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A WASTE INVENTORY REPORT FOR REACTOR AND FUEL-FABRICATION FACILITY WASTES

TECHNICAL REPORT

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A WASTE INVENTORY REPORT FOR REACTOR AND FUEL-FABRICATION FACILITY WASTES

PREPARED FOR

UNITED STATES DEPARTMENT OF ENERGY
OFFICE OF WASTE MANAGEMENT
AND ITS PRIME CONTRACTOR

BATTELLE MEMORIAL INSTITUTE
OFFICE OF NUCLEAR WASTE ISOLATION
PROJECT MANAGEMENT DIVISION

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TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF FIGURES

LIST OF FIGURES (Continued)

INTRODUCTION

The Office of Waste Isolation was established by the U.S. Energy Research and Development Administration (ERDA) in 1975. Pursuant to the Department of Energy Reorganization Act (P.C. 95-91), the functions and authority of ERDA were transferred to the U.S. Department of Energy (DOE). In 1978 the OWI was transferred from Oak Ridge National Laboratories to the Battelle Memorial Institute, Columbus, Ohio and renamed the Office of Nuclear Waste Isolation (ONWI). It was the responsibility of OWI, and now is the responsibility of ONWI, to manage the government program for geologic disposal of radioactive wastes in federal repositories. The repositories will receive all wastes containing radioactivity above the limit for safe surface (i.e. shallow-land) burial.

Commercial reactors and fuel-fabrication plants are producing wastes, some of which may be destined for geologic disposal when federal repositories become available. Spent fuel, both as a waste itself and as a source for reprocessing plant waste, is under investigation in other ONWI studies but few data are available on the wastes generated by reactors and fuel-fabrication plants.

The purpose of this Waste Inventory Study is to survey existing commercial nuclear facilities to determine actual wastes produced (with the exception of spent fuel and high-level waste), including the character, form and generation rates of the waste, and to develop projections of future waste production through the year 2000 based on assessments of waste treatment and packaging technology that is presently available or expected to be developed.

To obtain the data for the study, letters were sent to 35 operating light-water cooled nuclear power reactors (LWR) and 7 facilities involved in the fabrication of LWR fuel. These letters outlined the purpose, nature, and scope of the study. The letters were followed up with telephone calls to each facility superintendent and arrangements were made for one or more NUS personnel to visit the plant to obtain the required data. Copies of the questionnaire forms used in these interviews are included as appendices to this report. Information was collected on the equipment used in the systems that process and treat radioactive fluids including equipment installed but not used and equipment to be installed. Information was collected on the chemical, physical, and radioactive properties of concentrated liquids, filter sludge, spent resins, cartridge filters, compactible trash and noncompactible trash. This information included such things as annual volumes of waste, the total activity associated with this waste, the radionuclides present, density, type of solidification agent used, and the proportion of solidification agent to waste where this is applicable. All of the data used in this report are first-hand information from the files of each facility.

After collection the data were separated into three categories: pressurized water reactors (PWR), boiling water reactors (BWR), and fuel-fabrication facilities. The PWR category was further broken down into plants that process secondary system condensate and those that do not. The BWR category was separated into plants that process condensate through deep bed demineralizers versus plants that use precoat filters. Three of the seven fuel-fabrication facilities are complete processing plants that take enriched UF_6 and convert it to UO_2 pellets and manufacture fuel assemblies. Of the remaining four plants, two convert UF_6 to UO_2, which is then shipped to the remaining two plants for manufacturing into fuel assemblies. For the purposes of this study these facilities have been treated as five complete fuel-fabrication facilities.

The LWR data on waste volumes and waste activities were analyzed on the basis of annual generation rates per gigawatt of installed electrical capacity. Based on all the data collected, waste-volume and waste-activity generation rates were calculated to obtain an average of a number of operating plants. By selectively excluding specific data, which would not be considered to reoccur with reasonable frequency, typical waste-volume and waste-activity generation rates were also calculated. These values define the estimated waste volumes and activities of a single plant and are not used in the report beyond this point, but are presented to avoid the inaccurate use of the average values.

Also collected was a significant quantity of LWR data on the density of various waste forms, the types of shipping containers used, the volume in each container, the type of shielding used, radiation levels on contact or near a single container, and the method of solidification used, if any. For spent resins the type of resins used in the plant were tabulated as were the different types of filtering media used for precoat filters and filter demineralizers. Data on concentrated liquids include the chemicals concentrated and the average weight percent solids of the bottoms. These data were analyzed to determine average plant parameters for these quantities.

In order to provide a reasonable evaluation of the level of transuranic contamination in LWR wastes, the data collected in this survey were combined with two recent studies that looked specifically at this subject. Using this combined data base, the estimated concentrations of transuranic radionuclides in the various types of LWR wastes were calculated. The report also includes a listing of the relative levels of contamination in terms of gross activity concentration for various types of LWR waste.

For fuel-fabrication facilities the analysis was performed in terms of annual waste volumes and activities per metric ton of uranium (MTU) at the processing capacity. The data from the fuel-fabrication facilities were limited in detail yielding only gross estimates of volumes and activities, with some additional data on what items constitute combustible and noncombustible waste.

Once the chemical, physical, and radioactive characteristics of the waste were defined, projections of the waste volumes and activities through 2000 were calculated. Total LWR generating capacity through 1989 was based on plants now under construction. Projections from 1990 to 2000 are based on a 1977 report from Oak Ridge National Laboratory. The split in LWR capacity between BWRs and PWRs and the types of condensate polishing system each will have is based on operating plants and plants under construction. After the number and type of LWRs were defined, the total fuel-fabrication capacity necessary to support these reactors, including foreign sales, was estimated. The total annual waste volumes and activities from all fuel-fabrication facilities were added to LWR annual waste volumes and activities. These projections, combined with estimates of non-LWR fuel-cycle wastes, were used to determine when currently licensed burial site capacity will be exhausted.

The report also discusses basic liquid-waste processing techniques including filtration, demineralization, evaporation, and reverse osmosis. This is followed by discussions on various solidification techniques and the effect they have on overall waste volumes. Currently available waste volume-reduction techniques are discussed including solidification of the volume-reduction system product. This discussion covers the effect of these two processes on both waste volumes and activities, including transuranic radionuclides. Following volume reduction and solidification, average LWR and fuel-fabrication facility waste volumes are analyzed to determine the overall effect on burial site life assuming implementation of volume-reduction techniques at just LWRs and fuel-fabrication facilities and also at all types of facilities that generate waste buried at shallow-land burial sites.

1. SUMMARY

1.1 Introduction

This study describes the physical and chemical characteristics of wastes (other than spent fuel) that are generated at light-water-cooled nuclear reactor (LWR) power plants and nuclear fuel-fabrication facilities. The information used to define these characteristics is based on a survey of 30 nuclear power plants and 7 facilities involved in LWR fuel fabrication. Information on volumes and activity levels of five major categories of LWR and fuel fabrication facility waste was collected. The major categories of waste are spent resin, concentrated liquids, precoat filter sludge (including ground ion-exchange resin), cartridge filters, and compactible and non-compactible trash. The volumes and activity levels of these wastes were used to determine the average waste-generation rates for LWRs and fuel-fabrication facilities, and the typical waste-generation rates for LWRs. Average gross radioactive concentrations were calculated, including concentrations of transuranics, to determine annual activity-generation rates and the total radioactivity of the waste shipped. These generation rates were used in conjunction with projected increases in the gross electrical generating capacity of nuclear power plants to determine annual waste volumes through 2000. A determination of the effect of a broad application of current volume-reduction systems on the waste volumes is also included in this report.

The overall waste burial capacity of currently licensed burial site areas was examined and fuel-cycle wastes and nonfuel-cycle wastes were considered.

This chapter of the report presents the conclusions, a discussion of their bases, and recommendations regarding LWRs and fuel-fabrication facility wastes. The effect that the wastes generated at these facilities will have on current burial sites is also discussed with corresponding conclusions and recommendations.

1.2 Light-Water Reactors

1.2.1 Conclusions

1. Based on the data gathered for this study the following estimates are made:

    a. An average 1,000-MWe boiling water reactor (BWR) generates between 20,000 and 34,200 ft^3/yr of unsolidified waste. Implementation of federal government regulations requiring the solidification of all waste before transportation to burial sites could result in increased volumes ranging from 26,200 to 50,200 ft^3/yr.

b. An average 1,000-MWe pressurized-water reactor (PWR) generates between 16,700 and 17,200 ft^3/yr of unsolidified waste. Implementation of federal government regulations requiring the solidification of all waste before transportation to burial sites could result in a volume increase to 21,500 ft^3/yr.

2. Future plants are expected to generate the same types of waste (resin and trash) as current plants although the proportions of one waste type compared to another may change. Other types of waste such as dried salts from fluid bed dryers and ash from combustible waste incinerators are not currently generated at nuclear power plants. However, volume-reduction systems that process wastes to this form have been sold to U.S. utilities. These systems will be installed and operating within the next 5 to 6 years.

3. BWRs and PWRs produce almost the same quantities of compactible and noncompactible waste, approximately 11.5 ft^3/yr per MWe of installed capacity.

4. In BWRs, combustible trash is 17 to 32% of the total waste volume shipped with current solidification techniques. For PWRs, combustible trash accounts for approximately 36% of the total waste volume.

5. Of the BWRs using deep bed ion-exchange demineralizers for condensate polishing, those that use saltwater or brackish water for condenser cooling generate nine times the volume of spent resin that similar freshwater-sited plants generate.

6. BWRs using deep bed ion-exchange demineralizers for condensate polishing generate nearly three times the amount of process wastes (with four times the activity) that is generated at BWRs with precoat filters in the condensate polishing system. Process wastes are spent resins, filter sludges, spent cartridge filters, and concentrated liquids.

7. The total annual waste volume generated at PWRs does not appear to be dependent on whether or not the plant has a condensate polishing system.

8. The total quantity of radioactivity contained in BWR wastes ranges from 1.0 Ci/MWe-yr to 5.0 Ci/MWe-yr. For PWRs radioactivity ranges from 0.4 Ci/MWe-yr to 1.0 Ci/MWe-yr.

9. Transuranic radionuclides are present in both PWR and BWR wastes at concentrations ranging from 0.02 nCi/g to 8.5 nCi/g.

10. Following volume reduction and solidification of some LWR wastes the concentration of transuranic radionuclides may exceed 10 nCi/g.

11. By 2000 the total volume of waste shipped annually from LWRs will increase by a factor of 10 over the volume shipped in 1977. Based on current waste solidification practices this increase will be from 10^6 ft^3/yr in 1977 to 10^7 ft^3/yr in 2000.

    If all wastes are solidified, except trash, the annual volume of waste shipped from LWRs by 2000 will be only slightly higher at 10.8×10^6 ft^3/yr. The activity associated with these wastes will total over 600,000 Ci/yr by 2000.

    Broad application of volume reduction techniques could reduce the total quantity of solidified LWR waste requiring burial to less than 4×10^6 ft^3/yr in 2000.

The eleven conclusions noted are based on the data collected in the survey of 29 nuclear power plants including 18 PWRs and 12 BWRs. (One plant contains a PWR unit and a BWR unit.) These data are given in Section 4.2 of the report along with the results of the analysis of the data. Projections of waste volumes and waste activities through 2000 are based on these analyses and on the forecast of U.S. LWR generating capacity through 2000.

The following section is a discussion of each conclusion and the data that support it.

1.2.2 Discussion

1. Plants surveyed included 12 BWRs. Six of these plants use deep bed demineralizers for condensate polishing (henceforth termed deep bed plants). Of these six, four are single-unit plants, one is a two-unit plant, and the last plant consists of one BWR unit and one PWR unit. The other six plants use precoat filters for condensate polishing (henceforth termed precoat plants). These plants consist of five single-unit plants and one two-unit plant. When the data from these plants are analyzed as described in Appendix E the following estimates are calculated.

    a. For deep bed plants the largest source of process waste is the concentrated liquids at 12.7 ft^3/MWe-yr, 21 times the 0.6 ft^3/MWe-yr generated at precoat plants.

    b. Deep bed plants also produce 20 times the spent resin that precoat plants produce, 4.6 ft^3/MWe-yr versus 0.23 ft^3/MWe-yr, respectively.

c. Filter sludge volumes between the two types of plant are much closer to being equal, with 5.4 ft^3/MWe-yr produced at deep bed plants compared to 7.7 ft^3/MWe-yr for precoat plants.

d. Cartridge filters, although they are used in several BWRs, do not contribute significantly to the annual waste generation rates.

e. Trash at deep bed resin plants and precoat plants totals 11.5 ft^3/MWe-yr.

For precoat plants these figures represent a total of 20.0 ft^3/MWe-yr, or for a 1,000-MWe plant an annual waste generation rate of 20,000 ft^3. For deep bed plants the total is 71% greater primarily due to the greater quantity of spent resin and concentrated liquids. A 1,000-MWe BWR with a deep bed condensate polishing system could produce as much as 34,200 ft^3/yr of unsolidified waste. These data are summarized in Table 1.2-1.

When the process wastes (that is the spent resin, filter sludge, and concentrated liquids) are solidified, the volume of waste increases by 61% for spent resins, 72% for precoat filter sludge, and by 73% for concentrated liquids. Cartridge filters are either packaged as non-compactible trash or solidified in other wastes. Either way, the quantity of this type of waste from BWRs is extremely low. Trash is not solidified.

Based on these figures, if a plant were to solidify all its process wastes the annual volume shipped to a disposal site would be 26,200 ft^3 for a 1,000-MWe precoat plant and 50,200 ft^3 for a 1,000-MWe deep bed plant. See Table 1.2-2.

2. Of the 18 PWRs surveyed, 10 plants representing 12 units do not have a condensate polishing system. The other 8 plants representing 11 units do have condensate polishing systems. Data from these plants, analyzed as described in Appendix E, result in the following estimated annual waste generation rates:

   a. In PWRs, both with and without a condensate polishing system, concentrated liquids represent the largest source of process wastes. The PWRs without a condensate polishing system generate 3.9 ft^3/MWe-yr compared to 4.8 ft^3/MWe-yr for a PWR with a condensate polishing system.

   b. Spent resin wastes account for 0.94 ft^3/MWe-yr in a PWR without a condensate polishing system, approximately three times the 0.32 ft^3/MWe-yr from a plant with a condensate polishing system.

Table 1.2-1 Average Plant Untreated Waste Volumes

Waste Type	Waste Volumes (ft^3/MWe-yr) Boiling Water Reactors Deep Bed CPS[1]	Boiling Water Reactors Precoat CPS	Pressurized Water Reactors Without CPS	Pressurized Water Reactors With CPS
Deep bed resin	4.6	0.23	0.94	0.32
Concentrated liquids	12.7	0.6	3.9	4.8
Filter sludge	5.4	7.7	-	.015
Cartridge filters	-	-	0.39	0.39
Trash				
Total	11.5	11.5	11.5	11.5
Compactible	7.8	7.8	7.6	7.6
Noncompactible	3.7	3.7	3.9	3.9
Total	34.2	20.0	16.7	17.2
Annual volume (ft^3/yr) for a 1,000 MWe plant	34,200	20,000	16,700	17,200

1. Condensate polishing system.

Table 1.2-2 Solidified Waste Volumes With All Waste Solidified Ft^3/MWe Installed

Waste Type	Boiling Water Reactors		Pressurized Water Reactors	
	Deep Bed CPS(1)	Precoat CPS	With CPS(1)	Without CPS
Deep bed resin	7.4	0.37	0.54	1.6
Concentrated liquids	22.0	1.1	8.8	8.0
Filter sludge	9.3	13.2	0.25	-
Cartridge filters	-	-	0.39	0.39
Trash (all)	11.5	11.5	11.5	11.5
Total	50.2	26.2	21.5	21.5
1,000 MWe plant	50,200 ft^3/yr	26,200 ft^3/yr	21,500 ft^3/yr	21,500 ft^3/yr

1. Condensate polishing system.

c. Precoat filter sludge is limited to those plants at which precoat filters are used for condensate polishing. From the limited data available it is estimated that the annual generation rate is 0.15 ft^3/MWe-yr.

d. Cartridge filters, used extensively in PWRs in applications that are independent of whether or not the plant has a condensate polishing system, provide 0.39 ft^3/MWe-yr of waste.

e. Trash is generated at the same rate at PWRs as at BWRs, 11.5 ft^3/MWe-yr.

For PWRs without a condensate polishing system these figures total 16.7 ft^3/MWe-yr, or 16,700 ft^3 annually for a 1,000-MWe plant. A 1,000-MWe PWR with a condensate polishing system generates 17,200 ft^3/year based on 17.2 ft^3/MWe-yr. This data is also summarized in Table 1.2-1.

Solidification of spent resins at PWRs is less efficient than at BWRs because of the presence of boric acid on some resins. The increase in volume due to solidification is approximately 70% for spent resins and approximately 67% for precoat filter sludge. Concentrated liquids that have totally different characteristics in a PWR than in a BWR (concentrated boric acid versus concentrated sodium sulfate), increase in volume by 83 to 105%. These percentages are for PWRs without a condensate polishing system and with a condensate polishing system, respectively.

Cartridge filters were solidified in many cases. Utilities provided data on cartridge filter volume after the cartridge filters had been packaged. The reported cartridge filter waste volumes were not reduced to account for solidification because there is no practical method of reducing the unpackaged volumes of this waste once it is generated.

As is the case in BWRs, the trash in PWRs is not solidified. Totaling the solidified waste volumes, including cartridge filters and trash, results in an estimated 21,500 ft^3/year of waste for PWRs with and without a condensate polishing system. See Table 1.2-2.

3. The longest any of the plants in this survey has been operating is 10 years. Over this period of time, from 1968 through 1977, the types of waste reported have been essentially unchanged. The only change to take place in those 10 years is the introduction of urea-formaldehyde (UF) as a solidification agent in 1974. The types of waste have remained the same during this period, consisting of spent resin, filter sludge, and trash. These can be expected to change considerably as volume-reduction

systems are installed either as original equipment on new plants or backfits to older plants. Use of systems using asphalt for solidification will result in large quantities of asphalt being sent to disposal sites. Fluid bed dryers will generate large quantities of dried sodium sulfate and boric acid salts from the processing of concentrated liquid chemical wastes. Incinerators will turn the combustible trash into minute quantities of ash. Filter sludge and spent resins will also be converted to ash if processed in a fluid bed dryer.

Therefore, if the volume reduction of LWR wastes is practiced at a significant number of facilities, a large part of the waste requiring disposal will be solidified dry salts and ash rather than the sludges, resins, liquids, and bulk trash.

4. As noted previously, the total quantity of compactible and noncompactible trash produced by BWRs and PWRs is virtually the same, 11.5 ft^3/MWe-yr. The data on compactible and noncompactible trash collected in the survey are labeled throughout the report. In BWRs the compactible trash accounted for 7.8 ft^3/MWe-yr with the remaining 3.7 ft^3/MWe-yr being noncompactible trash. In PWRs the compactible trash accounts for 7.6 ft^3/MWe-yr with the remaining 3.9 ft^3/MWe-yr being noncompactible trash.

5. Specific data on what quantity of the compactible and noncompactible trash was combustible were not available. However, from the descriptions of the various items that compose the compactible and noncompactible trash it is estimated that two-thirds of the total waste volume was combustible. The similarity in the composition of the trash at BWRs and PWRs leads to the same estimate for both types of plants. With a total generation rate of 11.5 ft^3/MWe-yr it is estimated that 7.7 ft^3/MWe-yr was combustible and 3.8 ft^3/MWe-yr was noncombustible. For a 1,000-MWe plant the combustible trash accounts for 7,700 ft^3/yr of the total annual waste volume. This is 17% of the 45,900 ft^3/yr shipped from a BWR with a deep bed condensate polishing system, and 32% of the 23,700 ft^3/yr shipped from a BWR with a precoat filter condensate polishing system. The annual volumes quoted here are based on current solidification practices. For PWRs, the 7,700 ft^3/yr represents 36% of the total, 21,300 ft^3/yr or 21,100 ft^3/yr shipped from plants with and without a condensate polishing system, respectively.

6. Of the six deep bed BWRs, three are on saltwater sites and three are on freshwater sites. The quantity of resins shipped from the plants on saltwater sites averages 5.8 ft^3/MWe-yr. For the freshwater-sited plants, spent resins are generated at a rate of 0.64 ft^3/MWe-yr. One of the reasons for this large difference is the fact

that one of the saltwater plants is no longer regenerating its condensate resins because of operational problems with the chemical waste concentrator. The inability to concentrate the regenerant chemicals necessitates the disposal of the resins as they become exhausted.

7. The total quantity of process waste (spent resin, filter sludge, and concentrated liquids) from BWRs with a deep bed condensate polishing system is 22.7 ft^3/MWe-yr. BWRs with precoat filters in the condensate polishing system generate about one-third this volume, at 8.5 ft^3/ MWe-yr. The volume of filter sludge from the two types of plants is roughly equal, 5.4 ft^3/MWe-yr versus 7.7 ft^3/MWe-yr, with the larger quantity of filter sludge being generated at precoat plants.

   Initially it appears that deep bed plants produce 20 times the spent resin and concentrated liquids that precoat plants produce, 17.3 ft^3/MWe-yrs versus 0.83 ft^3/MWe-yr. However, when the spent resin data for deep bed plants sited on freshwater are compared to the spent resin data for precoat plants, the gap in the annual generation rates narrows, 0.64 ft^3/MWe-yr compared to 0.32 ft^3/MWe-yr.

   Compared to the spent resin volumes from the saltwater- and freshwater-sited deep bed plants there was no discernible difference in the generation rates for concentrated liquids.

8. For PWRs with a condensate polishing system, the total annual waste generation rate is 17.2 ft^3/MWe-yr, less than 3% greater than the 16.7 ft^3/MWe-yr for a PWR without a condensate polishing system. With current waste-solidification practices the difference is less than 1% at 21.1 ft^3/MWe-yr and 21.3 ft^3/MWe-yr, respectively.

   If all PWR wastes were solidified, plants with and without a condensate polishing system would ship identical waste volumes of 21.5 ft^3/MWe-yr.

9. The total activity associated with BWR and PWR wastes covers a range from 0.420 to 4.88 Ci/MWe-yr. The largest quantity of activity is shipped from BWRs with a deep bed condensate polishing system, 4.88 Ci/MWe-yr. BWRs with a precoat condensate polishing system and PWRs without a condensate polishing system ship 0.92 Ci/MWe-yr and 1.0 Ci/MWe-yr, respectively. The lowest quantity of activity is shipped from PWRs with a condensate polishing system; that activity is 0.42 Ci/MWe-yr. The only similarities in activity levels between reactor types is in the compactible trash, which for both BWRs and PWRs was approximately 5 Ci/yr for a 1,000-MWe plant. A breakdown of activities by waste type is given in Table 1.2-3.

Table 1.2-3 Average Plant Waste Activity

Waste Type	Waste Activity (Ci/MWe-yr) Boiling Water Reactors Deep Bed CPS[1]	Boiling Water Reactors Precoat CPS	Pressurized Water Reactors Without CPS	Pressurized Water Reactors With CPS
Deep bed resin	1.9	.0014	0.61	0.2
Concentrated liquids	0.58	0.016	0.20	0.024
Filter sludge	2.0	0.5	-	0.012
Cartridge filters	-	-	0.12	0.12
Trash				
Total	0.402	0.402	0.063	0.063
Compactible	0.0052	0.0052	0.0049	0.0049
Noncompactible	0.397	0.397	0.058	0.058
Total	4.88	0.92	1.00	0.42
Annual activity (Ci/yr) for a 1,000 MWe plant	4,880	920	1,000	420

1. Condensate polishing system.

10. Concentrations of transuranic nuclides found in samples taken at a number of LWR facilities for recent studies by the U.S. Environmental Protection Agency (EPA, 1977) and Electric Power Institute (EPRI, 1978) have been applied to the waste types identified in this study to determine the degree of transuranic contamination of LWR wastes. The highest concentration of transuranic contamination (8.5 nCi/g) was found in the filter sludge from a BWR with a precoat filter condensate polishing system. Spent resin wastes from a BWR with a deep bed condensate polishing system had the lowest reported concentration of transuranics, 0.019 nCi/g. These concentrations are for unsolidified wastes.

    These concentrations, and others reported in this report, are averages from as many as six samples taken at individual plants. Additional samples are essential in determining accurate, long-term, average transuranic concentrations. However, these data are important in that they define the relative levels of contamination and show that significant levels of transuranic contamination do exist.

11. As LWR process wastes are treated in volume-reduction systems, the concentrations of all radionuclides will increase inversely in proportion to the volume. In some cases the addition of the solidification agent results in a final volume equal to or greater than the original waste volume. In these cases the final transuranic concentrations are equal to or lower than in the original unsolidified waste. The PWR concentrated liquids, when completely dried in an extruder evaporator or fluidized bed dryer may decrease in volume by as much as a factor of 13. This would be a larger decrease than for any other waste type. When solidified with asphalt, concentrator bottoms from a PWR with a condensate polishing system are projected to have a final transuranic radionuclide concentration of 14 nCi/g. This is the highest transuranic concentration of any waste. The same waste solidified with cement will have a transuranic radionuclide concentration of 3.7 nCi/g because of the differences in density between asphalt and cement. The concentration in wastes solidified with urea-formaldehyde or a polyester resin would be similar to the concentration in asphalt.

    Table 1.2-4 shows the estimated concentrations of transuranic radionuclides in solidified wastes. When processed through a bitumen solidification system the concentrations will be as shown in Table 1.2-5. Table 1.2-6 shows the average concentrations for wastes treated in a fluidized bed dryer and solidified in cement.

12. The total quantity of waste shipped from U.S. LWRS during 1976 was 1.06×10^6 ft^3, including 173,000 ft^3 of

Table 1.2-4 Transuranic Radionuclide Concentrations in Solidified LWR Low Level Wastes, No Volume Reduction

Waste Type	Radionuclide Concentration (μCi/g)			
	BWRs		PWRs	
	Deep Bed CPS	Precoat CPS	Without CPS	With CPS
Deep bed resin	6.9(-6)	(1)	7.6(-4)	5.0(-5)
Concentrator bottoms	1.6(-5)	NA(2)	2.7(-5)	4.7(-4)
Filter sludge	5.4(-4)	2.8(-3)	NA	NA
Cartridge filters	NA	NA	NA	NA
Trash	NA	NA	NA	NA

1. Included in filter sludge.
2. Not available.

Table 1.2-5 Transuranic Radionuclide Concentrations: Extruder/ Evaporator VR with Asphalt Solidification Agent

Waste Type	Radionuclide Concentration (μCi/g)			
	BWRs		PWRs	
	Deep Bed CPS[1]	Precoat CPS	Without CPS	With CPS
Deep Bed resin	1.4(-5)	(2)	1.4(-3)	9.4(-5)
Concentrator bottoms	3.0(-5)	(3)	9.0(-4)	1.4(-2)
Filter sludge	1.1(-3)	6.4(-3)	NA[4]	(3)
Cartridge filters	NA	NA	(3)	(3)
Trash	(3)	(3)	(3)	(3)

1. Condensate polishing system.
2. Included in filter sludge.
3. Not available.
4. Not applicable.

Table 1.2-6 Transuranic Radionuclide Concentrations: Fluidized Bed Dryer VR With Cement Solidification Agent

Waste Type	Radionuclide Concentration (μCi/g)			
	BWRs		PWRs	
	Deep Bed CPS[1]	Precoat CPS	Without CPS	With CPS
Deep Bed resin	8.9(-6)	(2)	9.6(-4)	6.0(-5)
Concentrator bottoms	7.3(-5)	NA[3]	2.6(-4)	3.7(-3)
Filter sludge	7.5(-4)	4.1(-3)	NA	NA
Cartridge filters	NA	NA	NA	NA
Trash	NA	NA	NA	NA

1. Condensate polishing system.
2. Included in filter sludge.
3. Not available.

contaminated soil. The projection for 1977, based on current waste-solidification practices and the average waste-generation rates discussed in item 1 of this section is 1.05 x 10^6 ft^3. By 2000 these wastes will increase by a factor of 10, to 10.3 x 10^6 ft^3/yr. This estimate is based on the continuation of current waste-management techniques and solidification practices. The BWRs with a deep bed condensate polishing system will have a total of 85.3 GWe of installed generating capacity and ship 45.9 ft^3/MWe-yr for a total of 3.92 x 10^6 ft^3/yr. BWRs with a precoat condensate polishing system will account for 35.8 GWe of installed generating capacity and ship 23.7 ft^3/MWe-yr for a total of 0.85 x 10^6 ft^3/yr. The PWRs will represent over two-thirds of the total LWR generating capacity in 2000 but will ship only 15% more waste because of their lower waste-generation rates. By 2000 it is projected that PWRs with a condensate polishing system will total 124.6 Gwe of installed generating capacity and ship 21.3 ft^3/MWe-yr of waste resulting in 2.65 x 10^6 ft^3/yr. The PWRs without a condensate polishing system will represent 134.0 GWe of installed generating capacity in 2000 and will generate 21.1 ft^3/ MWe-yr for a total of 2.83 x 10^6 ft^3/yr.

If all the wastes except compactible and noncompactible trash are solidified, the waste volumes increase slightly. Deep bed BWRs will generate 50.2 ft^3/MWe-yr resulting in 4.29 x 10^6 ft^3/yr, and precoat BWRs will generate 26.2 ft^3/MWe-yr, 0.93 x 10^6 ft^3/yr in 2000. The PWRs currently solidify most of their waste, and the increase in annual volume resulting from the solidification of all of the waste will be minimal. In 2000 a PWR with a condensate polishing system will generate 21.5 ft^3/ MWe-yr for 2.68 x 10^6 ft^3/yr, and PWRs without a condensate polishing system will also generate 21.5 ft^3/MWe-yr for 2.88 x 10^6. The total annual waste generation will then be 10.8 x 10^6 ft^3/yr.

Applying currently available volume-reduction techniques to LWR wastes could result in more than a 60% decrease in the total quantity of waste to be buried by 2000. This decrease is primarily due to substantial reductions in the volume of concentrated liquid wastes and the volume of combustible waste. Assuming an equal split between the types of volume-reduction techniques implemented, and, to maximize the net effect (to develop a lower bond) assuming incineration of combustible trash at each plant, results in the following estimated generation rates. Deep bed BWRs will ship 20.8 ft^3/MWe-yr totaling 1.77 x 10^6 ft^3/yr by 2000 with 0.47 x 10^6 ft^3/yr from precoat BWRs, which generate 13.2 ft^3/MWe-yr. PWRs in 2000 will average 5.9 ft^3/MWe-yr, or 0.74 x 10^6 ft^3/yr for a plant with a condensate polishing plant and 6.2 ft^3/MWe-yr, resulting in 0.83 x 10^6 ft^3/yr for a PWR without a condensate polishing system. The lower limit waste volume in 2000 can then be estimated as 3.81 x 10^6 ft^3/yr. These figures are listed in Table 1.2-7.

Table 1.2-7 Summary of Waste Projections and Generating Capacity in 2000

	BWRs		PWRs		
	Deep Bed	Precoat	With CPS	Without CPS	Total
Project generating capacity, GWe	85.3	35.8	124.6	134.	379.9
Total waste volume shipped, 10^6 ft^3/yr					
Current waste solidification techniques	3.92	.85	2.65	2.83	10.3
All waste solidified (except trash)	4.29	.93	2.68	2.88	10.8
With broad application of volume reduction techniques	1.77	.47	.74	.83	3.81
Total activity shipped, 10^3Ci/yr	418.	32.9	134.	52.2	637.

1.2.3 Recommendations

1. The personnel of any LWR that generates significantly more waste than estimated for an average plant (as defined in this study) should seriously consider undertaking a detailed evaluation of their plant's radwaste system. This evaluation should include the following:

   a. A determination of the sources of liquid wastes requiring processing in the radwaste system, or other systems which result in the generation of solid wastes. Once identified, these sources should be evaluated to determine those that can be reduced or eliminated.

   b. Alternate processing methods, including additional equipment, should be evaluated in terms of their ability to reduce the quantity of solid waste generated within the plant. This evaluation should include volume-reduction systems, and alternate waste solidification methods.

2. A comprehensive analysis of the concentration of transuranic radionuclide contamination in LWR waste should be undertaken. This study should identify the source of transuranic contamination and develop recommendations for limiting transuranic contamination to as little of the LWR process waste as possible. Such a study would provide significant data required by any subsequent analysis of the costs associated with disposal of transuranic contaminated wastes.

