with hydraulic compactors operating at the higher end of this range. Standard compactors are reported to attain an average volume reduction factor of two.(1)

New hydraulic compactors insert a metal sleeve into the drum before compression and remove it at the end of the cycle. The metal sleeve allows greater compressive force to be applied without increasing the risk of drum failure. These modified compactors are capable of volume reduction factors of up to four.(1)

Several novel trash compactors have been used at LWR's. These include a double hydraulic ram device which uses a plywood box as the compaction vessel and a large compactor for use with 90 ft^3 liners.

Although not currently used for compaction of low-level wastes, industrial hydraulic presses of the type used to crush automobiles may be useful for compaction of metal items such as pipes.



HYDRAULIC TRASH COMPACTOR

DAMES & MOORE

FIGURE C-1

C.3 EVAPORATION

Evaporators concentrate liquid wastes by heating them to vaporize the volatile components. These wastes are almost always aqueous solutions. The vaporized water is relatively free of the dissolved and suspended solids and the radioactivity found in the input solution. In the nuclear industry, the vaporized water is rarely released directly to the environment but usually condensed and collected. After testing to determine whether the condensate requires additional treatment, it is discharged or recycled within the facility. The concentrated solution (bottoms) left in the evaporator retains the bulk of the solids and radioactivity, and it is usually solidified and shipped to a disposal site.

Although they are rarely used by non-fuel cycle waste generators, nearly all LWR's have at least one radioactive waste evaporator. These evaporators can be categorized according to their methods of heat transfer.

Natural circulation evaporators (Section C.3.1) use convection as the means of heat transfer. Forced circulation evaporators (Section C.3.2) use pumps to improve the flow of liquid over the heating surface. Evaporative crystallizers (Section C.3.3) are forced circulation evaporators specially designed to handle high concentrations of solids. Wiped film evaporators (Section C.3.4) mechanically spread a thin film of waste liquid on the heating surface. Each type of evaporator may by oriented horizontally or vertically and have external or internal heat exchangers.(2,3)

Volume Reduction Factor

The volume reduction factor (VRF) of an evaporator is defined as the ratio of the volume of liquid fed to the evaporator in a given time interval to the volume of bottoms produced in that time interval. The evaporator bottoms, also called concentrate or thick liquor, may be a more concentrated solution of the original waste, a slurry, or moist crystals of the original dissolved salts.

The volume reduction factor depends on the initial concentration and solubility of the dissolved salts and on the type of evaporator. The initial salt concentration and solubility determine the amount of volume reduction possible before the liquid becomes saturated and crystallization begins. Beyond this point, the ability of the evaporator to circulate the resultant slurry becomes the controlling factor. With these factors in mind, reported volume reduction factors (1,2) ranging from 3 to 1500 are not surprising. Volume reduction factors of 10 to 20 are typical with PWR bottoms averaging about 12 weight percent solids and BWR bottoms about 25 weight percent. For a given liquid waste, the volume reduction factor and final solids content are lowest for natural circulation evaporators, and highest for evaporative crystallizers and wiped film evaporators.

Decontamination Factors

An evaporator decontamination factor is defined as the ratio of the total amount of specified radioactivity fed to the evaporator in a time interval t to the total amount of that radioactivity in the condensate in the time interval t. As shown in Tables C-1 and C-2, decontamination factors vary with evaporator type, radioactive species, and waste liquid.

Decontamination factors are adversely affected by entrainment, splashover, foaming, and volatization of the solutes. Entrainment results in the carry-over of fine droplets of concentrated waste liquid to the condensate. Most evaporators have flash chambers and entrainment separators to prevent the droplets from reaching the condensate or being released if the vapor is not condensed. Entrainment can also be reduced by maintaining the boiling rate in a range low enough to

TABLE C-1

Mean Decontamination Factors by Evaporator Type

Type of Radioactivity	Overall	Natural Circulation	Forced Circulation[a]	Spray Film	Submerged U-tube
Gross Alpha & Beta	2.1E+04	4.2E+04	1.6E+04	3.0E+03	9.0E+03
Iodine	9.9E+02	1.1E+03	1.6E+03	3.2E+02	7.0E+02
Fission products	2.3E+04	4.3E+04	1.3E+04	1.6E+03	2.8E+03
Corrosion products	1.1E+04	2.5E+04	1.4E+04	3.3E+03	3.3E+03

(a) Includes evaporative crystallizers.

Source: Reference 2

TABLE C-2

Evaporator Decontamination Factors
by LWR Liquid Waste Type

Waste Stream	Decontamination Factor All Nuclides Except Iodine	Iodine
PWR		
Miscellaneous radwaste	10^4	10^3
Boric acid wastes	10^3	10^2
Laundry wastes	10^2	10^2
BWR		
Miscellaneous radwaste	10^4	10^3
Laundry wastes	10^2	10^2

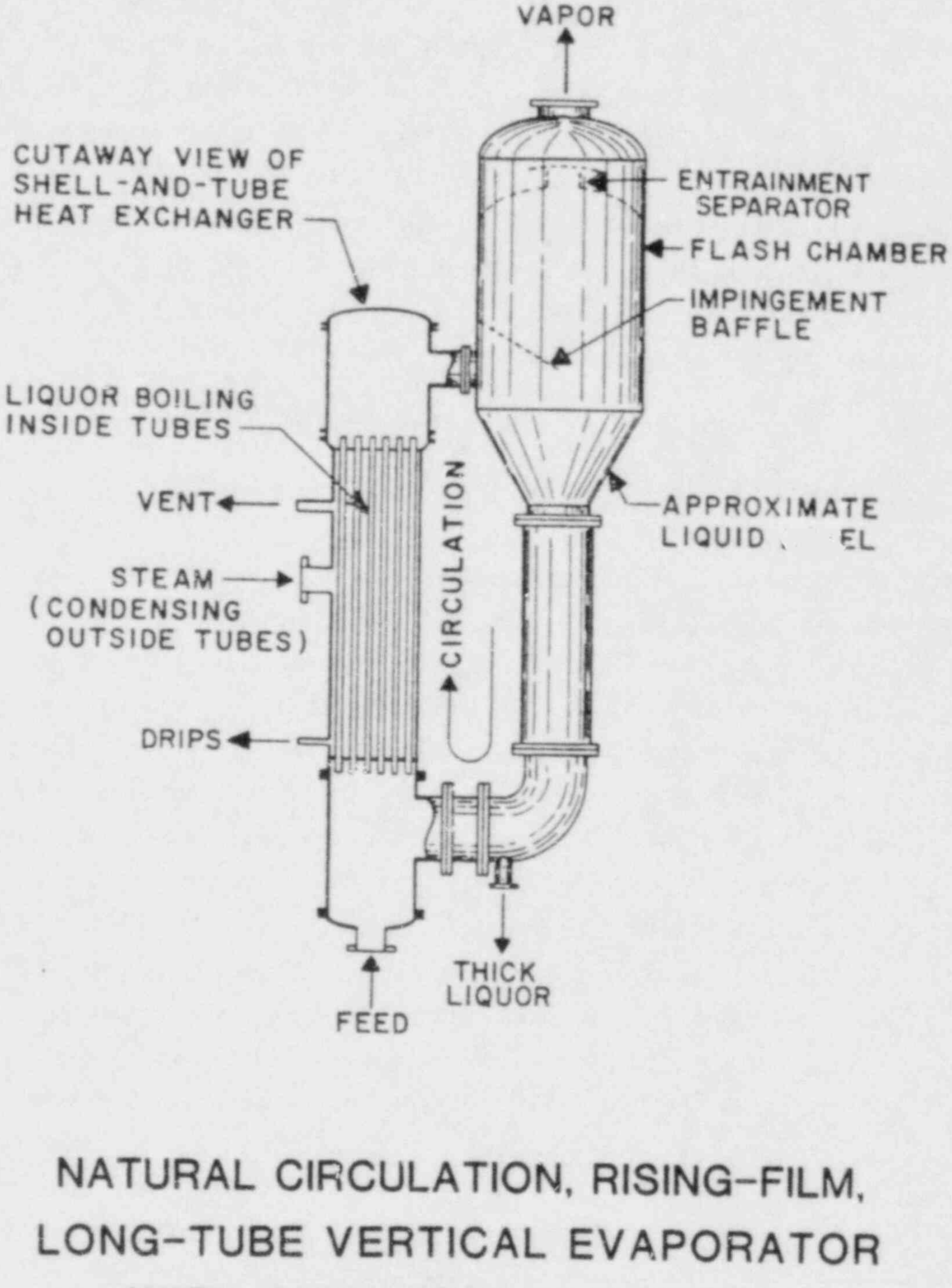
Source: References 4 and 5.

prevent entrainment of large droplets but high enough to prevent formation of very fine droplets which are easily entrained. At very high boiling rates, bulk liquid can escape the flash chamber (splash over) and contaminate the condensate. Foaming during boiling increases entrainment. Foaming is usually caused by organic compounds, finely divided solids, and dissolved gases. It is reduced by mechanical devices which break up the foam and by chemical antifoaming agents which prevent its formation. Solute volatilization is more difficult to control, especially for iodine which can escape as molecular iodine or by forming volatile compounds with organic solutes.

C.3.1 Natural Circulation

Use of natural circulation evaporators for radwaste treatment is decreasing and is largely confined to older LWR's. Since these evaporators rely on convection for heat exchange, they cannot tolerate high solids concentrations and as a result, they cannot match the volume reduction factors attained by other types of evaporators. Most types of natural circulation evaporators give DF's of about 10^3 for iodine and about 10^4 for other radionuclides (Table C-1). Volume reduction factors vary with the waste stream treated.

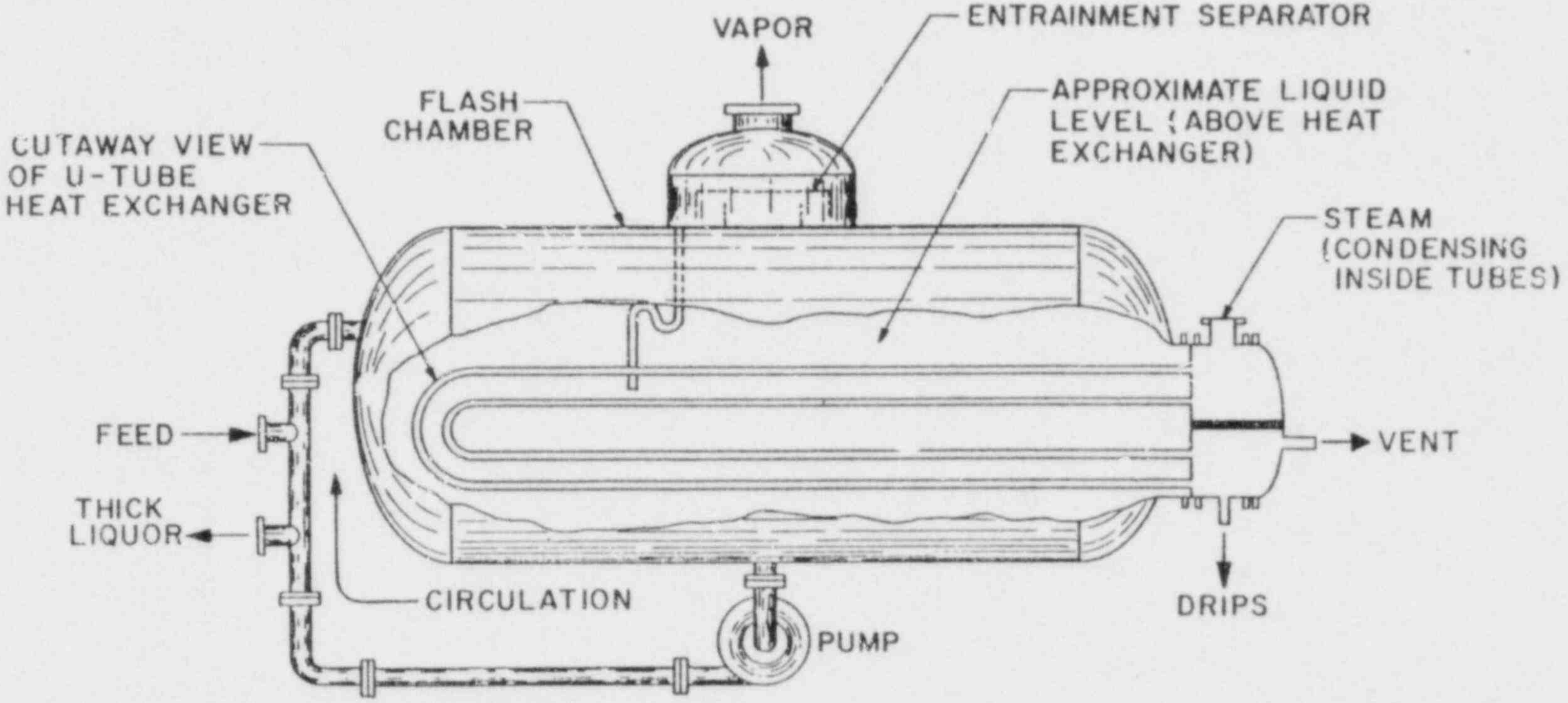
Natural circulation evaporators use either long-tube vertical heaters as shown in Figure C-2, or submerged U-tube heaters as shown in Figure C-3. Short tube heaters are less common. Since steam is abundant in an operating LWR, it is used as the heat exchange medium. Rising film evaporators introduce waste liquids at the bottom of the vertical heater. The heated liquid is confined within the heater tubes, sometimes rising very rapidly, as it boils into the flash chamber. Heat transfer is poor due to hydrostatic head friction and rapid acceleration of the liquid up the tubes as it boils. The problems are significantly reduced by introducing waste liquid at the top of the vertical heater. Such an evaporator is aptly called a falling film evaporator.



NATURAL CIRCULATION, RISING-FILM,
LONG-TUBE VERTICAL EVAPORATOR
WITH AN EXTERNAL HEATER

DAMES & MOORE

FIGURE C-2



SUBMERGED U-TUBE EVAPORATOR

FIGURE C-3

Submerged U-tube evaporators have horizontal U-tube heaters. In this case, the heat exchange medium (steam) is confined within the heater tubes. These evaporators have DF's approximately one order of magnitude less than the vertical heater type. This is due to the shorter distance between the surface of the boiling liquid and the vapor exit of the flash chamber. Although the submerged U-tube evaporator shown in Figure C-3 has a pump, it is a natural circulation evaporator since heat transfer occurs by convection.(2) The pump is used for circulation -- i.e..introducing feed and removing concentrate -- and is not used to improve heat exchange.

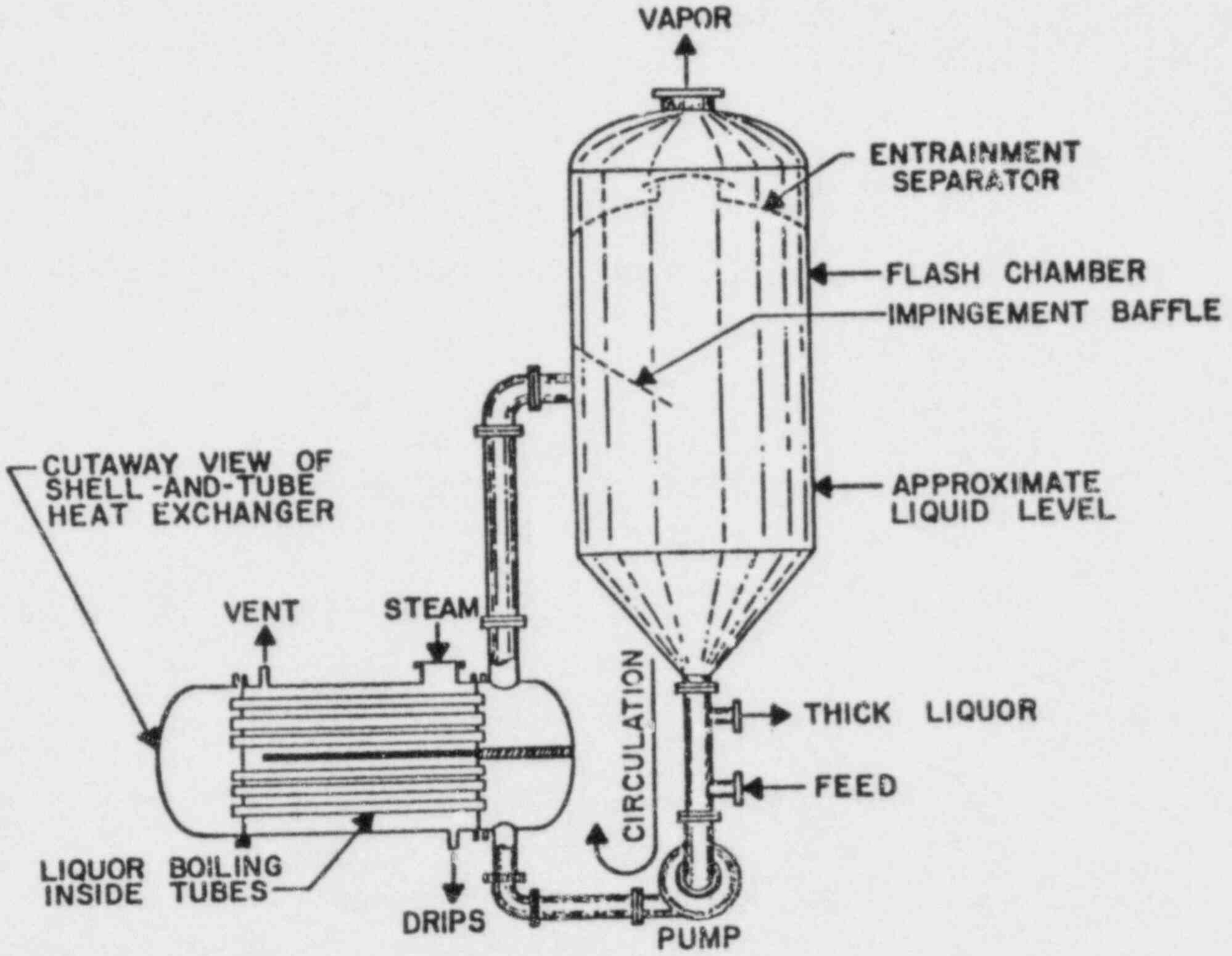
C.3.2 Forced Circulation

Forced circulation evaporators use mechanical devices to force liquid waste over the heating surface. This broad definition includes all evaporators other than natural circulation evaporators. For convenience, the discussion of forced circulation evaporators in this section is restricted to those which produce bottoms containing up to 25 weight percent solids.

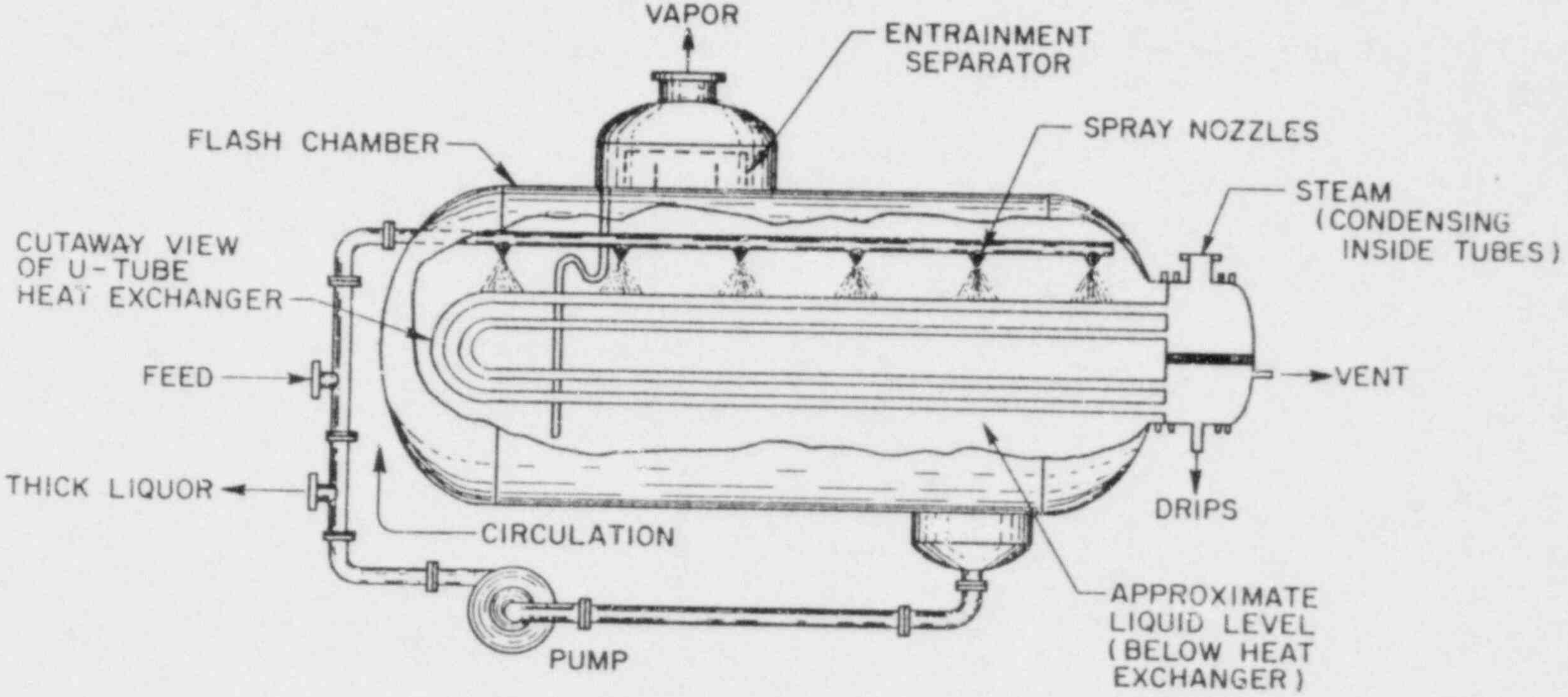
Figure C-4 shows a typical forced circulation evaporator. The pump which forces the waste liquid through the heater tubes distinguishes this evaporator from natural circulation types. Internal heaters are seldom used; the external heater may be oriented vertically or horizontally. Waste liquids normally are pumped in one direction in vertical heaters (single pass) while liquid normally flows in one direction in one section of a horizontal heater and in the opposite direction in the next section (two pass heater). This type of evaporator normally gives decontamination factors of about 10^3 for iodine and 10^4 for other radionuclides.

Spray film evaporators (Figure C-5) are a less common type of forced circulation evaporator. In this case a pump is used to force the waste liquid through spray nozzles directed onto the heating surface.

FIGURE C-4



FORCED-CIRCULATION EVAPORATOR WITH AN EXTERNAL, HORIZONTAL, SUBMERGED-TUBE, TWO-PASS HEATER



SPRAY-FILM EVAPORATOR WITH AN EXTERNAL SUBMERGED-TUBE, TWO-PASS HEATER

FIGURE C-5

The atomized liquid spreads rapidly into a thin film and evaporates rapidly on contact with the hot surfaces. Typical decontamination factors are about 10^2 for iodine and about 10^3 for other radionuclides.

C.3.3 Evaporative Crystallizers

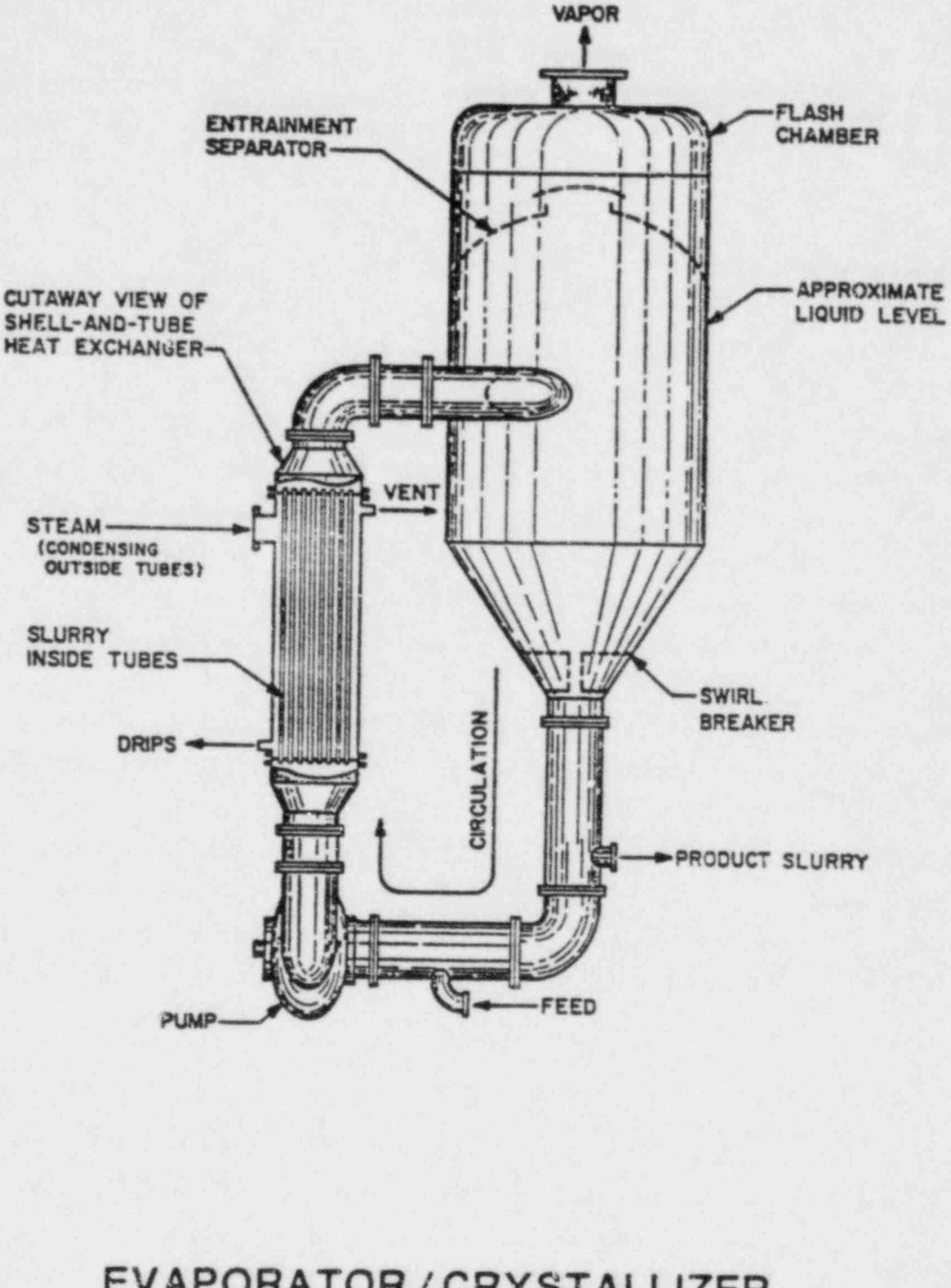
Evaporative crystallizers (Figure C-6) differ from the forced circulation evaporators just described in that crystallizers can handle bottoms containing about 50 weight percent solids. At this solids concentration, the bottoms are thick slurries containing large quantities of bulk solids. To accomodate these slurries, evaporative crystallizers use more powerful pumps and larger diameter pipes than are used in other forced circulation evaporators. Decontamination factors are about 10^3 for iodine and 10^4 for other radionuclides.

Depending on the initial concentrations and solubilities of dissolved solids in the liquid waste, an evaporative crystallizer may consist of two forced circulation evaporators operating in tandem. The first preconcentrates the waste for crystallization in the second evaporator. When added to existing evaporators an additional volume reduction factor of about 6 is attainable for 12 weight percent boric acid waste and about 2.4 for 25 weight percent sodium sulfate wastes.

C.3.4 Wiped Film Evaporators

Wiped film evaporators (Figure C-7) use a rotor to mechanically spread a thin film of waste liquid on the inside surface of a cylindrical heated surface. As the salts build up they are scraped off by the rotor and discharged from the evaporator. The evaporator may be horizontal or vertical.