1.3 Fuel-Fabrication Facilities

1.3.1 Conclusions

1. The total waste-generation rate for fuel-fabrication facilities is approximately 80 ft^3/MTU-yr, containing roughly 12 $\mu Ci/ft^3$ of U-235 and U-238.

2. Fuel-fabrication-facility wastes contribute 10 to 15% as much waste as do the LWRs they support.

These conclusions are based on the data collected in the survey from seven facilities involved in the manufacturing of fuel elements for light-water-cooled nuclear power reactors. These data are presented in Section 4.4 of the report along with the results of the analyses of the data. Projections of waste volumes through 2000 are based on these analyses, the forecast of U.S. nuclear generating capacity through 2000, and the average fuel-fabrication facility capacity required to support an average 1,000-MWe PWR or BWR annually.

The following section is a discussion of the data and information which lead to the previous conclusions.

1.3.2 Discussion

1. Fuel-fabrication facilities generate wastes such as contaminated clothing, wood from packaging crates, filter sludges, filters, and other combustible and noncombustible wastes. Several of the plants incinerate combustible trash to concentrate any residual uranium contamination. If the uranium concentration in the ash or in the filter sludge is high enough to allow economical recovery, the material is processed through a uranium recovery process. After processing through recovery, the wastes are packaged and shipped offsite for burial.

   The facilities surveyed ranged in capacity from 275 MTU/yr to 1,000 MTU/yr. The quantity of wastes shipped from these plants ranged from 4,200 ft^3/yr to 104,000 ft^3/yr. The average shipment was approximately 40,000 ft^3/yr generated at a rate of 80 ft^3/MTU.

2. The projected capacity of fuel-fabrication facilities in 2000 to support 380 GWe of installed nuclear power plant capacity is 12,500 MTU/yr. Current installed capacity is approximately 50% higher than domestic demand calls for. This increased capacity is necessary to cover facility down time, operation at less than full capacity, and allowance for delivery of full core loadings to new plants and foreign sales. As the need for nuclear fuels increase and new fuel-fabrication facilities are built, the requirement that the installed capacity be greater than demand will continue. Therefore, with a domestic demand of 12,500 MTU/yr in 2000, it is estimated that the total industry-installed capacity will be approximately 18,800 MTU/yr. Figure 1.3-1 shows the installed capacity of U.S. fuel-fabrication facilities to support U.S. LWRs through 2000.

   With an average waste-generation rate of 80 ft^3/MTU, the total volume of waste to be shipped to burial sites in 2000 will be approximately 1.5×10^6 ft^3. The waste volume generated to support the 380 GWe of installed nuclear power plant capacity is approximately 1.0×10^6 ft^3/yr. At the same time the total volume of waste being shipped from U.S. LWRs, under continued current waste management practices, is estimated to be 10×10^6 ft^3/yr. Thus, in 2000 fuel-fabrication facilities will generate 10 to 15% as much waste as the LWRs for which they produce fuel. The quantity of wastes produced in support of U.S. LWRs and in support of foreign sales through 2000 are shown in Figure 1.3-2.

1.3.3 Recommendations

Fuel-fabrication facilities should institute procedures to unpackage incoming equipment in uncontaminated areas

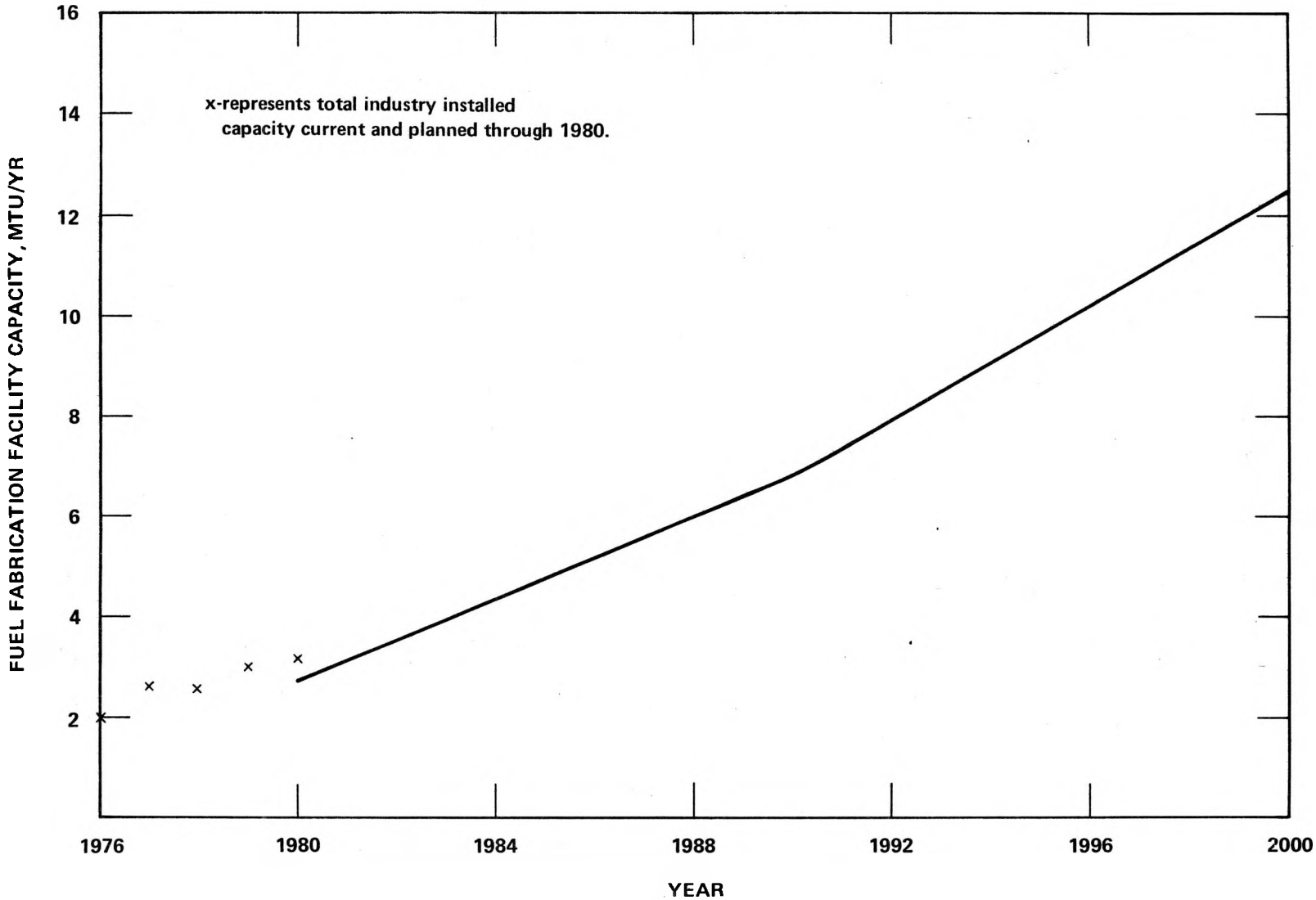

Figure 1.3-1 Total Required Throughput to Support U.S. LWRs (10^3 MTU).

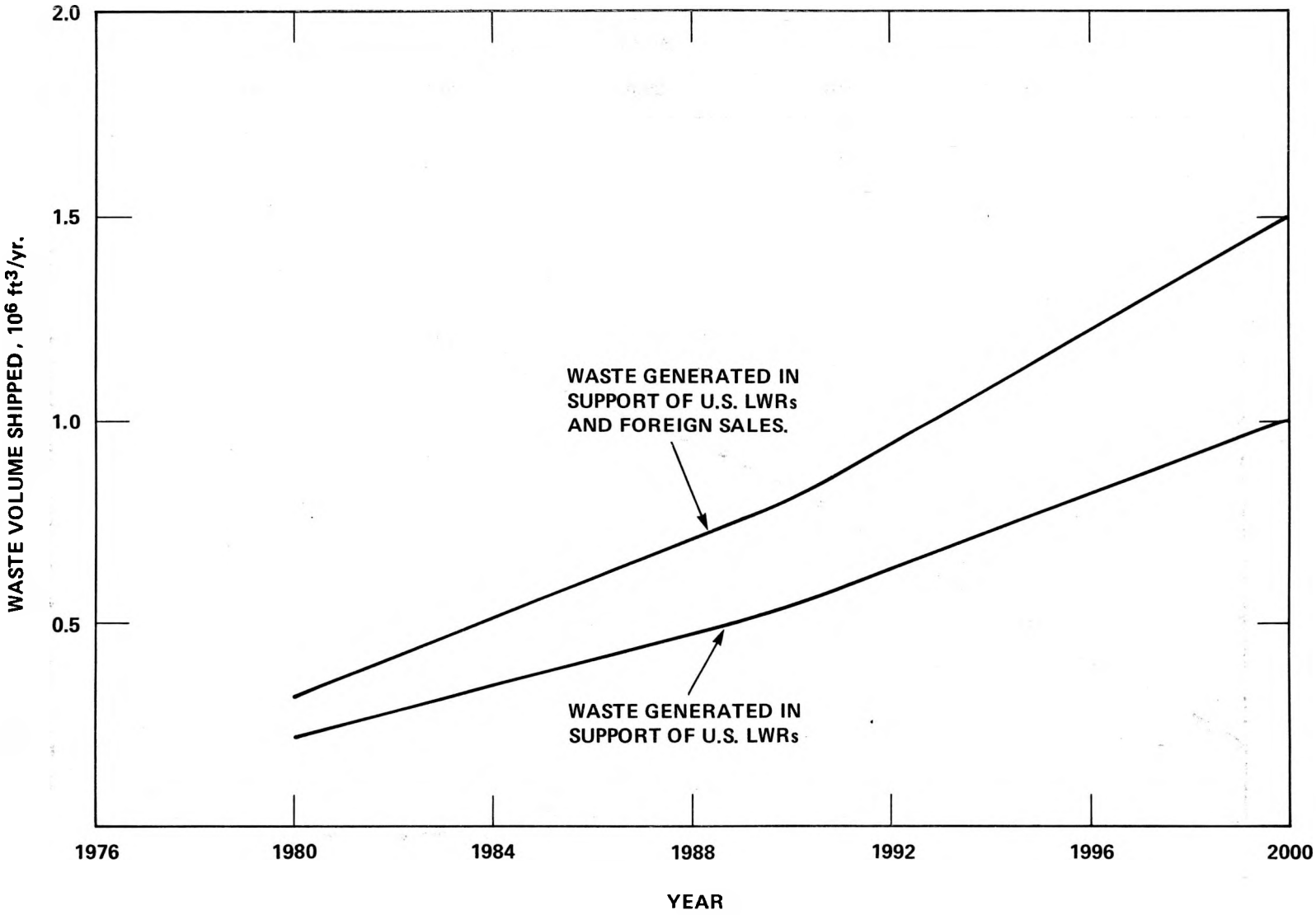

Figure 1.3-2 Annual Waste Volume from Fuel Fabrication Facilities.

so that the related packaging material may be disposed of as nonradioactive trash. Increased use of incinerators should be investigated as an additional method of waste-volume reduction and increased uranium recovery.

1.4 Burial Sites

1.4.1 Conclusions

These conclusions are based on the projected waste-generation rates for LWRs and fuel-fabrication facilities and on the assumption that these wastes contribute half of the total volume of waste buried at commercial shallow land burial sites. These projections are detailed in Chapter 5.

1. Currently licensed burial-site capacity will be exhausted in 1990 and an additional 600 acres will be required by 2000 if current waste management practices are continued.

2. With the initiation by 1985 of volume-reduction practices at all facilities that generate radioactive wastes, including LWRs, fuel-fabrication facilities and nonfuel cycle sources such as hospitals and laboratories, the currently licensed burial capacity will be exhausted by 1996. Under these conditions an additional 134 acres is required through 2000.

1.4.2 Discussion

Of the six commercial burial sites that have operated, only three are currently accepting waste for burial. These three facilities have approximately 283 acres of licensed land remaining for waste burial. With an average burial capacity of 325,000 ft^3/acre, the total remaining capacity is 92×10^6 ft^3.

Under current waste management practices the total radioactive waste shipped to these three burial sites will reach 92×10^6 ft^3 by 1990. Between 1978 and 1990 it is estimated that 46×10^6 ft^3 will be shipped by plants not in the uranium fuel cycle, 40×10^6 ft^3 by LWRs and the remaining 6×10^6 ft^3 by fuel-fabrication facilities.

Volume-reduction processes now available can reduce the cumulative volume of wastes from all U.S. LWRs through 1990 by a factor of 2, if they are phased into operation between 1980 and 1985. Similar reductions in the cumulative volume of radioactive waste from plants not in the uranium fuel cycle will result in extended burial site life. With these reductions it is estimated that currently licensed burial areas will be sufficient through 1996 Figure 1.4-1

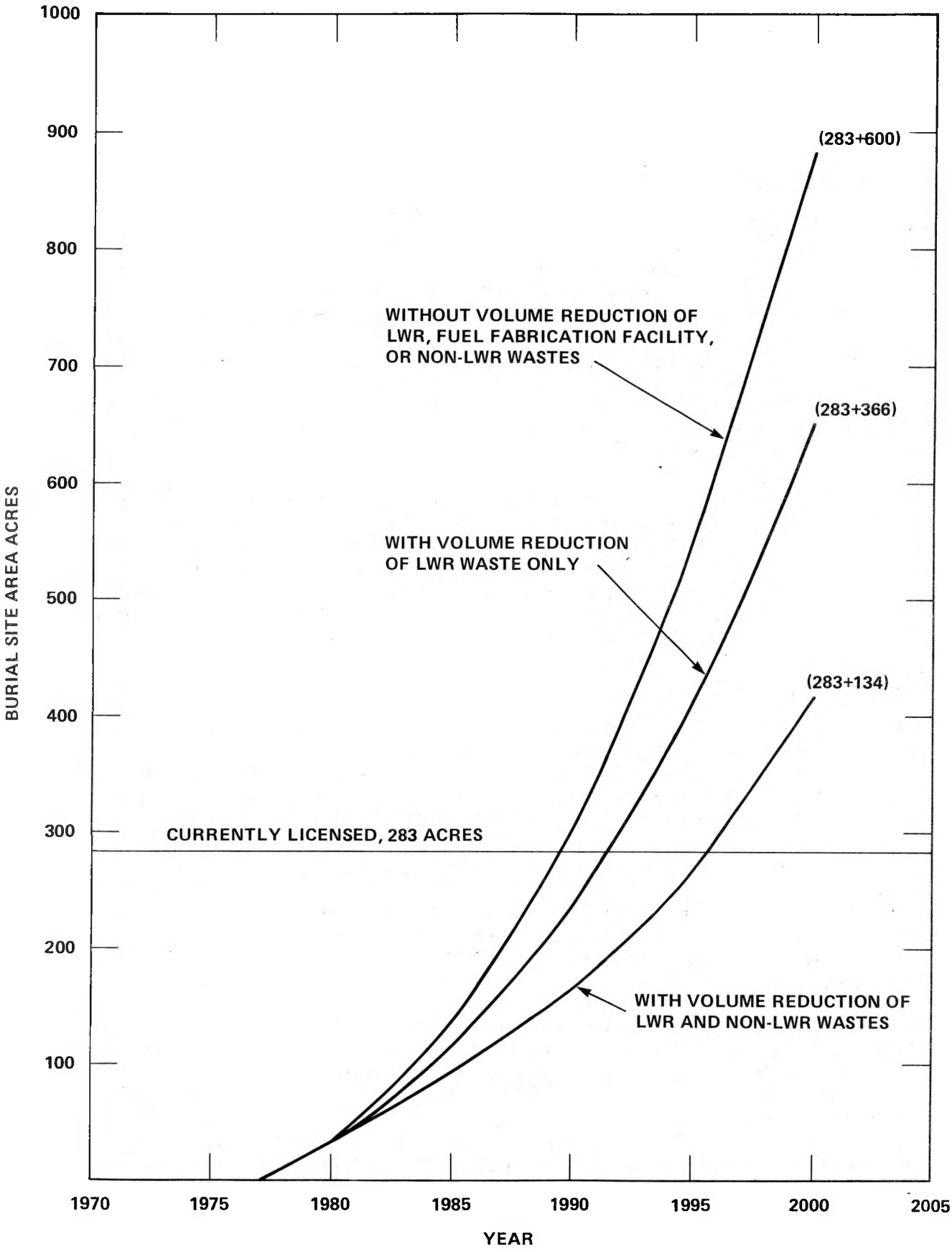

Figure 1.4-1 Burial Site Area Needed After 1977.

1086

shows the required burial site acreage needed through 2000 for each of these situations compared to the average currently licensed.

These estimates are highly dependent on the amount of wastes being buried at these facilities that comes from the sources not in the uranium fuel cycle.

1.4.3 Recommendations

1. A detailed evaluation of wastes that are not in the uranium fuel cycle, such as pharmaceutical, hospital, and laboratory waste should be undertaken to determine the nature, volumes, and activities of these wastes. Such an evaluation would determine the acceptable methods for the volume reduction of these wastes.

2. Alternate methods of disposal of LWR and fuel-fabrication facility wastes such as burial in geological respositories, sea disposal, deep well injection and hydrofracture should be investigated. These evaluations should include the geological requirements of the first and last two alternatives and the criteria for ocean disposal being developed by the EPA for the second. Adequate justification for licensing, procedures for ensuring adequate environmental protection, and the economics involved in each alternative should also be considered.

3. Current regulatory actions to incorporate the 1973 Revised Edition of the International Atomic Energy Agency (IAEA) Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials," should proceed as expeditiously as possible. Adoption of these revisions will allow increased activity limits per unit of volume for use in transportation and burial of waste. This in turn will result in higher concentrations of activity in the waste, lower volumes of waste and ultimately lower burial costs.

4. The possibility of expanding current burial sites should be evaluated, or new burial sites should be identified, licensed and developed as soon as possible. Any increase in burial area should be evaluated according to federal regulatory statutes being developed by both the NRC and the EPA.

2. WASTE CHARACTERIZATION

2.1 Introduction

This section is a discussion of the general chemical and physical properties of the process wastes identified in the survey. Process wastes are produced by the removal or concentration of radioactive contaminants from liquid waste streams. Five waste forms are discussed. They are, deep bed (ion-exchange) resins, precoat filter sludge, cartridge filters, concentrator bottoms, and reverse-osmosis sludge. Compactible and noncompactible waste, the major contributor to a plant's annual waste volume, is not discussed here because it is not a process waste. A complete discussion of compactible and noncompactible waste, or trash, is found in Section 4.2.1.5 for BWRs and in Section 4.2.2.5 for PWRs. The discussions center on specific resins, filter precoats, and cartridges identified as being used by the plants of this survey. The discussion on concentrators and concentrator bottoms is also based on information collected in the survey and on data recently reported by Godbee (1978). The discussion on reverse-osmosis sludge is based primarily on vendor data (Kremem, 1970) and the NRC (1976a).

As discussed in the Introduction to this report, this survey did not collect new data on the quantity of specific isotopes in the various types of waste. This report relies on other reports in which specific waste samples were analyzed for individual radionuclide content (EPA, 1977; EPRI, 1978). In this section only the relative concentrations are pertinent. These data are defined and then analyzed to determine the short-term and long-term (100 years) relative concentrations. Transuranic radionuclides are discussed also.

2.2 Chemical and Physical Characteristics of Process Wastes

2.2.1 Demineralizer Resins

2.2.1.1 Physical and Chemical Properties

Demineralizer, or ion-exchange resins are porous beads of polystyrene cross linked with divinylbenzene. Strong cation resins generally contain bound sulfonic acid functional groups, and to a lesser degree, carboxylic or phosphonic acid. Strong anion resins generally contain bound quaternary ammonium functional groups.

Rohm & Haas resin IRN-300 contains a weak base anion resin, XE- 236. This resin has an acrylic matrix structure and a polyamine functional group. Tables 2.2-1 through 2.2-3 contain most of the available information regarding the major resins in this study that are used in nuclear power plants.

Table 2.2-1 Physical and Chemical Characteristics of Selected Rohm & Haas Amberlite Resins

Nuclear Grade Resin Designation	Parent	Resin Type (Base-Anion) (Acid-Cation)	Functional Group	Matrix Structure	Mesh Range	Ionic Form	Total Exchange Capability (kgr/ft^3)	pH Range	Maximum Temperature ^{o}F	Shipping Weight (lb/ft^3)	Regenerative	Moisture wt%
IRN-77	IR-120	Strong acid	Sulfonic acid	Polystyrene-DVB[1]	16-50	H^+	39.2	0-14	250	53	Yes	44-48
IRN-78	IRA-400	Strong base	Quaternary ammonium	Polystyrene-DVB	20-50	80% OH^- 5% Cl^-, 5% CO^-	17.0	0-14	170	44	Yes	42-48
IRN-150	IRA-400	Strong base	Quaternary ammonium	Polystyrene-DVB	20-50	80% OH- 5% Cl-, OH- CO_3^-	12.0	0-14	170	48	Yes	42-48
	IR-120	Strong acid	Sulfonic acid		16-50	H^+						
IRN-217[2]	IRA-400	Strong base	Quaternary ammonium	Polystyrene-DVB	20-50	OH^-	10.0	0-14	170	48	Yes	42-48
	IR-120	Strong acid	Sulfonic acid		16-50	Li^{7+}						
IRN-218	IR-120	Strong acid	Sulfonic acid	Polystyrene-DVB	16-50	Li^{7+}	38.4	0-14	250	53	Yes	44-48
IRN-300[3]	IRN-77	Strong acid	Sulfonic acid	Polystyrene-DVB	16-50	H^+	22.2	0-14	250	43-47	-	46-52
	XE-236	Weak base	Polyamine	Acrylic	-	Free Base		0-7	140			
IRA-200C[4]	-	Strong acid	Sulfonic acid	Polystyrene-DVB	16-50	Na^+	38.2	0-14	300	50	Yes	46-51
IRA-900C[4]	-	Strong base	Quaternary ammonium	Polystyrene-DVB	20-50	Cl^-	21.8	0-14	170	42	Yes	58-64

1. DVB-divinylbenzene cross linking.
2. Effluent produced is dilute ^{7}LiOH.
3. Will deionize water without removing boric acid.
4. Particularly suited to high flow rate deionization such as condensate polishing.

Table 2.2-2 Physical and Chemical Characteristics of Selected Illinois Water Treatment Resins

Nuclear Grade Resin Designation	Parent	Resin Type (Base-Anion) (Acid-Cation)	Functional Group	Matrix Structure	Mesh Range	Ionic Form	Total Exchange Capability (kgr/ft^3)	pH Range	Maximum Temperature ^oF	Shipping Weight (lb/ft^3)	Regenerative	Moisture wt%
TC1		Strong acid		Polystyrene-DVB	16-50	H^+	39.3		200	50	Yes	50-55
TM-1	TC-1	Strong acid		Polystyrene-DVB	16-50	H^+	33.8		130	50	Yes	50-55
	TA-1	Strong base			16-50	OH^-						
TMD-12		Strong acid		Polystyrene-DVB	16-50	H^+	12.0			45	Yes	50-55
		Strong base		Polystyrene-DVB	16-50							
NR-1	TC-1	Strong acid		Polystyrene-DVB	16-50	H^+	39.3		200	50	Yes	50-55
NR-6	NR1	Strong acid		Polystyrene-DVB	16-50	H^+	33.8		130	50	Yes	50-55
	NR2	Strong base		Polystyrene-DVB	16-50	OH^-						

Table 2.2-3 Physical and Chemical Characteristics of Gravex Resins

Manufacturer Name	Resin Type (Base-Anion) (Acid-Cation)	Functional Groups	Matrix	Mesh Range	Ionic Form	Total Exchange Capability (kgr/ft^3)	Shipping Weight (lb/ft^3)	Percent Regeneration	Moisture wt%
Gravex									
GR-1	Strong base	Quaternary ammonium	Polystyrene	20-50	OH^-	26	43	90	60
GR-2	Strong acid	Sulfonic	Polystyrene	20-50	H^+	39.2	53	95	55

A hydrogen ion (H^+) is generally the ionic form for cation resins although Lithium-7 is also found. The ionic form of anion resins is usually a hydroxide (OH^-) form although chloride (Cl^-) and carbonate (CO_3^-) are also used. The total ion-exchange capability of a resin is defined in terms of equivalent kilograins of calcium carbonate ($CaCO_3$) per cubic foot of resin. Cation resins, ranging in size from 0.45 mm to 0.60 mm are rated at 30 to 40 kgrs/ft^3. Anion resins, with sizes of 0.38 mm to 0.45 mm are rated at 15 to 30 kgrs/ft^3. Mixed bed resins such as IRN-217 in Table 2.2-1 may have exchange capabilities as low as 10 kgrs/ft^3 to 33.8 kgrs/ft^3 for TM-1 in Table 2.2-2.

Anion resins can generally withstand temperatures as high as 200 to 250°F. Cation resins can generally take temperatures ranging from 130 to 170°F. If oxygen is present in the water, the temperature limit may be as low as 100°F, depending on the type of resin.

When shipped, the resins are dry and fully swollen. Dry means without free water. The actual material contains between 42 and 55% water by weight. The shipping weight is between 43 and 53 lb/ft^3.

2.2.1.2 Application of Ion Exchange in LWRs

Table 2.2-4 provides a listing of major applications of deep bed demineralizers in LWRs. This information is based on data collected in this survey and from the preliminary and final safety analysis reports of several plants under design and construction.

From the standpoint of waste-volume generation, the most significant application of deep bed resins is in the radwaste system and in the condensate polishing system. When exhausted, radwaste resins are not regenerated. These resins are flushed from the vessel to a spent-resin tank. After decay they are packaged for burial. When condensate polishing demineralizers are exhausted, they are regenerated and reused. The chemicals from regeneration are sent to the liquid radwaste system for processing and disposal.

Specialized applications in PWRs include the chemical- and volume-control system and the boron-control system. Boron control is accomplished by the use of thermal regeneration demineralizers, which remove boron from the water at low temperatures, 50°F, and return the boron at higher temperatures, 140°F. Deep bed demineralizers are also used to clean up steam generator blowdown.

A limited number of PWRs and BWRs use deep bed demineralizers in the fuel-pool cleanup system.

2.2.1.3 Application of Ion Exchange in Fuel-Fabrication Facilities

Section 4.4 of this report discusses the operations of a typical fuel fabrication facility. Liquid wastes contain minimal quantities of recoverable uranium and are not generally processed other than as noted in Section 4.4.3.2. Chemical wastes from the processing of UF_6 into UO_2 pellets are pumped to settling ponds. The survey did not discover any reportable quantities of demineralizer ion-exchange resins.

Table 2.2-4 Application of Deep Bed Demineralizers in LWRs

BWRs	PWRs
Liquid radwaste system	Liquid radwaste system
Condensate polishing	Condensate polishing
Spent fuel pool cleanup	Spent fuel pool cleanup
Chemical laboratory	Chemical laboratory
	Chemical and volume control system
	Boron control system

2.2.1.4 Deep Bed Demineralizer Decontamination Factors

Decontamination factors for deep bed demineralizers are a function of the application of the unit and the type of resin used. Based on data from a number of operating plants using a broad spectrum of resins the NRC has established a set of nominal decontamination factors to use in the evaluation of safety analysis reports and environmental reports. These decontamination factors are given in Table 2.2-5 for BWRs (NRC, 1976b) and Table 2.2-6 for PWRs (NRC, 1976a).

2.2.2 Precoat Filter Wastes

2.2.2.1 Chemical and Physical Properties

Sludge from precoat filters is a combination of the original precoat material, the insolubles removed from the influent stream such as dirt, corrosion particles, and other suspended solids and floculating agents (filter aid) used to extend the filter's life. Numerous types of precoat material are available. The most commonly used are Solka-Floc, diatomaceous earth, and Powdex. Solka-Floc is a cellulose fiber derived from wood pulp. The material is practically ashless and when completely dry is 99.5% pure cellulose. Diatomaceous earth, or diatomaceous silica, is the siliceous skeletons of microscopic aquatic plants which lived a hundred thousand to several millions of years ago. When these diatoms died they settled to the bottom where the organic material decomposed leaving only the silica skeleton. Powdex is ground ion-exhange resin manufactured specifically for use on precoat filters. It consists primarily of broken chains

Table 2.2-5 Demineralizer Decontamination Factors of BWRs

Demin Type	Anion	Cs, Rb	Other
Mixed bed (H^+OH^-)			
Reactor coolant	10	2	10
Condensate	10	2	10
Clean water	$10^2(10)$ (1)	10(10)	$10^2(10)$
Dirty water (floor drains)	$10^2(10)$	2(10)	$10^2(10)$
Cation bed (H^+)			
Dirty water	1 (1)	10(10)	$10^2(10)$

1. Decontamination factors in parenthesis are for evaporator polishing and second demineralizer in series.

Table 2.2-6 Demineralizer Decontamination Factors of PWRs

Demin Type	Anion	Cs, Rb	Other
Mixed bed (Li_3BO_3)	10	2	10
Mixed bed (H^+OH^-)			
Condensate	10	2	10
Radwaste	$10^2(10)$ (1)	2(10)	$10^2(10)$
Boron recycle system feed (H_3BO_3)	10	2	10
Steam generator blowdown	$10^2(10)$	10(10)	$10^2(10)$
Cation bed	1 (1)	10(10)	10(10)
Anion bed	$10^2(10)$	1 (1)	1 (1)

1. Decontamination factors in parentheses are for evaporator polishing and second demineralizer in series.

of polystyrene cross linked with 8 to 10% divinyl benzene. The ion-exchange properties of Powdex are unaltered when used as a precoat on a filter element (see Section 3.3.1.1 for the descriptions of various precoat filters); however, the contact time between the water and ground resin is significantly less than it is in a deep bed demineralizer. The reduced contact time results in reduced ion-exchange capability.

Table 2.2-7 lists the basic physical and chemical characteristics of the three basic precoat materials, Solka-Floc, diatomaceous earth, and Powdex.

2.2.2.2 Applications of Precoat Filters in LWRs

Table 2.2-8 (on the same page as Table 2.2-7) shows the major uses of precoat filters in BWRs and PWRs. Precoat filters are used primarily in BWRs because of their larger capacity and higher crud-holding capacity compared to cartridge filters.

In BWRs the majority of the precoat filter sludge volume is from the radwaste system filters. The waste from the reactor water cleanup system contributes the largest share of the activity. Powdex is used in almost all reactor water cleanup systems. This accounts for the presence of the soluble fission products such as cesium. Most spent fuel pool cleanup systems use precoat filters, many with Powdex, which may account for the relatively high concentration of transuranics in precoat filter wastes.

The precoat filters used for condensate polishing are the largest of all the precoat filters used in LWRs. As such, they also have the largest quantity of precoat and dirt at the time of backwash. However, unless there is significant inleakage of water with high undissolved content to the condenser, these filters are backwashed only about once every month or two. On the other hand, backwashing of the reactor water cleanup system occurs twice a week.

2.2.2.3 Application of Precoat Filters to Fuel-Fabrication Facilities

As discussed in Section 4.4.3.2 some filter sludge results from the filtration of floor drains, wash basin drains, and floor scrubber solutions. These wastes consist primarily of the filter precoat and dirt. The only radioactivity present is U-235 and U-238.

With no reason to keep detailed records of these wastes there is no available data pertaining to its specific characteristics.

2.2.2.4 Precoat Filter Decontamination Factors

Included in Table 2.2-7 are the decontamination factors (DF) accepted by the NRC for use in safety analysis reports and environmental reports (NRC, 1976b). For Solka-Floc and diatomaceous earth the

Table 2.2-7 Chemical and Physical Properties of Precoat Filter Media

Filter Media	Primary Constituent	Dirt Holding Capacity	Removal Efficiency	Shipping Density (lb/ft^3)	Allowable Inlet pH Range	Allowable Inlet Temperature, ^oF	Decontamination Factors
Solka-Floc	Cellulose			40			1 - All nuclides
Diatomaceous Earth	Silica (SiO_2)			53			1 - All nuclides
Powdex	Polystyrene and divinyl benzene			50-55	0-14	170	Anions - 10(10) Cs, Rb - 2(10) Others - 10(10)

Table 2.2-8 Application of Precoat Filters in LWRs

BWRs	PWRs
Condensate polishing	Condensate polishing
Radwaste	Spent fuel pool cleanup
Reactor water cleanup	
Spent fuel pool cleanup	

DF is 1, which means the filter does not remove any radioactive contaminants. Some credit can be taken when Powdex is used.