Wiped film evaporators can be operated so that dry crystals are produced. When operated in this way, a volume reduction factor of



EVAPORATOR / CRYSTALLIZER

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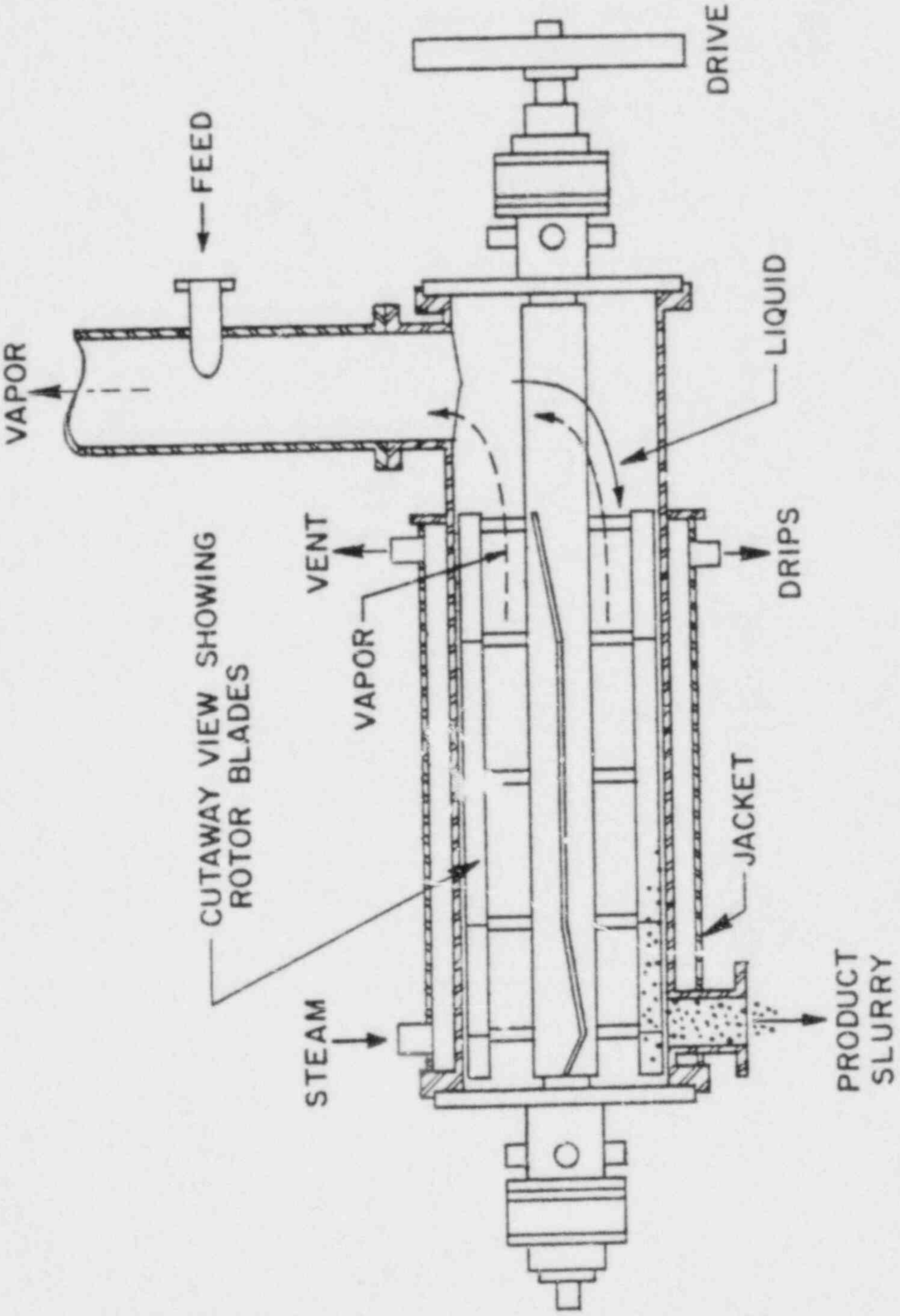


WIPED FILM EVAPORATOR

DAMES & MOORE

FIGURE C-7

about 8 is attained for 12 weight percent boric acid waste and about 6 for 25 weight percent sodium sulfate waste.

Use of wiped film evaporators in a single step evaporation/bitumenization process is described in Section C.5.2.

C.4 INCINERATION

Incinerators and related devices decompose combustible waste materials by thermal oxidation. Combustion or incineration involves complete oxidation of wastes by burning in an excess of oxygen (air). Pyrolysis involves partial oxidation in an oxygen deficient atmosphere. Oxidation can also be accomplished by introducing combustible wastes and air into a bath of molten salt. Acid digesters thermally and chemically oxidize wastes in a hot mixture of concentrated nitric and sulfuric acids.

Many types of incinerators, pyrolyzers, and other such devices are being developed for volume reduction of radioactive wastes. Table C-3 gives a partial listing of these devices. Many of the incinerators listed are being developed by the U.S. Department of Energy for processing TRU waste. Reference 23 summarizes the development and application of these and other incinerators.

Volume Reduction Factor

The volume reduction factor for incineration is defined as the ratio of the volume of combustible waste fed into the system during a given time interval to the volume of ash or residue produced in that same time interval. The ash produced may tend to compact on handling, introducing some error into measured VRF's.

Available volume reduction factors are presented in the appropriate section. Volume reduction factors of the various incinerators are expected to be roughly the same for a given waste. Volume reduction factors for pyrolyzers may be slightly lower than for incinerators due to incomplete oxidation of organic materials.

The volume of any secondary wastes produced by off-gas treatment is generally a small fraction of the total waste volume processed.

TABLE C-3

Incinerators in Use or Under Development for Radioactive Wastes

Incinerator Type	Described in Section	References
Acid Digestion	C.4.7	6,7,8,9,18
Agitated Hearth	C.4.2	9,10
Controlled Air	C.4.3	1,9,11,12,13,18
Cyclone Drum	C.4.4	9,10,14,15,18
Electromelt Furnace	C.5.4	18
Fluidized Bed	C.5.1	9,10,16,17,18
Molten Salt	C.4.8	1,9
Pathological	C.4.1	19,20,21
Pyrolysis/Controlled Air	C.4.6	10
Rotary Kiln	C.4.5	9,18

Decontamination Factors

The decontamination factor for incineration is defined as the ratio of the total amount of (mass or radioactivity) of a given species fed in time internal t to the total amount of that species in the treated off-gas in time interval t. Many types of incinerators are equipped with sophisticated off-gas treatment systems which significantly reduce the amount of radioactive particulates and iodine released. Release of tritium and carbon-14 as combustion products (H_2O, CO_2) is more difficult to control. Variations in the designs of off-gas treatment systems for a given type of incinerator also complicate estimation of DF's by incinerator type.

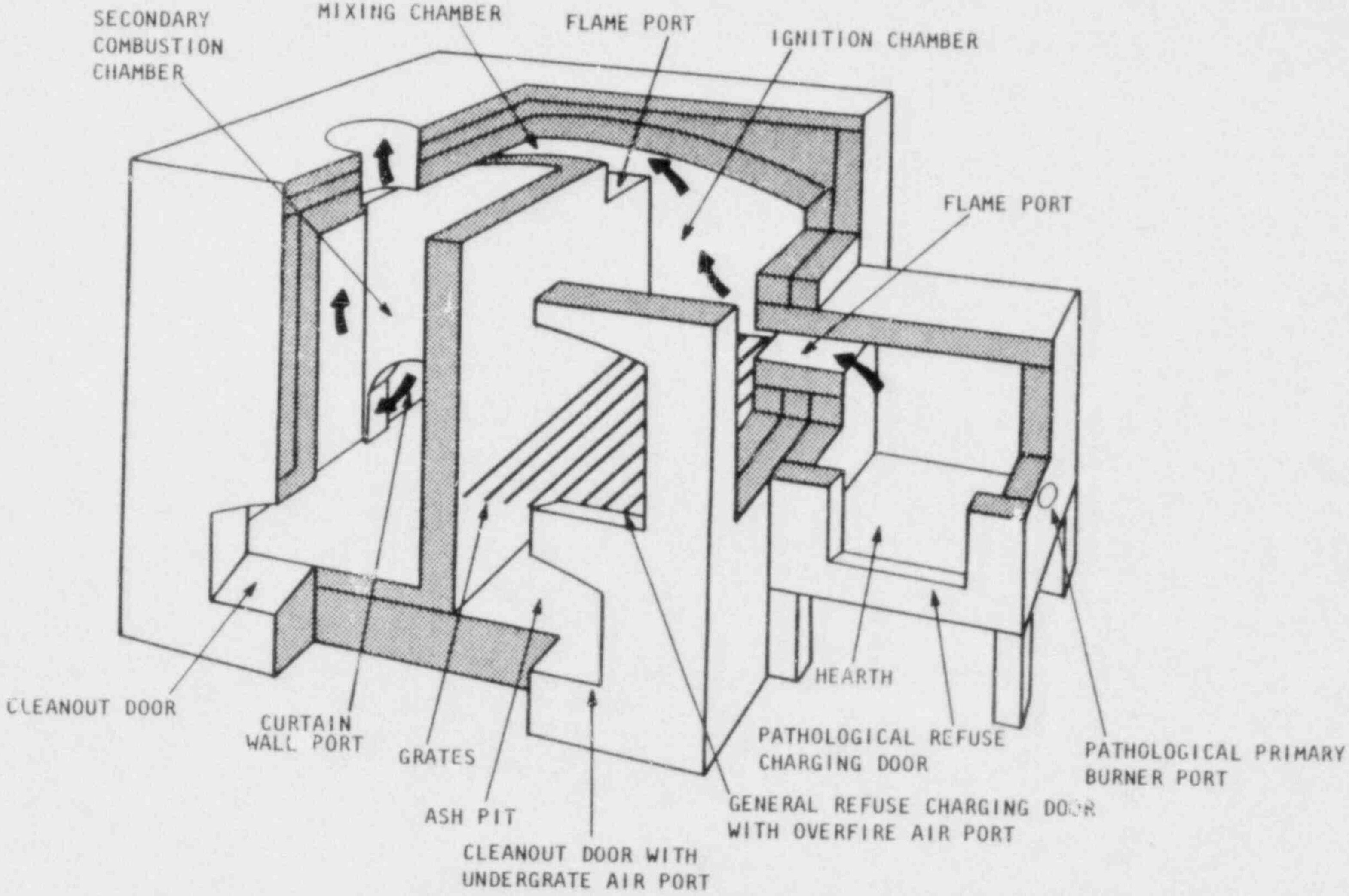
It is not uncommon for a single DF to be used for particulates and all radionuclides except iodine. This practice ignores differences in the volatility of radionuclides.

C.4.1 Pathological

Pathological incinerators are used by some insitutional waste generators for volume reduction of low-level wastes. These incinerators are typically multiple chamber incinerators with hot refractory hearths (Figure C-8) and operate at temperatures of 900 to 1000°C.(19,20) Off-gas treatment methods vary. Use of high efficiency air particulate filters (HEPA), vapor condensers, and wet scrubbers are common. Typical process rates range from 100-500 lbs/hr.

Pathological incinerators are used to volume reduce biowastes, scintillation fluids and other organic liquids, and trash.(21) Aqueous liquids can also be evaporated on the refractory hearth. Typical volume reduction factors are given in Table C-4. Institutional users of pathological incinerators generally control release of radioactivity to the atmosphere by controlling the rate of waste feed. Wastes incinerated are generally restricted to biowastes

FIGURE C-8



MULTIPLE-CHAMBER INCINERATOR
WITH A PATHOLOGICAL-WASTE RETORT

TABLE C-4

Volume Reduction Factors (VRF) for Pathological Incinerators

Waste Type	VRF
Trash, uncompacted	20
Full liquid scintillation vials	
Glass vials	4
Polyethylene or nylon vials	>100
Scintillation fluids and organic liquids	>100
Aqueous liquids	>100
Biowaste	15

Source: Reference 21.

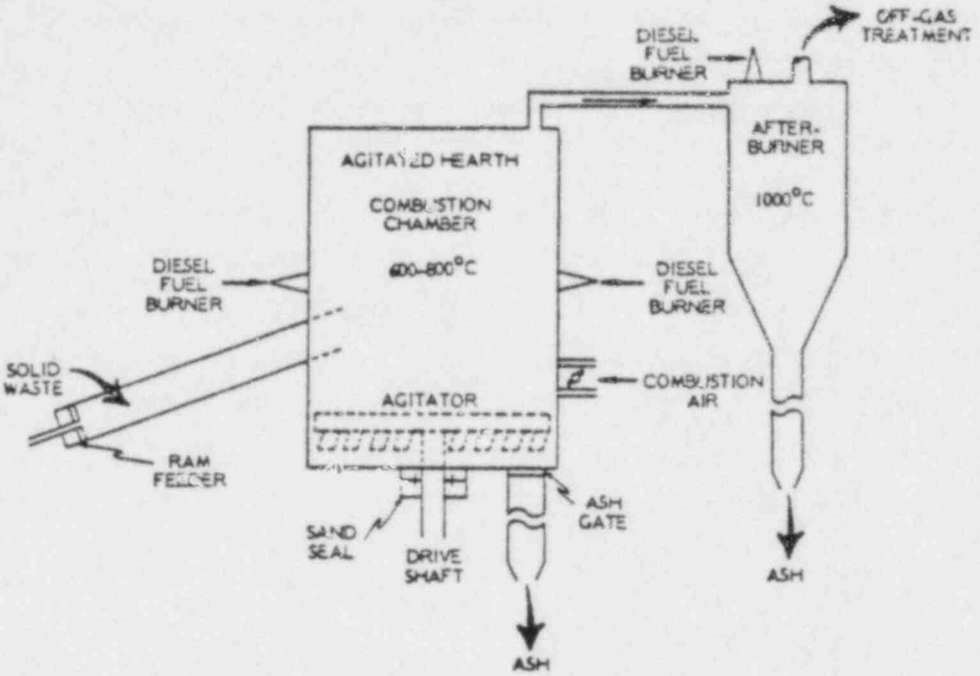
and scintillation fluids which contain small amounts of beta-emitting radionuclides (e.g., H-3, C-14).(21) Incineration of wastes containing I-125, I-131, Co-60, or Cs-137 is generally avoided.

Decontamination factors for pathological incinerators are estimated below. Data given in Table 129 of Reference 19 for incineration of human tissue and animal carcasses in eight pathological incinerators was used to calculate an average feed rate of 41.2 kg/hr, a particulate release rate of 6.54×10^{-5} kg/m^3 off-gas, and an off-gas flow rate of 1580 m^3/hr. These values indicate that about 2.51×10^{-3} kg of particulates were released per kilogram of waste burned. A particulate decontamination factor of about 400 is obtained from the inverse of this ratio.

It is assumed that the particulate DF is applicable to all radioactivity species except iodine (I-129), tritium (H-3), and carbon (C-14). On the basis of the DFs reported for a fluidized bed calciner/incinerator(16,22) (see Section C.5.1), a DF of 100 is assumed for iodine. Decontamination factors of 1.1 and 1.3 are arbitrarily assumed for tritium and carbon-14, respectively. These factors correspond to release of 90 percent of the tritium and 75 percent of the carbon-14 initially present in the waste.

C.4.2 Agitated Hearth Incinerator

A 4 kg/hr agitated hearth incinerator is being scaled up for processing low activity TRU trash at the Rocky Flats Arsenal.(24) The planned capacity of the Rocky Flats unit is 70 kg/hr. The incinerator (Figure C-9) is a multiple chamber, refractory lined, oil-fired incinerator. The primary combustion chamber operates at 600 to 800°C and is equipped with rotating arms which improve combustion by agitating the waste. The second chamber (afterburner) operates at 1000°C. Wet scrubbers are used to treat the off-gas. The unit is reported to have good tolerance for non-combustible materials other than glass.



AGITATED HEARTH INCINERATOR

DAMES & MOORE

FIGURE C-9

C.4.3 Controlled Air

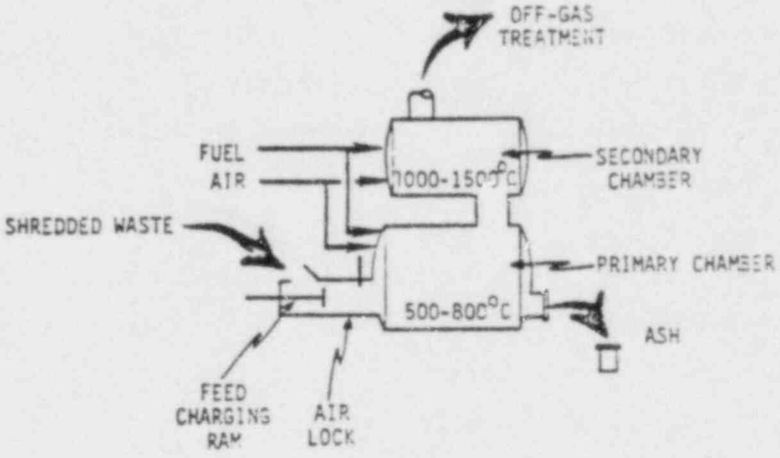
Several types of controlled air incinerators are either in use or under development at DOE facilities.(18) The demonstration unit at Los Alamos is designed to process TRU contaminated trash at 45 kg/hr and is fueled by natural gas. A volume reduction factor of greater than 40 has been attained for trash.

As shown in Figure C-10, pre-sorted shredded trash is charged to the primary chamber which operates at 800-1000°C. The primary chamber operates in a starved air condition. Unburned volatiles and particulates are swept into the upper secondary chamber which operates at about 1100°C with a slight excess of oxygen. The off-gas treatment system consists of a quench column, a venturi scrubber, packed columns, and HEPA filters.

C.4.4 Cyclone Drum

A cyclone drum incinerator developed at the Mound Facility has processed over nine tons of compacted (by a volume reduction factor of 2) TRU contaminated trash with an average volume reduction factor of 43 based on the volume of compacted trash.(15,18) The process rate for uncompacted trash is 27 kg/hr.(18)

The incinerator and off-gas treatment system for a cyclone drum incinerator are diagrammed in Figure C-11. The most interesting feature of this incinerator is that it can use either a permanent steel chamber or a standard 55 gallon drum as the process vessel. Combustion air is injected tangentially through an induction cover atop the drum, thus creating a downward spiral. The waste is ignited by a small quantity of liquid fuel and burns downward uniformly while combustion gases move upward inside the spiral. These gases exit the vessel at about 1300°C and pass through a spray scrubber/mist eliminator, a prefilter, a silver zeolite bed, and HEPA filters.



CONTROLLED AIR INCINERATION

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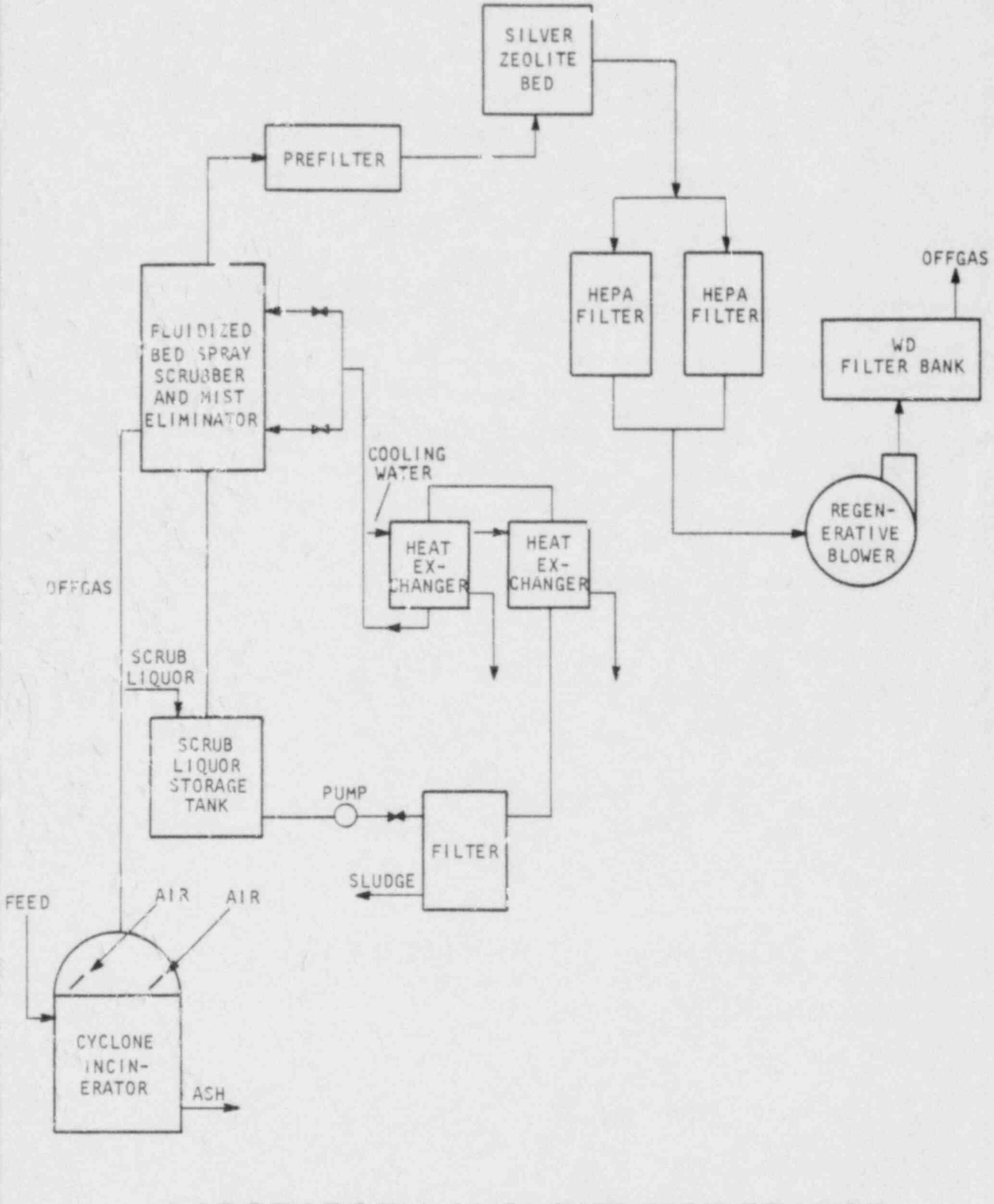
FIGURE C-10



LABORATORY CYCLONE INCINERATOR FLOW DIAGRAM

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FIGURE C-11

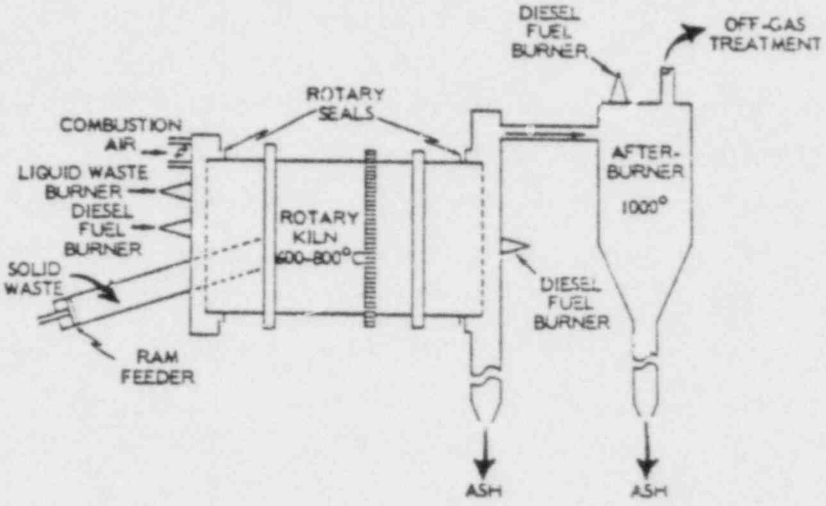
Studies of decontamination factors for incineration of trash spiked with Co-60, Cs-137 and I-131[15] are inconclusive. The available data suggests that as much as 3% of the Co-60, 12% of the Cs-137, and 13% of the I-131 may be released. The corresponding DFs would be 39, 7.6, 6.7. Further work is needed to better define decontamination factors.

The cyclone drum tolerates non-combustible waste well; for example, trash containing 6 wt% metal can be processed.[18] If problems with decontamination factors are resolved, this type of incinerator could be used to process trash at a central facility. Drums received from waste generators could be used as the process vessel, ash and non-combustible material emptied into larger disposal containers, and the drum reused.

C.4.5 Rotary Kiln

Rotary kiln incinerators have been used to process municipal solid waste and industrial solid, liquid, and gaseous wastes including chemical warfare agents.[24] The Department of Energy program to adapt rotary kilns for processing of TRU wastes is now in the production stage.[18] The production unit being installed at Rocky Flats is designed to process trash, organic liquids and ion exchange resins at a nominal rate of 40 kg/hr.

As shown in Figure C-12, the rotary refractory-lined kiln is fired by two axial diesel fuel burners and operates at about 800°C. Liquid wastes are injected through a separate burner while solid wastes are charged with a ram feeder. The afterburner operates at about 1000°C. Off-gases are treated by two venturi scrubbers and four stage HEPA filtration. Ash is continuously discharged from the kiln.



ROTARY KILN INCINERATOR

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FIGURE C-12

C.4.6 Pyrolysis - Controlled Air

The pyrolysis - controlled air incinerator, also known as the electric controlled air incinerator, is a small (5 kg/hr) unit being developed for use at the Savannah River Plant.(18) It is designed to process solid waste containing up to 1 nCi/g of transuranics. It may be useful for processing high activity commerical wastes such as ion exchange resins generated during decontamination of LWR primary cooling systems.

Oxygen deficient conditions are maintained in the first stage of the unit so that pyrolysis rather than combustion occurs. Pyrolysis gases are fed to a vertical labrynith afterburner. The primary chamber is lined with silicon carbide and operates at 700 to 900°C. The afterburner is constructed of cast alumina tubes and operates at 1000°C. Both chambers are electrically heated. The independent scrubber loops, a venturi quench, a fibrous-bed scrubber and a padded bed contactor, are used to minimize buildup of TRU salts.

C.4.7 Acid Digestion

Several acid digestion systems have been developed at Hanford Engineering Development Laboratory (HEDL) for volume reduction and plutonium recovery from TRU waste.(6,7,8,18) The high rate digester (Figure C-13) is designed for a throughput of 10 kg/hr. Pre-sorted, shredded waste is charged by a ram and rapidly mixed with concentrated sulfuric and nitric acids at 250°C. The interior surfaces are lined with Glasteel.* The hot sulfuric acid carbonizes organic materials which are then oxidized by nitric acid. Residues are removed from an acid slurry side-stream by centrifuges or by evaporation of the sulfuric acid. A volume reduction factor of about 23 is typical for trash. The off-gas treatment system consists of two scrubbers, each

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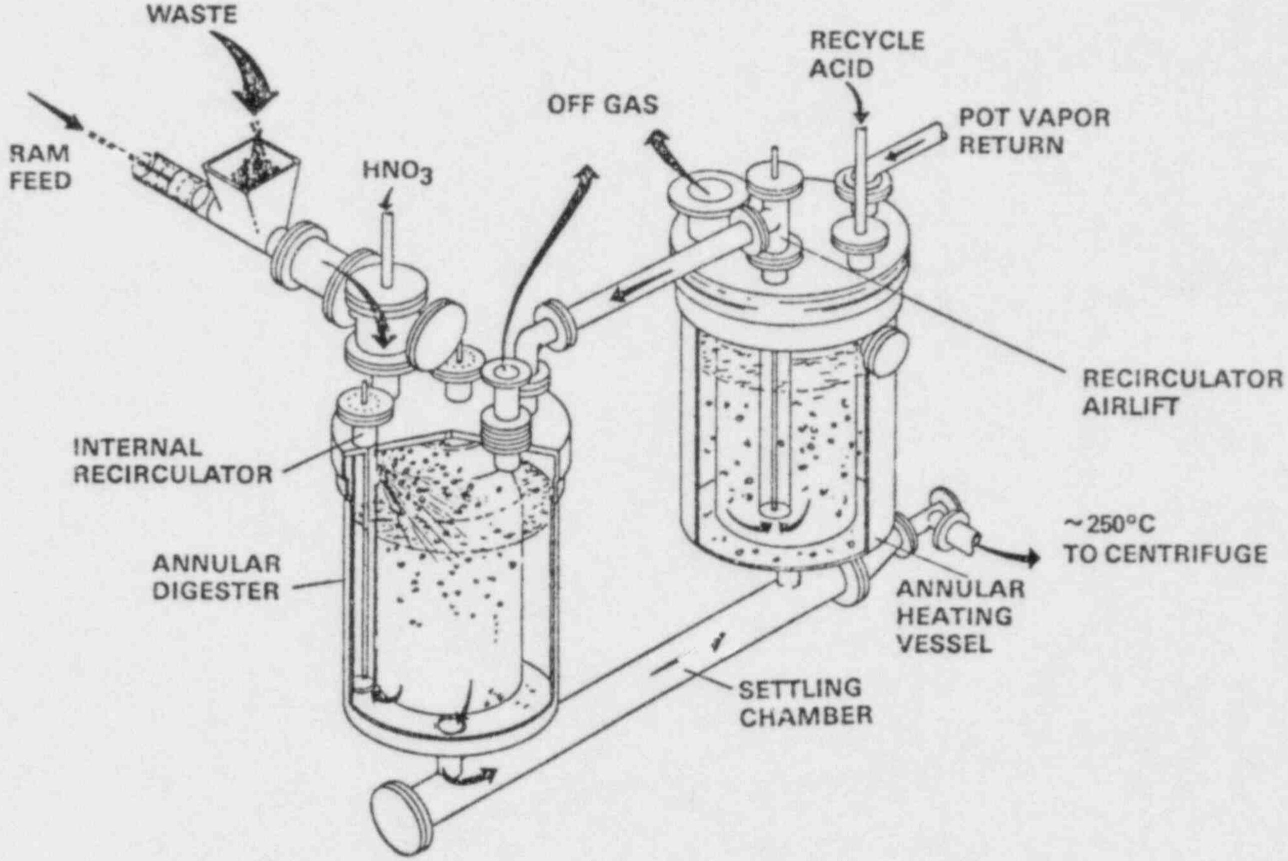
* Registered trademark of the Pfaudler Co., Elyria, Ohio.



HIGH RATE ACID DIGESTION SYSTEM

FIGURE C-13

DAMES & MOORE

followed by a demister and HEPA filters, and an acid fractionator with a demister and HEPA filters. The fractionator recovers nitric acid for reuse.

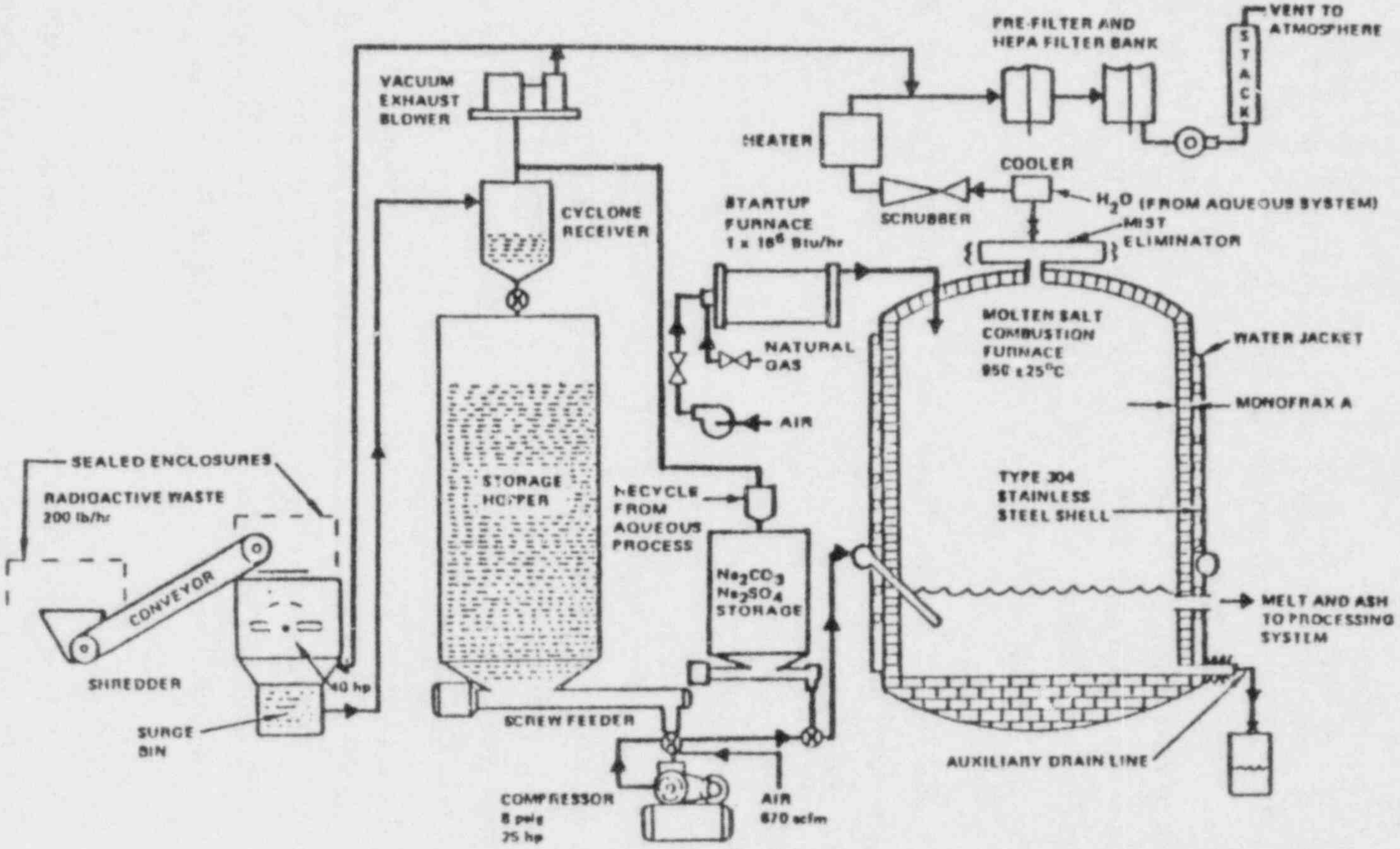
Several organic liquids have been processed with varying degrees of success.(25) These include pump oil, tri-n-butyl phosphate (TBP), normal paraffin hydrocarbons (NPH), carbon tetrachloride (CCl_4), trichloroethane, toluene, hexone, and polychlorobiphenyl (PCB). Toluene was digested with sulfuric acid alone to avoid production of trinitrotoluene (TNT). Digestion of NPH, CCl_4, trichloroethane, and PCB was less than 50 percent complete.

The residues obtained for the digestion may contain residual acids. HEDL packs residues in special cannisters. A standard 55 gallon drum holds nine of these cannisters.

C.4.8 Molten Salt Combustion

Use of molten salt for combustion of low-level radioactive wastes is being developed by the Atomics International Division of Rockwell International. Other companies and laboratories are developing similar systems for the combustion of municipal wastes and hazardous chemicals. The Rockwell process (Figure C-14) uses a molten pool of sodium salts, primarily sodium carbonate, at a temperature of 800 to 1000°C. Combustible wastes are shredded and carried into the molten salt via a stream of compressed air. Most of the combustion products are absorbed by the molten sodium carbonate and the remaining gas processed through a venturi scrubber and HEPA filters before release.

The molten salt process will tolerate up to 20 weight percent noncombustibles in the salt pool. The pool must be replaced when this limit is reached. Pool replacement can be avoided by withdrawing a small stream of the melt, quenching with an aqueous solution, and



MOLTEN SALT COMBUSTION SYSTEM

DAMES & MOORE

FIGURE C-14

filtering to remove noncombustibles. The sodium salts are then returned to the process vessel.

The off-gas is reported to contain practically no radioactivity and undesirable gases but DFs are not given.(1,24) A VRF of 46 is reported for an unspecified combustible waste stream with the molten salt being dumped directly into disposal containers.(24) The molten salt can also be glassified by raising the temperature and adding borosilicate glass or other suitable materials before dumping the pool. Glassification is estimated to reduce the VRF to 10 to 20.

C.5 DUAL FUNCTION SYSTEMS

Four additional types of volume reduction systems are designed to function in two modes. Well-established fluidized bed technology has been adapted to calcine liquid waste and incinerate wet and dry solid wastes generated by LWRs. Calcination is a high-temperature process where liquid wastes are evaporated and thermally decomposed to form stable, nonfused compounds such as oxides. Less versatile systems are available for TRU waste.(18,24) Bitumenization systems are available which evaporate virtually all water from liquid and wet solid wastes and which can also handle several types of dry solid waste. The inert carrier radwaste process (ICRP) also evaporates liquids and wet solid waste and solidifies the residues and other finely divided dry solids in epoxy. The Electromelt process uses molten glass to evaporate waste liquids and incinerate solids, solidifying the residues in the glass.

The properties of the solidification agents and of the final waste forms are discussed in Appendix D. This section discusses the volume reduction aspects of these systems.

C.5.1 Fluidized Bed Systems

Fluidized bed systems use a heated bed of a inert granular material as the heat exchange medium. The bed is suspended (fluidized) by a stream of hot air. Aqueous wastes are sprayed into the bed, flash evaporate on contact, and thermally decompose leaving behind dry salts as a coating on the bed particles. Organic liquids, shredded trash, and wet solid waste are also incinerated in the bed. Most of the ash formed exits with the off-gases.

Fluidized bed volume reduction systems designed for use in LWRs are available from Aerojet Energy Conversion Company(16,22) and Newport News Industrial Corporation.(17) The Aerojet system uses separate process vessels for incineration and calcination while the Newport News system performs both functions in a single vessel. These two systems are described below.

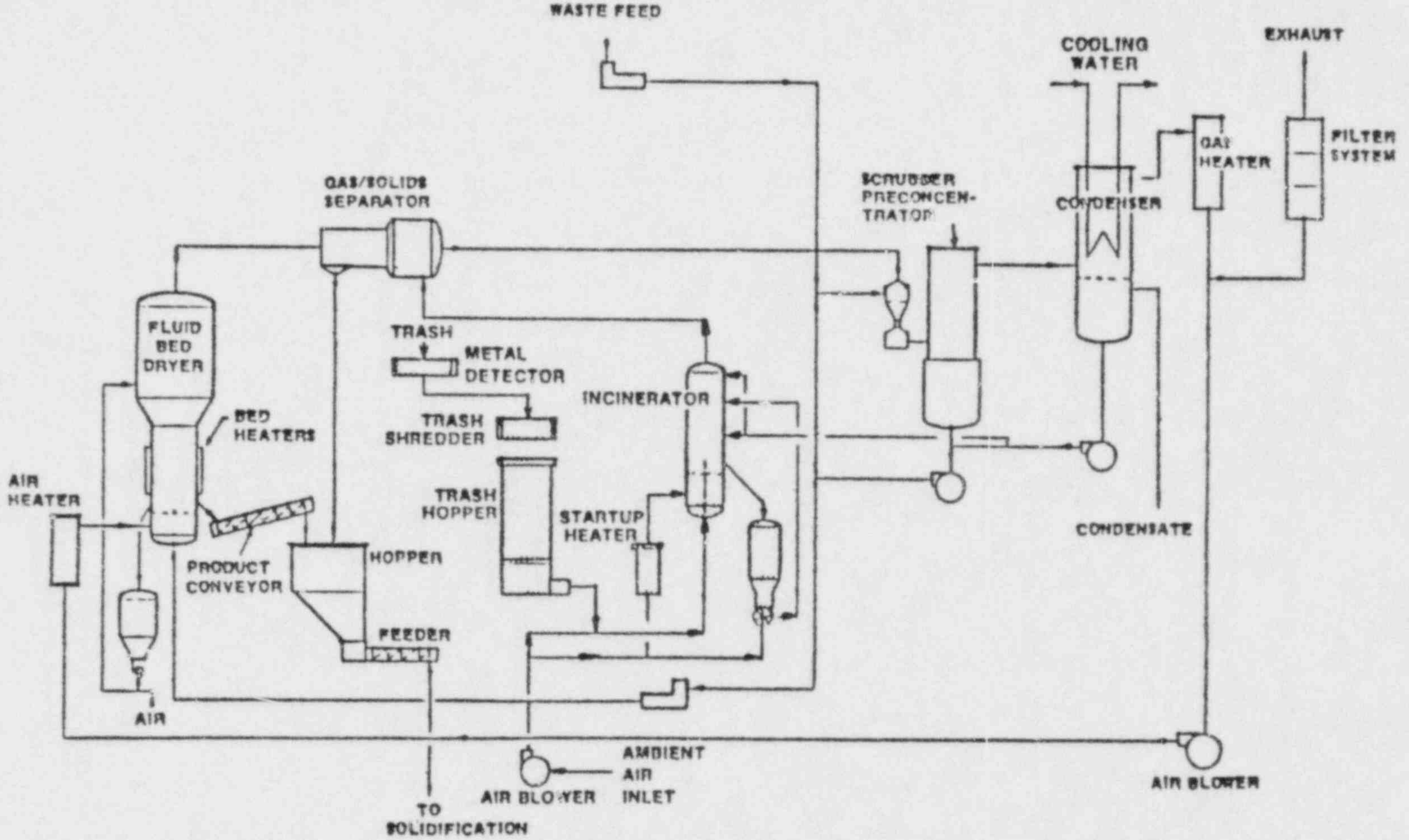
Aerojet System

The calciner and incinerator vessels of the Aerojet system can be operated independently or simultaneously. Simultaneous operation is the preferred mode since the system is designed to use incoming liquid waste bound for the calciner to scrub and condense incinerator off-gases. This arrangement also preconcentrates the liquid waste. The calciner uses electrically heated air to fluidize the bed. Supplemental electric heaters are attached directly to the vessel to maintain a bed temperature of about 480°C. Most of the residual solids accumulate in the bed material and are removed via a conveyor system. Any residues or bed materials in the off-gas are collected in a gas/solids separator (Figure C-15). The off-gas then passes through a venturi scrubber, a preconcentrator, a condenser and HEPA filters before release.

The incinerator vessel is equipped with electric startup heaters. After ignition, combustion of the waste materials maintains the bed temperature at 800-1000°C. Ashes are collected in the gas/solid separator and off-gases treated in the common system. In the event that the calciner is not in operation during incineration, dilute liquid waste is circulated through the scurbber, preconcentrator, and condenser and returned to a holding tank. The returned waste is more concentrated than the original waste.

The Aerojet system can process trash, aqueous and organic liquids, and diatomaceous and Solka-Floc filter sludges. Aerojet does not recommend incineration of organic ion exchange resins or filter sludges containing powdered resins. Typical volume reduction factors are given in Table C-5.

FIGURE C-15



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AEROJET FLUIDIZED BED CALCINER/ INCINERATOR

TABLE C-5

Volume Reduction Factors (VRF) for the Aerojet Fluidized Bed Dryer/Incinerator

Waste Type	VRF
Resins	Not Recommended
Filter Sludge	
50 weight percent solids	5
Evaporator Bottoms	
12.5 weight percent boric acid	9.3
25 weight percent sodium sulfate	4.6
Crystalizer Bottoms	
50 weight percent boric acid	2.9
50 weight percent sodium sulfate	2.9
Combustible Trash	
Uncompacted	80

Source: Reference 1.

Decontamination factors of 10^4 for iodine and 10^6 for particulates are estimated.(16) The factors are applicable for both calcination and incineration.

Newport News System

The Newport News fluidized bed system may have one or two process vessels depending on the required process rate. Each vessel is capable of both calcination and incineration. The bed material is heated initially with a mixture of air and burning fuel oil. The calcination temperature of about 400°C and incineration temperature of 800-1000°C are maintained thereafter by burning fuel oil as needed.

The Newport News system is designed to agitate the bed materials to prevent buildup of salt residues during calcination. These residues and ashes from incineration are collected by a dry cyclone (Figure C-16). Off-gases pass through a quench tank, a venturi scrubber, a wet cyclone, a condenser/mist eliminator, HEPA filters and iodine absorbers before release. Decontamination factors of 4×10^4 to 7×10^6 for particulates and 1×10^4 to 1×10^5 for iodine have been reported.(17)

Volume reduction factors for this system are given in Table C-6. Differences in VRFs between the two fluidized bed systems for liquid wastes are probably due to the method of extrapolating available data to the desired waste concentrations rather than to real differences in equipment capabilities. The Newport News system can incinerate ion exchange resins.

C.5.2 Bitumenization

The bitumen solidification systems marketed by Werner-Pfleiderer Corporation(26) (WPC) and by Associated Technologies, Incorporated (27) (ATI) evaporate liquid and wet solid wastes while simultaneously

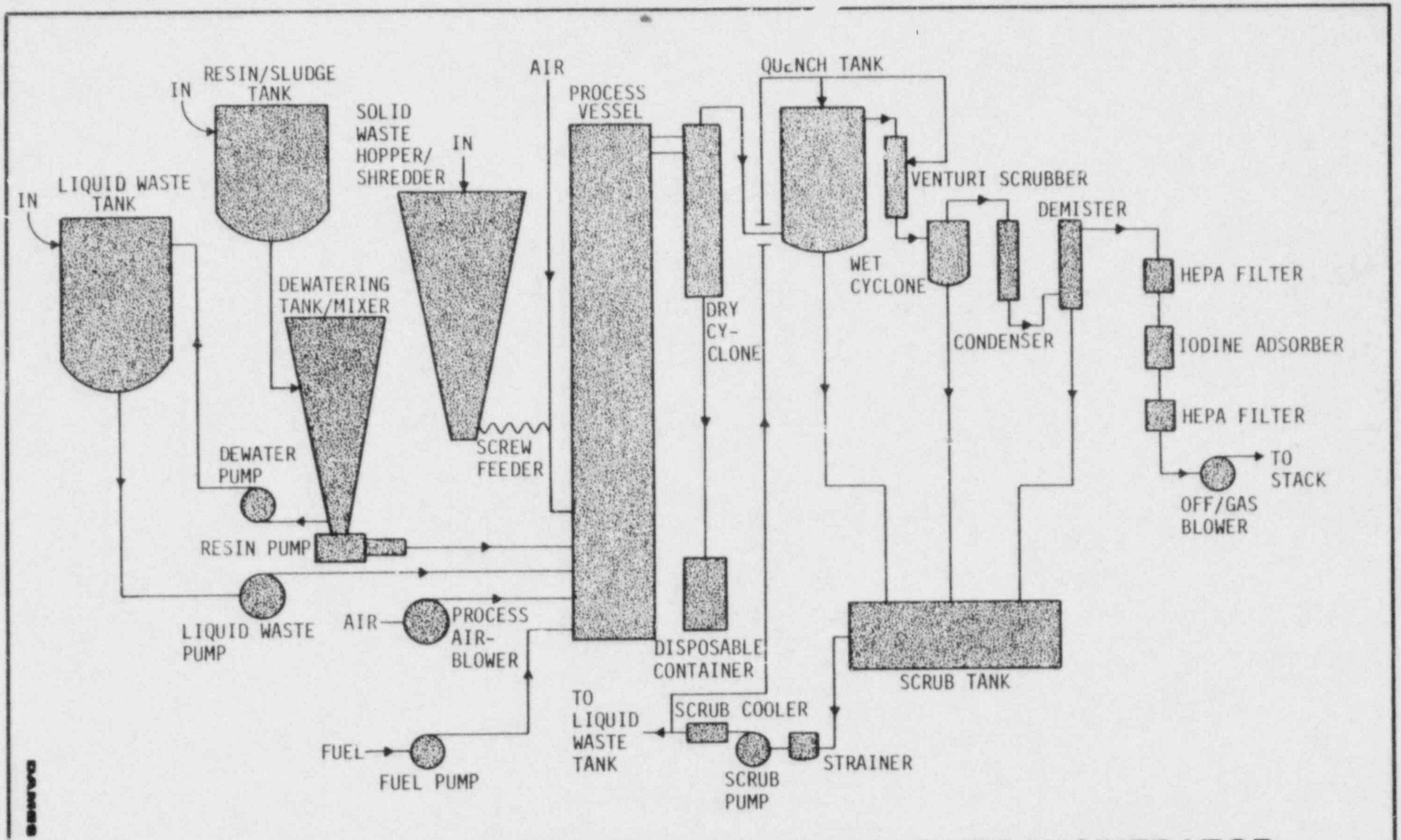

FIGURE C-16

FLOW DIAGRAM OF NEWPORT NEWS CALCINER/INCINERATOR

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TABLE C-6

Volume Reduction Factors (VRF) for the Newport News Fluidized Bed Dryer/Incinerator

Waste Type	VRF
Resins	
33 weight percent solids	18
Filter Sludge	
50 weight percent solids	5
Evaporator Bottoms	
12.5 weight percent boric acid	8
25 weight percent sodium sulfate	6.4
Crystalizer Bottoms	
50 weight percent boric acid	2
50 weight percent sodium sulfate	3.2
Combustible Trash	
Uncompacted	80

Source: Reference 1.

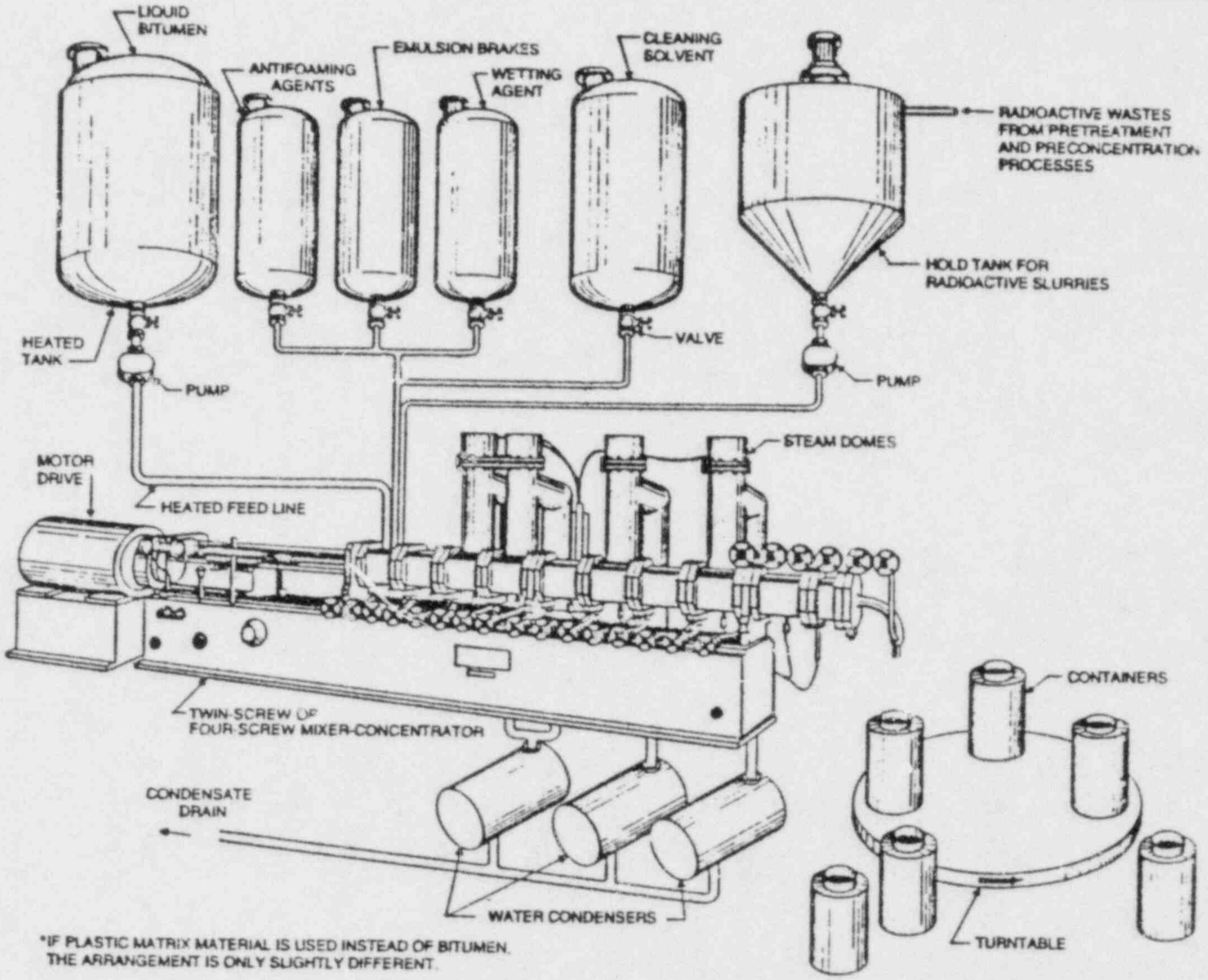
solidifying the residues in bitumen.(27) Both systems can also process finely divided dry solids but neither perform incineration. The properties of the product waste forms are discussed in Appendix D. This describes the process equipment and overall changes in waste volumes.

Extruder/Evaporator

The WPC system shown in Figure C-17 uses a heated screw extruder to mix liquid and solid wastes with hot bitumen. Temperatures in the extruder increase from 70 to 175°C moving down the extruder and are sufficient to evaporate greater than 99 percent of the water from the waste. Steam is used to preheat the bitumen and to heat the screw extruder. Waste, bitumen, and chemicals to improve mixing enter the low temperature end of the extruder. Evaporated water is collected by steam domes and routed to the effluent treatment system. The overall volume reduction factors for wastes processed by the WPC system are presented in Table C-7. These factors are the ratios of initial waste volume to the volume of the final bitumen product. Factors less than one indicate a net increase in volume.

Wiped Film System

The heart of the ATI system shown in Figure C-18 is a Luwa wiped film evaporator (see Section C.3.4). The system can process the liquid and wet solid wastes listed in Table C-7. Figure C-19 shows the evaporator in more detail. Waste and bitumen enter at the top of the unit so that evaporation and encapsulation occurs simultaneously. The product is discharged directly into disposal containers from the bottom of the evaporator. Steam is used as the heat exchange medium. Volume reduction factors are proprietary but are expected to be similar to those given in Table C-7 for the WPC system. Decontamination factors for the ATI system are proprietary.



*IF PLASTIC MATRIX MATERIAL IS USED INSTEAD OF BITUMEN. THE ARRANGEMENT IS ONLY SLIGHTLY DIFFERENT.

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EXTUDER/EVAPORATOR BITUMERIZATION SYSTEM

FIGURE C-17

TABLE C-7

Volume Reduction Factors for the WPC Extruder/Evaporator

Waste (pH = 7)	Volume Reduction Factor Supplied by Vendor	 BNL Data[a]
1) Resin Slurry		
a) 33% by weight	1.56	2.1
b) 50% by weight	1.03	--
2) Aqueous Sodium Sulfate		
a) 23% by weight + 2% other solids	--	2.6
b) 25% by weight	2.1	--
c) 50% by weight	0.88	--
3) Aqueous Boric Acid (12% by weight)	4.7	7.7
4) Filter Sludge		
a) 40% by weight powdered resin + 10% other solids	--	2.5
b) 50% by weight unspecified sludge	1.03	--
5) Dry Salt	0.42	--
6) Incinerator Ash	0.43	--

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[a] For a product containing 50 weight percent waste solids.

Source: Vendor supplied data taken from Reference 1.
Brookhaven National Laboratory data taken from Reference 28.