2.2.3 Cartridge Filters

2.2.3.1 Chemical and Physical Properties

Unlike spent resins, filter sludge, and concentrator bottoms, cartridge filters cannot be pumped, poured, or processed. A cartridge filter which has reached the end of its life due to either plugging of the filter or high radiation from the contaminants is, by itself, the final product. Most cartridge filters are made from a few basic materials such as stainless steel, cotton, viscose rayon acrylic, rayon, glass, nylon, and cellulose. The material used to bind the fiber is usually a melamine or phenolic resin. The variety of cartridges on the market is too great to identify specific cartridges by the manufacturer's catalog numbers that were collected as part of the survey data. Instead Table 2.2-9 identifies the properties of filter cartridges typical of those used in LWRs. Cartridge filters as shown in Table 2.2-9 have micron ratings from 0.5 to 350 and can withstand temperatures up to 300°F.

2.2.3.2 Application of Cartridge Filters to LWRs

The commonest method of water treatment in LWRs is the use of cartridge filters. They are used mainly for small flows of relatively clean water where even small amounts of dirt could be harmful, such as flows used for mechanical seals on pumps. They are preferable to precoat filters for high operating temperatures.

Table 2.2-10 lists some of the most frequently reported uses of cartridge filters in both BWRs and PWRs. The only uses common to both types are in cleanup of the spent fuel pool and in liquid radwaste systems.

2.2.3.3 Application of Cartridge Filters to Fuel-Fabrication Facilities

The survey discovered no evidence indicating that cartridge filters are used in any of the liquid systems. The only use of filters is noted in Section 4.4.3.2.

2.2.3.4 Cartridge Filter Decontamination Factors

Attributing the removal of radioactive contaminants to cartridge filters is not permitted by the NRC (1976a). The decontamination factor is one. However, it is apparent from the data on the radiation dose rates (from packaged and sometimes unpackaged cartridge filters) gathered in this survey that the actual DF must be significant to account for the high dose rates.

Table 2.2-9 Chemical and Physical Properties of Cartridge Filter Elements

Cartridge Type	Materials of Construction	Micron Rating	Maximum Temperature, ^{o}F	Dimensions diameter/height (in.)	Effective Surface Area ft^2 per cartridge	Filtration Mechanism
Cuno: Micro-Klean II	Cellulose fiber with melamine resin	5,25,50	250	2-5/8 by 9-3/4 or 10	0.89	Depth
	Cellulose fiber with phenolic resin	5,25,50	250	2-5/8 by 9-3/4 or 10	0.89	Depth
	Acrylic fiber with phenolic resin	10,25,50,75	250	2-5/8 by 9-3/4 or 10	0.89	Depth
	Viscose rayon fiber with phenolic resin	125	250	2-5/8 by 9-3/4 or 10	0.89	Depth
	Acrylic viscose rayon fiber with phenolic resin	125	250	2-5/8 by 9-3/4 or 10	0.89	Depth
Cuno: Micro-Wynd II	Polypropylene media and matrix on tinned or stainless steel core	1 to 350	175	2-5/8 by 10	0.56	Depth
	Cotton media and matrix on tinned or stainless steel core	1 to 350	250	2-5/8 by 10	0.56	Depth
Pall: Epocel	Epoxy resin impregnated cellulose fiber on steel core (corrugated cellulose)	3,10,30	NA(1)	2-3/4 by 9-13/16	4.5	Surface
Filterite: Cotton Wound	Cotton winding on steel core	0.5 to 200	300	2-1/2 by 10	0.55	Depth

1. Not available.

2.2.4 Concentrator Bottoms

2.2.4.1 Chemical and Physical Properties of Boric Acid Concentrates

Concentrators used in most PWRs are not capable of concentrating boric acid solutions beyond 12.5% by weight. At higher concentrations the boric acid begins to crystallize. This is a condition for which these concentrators were not designed. Newer concentrators, aptly named crystallizers, are designed to handle concentrator bottoms of up to 50% boric acid by weight. Several plants have also reported concentrations of detergents, such as lithium hydroxide and anti-foam agents. The pH of the concentrator bottoms averages 6.5 and ranges from 4 to 9. The density of this waste is 1 g/cc increasing to approximately 1.6 g/cc when solidified.

Table 2.2-10 Application of Cartridge Filters in LWRs

BWRs	PWRs
Control rod drive filters	Reactor coolant purification filter
Spent fuel pool filter	Reactor coolant pump seal water injection
Laundry filter	Reactor coolant pump seal water return
Liquid radwaste	Spent fuel pool filter
	Boric acid filter
	Letdown demineralizer filter
	Liquid radwaste
	Aerated waste tank filter
	Evaporator condensate
	Reactor coolant drain filter
	Steam generator for blowdown holdup tank
	Reactor cavity cleanup

2.2.4.2 Chemical and Physical Properties of Sodium Sulfate Concentrates

Virtually all concentrator bottoms containing sodium sulfate are from BWRs although plant P18 does report some sodium sulfate in its concentrator bottoms. Current generation radwaste concentrators are limited to producing bottoms at 25% sodium sulfate by weight. Again, crystallizers will concentrate these wastes up to 50% sodium sulfate by weight. Other materials in the concentrator bottoms include defoaming agents, cleaning solutions, and possibly some excess sodium hydroxide from the regeneration of deep bed demineralizer resins. Sodium sulfate wastes range in

pH from 4.5 to 9 or above, and range in density from 1.2 g/ci (unsolidified) to 1.45 g/ci when solidified.

2.2.4.3 Chemical and Physical Properties of Miscellaneous Chemical Concentrates

These wastes are found in both PWRs and BWRs and are the result of the concentration of chemical wastes such as decontamination solutions, chemical laboratory drains, dilute water chemistry chemicals such as those to adjust the primary coolant system pH, and laundry waste water. When concentrated, these wastes are not limited by the crystallization of boric acid and can be concentrated up to 25% solids by weight. The presence of laundry soap in the waste may account for the low weight percent solids at facility P3. The pH of these wastes is between 8 and 10 although P3 shows a pH less than 6.5 probably because of the reported boron in the form of boric acid.

2.2.4.4 Applications of Concentrators in LWRs

Concentrators in BWRs are found only in the radwaste system where they are used to concentrate sodium sulfate (resulting from the regeneration of condensate demineralizers), decontamination solutions, laundry waste water, and floor drains. In BWRs that do not regenerate ion-exchange resins the chemical wastes consist of only the dilute decontamination solutions. Regeneration wastes and decontamination solutions may be collected in separate tanks to allow for independent pretreatment, such as pH adjustment, but they are usually processed through the same evaporator. The bottoms are then collected in the same tank making identification of the concentrators operating characteristics (relative to the waste processed) almost imposssible.

In PWRs, concentrators are used to concentrate boron in the boron-recovery system for storage and reuse, and to remove impurities from the steam generator blowdown. The condensed vapor from the steam generator blowdown may be either continuously released or recycled to the plant. Concentrators are also used in the liquid and chemical radwaste systems much the same as is the laundry waste concentrator of BWRs. Table 2.2-11 lists the systems in which concentrators are used in LWRs.

2.2.4.5 Applications of Concentrators in Fuel-Fabrication Facilities

Section 4.4 of this report is a discussion of the operations of a typical fuel-fabrication facility. Liquid wastes contain minimal quantities of recoverable uranium and are processed solely by filtration as noted in Section 4.4.3.2. Chemical wastes from the processing of UF_6 into UO_2 pellets are pumped to settling ponds. Concentrators are not used to process liquid wastes in fuel fabrication facilities.

Table 2.2-11 Application of Concentrators in LWRs

BWRs	PWRs
Chemical waste system	Boron recovery system
Floor drain waste system	Steam generator blowdown
Laundry waste system	Liquid waste system
	Chemical waste system
	Laundry waste system

2.2.4.6 Concentrator Decontamination Factors

Concentrators provide the best overall decontamination factors of any single piece of process equipment used for the removal of radioactive and nonradioactive contaminants from liquid process streams. The decontamination factors allowed by the NRC are a function of the application of the concentrator and the nuclide (NRC, 1976a, 1976b). Table 2.2-12 lists NRC-accepted decontamination factors for BWRs and PWRs.

Table 2.2-12 Concentrator Decontamination Factors

Application	All Nuclides Except Iodine	Iodine
PWR		
Miscellaneous radwaste	10^4	10^3
Boric acid recovery	10^3	10^2
Laundry wastes	10^2	10^2
BWR		
Miscellaneous	10^4	10^3
Laundry wastes	10^2	10^2

2.2.5 Reverse Osmosis

2.2.5.1 Chemical and Physical Properties of Reverse Osmosis Sludge

Reverse osmosis is a method of waste treatment that has found only limited acceptance in the nuclear industry. Units in operation are used to treat laundry waste water at Ginna and Point Beach. Plants under construction which have announced plans to

use reverse osmosis in the treatment of laundry wastes include Byron, Braidwood, Tyrone Energy Park, and Allens Creek.

The liquor, or concentrate, from operating reverse-osmosis units has not attracted sufficient attention for any data on its chemical or physical characteristics to be available. From its application it is apparent that the sludge will contain dirt, lint, detergent, and concentrated radioactive contamination.

2.2.5.2 Applications of Reverse Osmosis in LWRs

As mentioned in the previous section reverse-osmosis systems have been used on laundry wastes exclusively.

2.2.5.3 Application of Reverse Osmosis in Fuel-Fabrication Facilities

There are no reported uses of reverse-osmosis units for the treatment of liquid wastes in fuel-fabrication facilities.

2.2.5.4 Reverse-Osmosis Decontamination Factors

The DF accepted by the NRC for use in safety analysis reports and environmental reports is 30 for all nuclides in laundry wastes and 10 for all nuclides when used in other systems (NRC 1976a, 1976b). These accepted DF are called the system DF. The system DF is the ratio of the nuclide concentration in the inlet stream to those in the clean effluent. A membrane DF is also defined as the ratio of the nuclide concentrations in the concentrated liquor to those in the inlet. The relationship between the system DF and the membrane DF is not linear. It is a function of the percent recovery. The percent recovery is the ratio of the final clean effluent volume to the initial volume of waste to be treated. The relationship between the membrane DF and the system DF is given by:

$$DF_s = \frac{F}{1-(1-P)^{1/DF_m}}$$

where

DF_s = System DF
DF_m = Membrane DF
F = Percent recovery, $\left(\frac{\text{Effluent volume}}{\text{Inlet volume}}\right)$

Data collected at Ginna show individual isotopic membrane DF ranging from 60 to 12,000, with a calculated average of approximately 200. Point Beach data on gross membrane DF range from 16 to 1,500 with an average of 160. In calculating the overall system DF the NRC assumes an average membrane DF of 100 and 95% recovery. Substituting these values into the equation results in an average system DF of 30. For other types of waste the average membrane

DF are expected to be lower due to higher concentrations of iodine and cesium, both of which have membrane DFs lower than the average membrane DF (NRC, 1976a).

2.3 Radioactive Properties of Low- and Intermediate-Level Wastes

2.3.1 Radionuclide Concentrations in LWR Waste

According to the analysis in Section 4.2.3, radionuclide concentrations in LWR wastes range from 0.00064 Ci/ft^3 for PWR compactible trash to 0.65 Ci/ft^3 for PWR resins. Detailed breakdowns by isotope are listed in Table 2.3-1. The information in the following two sections is based on the concentrations in this table.

2.3.1.1 Heat Generation From Radioactive Decay

As the radionuclides in the waste decay, they emit alpha, gamma, and beta radiation ranging in energy from approximately 0.01 to 2.5 MeV. Assuming that all of the decay energy is absorbed within the waste and knowing the concentration of the majority of the isotopes in the waste (as shown in Table 2.3-2), the decay heat generation rates can be calculated. For LWR wastes, which contain fission products and activated corrosion products at concentrations of 640 μCi/ft^3 to 0.65 Ci/ft^3, the decay heat generation rates are between 1.1×10^{-6} Btu/hr/ft^3 to 1.7×10^{-2} Btu/hr/ft^3. For a 55-gallon drum this is equivalent to 0.006 Btu/hr per square foot of surface area. In 50-ft^3 liners or 100-ft^3 liners the heat rejection rate is 0.01 Btu/hr per square foot of surface area. These heat transfer rates, for unsolidified waste packages, are undetectable. When solidified these values will be even lower due to the lower concentration of radionuclides.

2.3.1.2 Short-Term Versus Long-Term Radionuclide Predominance

Of the several radionuclides that this and other studies (EPRI, 1978; EPA, 1977; and Phillips, 1977) have identified as the major nontransuranic radionuclides in LWR wastes, only three have half-lives greater than one year, Co-60 (5.26 years), Cs-134 (2.05 years), and Cs-137 (30.0 years). Table 2.3-1 shows the radionuclide concentrations of unsolidified waste at the time of packaging for burial, after 30 years of decay, and following 100 years of decay. After a period of 30 years only 1.9% of the Co-60 will remain, .0039% of the Cs-134 will remain, and 50% of the Cs-137 will remain. For periods of time longer than 30 years essentially only the Cs-137 will remain. After 100 years' decay the only remaining nontransuranic activity is approximately 10% of the original Cs-137 activity.

2.3.1.3 Transuranic Radionuclides in LWR Waste

Section 4.3 of this report presents specific data on the existence of transuranic radionuclides in LWR wastes. Based on plant analyses of LWR wastes it appears that transuranic radionuclides

Table 2.3-1 Radionuclide Concentrations in Unsolidified Waste at Packaging, With 30 Years Decay, and With 100 Years Decay

Nuclide	BWRs With Deep Bed CPS[1] (Ci/ft^3) Spent Resin	Precoat Filter Sludge	Concentrator Bottoms	BWRs With Precoat Filter CPS (Ci/ft^3) Compactible[2] Trash	Combined[3] Waste	PWRs Without CPS (Ci/ft^3) Spent Resin	Concentrator Bottoms	Compactible[4] Trash	PWRs With CPS (Ci/ft^3) Spent Resin	Concentrator Bottoms
At Packaging, Ci/ft^3										
Cr-51		.0063			.00012	.070	.023			.0029
Mn-54	.0028	.085	.0012	.00014	.00082	.065	.00051	.000026	.0031	.00020
Fe-59		.0052			.00012	.0053	.0021			.00020
Co-58	.00043	.0022		.000013	.0018	.11	.013	.00017	.0062	.00062
Co-60	.018	.19	.0052	.00028	.022	.081	.0003	.00013	.031	.00062
Zn-65	.00087	.0015	.00042		.019	.0021	.0003		.0031	.0002
Zr-95		.0011			.00047	.0032	.0054			.0002
Cs-134	.11	.023	.0099	.000039	.0056	.052	.0003	.000066	.19	.0002
Cs-137	.28	.051	.012	.000077	.0093	.23	.0003	.00013	.33	.0002
Other	.0017	.0048	.017	.00013	.0018	.026	.0067	.00012	.059	.0002
Total	.41	.37	.046	.00067	.061	.65	.051	.00064	.62	.005
After 30 years decay, μCi/ft^3										
Co-60	350.	3,600.	100.	54.	420.	1,600.	5.8	2.5	590.	12.
Cs-134	4.3	.9	.4	.0015	.2	2.	.01	.0026	7.4	.008
Cs-137	140,000.	25,000.	6,000.	38.5	4,600.	115,000.	150.	65.	165,000.	100.
Other	.01	.01	.01	.01	.01	.01	.01	.01	.01	.01
Total (Ci/ft^3)	1.4 (-1)	2.9 (-2)	6.1 (-3)	9.3 (-5)	5.0 (-3)	1.2 (-1)	1.5 (-4)	6.8 (-5)	1.7 (-1)	1.1 (-4)
% of total at packaging	34.	8.	13.	14.	8.	18.	1.0	11.	27.	2.
After 100 years decay, μCi/ft^3										
Cs-137 (Total)	28,000.	5,100.	1,200.	7.7	930.	23,000.	30.	13.	33,000.	20.
Other	.01	.01	.01	.01	.01	.01	.01	.01	.01	.01
% of total at packaging	7.0	1.4	2.6	1.1	1.5	3.5	1.0	2.0	5.3	1.0
Transuranics (μCi/ft^3)[5]	.45[6]	39.[7]	1.1[8]	NA[9]	210.[7]	47.[6]	2.5[10]	NA	3.0[6]	40.[10]

1. Condensate polishing system.
2. Applicable to both types of BWRs.
3. Composite of spent resin, filter sludge, and concentrator bottoms.
4. Applicable to both types of PWRs.
5. At packaging, with 30 years decay and with 100 years decay.
6. Based on an unsolidified density of .83 gm/cc.
7. Based on an unsolidified density of .86 gm/cc.
8. Based on an unsolidified density of 1.2 gm/cc.
9. Not available.
10. Based on an unsolidified density of 1.0 gm/cc.

Table 2.3-2 Decay Heat Generation Rates for LWR Wastes
($Btu/hr/ft^3$)

Waste Type	BWR		PWRs	
	Deep Bed CPS[1]	Precoat CPS	Without CPS	With CPS
Spent resin	1.0×10^{-2}	(2)	1.0×10^{-2}	1.7×10^{-2}
Concentrated liquids	1.1×10^{-3}	(2)	-	-
Filter sludge	5.3×10^{-3}	1.1×10^{-3}	$<5.3 \times 10^{-4}$	$<1.2 \times 10^{-4}$
Compactible trash	1.1×10^{-5}	1.1×10^{-5}	1.1×10^{-5}	1.1×10^{-5}

1. Condensate polishing system.
2. Included with filter sludge.

are not present in LWR wastes. This, however, is not true. Studies by EPRI (1978) and EPA (1977) both indicate that transuranic radionuclides are present in all forms of LWR waste in concentrations as high as 8.5×10^{-3} μCi/gm (BWR filter sludge) and as low as 1.9×10^{-5} μCi/gm (BWR spent resin). Data for compactible and noncompactible trash and PWR cartridge filters were not available.

The last line in Table 2.3-1 lists the estimated total concentrations of the transuranic radionuclides as identified in Section 4.3 of this report. The second-to-last line lists the total concentrations of the nontransuranic radionuclides, specifically Cs-137 after 100 years' decay.

Any specific statement about the relative magnitude of these numbers would be speculative. However, it can be said that following 100 years' decay the total concentration of nontransuranic radionuclides is approaching the concentration of the transuranic radionuclides.

2.3.2 Radionuclide Concentrations in Fuel-Fabrication Facility Wastes

The only radionuclides present in waste from fuel-fabrication facilities are U-238 and U-235. Detailed discussions of the nature of wastes from these facilities are given in Section 4.4. The concentration of two isotopes averages 12 μCi/ft^3.

2.3.2.1 Transuranic Radionuclides in Fuel-Fabrication Facility Wastes

No specific information is available on what percentage of the total activity in the waste is attributable to either U-235 or U-238. However, with enrichments averaging between three and four percent U-235 in U-238 the activity in the waste should exhibit the same proportions. Thus the U-235 concentration may be approximately 4×10^{-1} μCi/ft^3 and the U-238 concentration would then be approximately 11 μCi/ft^3.

2.3.2.2 Heat Generation From Radionuclide Decay

At the uranium concentrations noted in the previous section the heat generated from radionuclide decay will be less than 2×10^{-7} Btu/hr.

2.3.2.3 Long-Term Radionuclide Predominance

The half-life of U-235 is 7.1×10^8 years and that of U-238 is 4.51×10^9 years. The daughter, granddaughter, and progeny of the U-235 and U-238 all have shorter half-lives than the original parent nuclides. Thus, for time periods of less than a few million years the predominant radionuclides will continue to be U-235 and U-238.

2.4 Classifications of Radioactive Wastes

There are presently four classifications of waste as defined by 10 CFR 71, Packaging for Transport and Transportation of Radioactive Material Under Certain Conditions. These four classifications are, Low Specific Activity (LSA), Type A, Type B, and Large Quantities. Classifications are determined by the total activity of radionuclides in each of seven transport groups. Table 2.4-1 gives the transport group assigned to almost 300 radionuclides. Any radionuclide that is not included in the table is assigned to one of the groups in accordance with Table 2.4-2. Reactor wastes, as shown by this report in Section 4.2.3, are predominantly Mn-54, Co-58, Co-60, Cs-134, and Cs-137. Mn-54 and Co-58 belong to transport group IV and Co-60, Cs-134, and Cs-137 belong to transport group III. Mixed fission products have a separate listing in the table and are assigned to transport group II.

2.4.1 Low Specific Activity (LSA)

As defined by Title 10 CFR 71, in 71.4(g), low specific activity is:

1. Uranium or thorium ores and physical or chemical concentrates of those ores;

2. Unirradiated natural or depleted uranium or unirradiated natural thorium;

3. Tritium oxide in aqueous solutions, provided the concentration does not exceed 5.0 millicuries per milliliter;

4. Material in which the activity is essentially uniformly distributed and in which the estimated average concentration per gram of contents does not exceed:

   i. .0001 millicurie of Group I radionuclides; or

   ii. .005 millicurie of Group II radionuclides; or

   iii. .3 millicurie of Group III or IV radionuclides.

Note: This includes, but is not limited to, materials of low radioactivity concentration such as residues or solutions from chemical processing; wastes such as building rubble, metal, wood, and fabric scrap; glassware, paper, and cardboard; solid or liquid plant waste, sludges, and ashes.

Table 2.4-1 Radionuclide Transport Groups by Nuclide

Element[1]	Radionuclide[2]	Group	Element[1]	Radionuclide[2]	Group
Actinium (89).....	Ac 227.....	I	Californium (98)...	Cf 249.....	I
	Ac 228.....	I		Cf 250.....	I
Americium (95).....	Am 241.....	I		Cf 252.....	I
	Am 243.....	I	Carbon (6).....	C 14.....	IV
Antimony (51).....	Sb 122.....	IV	Cerium (58).....	Ce 141.....	IV
	Sb 124.....	III		Ce 143.....	IV
	Sb 125.....	III		Ce 144.....	III
Argon (18).....	Ar 37.....	VI	Cesium (55).....	Cs 131.....	IV
	Ar 41.....	II		Cs 134 m.....	III
	Ar 41 (uncompressed)[3].....	V		Cs 134.....	III
				Cs 135.....	IV
Arsenic (33).....	As 73.....	IV		Cs 136.....	IV
	As 74.....	IV		Cs 137.....	III
	As 76.....	IV	Chlorine (17)	Cl 36.....	III
	As 77.....	IV		Cl 38.....	IV
Astatine (85).....	At 211.....	III	Chromium (24).....	Cr 51.....	IV
Barium (56).....	Ba 131.....	IV	Cobalt (27).....	Co 56.....	III
	Ba 133.....	II		Co 57.....	IV
	Ba 140.....	III		Co 58 m.....	IV
Berkelium (97).....	Bk 249.....	I		Co 58.....	IV
Beryllium (4).....	Be 7.....	IV		Co 60.....	III
Bismuth (83).....	Bi 206.....	IV	Copper (29).....	Cu 64.....	IV
	Bi 207.....	III	Curium (96).....	Cm 242.....	I
	Bi 210.....	II		Cm 243.....	I
	Bi 212.....	III		Cm 244.....	I
Bromine (35).....	Br 82.....	IV		Cm 245.....	I
Cadmium (48).....	Cd 109.....	IV		Cm 246.....	I
	Cd 115 m.....	III	Dysprosium (66)	Dy 154.....	III
	Cd 115.....	IV		Dy 165.....	IV
Calcium (20).....	Ca 45.....	IV		Dy 166.....	IV
	Ca 47.....	IV			

Table 2.4-1 Radionuclide Transport Groups by Nuclide (Cont'd)

Element[1]	Radionuclide[2]	Group
Erbium (68).....	Er 169.....	IV
	Er 171.....	IV
Europium (63).....	Eu 150.....	III
	Eu 152 m.....	IV
	Eu 152.....	III
	Eu 154.....	II
	Eu 155.....	IV
Fluorine (9).....	F 18.....	IV
Gadolinium (64)....	Gd 153.....	IV
	Gd 159.....	IV
Gallium (31).....	Ga 67.....	III
	Ga 72.....	IV
Germanium (32).....	Ge 71.....	IV
Gold (79).....	Au 193.....	III
	Au 194.....	III
	Au 195.....	III
	Au 196.....	IV
	Au 198.....	IV
	Au 199.....	IV
Hafnium (72).....	Hf 181.....	IV
Holmium (67).....	Ho 166.....	IV
Hydrogen (1).....	H 3 (see tritium).	
Indium (49).....	In 113 m.....	IV
	In 114 m.....	III
	In 115 m.....	IV
	In 115.....	IV
Iodine (53).....	I 124.....	III
	I 125.....	III
	I 126.....	III
	I 129.....	III
	I 131.....	III
	I 132.....	IV
	I 133.....	III
	I 134.....	IV
	I 135.....	IV
Iridium (77).....	Ir 190.....	IV
	Ir 192.....	III
	Ir 194.....	IV
Iron (26).....	Fe 55.....	IV
	Fe 59.....	IV
Krypton (36).....	Kr 85 m.....	III
	Kr 85 m (uncompressed)[3].....	V
	Kr 85.....	III
	Kr 85 (uncompressed)[3].....	VI
	Kr 87.....	II
	Kr 87 (uncompressed)[3].....	V
Lanthanum (57).....	La 140.....	IV
Lead (82).....	Pb 203.....	IV
	Pb 210.....	II
	Pb 212.....	II
Lutecium (71).....	Lu 172.....	III
	Lu 177.....	IV
Magnesium (12).....	Mg 28.....	III
Manganese (25).....	Mn 52.....	IV
	Mn 54.....	IV
	Mn 56.....	IV
Mercury (80).....	Hg 197 m.....	IV
	Hg 197.....	IV
	Hg 203.....	IV
Mixed fission products MFP.....	.....	II
Molybdenum (42)	Mo 99.....	IV
Neodymium (60)	Nd 147.....	IV
	Nd 149.....	IV
Neptunium (93).....	Np 237.....	I
	Np 239.....	I

Table 2.4-1 Radionuclide Transport Groups by Nuclide (Cont'd)

Element[1]	Radionuclide[2]	Group
Nickel (28)	Ni 56	III
	Ni 59	IV
	Ni 63	IV
	Ni 65	IV
Niobium (41)	Nb 93 m	IV
	Nb 95	IV
	Nb 97	IV
Osmium (76)	Os 185	IV
	Os 191 m	IV
	Os 191	IV
	Os 193	IV
Palladium (46)	Pd 103	IV
	Pd 109	IV
Phosphorus (15)	P 32	IV
Platinum (78)	Pt 191	IV
	Pt 193	IV
	Pt 193 m	IV
	Pt 197 m	IV
	Pt 197	IV
Plutonium (84)	Pu 238 (F)	I
	Pu 239 (F)	I
	Pu 240	I
	Pu 241 (F)	I
	Pu 242	I
Polonium (84)	Po 210	I
Potassium (19)	K 42	IV
	K 43	III
Praseodymium (59)	Pr 142	IV
	Pr 143	IV
Promethium (61)	Pm 147	IV
	Pm 149	IV
Protactinium (91)	Pa 230	I
	Pa 231	I
	Pa 233	II
Radium (88)	Ra 223	II
	Ra 224	II
	Ra 226	I
	Ra 228	I
Radon (86)	Rn 220	IV
	Rn 222	II
Rhenium (75)	Re 183	IV
	Re 186	IV
	Re 187	IV
	Re 188	IV
	Re Natural	IV
Rhodium (45)	Rh 103 m	IV
	Rh 105	IV
Rubidium (37)	Rb 86	IV
	Rb 87	IV
	Rb Natural	IV
Ruthenium (44)	Ru 97	IV
	Ru 103	IV
	Ru 105	IV
	Ru 106	III
Samarium (62)	Sm 145	III
	Sm 147	III
	Sm 151	IV
	Sm 153	IV
Scandium (21)	Sc 46	III
	Sc 47	IV
	Sc 48	IV
Selenium (34)	Se 75	IV
Silicon (14)	Si 31	IV
Silver (47)	Ag 105	IV
	Ag 110 m	III
	Ag 111	IV
Sodium (11)	Na 22	III
	Na 24	IV

Table 2.4-1 Radionuclide Transport Groups by Nuclide (Cont'd)

Element[1]	Radionuclide[2]	Group
Strontium (38).....	Sr 85 m.....	IV
	Sr 85.....	IV
	Sr 89.....	III
	Sr 90.....	II
	Sr 91.....	III
	Sr 92.....	IV
Sulfur (16).....	S 35.....	IV
Tantalum (73).....	Ta 182.....	III
Technetium (43)....	Tc 96 m.....	IV
	Tc 96.....	IV
	Tc 97 m.....	IV
	Tc 97.....	IV
	Tc 99 m.....	IV
	Tc 99.....	IV
Tellurium (52).....	Te 125 m.....	IV
	Te 127 m.....	IV
	Te 127.....	IV
	Te 129 m.....	III
	Te 129.....	IV
	Te 131 m.....	III
	Te 132.....	IV
Terbium (65).....	Tb 160.....	III
Thallium (81).....	Tl 200.....	IV
	Tl 201.....	IV
	Tl 202.....	IV
	Tl 204.....	III
Thorium (90).....	Th 227.....	II
	Th 228.....	I
	Th 230.....	I
	Th 231.....	I
	Th 232.....	III
	Th 234.....	II
	Th Natural.....	III
Thulium (69).....	Tm 168.....	III
	Tm 170.....	III
	Tm 171.....	IV
Tin (50).....	Sn 113.....	IV
	Sn 117 m.....	III
	Sn 121.....	III
	Sn 125.....	IV
Tritium (1).....	H 3.....	IV
	H 3 (as a gas, as luminous paint, or absorbed on solid material).	VII
Tungsten (74).....	W 181.....	IV
	W 185.....	IV
	W 187.....	IV
Uranium (92).....	U 230.....	II
	U 232.....	I
	U 233 (F).....	II
	U 234.....	II
	U 235 (F).....	III
	U 236.....	II
	U 238.....	III
	U Natural.....	III
	U Enriched (F)....	III
	U Depleted.....	III
Vanadium (23).....	V 48.....	IV
	V 49.....	III
Xenon (54).....	Xe 125.....	III
	Xe 131 m.....	III
	Xe 131 m (uncompressed)[3]	V
	Xe 133.....	III

Table 2.4-1 Radionuclide Transport Groups by Nuclide (Cont'd)

Element[1]	Radionuclide[2]	Group	Element[1]	Radionuclide[2]	Group
Xenon (54) cont'd..	Xe 133 (uncompressed)[3]	VI	Yttrium (39).cont'd.	Y 91.....	III
	Xe 135.....	II		Y 92.....	IV
	Xe 135 (uncompressed)[3]	V		Y 93.....	IV
Ytterbium (70).....	Yb 175.....	IV	Zinc (30).....	Zn 65.....	IV
Yttrium (39).....	Y 88.....	III		Zn 69 m.....	IV
	Y 90.....	IV		Zn 69.....	IV
	Y 91 m.....	III	Zirconium (40).....	Zr 93.....	IV
				Zr 95.....	III
				Zr 97.....	IV

1. Atomic number shown in parentheses.
2. Atomic weight shown after the radionuclide symbol.
   m - Metastable state.
   (F) - Fissile material.
3. Uncompressed means at a pressure not exceeding one atmosphere.

Table 2.4-2 Radionuclide Transport Groups by Atomic Number

Radionuclide	Radioactive half-life		
	0 to 1000 days	1000 days to 10^6 years	Over 10^6 years
Atomic number 1-81	Group III	Group II	Group III
Atomic number 82 and over	Group I	Group I	Group III

5. Objects of nonradioactive material externally contaminated with radioactive material, provided that the radioactive material is not readily dispersible and the surface contamination, when averaged over an area of 1 square meter, does not exceed 0.0001 millicurie (220,000 disintegrations per minute) per square centimeter of Group I radionuclides or 0.001 millicurie (2,200,000 disintegrations per minute) per square centimeter of other radionuclides.

2.4.2 Type A and Type B Waste

Title 10 CFR 71, in part 71.4(q), defines Type A and Type B waste as follows:

Type A quantity and type B quantity means a quantity of radioactive material the aggregate radioactivity of which does not exceed that specified in the following information:

Transport groups	Type A quantity (Ci)	Type B quantity (Ci)
I	0.001	20
II	0.05	20
III	3	200
IV	20	200
V	20	5,000
VI and VII	1,000	50,000
Special form	20(1)	5,000

1. Except that for californium-252, the limit is 2 Ci.

2.4.3 Large Quantities

Title 10 CFR 70 defines large quantities, in 71.4(f), as any "quantity of radioactive material, the aggregate radioactivity of which exceeds any one of the following:

1. For transport groups as defined in paragraph (p) of this section:

   i. Group I or II radionuclides: 20 curies;

   ii. Group III or IV radionuclides: 200 curies;

   iii. Group V radionuclides: 5,000 curies;

   iv. Group VI or VII radionuclides: 50,000 curies; and

2. For special form material as defined in paragraph (o) of this section: 5,000 curies."

Paragraph (p) states that the term "transport group means any one of the seven groups into which radionuclides in normal form are classified, according to their toxicity and their relative potential hazard in transport...." Paragraph (p) of Title 10 CFR 70 continues as follows:

> 2. For mixtures of radionuclides the following shall apply:
>
> i. If the identity and respective activity of each radionuclide are known, the permissible activity of each radionuclide shall be such that the sum, for all groups present, of the ratio between the total activity for each group to the permissible activity for each group will not be greater than unity.
>
> ii. If the groups of radionuclides are known but the amount in each group cannot be reasonably determined, the mixture shall be assigned to the most restrictive group present.
>
> iii. If the identity of all or some of the radionuclides cannot be reasonably determined, each of those unidentified radionuclides shall be considered as belonging to the most restrictive group which cannot be positively excluded.
>
> iv. Mixtures consisting of a single radioactive decay chain where the radionuclides are in the naturally occurring proportions shall be considered as consisting of a single radionuclide. The group and activity shall be that of the first member present in the chain, except that if a radionuclide "x" has a half-life longer than that of that first member and an activity greater than that of any other member, including the first, at any time during transportation, the transport group of the nuclide "x" and the activity of the mixture shall be the maximum activity of that nuclide "x" during transportation.

2.4.4 Proposed Changes to 10 CFR Part 71

Proposed changes to 49 CFR 127,171-177 (FR Vol 44 No. 5 1/8/79) and similar proposed changes expected in 10 CFR 71 will eliminate the seven transport groups and establish curie limits on each radionuclide based on its own toxicity. The authors of a paper in which these proposed changes are discussed (Weller, 1978) state that one of the effects of this change will be to increase the quantity of less toxic members that can be shipped in a given container. For example, the 3 curie limit for Co-60 in Type A packages will increase to 7 curies. On the other hand, use of existing DOT Spec 55 packages will no longer be authorized. Other changes are a revised definition of LSA material and a new classification called "low level solids" (LLS). The changes to the LSA definition will essentially eliminate bulk liquid shipments as LSA material. For both

the LSA and the LLS categories the specific activity limits are tied to the individual isotopic curie limits. Furthermore, the LLS definition considers the leachability characteristics of the solidification agent used to immobilize the radioactive waste. Under these revised regulations most of the LLS material from power reactors would be shipped as Type A material rather than LSA as in the past.

CHAPTER 2 REFERENCES

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3. WASTE TREATMENT AND PACKAGING

3.1 Introduction

Current waste management practices are aimed at removing radioactive contaminants from liquid waste streams so the remaining purified water may be recycled to the plant or discharged into the environment. Federal and state statutes define the quality of water that is acceptable for discharge. To achieve these standards nuclear power plants have been designed with special water treatment systems to remove radioactive and chemical contaminants.

The radioactive contaminants in nuclear power plants have two sources. They may originate as fission products in the nuclear fuel, or they may develop from corrosion products that plate out in the reactor core. These corrosion products are irradiated in the high neutron flux, activated to a radioactive state, and then break loose. As these crud particles break loose, and the fission products escape from the fuel, the primary system becomes contaminated. Leakage from pump seals, valve stems, broken heat exchanger tubes, and processing through primary system cleanup (reactor water) permits these radioactive contaminants to spread to other systems in the plant. In order to minimize the spread of these contaminants and collect the waste byproducts of the reactor water cleanup system, plants are designed with a radioactive waste management system. This system collects, processes, and stores all liquid and solid radioactive waste (including radioactively contaminated chemical waste) produced in the plant. After these radioactive wastes have been collected, processed, and stored, they are disposed of. Processed liquid wastes--once they meet federal and state standards--are discharged into the environment through a natural body of water or recycled to the plant. Solid waste is shipped to a shallow land burial site for burial.

In fuel-fabrication plants various chemical streams used in processing uranium into fuel become contaminated with uranium (refer to Section 4.4.2). Chemical wastes consisting of ammonium fluoride, ammonium nitrate, and ammonium hydroxide are pumped to settling ponds and are not treated to remove the residual uranium. Nonchemical wastes from floor drains, wash basin drains, and floor cleaning solution are processed through filters to collect recoverable uranium. If the uranium concentration is high enough to permit economical uranium recovery the filter sludge is sent to recovery. Otherwise the sludge is packaged for burial at a shallow land burial site.

3.2 Liquid Waste Processing, A Starting Point

3.2.1 Light Water Reactor

Attempting to describe a typical liquid radwaste management system (LRMS) would be futile. Not only has design philosophy changed,

but each system reflects the personal characteristics of the utility for which the system was designed and/or the design preference of the engineer who designed it. This section gives brief descriptions of the various systems that process liquid wastes and ultimately contribute to the volume and activity associated with PWR solid radwaste.

3.2.1.1 Pressurized Water Reactor

3.2.1.1.1 Chemical and Volume Control System

In order to maintain water quality in the primary system a small stream is taken from the discharge side of one of the reactor recirculation pumps, cooled, and processed through the chemical and volume control system (CVCS). The purpose of the CVCS is to remove impurities, such as corrosion products and fission products, from the reactor water in order to prevent their buildup in the primary system. These contaminants are removed by filtration and/or demineralization.

3.2.1.1.2 Boron Recovery System

Connected to the CVCS is the boron recovery system (BRS). The BRS consists of a number of demineralizers, connected in parallel, which can be used either to decrease or increase the boron concentration in the primary system.

3.2.1.1.3 Steam Generator Blowdown System

The PWR is an indirect cycle. The primary system is kept under sufficient pressure to prevent vapor formation while giving up its heat energy to a secondary system in a steam generator. The secondary side pressure is such that boiling does occur. This steam is used to roll the turbines, and after it is condensed it is pumped back to the steam generator. Most operating plants bleed off a portion of this secondary system flow to prevent the buildup of corrosion products, scale, and impurities from condenser cooling water inleakage. This bleed stream is processed through a filter and/or demineralizer and discharged if possible. If any of the steam generator tubes leak, radioactive contamination from the primary system will enter the secondary system, thus contaminating it. More recent designs have eliminated this bleed line in favor of full-flow condensate filtration systems located down-stream of the turbine condenser. As these filters and demineralizers become loaded with crud and radioactive ions the pressure drop across the units and the dose rates in the vicinity of the units will increase to preset limits at which time the filter precoat or cartridge or demineralizer resin will be replaced. The expended material will be transported to the solid radwaste system for disposal.

3.2.1.1.4 Spent Fuel Pool Cleanup System

The spent fuel pool cleanup system is designed to process a small portion of the fuel pool water to remove activated corrosion products, fission products, and dirt which settles on the surface of the pool. The activated corrosion products enter the pool as crud stuck to the fuel. Fission products continuously leak from the fuel as long as it is in the pool. Removal of the radioactive contaminants is necessary to minimize the doses to plant personnel working in the vicinity of the pool and to maintain water clarity. With few exceptions spent fuel pool cleanup systems consist of a cartridge filter backed up by a nonregenerative deep bed demineralizer.

3.2.1.1.5 Liquid Radwaste Management System

The liquid radwaste management system (LRMS) collects, processes, and prepares liquid wastes for release to the environment or for recycling to the plant. Inputs to the LRMS consist of various equipment leakages, floor drain wastes, chemical wastes, and laundry waste water. Liquid wastes are collected based on the expected quality of the water.

Equipment leakages and the discharge from the floor drain sumps are collected in the waste holdup or collection tank. After processing, these wastes are collected in the waste monitor or sample tanks and either released to the environment, or recycled to the plant for reuse or further processing.

In most operating PWRs the liquid radwaste system designed for the processing of these wastes almost always includes an evaporator, and in a few plants the evaporator may be the only processing equipment used, as is the case with Maine Yankee. The most common combination of equipment is a cartridge filter followed by an evaporator and a polishing demineralizer for the condensate. Tables C-3 and C-4 in Appendix C show the equipment available in each plant surveyed for this study, and for a number of plants now under construction or which have not been operating long enough to have been included in the survey.

3.2.1.1.6 Chemical Waste System

Chemical wastes from the regeneration of demineralizer resins, chemical decontamination of equipment, or from the chemical laboratory are collected separately from other waste, and in some plants they are collected separately from each other. After processing, chemical wastes are sampled, and discharged or recycled. The most frequently used process method for chemical wastes is straight evaporation. Some plants, Ginna and Palisades for example, have installed evaporators specifically for chemical wastes whereas many other plants are using the same evaporator that is used for liquid wastes; Robinson and Zion are examples. The tendency with new plants is to separate evaporators for chemical waste as can be seen in Tables C-3 and C-4 in Appendix C.

3.2.1.1.7 Laundry Waste System

Laundry wastes, which also are collected separately, are processed, collected in a sample or monitor tank, and discharged or recycled. Most designs use only a cartridge filter to remove undissolved solids because the radionuclide concentrations normally are well below discharge limits before reaching the filter. Reverse osmosis is downstream from a cartridge filter at Ginna, and it is planned to be used in the same manner at Byron 1 & 2. Some plants, such as Surry and Robinson, process their laundry wastes through the same equipment as their liquid and chemical wastes. The expected daily volumes of waste going into the liquid waste system, the chemical waste system, and the laundry system for a reference 1,000-MWe PWR are shown in Table 3.2-1 (NRC, 1976a).

3.2.1.2 Boiling Water Reactor

3.2.1.2.1 Reactor Water Cleanup System

Water quality in the primary BWR system is maintained by the reactor water cleanup system. Once cooled to system operating temperature the water is routed through the cleanup equipment, reheated to the primary system operating temperature, and then returned to the primary system. Most operating BWRs use a filter/demineralizer alone, or in combination with a cartridge filter, precoat filter, or a deep bed demineralizer. Newer plants appear to be using only the filter/demineralizer.

3.2.1.2.2 Spent Fuel Pool Cleanup System

Spent fuel pool cleanup systems in BWRs use a greater variety of process equipment than those used in PWRs. The cleanup methods in order of greatest use to least use are the filter/demineralizer, precoat filter, cartridge filter, and the deep bed demineralizer preceded by a cartridge filter.

3.2.1.2.3 Condensate Polishing System

The BWR cycle is a direct steam cycle. High quality steam produced within the reactor vessel is routed directly to the turbines. The steam is then condensed in the turbine condenser and returned to the reactor vessel through the feedwater system. Reactor water quality is maintained by a partial- or full-flow condensate polishing system. Early BWRs were designed either with regenerative or nonregenerative deep bed demineralizers for condensate polishing whereas plants of later design tended more towards filter/demineralizers. Of the plants now under construction the majority will be using deep bed demineralizers of the regenerative type in conjunction with ultrasonic resin cleaners to extend the time between regenerations.

Tables C-1 and C-2 in Appendix C show which pieces of equipment are used in operating plants and which have been selected for

Table 3.2-1 PWR Liquid Radwaste Inputs[1]

Source	Flow rate (gal/d)	Fraction of Primary Coolant Activity[2]
Liquid waste system		
Containment building sump	40	1
Auxiliary building floor sump	200	0.1
Laboratory drains	400	0.002
Sampling drains[3]	35	1
Miscellaneous	700	0.01
Turbine building floor drains	7,200	(4)
Subtotal	8,575	
Chemical waste[5]		
Condensate demineralizer regenerant waste	3,400	(4)
Laundry waste	450	(6)
Total with deep bed condensate demineralizers	9,025	
Total without deep bed condensate demineralizers	12,424	

1. Taken from NRC, 1976a.
2. See Tables 3.2-2 and 3.2-3 for primary and secondary coolant activities.
3. 15 gal/d for continuous purge cycle.
4. Calculated by PWR-GALE Code.
5. Only for plants with deep bed condensate demineralizers.
6. See Table 3.2-4.

Table 3.2-2 PWR Primary and Secondary Coolant Activities (U-Tube Steam Generators) [1]

Isotope	Reactor Coolant [3] (μCi/g)	Secondary Coolant (μCi/g) [2]			
		Water [4]		Steam [5]	
		Phosphate	Volatile	Phosphate	Volatile
Noble gases					
Kr-83m	2.1(-2) [6]	Nil	Nil	5.8(-9)	5.8(-9)
Kr-85m	1.1(-1)	Nil	Nil	3.1(-8)	3.1(-8)
Kr-85	1.5(-1)	Nil	Nil	4.2(-8)	4.2(-8)
Kr-87	6.0(-2)	Nil	Nil	1.6(-8)	1.6(-8)
Kr-88	2.0(-1)	Nil	Nil	5.5(-8)	5.5(-8)
Kr-89	5.0(-3)	Nil	Nil	1.4(-9)	1.4(-9)
Xe-131m	1.1(-1)	Nil	Nil	3.1(-8)	3.1(-8)
Xe-133m	2.2(-1)	Nil	Nil	6.2(-8)	6.2(-8)
Xe-133	1.8(+1)	Nil	Nil	5.0(-6)	5.0(-6)
Xe-135m	1.3(-2)	Nil	Nil	3.6(-9)	3.6(-9)
Xe-135	3.5(-1)	Nil	Nil	9.7(-8)	9.7(-8)
Xe-137	9.0(-3)	Nil	Nil	2.5(-9)	2.5(-9)
Xe-138	4.4(-2)	Nil	Nil	1.2(-8)	1.2(-8)
Halogens					
Br-83	4.8(-3)	1.5(-7)	6.9(-8)	1.5(-9)	6.9(-10)
Br-84	2.6(-3)	2.0(-8)	1.5(-8)	2.0(-10)	1.5(-10)
Br-85	3.0(-4)	2.0(-10)	2.0(-10)	2.0(-12)	2.0(-12)
I-130	2.1(-3)	2.5(-7)	4.6(-8)	2.5(-9)	4.6(-10)
I-131	2.7(-1)	1.1(-4)	6.8(-6)	1.1(-6)	6.8(-8)
I-132	1.0(-1)	1.1(-5)	1.9(-6)	1.1(-7)	1.9(-8)
I-133	3.8(-1)	6.5(-5)	8.9(-6)	6.5(-7)	8.9(-8)
I-134	4.7(-2)	5.7(-7)	3.8(-7)	5.7(-9)	3.8(-9)
I-135	1.9(-1)	1.4(-5)	3.8(-6)	1.4(-7)	3.8(-8)
Cs, Rb					
Rb-86	8.5(-5)	4.0(-8)	4.4(-9)	4.0(-11)	4.4(-12)
Rb-88	2.0(-1)	8.0(-7)	7.4(-7)	8.0(-10)	7.4(-10)
Cs-134	2.5(-2)	1.2(-5)	1.3(-6)	1.2(-8)	1.3(-9)
Cs-136	1.3(-2)	5.0(-6)	6.7(-7)	5.0(-9)	6.7(-10)
Cs-137	1.8(-2)	8.0(-6)	9.4(-7)	8.0(-9)	9.4(-10)
Water activation products					
N-16	4.0(+1)	1(-6)	1(-6)	1(-7)	1(-7)

Table 3.2-2 PWR Primary and Secondary Coolant Activities (U-Tube Steam Generators)[1] (Cont'd)

Isotope	Reactor Coolant[3] (μCi/g)	Secondary Coolant (μCi/g)[2]			
		Water[4]		Steam[5]	
		Phosphate	Volatile	Phosphate	Volatile
Tritium					
H-3	1(0)	1(-3)	1(-3)	1(-3)	1(-3)
Other nuclides					
Cr-51	1.9(-3)	8(-7)	9(-8)	8(-10)	9(-11)
Mn-54	3.1(-4)	2(-7)	2(-8)	2(-10)	2(-11)
Fe-55	1.6(-3)	7(-7)	8(-8)	7(-10)	8(-11)
Fe-59	1.0(-3)	5(-7)	6(-8)	5(-10)	6(-11)
Co-58	1.6(-2)	7(-6)	8(-7)	7(-9)	8(-10)
Co-60	2.0(-3)	9(-7)	9(-8)	9(-10)	9(-11)
Sr-89	3.5(-4)	2(-7)	2(-8)	2(-10)	2(-12)
Sr-90	1.0(-5)	5(-9)	4(-10)	5(-12)	4(-13)
Sr-91	6.5(-4)	6(-8)	2(-8)	6(-11)	2(-11)
Y-90	1.2(-6)	2(-9)	8(-11)	2(-12)	8(-14)
Y-91m	3.6(-4)	3(-8)	1(-8)	3(-11)	1(-11)
Y-91	6.4(-5)	3(-8)	3(-9)	3(-11)	3(-12)
Y-93	3.4(-5)	4(-9)	1(-9)	4(-12)	1(-12)
Zr-95	6.0(-5)	3(-8)	4(-9)	3(-11)	4(-12)
Nb-95	5.0(-5)	3(-8)	4(-9)	3(-11)	4(-12)
Mo-99	8.4(-2)	3(-5)	4(-6)	3(-8)	4(-9)
Tc-99m	4.8(-2)	3(-5)	3(-6)	3(-8)	3(-9)
Ru-103	4.5(-5)	2(-8)	2(-9)	2(-11)	2(-12)
Ru-106	1.0(-5)	5(-9)	4(-10)	5(-12)	4(-13)
Rh-103m	4.5(-5)	2(-8)	2(-9)	2(-11)	2(-12)
Rh-106	1.0(-5)	5(-9)	4(-10)	5(-12)	4(-10)
Te-125m	2.9(-5)	9(-9)	1(-9)	9(-12)	1(-12)
Te-127m	2.8(-4)	9(-8)	1(-8)	9(-11)	1(-11)
Te-127	8.5(-4)	2(-7)	3(-8)	2(-10)	3(-11)
Te-129m	1.4(-3)	6(-7)	6(-8)	6(-10)	6(-11)
Te-129	1.6(-3)	6(-7)	6(-8)	6(-10)	6(-11)
Te-131m	2.5(-3)	5(-7)	1(-7)	5(-10)	1(-10)
Te-131	1.1(-3)	5(-7)	2(-8)	5(-10)	2(-11)
Te-132	2.7(-2)	8(-6)	1(-6)	8(-9)	1(-9)
Ba-137m	1.6(-2)	8(-6)	9(-7)	8(-9)	9(-10)
Ba-140	2.2(-4)	9(-8)	1(-8)	9(-11)	1(-11)

Table 3.2-2 PWR Primary and Secondary Coolant Activities (U-Tube Steam Generators)[1] (Cont'd)

Isotope	Reactor Coolant[3] (μCi/g)	Secondary Coolant (μCi/g)[2]			
		Water[4]		Steam[5]	
		Phosphate	Volatile	Phosphate	Volatile
Other nuclides (cont'd)					
La-140	1.5(-4)	8(-8)	7(-9)	8(-11)	7(-12)
Ce-141	7.0(-5)	3(-8)	4(-9)	3(-11)	4(-12)
Ce-143	4.0(-5)	9(-9)	1(-9)	9(-12)	1(-12)
Ce-144	3.3(-5)	2(-8)	2(-9)	2(-11)	2(-12)
Pr-143	5.0(-5)	2(-8)	2(-9)	2(-11)	2(-12)
Pr-144	3.3(-5)	2(-8)	2(-9)	3(-11)	2(-12)
Np-239	1.2(-3)	3(-7)	6(-8)	3(-10)	6(-11)

1. Taken from NRC, 1976a.
2. Based on a primary-to-secondary leak of 100 lb/day.
3. The concentrations given are for reactor coolant entering the letdown line.
4. The concentrations given are for water in a steam generator.
5. The concentrations given are for steam leaving a steam generator.
6. 2.1(-2) = 2.1×10^{-2}.

Table 3.2-3 PWR Primary and Secondary Coolant Activities (Straight Tube Steam Generators)[1]

Isotope	Reactor Coolant[2] (μCi/g)	Secondary Coolant[3] (μCi/g)
Noble gases		
Kr-83m	2.1(-2)[4]	5.8(-9)
Kr-85m	1.1(-1)	3.1(-8)
Kr-85	1.5(-1)	4.2(-8)
Kr-87	6.0(-2)	1.6(-8)
Kr-88	2.0(-1)	5.5(-8)
Kr-89	5.0(-3)	1.4(-9)
Xe-131m	1.1(-1)	3.1(-8)
Xe-133m	2.2(-1)	6.2(-8)
Xe-133	1.8(+1)	5.0(-6)
Xe-135m	1.3(-2)	3.6(-9)
Xe-135	3.5(-1)	9.7(-8)
Xe-137	9.0(-3)	2.5(-9)
Xe-138	4.4(-2)	1.2(-8)
Halogens		
Br-83	4.8(-3)	2.3(-9)
Br-84	2.6(-3)	1.2(-9)
Br-85	3.0(-4)	1.4(-10)
I-130	2.1(-3)	1.3(-9)
I-131	2.7(-1)	1.3(-7)
I-132	1.0(-1)	4.7(-8)
I-133	3.8(-1)	1.8(-7)
I-134	4.7(-2)	2.2(-8)
I-135	1.9(-9)	9.0(-8)
Cs, Rb		
Rb-86	8.5(-5)	7.0(-11)
Rb-88	2.0(-1)	2.0(-7)
Cs-134	2.5(-2)	2.0(-8)
Cs-136	1.3(-2)	1.0(-8)
Cs-137	1.8(-2)	1.5(-8)
Water activation products		
N-16	4(+1)	1(-6)

Table 3.2-3 PWR Primary and Secondary Coolant Activities (Straight Tube Steam Generators)[1] (Cont'd)

Isotope	Reactor Coolant[2] (μCi/g)	Secondary Coolant[3] (μCi/g)
Tritium		
H-3	1(+1)	1(-3)
Other nuclides		
Cr-51	1.9(-3)	9(-10)
Mn-54	3.1(-4)	2(-10)
Fe-55	1.6(-3)	8(-10)
Fe-59	1.0(-3)	5(-10)
Co-58	1.6(-2)	8(-9)
Co-60	2.0(-3)	9(-10)
Sr-89	3.5(-4)	2(-10)
Sr-90	1.0(-5)	5(-12)
Sr-91	6.5(-4)	3(-10)
Y-90	1.2(-6)	6(-13)
Y-91m	3.6(-4)	2(-10)
Y-91	6.4(-5)	3(-11)
Y-93	3.4(-5)	2(-11)
Zr-95	6.0(-5)	3(-11)
Nb-95	5.0(-5)	2(-11)
Mo-99	8.4(-2)	4(-7)
Tc-99m	4.8(-2)	2(-7)
Ru-103	4.5(-5)	2(-11)
Ru-106	1.0(-5)	5(-12)
Rh-103m	4.5(-5)	2(-11)
Rh-106	1.0(-5)	5(-12)
Te-125m	2.9(-5)	1(-11)
Te-127m	2.8(-4)	1(-10)
Te-127	8.5(-4)	4(-10)
Te-129m	1.4(-3)	7(-10)
Te-129	1.6(-3)	8(-10)
Te-131m	2.5(-3)	5(-10)
Te-131	1.1(-3)	5(-10)
Te-132	2.7(-2)	1(-8)
Ba-137m	1.6(-2)	8(-9)
Ba-140	2.2(-4)	1(-10)
La-140	1.5(-4)	7(-11)
Ce-141	7.0(-5)	3(-11)

Table 3.2-3 PWR Primary and Secondary Coolant Activities (Straight Tube Steam Generators)[1] (Cont'd)

Isotope	Reactor Coolant[2] (μCi/g)	Secondary Coolant[3] (μCi/g)
Other nuclides (cont'd)		
Ce-143	4.0(-5)	2(-11)
Ce-144	3.3(-5)	2(-11)
Pr-143	5.0(-5)	2(-11)
Pr-144	3.3(-5)	2(-11)
Np-239	1.2(-3)	6(-10)

1. Taken from NRC, 1976a.
2. The concentrations given are reactor coolant entering the letdown line.
3. Based on primary-to-secondary leakage of 100 lb/day. The concentrations given are for steam leaving a steam generator.
4. $2.1(-2) = 2.1 \times 10^{-2}$.

Table 3.2-4 Radionuclide Concentrations in Untreated Detergent Waste[1]

Nuclide	Average Concentration ($\mu Ci/cm^3$)
Mn-54	1.6×10^{-6}
Co-58	6.4×10^{-6}
Co-60	1.4×10^{-5}
Zr-95	2.3×10^{-6}
Nb-95	3.2×10^{-6}
Ru-103	2.3×10^{-7}
Ru-106	3.9×10^{-6}
Ag-110m	7.1×10^{-7}
I-131	9.6×10^{-7}
Cs-134	2.1×10^{-5}
Cs-137	3.9×10^{-5}
Ce-144	8.0×10^{-6}
Total	1.0×10^{-4}

1. Taken from NRC, 1976a.

plants under construction. Table 3.2-5 shows the estimated radionuclide concentrations in BWR primary coolant and main steam (NRC, 1976b).

3.2.1.2.4 Liquid Waste Processing System

Historically, BWR liquid waste processing systems are divided into four distinct subsystems:

a. Clean radwaste (CRW)

b. Dirty radwaste (DRW)

c. Chemical waste

d. Laundry waste

The following sections describe the inputs to these subsystems. In addition, these sections give the basis for deciding to which subsystem a given waste stream will be routed for processing. Table 3.2-6 shows the expected daily input to each of these subsystems based on NUREG-0016 (NRC, 1976b).

Clean Radwaste

The clean radwaste system (CRW) collects wastes from equipment leakages and drainages of equipment during maintenance. The clean radwaste system wastes may vary in radionuclide concentration from 1% of primary coolant concentration to 100% of primary coolant concentration. During system design wastes are selected as CRW waste when it is expected that the conductivity will be less than 10 mho/cm (NRC, 1976b). Processing usually will consist of filtration and demineralization only. Processed waste is collected in a sample tank and either discharged to the environment or recycled to the plant for reuse or for further treatment.

Dirty Radwaste

The dirty radwaste system (DRW) collects equipment drainages from systems that are expected to have high conductivity, between 10 and 200 mho/cm, and from floor drains (NRC, 1976b). Most plants are equipped with filters and demineralizers for DRW waste processing. Approximately half of the operating BWRs also have an evaporator available for processing DRW waste if necessary. Newer plants (i.e., those coming on line in the next 10 years), are eliminating the demineralizer and only using a filter upstream of the evaporator.

Chemical Waste System

Early BWR designs did not contain separate equipment for the processing of chemical wastes. These plants treat chemical wastes as high-conductivity wastes and collect them along with other high-conductivity wastes in the floor drain system. Plants under

Table 3.2-5 Radionuclide Concentrations in Boiling Water Reactor Coolant and Main Stream[1,2]

Isotope	Reactor Water (μCi/g)	Reactor Steam (μCi/g)
Noble Gases		
Kr-83m		1.1(-3)[3]
Kr-85m		1.9(-3)
Kr-85		6.0(-6)
Kr-87		6.6(-3)
Kr-88		6.6(-3)
Kr-89		4.1(-2)
Kr-90		9.0(-2)
Kr-91		1.1(-1)
Kr-92		1.1(-1)
Kr-93		2.9(-2)
Kr-94		7.2(-3)
Kr-95		6.6(-4)
Kr-97		4.4(-6)
Xe-131m		4.7(-6)
Xe-133m		9.0(-5)
Xe-133		2.6(-3)
Xe-135m		8.4(-3)
Xe-135		7.2(-3)
Xe-137		4.7(-2)
Xe-138		2.8(-2)
Xe-139		9.0(-2)
Xe-140		9.6(-2)
Xe-141		7.8(-2)
Xe-142		2.3(-2)
Xe-143		3.8(-3)
Xe-144		1.8(-4)
Halogens		
Br-83	3(-3)	6(-5)
Br-84	5(-3)	1(-4)
Br-85	3(-3)	6(-5)
I-131	5(-3)	1(-4)
I-132	3(-2)	6(-4)
I-133	2(-2)	4(-4)
I-134	5(-2)	1(-3)
I-135	2(-2)	4(-4)

Table 3.2-5 Radionuclide Concentrations in Boiling Water Reactor Coolant and Main Stream[1,2] (Cont'd)

Isotope	Reactor Water (μCi/g)	Reactor Steam (μCi/g)
Cesium and rubidium		
Rb-89	5(-3)	5(-6)
Cs-134	3(-5)	3(-8)
Cs-136	2(-5)	2(-8)
Cs-137	7(-5)	7(-8)
Cs-138	1(-2)	1(-5)
Water activation products		
N-13	5(-2)	7(-3)
N-16	6(+1)	5(+1)
N-17	9(-3)	2(-2)
O-19	7(-1)	2(-1)
F-18	4(-3)	4(-3)
Tritium[4]		
H-3	1(-2)	1(-2)
Other nuclides		
Na-24	9(-3)	9(-6)
P-32	2(-4)	2(-7)
Cr-51	5(-3)	5(-6)
Mn-54	6(-5)	6(-8)
Mn-56	5(-2)	5(-5)
Fe-55	1(-3)	1(-6)
Fe-59	3(-5)	3(-8)
Co-58	2(-4)	2(-7)
Co-60	4(-4)	4(-7)
Ni-63	1(-6)	1(-9)
Ni-65	3(-4)	3(-7)
Cu-64	3(-2)	3(-5)
Zn-65	2(-4)	2(-7)
Zn-69	2(-3)	2(-6)
Sr-89	1(-4)	1(-7)

Table 3.2-5 Radionuclide Concentrations in Boiling Water Reactor Coolant and Main Stream[1,2] (Cont'd)

Isotope	Reactor Water (μCi/g)	Reactor Steam (μCi/g)
Other nuclides (cont'd)		
Sr-90	6(-6)	6(-9)
Sr-91	4(-3)	4(-6)
Sr-92	1(-2)	1(-5)
Y-91	4(-5)	4(-8)
Y-92	6(-3)	6(-6)
Y-93	4(-3)	4(-6)
Zr-95	7(-6)	7(-9)
Zr-97	5(-6)	5(-9)
Nb-95	7(-6)	7(-9)
Nb-98	4(-3)	4(-6)
Mo-99	2(-3)	2(-6)
Tc-99m	2(-2)	2(-5)
Tc-101	9(-2)	9(-5)
Tc-104	8(-2)	8(-5)
Ru-103	2(-5)	2(-8)
Ru-105	2(-3)	2(-6)
Ru-106	3(-6)	3(-9)
Ag-110m	1(-6)	1(-9)
Te-129m	4(-5)	4(-8)
Te-131m	1(-4)	1(-7)
Te-132	1(-5)	1(-8)
Ba-139	1(-2)	1(-5)
Ba-140	4(-4)	4(-7)
Ba-141	1(-2)	1(-5)
Ba-142	6(-3)	6(-6)
La-142	5(-3)	5(-6)
Ce-141	3(-5)	3(-8)
Ce-143	3(-5)	3(-8)
Ce-144	3(-6)	3(-9)
Pr-143	4(-5)	4(-8)

Table 3.2-5 Radionuclide Concentrations in Boiling Water Reactor Coolant and Main Stream[1,2] (Cont'd)

Isotope	Reactor Water (μCi/g)	Reactor Steam (μCi/g)
Other nuclides (cont'd)		
Nd-147	3(-6)	3(-9)
W-187	3(-4)	3(-7)
Np-239	7(-3)	7(-6)

1. Taken from NRC, 1976b.
2. The reactor water concentration is specified at the nozzle where reactor water leaves the reactor vessel. Similarly, the reactor steam concentration is specified at time 0.
3. 1.1(-3) = 1.1×10^{-3}.
4. Measured values increased to account for liquid recycle.