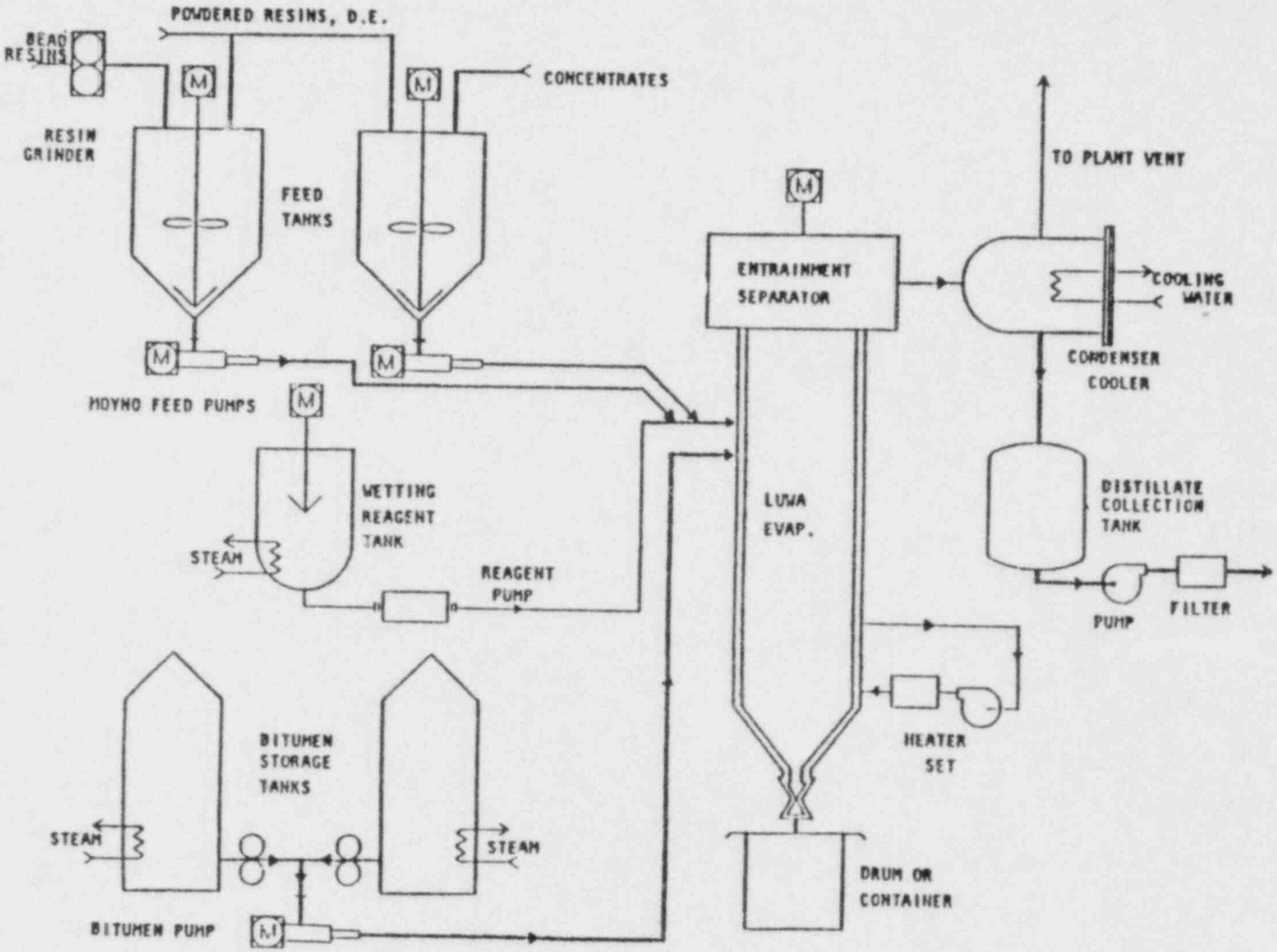
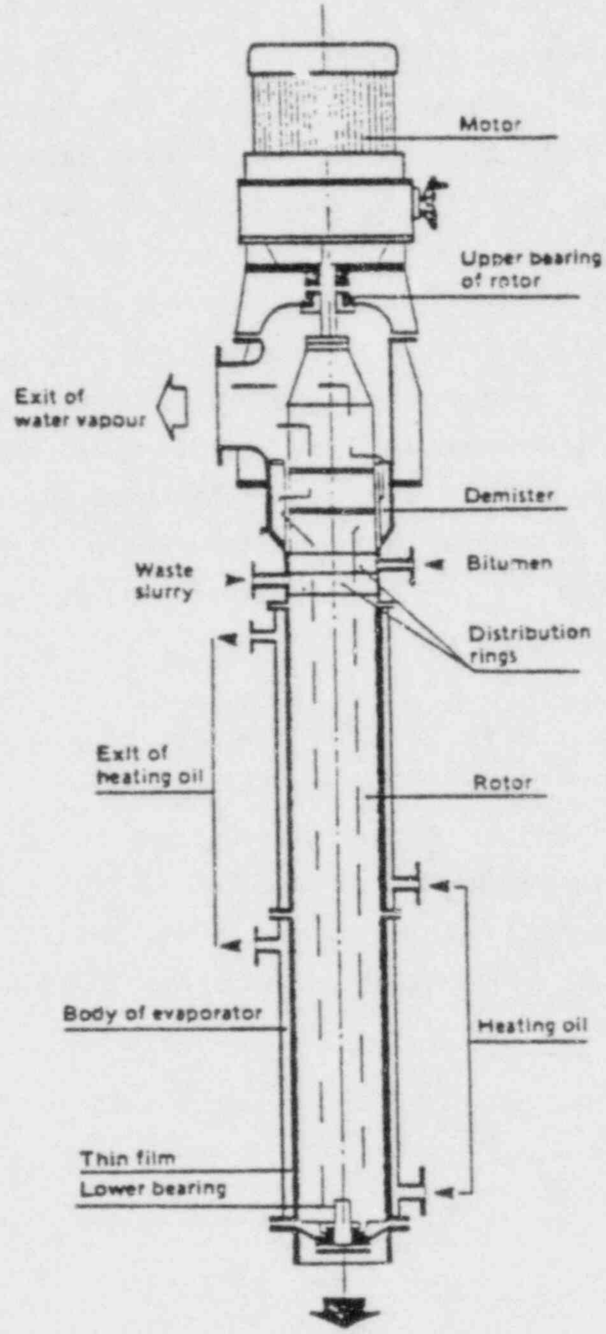


FIGURE C-18

WIPED FILM EVAPORATOR BITUMERIZATION SYSTEM



LUWA WIPED FILM EVAPORATOR

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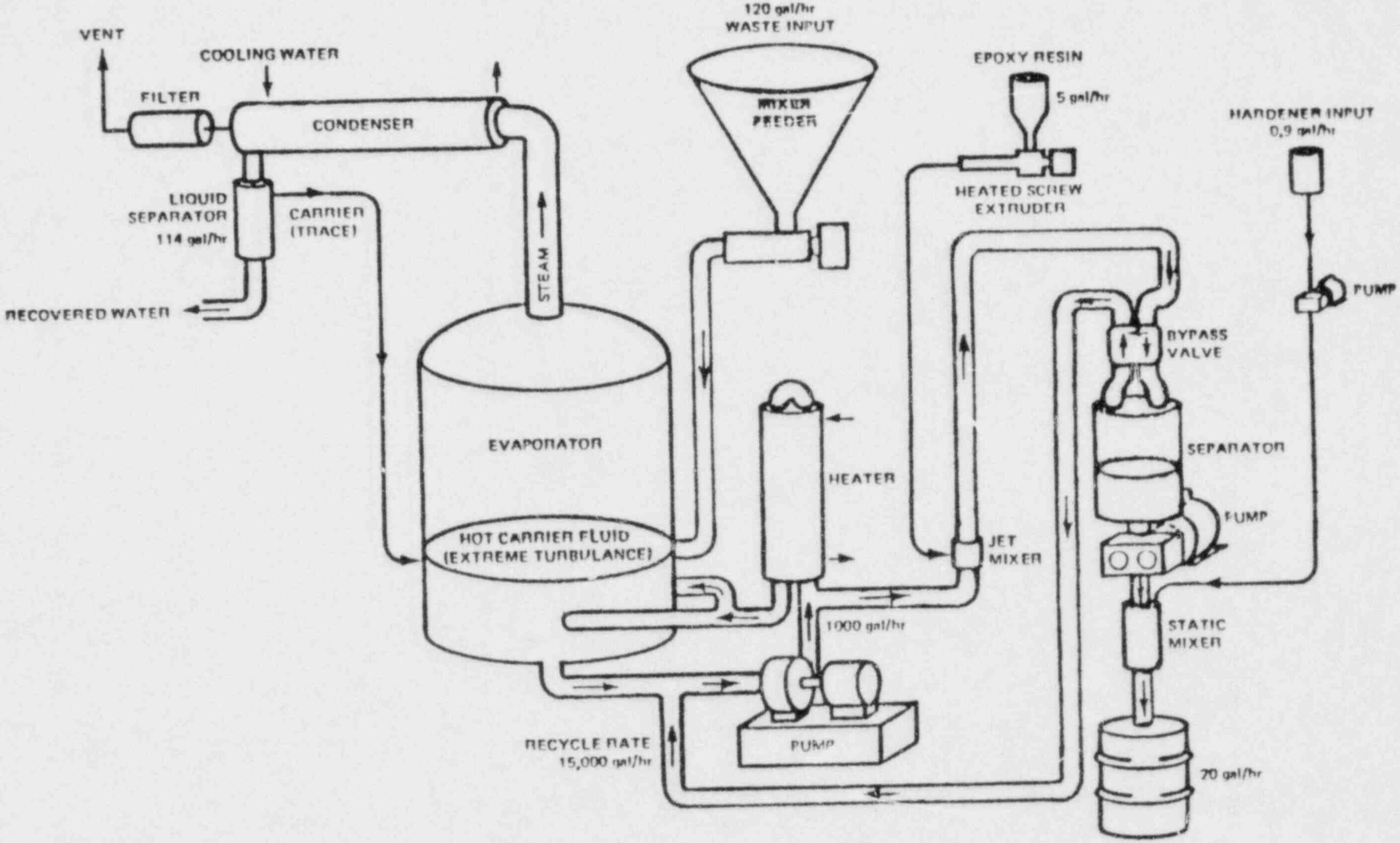
FIGURE C-19

C.5.3 Inert Carrier Radwaste Process

The inert carrier radwaste process is being developed by United Technologies (UT) to process LWR liquid and wet solid wastes at a nominal rate of 120 gal/hr.(29,30) The system shown in Figure C-20 uses an inert silicone oil as the heat exchange medium. The oil is heated to about 300°F and circulated at a high velocity. The water in liquid and wet solid wastes flash-evaporates on contact with the turbulent fluid. A side stream of the residue/fluid slurry is mixed with epoxy resin and sent to a solids separator. The resin coated residues are mixed with a hardener to initiate curing of the resin and discharged to a disposal container. The properties of the fluid waste form are discussed in Appendix D.

Overall volume reduction factors for the ICRP system are somewhat higher than those for the bitumen systems. Reported VRFs for ion exchange resins, 25 weight percent aqueous sodium sulfate, and 12 weight percent of aqueous boric acid are 1.2, 4.3, and 8.3, respectively. The same amount of water is removed by the ICRP and bitumen systems; however, less epoxy resin is required to obtain a satisfactory final product.



INERT CARRIER RADWASTE PROCESS

FIGURE C-20

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APPENDIX D

WASTE FORM AND WASTE BINDER CHARACTERISTICS

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APPENDIX D : WASTE FORM AND WASTE BINDER CHARACTERISTICS

This appendix contains a summary of the available information on low-level radioactive waste (LLW) form (waste and containers), waste binders (solidification agents utilized to change and/or improve various characteristics of LLW), and the characteristics of these wastes after solidification.

D.1 INTRODUCTION

The radioactivity contained in LLW can be mobilized in a variety of ways. The radioactivity contained in wet and dry solid wastes can be dispersed by wind and fire during transportation accidents and released to groundwater by leaching of the wastes after disposal. The mobility of the radioactive species can be significantly reduced in many cases by mixing the waste with a solidification agent which physically and/or chemically binds these species within a free-standing monolithic waste form. Commercial LLW disposal sites currently require solidification of liquid wastes from light water reactors (LWR's) and will soon require solidification of spent ion exchange resins and filter media.(1,2)*

This appendix describes solidification agents which are now in use or being actively developed for routine use and discusses the properties of the solidified waste forms. The characteristics of solidified and unsolidified wastes are also discussed. Included in the discussion are wastes such as resins, sludges, trash, and organic liquids.

Three general types of solidification agents (binders) are considered in this appendix for use with LLW: (1) Portland cements, (2) bitumen, and (3) synthetic organic polymers. Each of these three general types

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(*) An option is provided at one disposal facility (Barnwell) to package ion exchange resins and filter media within high integrity containers.

of binders can be further subdivided. For example, there are five major types of Portland cement, each of which may be used with additives, enumerable emulsified and molten bitumens, and four types of synthetic organic polymers now being used or actively developed. These waste binders and the processes used to incorporate LLW within them are described in Section D.2.

The remainder of this appendix discusses available information on waste form characteristics which allow assessment of the ability of a given waste-binder combination to immobilize radioactivity. These characteristics are: free-standing water which is discussed in Section D.3, leachability which is discussed in Section D.4, mechanical properties which are discussed in Section D.5, thermal properties which are discussed in Section D.6, corrosion of mild steel which is discussed in Section D.7, radiation effects which are discussed in Section D.8, and biological and chemical degradation which is discussed in Section D.9. Much of the data presented in these sections is taken from a series of reports by the Nuclear Waste Management Research Group of Brookhaven National Laboratory entitled "Properties of Radioactive Wastes and Waste Containers."(3-20)

D.2 Solidification Agents

Several solidification agents (binders) may be used to immobilize the radioactivity contained in LLW and/or to improve the waste form stability. Among these solidification agents are Portland cement, bitumen, and synthetic organic polymers which can physically encapsulate or entrap waste liquids and solids. In addition, cement has the ability to chemically bind radioactive species dissolved in liquids and wet solid wastes.

Of the available binders, only Portland cement, vinyl ester-styrene, and urea-formaldehyde are routinely used for solidification of LLW in the United States. Bitumen is widely used in Europe. Vinyl ester-styrene has been used on a limited scale and is scheduled for use during the decontamination of Dresden Unit 1. Polyester and epoxy are still in the development and testing stages for LLW application.

The chemical reactions which occur during solidification of cement and synthetic polymers are exothermic (generate heat). Bitumen must be heated to obtain a satisfactory waste form. These and other properties of the solidification agents and processes are described in detail in the remainder of this section.

D.2.1 Portland Cement

Portland cement, a hydraulic cement, is the most commonly manufactured hydraulic cement and is frequently used for solidification of radioactive waste. Hydraulic cements react with water which is either in the waste or added to it, to form hydrated silicate and aluminate compounds which ultimately solidify to produce a monolithic solid.

Portland cements are complex mixtures of compounds formed from simple oxides, predominately silica (SiO_2), lime (CaO), and alumina (Al_2O_3) with lesser amounts of magnesia (MgO), ferric oxide (Fe_2O_3), and sulfur trioxide (SO_3). The major compounds formed from these oxides are tricalcium silicate ($3CaO.SiO_2$), dicalcium silicate ($2CaO.SiO_2$), tricalcium aluminate ($3CaO.Al_2O_3$), and tetracalcium aluminoferrite ($4CaO.Al_2O_3 Fe_2O_3$). There are five major types of Portland cement which are made by varying the relative amounts of these four compounds. Their composition and properties are listed in Table D-1.

Of the 5 major types of Portland cements available, Portland Types I, II, and III cements are used most frequently for solidification of radioactive wastes. Type I is a common cement used for general construction applications and is used as a solidification agent where it is not subject to attack by sulfates and where the heat released during curing is acceptable. Type II has a lower heat of hydration and better sulfate resistance than Type I. Type III gives high early (within one to three days) mechanical strength. Type IV is used for special applications requiring a slow rate of hydration with minimum heat generation. Type V is used when severe sulfate attack is expected.

The processes involved in the hydration, setting and curing of Portland cement are not completely understood.(21) On mixing with water the four compounds listed in Table D-1 begin to hydrate, forming a colloidal-disperse "sol". During this phase, hydration of tricalcium silicate and tricalcium aluminate predominates. The "sol" coagulates into a "gel" which subsequently precipitates. Setting of the cement begins with gelation and ends when precipitation is complete. The strength of cement during setting is due to the presence of tricalcium silicate and tricalcium aluminate. Once setting is complete, the cement begins to cure (dry), and to produce crystalline slabs and needles. Dicalcium and tricalcium silicates are responsible for the ultimate strength of the cement. A minimum water/cement weight ratio of 0.25 is required to obtain a free standing product.

Radioactive wastes (liquids, slurries, dewatered resins and sludges,

TABLE D-1 . Composition and Properties of Portland Cements
(percent by weight)

Type	Properties	Tricalcium Silicate	Dicalcium Silicate	Tricalcium Aluminate	Tetracalcium Aluminoferrite
I	Normal, general purpose	45	27	11	8
II	Low heat of hydration improved sulfate resistance	44	31	7	13
III	High early strength	53	19	10	7
IV	Low heat of hydration	20	52	6	14
V	High sulfate resistance	38	43	4	8

dry solids) can be mixed with cement either in the waste container or in-line and poured into the container. Addition of water may be necessary for dewatered and dry wastes and pretreatment (pH adjustment) may be necessary for acidic liquids.

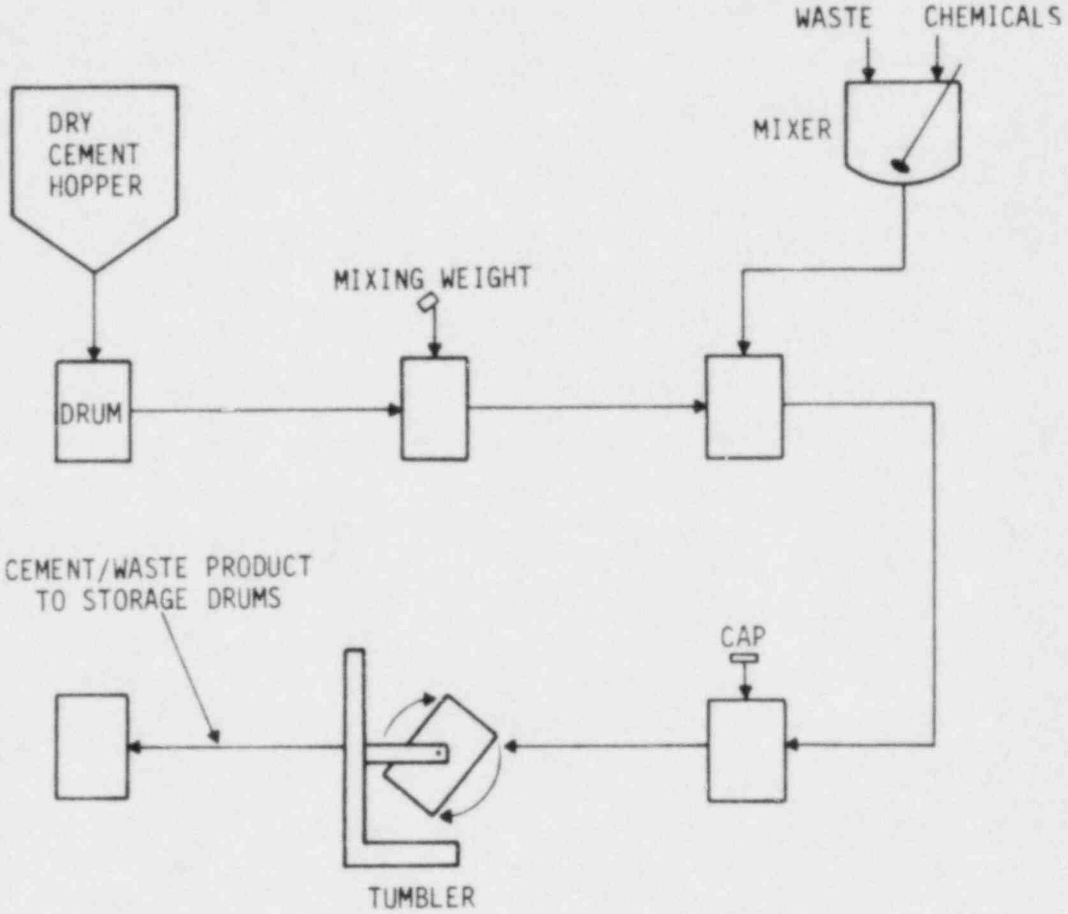
Gravity mixing, tumbling/rolling, or external agitation are employed when using the container as the mixing vessel. In the gravity mixing procedure, liquid waste is added directly to a pre-mixed blend of cement and a light-weight absorbent, such as vermiculite, which absorbs the liquid and disperses it throughout the mixture.(22) In the tumbling/rolling method, which is shown in Figure D-1, a mixing weight is added to the drum which is capped and transferred to a tumbling or rolling station where its contents are mixed. In the external agitation process, a mixing blade lowered into the drum during or after waste addition blends the waste with cement.

In-line mixing can be performed on either a batch or continuous basis. In this process (Figure D-2), cement and slurry containing appropriate amounts of liquid and solid wastes are fed into a mixer (usually a powered screw dynamic mixer) at predetermined rates, and the mixture is discharged directly into the shipping container.

Commercial cement solidification systems frequently include equipment to control vapors and fumes generated by heat released during solidification and by chemical reactions between the waste and cement.

D.2.2 Cements and Additions

The properties of Portland cements listed in Table D-1 can be modified by the use of additives. These additives may improve waste form homogeneity, speed solidification, increase mechanical strength, and/or reduce leachability. Materials tested as additives include sodium silicate, lime, clays, zeolites, and styrene.



IN-DRUM CEMENT INCORPORATION PROCESS

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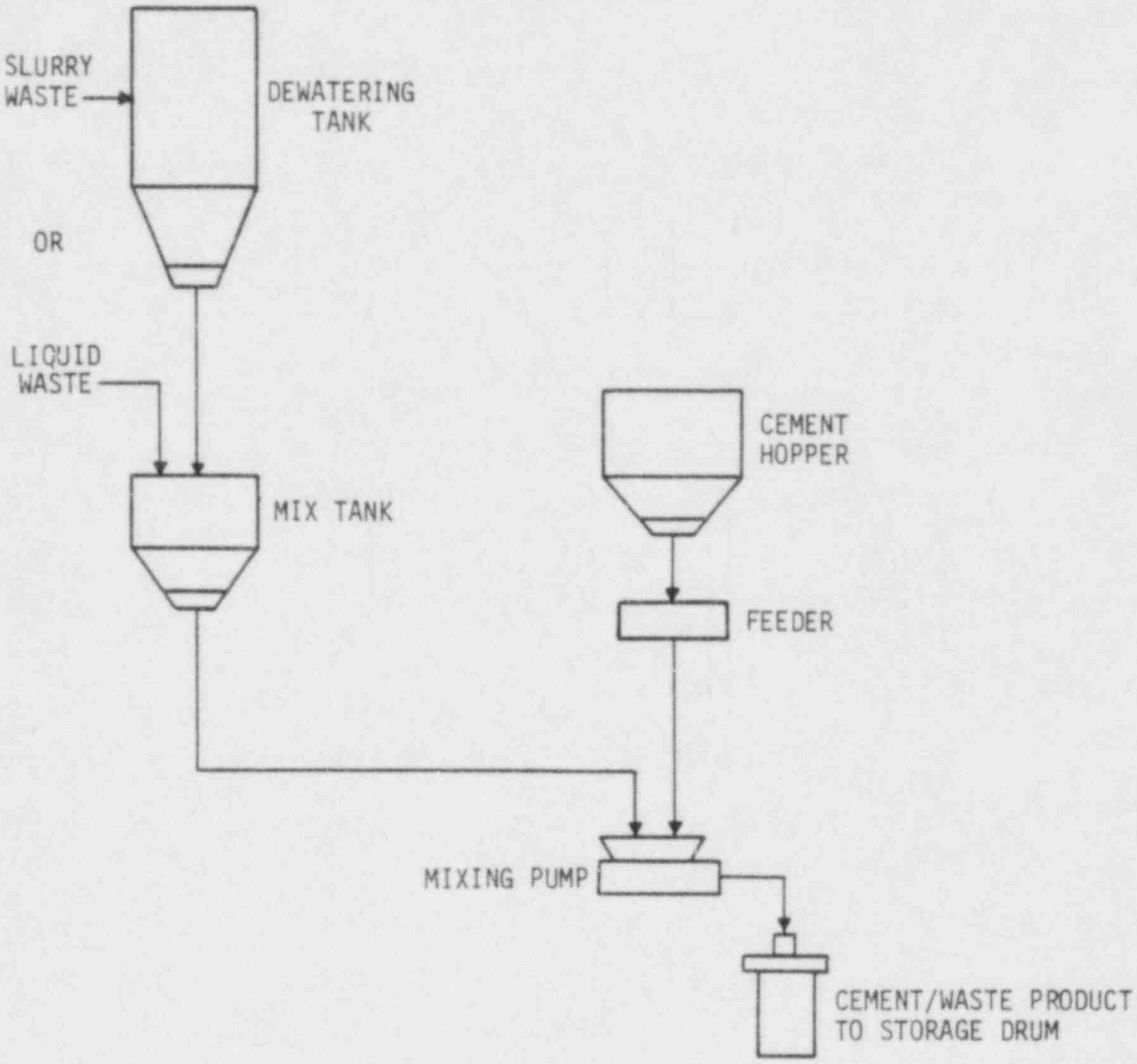
FIGURE D-1



EXTERNAL CEMENT INCORPORATION PROCESS

FIGURE D-2

Sodium silicate is sometimes added immediately after mixing the cement with the waste. The sodium silicate reacts to form gelatinous precipitates with multi-valent metal ions. Precipitation is rapid and accelerates gelation of the mixture but care must be taken not to agitate the mixture to prevent breakup of the gel. Addition of sodium silicate is reported to aid in the solidification of boric acid wastes and to increase the waste/binder ratio.(17) Several companies market cement solidification systems which use sodium silicate or sodium meta-silicate.(23,24)

Lime is used in masonry cement which is a mixture of slaked lime and Portland cement. Masonry cement has been studied for use with liquids containing boric acid.(6,14) Boric acid wastes are routinely generated by PWRs and are difficult to solidify with Portland cement alone because the acidic waste interferes with the alkaline processes involved in cement solidification. Addition of lime helps to maintain the alkaline environment during solidification by neutralizing the boric acid.

A large number of clays and zeolites have been tested as additives.(25-27) Both types of materials assist in immobilizing radioactive cations by undergoing ion exchange reactions with waste liquids in cements. These waste forms are often more resistant to leaching, especially of cesium, than the corresponding cement waste form without additives.