Table 3.2-6 BWR Liquid Radwaste Inputs[1]

Source	Regenerative Deep Bed Condensate Demineralizers (gal/d)		Filter/ Demineralizer (Powdex) Condensate Demineralizer (gal/d)	Fraction of Primary Coolant Activity[2]
	Plant with Ultrasonic Resin Cleaner	Plant Without Ultrasonic Resin Cleaner		
Equipment drains				
Drywell	3,400	3,400	3,400	1
Containment, auxiliary building, and fuel pool	3,720	3,720	3,720	0.01
Radwaste building	1,060	1,060	1,060	0.01
Turbine building	2,960	2,960	2,960	0.01
Ultrasonic resin cleaner[3]	15,000	-	-	0.05
Resin rinse	2,500	5,500	-	0.002
Subtotal	28,640	16,140	11,140	-
Floor drains				
Drywell	700	700	700	1
Containment, auxiliary building, and fuel pool	2,000	2,000	2,000	0.01
Radwaste building	1,000	1,000	1,000	0.01
Turbine building	2,000	2,000	2,000	0.01
Subtotal	5,700	5,700	5,700	-
Other				
Cleanup phase separator decant	640	640	640	0.002
Laundry drains	450	450	450	(4)
Lab drains	500	500	500	0.02
Regenerants[3]	1,700	3,400	-	(5)
Condensate backwash[6]	-	-	8,100	2×10^{-6}
Chemical lab waste	100	100	100	0.02
Subtotal	3,390	5,090	9,790	
Total	37,730	26,930	26,630	

1. Taken from NRC, 1976b.
2. See Table 3.2-5 for reactor water and reactor steam activities.
3. Deep-bed condensate demineralizers.
4. See Table 3.2-4.
5. Calculated by BWR-GALE Code.
6. Filter/demineralizer(Powdex) condensate demineralizer.

design in the early to mid-seventies added evaporators and separate collection tanks specifically for chemical waste. Because most future plants are being designed with regenerative deep bed condensate demineralizers, separate chemical waste treatment equipment will be essential.

Laundry Wastes

Laundry wastes are collected, processed, and sampled before disposal or recycling. Processing is normally achieved through simple filtration since, as with PWRs, the laundry wastes are almost always below the discharge limits for radioactivity prior to filtration. A few plants are using reverse osmosis to concentrate the sludge, and even fewer plants are using evaporation.

3.2.2 Fuel Fabrication Plants

As mentioned in the introduction to this chapter the only liquid waste treatment system in a fuel fabrication facility analogous to the liquid waste treatment system of an LWR is the filtration of floor cleaning solutions, floor drains, and wash basin drains. These wastes are filtered to recover any uranium that may be contaminating these wastes.

3.3 Current Waste Management Practices at LWRs

3.3.1 Radioactive Waste Concentration Techniques

There are three widely accepted methods or techniques used to concentrate radioactive contaminants in LWR liquid wastes: filtration, demineralization, and evaporation. This section of the report discusses the equipment used in each of these processes.

3.3.1.1 Filtration

Filtration is the process of passing a liquid or gaseous stream through a porous medium or mass to filter out suspended matter. The types of filters available are numerous, including those that are backflushable, not backflushable, disposable, reusable, and those requiring precoat and those not requiring precoat.

As seen from the survey of LWRs conducted for this study the filters used in PWRs are predominantly cartridge type disposable filters. IN BWRs reusable precoat filters are predominant. Both types of filters have been used for many years prior to their application to the nuclear industry. The additional problem of shielding operating personnel from the filters, especially during cartridge replacement for cartridge filters, is a problem unique to the nuclear industry.

The filters that are reportedly used in the plants surveyed for this study are

- Disposable cartridge filters
- Vertical tube precoat filters, and
- Flat-bed filters.

Other filters that are available or which have been specifically marketed for use in LWRs are as follows:

- Centrifugal-discharge filters
- Stacked etched-disc filters

3.3.1.1.1 Disposable Cartridge Filters

Figure 3.3-1 shows a cross-sectional view of a typical disposable cartridge filter. The body, or filter housing, consists of a vertical cylinder with a rounded bottom and removable top. Mounted inside the body are the removable disposable cartridges. Each cartridge is approximately 3 inches in diameter and from 1 to 3 feet long. Anywhere from one to several dozen cartridges may be fitted into the filter body. Water enters the body of the filter from the side and is distributed evenly throughout the filter. The water passes through the outer wall of the cartridge into its interior and down the cartridge to the outlet. As seen in Figure 3.3-1 the cartridges actually sit in a lifting basket for ease of installation and removal.

The cartridges used in these filters and specific applications have been noted in Tables 4.2-11, 4.2-35, and 4.2-36. Specific details on representative cartridges are given in Section 2.2.3. Table 3.3-1 lists various factors to be considered when selecting the proper filter for a given application.

3.3.1.1.2 Vertical Tube Precoat Filter

The vertical tube precoat filter is the second most widely used type of filter in LWRs. Also known as a candle filter, it is used in almost all BWR radwaste systems and in BWR and PWR precoat filter condensate polishing systems. The tubes are permanent and backflushable. Figure 3.3-2 is a cross-sectional view of a candle filter. The types are porous, covered with either a wire screen or a wedge-wire winding. The tubes are suspended from the tube sheet, with the number of tubes dependent on the expected or rated flow rate. The bottom ends of the tubes are plugged, with a hole at the top extending through the otherwise solid tube sheet. Before use, the tubes are precoated with one of the precoat materials described in Section 2.2.2. This material builds on the tube surfaces creating a porous cake that performs the actual filtration.

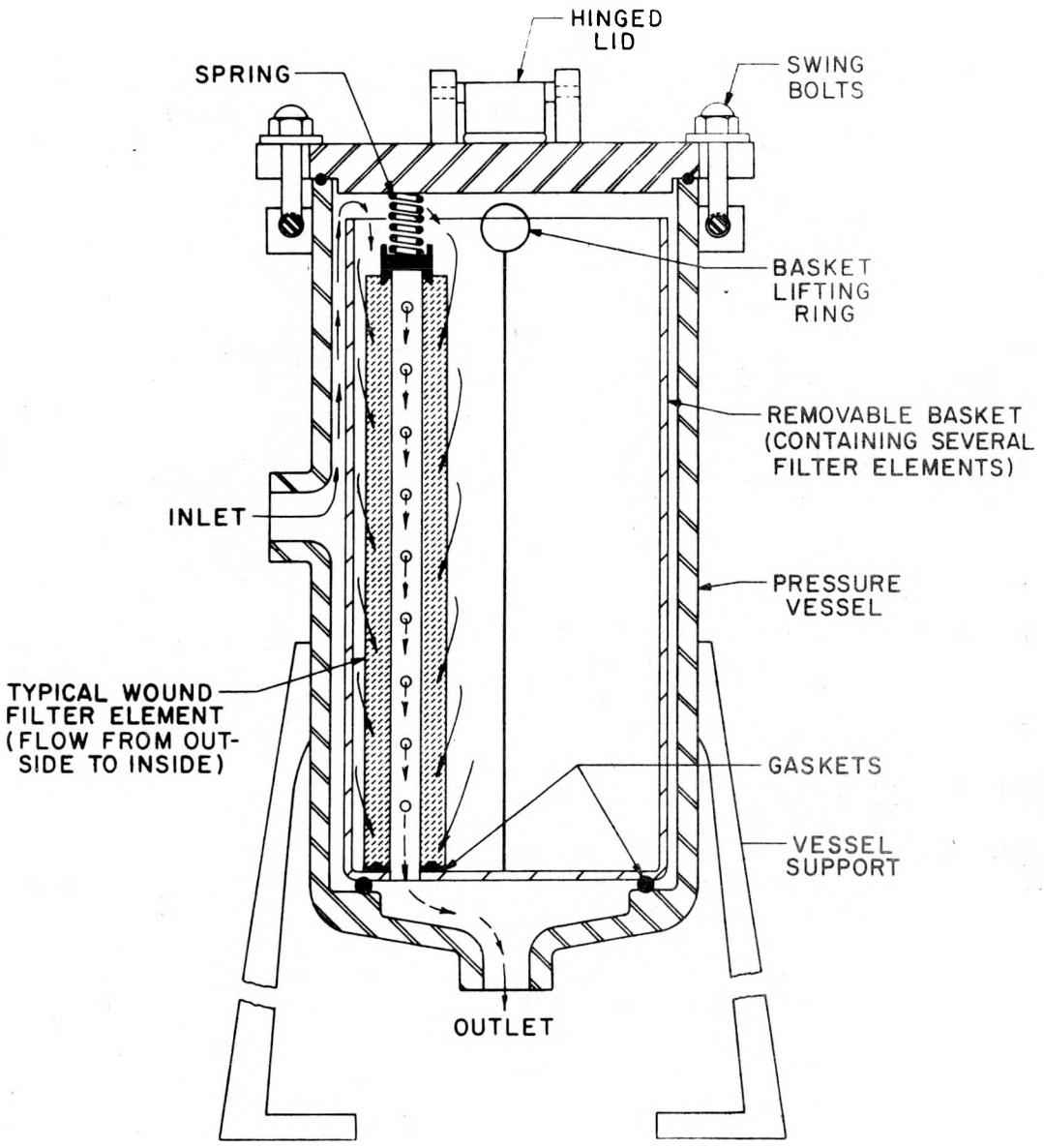


Figure 3.3-1 Typical Disposable Cartridge Filter[1]

1. Taken from Kibbey, 1978.

Table 3.3-1 Potential Advantages and Disadvantages of Filters for Liquids in LWR Nuclear Power Plants[1]

Type of Filter	Advantages	Disadvantages
Disposable		
Wound cartridge	Compact Low solid waste volume; No backflush gas or liquid to treat; Good solids removal.	Remote and/or automatic changeout difficult because of nonuniformity and poor arrangement; changeout frequently done on radiation level rather than pressure drop; media migration may occur.
Pleated paper cartridge	Compact Low solid waste volume; No backflush gas or liquid to treat; Good solids removal.	Remote and/or automatic changeout difficult because of nonuniformity and poor arrangement; changeout frequently done on radiation level rather than pressure drop; media migration may occur.
Pleated wire screen	Can operate at elevated temperatures; Good solids removal; Little or no media migration.	Fair mechanical strength when adequately supported; Plugging may cause uneven flow and nonuniform cake buildup.
Reusable without precoat		
Stacked etched-disc	Short backflush time with thorough cleaning; expected to last for plant life; Amenable to automatic and/or remote operation; Low solid waste volume; Compact; high mechanical strength.	Low crud-holding capability; Corrosion characteristics unknown; Backwash waste to treat; Low oil-holding capacity.

Table 3.3-1 Potential Advantages and Disadvantages of Filters for Liquids in LWR Nuclear Power Plants[1] (Cont'd)

Type of Filter	Advantages	Disadvantages
Reusable with precoat		
Backflushable tubular bundle	Amenable to automatic and/or remote operation; Powdered resin and/or diatomaceous earth precoat can be used; Relatively compact.	Precoat loss upon loss of flow or fluctuation in pressure; Excessive or uneven cake can cause strain and possible collapse of supporting screen; Incomplete backflushing causes uneven precoat.
Dry cake discharge		
Centrifugal discharge	High crud-holding capacity; Can handle automatically and remotely all plant wastes with same filter; Low maintenance requirements; No precoat loss caused by loss of flow, pressure or power.	Relatively high headroom; Cake overloading can cause distortion; Generates large sludge volume; Some cake difficulty with Solka-Floc or resins alone.
Flat bed	High crud-holding capacity; Can handle automatically and remotely all plant wastes with same filter; No precoat loss caused by loss of flow, pressure or power.	Relatively large floor space and high headroom; Cake overloading can cause belt wear; Generates large sludge volume; Some cake difficulty with resin alone; May require fairly high belt maintenance.

1. Taken from Kibbey, 1978.

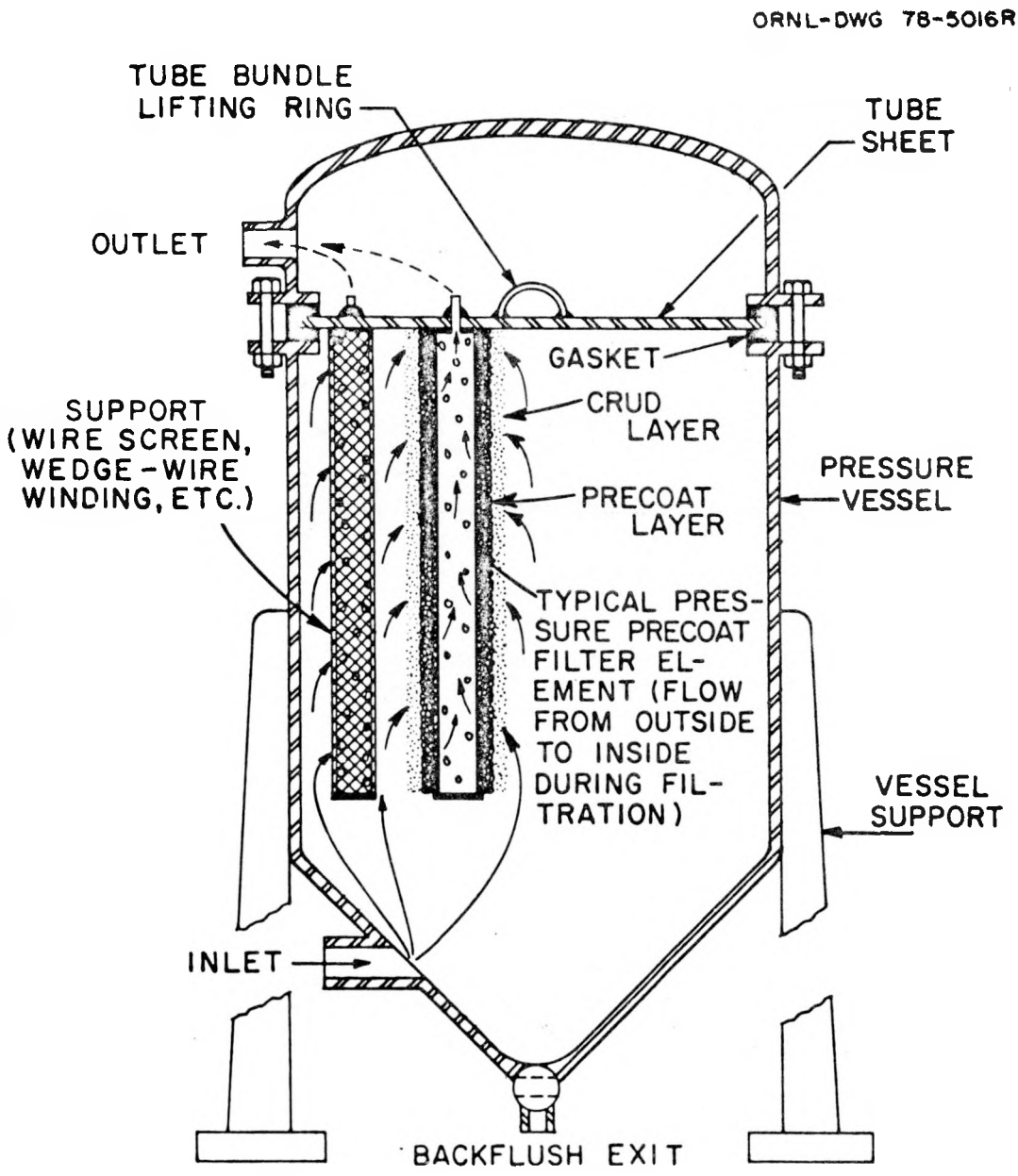


Figure 3.3-2 Typical Tubular-Support Pressure-Precoat Filter[1]

1. Taken from Kibbey, 1978.

As shown in Figure 3.3-2 the water enters through the inlet at the bottom of the filter and travels through the filter cake, then through the precoat medium, before moving up the tube. The filtered water, or filtrate, exits through the top of the vessel. When the filter is not in use a small holding pump maintains a small constant flow through the unit to maintain a minimal pressure differential across the candle. If this differential pressure were lost, the filter cake would also be lost.

As the filtration process continues, the filter cake builds, resulting in increased differential pressure across the filter. In most systems filter life is determined by this differential pressure rather than effluent quality. If the solids concentration of the feed is high (more than several hundred ppm), small amounts of additional filter aid may be added to the feed stream. This additional filter aid prolongs the filter life by dispersing throughout the thicker filter cake the solids that are removed. To backflush the filter the inlet valve (not shown) is closed, and the backflush exit valve is opened. After an initial air or nitrogen bump loosens the filter cake, water is pumped back into the unit through the outlet. This forces the water to flow from the inside of the tubes to the outside, thus washing off the built-up filter cake and precoat and washing it out of the filter. The backwash water is collected in a settling tank, or phase separator, where the solids are allowed to settle out and the excess water drained off.

3.3.1.1.3 Flat Bed Filters

The flat bed or traveling belt filter uses a fine wire screen or woven fabric, usually precoated with diatomaceous earth, as the filter support medium. The filter screen may be a continuous belt or may wind back and forth to discharge the filter cake. The screen separates the upper housing, which contains the filter inlet, from the lower housing which contains the filter outlet. The unit is sealed on all sides by a gasket.

After precoating, the filter cycle continues until a preset pressure differential is reached. Then air is passed through the unit until the cake is dry. After this, the upper housing is raised, and the belt is advanced forward, thus causing the dried cake to discharge directly into the shipping container. Figure 3.3-3 shows a cross-sectional view of the filter during the filtration stage and the cake discharge stage.

3.3.1.1.4 Centrifugal Discharge Filters

In centrifugal discharge filters the precoat material collects on a wire mesh screen on horizontally mounted discs. These discs are attached to a vertically mounted hollow shaft. Figure 3.3-4 is a cross-sectional view of a centrifugal discharge filter.

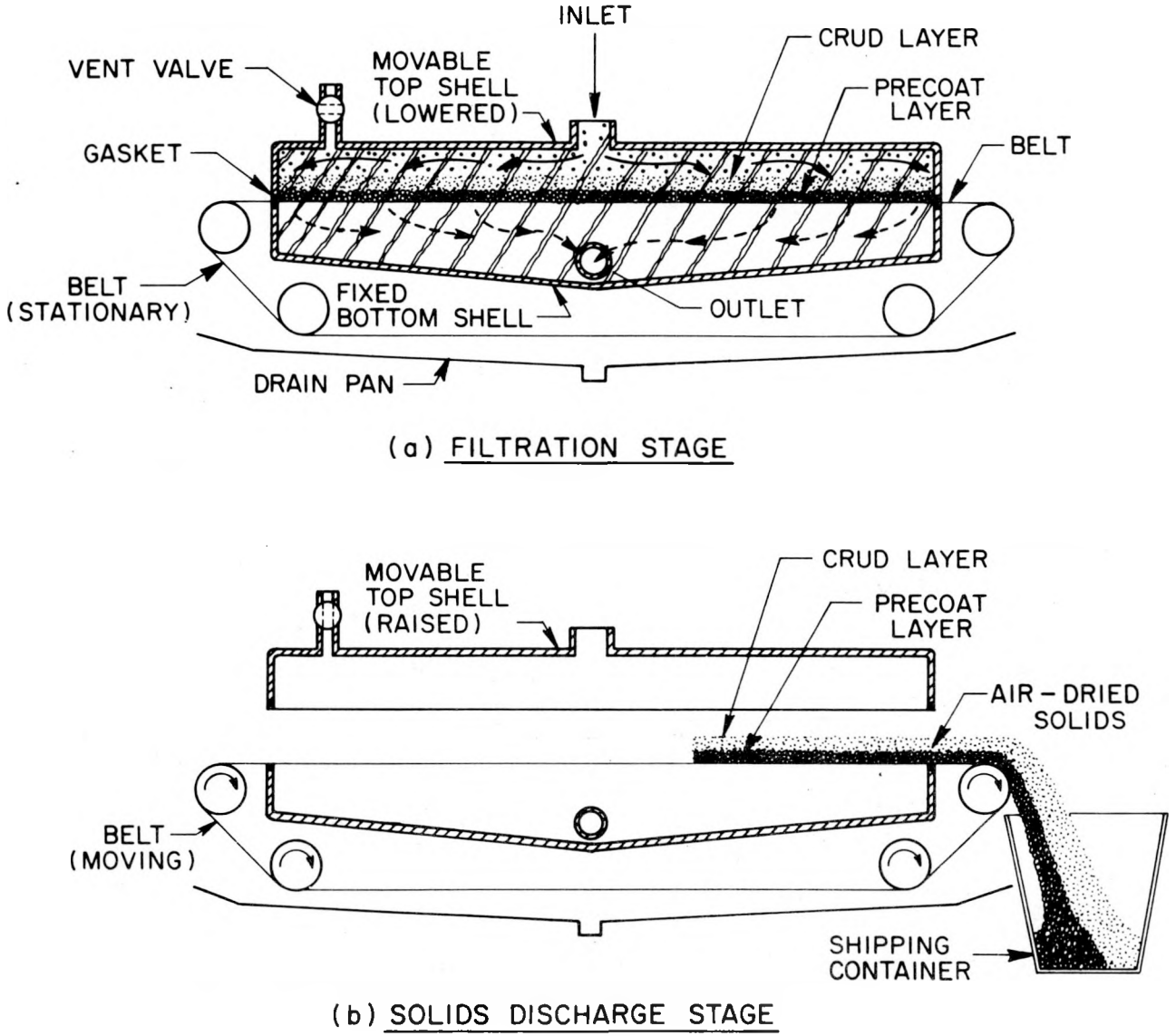


Figure 3.3-3 Typical Flat-Bed Filter[1]

1. Taken from Kibbey, 1978.

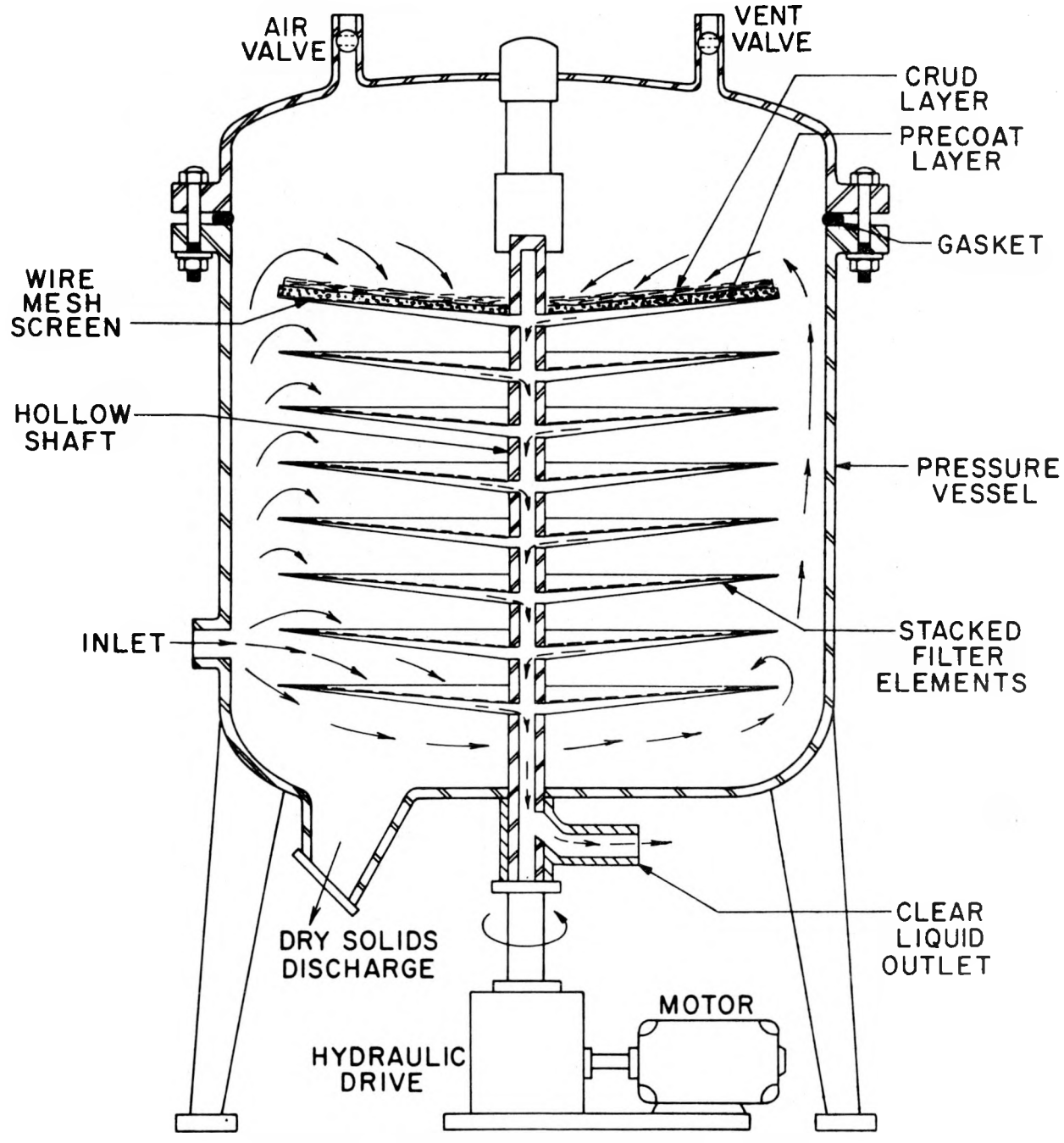


Figure 3.3-4 Typical Centrifugal-Discharge Filter[1]

1. Taken from Kibbey, 1978.

Centrifugal discharge filters are precoated in much the same way as tubular precoat filters. Precoat is added to a loading of approximately 0.2 lb/ft^2. During filtration body feed is added to the inlet stream continuously to prevent plugging. When the differential pressure across the vessel reaches a preset limit, the inlet feed is terminated, and the unit is drained. The filter cake is then dried with ambient-temperature compressed air for 30 to 40 minutes. Then the entire internals are spun at 200 to 300 rpm, throwing the filter cake against the interior walls. The cake drops to the bottom and is mechanically pushed to the discharge hopper.

Diatomaceous earth is the best precoat material to use in centrifugal discharge filters. Solka Floc and other cellulose precoats tend to harden during the drying cycle and therefore are very difficult to force loose during the discharge cycle. Powdex and other powered resins do not always form even precoats and can be unstable. Dewatering of cellulose sludge and spent demineralizer resins can be accomplished if they are mixed in proper proportions.

Centrifugal discharge filters are capable of removing 98 to 99% of all particles ranging in size from 1 μm to 5 μm.

3.3.1.1.5 Stacked Etched-Disk Filters

Etched-disk filters consist of numerous chemically etched stainless-steel disks. The disks are compressed together and mounted vertically in the filter housing. Disks are etched on one side only and stacked with the etched side of one in contact with the unetched side of the next. Disks are normally etched to a depth of 5 μm. Figure 3.3-5 is a cross-sectional view of an etched-disk filter.

Water enters the filter from the bottom and flows from the outside of the stack through the etched passages and out the top of the filter. As the particles and crud build up on the outside of the stack, the pressure drop increases. At a preset pressure drop, flow is terminated and the unit is backwashed. The backwash cycle begins with a bump of high-pressure air or nitrogen and continues with a water wash. These units are usually used without a precoat although diatomaceous earth may be used for the removal of small quantities of oil.

3.3.1.2 Demineralization

Demineralization is the process of removing dissolved mineral and other ions from a solution by passing it through a demineralizer resin. This process is called ion exchange. Figure 3.3-6 shows a typical deep bed demineralizer vessel. Resins are manufactured in two basic types: anion resin and cation resin. Anion and cation resins are combined to form a mixed bed resin in one vessel. Section 2.2.1 discusses the various physical and chemical properties of several resins used in LWRs. Demineralizers are used

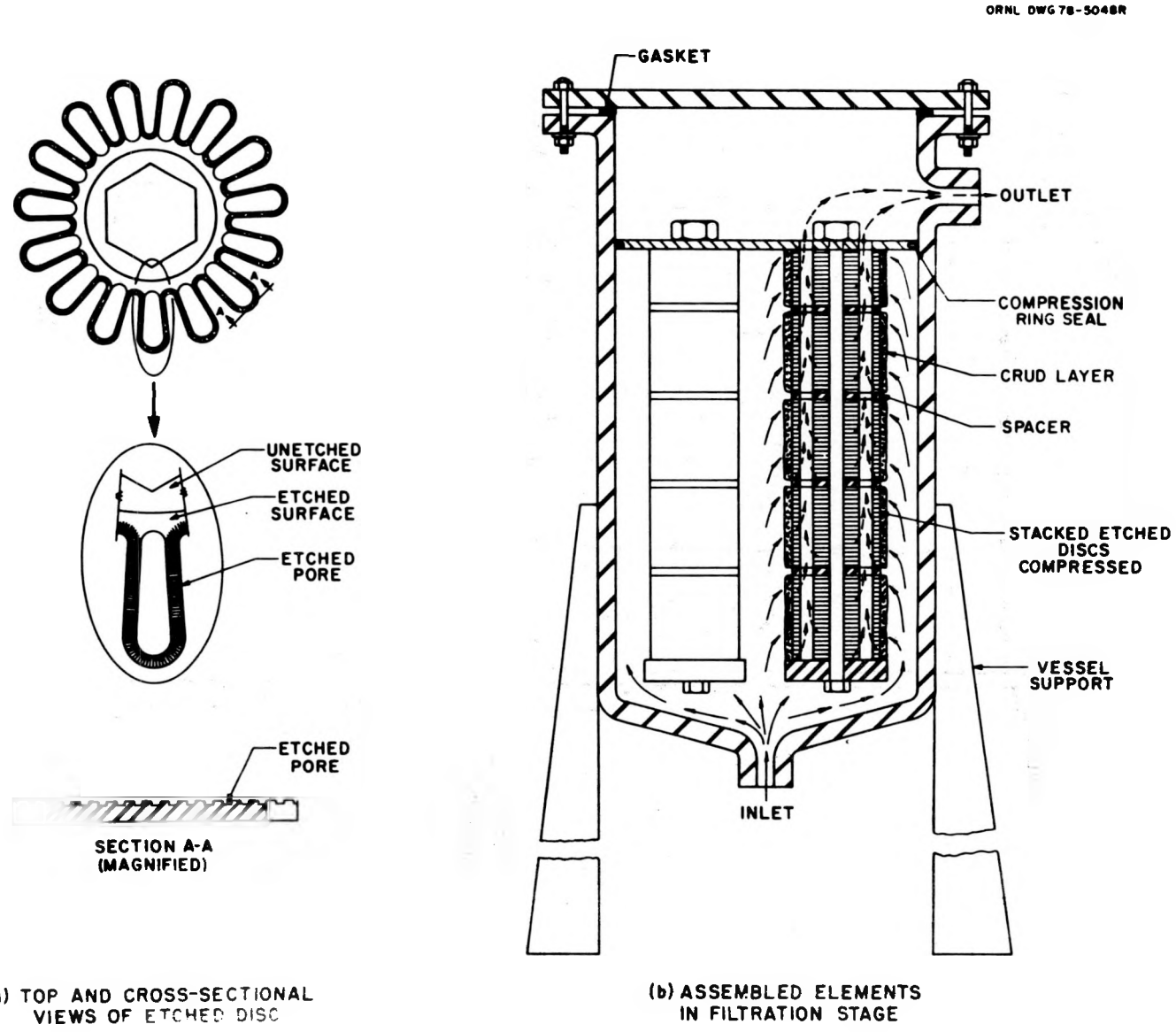


Figure 3.3-5 Typical Etched-Disc Filter[1]

1. Taken from Kibbey, 1978.

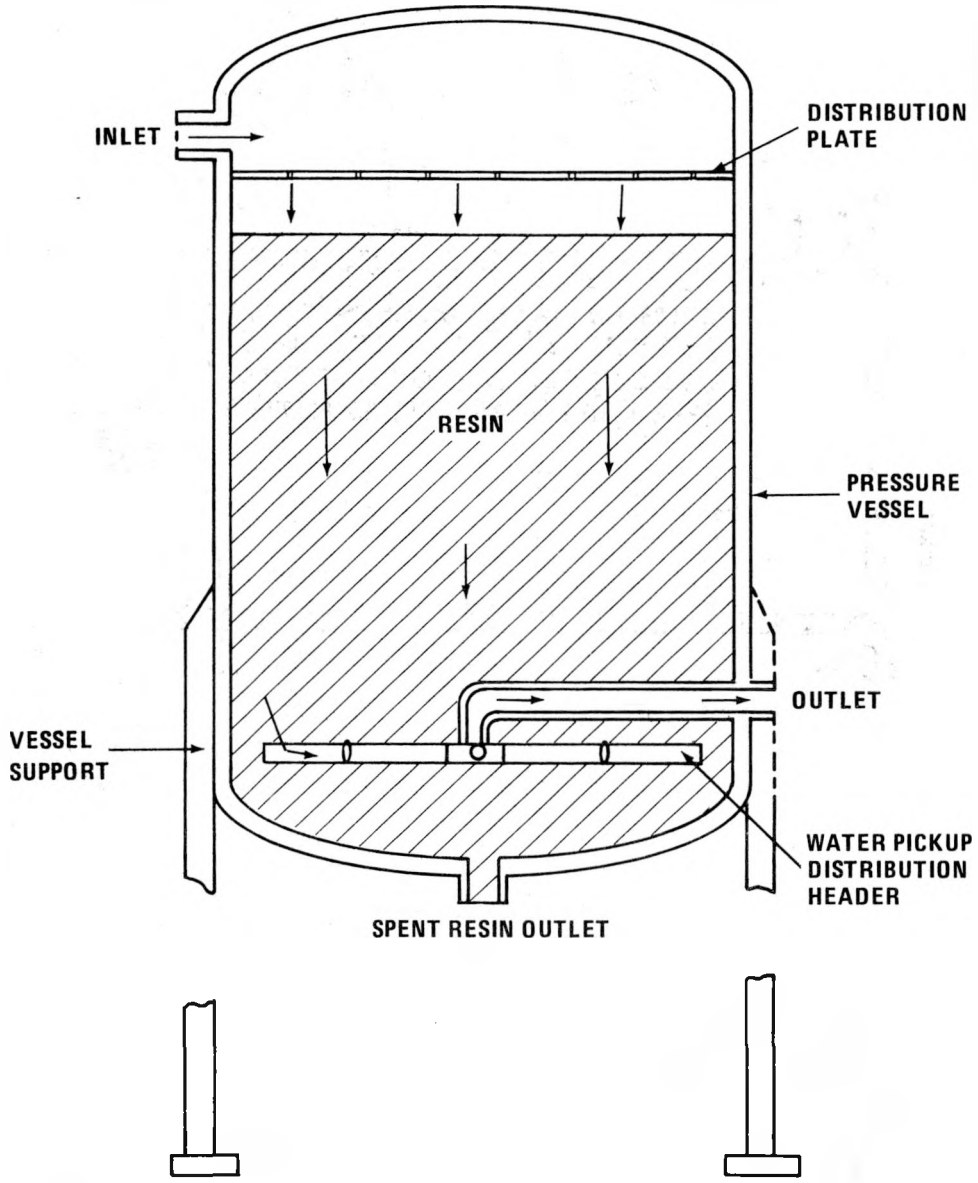


Figure 3.3-6 Typical Deep Bed Demineralizer

extensively throughout PWRs and BWRs to remove impurities including chlorides, borates, cesium, and almost all the other fission products. Experience has also shown demineralizers to be fairly efficient filters. In most applications demineralizer resins are replaced, cleaned, or regenerated based on the pressure drop across the vessel.