Incorporation of styrene monomer into concrete waste forms has been shown to reduce the mobility of radioactive cations(25) but was ineffective in immobilizing tritium.(28) Solidified waste forms consisting of a mixture of cement, zeolite sand, water, and sludge were soaked in a mixture of styrene monomer and a polymerization catalyst. After soaking, the monomer impregnated concrete was heated at 50 to 70°C to induce polymerization of the styrene. Leachability of cesium and strontium from the polymer impregnated forms was about two orders of magnitude less than that of the unimpregnated waste forms.(25)

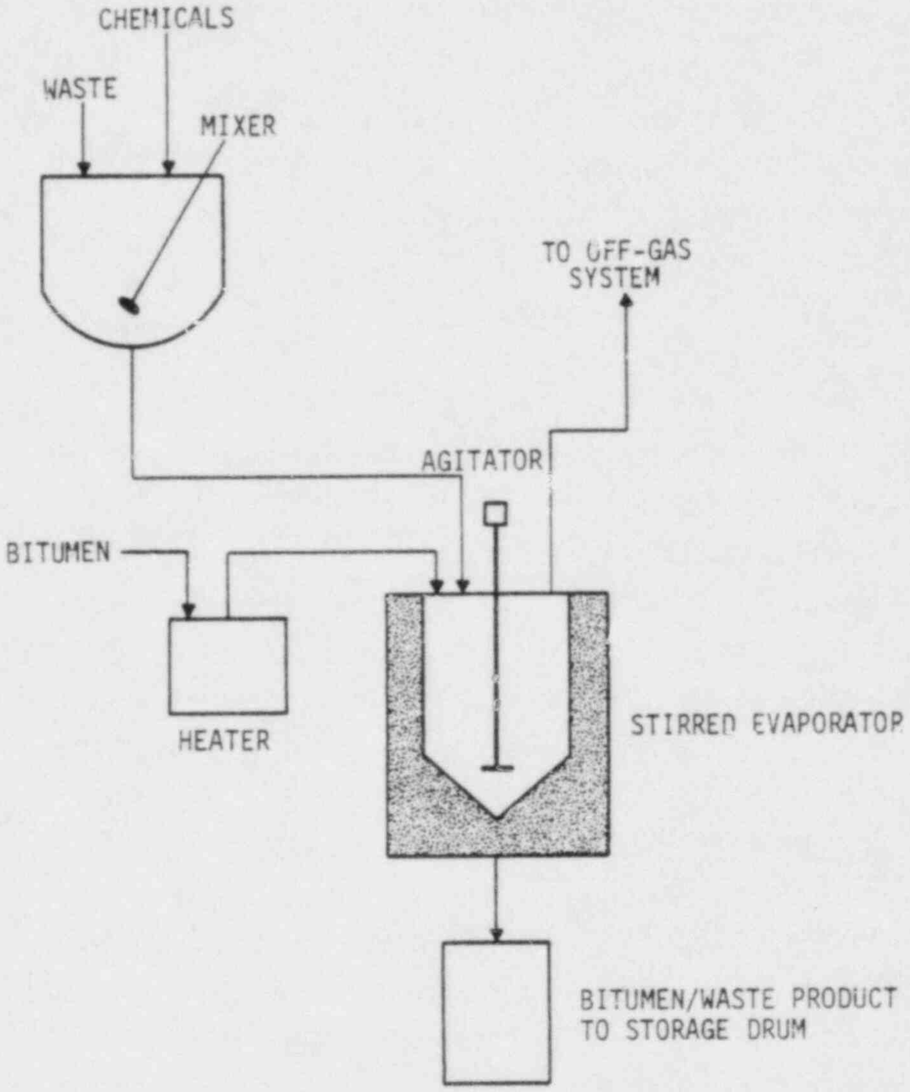
D.2.3 Bitumen

Bitumen (or asphalt) is a mixture of two types of high molecular weight hydrocarbons, asphaltenes and malthenes, which are obtained as a residue in petroleum and coal tar refining. The malthene component behaves as a viscous liquid in which the asphaltenes tend to form colloidal aggregates. These aggregates are more or less mobile depending on the amount and composition of the malthenes. At ambient temperatures bitumen behaves as an elastic solid and at elevated temperature as a viscous liquid.

Four types of bitumen solidification processes have been developed:(22,29) (1) stirred evaporation, (2) emulsified bitumen, (3) wiped-film evaporation, and (4) screw extrusion. All of these processes operate at temperatures of 150 to 230°C, so that any water in the waste may be evaporated. The chemical composition of bitumen and the temperatures used in these processes create the potential for vigorous, if not violent, reactions in the presence of strong oxidizers. Bitumen waste forms tend to contract on cooling so that disposal containers are normally filled more than once to avoid large void volumes.

Stirred Evaporator Process(22)

The stirred evaporator bitumen process (Figure D-3) was originally developed for immobilization of radioactive chemical sludges and later expanded to include concentrated liquids, incinerator ash, and ion exchange resins. The process involves charging an evaporator with preheated bitumen. The waste is introduced and blended with the bitumen using an adjustable blade stirrer. After several hours of blending, the mixture is discharged into a disposal container.



STIRRED-EVAPORATOR BATCH PROCESS

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FIGURE D-3

Emulsified Bitumen Process(22)

In this process (Figure D-4), radioactive waste is mixed with bitumen and surface-active agents in a heated mixer. The hot mixture is passed to a dryer to complete evaporation of water and then discharged to disposal containers.

Wiped-Film Process(30,31)

Wiped-film evaporators crystallize liquid waste by using a rotor to spread a thin film of liquid on a hot metal surface. As the crystalline layer builds up, it is removed by the rotor. Wiped-film evaporators are now available which spread a thin film of a mixture of bitumen and waste on the heated surface (Figure D-5). Bitumen containing the radioactive solids crystallized from the waste liquids is discharged to disposal containers from the bottom of the evaporator.

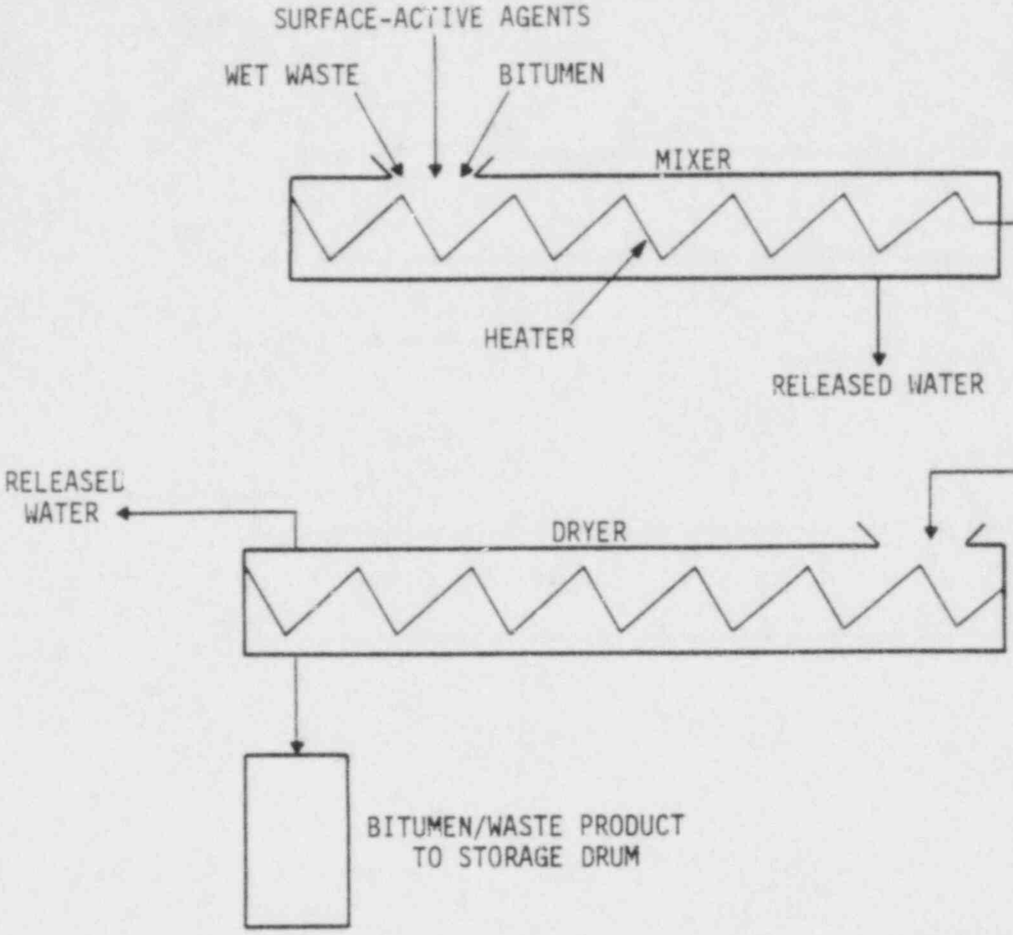
Screw Extruder Process

This process is used at several nuclear and non-nuclear facilities in Europe and handles liquids and wet and dry solid wastes. Waste and preheated bitumen are discharged to a heated steel barrel containing two to four screw extruders (Figure D-6) which mix the materials. The extruders discharge directly into disposal containers.

D.2.4 Urea-Formaldehyde

Urea-formaldehyde (UF) is one of a group of polymers formed by condensation reactions of formaldehyde (CH_2O) with amino compounds ($R-NH_2$). The reaction between urea and formaldehyde ultimately yields a three-dimensional polymer and produces water as a by-product.

Urea-formaldehyde has been used to solidify radioactive wastes for several years and a number of proprietary UF/catalyst systems are



EMULSIFIED BITUMEN PROCESS

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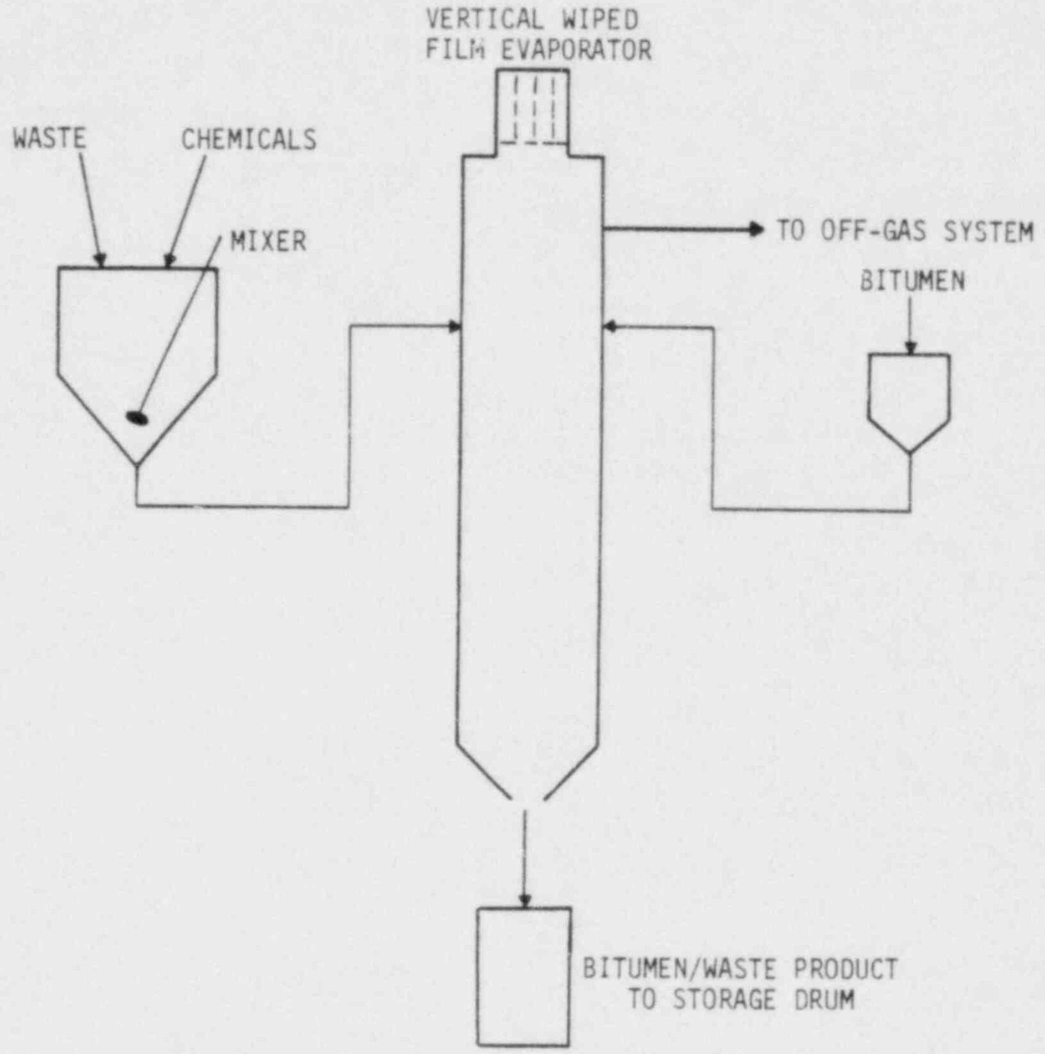
FIGURE D-4



TURBULENT-FILM EVAPORATION PROCESS

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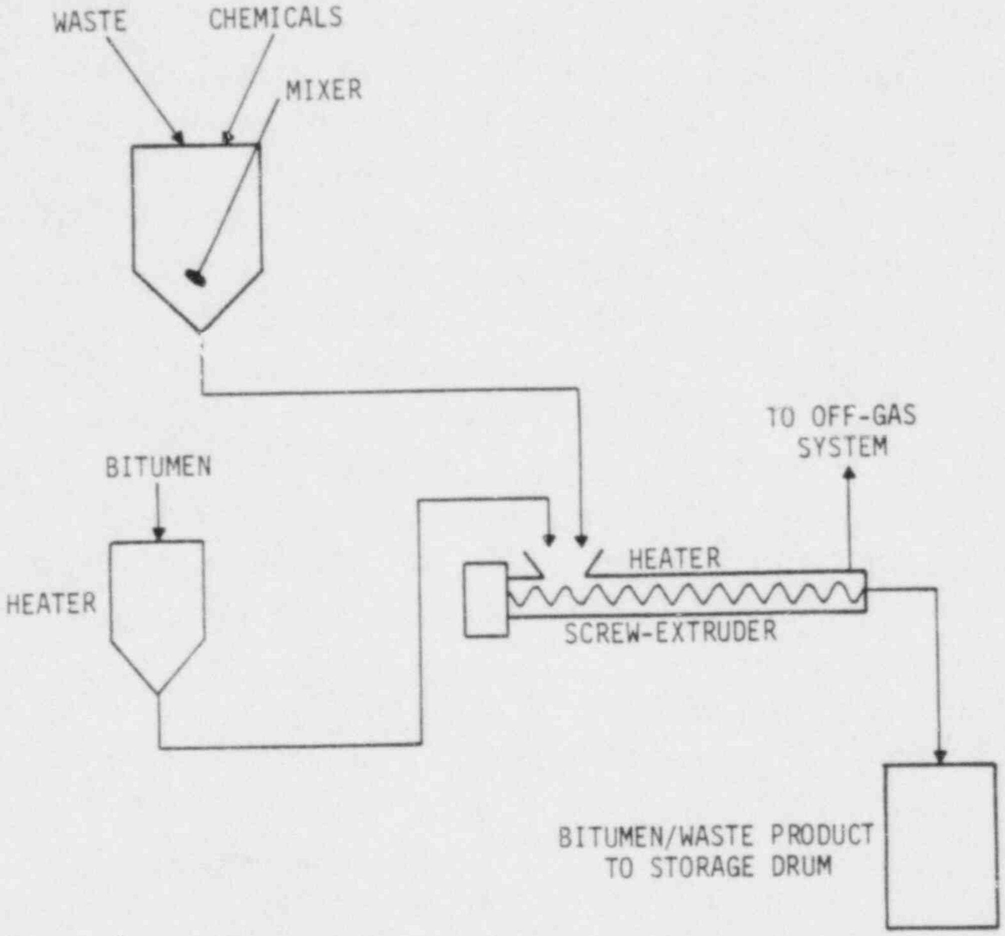
FIGURE D-5



SCREW EXTRUDER EVAPORATION PROCESS

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FIGURE D-6

available.(22) Since formaldehyde is a gas and inconvenient to handle, UF is usually supplied as a partially polymerized emulsion.(14) A typical emulsion consists of a partially polymerized mixture of monomethylol urea ($NH_2CONHCH_2OH$), dimethylol urea ($CH_2OHNHCONHCH_2OH$), and a small amount (<3 wt%) of free formaldehyde (CH_2O). After mixing with the waste material, polymerization is completed by addition of a weak acid catalyst (e.g., sodium bisulfate or phosphoric acid). The final UF polymer has a three dimensional structure which physically entraps the waste. Since polymerization can also be induced by heat or oxidation (contact with air), partially polymerized emulsions have limited shelf-lives. The properties of the final UF polymer can be controlled to an extent by varying the nature and relative amounts of the components of the emulsions.

Processing equipment used for UF solidification is similar to that used for cement. Waste and the partially polymerized emulsion may be mixed either before or after discharge to the disposal container. Batch and continuous systems are available. These materials must be thoroughly mixed before addition of the catalyst to prevent phase separation and incomplete polymerization. Best results are obtained when sufficient catalyst is added to lower the pH of the mixture to about 1.5. Free-standing solid waste forms are normally obtained in less than an hour and quickly harden.

D.2.5 Vinyl Ester-Styrene(14,33)

Vinyl ester-styrene (VES) is a proprietary thermosetting polymer used in a proprietary solidification process, both developed by Dow Industrial Service. Polymerization proceeds by an exothermic addition mechanism using a promoter-catalyst system which permits curing without external heat. Radioactive waste liquids and solids are physically entrapped in the polymer matrix. Free-standing solid waste forms are normally obtained in less than an hour.

D.2.6 Polyester

Incorporation of radioactive hazardous wastes in a polyester-stryrene polymer has been investigated at Washington State University.(34,35) The process uses a water-extensible polyester, a promoter/initiator, and sytrene monomer to produce a three dimensional polymer matrix which physically entraps liquid and solid wastes.

Water extensible polyesters are especially formulated to allow preparation of water-in-polyester emulsions. They have been proposed for use with chemical wastes and oils. Those used in the Washington State University studies were unsaturated linear polyesters made by polymerizing maleic or furmaric acids (unsaturated dicarboxylic acids) with saturated dicarboxylic acids and glycols. The unsaturated acids provide sites for cross-linking of the linear polymer chains, the saturated acids separate these sites, and glycols provide linkages to form the linear polyester chain.

The water extensible polyester, waste, and a promoter are mixed to form an emulsion and the styrene and an initiator are mixed in to produce the final waste form. The curing reaction (cross-linking of the linear polyester by styrene) proceeds by a free radical mechanism and is initiated by peroxides free radicals. These peroxide radicals can be formed in several ways. A convenient method is to add an easily reduced material (promoter) such as cobalt naphthenate or dimethyl aniline which reacts with the peroxide at ambient temperatures to generate free radicals. Under these conditions, the exothermic curing reaction is rapid (complete in about an hour).

It was found that the properties of the waste form are sensitive to the rate of mixing (mixer speed). Gas generation was also observed with boric acid waste but was eliminated by modifying the promoter/initiator composition.

D.2.7 Epoxy

Epoxy resin is used with the inert carrier radwaste process (ICRP) under development by United Technologies.(36) The resin used is commercially available. Commercial resins are supplied as linear pre-polymers made from condensation of the sodium salt of bisphenol A and epichlorohydrin. Curing (foundation of a three-dimensional cross-linked polymer) is accomplished by addition of either tri-functional amines or polybasic acid anhydrides.

The ICRP system performs both volume reduction and solidification. The volume reduction process, described in Appendix C, produces dry waste residues slurried in a hot inert fluid. The residues are kept suspended by high velocity recirculation. A side stream of this fluid is routed through a jet mixer where epoxy pre-polymer is added. The resin-coated residues are separated from the fluid in a separator column, mixed with a hardening agent, and discharged to a disposal container for curing. The final waste form is reported to be very hard and to have a low leachability.

D.3 FREE-STANDING WATER

For purposes of this discussion, free-standing water (FSW) is defined as any liquid not physically or chemically bound within a solidified waste form. Free-standing water is of concern during storage, transportation, and disposal of LLW. It can be corrosive to disposal containers, and may contain higher concentrations of nuclides than the original waste liquid. The contribution of free-standing liquids to leaching rates is discussed in Section D.4 and corrosion of disposal containers in Section D.7.

The presence, amount, and characteristics of free-standing water are dependent on waste and binder types, waste/binder ratios, and waste pretreatment (usually consisting of pH adjustment) and curing time. As discussed below, free-standing water is frequently observed in UF waste forms, is less frequently observed with cement waste forms, and rarely observed in VES and polyester waste forms. The processes for bitumen and epoxy solidification preclude free-standing water when provisions are made to prevent condensation as the waste form cools.

Most of the data presented in this and other sections of this Appendix are taken from a series of reports by Brookhaven National Laboratories.(3-20) The waste formulations used in these studies to represent typical LWR wastes are given in Table D-2.

D.3.1 Cement

Virtually all liquid and wet solid waste can be solidified in Portland cement using reasonably high waste/binder weight ratios without generating free-standing water. For neat cement, which is a mixture of cement and water with no waste, minimum water/cement weight ratios of 0.27, 0.26, and 0.32 are required to obtain workable mixtures for Portland I, II, and III cements, respectively, while the respective

TABLE D-2 . Waste Formulations Used by BNL

1a.	BEAD RESIN WASTE (Slurry)	Weight Percent
	Water	50.0
	Bead Resin (IRN-150)[a]	50.0
	Temperature	70°F
	pH	7
1b.	BEAD RESIN WASTE (Dewatered)	Weight Percent
	Water	35.0
	Bead Resin (IRN-150)[a]	65.0
	Temperature	70°F
	pH	7
2a(1).	BWR PRECOAT FILTER CAKE WITH POWDERED RESIN (Slurry)	Weight Percent in Filter Cake
	Water	50.0
	Anion Powdered Resin (PAO)[b]	20.0
	Cation Powdered Resin (PCH)[b]	20.0
	Crud[c]	5.0
	Sodium Chloride	5.0
	Temperature	70°F
	pH	7
2a(2).	BWR PRECOAT FILTER CAKE WITH POWDERED RESIN (Dewatered)	Weight Percent in Filter Cake
	Water	32.0
	Anion Powdered Resin (PAO)[b]	30.0
	Cation Powdered Resin (PCH)[b]	30.0
	Crud[c]	6.0
	Sodium Chloride	2.0
	Temperature	70°F
	pH	7
2b(1).	BWR PRECOAT FILTER CAKE WITH DIATOMACEOUS EARTH (Slurry)	Weight Percent in Filter Cake
	Water	75.0
	Diatomaceous Earth	20.0
	Crud[c]	5.0
	Temperature	70°F
	pH	7
2b(2)	BWR PRECOAT FILTER CAKE WITH DIATOMACEOUS EARTH (Dewatered)	Weight Percent in Filter Cake
	Water	60.0
	Diatomaceous Earth	30.0
	Crud[c]	10.0
	Temperature	70°F
	pH	7

TABLE D-2 (continued)

3a. BWR CHEMICAL REGENERATIVE WASTE OF A FORCED RECIRCULATION EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	75.0
Sodium Sulfate	22.9
Sodium Chloride	2.0
Crud[c]	0.1
Temperature	170°F
pH	6

3b. PWR CHEMICAL REGENERATIVE WASTE OF A FORCED RECIRCULATION EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	73.4
Sodium Sulfate	14.9
Ammonium Sulfate	9.6
Sodium Chloride	2.0
Crud[c]	0.1
Temperature	170°F
pH	2.5 to 4.0

3c. BORIC ACID WASTE OF A FORCED RECIRCULATION EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	87.9
Boric Acid	12.0
Crud[c]	0.1
Temperature	170°F
pH	3.5

3d. DECONTAMINATION WASTE OF A FORCED RECIRCULATION EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	80.0
Nutek NT-700[d]	9.4
EDTA	5.0
Citric Acid	5.0
Crud[c]	0.2
Hydraulic Oil No. 2	0.2
Lubricating Oil No. 20	0.2
Temperature	170°F
pH	5

4a. BWR CHEMICAL REGENERATIVE WASTE OF A THIN FILM EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	50.0
Sodium Sulfate	45.8
Sodium Chloride	4.0
Crud[c]	0.2
Temperature	150°F to 250°F
pH	6

TABLE D-2 (continued)

4b. PWR CHEMICAL REGENERATIVE WASTE OF A THIN FILM EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	50.0
Sodium Sulfate	29.0
Ammonium Sulfate	16.8
Sodium Chloride	4.0
Crud[c]	0.2
Temperature	150°F to 250°F
pH	1.8 to 4.0

4c. BORIC ACID WASTE OF A THIN FILM EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	50.0
Boric Acid	49.8
Crud[c]	0.2
Temperature	150°F to 250°F
pH	2.5 to 3.5

4d. DECONTAMINATION WASTE OF A THIN FILM EVAPORATOR	Weight Percent in Evaporator Bottoms
Water	50.0
Nutek NT-700[d]	20.0
EDTA	9.8
Citric Acid	19.0
Crud[c]	0.2
Hydraulic Oil No. 2	0.5
Lubricating Oil No. 20	0.5
Temperature	150°F to 250°F
pH	5

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[a] Rohm and Haas Co., Philadelphia, PA 19105. IRN-150 is a mixture of a cation resin (IRN-77) and an anion resin (IRN-78)

[b] Ecodyne Corp., Union, NJ 07083

[c] Fine air cleaner test dust no. 1543094, AC Spark Plug Division, General Motors Corp., Flint, MI 48556

[d] Nuclear Technology Corp., Amston, CT 06231

Source: Reference 14.

maximum ratios to prevent formation of free-standing water are 0.64, 0.68, and 0.96.(16,17)

Two types of data are available for Portland cements: (1) quantity of free-standing water for a given waste/binder ratio, waste type, and pH and (2) range of waste/binder ratios which yield no free-standing water.

The first type of data is presented in Table D-3. As expected, the data shows that high waste/binder ratios favor the formation of free-standing water. It is interesting to note that although increasing the pH of wastes before solidification in Portland II cement may improve waste form integrity and reduce cure times (time required to form a free-standing solid),(14) it is not a universal solution to the problem of free-standing water.

Ion exchange resins can be solidified in cement without free-standing water but it is difficult to obtain waste forms with reasonable integrity. This problem is discussed in Section D.5.

The data given in Tables D-4 through D-6 defines the range of waste/binder weight ratios within which a free-standing Portland cement waste form can be obtained without the formation of free-standing water. As used in these three tables the term "workability" means a cement-waste blend which can be mixed with a mechanical blade mixer. Waste/binder limits for free-standing water represent the waste/binder ratio above which free-standing water is formed in amounts that can be drained from the sample container.

As shown in Tables D-4 and D-5, the ranges of acceptable waste/binder weight ratios for Portland I and II cements are very similar for diatomaceous earth and sodium sulfate wastes, while the same weight Portland III cement can accommodate a larger quantity of either waste. It was found that thorough mixing is essential for successful

TABLE D-3

Free Standing Water in Portland Cement II Waste Forms After Eleven Days

Waste Type[b]	Waste/Binder Weight Ratio	Free Standing Water (wt %)[a]	
		pH=7	pH=10
1. Bead Resin (Slurry)	1.8	5.5±2.5	1.0±0.5
	1.6	0.0	-
	1.5	0.0	0.0
2. BWR Precoat Filter Cake			
a. Powdered Resin[c]	1.8	0.0	0.0
b. Diatomaceous Earth[d]	1.6	1.0±0.3	4.9±0.7
3. Forced Recirculation Evaporator Concentrates			
a. BWR Chemical Regenerative Waste	1.7	8.0±1.0	13.5±3.0
	1.0	4.0±1.0	3.2±3.0
b. PWR Chemical Regenerative Waste	1.0	2.5±0.8	0.0
	0.6	0.0	-
c. Boric Acid Waste	0.6	0.0	3.5±1.0
	0.5	-	0.0
d. Decontamination Waste	1.0	4.0±1.5	0.0
	0.6	0.0	-
4. Thin Film Evaporator Concentrates			
a. BWR Chemical Regenerative Waste	1.5	0.0	0.0
b. PWR Chemical Regenerative Waste	1.0	0.0	0.0
c. Boric Acid Waste	0.7	0.0	2.5±0.5
	0.5	-	0.0
d. Decontamination Waste	1.4	0.0	0.0

(a) Expressed as a weight percent of the total waste form weight.

(b) Waste types and numbering correspond to those listed in Table D-2.

(c) Waste consisted of 70 wt % water, 12 wt % powdered cation resin, 12 wt % powdered anion resin, 3 wt % crud and 3 wt % NaCl. Waste content increased to improve workability.

(d) Waste consisted of 70 wt % water, 24 wt % diatomaceous earth, and 5% crud.

Source: Reference 5.