3.3.1.2.1 Nonregenerative Demineralizers

Nonregenerative demineralizers use resins that are backflushed to a spent resin tank for disposal when they become depleted. Depletion occurs either through exhaustion of all available ion-exchange sites or because of a high pressure drop across the bed. After backflushing, the vessel is filled with new resin and returned to service.

3.3.1.2.2 Regenerative Demineralizers

In regenerative demineralizers depleted resins are backwashed to a resin regeneration tank. At that point the mixed bed resins are mechanically separated, and the anion resin is moved to another tank. (If the demineralizer is not a mixed bed type this step is not necessary.) Regeneration is a chemical process using sulfuric acid (H_2SO_4) to regenerate the cation resins and sodium hydroxide (NaOH) to regenerate the anion resins. The resultant sodium and/or sulfate salts are sent to the radwaste system for concentration and disposal. In mixed bed systems these chemicals form sodium sulfate when mixed in the chemical waste tank of the radwaste system.

3.3.1.2.3 Ultrasonic Resin Cleaning

Ultrasonic resin cleaning is a process used primarily on BWR regenerative condensate resins. These resins are generally expected to contain little radioactive contamination, but they may pick up significant quantities of crud. Regeneration will remove the crud, but the process generates significant quantities of unnecessary chemical waste. To avoid this, several plants have installed ultrasonic resin cleaners to remove dirt and crud from the resins, thereby extending the time between regenerations. In an ultrasonic resin cleaner the resins enter a vertical column at the top and fall to the bottom because of gravity. Water entering the bottom is pumped against the flow of the resins at a velocity slightly slower than the falling resins. Ultrasonic vibrators attached to the column cause the crud and other particulates to break loose from the resin. The vibrations also break up any cracked resins. These items are small enough and light enough to be carried away in the water. Resin loss is approximately 1% by volume. The dirty water is sent to the radwaste system for treatment, and the resins are returned to the system for service. Figure 3.3-7 is a simplified flow diagram of an ultrasonic resin cleaner.

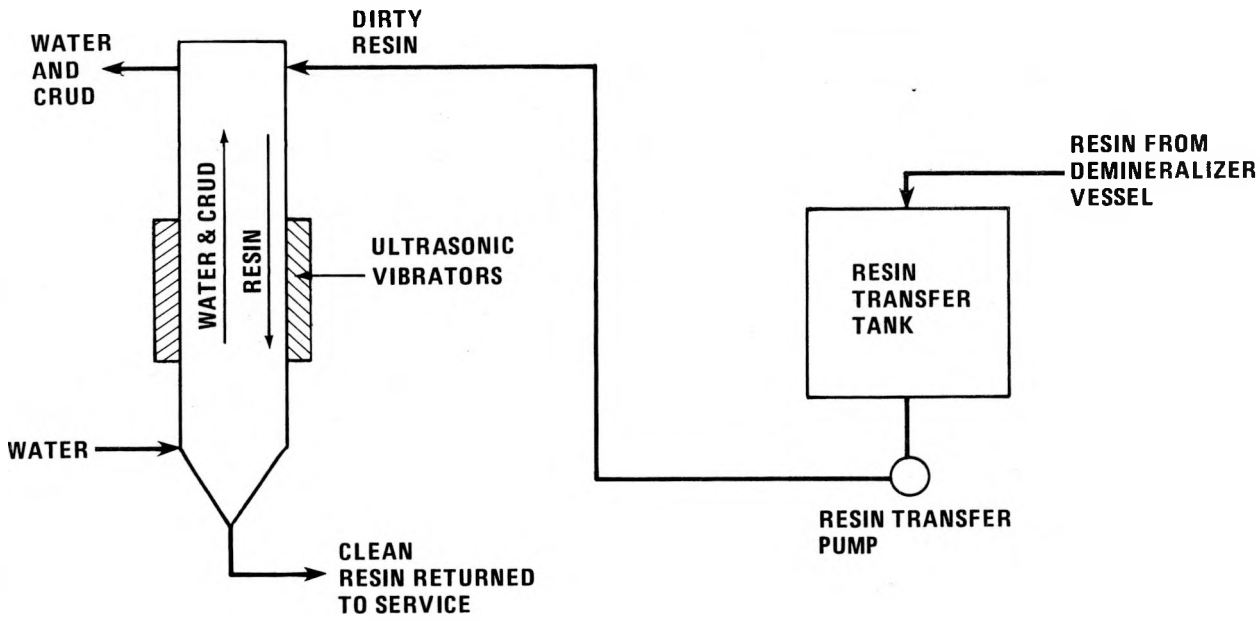


Figure 3.3-7 Deep Bed Resin Ultrasonic Resin Cleaner

3.3.1.3 Evaporation

In its simplest context an evaporator boils away the water from a liquid solution or slurry. The prime function of the evaporator is to produce a condensed vapor as free of the original contaminants as possible. Thus, the efficiency of an evaporation is rated in terms of its decontamination factor, as defined in Section 2.2.4.6 and listed in Table 2.2-12, rather than its volume reduction. Volume reduction is an important aspect of this study; therefore evaporation will be evaluated for various types of wastes. Although varying in size, shape, and type, all evaporators have the same basic parts: a heating section where the waste is heated; a vapor head or flash chamber where the vapor collects; a demister or similar device to remove any mist or small droplets from the steam; and a condenser to condense the steam. The tubes in the heating section may be either horizontal or vertical. The waste requiring concentration flows through the tubes, and in most evaporators is heated by steam. The tubes are located either directly below the vapor body and demister or separate from them. Circulation of the waste liquid is either induced by the density variations resulting from boiling, called natural circulation, or pumped, referred to as forced circulation. Figures 3.3-8 through 3.3-12 show the typical arrangement of equipment for a number of various evaporator configurations.

Evaporators are used in BWRs and PWRs to concentrate liquid wastes that are not conducive to other treatment methods such as filtration or demineralization. These waste streams are

a. Regeneration solutions from the regeneration of deep bed demineralizer resins

b. Boron or boric acid waste from the primary system leakage or backflushing of boron recovery demineralizers

c. Low-purity wastes (BWR floor drains, decontamination solutions, and detergent wastes)

d. Miscellaneous waste (chemicals, decontamination solutions, and detergent wastes).

In BWRs where miscellaneous waste and regeneration wastes are often collected in the same tank, volume reduction factors ranging from 15 to 20 have been observed (Godbee, 1978). The solids content of the concentrated waste, as reported in Section 4.2.1.2, averages 25 percent by weight. BWR low-purity wastes are concentrated between 40 and 80 times, yielding similar volume reduction factors.

PWR miscellaneous waste is similar to BWR low-purity waste except that the PWR miscellaneous waste contains boron, ammonia, and hydrazine. The volume reduction factors for PWR miscellaneous wastes have ranged from 10 to 100. PWR wastes are principally boron-contaminated wastes and their volume reduction factor ranges

ORNL DWG. 73-5932R2

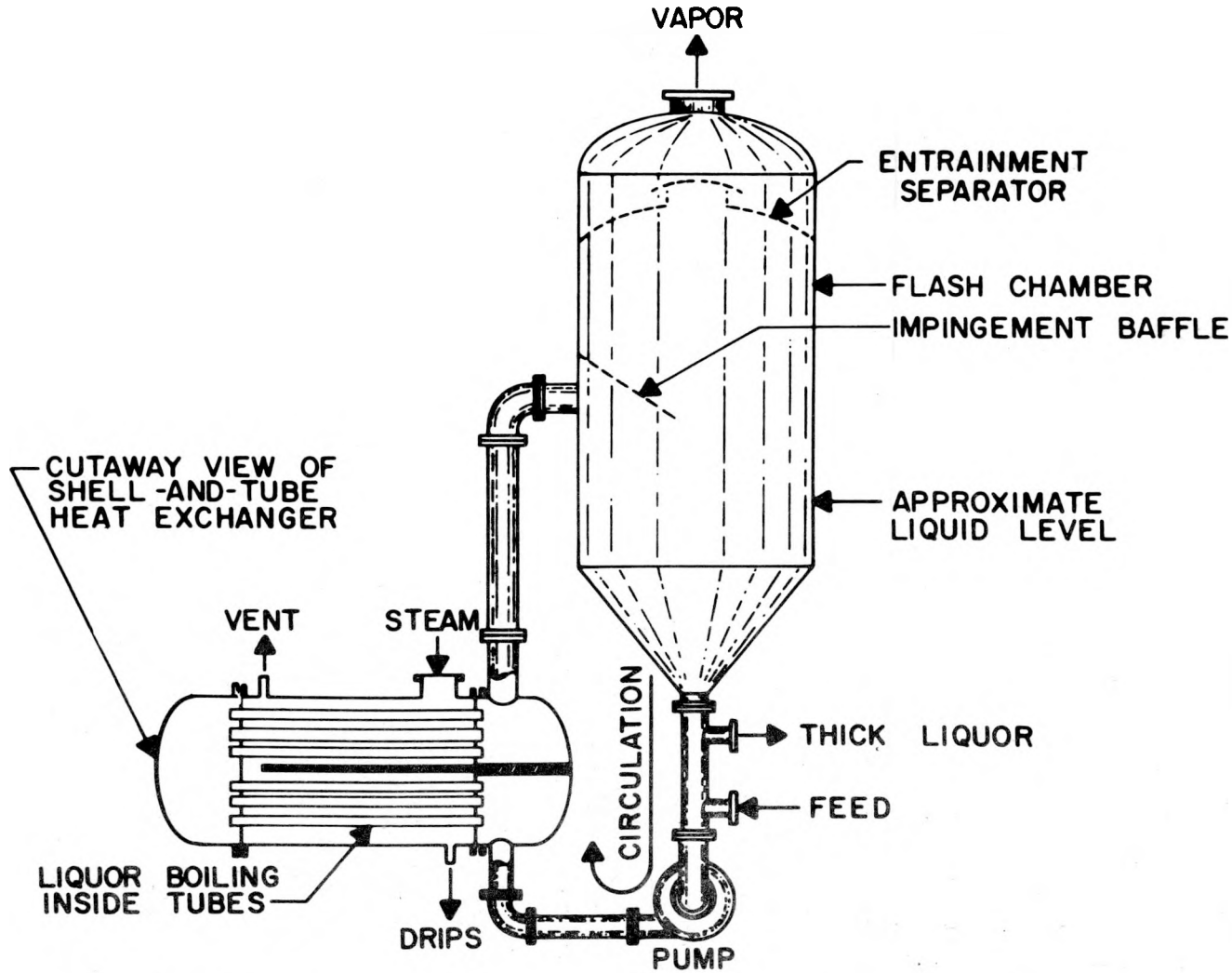


Figure 3.3-8 Forced-Circulation Evaporator With an External, Horizontal, Submerged-Tube, Two-Pass Heater[1]

1. Taken from Godbee, 1978.

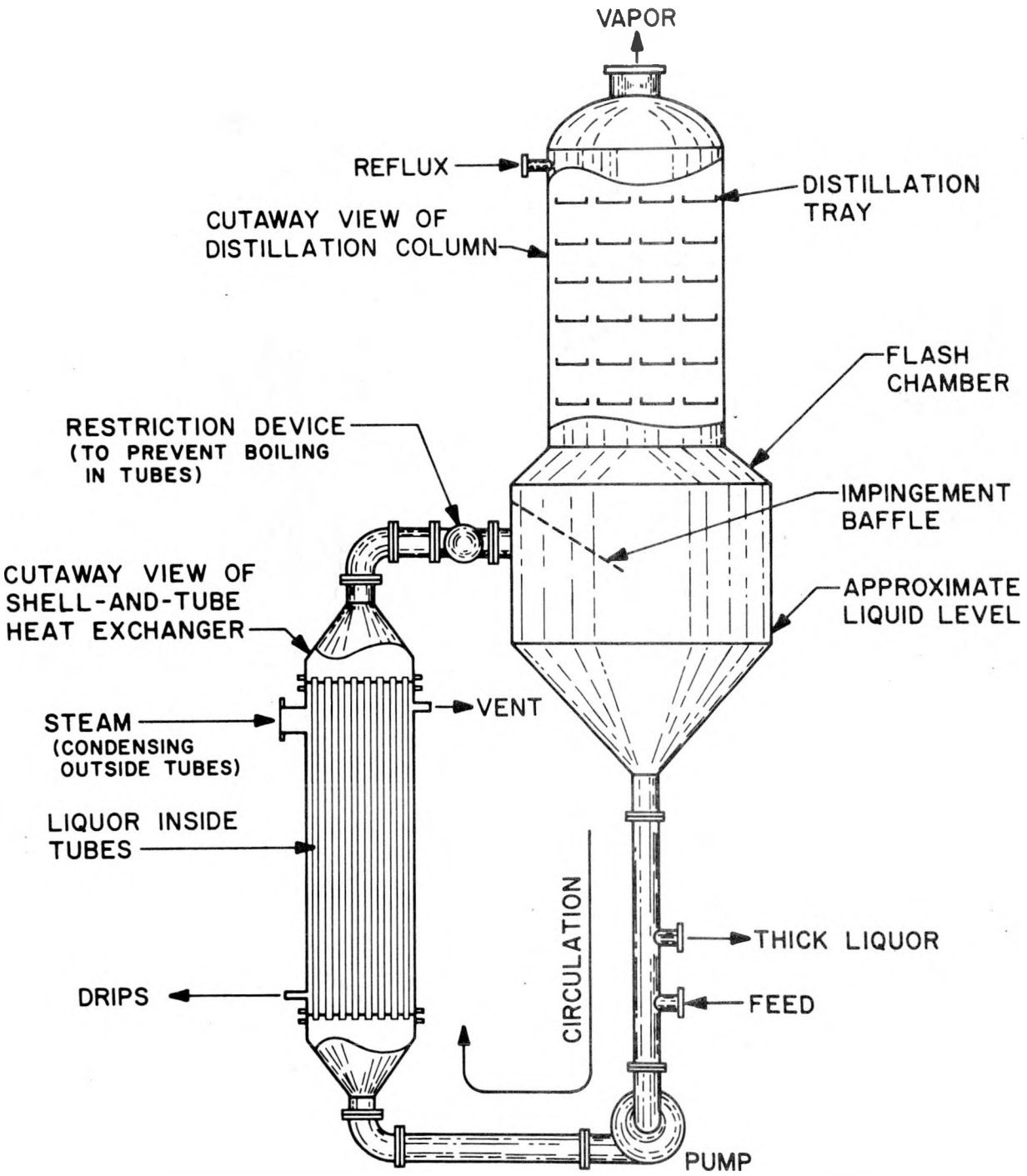


Figure 3.3-9 Forced-Circulation Evaporator with an External, Vertical, Single-Pass Heater and Restrictive Device to Prevent Boiling in Tubes[1]

1. Taken from Godbee, 1978.

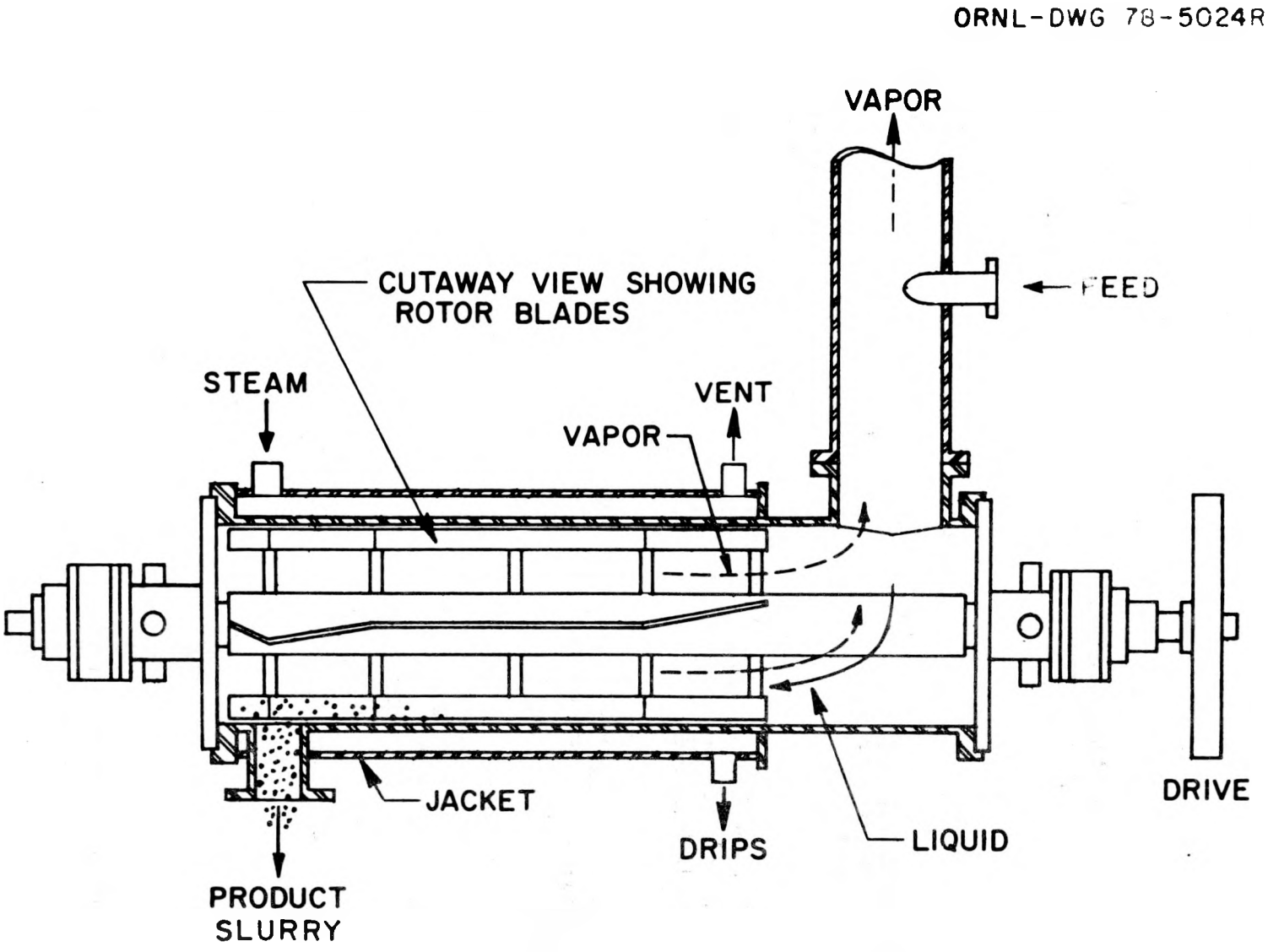


Figure 3.3-10 Wiped-Film Evaporator[1]

1. Taken from Godbee, 1978.

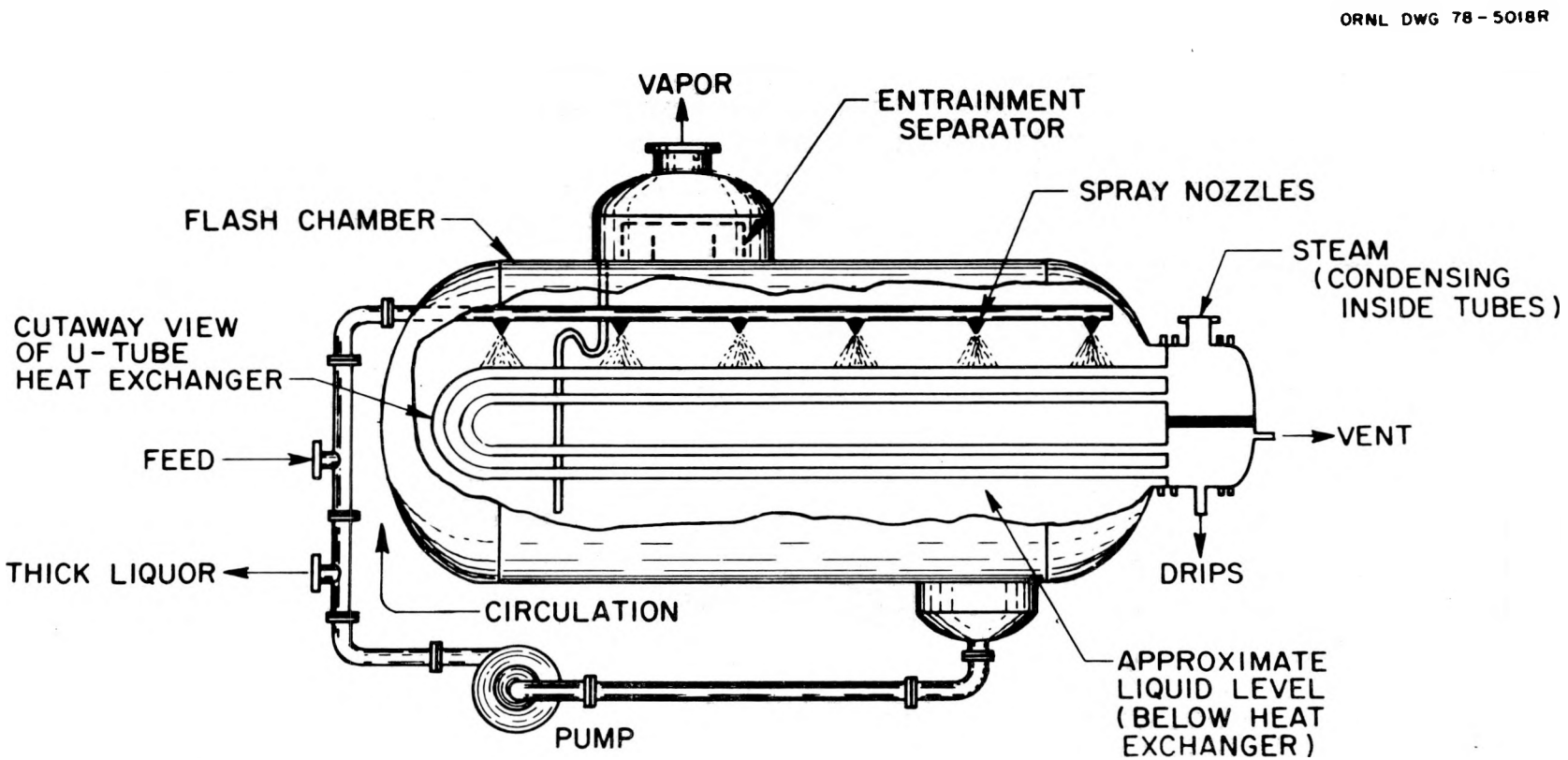


Figure 3.3-11 Spray-Film Evaporator with Horizontal U-Tube Heater[1]

1. Taken from Godbee, 1978.

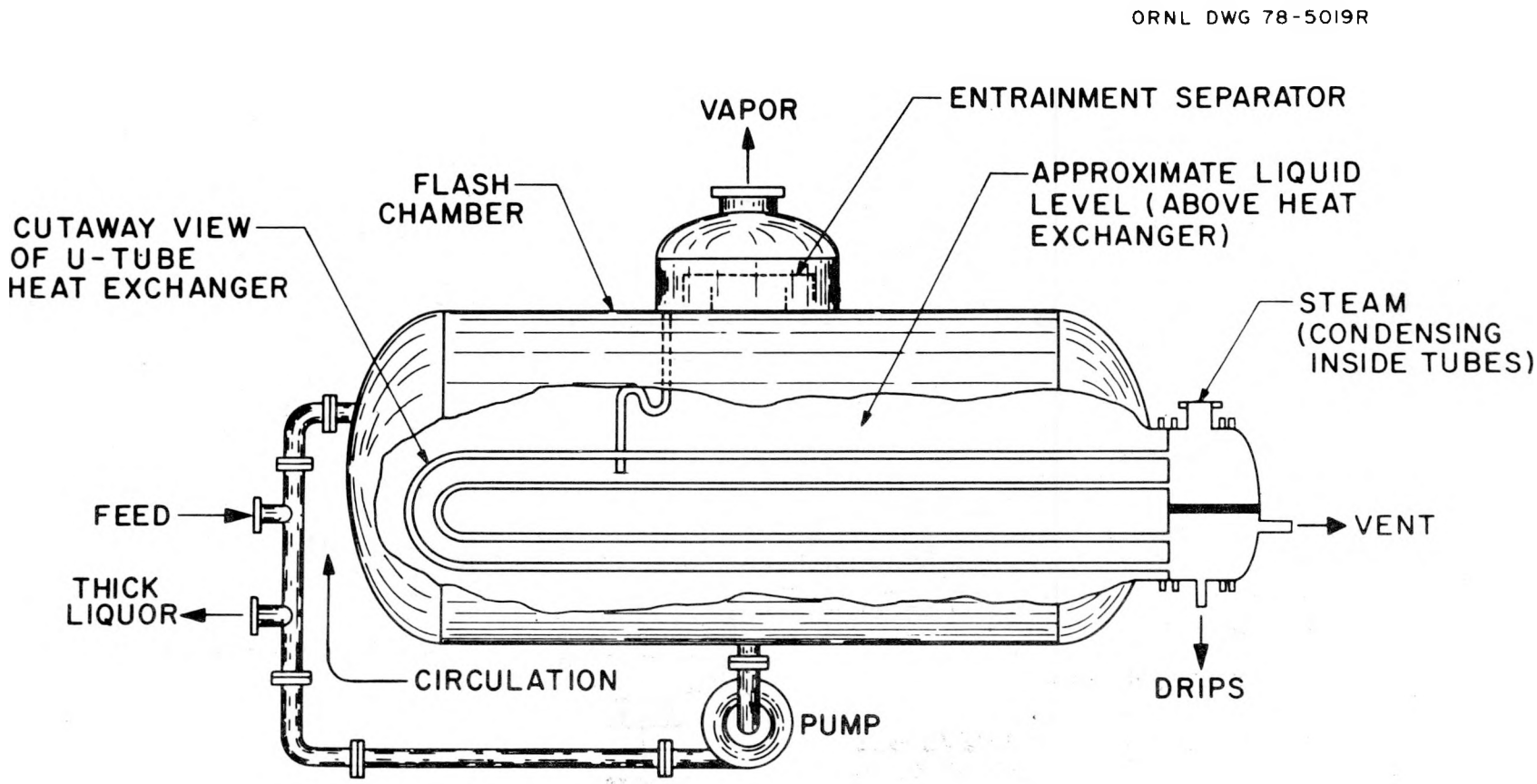


Figure 3.3-12 Submerged U-Tube Evaporator[1]

1. Taken from Godbee, 1978.

between 10 and 20, with the solids contents averaging approximately 12.5 percent by weight (Godbee, 1978). Table 3.3-2 summarizes the performance of evaporators used in LWRs. Table 3.3-3 lists the advantages and disadvantages of natural circulation, forced circulation, and submerged U-tube evaporators (Godbee, 1978).

Table 3.3-2 Performance Characteristics of Evaporators Used in LWRs[1]

Installation [Capacity] Reactor type	Evaporator Category[2]	Evaporator Designer and/or mfgr.	Stream treated	Evaporator capacity (gpm) Design	Operating Max	Operating Min	Operating Avg	Operating time (%)	Maintenance time (%)	Feed pH	Feed Cl (ppm)	Feed Pretreatment Filtered	Feed Pretreatment Degassed	Antifoam agent	Volume reduction (feed/thick liquor)	Notes
Beaver Valley 1 [2660 MW(t)] PWR	FC	Stone & Webster	Miscellaneous, chemical, and secondary system wastes	6	5				See comment			No	No	No		Loss of capacity due to solidification in thick-liquor pump and lines.
Brunswick 1 & 2 [2436 MW(t)] ea. BWR	NC	Aqua-Chem	Chemical and low-purity wastes	20				3–8 (5 avg)				No	No	Betz HT		Problems with entrainment, foaming, instrumentation, and plugged tubes.
	NC	Swenson	Chemical and low-purity wastes	50				55–82 (68 avg)				No	No	Betz HT	20–80	Problems with entrainment, foaming, and extensive corrosion (evaporator will be replaced).
Cook 1 [3250 MW(t)] PWR	SU	Westinghouse	Miscellaneous, chemical, detergent, and secondary system wastes	2	1.5	0.5	1.3	72.2	2.9	7.5–8.3 (8.0 avg)	10–80 (15 avg)	Yes	No	Dow H-10	20–100 (50 avg)	Poor condensate quality; NaOH added for pH control.
Crystal River 3 [2452 MW(t)] PWR	SU	(AMF)-Riley Beaird	Miscellaneous, chemical, and secondary system wastes	20	20	1	12.5	10–20 (15 avg)				No	No	Dow	10 avg	Components not readily accesible for decontamination and maintenance.
Ft. Calhoun [1420 MW(t)] PWR	SF	Aqua-Chem	Miscellaneous, chemical, detergent and secondary system wastes	17	14			44–66	8–9			Optional	No	GE		Gas stripper inoperable; low capacity due to plugging of spray nozzles; poor condensate quality; highly congested component arrangement; $Na_2S_2O_3$ is added for iodine control.
Maine Yankee [2440 MW(t)] PWR	FC	Stone & Webster	Boric acid	25								Yes	Yes	No		No corrosion problems to date.
	FC	Stone & Webster	Miscellaneous, chemical, and secondary system wastes	6			5–6							Ameril	100	Performance as expected.
Nine Mile Point 1 [1850 MW(t)] BWR	FC	HPD	Chemical wastes	20	20	5	15	30–60		6–9		No	No	No	15	Only casual operator attention is needed.
			Low-purity waste	20	20	5	15	40		6–9		Yes	No	Dow B	400	This is the same evaporator that is used for chemical waste; Na_2HPO_4 is added for chloride control.
Oconee 1, 2 & 3 [2568 MW(t)] ea. PWR	SF	Aqua-Chem	Miscellaneous, chemical, and secondary system wastes	7.5	5	2.5							No	Yes	10	System is being modified.
	SU	Westinghouse	Same as above	15	15	10	12	82						Yes	10	DF approximately a factor of 10^3 less than design DF.
Oyster Creek [1930 MW(t)] BWR	NC		Chemical and low-purity wastes	15	10	6	8			8–9.5					60	Tube plugging led to reduced waste processing.
Palisades [2212 MW(t)] PWR	SU	(AMF)-Riley Beaird	Boric acid and reactor makeup quality waste	20			20	20		5.1	0–0.4 (0.3 avg)	Yes	No	Yes	14	High degree of operator attention required; maintenance time is high.
	SU	(AMF)-Riley Beaird	Miscellaneous, chemical, and secondary system wastes	20			15	20		5.75–7.2		Yes	No	Yes	16	High degree of operator attention required; maintenance time is high; NaOH is added for pH control.
St. Lucie 1 [2570 ME(t)] PWR	NC	Aqua-Chem	Boric acid	20	20	2	10	~100	2	8.5 avg	35 avg	Yes	Yes	No	16–18 (17 avg)	Requires fairly close operator attention for satisfactory performance.