TABLE D-4

Waste/Binder Weight Ratio Limits for Solidification of Diatomaceous Earth Waste in Portland Cements

PORTLAND I CEMENT

Weight Percent Diatomaceous Earth	Waste/Binder Weight Ratio: Minimum for Workability	Waste/Binder Weight Ratio: Maximum to Preclude Free Standing Water[a]
0	0.27	0.64
10	0.40	0.80
25	0.95	1.20

PORTLAND II CEMENT

Weight Percent Diatomaceous Earth	Waste/Binder Weight Ratio: Minimum for Workability	Waste/Binder Weight Ratio: Maximum to Preclude Free Standing Water[a]
0	0.26	0.68
10	0.35	0.80
25	0.90	1.20

PORTLAND III CEMENT

Weight Percent Diatomaceous Earth	Waste/Binder Weight Ratio: Minimum for Workability	Waste/Binder Weight Ratio: Maximum to Preclude Free Standing Water[a]
0	0.32	0.96
10	0.45	1.40
25	1.00	2.40

(a) After three days curing

Source: Reference 17

TABLE D-5

Waste/Binder Weight Ratio Limits for Solidification of Sodium Sulfate Waste in Portland Cements

PORTLAND I CEMENT

Weight Percent Na_2SO_4	Minimum for Workability	Maximum to Preclude Free Standing Water[a]	Crystal Layer[b]
0	0.27	0.64	-
10	0.36	0.80	-
20	0.38	0.80	-
25	0.38	0.80	-
35	0.45	5.0	2.5
50	0.60	8.5	3.8

PORTLAND II CEMENT

Weight Percent Na_2SO_4	Minimum for Workability	Maximum to Preclude Free Standing Water[a]	Crystal Layer[b]
0	0.26	0.68	-
10	0.34	0.80	-
20	0.36	0.80	-
25	0.36	0.80	-
35	0.46	4.5	2.6
50	0.60	7.5	3.7

PORTLAND III CEMENT

Weight Percent Na_2SO_4	Minimum for Workability	Maximum to Preclude Free Standing Water[a]	Crystal Layer[b]
0	0.32	0.96	-
10	0.40	0.90	-
20	0.40	1.0	-
25	0.40	1.2	-
35	0.50	5.5	2.7
50	0.70	8.0	4.0

(a) After three days curing.

(b) Crystal layer with thickness greater than 0.5 mm after three days curing.

Source: Reference 16.

TABLE D-6

Waste/Binder Weight Ratio Limits for Solidification of Boric Acid Waste in Portland III Cements[a]

Weight Percent	pH 3.0		pH 7.0		pH 10.0		pH 12.0	
Boric Acid	Min.[b]	Max.[c]	Min.	Max.	Min.	Max.	Min.	Max.
3 wt. %	0.32	0.70	0.32	0.70	0.34	0.80	0.34	0.80
6 wt. %	0.32	0.70	0.32	0.70	0.34	0.80	0.34	0.80
12 wt. %	0.35	-	0.35	0.50	0.38	0.70	0.38	0.90

(a) Cure times range from two to ten days; pH adjustments made with sodium hydroxide pellets.

(b) Minimum for workability.

(c) Maximum to preclude free standing water.

Source: Reference 19.

solidification of diatomaceous earth in all three Portland cements and that simulated wastes containing 50 weight percent diatomaceous earth did not contain enough water to permit mixing.(17)

The sodium sulfate/Portland cement waste forms exhibited several types of unusual behavior.(16) Frequent partial phase separations were observed early in the curing process but the water was usually reabsorbed within 24 hours. With 35 and 50 weight percent sodium sulfate solutions, it was observed that while very high waste/binder ratios could be used without producing free-standing water, sodium sulfate crystals were formed on the surface of the waste forms. Crystal formation was observed at a waste/binder ratio of about 50 percent of the limiting ratio for free-standing water production.

The presence of these crystal layers is considered as detrimental as the presence of free-standing water. It is probable that a significant amount of the radioactivity of the sodium sulfate waste would be contained in the crystal layer. Since sodium sulfate is readily soluble in water, this radioactivity is expected to be highly mobile. The waste forms obtained near the waste/binder limit for free-standing water contain little cement and are also expected to have poor mechanical properties.

The range of acceptable waste/binder ratios for solidification of boric acid waste in Portland III cement is given in Table D-5. The range of acceptable ratios appears to be insensitive to pH but curing times are quite sensitive. In an earlier study(18) of boric acid/Portland III waste forms, cure times were found to decrease from 40 to 14 days as the pH was increased from 3.9 to 12.0 using a 10 M sodium hydroxide solution. A later study(19) (Table D-5) reported cure times ranging from two to ten days for all successfully solidified waste/binder ratios. Boric acid wastes have also been successfully solidified in masonry cement at waste/binder weight ratios of from 0.6 to 1.0.(6)

Limited work with Portland II cement and sodium silicate additive is inconclusive. Free-standing water appears to have been present in some samples but in amounts described as insignificant.(9)

D.3.2 Urea-Formaldehyde

Since the polymerization reactions which produce urea-formaldehyde generate water as a by-product it is not surprising that free-standing water is frequently observed in UF waste forms. This water contains the polymerization catalyst (catalysts are not consumed in chemical reactions) and, as a result, is acidic with pH's ranging from 1.5 to 3.8.(4,5) Such acidic water is corrosive and increases the solubility (and solution stability) of many of the radionuclides found in low-level waste.

The data presented in Table D-7 is indicative of the frequency of free-standing water formation and of its acidity in UF waste forms. Only four of 37 waste forms did not contain free-standing water. Reducing the pH of the UF/waste emulsion does decrease the amount of free-standing water but at the expense of increasing its acidity.

As shown in Figure D-7 and D-8, the quantity of free-standing water does not increase monotonically with increasing waste/UF weight ratio but passes through a maximum. Figure D-8 and Table D-8 suggest that this behavior is related to shrinkage of the waste forms. A possible explanation of the shape of the curves shown in Figures D-7 and D-8 is that at low waste/binder ratios, incorporation of increasing amounts of waste in UF causes an increasing amount of strain in the polymer network. This strain is relieved by squeezing out increasing amounts of water. Beyond a certain waste/binder ratio, the polymer network is forced by the increasing volume of waste to a less-strained structure. Such structural changes would be expected to affect waste form leachability and integrity.

TABLE D-7 . Free Standing Water in Urea-Formaldehyde Waste Forms After Seven Days

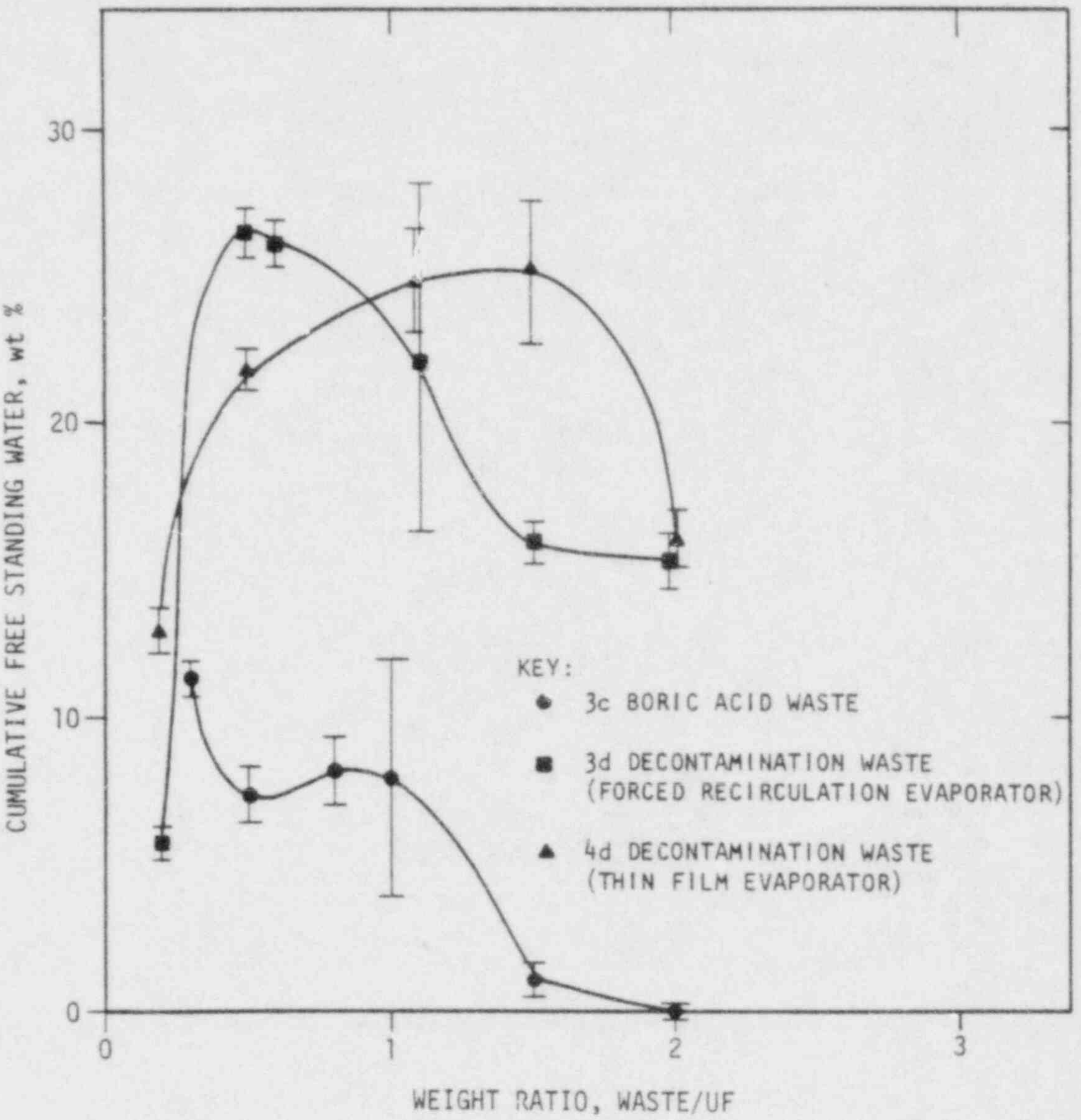
Waste Type[a]	Waste/UF Weight Ratio	2% Catalyst[b] (by volume)		Catalyst Added to Achieve pH = 1.5+0.5		
		Free Standing Water		Volume Percent Catalyst Added	Free Standing Water	
		Weight Percent[c]	pH		Weight Percent[c]	pH
1) Bead Resin (Slurry)	2.2	0.5	--	1.8	0.45	--
2) BWR Filter Cake						
a) Powdered Resin (Slurry)	2.0	0.3	--	1.9	0.25	--
b) Diatomaceous Earth (Dewatered)	2.0	8.8	2.8	3.0	0	--
	1.0	3.1	--	--	--	--
3) Forced Recirculation Evaporator Concentrates						
a) BWR Regenerative Waste	1.2	1.6	2.3	--	--	--
	1.3	--	--	3.1	0.23	--
b) PWR Regenerative Waste	1.2	1.8	1.5	--	--	--
	1.3	--	--	2.9	0.80	1.5
c) Boric Acid Waste	2.0	0	--	--	--	--
	1.5	1.2	--	--	--	--
	1.2	--	--	0.8	1.5	1.6
	1.0	8.0	1.7	--	--	--
	0.8	8.1	--	--	--	--
	0.5	7.4	--	--	--	--
	0.3	11.3	--	--	--	--
d) Decontamination Waste	2.0	15.5	--	--	--	--
	1.5	16.0	--	--	--	--
	1.2	--	--	10.5	4.0	2.0
	1.1	22.2	3.7	--	--	--
	0.6	26.1	--	--	--	--
	0.5	26.4	--	--	--	--
	0.2	26.1	--	--	--	--

TABLE D-7 (continued)

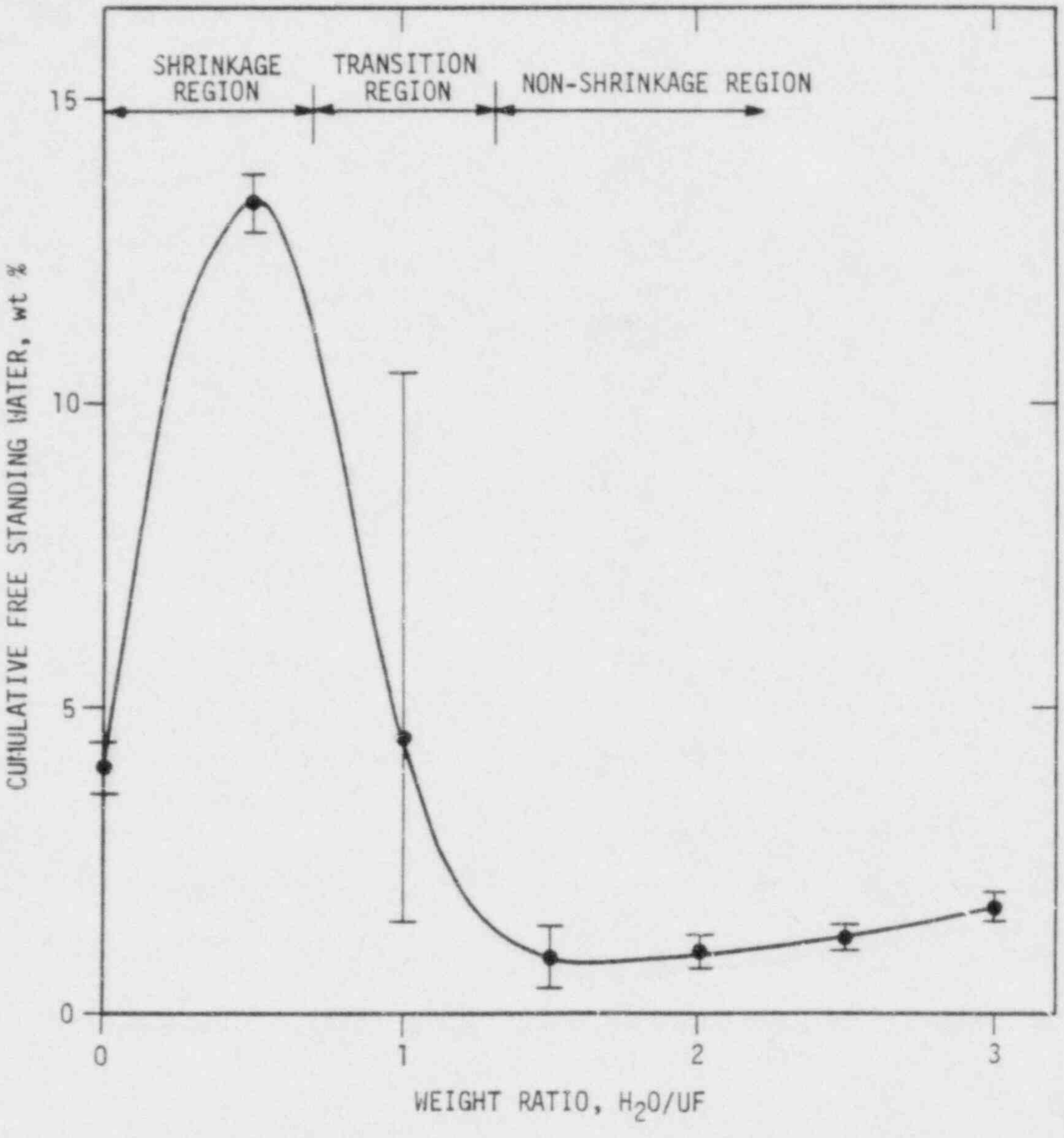
Waste Type[a]	Waste/UF Weight Ratio	2% Catalyst[b] (by volume)		Catalyst Added to Achieve pH = 1.5+0.5		
		Free Standing Water Weight Percent[c]	pH	Volume Percent Catalyst Added	Free Standing Water Weight Percent[c]	pH
4) Thin-Film Evaporator Concentrates						
a) BWR Regenerative Waste	1.0	0	--			
	1.5	--	--	1.5	0	--
b) PWR Regenerative Waste	0.7	4.0	2.0	--	--	--
	1.0	--	--	1.7	7.2	1.6
c) Boric Acid Waste	1.0	1.1	2.0	--	--	--
	1.2	--	--	1.4	0.55	1.4
d) Decontamination Waste	2.0	16.0	--	--	--	--
	1.5	25.4	3.8	13.3	11.0	2.6
	1.0	25.5	--	--	--	--
	0.5	21.8	--	--	--	--
	0.2	13.0	--	--	--	--

(a) Composition of waste types is given in Table D-2.
(b) 24 weight percent aqueous sodium bisulfate.
(c) As a percent of the total weight of the waste form.

Source: References 4, 5, and 14.



CUMULATIVE FREE STANDING WATER AS A FUNCTION OF THE WASTE/UF WEIGHT RATIO FOR SELECTED WASTES



CUMULATIVE FREE STANDING WATER
AS A FUNCTION OF THE
WATER/UF WEIGHT RATIO

DAMES & MOORE

FIGURE D-8

TABLE D-8

Decrease in Diameter (Shrinkage) of Urea-Formaldehyde Waste Forms

Waste Type	Waste/Binder Weight Ratio	Percent Decrease in Diameter	Weight Percent Free Standing Water
3.c. Boric Acid Waste (forced recirculation evaporator)	1.0	5.5	8.0
	0.8	6.7	8.1
	0.5	6.7	7.4
	0.3	6.3	11.3
3.d. Decontamination Waste (forced recirculation evaporator)	2.0	6.7	15.5
	1.5	8.3	16.0
	1.0	7.1	22.2
	0.5	13.4	26.4
	0.2	3.9	5.8
4.d. Decontamination Waste (thin film evaporator)	2.0	6.7	10.0
	1.5	9.1	25.4
	1.0	8.7	25.5
	0.5	10.6	21.8
	0.2	8.3	13.0
Distilled Water	1.0	5.5	4.0
	0.5	9.1	13.4
	0.0	6.7	4.6

Source: Reference 14.

Additional studies(6) with UF waste forms have shown that unsatisfactory waste forms are obtained with Solka-Floc and with alkaline regenerative wastes (regenerative wastes listed in Table D-6 but adjusted to a pH of 10). Solka-Floc is a fibrous high purity cellulose material used as a precoat filter media in LWRs. At a waste/binder weight ratio of 2.0, the Solka-Floc waste forms were either incompletely solidified or did not harden. Alkaline regenerative wastes could not be solidified using two volume percent of acid catalyst.

Use of waste/binder ratios of 3.0 were investigated(6) for regenerative, boric acid, and decontamination wastes. The regenerative waste form did not solidify. The boric acid waste form solidified without free-standing water although water could be easily squeezed from the final waste form. The decontamination waste form contained 16 weight percent free standing water.

Studies of bead resin, sodium sulfate, and boric acid wastes solidified with a new proprietary "two-part" urea-formaldehyde process showed that free-standing water was formed (<1 weight percent) and sample shrinkage occurred.(19) The pH of the free-standing liquids was 2 or less.

D.3.3 Bitumen

Available waste bitumenization systems are designed to completely evaporate any water in the waste being processed. Free-standing water could possibly be formed if system throughput rates are exceeded or if containers are sealed while hot, thus allowing condensation of water vapor within the container.

D.3.4 Vinyl Ester-styrene

Vinyl ester-styrene (VES) waste forms have not been studied as extensively as cement or urea formaldehyde waste forms, however, free-standing water has not been observed with common wastes.(7,14,33,38) Bead resins, chemical regenerative, diatomaceous earth, boric acid and dry solid wastes have all been successfully solidified at waste/binder weight ratios ranging from 1.5 to 2.5. Proprietary pretreatment was required for boric acid waste. Dow Industrial Service's proprietary decontamination solvent has also been successfully solidified. The available data is insufficient to determine the range of acceptable waste/binder weight ratios.

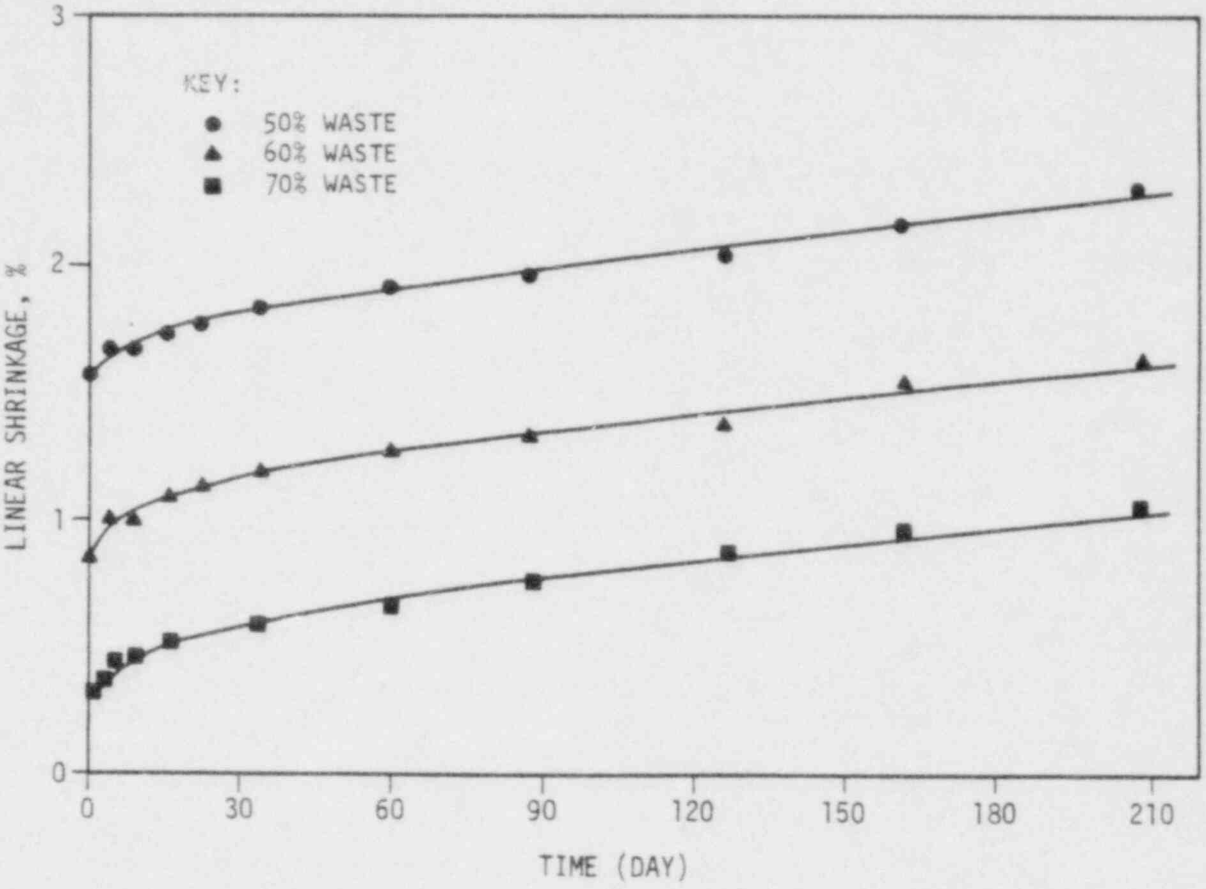
Water/VES waste forms have been observed to lose up to 42 percent of their original weight after 70 days exposure to ambient air.(5,14) These weight losses were attributed to evaporation. No free-standing water or waste from shrinkage was reported.

Evaporation of water from radioactive VES waste forms could result in deposition of the radioactive species as salts on the outer surfaces of the waste forms where they would be highly mobile. It is expected that, after disposal, the natural presence of moisture in soils would hinder this type of evaporative process.

D.3.5 Polyester

Aqueous solutions of sodium sulfate and boric acid with anhydrous sodium sulfate, sodium borate and sodium meta-borate have been solidified in polyester without formation of free-standing water or surface crystals.(34,35) Waste/binder ratios ranged from 1.2 to 2.3. Boric acid and borate waste forms required about 24 hours to cure.

Some shrinkage of sodium sulfate/polyester waste forms have been reported.(35) As shown in Figure D-9, shrinkage amounted to in the worst case slightly more than two percent of the sample length after 210 days.



LINEAR SHRINKAGE
OF POLYESTER-ENCAPSULATED
SODIUM SULFATE WASTE COMPOSITE

DAMES & MOORE

FIGURE D-9

D.3.6 Epoxy

Free-standing water is not expected with the epoxy solidification system being developed since the system is designed to completely evaporate any water in the waste.(36,37)

D.4 LEACHABILITY

Leaching of low level radioactive waste forms is of primary concern in the management of these wastes. The leached radioactive species are potentially highly mobile in the environment.

The processes by which water reaches buried waste forms and by which the leached radioactive species migrate out of the immediate disposal environment (i.e., the disposal trench) are discussed in Volume 3 of this series of reports. This appendix is concerned with the rate of leaching.

Once water (leachant) has reached and penetrated into a buried waste form, the rate of leaching is controlled by three major processes: (1) dissolution, (2) ion-exchange, and (3) diffusion. These processes are sensitive to temperature, pH, ionic concentrations, oxidation-reduction (redox) potential and other effects.

The processes involved in dissolution can range from simple hydration to chelation and redox reactions. Ion exchange reactions during leaching are not restricted to waste forms containing ion exchange resins. For example, diatomaceous earths are capable of ion exchange as are cement waste forms. Radioactive species transferred to the aqueous phase by dissolution and ion exchange, as well as those already in solution (entrapped waste liquids), escape from the waste form by diffusion through the leachant.

The driving force for diffusion is the net decrease in free energy as the leached species moves from the region of high concentration inside the waste form to the region of lower concentration outside. The rate of diffusion is dependent on a number of parameters which include the viscosity of the leachant and the effective porosity and geometry of the waste form. As discussed in Section D.4.1, the common methods of leach data analysis assume that diffusion is the process controlling the rate of leaching.

D.4.1 Leach Testing Proceedures and Data Analysis

Several experimental procedures have been used in laboratory studies of waste form leachability. Those considered here are similar in that they all involve complete immersion of a right-cyclindrical sample in the leachant. The differences among these procedures include differences in leachant replacement frequency, leachant volume to sample surface area ratios (V_L/S), sample volume to sample surface area ratios (V/S), sampling frequency, type of leachant, and length of tests.

Leach testing procedures can be categorized on the basis of leachant replacement frequency. In equilibrium procedures,(14,34) the leachant is not replaced. The waste form or an aliquot of the leachant is removed, analyzed non-destructively, and returned to the original container. In many cases, the system moves toward equilibrium which, in turn, limits the amount of species leached.

For leaching by diffusion the driving force for the process decreases as the concentration of the leachant increases and becomes zero at equilibrium. For this reason the equilibrium procedure minimizes leachability. It is also clear that increasing either V/S or V_L/S while holding waste form geometry constant will increase the total amount of a species leached. Variations in both sample geometry and V/S are common and make comparison of results for different studies difficult.

When static procedures are used, the leachant is completly replaced each time a sample is taken. With the modified IAEA procedure used by BNL,(14) sampling frequency decreases as the experiment progresses. Four samples are collected during the first day, one each day during the next week, and one each day during the next five months. The V_L/S ratio is normally 10 cm. This procedure obviously maximizes leaching. Other researchers (33,40) sample daily for the duration of the test and use other V_L/S ratios (frequently less than 10 cm). As a result of smaller leachant volumes and less frequent replacement, the experimental results obtained fall between those obtained with the modified IAEA and equilibrium procedures. It should be noted that the leach testing procedures used by BNL specify that any free-standing water formed during sample preparation be transferred to the leaching container.(14)

Data from laboratory studies of waste form leaching (both static and equilibrium) is frequently treated using the semi-infinite model for mass transport by diffusion. This model assumes that at least a part of the waste form retains its initial concentration during the entire leaching period. For a homogeneous semi-infinite medium with zero surface concentration at $t>0$, the leaching rate due to diffusion is (neglecting radioactive decay)(39):

$$(\sum a_n/A_o)\ (V/S) = 2\sqrt{(D_e/\pi)\sum t_n} \qquad \text{(D.1)}$$

where

$\sum a_n$ = cumulative radioactivity leached
A_o = initial radioactivity
V = waste form volume
S = waste form surface area
D_e = Effective diffusivity
$\sum t_n$ = cumulative leach time

Thus, plotting the left-hand side of Equation D.1 versus the square root of the cumulative leaching time should yield a straight line, and the effective diffusivity, D_e, can be derived from the slope of the line. Plots of this type are rarely linear at short leaching times. If the nonlinear region represents a small part of the total cumulative fraction leached, it can be handled by adding a constant to the right-hand side of Equation D.1. The modified equation is then used to predict leachability at longer times.

Another common method of presenting leach test data is to plot ($\sum a_n/A$) X (V/S) versus t. Some typical curves obtained are shown in Figure D-10. Curves of the type labelled (a) are common for leaching of some species from urea-formaldehyde, bitumen, and cement waste forms and cannot be used to predict long term leachabilities. Testing of samples with larger V/S ratios can be useful in these cases.

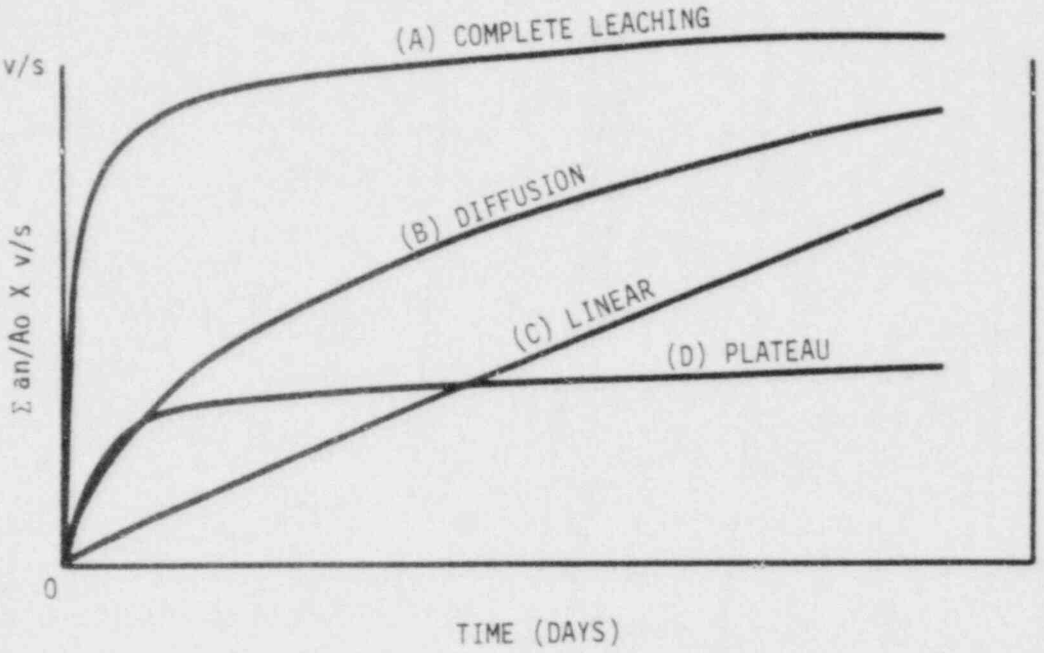
When expressed in the form of Equation D.1, the cumulative fraction leached ($\sum a_n/A$) X (V/S) should be independent of waste form dimensions so that;

$$(\sum a_n/A_o)_1 \times (V/S)_1 = (\sum a_n/A_o)_2 \times (V/S)_2 \qquad \text{(D-2)}$$

Equations D.1 and D.2 should permit prediction of the long-term leachability of full-sized waste forms such as 55-gallon drums (V/S = 10.8 cm) from small laboratory samples (V/S = 0.5 cm). Of course, samples exhibiting the leaching behavior represented by curve (a) in Figure D-10 cannot be treated in this way since they do not meet the requirements of a semi-infinite medium.

Recent preliminary studies at BNL(20) show that Equation D.2 is not valid for leaching of Cs-137 and Sr-90 from BWR regenerative/ Portland II cement waste forms.

Representative experimental leaching curves are presented in Figure D-11. Equation D.2 predicts that all six samples yield a single curve which is not the case in Figure D-11. A more complex expression is available to describe diffusion mass transport from a finite medium(14,39) and involves a multiple summation term. The value of $\sum a_n/A_o$ tends to converge to a minimum as the number of terms included increases. However, the number of terms becomes so large (on the order of 10^6 terms) that computer round-off error can become a problem.(14)



REPRESENTATIVE LEACHING CURVES

DAMES & MOORE

FIGURE D-10

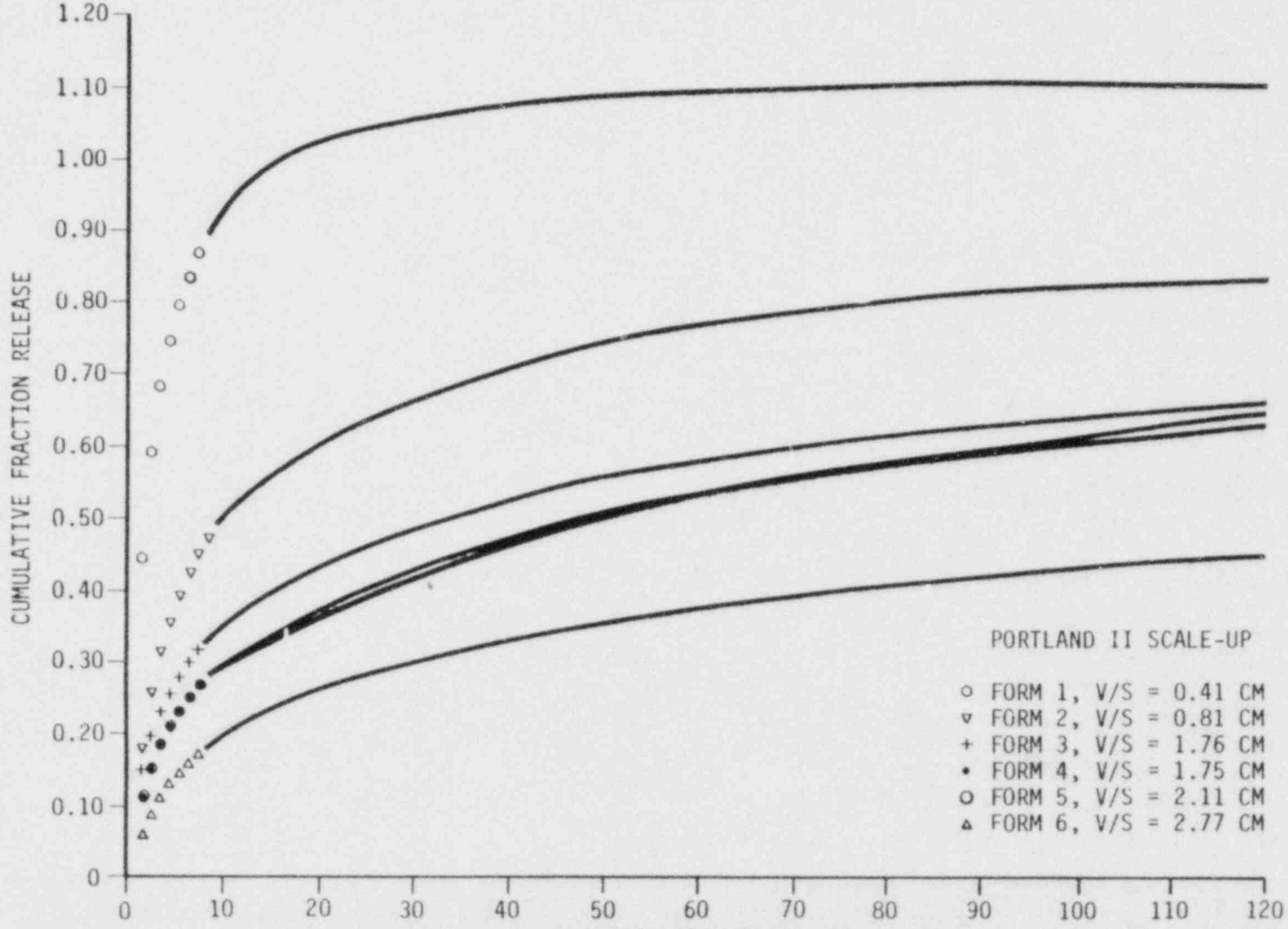


FIGURE D.11

DATA FOR LEACHING OF CS-137 FROM BWR REGENERATIVE/PORTLAND II CEMENT WASTE FORMS SHOWING DEPENDENCE ON SAMPLE DIMENSIONS

D.4.2 Leaching Data

As discussed in the previous section, breakdown of the assumptions of the semi-infinite model and the computational difficulties of the finite model preclude meaningful projections of long-term leachabilities of full-sized waste forms from laboratory studies of small samples. Furthermore, variations in leach testing procedures and conditions make comparisons of the leachabilities of different waste binders difficult.

Useful insights, however, can be gained by consideration of the available data. For example, Table D-9 gives an indication of a basic difference between cement and urea-formaldehyde. Both binders frequently yield free-standing water; however, free standing water associated with UF contains more radioactivity.

As expected for the acidic and non-ionic environment in UF, selective retention of radionuclides does not occur. The situation is considerably different in the alkaline and highly ionic cement environment. General rules for the selectivity of ion exchangers in simple systems are: (1) more highly charged ions are held more strongly than ions with lower charges, at the same concentrations; and (2) for ions of the same charge, the larger (less hydrated) ions are held more strongly, at the same concentrations. These rules predict that retention decreases in the order $Sr^{+2} > Co^{+2} > Cs^{+}$. The observed strong retention of cobalt (Co^{+2}), weaker retention of strontium (Sr^{+2}), and indifferent retention of cesium (Cs^{+}) in cement suggests that the effects of solubility under alkaline conditions ($Cs^{+} >> Sr^{+2} > Co^{+2}$) and competition with high concentrations of non-radioactive ions generated during the solidification process cause a breakdown of general selectivity rules.

The importance of ion exchange in cement waste forms is also evidenced by the data presented in Tables D-10 and D-11. During the early

TABLE D-9 . Decontamination Factors for Free-Standing Water from Portland II Cement and Urea-Formaldehyde(5,7)

Binder	Waste/Binder Weight Ratio	Decontamination Factor[a]			
		Cs-137	Sr-85	Co-60	Fe-59
Portland II Cement	1.0	0.91	11.0	200	-
Urea-Formaldehyde	2.0	0.96	1.93	0.97	1.12
	3.0	0.93	1.60	1.20	1.02

(a) The decontamination factor is the ratio of the activity (μCi/ml) of each isotope initially in the water to that in the free standing water.

TABLE D-10 . Cesium-137 Activity Remaining on IRN-77 Cation Exchange Resin as a Function of Contact Time with Portland II and Luminite Cements(20)

Cement Type	Contact Time	Percent Cs-137 Remaining on Resin
Portland II	5 min	83.0 ± 12.3
	1 hr	73.0 ± 7.4
	2 hrs	73.0 ± 12.3
Luminite (HAC)	5 min	87.3 ± 11.3
	1 hr	61.1 ± 9.5
	2 hrs	57.4 ± 12.9

TABLE D-11 . Composition of Portland II and Luminite Cements(41)

Cement Type	Composition (weight percent)						
	CaO	SiO_2	Al_2O_3	FE_2O_3	MgO	Si_2O_3	Other
Portland II	63.3	22.4	4.6	4.3	2.5	1.7	1.2
Luminite (HAC)	36.5	8.5	40.5	5.5	1.0	0.2	7.8[a]

(a) Includes Fe_2O_3 5.5% and TiO_2 2%.

stages of cement solidification many multi-valent cations (e.g. Al^{+3}, Fe^{+3}) are in a semi-soluble state and available for ion exchange. As seen in Table D-11, Luminite cement contains more of these species than Portland II cement and accordingly displaces a larger amount of Cs^+ from a cation exchange resin. This behavior strongly suggests that cement solidification of cation exchange resins mobilizes rather than immobilizes the Cs^+. The generally poor mechanical integrity of resin/cement waste forms has prevented extensive leach testing.

Results of static and equilibrium leach testing of simulated waste forms are complied in Table D-12 and the leachant composition used during the tests is presented in Table D-13.

In cases when experimental data was not available for a cumulative leaching time of 100 days, the linear portion of the available data was either extrapolated graphically or by regression analysis. Bitumen and epoxy waste forms are not included since leaching of non-radioactive species was studied (Na^+, SO_4^{-2}, boron compounds).

The data presented in Table D-12 for bead resins, when considered in light of that in Table D-10, indicates that, of the waste forms tested, only vinyl ester-styrene is effective in reducing the leaching of resins. The data also show the desirability of isolating unsolidified ion exchange resins from leachants containing elevated levels of dissolved solids. As expected on the basis of the higher V/S ratio of powdered resins relative to bead resins, a larger cumulative fraction release ($\sum a_n/A_o$) of cesium and strontium are leached from powdered resin.

Data for equilibrium leaching of sodium sulfate wastes indicate that leachabilities of vinyl-ester styrene and polyester waste forms are low for all of three elements tested (strontium, cesium, and cobalt). Portland II cement shows good retention of strontium and

TABLE D-12 . Leachability of Cesium, Strontium and Cobalt from Simulated Waste Forms

Waste Type[a]	Binder[b]	Leachant[c]	Method[d]	W/B[e]	V/S[f] (cm)	Cumulative Fraction Leached X(V/S) @ 100 days			Ref.
						Cs	Sr	Co	
A. Bead resin (no free water)	None	DS	MS	NA[g]	8.75E-3[h]	1.42E-4	9.16E-6	-	6
	None	G	MS	NA	8.75E-3	3.43E-3	1.01E-4	-	6
	None	S	MS	NA	8.75E-3	8.75E-3	8.75E-3	-	6
Bead resin slurry [1a]	UF	G	MS	2.6	0.507	1.20E-1	2.42E-2	-	10
	UF	S	MS	2.7	0.507	5.07E-1	3.68E-1	-	10
90 wt % bead resin	VES	DI	DS	2.2	0.263	1.25E-5	-	1.36E-5	40
B. Powdered resin (no free water)	None	DS	MS	NA	9.25E-4	1.64E-4	2.18E-4	-	6
	None	G	MS	NA	9.25E-4	9.25E-4	9.25E-4	-	6
	None	S	MS	NA	9.25E-4	9.25E-4	9.25E-4		6
C. BWR regenerative waste [3a]	UF	DS	E	1.3	0.501	4.43E-1	3.86E-1	-	12
	UF	DS	MS	1.3	0.503	5.03E-1	3.43E-1	-	12
	UF	G	MS	1.3	0.504	2.78E-1	4.53E-1	-	10
	UF	S	MS	1.3	0.504	3.26E-1	4.92E-1	-	10
	PCII	DS	E	1.0	0.495	4.95E-1	3.57E-3	0[i]	12
	PCII	DS	MS	1.0	0.493	4.64E-1	3.29E-1	1.00E-1	12
	PCII	DS	MS	1.0	0.495	4.72E-1	2.87E-1	-	10
	PCII	G	MS	1.0	0.495	4.57E-1	3.22E-1	-	10
	PCII	S	MS	1.0	0.495	4.62E-1	1.94E-1	-	10
	VES	DS	E	1.9	0.503	2.86E-2	3.58E-2	4.29E-2	12
	VES	DS	MS	1.9	0.491	3.57E-2	4.29E-2	2.14E-3	12
12 wt % Na_2SO_4	VES	DI	DS	2.0	0.268	1.60E-2	-	1.05E-2	40
24 wt % Na_2SO_4	PE	DI	E	1.0	0.376	2.38E-3	2.18E-3	1.51E-3	34
	PE	DI	E	1.5	0.368	5.52E-3	3.22E-3	3.02E-3	34

TABLE D-12 (cont'd)

Waste Type [a]	Binder[b]	Leachant[c]	Method[d]	W/B[e]	V/S[f] (cm)	Cumulative Fraction Leached X(V/S) @ 100 days			Ref.
						Cs	Sr	Co	
D. Boric acid waste	UF	DS	MS	2.0	0.509	4.79E-1	4.39E-1	-	10
[3c]	UF	G	MS	2.0	0.509	4.83E-1	4.74E-1	-	10
	UF	S	MS	2.0	0.509	4.91E-1	4.78E-1	-	10
[3c]	PCIII	DS	MS	0.5	0.514	-	-	1.02E-2	15
	PCIII	G	MS	0.5	0.514	-	-	6.46E-3	15
	PCIII	S	MS	0.5	0.514	-	-	1.35E-2	15
6 wt % H_3BO_3	VES	DI	DS	1.7	0.267	1.44E-2	-	7.30E-3	40
20 wt % H_3BO_3	PE	DI	E	1.0	0.377	4.48E-2	8.74E-3	2.67E-3	34
E. Diatomaceous	UF	DS	MS	2.0	0.509	2.71E-2	3.52E-1	-	10
Earth [2b(1)]	UF	G	MS	2.0	0.510	3.31E-2	4.59E-1	-	10
	UF	S	MS	2.0	0.511	2.80E-1	5.05E-1	-	10
[2b(1)]	PCII	DS	MS	1.6	0.495	4.25E-1	2.97E-1	-	10
	PCII	G	MS	1.6	0.495	4.25E-1	2.70E-1	-	10
	PCII	S	MS	1.6	0.495	4.62E-1	2.20E-1	-	10
90 wt % D.E.	VES	DI	DS	1.6	0.783	5.68E-2	-	3.40E-2	40

(a) Numbers in brackets refer to waste formulations given in Table D-2.

(b) UF=urea-formaldehyde; PCII, PCIII=Portland cement, Type II and Type III; VES=vinyl ester-styrene; PE=polyester.

(c) DS=distilled water; G=groundwater (see Table D-13); S=sea water; DI=deionized water.

(d) MS=modified IAEA static leaching; DS=static leaching with daily leachant replacement; E=equilibrium leaching.

(e) Waste/binder weight ratio.

(f) Volume to surface area ratio (cm) of waste form.

(g) NA=not applicable. Note that V/S is independent of sample geometry and quantity for these wastes.

(i) Amount of cobalt leached was too small to read from graph.

TABLE D-13

Composition of Groundwater Leachant Used in BNL Studies

pH	6.2
Conductivity, umho	130

Constituent	Content, ppm
Dissolved oxygen	9.4
Chloride	18.1
Total phosphorus	<0.05
Total nitrogen	1.1
Dissolved solids	92
Copper	0.12
Silver	<0.005
Lead	0.004
Zinc	0.045
Cadmium	<0.008
Chromium	<0.008
Iron	0.061

Source: Reference 10.

cobalt but poor retention of cesium. Under static leaching conditions only VES shows good leach resistance. Leaching of sulfate from sodium sulfate/bitumen waste forms is complete (100 percent) within about 10 days,(14) while only 11 weight percent is leached from epoxy within 100 days.(36)

The limited data for boric acid waste forms shows that the leachability of UF is very high. The lower leachability of cobalt from VES and polyester relative to Portland III cement may be due to less frequent leachant replacement rather than the properties of the binders. About 15 percent of the total boron content of a boric acid/bitumen waste form was leached in 100 days.(14)

Urea-formaldehyde, Portland II cement, and vinyl ester-styree are not particularly effective in immobilizing diatomaceous earth, although UF showed some retention of cesium and VES some retention of cobalt.

D.5 WASTE FORM INTEGRITY

The ability of a waste form to immobilize radioactivity is decreased if it has a tendency to crumble or fracture. Such tendencies increase leachability by increasing surface area, decrease the stability of the disposal cell causing subsidence and increased water infiltration, and can lead to suspension of the deteriorated waste form by wind during transportation accidents or should the waste form be unearthed at some point after disposal. After disposal the wastes are subjected to static loading which can cause compressive failure. Waste containers are subject to handling mishaps and are frequently dropped into disposal trenches which create the potential for fracturing. Compressive strengths of Portland II cement, urea-formaldehyde, vinyl ester-styrene, and polyester waste forms are given in Table D-14.

The data for bead resin solidified in Portland cement Type II provides a convenient standard for comparison of waste form strengths. These samples showed extensive cracking and swelling. Their integrity was so low that they could not survive handling during leach testing. On this basis, waste forms with compressive strengths less than 50 psi under the test conditions are considered too fragile to arrive at the disposal site in one piece. As a consequence, any reduction in leachability which might have been realized from increasing the V/S ratio is nullified.

The data given in Table D-14 for Portland II cement shows the compressive strengths are greatest for the lowest waste/binder ratios. It should be noted that free-standing water is present in BWR regenerative/cement waste forms above a waste/binder ratio of 0.80 (see Table D-4) and in PWR regenerative/cement above a waste/binder ratio of 1.0 (see Table D-3). Although compressive strengths are not available, the data presented in Table D-15 shows that the integrity of resin/Portland II cement waste forms can be improved by using low resin/cement ratios and especially by using low water/cement ratios.

TABLE D-14 . Compressive Strengths of Simulated Waste Forms[a]

Waste and Binder[b]	pH	Waste/Binder Weight Ratio	Compressive Strength (psi)	Ref.
1. Portland II Cement				
(a) Bead resin slurry [1a]	10	2.0	48	7,14
	10	2.4	68	7,14
	10	2.6	41	7,14
(b) Powdered resin slurry [2a(1)]	10	1.8	48	7,14
	10	2.0	45	7,14
(c) Diatomaceous earth dewatered [2b(2)]	7	1.6	482	7,14
	7	2.0	420	7,14
	7	2.4	103	7,14
(d) BWR regenerative waste [3a]	6	0.6	3270	7,14
	6	1.2	580	7,14
	6	1.7	177	7,14
(e) PWR regenerative waste [3b]	3	0.6	3160	7,14
	3	1.2	72	7,14
	3	1.7	40	7,14
(f) Boric acid waste [3c]	3	c	c	7,14
2. Urea-Formaldehyde				
(a) Bead resin slurry [1a]	7	2.6	78	7,14
(b) Powdered resin slurry [2a(1)]	7	2.0	384	7,14
(c) Diatomaceous earth dewatered [2b(2)]	7	2.0	387	7,14
(d) BWR regenerative waste [3a]	6	1.2	67	7,14
(e) PWR regenerative waste [3b]	3	1.2	61	7,14

TABLE D-14 (cont'd.)

Waste and Binder[b]	pH	Waste/Binder Weight Ratio	Compressive Strength (psi)	Ref.
3. Vinyl Ester-Styrene[d]				
(a) Bead resin dewatered (90 wt % resin)	7	2.4	1761	33,40
(b) Diatomaceous earth dewatered (90 wt % DE)	7	1.8	4210	33,40
(c) Aqueous sodium sulfate (5 wt % Na_2SO_4)	11	1.8	3952	33,40
(d) Aqueous boric acid (5 wt % H_3BO_3)	3	1.8	2790	33,40
(e) Dow NS-1 decontamination solvent	e	1.8	3312	33,40
(f) Anhydrous sodium sulfate	NA[f]	2.5	6130	42
(g) Anhydrous sodium & lithium borates	NA	2.0	5425	42
(h) Anhydrous sodium sulfate & sodium & lithium borates	NA	2.0	7350	42
4. Polyester[g]				
Aqueous sodium sulfate (24 wt % Na_2SO_4)	e	1.0	413	35
	e	2.0	310	35
	e	2.3	186	35

(a) Measured after 28 days curing unless specified otherwise. Cement and UF samples prepared in accordance with ASTM method C192-69.
(b) Numbers in brackets refer to waste formulation given in Table D-2.
(c) Waste/binder ratio not specified. Samples did not solidify within 28 days.
(d) Minimum curing time is 24 hours. Exact time not specified.
(e) pH not specified.
(f) Not applicable.
(g) Cure time not specified.

TABLE D-15

Heights (cm) of 4.5 cm Diameter Bead Resin/ Portland II Cement Waste Forms After 28 Days Curing[a]

Water/Cement Weight Ratio	0.4	0.6	0.8	1.0	1.2	1.4	1.6	1.8
0.3	6.4	6.9	7.3	7.8	7.9	8.2	8.6	8.8
0.4	6.7	7.3	7.7	7.9	8.2	8.3	8.6	9.0
0.5	NE[b]	7.6	7.9	8.2	8.3	8.7	8.8	8.9
0.6	NE	7.6	7.9	8.1	8.3	8.7	9.1	9.7
0.7	NE	7.5	7.8	8.3	8.5	--	9.7	>9.7
0.8	NE	NE	7.8	8.2	8.6	>9.7	9.7	>9.7
0.9	NE	NE	NE	8.0	8.3	9.3	>9.7	>9.7
1.0	NE	NE	NE	6.9	7.6	8.8	9.0	>9.7

(a) Samples above the solid line exhibited little or no swelling. Those below the line exhibited considerable swelling, cracking or splitting.

(b) NE=Not examined.

Source: Reference 19.

The compressive strengths of the vinyl ester-styrene studied are all high. It would appear that waste forms having good integrity are readily obtained with VES but require careful control of waste/binder ratios for cement. Use of lower waste/binder ratios could improve the compressive strengths of urea-formaldehyde waste forms. Compressive strengths of bitumen waste forms could not be measured at ambient temperatures since bitumen deforms rather than fractures.(14) A tensile strength (ASTM D-638) ranging from 3100 to 4200 psi was reported for anhydrous sodium sulfate in epoxy at a waste/binder weight ratio of 2.3.

Unnotched Izod impact strengths for waste forms containing only water were measured by ASTM method D256-73 (Part C) and are given in Table D-16. Again vinyl ester-styrene shows the highest integrity. Evaporative water loss from UF is parallelled by a decrease in impact strength of about 50 percent. Vinyl ester-styrene also loses water when exposed to air, but the effect on impact strength has not been determined. Water loss from VES is considerably less than that from UF (see Section D.6).

An impact strength (ASTM D-3029) of 10.8 in-lb/in was measured for a sodium sulfate/epoxy waste form with a waste/binder ratio of 2.3.(26)

The size distribution of fragments produced by impact loading of waste form is another guide to their effectiveness in immobilizing radioactivity. The distribution of fragments of cement and urea-formaldehyde waste forms produced by a single impact loading of 3.2 kg-m are given in Table D-17. The weight percent of fragments is a linear function of particle size in the range of 30 um to about 10,000 um for the waste forms studied. The percent of fragments less than 1000 um may seem low, however, they were produced by a single impact loading. These fragments are all highly leachable, and may become airborne if exposed to strong winds.

TABLE D-16

Unnotched Izod Impact Strengths of Portland II Cement, Urea-Formaldehyde and Vinyl Ester-Styrene Containing Water

Binder	Water/Binder Weight Ratio	% Initial Weight	Impact Strength[a] in-lb/in.
Portland II Cement [b]	0.20		3.3 ± 0.5
	0.30		4.9 ± 2.0
	0.40		5.0 ± 1.6
	0.50		4.4 ± 0.9
	0.60		4.2 ± 1.2
Urea Formaldehyde[c]	1.0		3.0 ± 0.9
	2.0		1.8 ± 0.2
	3.0		1.2 ± 0.1
After drying in air[d]		% Initial Weight	
	1.0	31	1.1 ± 0.3
	2.0	22	0.93 ± 0.07
	3.0	16	0.68 ± 0.04
Vinyl Ester-Styrene[c]	1.0		8.6 ± 1.1
	1.5		6.8 ± 1.4
	2.0		5.1 ± 0.6

(a) Values given are mean ± standard deviation of ten measurements.

(b) Cured for 7 days.

(c) Cured for 24 hours.

(d) Exposed to ambient air for two days.

Source: Reference 14

TABLE D-17

Fragmentation of Portland Cement and Urea-Formaldehyde
Waste Forms Under an Impact Loading of 3.2 kg-m

Waste and Binder[a]	Water/Binder Weight Ratio	Weight Percent Fragments Less Than	
		100 um	1000 um
1. Portland II Cement[b]			
(a) Bead resin slurry [1a]	1.5	1.2	57
(b) Diatomaceous earth slurry [2b(1)]	1.2	0.20	1.8
	1.6	0.34	2.7
(c) BWR regenerative waste [3a]	2.0	0.80	5.0
	1.0	0.20	1.7
(d) Water	0.5	0.32	2.7
2. Portland III Cement[b]			
(a) Boric acid waste [3c]	0.5	0.05	1.2
3. Urea-Formaldehyde[c]			
(a) Diatomaceous earth dewatered [2b(2)]	1.6	0.26	2.5
	2.0	0.40	5.5
	2.4	1.0	7.5

(a) Numbers in brackets refer to waste formulations given in Table D-2.
(b) Cure time 102 to 122 days.
(c) Cure time 202 to 206 days.

Sources: Cement, Reference 13; Urea Formaldehyde, Reference 18.