1. Taken from Godbee, 1978.
2. FC = forced circulation; NC = natural circulation; SF = spray film; SU = submerged U-tube.

Table 3.3-2 Performance Characteristics of Evaporators Used in LWRs (Cont'd)

Installation [Capacity] Reactor type	Evaporator Category	Evaporator Designer and/or mfgr.	Stream treated	Evaporator capacity (gpm) Design	Evaporator capacity (gpm) Operating Max	Min	Avg	Operating time (%)	Maintenance time (%)	Feed pH	Feed Cl (ppm)	Feed Pretreatment Filtered	Feed Pretreatment Degassed	Antifoam agent	Volume reduction (feed/thick liquor)	Notes
Three Mile Island 1 [2535 MW(t)] PWR	SU	(AMF)-Riley Beaird	Miscellaneous, chemical, and secondary system wastes	12.5				48				Optional	No	Dupont		Components are not easily accessible; excessive operator attention is required.
Turkey Point 3 & 4 [2200 MW(t)] ea. PWR	NC	Aqua-Chem	Boric acid	5			4	55				Yes	Yes	No	18–20	Both boric acid evaporators meet design specifications.
	SU	Westinghouse	Miscellaneous, chemical, detergent and secondary system wastes	15			12	55				Yes	Yes	Optional	18–20	Both waste evaporators perform below design specifications. NaOH can be added for pH control.
Yankee-Rowe [600 MW(t)] PWR	FC	Pantex	Boric acid, miscellaneous, chemical, and secondary system wastes	5	5	2.5	5	20–33 (25 avg)	<1	6–10 (8 avg)		No	No	No	20–100 (50 avg)	Drew Chemical Corp. L-113 antifoam being tested.
Zion 1 & 2 [3250 MW(t)] ea. PWR	SF	Aqua-Chem	Miscellaneous, chemical, detergent, and secondary system wastes	12			~12	92	8	7.3–8.4		No	No	Calgon C1	75–100	Components are not easily accessible.
	SU	Westinghouse	Same as above	15	10	6	8	82	16	7.3–8.4		No	No	Calgon C1	50–1000	Performs below design specification; condensate quality varies randomly; constant operator attention required.

Table 3.3-3 Advantages and Disadvantages of theTypes of Evaporators Used in LWRs[1]

	Evaporator type		
	Natural circulation	Forced circulation	Submerged U-tube
Advantages	1. Low cost 2. Large heating surface in one body 3. Low hold-up 4. Small floor space 5. Good heat-transfer coefficients at reasonable temperature differences (rising film) 6. Good heat-transfer coefficients at all temperature differences (falling film)	1. High heat-transfer coefficients 2. Positive circulation 3. Relative freedom from salting, scaling, and fouling	1. Very low headroom 2. Large vapor-liquid disengaging area 3. Good heat-transfer coefficients 4. Easy semiautomatic descaling
Disadvantages	1. High headroom 2. Generally unsuitable for salting and severely scaling liquids 3. Poor heat-transfer coefficients of rising-film version at low temperature differences 4. Recirculation usually required for falling-film version	1. High cost 2. Power required for circulating pump 3. Relatively high hold-up or residence time	1. Unsuitable for salting liquids 2. High cost 3. Relatively high hold-up or residence time
Best applications	1. Clear liquids 2. Foaming liquids 3. Corrosive solutions 4. Large evaporation loads 5. High temperature differences–rising film, low temperature differences–falling film 6. Low-temperature operation–falling film	1. Crystalline product 2. Corrosive solutions 3. Viscous solutions	1. Limited headroom 2. Small capacity 3. Severely scaling liquids
Frequent difficulties	1. Sensitivity of rising-film units to changes in operating conditions 2. Poor feed distribution to falling-film units	1. Plugging of tube inlets by salt deposits detached from walls of equipment 2. Poor circulation due to higher than expected head losses 3. Salting due to boiling in tubes 4. Corrosion–erosion	1. Slow response to changes in control settings 2. Poor level control in vacuum units

1. Taken from Godbee, 1978.

3.4 Solidification of Low- and Intermediate-Level Wastes

3.4.1 Introduction

There are five solidification agents that are currently considered for use in commercial nuclear power plants. They are as follows:

a. Cement
b. Urea-formaldehyde (UF)
c. Bitumen
d. Polyester resin
e. Dow system resin

Absorbent materials such as vermiculite, which had been used extensively in LWRs, are no longer used. When mixed with waste these porous materials will soak up the free water and retain it. This being the case, there is no chemical or physical binding of the waste and the final product is not a monolithic solid. This method of waste fixation is no longer used because of limitations imposed by the burial sites.

Of the five solidification agents listed above, cement is the only nonorganic binding material that reacts chemically with the water contained in the waste to form an inert solid product.

Urea-formaldehyde, polyester, and Dow system agents are thermosetting polymers. Thermosetting polymers are usually stronger at higher temperatures and set irreversibly because they are not softened by increased temperature.

Ordinarily, bitumen behaves as a thermoplastic polymeric material and is sometimes so categorized. Most thermoplastic polymers are synthetic organic materials which can be reversibly softened by heating and formed in the softened state by processes such as extrusion.

Systems using cement or urea-formaldehyde have been installed in many United States plants. Bitumen systems have been used in almost all European plants.

So far, systems using Dow System resin or polyester have not been installed in any United States plant, and their use has been limited to prototype systems and isolated special applications.

3.4.1.1 Why Waste Is Solidified

In addition to treating liquid waste streams to maximize the quantity of water recycled to the plant and to minimize the quantity of waste requiring disposal, the objectives of low-level waste management are as follows:

a. To package the by-product so it is safe for transportation and disposal

b. To provide transportation that protects the public from radiation exposures and hazards in the event of an accident

c. To provide disposal that is safe for the environment.

Means for the stabilization of low-level waste containing free liquids are needed to minimize the potential release of radionuclides to the biosphere during in-plant handling, offsite shipment, and disposal. Minimizing the potential for radionuclide release will guard the public health and safety.

However, to reach these goals a stabilized waste must possess certain qualities. Mechanical strength is of primary importance during in-plant handling, transportation, and disposal. During an accident a waste with poor mechanical properties may fracture and disperse into the environment. Also, because of its increased surface area a fragile waste form would result in increased leachability. The thermal stability of a solidified waste form is a concern primarily because accident conditions involving fire are possible. The accident may cause decomposition, degradation of mechanical properties, and dispersion of radionuclides as gas or aerosol. Also, leachability is a primary concern because in shallow land burial, radionuclide release is principally the result of groundwater interactions. Leachability refers to the removal of radionuclides from the solidified waste package by fluids. Dissolution, diffusion, and chemical reactions may contribute to this release.

3.4.1.2 NRC Requirements

As part of the licensing procedure the NRC has established criteria for acceptable methods of operating the solid radioactive waste system in LWRs. In November 1975 the NRC issued a Regulatory Standard Review Plan and Branch Technical Position for Section 11.4 of the Safety Analysis Report, which covers solid waste systems (NRC, 1975a, 1975b). The purpose of these documents is to inform the nuclear industry and the general public of regulatory procedures and policies. Put simply, the NRC position on the treatment of solid radwaste is that all waste should be in a solid, immobile form before shipment from the facility generating the waste.

The NRC Branch Technical Position applies to the waste solidification systems installed in plants that were licensed after the Branch Technical Position document was issued. There are no NRC requirements for the installation of solidification systems in plants licensed before the position document was issued.

A summary of the criteria established by the NRC's Branch Technical Position document follows:

- All waste should be in a solid, immobile form before shipment from the site.

- Spent resin and filter sludges should be combined with a suitable binding agent (such as cement or urea-formaldehyde) and formed into a solid matrix, thereby mitigating the consequences if shipping containers are ruptured.

- For normal operation, shipment of liquids offsite is unacceptable. Means should be provided for the complete solidification of all wastes that can be reasonably expected to be generated during normal operation, including anticipated operational occurrences.

- The use of absorbents, such as vermiculite, is not an acceptable substitute for solidification.

- Complete solidification of wastes should be ensured by the implementation of process control programs or methods to detect free liquids within container contents prior to shipment.

The NRC states that the waste should be solidified and that complete solidification should be ensured, but it does not define the thermal, mechanical, chemical, physical, and leachability criteria for an acceptable solidified mass.

3.4.1.3 DOT Regulations

The current DOT regulations do not require wastes to be solidified. According to the DOT regulations, most of the liquid waste produced in LWRs can be shipped in bulk tanks or be packaged with excess absorbent material.

The International Atomic Energy Agency (IAEA) safety standards, safety series No. 6, "Regulations for the Safe Transportation of Radioactive Materials," 1973 Revised Edition (IAEA, 1973), includes a new classification of radioactive material under the category of low-level solids (LLS) for which a leachability criterion has been established.

The LLS category is defined in the IAEA regulations as follows:

1. Solids (for example, consolidated wastes, activated materials) in which

   a. The activity under normal transport conditions is, and remains, distributed throughout a solid or a collection of solid objects; or is, and remains, uniformly distributed in a solid compact binding agent (such as concrete, bitumen, or ceramic);

   b. The activity is, and remains, insoluble so that, even under loss of packaging, the loss of radioactive material per package resulting from the effects of wind

and rain, and from total immersion in water, is limited to less than $0.1A_2$ in a period of one week; and

c. The estimated activity averaged throughout the radioactive material does not exceed $2 \times 10^{-3}A_2$ Ci/g. (A_2 is the number of curies in a Type A quantity of normal form radioactive material.)

2. Objects of nonradioactive material contaminated with radioactive material, provided that the radioactive contamination is not in a readily dispersible form, and the level of contamination averaged over $1m^2$ (or the area of the surface if this is less than $1m^2$) does not exceed 20 $\mu Ci/cm^2$ (4.4×10^9 dpm/100 cm^2) for beta and gamma emitters and the low toxicity alpha emitters; or 2 $\mu Ci/cm^2$ (4.4×10^8 dpm/100 cm^2) for other alpha emitters.

The LLS category was established for radioactive material that did not meet low specific activity (LSA) criteria, but still did not need to depend on stricter specification packaging for its safe transport. As a result, liquid waste or solid waste with entrained or entrapped liquid (such as spent resin and filter sludge) that is solidified in such substances as cement, plaster of paris, or urea-formaldehyde will be the main type of radwaste shipped as LLS. Currently, most such wastes are being shipped as Type A.

In accordance with the IAEA criteria, the leachability of LLS is limited to less than $0.1A_2$ in a period of 1 week. The A_2 values of the IAEA standard and the weekly allowable leach rates are given in Table 3.4-1.

3.4.1.4 Limitation by the Burial Sites

All United States commercial disposal sites require that wastes in liquid form be solidified before arrival at the site. The three operating sites (Barnwell, Beatty, and Richland) accept wet solids (for example, dewatered resins).

The current disposal-site criteria for solidified waste do not define detailed characteristics of an acceptable solidified mass. The state health agencies have adopted an interim policy under which they review and accept for burial those wastes that are immobilized with specified solidification agents.

In view of the current studies related to radionuclide migration at the shallow land burial sites (Meyer, 1976), it can be expected that a more detailed criterion for acceptance of a solidified mass will be imposed by the regulatory agencies that have jurisdiction over waste disposal (DOE, NRC, EPA, and state agencies).

Table 3.4-1 IAEA Allowable Leach Rate for Low-Level Solid Waste[1]

Nuclide	Package Limit A_2 (Ci)	Allowable Leach Rate 0.1 x A_2 (Ci/week)
Sr-89	40	4
Sr-90	0.4	0.04
Zr-95	20	2
Ru-103	30	3
Ru-106	7	0.7
Te-127m	40	4
Te-129m	30	3
Cs-134	7	0.7
Cs-137	9	0.9
Ce-141	200	20
Ce-144	7	0.7
Cr-51	600	60
Mn-54	20	2
Fe-55	1,000	100
Fe-59	10	1
Co-58	20	2
Co-60	7	0.7

1. Taken from IAEA, 1973.

Table 3.4-2 Unnotched IZOD Impact Strength of Portland Type II Neat Cements[1]

Water/Cement Ratio	Impact Strength in.-lb/in.	Impact Strength Standard Deviation in.-lb/in.
0.20	3.38	0.46
0.30	4.89	1.96
0.40	4.97	1.64
0.50	4.35	0.88
0.60	4.16	1.15

1. Taken from Colombo, 1977a.

3.4.2 Cement

3.4.2.1 Process and Material Description

Cement is the most commonly used material for the immobilization of radwaste. Of the plants surveyed, 52% of those with operating solidification systems use Portland Type I cement. However, type II may be used where a moderate sulfate concentration exists. Cement is made from a mixture of approximately 80% carbonate of lime (from limestone, chalk, or marl) and 20% clay (in the form of clay, slag, or shale). A chemical analysis of Type I cement (Baumeister, 1967) shows the following compounds in the approximate amounts listed:

a. Silica SiO_2 - 21.9%
b. Alumina Al_2O_3 - 6.9%
c. Iron oxide Fe_2O_3 - 2.9%
d. Calcium oxide CaO - 62.9%
e. Magnesium oxide MgO - 2.5%
f. Sulfuric oxide SO_3 - 1.7%
g. Alkalies[1] R_2O_3 - 1.0%
h. Insoluble residue - 0.2%

Loose Portland cement has a density of 94 lb/ft^3. When set, the density is about 196 lb/ft^3. In order to form a workable mixture, the minimum water-to-cement ratio needed is 0.25 by weight.

This water reacts chemically with the cement, in a reaction called hydration, and becomes part of the solidified product. Additional water is needed if the material to be solidified absorbs water.

For use in radioactive waste solidification the reaction must result in a final product in which the radioactive ions are firmly bound in stable chemical combination with the cement ions. This bonding results in a product in which the radioactive ions are not easily leached out on exposure to water. Retention by cement is good for transition metals (for example, cobalt and manganese), but it is poor for alkali metals and alkaline earths such as cesium and strontium. Acid solutions do not yield a solid mass with cement. Moreover, solutions containing borates (boric acid and sodium borate), which are common ingredients in PWR liquid wastes, slow down the setting of cement. Any ammonium ion present in the wastes will react with the cement and release ammonia gas.

Several additives have been used to improve the setting properties, fission-product retention, and packaging efficiency of cement. These include:

- Vermiculite (used by Westinghouse systems)

1. Represents any of the alkalai materials, Li, Na, K, Rb, or Cs.

- Sodium silicate (used by UNI and Delaware Custom Materials systems)

- Metso Beads[1] (sodium metasilicates used in HNDC systems).

The most commonly used additive is sodium silicate, which has four major advantages: it reduces the set up time; it reduces the chance of free water; it tends to neutralize wastes containing acids; and it improves the packaging efficiency.

Solidification systems using cement can be divided into two categories: those in which the waste is mixed with the cement prior to being placed in the container; and those in which the waste and the cement are placed in the container separately and then mixed. Systems designed for commercial nuclear power plants several years ago were of the first type. Systems sold recently have been the second type. In these systems the dry cement is placed in the container before the waste is added. The quantity of cement is dependent on the type of waste to be solidified.

The container is moved to the fill position and the waste is added, the container is sealed, and the waste cement mixture is mixed thoroughly.

The reaction of cement with water is exothermic; that is, heat is generated in the reaction. The average amount is about 120 cal/g for complete hydration of cement.

3.4.2.2 Waste Characteristics

A number of studies have been performed to determine the properties of cement/waste solidified mixtures. The most extensive investigation was conducted by Colombo and Neilson (1976a, b, c, 1977a, b, c). A summary of their findings has been included here in Tables 3.4-2 through 3.4-5, and Figures 3.4-1 and 3.4-2. Properties of cement-waste forms are also summarized in Table 3.4-6.

3.4.2.3 Volume Effect of Solidification With Cement

The amount of water needed to solidify one 94-pound bag of cement can vary from approximately 4 to 10 gallons. The resultant volume can be estimated by the absolute-volume computation method. The

1. Metso Beads is the registered trademark of Philadelphia Quartz Company.

Table 3.4-3 Portland Type I Cement Composition (Normalized)

Constituent	Wt%
Calcium	0.481
Oxygen	0.355
Silicon	0.109
Aluminum	0.034
Iron	0.020

Table 3.4-4 Compression Strength of Portland Type II Cement Waste Forms[1]

Waste Type[2]	Waste/Cement Weight Ratio	Waste Form Density, g/cm^3	Packing Efficiency, %[3]	Compression Strength, psi ± 1
1	2.0	1.29	81.8	48 ± 5
1	2.4	1.33	90.2	68 ± 10
1	2.6	1.35	92.8	41 ± 8
2A	1.8	1.23	72.8	48 ± 4
2A	2.0	1.22	75.7	45 ± 7
2B	1.6	1.65	89.4	482 ± 98
2B	2.0	1.57	92.1	420 ± 17
2B	2.4	1.53	94.3	103 ± 25
3A	0.6	2.03	63.2	3271 ± 262
3A	1.2	1.84	83.4	576 ± 100
3A	1.7	1.74	91.7	177 ± 61
3B	0.6	2.09	67.4	3161 ± 257
3B	1.2	1.77	83.2	72 ± 27
3B	1.7	1.67	90.7	40 ± 22

1. Taken from Colombo 1977c.

2. Waste types as defined in Table 3.4-5.

3. Packing efficiency = $\frac{\text{Initial waste volume}}{\text{Waste form volume}} \times 100$.

Table 3.4-5 Simulated Waste Formulations[1]

1. Bead Resin Waste

Material	Resin Properties
Water (wt%)	50.
Bead resin (IRN-150)[2] (wt%)	50.
Temperature (°F)	70
pH	7

2a. BWR Precoat Filter Cake (With Powdered Resin)

Material	Filter Cake Properties
Water (wt%)	50.
Anion powdered resin (PAO)[3] (wt%)	20.
Cation powdered resin (PCH)[3] (wt%)	20.
Crud[4] (wt%)	5.
Sodium chloride (wt%)	5.
Temperature (°F)	70
pH	7

2b. BWR Precoat Filter Cake (With Diatomaceous Earth)

Material	Filter Cake Properties
Water (wt%)	50.
Diatomaceous earth (wt%)	40.
Crud[4] (wt%)	10.
Temperature (°F)	70
pH	7

3a. BWR Chemical Regenerative Waste of a Forced Recirculation Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	75.
Sodium sulfate (wt%)	22.9
Sodium chloride (wt%)	2.0
Crud[4] (wt%)	0.1
Temperature (°F)	170
pH	6

Table 3.4-5 Simulated Waste Formulations[1] (Cont'd)

3b. PWR Chemical Regenerative Waste of a Forced Recirculation Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	73.4
Sodium sulfate (wt%)	14.9
Ammonium sulfate (wt%)	9.6
Sodium chloride (wt%)	2.0
Crud[4] (wt%)	0.1
Temperature (°F)	170
pH	2.5 to 4.0

3c. Boric Acid Waste of a Forced Recirculation Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	87.9
Boric acid (wt%)	12.0
Crud[4] (wt%)	0.1
Temperature (°F)	170
pH	3.5

3d. Decontamination Waste of a Forced Recirculation Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	80.
NUTEK-700[5] (wt%)	9.4
EDTA (wt%)	5.
Citric acid (wt%)	5.
Crud[4] (wt%)	0.2
Hydraulic Oil No. 2 (wt%)	0.2
Lubricating Oil No. 20 (wt%)	0.2
Temperature (°F)	170
pH	5

Table 3.4-5 Simulated Waste Formulations[1] (Cont'd)

4a. BWR Chemical Regenerative Waste of a Thin Film Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	50.
Sodium sulfate (wt%)	45.8
Sodium chloride (wt%)	4.0
Crud[4] (wt%)	0.2
Temperature (°F)	150 to 250
pH	6

4b. PWR Chemical Regenerative Waste of a Thin Film Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	50.
Sodium sulfate (wt%)	29.
Ammonium sulfate (wt%)	16.8
Sodium chloride (wt%)	4.0
Crud[4] (wt%)	0.2
Temperature (°F)	150 to 250
pH	1.8 to 4.0

4c. Boric Acid Waste of a Thin Film Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	50.
Boric acid (wt%0	49.8
Crud[4] (wt%)	0.2
Temperature (°F)	150 to 250
pH	2.5 to 3.5

Table 3.4-5 Simulated Waste Formulations[1] (Cont'd)

4d. Decontamination Waste of a Thin Film Evaporator

Material	Evaporator Bottom Properties
Water (wt%)	50.
NUTEK-700[5] (wt%)	20.
EDTA (wt%)	9.8
Citric Acid (wt%)	19.
Crud[4] (wt%)	0.2
Hydraulic Oil No. 2 (wt%)	0.5
Lubricating Oil No. 20 (wt%)	0.5
Temperature (°F)	150 to 250
pH	5

1. Taken from Colombo, 1977a.
2. Rohm and Haas Co., Philadelphia, Pa. 19105
3. Ecodyne Corp., Union, N.J. 07083
4. Fine air cleaner test dust no. 1543094, AC Spark Plug Division, General Motors Corp., Flint, Michigan 48556
5. Compound for the dissolution of calcium sulfate scale, Nuclear Technology Corp., Amston, Conn. 06231

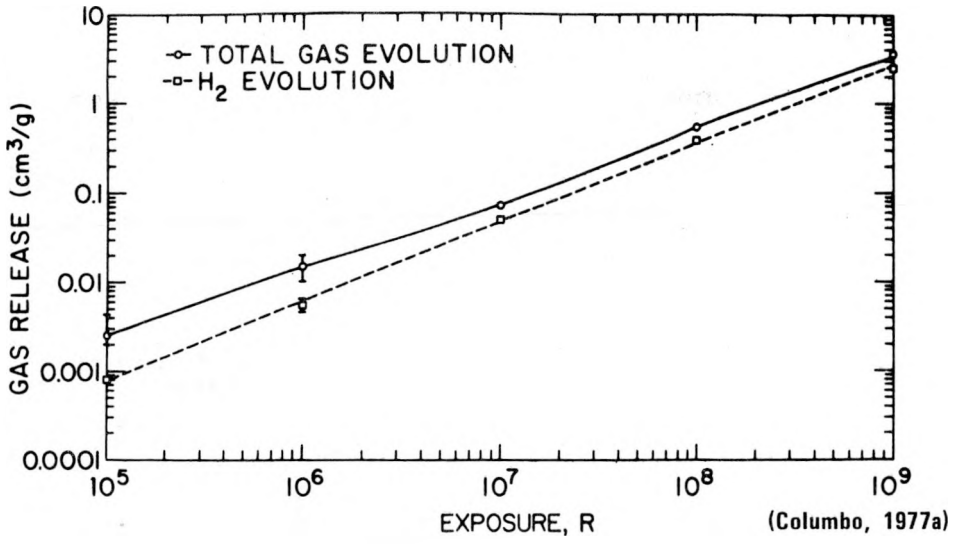


Figure 3.4-1

COBALT-60 RADIOLYSIS GAS RELEASE FROM PORTLAND TYPE II NEAT CEMENT (W/C = 0.5) AT 250°C, DOSE RATE = 4.74×10^6 R/HR

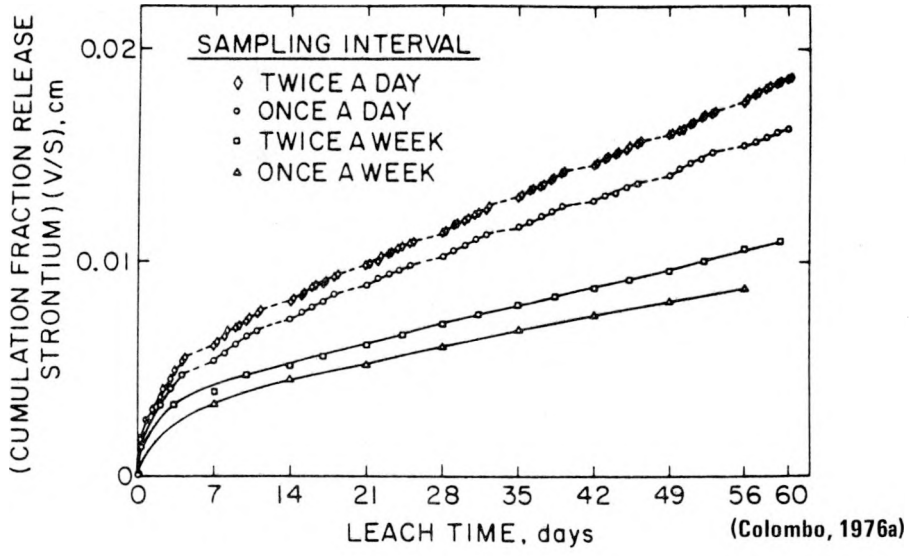


Figure 3.4-2

RELEASE OF STRONTIUM FROM PORTLAND TYPE II NEAT CEMENT BY STATIC LEACHING IN DISTILLED WATER AS A FUNCTION OF LEACHANT CHANGING INTERVAL

NOTE: Leach rates are reported in this figure as [(cumulative fraction cesium release) x (specimen volume to exposed surface ratio)] or $(\Sigma a_n/A_o)(V/S)$ versus Σt_n where,

a_n = amount of species removed during leaching period n

Σa_n = cumulative amount of the species of interest during all leaching periods

A_o = amount of the species of interest initially present in the speciment

V = volume of the specimen, cm^3

S = exposed geometric surface area of the specimen, cm^2

t_n = leachant renewal period, days

Σt = cumulative leach time, days

Table 3.4-6 Properties of Waste Forms Solidified With Cement

Property	Description
Leachability	Leachability depends on the radionuclides present, the leachant composition, and the chemical content of waste. Cement retention is good for transition metals (e.g., cobalt and manganese) but it is poor for alkali metal and alkaline earth compounds (e.g., cesium and strontium). The retention of the latter compounds can be improved by the use of additives, such as sodium silicate.
Thermal	Cement-waste forms are thermally suitable and incombustible. It has good fire resistance. During a fire, the solidified waste form may fail due to gas generation as a result of waste decomposition or because of differences in thermal expansion between the concrete and the waste. The thermal conductivity depends on the aggregates and is generally 3.4 - 8.6 x 10^{-3} Cal/sec-cm-^{o}C for commercial construction concrete.
Mechanical	Strength depends on the waste material and cement/waste ratio. Cement solidification with alkaline waste solutions has good compression strength. Cellulose filter media solidified with cement generally has poor mechanical strength. Acidic wastes solidified with cement also give a product that is mechanically weak.
Product stability	Stability is good both in a sealed system and exposed to air. Exposure to moisture helps the curing process. Concrete waste forms may be affected due to gas generation as a result of waste decomposition. It is susceptible to cracking and degradation as a result of freeze-thaw cycling.

Table 3.4-6 Properties of Waste Forms Solidified With Cement (Cont'd)

Property	Description
Radiation resistance	Normally, it is not significantly affected by the activity in the range of that contained in the power-reactor low-level waste. However, the pressure of water and other components susceptible to radiolysis could result in gas generation and pressurization problems.
Free water	Free water is not generally a problem although improper cement/waste mixing ratios or the presence of some acids in the waste solution can produce free water in a package.
Interaction with container	Portland cement is normally compatible with steel. Furthermore, it contains some calcium hydroxide formed during hydration. This promotes an alkaline reaction at the concrete-steel interface and tends to inhibit corrosion.
Resistance to chemical attack	Portland cement is not resistant to acids. Constant acid attack could result in softening of the surface layers. Deterioration of concrete may take place in regions where groundwater and alkali soil contains sulfates of magnesium and sodium. These salts react with the hydrated cacium aluminate to form crystals of calcium sulfoaluminates accompanied by considerable expansion that may result in eventual disintegration.
Resistance to biodegradation	Concrete itself is not biodegradable but bacteria and fungi could cause damage by mechanical action and by secretion of organic acids.

specific gravity (S.G.) of cement can be taken as 3.15 with reasonable accuracy. For example, the volume that results from mixing 94 pounds of cement with waste slurry containing 9 gallons of water (75 pounds) and 25 pounds of bead resins can be computed as follows:

$$\text{Total volume} = \text{water volume } (ft^3) + \frac{\text{Cement wt (lb)}}{\text{Cement S.G. x 62.4 } (lb/ft^3)}$$

$$+ \frac{\text{Resin wt (lb)}}{\text{Resin S.G. x 62.4 } (lb/ft^3)} \qquad (3.4.1)$$

Assuming 3.15 as the S.G. of cement and 1.21 as the S.G. of bead resin, the total solid mass volume is

$$1.20 + 0.48 + 0.4 = 2.08 \text{ ft}^3. \qquad (3.4.2)$$

The volume increase factor, also called the packaging factor in this case, is 2.4 and the packaging efficiency is 1/2.4, or 0.42. Table 3.4-7 is a list of typical packaging efficiencies for various types of waste. Despite years of experience, solidification of wastes with cement is still an art. Because reactor waste composition and chemistry are highly variable, each new waste application must be considered individually to determine the optimum ratio.

3.4.3 Urea-Formaldehyde (UF)

3.4.3.1 Process Description

Urea-Formaldehyde has been offered by Protective Packaging Incorporated (a subsidiary of Nuclear Engineering Company) as a solidification agent for LWR wastes since 1971. Other firms that have marketed solidification systems using UF include Chem-Nuclear Systems, Hittman Nuclear and Development Corporation, and United Nuclear Incorporated. Urea-Formaldehyde resin is commercially available from Dow Chemical Company (Cynaloc 62), Borden Chemical Company (Coso Resin No. 2), and Protective Packaging Incorporated (Tigerlok).

UF used in LWR radwaste solidification processes is a viscous, water-soluble liquid containing partially polymerized monomethylolurea, dimethylolurea, and formaldehyde. Complete polymerization is initiated by the addition of an acid catalyst. The chemical composition of the urea molecule is H_2NCONH_2, and the chemical composition of the formaldehyde molecule is HCHO.

In a typical solidification process, the UF is first mixed with neutral (pH adjusted to approximately 7) waste solutions. Approximately one part UF by volume is mixed with two parts waste by volume. A concentrated solution of an acid such as sodium bisulfate is then added to initiate polymerization. The mixture will

Table 3.4-7 Volume Increase Factors for Waste Solidification in Cement

Waste Type	Volume Increase Factor[1]	Packaging Efficiency[2]
Spent resin		
(33 wt% solids)	1.30	.76
(50 wt% solids)	1.23	.81
25 wt% $NaSO_4$	1.50	.66
12.5 wt% Boric acid	2.45	.41
50 wt% mixed sulfate solution	1.30	.76
Dry salt, (calcium sulfate)	1.74	.57
Filter sludge (50 wt% solids)	1.23	.81
Incinerator ash[3]	1.75	.57

1. Volume increase factor = $\frac{\text{Waste volume + binder volume}}{\text{Waste volume}}$

2. Packaging efficiency = $\frac{\text{Waste volume}}{\text{Waste volume + binder volume}}$

3. For 1,000 ft^3 of uncompacted combustible trash burned in an incinerator, with a volume reduction factor of 80, a cement system will generate 22 ft^3 of solidified waste.

start gelling after about 3 minutes and cures into a relatively hard solid mass within a few hours. The initial gelling time can be controlled by acidity of the catalyst.