D.6 THERMAL PROPERTIES

The flammability of a waste form and its ability to withstand elevated temperatures without decomposition or excessive loss of waste liquids are important in assessing the ability of the waste form to immobilize the radioactivity under accidental fire conditions. Obviously, a waste form which sustains burning or decomposes when heated will release a fraction of its radioactivity. Waste forms which contain liquids and which rapidly release the liquids when heated also present a hazard since radioactivity may be entrained in the vapor or deposited on the surface of the waste form. This section considers the flammability and thermogravimetric behavior of waste forms. The data presented here, combined with the properties of the untreated wastes presented in Chapter 2 of this report are used to assign values to the flammability indices discussed in Chapter 5 of this report.

D.6.1 Flammability

The discussion on flammability is restricted to final waste forms and excludes consideration of the separate chemicals combined to produce the final waste forms. As defined here, a flammable or partially flammable waste form is one which burns, chars, melts or decomposes on exposure to a temperature of approximately 1000°F. This imprecise definition is required by variations in test conditions.

Portland cement is non-flammable at this temperature. No data is available for polyester or epoxy waste forms but it is assumed that they, like similar materials used in other industries, will char and lose weight on heating. The flammability of polyester is expected to resemble that of vinyl ester-styrene.

Flammabilities of selected waste forms as determined by ASTM Standard D635-74 are presented in Table D-18. This test involves a 30 second exposure of small samples to a bunsen burner flame. As seen in the

TABLE D-18

Flammability of Selected Waste Formulations (ASTM Standard D 635-74)

Specimen	Weight loss percent	Weight of ash grams	Extent of burning mm	Burning time sec	Comments
Urea-formaldehyde	8.5	0.01	0.0	0.0	Minor surface charring
UF-bead resin	7.2	0.04	0.0	0.0	Minor surface charring
UF-powdered resin	6.4	0.00	0.0	1.3	Minor surface charring
Portland type II neat cement	1.0	0.00	0.0	0.0	----
Pioneer 221 asphalt-bead resin	12.1	1.3	4.0	-	Specimens melted out of position of applied flame. "Ash" consists of drippings

Source: Reference 14

TABLE D-19

Cleveland Open Cup Flash and Flame Point Determinations for Bitumen Waste Forms Made with Pioneer 221 Asphalt (ASTM Standard D-92-72)

Waste Type	Flash Point, °F	Flame Point, °F
None	610 ± 2	668 ± 4
Water	630 ± 10	665 ± 3
BWR chemical regenerative waste	625 ± 5	670 ± 10
Boric acid waste	Determinations could not be made because of specimen foaming.	

Source Reference 14

table, all samples, except cement, charred and all lost weight. The bitumen sample melted and the UF samples showed evidence of combustion. It has been reported that UF waste forms are self-extinguishing since they contain water; however, studies by the same researchers show that this water is lost rapidly on exposure to heat or to ambient air.(14)

In addition to melting, bitumen waste forms will also ignite (see Table D-19). Many grades of bitumen, some with higher flash and flame points, are readily available. Bitumen can also react vigorously, if not violently, with strong oxidizers such as nitrates, peroxides, and permanganates. Results of flammability testing of vinyl ester-styrene are summarized in Table D-20. All waste forms charred and lost weight but none were reported to support combustion.(40)

D.6.2 Decomposition and Weight Loss Due to Evaporation

Testing of urea-formaldehyde and vinyl ester-styrene waste forms shows that both lose weight on exposure to ambient air (14) (see Section D.5). The weight loss of UF samples is about twice that of VES samples. It is reasonable to assume that this outward flow of water from the waste form will result in deposition of evaporated radioactive salts on the surfaces of these waste forms.

Weight losses of similar magnitude have been observed to occur over much shorter times during flammability testing (see Tables D-18 and D-20). Under these conditions, the evaporated water may contain entrained radioactivity and any deposited salts may be dislodged or volatized by the hot turbulent gases.

Thermogravimetric studies of UF and VES waste forms show that the total weight loss due to evaporation can amount to 80 percent of the initial sample weight(5,6,14). These same studies showed that thermal decomposition begins at about 290°C for UF and about 350°C for VES.

TABLE D-20

Flammability of Vinyl Ester-Styrene Waste Forms

A. Ten Minute Exposure in 1000°F Muffle Furnace

Waste	Average Weight Loss (%)	Comments
Bead Resin	22.8	Charred
BWR regenerative waste	27.1	Charred
Boric acid waste	27.8	Charred
Diatomaceous Earth	27.3	Charred
Dow decon solvent	27.5	Charred

B. Exposure (7 minutes) to One Gallon of Burning #2 Fuel Oil

Waste	Average Weight Loss (%)	Comments
Bead Resin	8.1	Charred
BWR regenerative waste	8.4	Charred
Boric acid waste	6.3	Charred
Dow decon solvent	9.9	Charred

Source: Reference 40

D.7 CORROSION OF MILD STEEL

Fifty-five gallon drums made from mild steel are frequently used as containers for solidified low-level wastes. These containers can be corroded externally by water in burial trenches and internally by the waste form. External corrosion of containers by the burial environment is discussed in BNL-NUREG-50774(8) and NUREG/CR-0619(14) and is not considered here. Results of BNL(10,14) testing of mild steel corrosion by urea-formaldehyde and cement waste forms are summarized here. This information is not used in assigning waste form behavior indices discussed in Chapter 5, since no credit is taken for container performance but is presented for completness.

Mild steel containers can be corroded by the waste form itself, by free-standing water, and by vapors enclosed in the container. Corrosion may be spread over the exposed surface (uniform) or be highly localized (pitting). Pitting corrosion is more serious since it causes more rapid loss of container integrity than uniform corrosion.

Corrosion by solidified waste is an electrochemical process in which the waste container functions as an anode and is corroded by the loss of metal cations to the electrolyte (free-standing water or other moisture in the container). Corrosion generates hydrogen gas in acidic solutions and consumes oxygen in neutral and basic solutions. Corrosion rates frequently decrease with time as a protective layer of corrosion products builds up.

The results of corrosion testing of mild steels by selected waste forms are summarized in Tables D-21 through D-23. Since in several cases the amount of data available does not allow accurate determination of corrosion rates over long time periods, the mean and standard deviation of the available data were calculated for each type of sample. The mean values are indicative of relative corrosion rates while the standard deviation suggests the change in corrosion rate as a function of time.

TABLE D-21 Corrosion of Mild Steels Totally Immersed in Selected Wastes

	Waste Only		Waste and Sodium Bisulfate		Waste and UF	
	Rate[a] (mpy)	Type[b]	Rate (mpy)	Type	Rate (mpy)	Type
A. Mild Steel						
Powdered Resin	0.88 ± 0.28	U	4.6 ± 2.0	U	0.95 ± 0.04	U
BWR Regenerative Waste	0.42 ± 0.15	U	15.9 ± 3.6	U	1.38 ± 0.40	U
Boric Acid Waste	4.3 ± 1.3	U	10 ± 13	U	3.8 ± 1.2	U/P
B. Drum Steel[c]						
Powdered Resin			2.52 ± 0.47	U	2.58 ± 0.60	U
BWR Regenerative Waste			3.09 ± 0.43	U[d]	2.35 ± 0.22	U
Boric Acid Waste			5.8 ± 2.7	U[d]	3.8 ± 1.0	U

(a) Rate (in mils per year) is expressed as average ± standard deviation of three to seven measurements.
(b) U = uniform, P = pitting
(c) Mild steel with zinc coating
(d) Some evidence of non-uniform corrosion

Source: Reference 14

TABLE D-22 Corrosion of Mild Steels In The Vapor Phase of Urea-Formaldehyde Waste Forms

Waste	Rate[a] (mpy)	Type[b]
A. Mild Steel		
Powdered Resin	12.7 ± 7.1	U
BWR Regenerative Waste	8.2 ± 3.4	U
Boric Acid Waste	7.6 ± 4.7	U
B. Drum Steel[c]		
Powdered Resin	6.75 ± 0.6	P
BWR Regenerative Waste	9.1 ± 5.2	P
Boric Acid Waste	4.0 ± 1.6	P

(a) Mean ± standard deviation of three measurments (mils per year)

(b) U = uniform, P = pitting

(c) Mild steel with zinc coating

Source: Reference 14

TABLE D-23 Corrosion of Mild Steels Partially Immersed in Selected Wastes

	Urea-Formaldehyde			Portland II Cement		
	Rate[a]	Corrosion Type[b]		Rate[a]	Corrosion Type[b]	
Waste	(mpy)	Uniform	Pitting	(mpy)	Uniform	Pitting
Mild Steel						
Powdered Resin	7.1±2.9	B	A	1.46±0.90	A	-
BWR Regenerative	6.8±2.6	B	A	0.41±0.36	A	A
Boric Acid Waste	7.54±0.21	B	A	0.22±0.11	A	A
Drum Steel						
Powdered Resin	2.95±0.30	B	A			
BWR Regenerative	1.64±0.58	B	A			
Boric Acid Waste	2.74±0.97	B	A			

(a) Mean ± standard deviation of three to four measurements (mils per year)

(b) A = above solid/vapor interface, B = below interface

Source: Reference 14

As expected on the basis of the acidity of liquids contained in urea-formaldehyde, these waste forms are more corrosive to mild steel than the corresponding cement waste forms (Table D-23). Comparison of results given in Tables D-21 and D-23 confirms those given in Table D-22, and shows that most of the corrosivity of UF waste forms is due to the vapor phase. The data in these two tables also shows that the zinc coating on samples cut from 55 gallon drums does provide some protection against corrosion. In most cases the coating was removed within about 50 days, indicating that its effectiveness is short lived(14).

The ion exchange resins used in these studies were expended with sodium chloride before testing(14). Ion exchange resins which are not expended are capable of generating acids on contact with groundwater and accelerated corrosion is expected.

Gas generation by corrosion was not measured in these studies; however, an estimate can be made based on the immersion corrosion rate of about 16 mpy (Table D-21). If it is assumed that metallic iron is being corroded, then hydrogen is produced at the rate of 0.127 moles/year. If this gas is confined in 5% of the drum volume (about 10 liters) at 25°C, the expected pressure due to hydrogen is only 0.31 atmospheres. This low pressure combined with the ability of hydrogen to diffuse through metals indicates that container pressurization will not occur as the result of corrosion.

D.8 RADIATION EFFECTS

Decay of the radioactive species contained in waste forms generates gases (mainly hydrogen) and can also cause changes in their mechanical strengths and leachabilities.

D.8.1 Radiolytic Gas Generation

Radiolytic gas generation is usually quantified by G values which express the number of molecules of gas produced per 100 eV of absorbed energy. The amount of energy absorbed varies with the type of radiation and the average atomic number of the medium. G values may vary with the intensity and type of ionizing radiation.

Table D-24 presents G values for total gas production and for hydrogen production from "neat" waste binders. Neat means that the binder contains no waste (bitumen) or only water (UF and cement). G values are highest for water/UF and, decrease with increasing dose for both UF and cement. These relatively large G values are due to radiolysis of water rather than the waste binder. It should be noted that the UF sample contains four times as much water as the cement sample. Decreasing G values with increasing dose have been observed for radiolytic production of hydrogen from water(42,43) and are attributed to recombination of radiolysis products (e.g., H_2 + O_2) to form water.

The G values for bitumen increase with increasing dose and are initially lower than those measured for UF and cement. The absence of water and the ability of the large aromatic malthenes and asphaltenes to absorb radiation without decompositon are responsible for these observations.

The range of doses in Roentgen (10^5 to 10^9 R) covered in Table D-24 corresponds roughly to the estimated cumulative 1000 year self-dose in rads for typical LWR wastes. These cumulative self-doses range

TABLE D-24 G Values for Total Gas Production and for Hydrogen Production From Neat Waste Binders

(Co-60 dose rate = 4.8×10^6 R/hr)

	10^5 R	10^6 R	10^7 R	10^8 R	10^9 R
	G(total gas), molecules/100 eV				
Portland Type II Neat Cement (w/c = 0.5)	1.5	0.65	0.32	0.24	0.16
Urea-formaldehyde (w/UF = 2.0 by vol)	21	8.6	2.8	2.0	0.36
Pioneer 221 asphalt	NT*	NT	0.099	0.029	0.43
	$G(H_2)$, molecules/100 eV				
Portland Type II Neat Cement (w/c = 0.5)	0.35	0.24	0.22	0.17	0.11
Urea-formaldehyde (w/UF = 2.0 by vol)	4.8	6.5	2.4	1.3	0.24
Pioneer 221 asphalt	NT	NT	0.069	0.028	0.41

(*) Not tested.

Source: Reference 14.

from about 10^4 rads for waste containing a total activity 0.01 Ci/ft^3 to about 10^9 rads for wastes containing 100 Ci/ft^3.(14)

The amount of radiolytic gas generated in actual waste forms is sensitive to the chemical nature of the waste. As indicated by Table D-24, waste forms containing large amounts of water will produce large amounts of gas. The presence of sulfates and nitrates also tends to increase G values. Although radiolysis of simulated solidified waste has not been extensively studied, some data is available for unsolidified wastes. The data presented in Table D-25 was obtained for transuranic (TRU) contaminated defense waste containing an average of 0.19 Ci/m^3 of alpha activity, primarily weapons grade plutonium.

For estimation of any pressurization of the waste container that may occur as a result of radiolytic gas generation, consider a 55 gallon drum containing 10 Ci/m^3 of cobalt-60, a relatively high activity waste, solidified in UF. Assuming an average photon energy of 1.25 MeV for cobalt-60 and a conservative density of 1.0 g/cm^3 for the solidified waste, a dose of 1.34×10^5 rads/yr is obtained (neglecting decay). Using a rad/R conversion of 0.947(14) yields an annual dose of 1.41×10^5R and indicates a G value of 21 is appropriate (Table D-24). On this basis approximately 1.2 moles of gas will be generated in a year. If 10 liters (about 5%) of the drum volume is available to the gas, it will exert a pressure of about 2.6 atm (38 psi) at 0° C. Under similar conditions, the cement waste form (Table D-24) would generate a pressure of 0.18 atm (2.6 psi). Bitumen G values were not measured at this low dose.

Hydrostatic testing of DOT 17C and 17H open head steel drums revealed pressure leakage at an average pressure of 16 psi and 1.3 psi, respectively.(5,14) Pressure leakage occurred at 71 psi and 12 psi for closed head DOT 17C and 17H drums, respectively. Thus, although the above estimations of pressurization are based on some rather gross approximations, they do indicate that a potential for drum failure

TABLE D-25 Range of G Values for Total Gas Production in Unsolidified TRU Contaminated Wastes

Waste Type[a]	Range of G (total gas)
Cellulosics[b]	0.31-2.9
Ion Exchange Resins	0.11[c]
Polyethylene	0.73-1.9
Polyvinyl Chloride	0.43-11
Rubbers (unspecified)	0.37[c]
Pump Oil	1.3-3.1
Plexiglass	19[c]
Octane	4.5
Asphalt	0.2-1.0

(a) Waste contain an average of 0.19 Ci/m^3 of alpha activity.

(b) Paper and cotton.

(c) Single value given.

Source: Reference 44.

exists when high activity wastes containing large amounts of water are present.

D.8.2 Leachability

Little data is available relating leachabilities to absorbed dose of waste forms. One such study investigated the leachability of slurried diatomaceous earth solidified in Portland II cement (W/B=1.6) and in urea formaldehyde (W/B=2.0).(13) Leaching of cesium and strontium from the cement waste form was unaffected by doses of up to 10^8 rads. The cumulative fraction leached x (V/S) for strontium for the UF waste form (V/S = 0.508 cm) gradually increased from 0.25 cm for the control to 0.36 cm for a dose of 10^8 rads. The effect was more dramatic for cesium. The cumulative fraction x (V/S) gradually increased from 0.018 cm for the control to 0.024 cm for a dose of 10^7 but jumped to 0.078 cm for a dose of 10^8. All samples were leached for about 20 days.

The difference in leachabilities of the cement and UF waste forms as a function of absorbed dose may be related to differences in G values discussed in the previous section. Leachability of the UF waste form showed a dependence on dose rate. At a total absorbed dose of 10^7 rads, the leachabilities of both cesium and strontium from the diatomaceous earth/urea-formaldehyde waste form were found to increase as the dose rate was increased from 1.0×10^4 R/hr to 1.3×10^6 R/hr.(16)

D.9 CHEMICAL AND BIOLOGICAL DEGRADATION

Chemical and biological degradation can affect solidified wastes in several ways. The integrity of the waste form is obviously reduced by such degradation. This reduction of integrity, and the ensuing trench instability, leads to an increase in leachability (due to changes in geometry which expose more surface area) and in most cases, is accompanied by gas generation.

Decomposition gas consists primarily of hydrogen (H_2), carbon dioxide (CO_2), carbon monoxide (CO), water vapor (H_2O), methane (CH_4), oxygen (O_2), and oxides of nitrogen (NO_x) and sulfur (SO_x). The principal radioisotopes associated with these decomposition gases are expected to be tritium and carbon-14, since the radioactive isotopes of oxygen, sulfur and nitrogen likely to be present are short-lived. It has been estimated that 0.1 to 2.0 Ci/yr of tritiated methane (CH_3T) is released from all the burial trenches at West Valley, New York.(45) However, container failure due to pressurization by these gases can cause an additional increase in leachability by allowing larger amounts of water to contact the waste form. Several of the gases also represent potential fire and/or explosion hazards.

Degradation of waste forms can also increase the mobility of any leached radioactive species by chelation. Carboxylic acids are common degradation products and share to varying degrees the chelating abilities of EDTA and other carboxylic acids used as decontamination chemicals. Detailed discussions of the effects of carboxylic acids on chelation are beyond the scope of this report.

Thermal degradation of waste form is discussed in Section D.5. Studies of unsolidified organic wastes indicate that the rate of thermal degradation is very slow below 70°C.(44) Chemical degradation of waste forms has not been extensively studied. Most of the available information on chemical degradation is presented from the viewpoint of waste-binder compatability. In general, organic waste binders (UF, VES, polyester, bitumen, and epoxy) react chemically with (are degraded by) oxidizers (nitrates, permanganates) and strongly acidic wastes (boric acid, UF catalysts).

The available data on degradation of all types indicates that biological (bacterial) degradation is the most important type. Most of the available information relates to bacterial degradation of waste materials(44) rather than waste forms or binders. Although this information is not directly relevant to the characteristics of waste forms and binders, it is presented here for convenience.

A very large number of bacterial species which are present in air and soil are capable of metabolizing both organic and inorganic components of waste forms. These bacteria may be aerobic or anaerobic, sulfate-reducing, denitrifying, or methanogenic. As indicated in Tables D-26 and D-27, these bacteria thrive in trench leachate. Species identified in these leachate samples include Bacillus, Pseudomonas, Citrobacter, and Clostridium.

These and other species metabolize organics in trench leachate and in wet wastes. Table D-28 shows that these organics may be both consumed and produced by the bacteria. The quantity of methane produced and its carbon-14 and tritium content are influenced by the composition of the overall gas environment (Table D-29).

The organics on which the bacteria feed may be constituents of the original waste, produced by other bacteria (Table D-28), or leached from the waste binder (Table D-30). On the basis of the results in Table D-30, UF is expected to support a much larger bacterial population than the other waste binders. Since UF waste forms frequently contain free-standing water and the bacteria are present in air and in waste materials, bacterial growth is a potential cause of rather than a result of container failure. Bacterial growth in other waste forms

TABLE D-26

Population of Bacteria in Leachate Samples From Low-Level Radioactive Waste Disposal Sites

Sample	Collection Date	Aerobic CFU/mL[a]	Anaerobic CFU/mL[a]
Maxey Flats			
Trench 2	7/77	1.2×10^3	1.0×10^2
Trench 26	7/77	4.7×10^3	4.1×10^2
Trench 32b	7/77	4.8×10^4	1.2×10^4
Trench 19S	5/78	2.2×10^2	3.2×10^2
Well UB1-A	5/78	3.4×10^3	N.D.[c]
West Valley			
Trench 3	10/78	5.0×10^4	4.0×10^3
Trench 4	10/78	2.3×10^3	3.3×10^3
Trench 5	10/78	1.6×10^3	3.5×10^2
Trench 8	10/78	1.4×10^3	7.6×10^2
Trench 9	10/78	5.0×10^2	7.3×10^3
Barnwell			
Trench 8D2	3/79	2.0×10^8	1.0×10^4
Trench 6D1	3/79	3.3×10^3	1.3×10^2
Trench 25/21-D1	3/79	3.5×10^4	2.2×10^3
Trench 3D1	3/79	1.5×10^5	1.2×10^3
Sheffield			
Trench 14A	4/79	1.7×10^5	4.4×10^4
Trench 18	4/79	7.1×10^2	6.9×10^1
Well 52b	4/79	6.3×10^2	4.2×10^2

(a) Colony forming units per milliliter.
(b) Sample analyzed 7 days after collection.
(c) N.D. - not detected.

Source: Reference 45.

TABLE D-27

Population of Denitrifying, Sulfate Reducing, and Methanogenic Bacteria in Leachate Samples From Low-Level Radioactive Waste Disposal Sites

Sample	Collection Date	Dentrifiers MPN/mL[a]	Sulfate Reducers MPN/mL[a]	Methanogens MPN/mL[a]
Maxey Flats				
Trench 19S	5/78	3.3×10^1	4.0×10^0	4.9×10^0
Well UB1-A	5/78	4.6×10^2	N.D.[b]	1.0×10^0
West Valley				
Trench 3	10/78	1.3×10^4	7.0×10^1	2.3×10^1
Trench 4	10/78	2.3×10^3	4.9×10^2	1.7×10^0
Trench 5	10/78	3.3×10^2	1.1×10^1	N.D.
Trench 8	10/78	7.9×10^2	1.7×10^2	1.0×10^0
Trench 9	10/78	1.3×10^2	3.5×10^2	4.5×10^0
Barnwell				
Trench 8D2	3/79	2.3×10^5	1.1×10^0	0.8×10^0
Trench 6D1	3/79	1.1×10^3	N.D.	N.D.
Trench 25/21-D1	3/79	1.3×10^4	1.3×10^2	0.2×10^0
Trench 3D1	3/79	5.4×10^4	N.D.	N.D.
Sheffield				
Trench 14A	4/79	2.4×10^5	N.D.	0.2×10^0
Trench 18	4/79	9.5×10^2	4.9×10^1	N.D.
Well 525	4/79	1.7×10^3	2.3×10^0	N.D.

(a) Most probable number per milliliter.
(b) N.D. - not detected.

Source. Reference 45.

TABLE D-28

Anaerobic Degradation of Organic Compounds Present in Maxey Flats Trench 26 Leachate Sample by a Mixed Culture Bacteria

Compound	Initial Concentration (mg/L)	Change in Concentration (%)
2-Methylpropionic acid	3.5	+ 31
2-Methylbutanoic acid	19	+ 16
Valeric acid	4.6	- 100
C_6 acid (unidentified)[a]	N.Q.[b]	+ 5.8
C_6 acid (unidentified)[a]	N.Q.	+ 3.6
Hexanoic acid	1.8	- 100
2-Methylhexanoic acid	1.3	+ 8
Cresol	1.8	+ 11
C_8 acid (unidentified)[a]	N.Q.	- 4
C_8 acid (unidentified)[a]	N.Q.	- 0.5
Benzoic acid	1.1	- 0
Phenylacetic acid	1.4	- 7
Phenylpropionic acid	1.2	- 100
α-Terpineol	0.16	- 6

(a) Percent change in concentration was determined on the basis of the ratio of the compound with the internal standard.

(b) N.Q. = Not Quantified.

Source: Reference 45.

TABLE D-29

Microbial Production of $^{14}CH_4$ and CH_3T From Maxey Flats Trench 19S Leachate Sample

Sample[a]	Methane Produced[b] (nmol)	Total Activity (pCi)[c]	
		$^{14}CH_4$	CH_3T
Control (10% formaldehyde)	980	0.5	0.03
Inoculated 85% (N_2+ 10% CO_2+5% H_2)	18,000	0.59	1.0
Inoculated 80% (CO_2+20% H_2)	68,000	12	57

(a) 30 mL of trench leachate in 60-mL bottle

(b) Time required for production not specified.

(c) Total initial activity not specified.

Source: Reference 45.

TABLE D-30

Analysis for Organic Carbon Removed From Solidification Matrix Materials by Leaching in 300 ml Distilled Water for Ten Days

	Portland Type II Cement	Urea Formaldehyde	Pioneer 221 Bitumen	Vinyl Ester-styrene
Specimen mass, g	296.0	218.3	103.1	198.2
Ratio of specimen volume to geometric surface area, cm	0.98	1.1	0.75	1.1
Leachate content				
a. Total C, ppm	14.8	9540	3.4	34.2
b. Inorganic C, ppm	14.8	39	<2.0	<2.0
c. Organic C, ppm	0.0	9500	3.4	34.2
d. Organic C, g	0.0	2.85	0.0010	0.010

Source: Reference 14.

which contain no free-standing water and produce leachates with low organic content is less significant before the container fails.

The data presented in Table D-31 is relevant to decomposition of trash, plywood boxes (frequently used as disposal containers), and bitumen. Carbon dioxide (CO_2) is expected as a major product of aerobic bacteria while anerobic bacteria produce larger amounts of methane. The fact that little or no methane was observed in these studies was attributable to loss of anerobic conditions and/or lack of methanogenic bacteria(44) and calls attention to the sensitivity of the composition of the decomposition gases to local conditions.

Table D-31 shows that water-saturated wastes generally, but not always, produce more carbon dioxide at 25°C than those with less water. This trend becomes less well-defined at 70°C. These results indicate that bacterial gas production is the most significant cause of container pressurization. A comparison with the estimated gas production of 2.6 moles/yr from radiolysis of UF (Section D.8) indicates that the potential for loss of container integrity due to pressurization by decomposition gases is substantial.

TABLE D-31

Net Bacterial CO_2 Gas Generation for Various Environments (moles/year per container)

Environment	Aerobic 25°C	Aerobic 40°C	Aerobic 70°C	Anaerobic 25°C	Anaerobic 40°C	Anaerobic 70°C
LASL Composite[a] (51.4 kg/drum)						
Water, Saturated	1.6	1.8	3.1	4.2	0.6	3.4
Brine	(0)	5.2	5.5	1.2	7.8	(0)
Nutrient	3.1	1.5	(0)	3.6	1.4	7.3
Water, 1%	(0)	1.3	4.2	0.3	2.6	2.5
Sawdust-Plywood (153 kg box)						
Water, Saturated	14.3	10.3	14.0	26.2	4.7	23.0
Brine	(0)	11.6	18.6	11.8	(25)	6.8
Nutrient	13.5	2.8	18.0	12.2	6.9	12.6
Water, 1%	3.0	9.7	2.8	11.3	17.3	17.3
Sawdust-Plywood (18 kg)						
Water, Saturated	1.7	1.2	1.7	3.1	0.6	2.7
Brine	(0)	1.4	2.2	1.4	(3)	0.8
Nutrient	1.6	0.3	2.1	1.4	0.8	1.5
Water, 1%	0.4	1.1	0.3	1.3	2.0	2.0
Asphalt (135 kg/drum)						
Water, Saturated	2.1	(0)	(0)	0.6	1.9	1.9
Brine	2.6	(0)	8.4	(0)	0.9	1.6
Nutrient	3.7	1.0	0.9	4.3	0.3	(0)
Water, 1%	0.01	0.9	0.03	4.8	0.9	(0)

(a) Mixture of cellulosics and rubbers (LASL = Los Alamos Scientific Laboratories).

Source: Reference 44.

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This document is prepared in three volumes and provides part of the technical support to the draft environmental impact statement (NUREG-0782) on a proposed regulation, 10CFR Part 61, setting forth licensing requirements for land disposal of low level radioactive waste. Volume 1 is a summary and analysis of the history of low level waste disposal at both commercial and government disposal facilities. Volume 2 provides a summary of low level waste volumes and characteristics as projected to the year 2000, in addition to characterizing treatment options for this waste. Volume 3 provides a methodology for analyzing the impacts of handling and disposing of low level waste based upon consideration of alternative waste forms, disposal facility design and operating practices, disposal facility environmental characteristics, and institutional control considerations.

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