The UF does not react chemically with waste. It polymerizes into a honeycomb-type microstructue within whose interstitial spaces the waste is confined.

3.4.3.2 Waste Characteristics

Colombo and Neilson (1976a, b; 1977a, b) have investigated various properties of UF waste products. Table 3.4-8 shows impact/strength and weight loss properties. Table 3.4-9 shows the compressive strength of six generic LWR waste types solidified with UF. Leachability properties for cesium-137 and strontium-85 are given in Figures 3.4-3 and 3.4-4 respectively. Figure 3.4-5 shows the radiolysis gas release of UF. Table 3.4-10 gives the basic composition of urea-formaldehyde. This list does not include any additives that may be added to enhance the gelling properties of UF. These additives are proprietary. A summary of various properties of UF waste products is given in Table 3.4-11.

During the actual field application, the UF solidification process may encounter some operational difficulties if no means are provided to eliminate the following adverse conditions.

a. In solidification of wastes containing ion-exchange bead resin the denser resin beads tend to settle to the bottom of the container. If this action takes place before the UF begins to gel, all the resins will be segregated at the bottom and the UF at the top. The result will be a two-phase inhomogeneous solid mass.

b. Because of the sensitivity of UF to acidic solutions, the pH of the waste must be adjusted to a neutrality. If this is not done then gelling will either be initiated prematurely or will not occur at all. The pH must also be considered in the solidification of ion-exchange resins that may interact with the catalyst and prevert the polymerization and gelling.

Table 3.4-8 Unnotched IZOD Impact Strength and Weight Loss of Urea-Formaldehyde Specimens on Exposure to Ambient Air (66°F, 48% R.H.). Specimen Width Is 0.5 Inch[1]

Time in Ambient Air, Days	% Original Weight	Impact Strength in-lb/in	Impact Strength Standard Deviation
0	100	1.2 ± 0.1	.047
1	61 ± 4	1.4 ± 0.1	.087
2	24 ± 1	0.9 ± 0.1	.057
3	24 ± 1	0.8 ± 0.1	.070
5	23 ± 1	0.8 ± 0.1	.022

1. Taken from Colombo, 1976a.

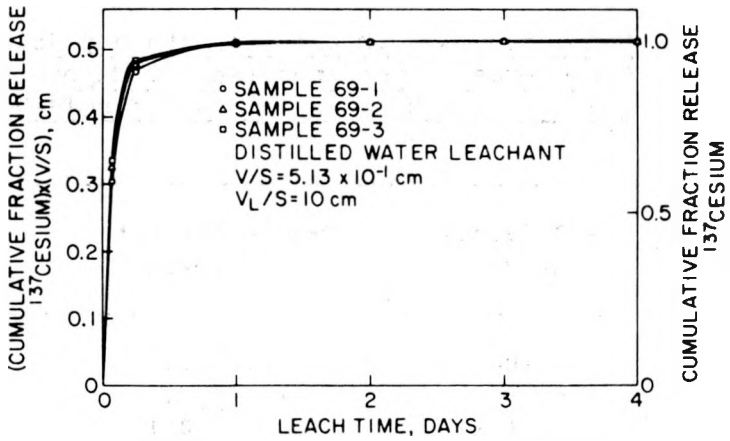
Table 3.4-9 Compression Strength of Urea-Formaldehyde Waste Forms[1]

Waste Type[2]	Waste/UF Weight Ratio	Waste Form Density, g/cm^3	Packing Efficiency, %[3]	Compression Strength psi ±1
1	2.6	1.13	77.8	78 ± 5
2A	2.0	1.18	72.9	384 ± 47
2B	2.0	1.21	70.9	387 ± 53
3A	1.2	1.23	56.1	67 ± 12
3B	1.2	1.25	58.3	61 ± 22
3C	2.0	1.12	73.4	95 ± 43

1. Taken from Colombo, 1977c.
2. Waste types as defined in Table 3.4-5
3. Packing efficiency = $\frac{\text{Initial waste volume}}{\text{Waste form volume}} \times 100.$

Table 3.4-10 Urea-Formaldehyde Composition

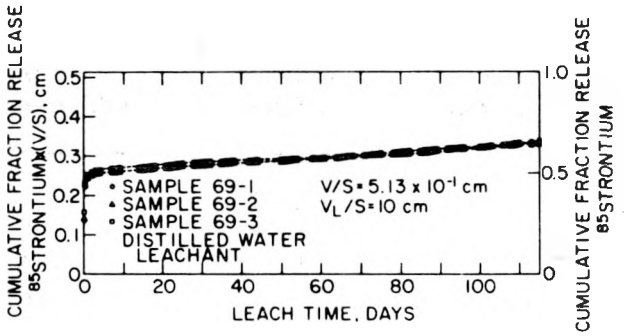
Constituent	Weight Percent
C	33
H	3
O	44
N	20



(Colombo, 1977b)

Figure 3.4-3

RELEASE OF CESIUM-137 FOR STATIC LEACHING OF UREA-FORMALDEHYDE SAMPLES IN DISTILLED WATER



(Colombo, 1977b)

Figure 3.4-4

RELEASE OF STRONTIUM-85 FOR STATIC LEACHING OF UREA-FORMALDEHYDE SAMPLES IN DISTILLED WATER

NOTE: Leach rates are reported in these figure as [(cumulative fraction cesium release) x (specimen volume to exposed surface ratio)] or $(\Sigma a_n/A_o)(V/S)$ versus Σt_n where,

a_n = amount of the species of interest removed during leaching period n

Σa_n = cumulative amount of the species of interest during all leaching periods

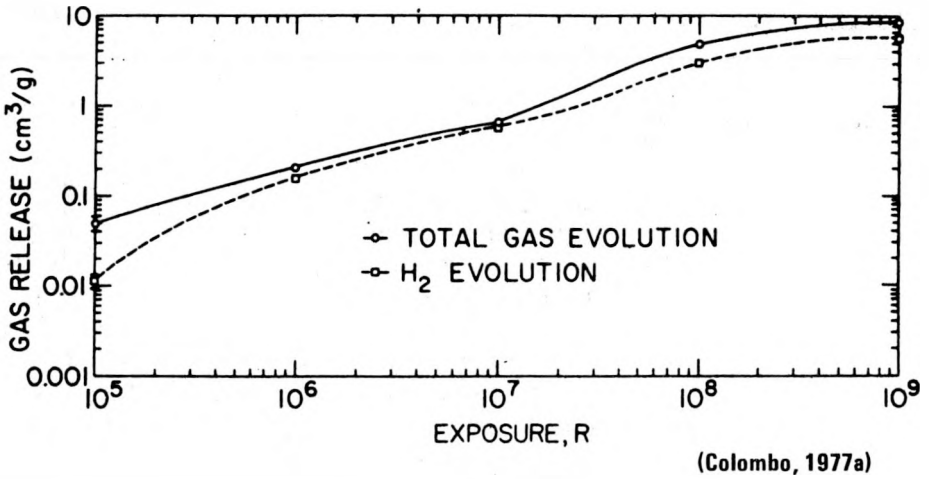
A_o = amount of the species of interest initially present in the specimen

V = volume of the specimen, cm^3

S = exposed geometric surface area of the specimen, cm^2

t_n = leachant renewal period, days

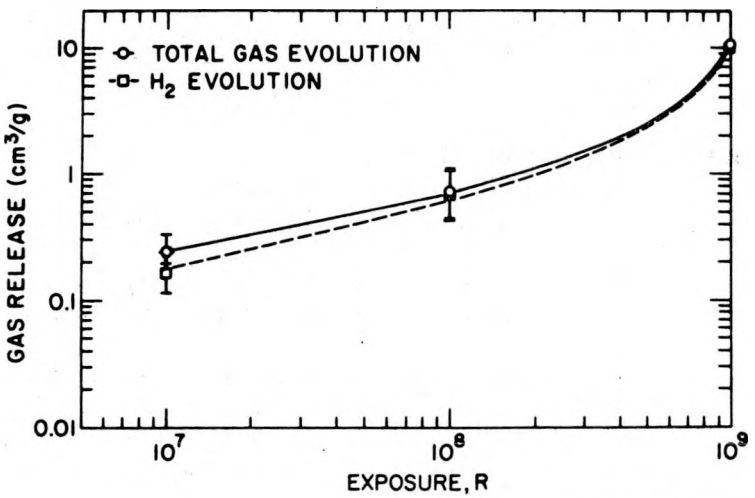
Σt_n = cumulative leach time, days



(Colombo, 1977a)

Figure 3.4-5

COBALT-60 RADIOLYSIS GAS RELEASE FROM UREA-FORMALDEHYDE AT 250°C (1 PART RESIN: 2 PARTS WATER, BY VOLUME), DOSE RATE = 4.84×10^6 R/HR



(Colombo, 1977a)

Figure 3.4-6

COBALT-60 RADIOLYSIS GAS RELEASE FROM PIONEER 221 ASPHALT AT 250°C, DOSE RATE = 4.78×10^6 R/HR

Table 3.4-11 Properties of Waste Forms Solidified With UF

Property	Description
Leachability	The rate of radionuclide release depends on the chemical content of waste form. Increasing this waste content will increase leachability. For strontium-85 and cesium-137 the leachability properties of UF are poor.
Thermal	Will burn if exposed to a flame but is also self-extinguishing. Will reduce to ash if exposure to flame is prolonged. The toxicity of the gases released is low. Thermal conductivity is low (7.0-10.0 x 10^{-4} Cal/sec-cm-^{o}c for UF molding compounds).
Mechanical	A product with a one-to-one UF to liquid waste ratio exhibits a compression strength of about 50 kg/cm^2. Strength is significantly reduced when the proportion of UF is decreased.
Product stability	Stability is good in a sealed system, but if exposed to air, it loses water by evaporation leading to product degradation, decreased mechanical strength and increased leachability. This waste form may be affected by gas generation as a result of waste decomposition. Freeze-thaw cycling can cause degradation.
Radiation resistance	The threshold dose for mild to moderate radiation damage is approximately 3 x 10^6 rads. For moderate to severe damage it is 2 x 10^7 rads. Hydrogen gas is evolved during radiolysis. The radiolysis properties are greatly dependent on the waste type and content.
Free water	UF is polymerized by acid catalyst (pH 2). The monomer polymerizes by a condensation reaction which releases water, some of which is present as free water in the waste package. The volume of free water released is dependent on the ratio of UF to waste. With proper mixture selection, the free water content can be kept as low as 1%.
Interaction with container	The free water released in UF solidification contains acid and may also contain free formaldehyde which can lead to severe container corrosion problems.

Table 3.4-11 Properties of Waste Forms Solidified With UF (Cont'd)

Property	Description
Resistance to chemical attack	UF is resistant to chemical attack by oils, solvents, and greases. It is decomposed by strong acids and alkalis and is attacked by weak acids and alkalis.
Resistance to biodegradation	Although very little is known, UF is believed to be susceptible to biodegradation. The rate depends on the biodegradable material contained in the waste.

3.4.3.3 Volume Effect of Solidification With UF

As mentioned in Section 3.4.3.1, solidification with UF is not a chemical reaction. The UF-solidification process involves the physical encapsulation of the waste within the polymer structure. Under these conditions the final product volume is the sum of the volume of waste and the volume of UF. The proportions by which the waste and UF are mixed is based on their respective weights. In other words 1 pound of waste is mixed with 1 pound of UF. These proportions are different for different types of waste. Volume increase factors for various types of wastes are given in Table 3.4-12. These figures are based on data provided by Protective Packaging Inc.

3.4.4 Bitumen

3.4.4.1 Process and Material Description

Solidificaton systems using bitumen have been sold to two United States utilities for use in plants now under construction. The design of these systems is based on similar systems that have been operating in Europe for several years. The system uses bitumen, or asphalt, as a binding agent to encapsulate the waste. Systems marketed by Werner and Pfleiderer use Steep Roofing Asphalt, ASTM D-312-71 Type 3. The composition of this material is 39% by weight carbon disulfide with the remainder being various hydrocarbons. In the bituminization process molten bitumen is mixed with the waste and processed through either a batch-type roll dryer, a thin film evaporator, or an extruder. Mixing the waste with the hot bitumen and the external heating (usually steam), applied to the mixing device drives off all moisture. This in turn results in a partial volume reduction. As the hot mixture cools, it solidifies. Since no chemical reaction is involved the process is reversible, like those using thermoplastic polymers. Normally, a rotary turntable that holds several waste containers is used to maintain throughput.

3.4.4.2 Waste Characteristics

The major advantages of the bituminization process is that both volume reduction and solidification take place in one process step. Also, bitumen has certain properties that are advantageous in the immobilization of low- and intermediate-level wastes: it is chemically inert; it has good coating properties; and it is somewhat plastic.

One of the major disadvantages of bitumen is its potential fire hazard. The solvent normally used for cleaning in the process is also subject to fire hazards. Fires have occurred in bituminization facilities but they were readily controlled.

The potential combustion problem is minimized by the use of bitumen grades having high flash point (approximately 600°F). The use of fire-protection systems also improves the safety of the

Table 3.4-12 Volume Increase Factors for Waste Solidified in UF

Waste Type	Volume Increase Factor[1]	Packaging Efficiency[2]
Spent resin (33 wt% water)[3]	1.47	.68
25 wt% $NaSO_4$	1.45	.69
12.5 wt% boric acid	1.45	.69
Filter sludge (50 wt% water)	1.45	.69

1. Volume increase factor = $\frac{\text{Waste volume + binder volume}}{\text{Waste volume}}$

2. Packaging efficiency = $\frac{\text{Waste volume}}{\text{Waste volume + binder volume}}$

3. It is recommended that resins be mixed with evaporator bottoms prior to solidification. When resin alone is mixed with UF the resultant viscosity is too high to mix. Adding evaporator bottoms lowers the viscosity which is easier to mix and easier to make into a homogeneous mixture. The recommended ratio is approximately 2.7 parts resin to 1 part evaporator bottoms to 1 part UF (including catalyst).

bituminization process but increases the capital cost. The extent of fire-protection system requirements will depend on the bitumen process layout and the effect of a potential fire on the plant's safety-related systems. The container storage area must be adequately protected because of the potential spread of fire to the adjacent containers.

Because of the handling restrictions, bitumen must be heated and kept molten while in storage. Storage tank vents and the bituminization area must be adequately vented to remove bitumen vapors. The bitumen product shrinks up to 30% on cooling. To achieve maximum packaging efficiency, the container must be topped off and cooled as many as five times. The properties of bitumen waste forms have been investigated in several studies (Colombo, 1976a, b; 1977a, b, c; 1978). Tables 3.4-13 and 3.4-14, and Figures 3.4-6 and 3.4-7 summarize some of these findings.

Table 3.4-13 Composition of Bitumen

Constituent	Wt%
Carbon	83
Hydrogen	11
Sulfur	6
Nitrogen	1

3.4.4.3 Volume Effect of Solidification With Bitumen

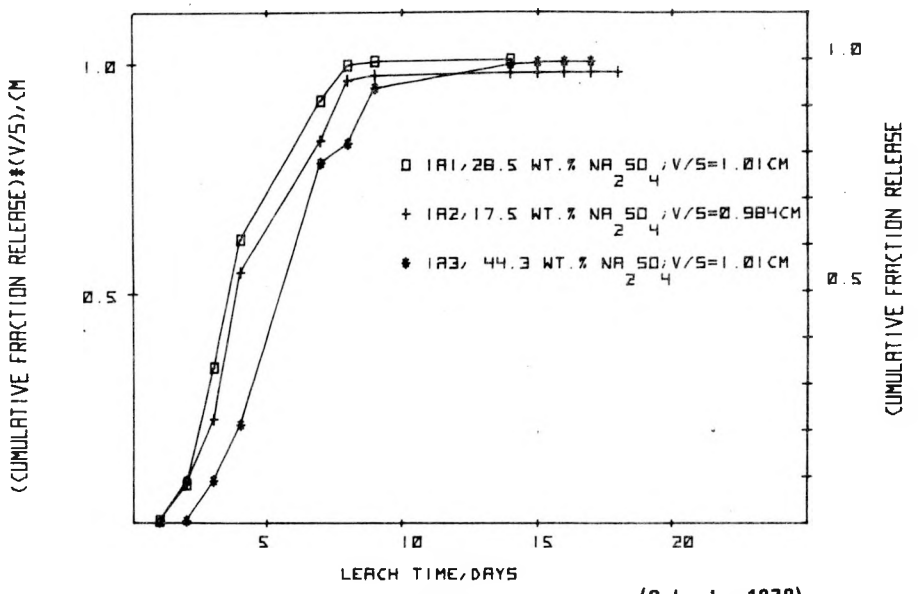
Solidification of waste with bitumen is a physical process in which the waste particles are coated with bitumen and held in place when the bitumen solidifies. The final waste volume is equal to the initial waste volume plus the volume of bitumen added minus the volume of water lost through evaporation. If the volume of water driven off is equal to the volume of bitumen added, the net volume increase factor is one. The volume increase factors for various waste types are given in Table 3.4-15. These figures are based on data provided by Werner & Pfleiderer.

Table 3.4-14 Properties of Waste Forms Solidified With Bitumen

Property	Description
Leachability	The rate of radionuclide release depends on the concentration and the size of the salt particles confined within the bitumen matrix. Recent studies indicate that bitumen waste forms containing sodium sulfate have extremely poor leachability characteristics. Typical Na_2SO_4-bitumen waste samples exhibited a tendency to swell, crack, and break up during leach testing. Other studies indicate that leach rates for alkali metal and alkaline earth elements can be initially as high as 10^{-2} g/cm^2/day, and as low as 3 x 10^{-7} g/cm^2/ day. The leachability of the waste depends on the solubility of the salts.
Thermal	Bitumen's flash point is about 280°C. The presence of oxidizing compounds can cause vigorous burning of bitumen. The pressures and temperatures generated under conceivable storage and transport conditions could potentially cause explosion of oxidizing compounds. Because of the low melting temperature of bitumen (60¼C), phase separation may occur in bitumen waste forms due to temperatures encountered during normal transportation and storage.
Mechanical	Strength depends on the type of bitumen used, the salts that are incorporated, and the temperature. Basically, bitumen has low mechanical strength.
Product Stability	Bitumen is a stable material. However, certain salts can cause rapid degradation of bitumen waste forms. Certain bitumen waste forms exposed to water have shown volume changes as a result of swelling.
Radiation Resistance	Radiolytic gas evolution can occur at as low as 10^6 rads, integrated dose. In addition to gas evolution, the irradiation of bitumen can result in chemical reaction between bitumen and incorporated solids with radiation-produced radicals, and oxidation of the bitumen. The gases produced are potentially explosive.
Free Water	Free water has not been a problem because bitumen waste forms reject free water. With adequate processing, free water in the package can be avoided.

Table 3.4-14 Properties of Waste Forms Solidified With Bitumen (Cont'd)

Property	Description
Interaction with Container	Bitumen does not interact with container.
Resistance to Chemical Attacks	It efficiently resists the action of most acids, alkalis, and salts.
Resistance to Biodegradation	Insignificant bacteriological attack on bitumen waste forms has been noted.



(Colombo, 1978)

Figure 3.4-7

CUMULATIVE SODIUM SULFATE FRACTION RELEASE x (V/S), STATIC LEACHING OF SERIES 1A WASTE FORMS IN DISTILLED WATER

NOTE: Leach rates are reported in this figure as [(cumulative fraction cesium release) x (specimen volume to exposed surface ratio)] or $(\Sigma a_n/A_o)(V/S)$ versus Σt_n where,

a_n = amount of species removed during leaching period n

Σa_n = cumulative amount of the species of interest during all leaching periods

A_o = amount of the species of interest initially present in the speciment

V = volume of the specimen, cm^3

S = exposed geometric surface area of the specimen, cm^2

t_n = leachant renewal period, days

Σt = cumulative leach time, days

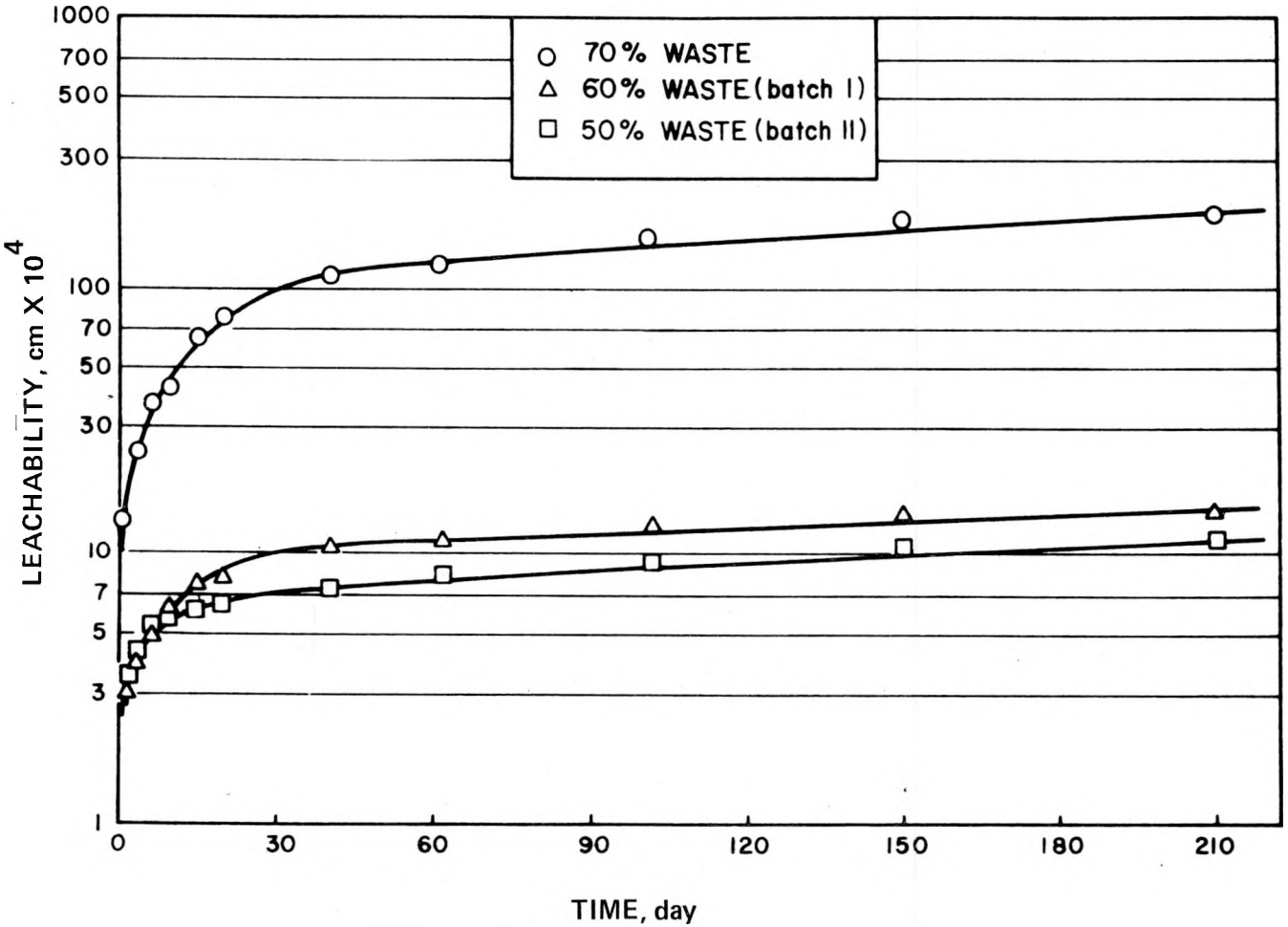


Figure 3.4-8
Leachability Curves for Sodium from 24% Sodium Sulfate Encapsulated in Water-Extended Polyester. (1)

1. Taken from Subramanian, 1977.

Table 3.4-15 Volume Increase Factors for Waste Solidified in Bitumen

Waste Type	Volume Increase Factor[1]	Packaging Efficiency[2]
Spent Resin		
(50 wt% solids)	0.97	1.03
(33 wt% solids)	0.64	1.56
25 wt% $NaSO_4$	0.47	2.1
50 wt% $NaSO_4$	1.13	.88
12.5 wt% boric acid	0.21	4.7
Filter sludge		
(50 wt% solids)	0.97	1.03
Dry salt	2.4	.42
Incinerator ash[3]	2.3	.43

1. Volume increase factor =

$$\frac{\text{Waste volume - water volume + bitumen volume}}{\text{Waste volume}}$$

2. Packaging efficiency =

$$\frac{\text{Waste volume}}{\text{Waste volume - water volume + bitumen volume}}$$

3. For 1,000 ft^3 of uncompacted combustible trash burned in an incinerator with a volume reduction factor of 80 a bituminization system will generate 29 ft^3 of solidified waste.

3.4.5 Polyester Resin

3.4.5.1 Process and Material Description

The process of solidifying low-level radwaste using water-extendible polyester (WEP) resins is being developed at the Washington State University. The process involves the use of unsaturated polyesters with polymer chain segments similar to the following:

$$\left[-O-CO-CH=CH-CO-O-R\right]_n O$$

where

R = any of the alkane groups such as methane, butane, or others.

The unsaturated polyesters are dissolved in polymerizable monomer, usually styrene, $C_6H_5CH=CH_2$. When a catalyst is added to the solution, polymerization of the styrene monomer is initiated. The catalyst consists of a promoter and an initiator. The promoter, which already may be present in the polyester, is a reducing agent such as cobalt naphthenate, cobalt octoate, or dimethylaniline. The initiator is any of a variety of peroxides such as methyl ethyl ketone or hydrogen peroxide.

As with systems using urea-formaldehyde, the waste is bound in a closed-cell structure formed during the polymerization. In a proposed solidification system the polyester and waste would be mixed until the waste is thoroughly emulsified in the polyester. The initiating peroxide is then added to initiate curing. The polyester polymerization reaction leading to curing is an exothermic reaction. However, the encapsulated aqueous waste serves as an efficient heat sink and heat removal is not usually necessary.

The factors that must be considered for compatibility of polyester and a given waste type are

- The ease of dispersion of waste into polyester resin
- The stability of emulsion thus formed
- The curing reaction leading to solidification of emulsion
- The product quality of the resultant solid structure.

Most types of reactor waste are believed to be compatible with polyester except that certain acidic wastes appear to be unfavorable for the formation of an emulsion with water-extensible polyester. Solidification of the acidic wastes requires proper control of pH and choice of a suitable catalyst. In pilot-plant test runs successful dispersion and solidification of several simulated power-plant wastes have been readily achieved. The simulated wastes used include a 24 wt% aqueous solution of sodium sulfate; a 12 wt% aqueous solution of boric acid adjusted to pH 8.0; and a 20 wt% aqueous solution of boric acid adjusted to pH 8.0.

3.4.5.2 Waste Characteristics

As with bitumen, the polyester resin and its catalyst system are flammable. Furthermore, the process of dispersing waste into polyester resin produces fumes that must be adequately controlled.

When the polyester waste forms were exposed to the open environment, evaporation of water from aqueous material in the closed cell caused a 1% per month weight loss. A slight amount of shrinkage has also been observed.

A prototype polyester waste processing module was developed at Washington State University by Subramanian and Mahalingam. Several waste samples have been prepared and tested to determine the product quality. Figure 3.4-8 and Table 3.4-16 present some of their findings.

3.4.5.3 Volume Increase Factors for Solidification in Polyester Resin

Recommended mixing ratios supplied by Dr. Subramanian of Washington State University are based on waste volumes. These ratios generally range from one part waste to one part polyester resin up to two and a quarter parts waste to one part polyester resin.

Table 3.4-16 Compressive Strength of 24% Sodium Sulfate Solution Encapsulated in Polyester Resin[1]

Proportion of 24% Sodium Sulfate Solution (%)	Gamma Radiation (Mrad)	Compressive Strength (N/mm^2)
50	-	20.9 -
60	-	15.4 ± 0.42
60	3.8	17.3 ± 0.41
60	7.9	17.4 ± 0.55
60	23.7	18.3 ± 0.38
60	134.0	20.2 ± 0.47
60	326.0	21.4 ± 0.37
60	466.0	22.1 ± 0.41

1. Taken from Subramanian, 1977.

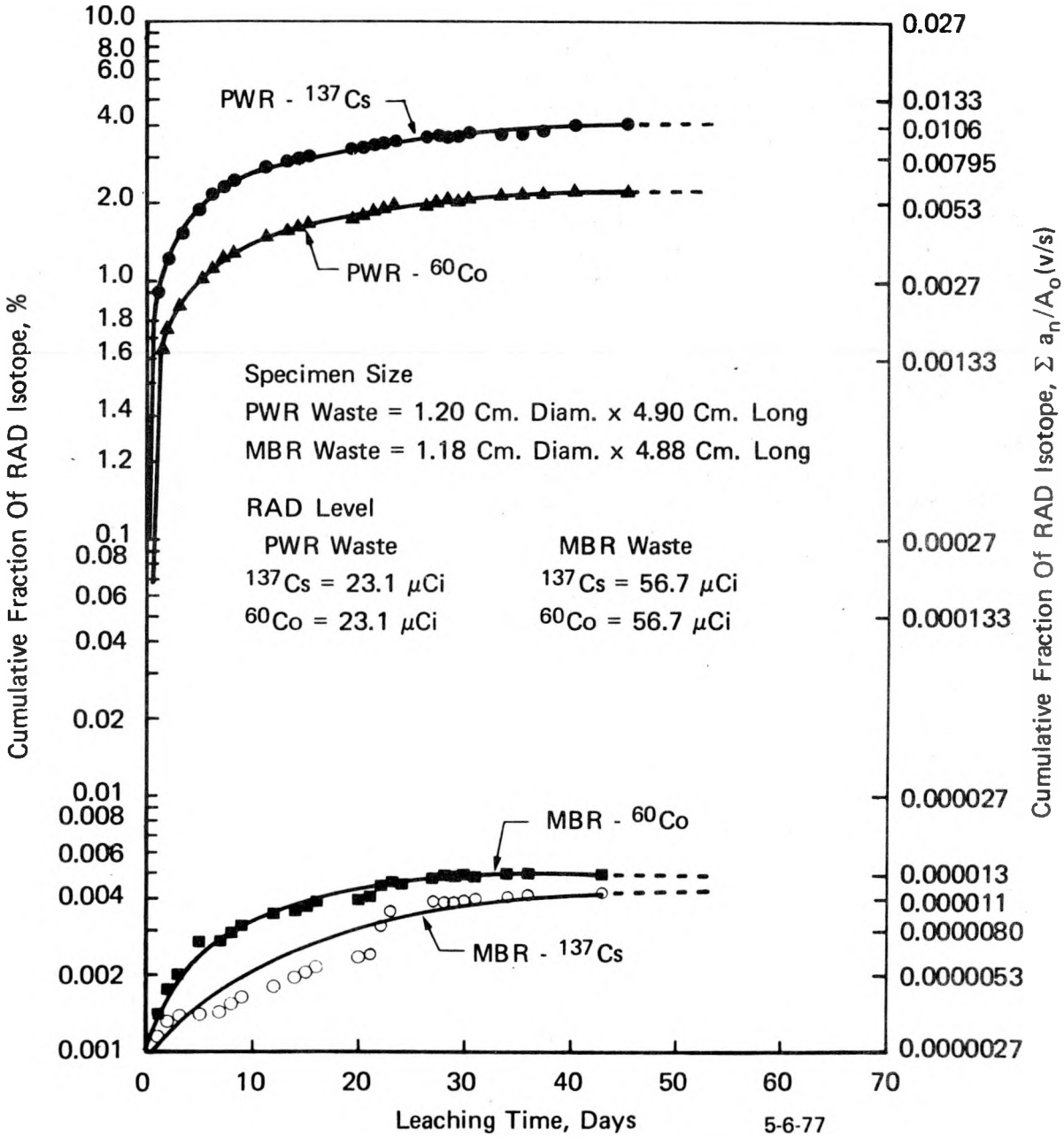


Figure 3.4-9

Leach Test Results for Simulated PWR Evaporator Bottoms and Mixed Bed Resins at 1.65/1.0 & 2.0/1.0 Waste/Binder Ratios Respectively. (1)

1. Taken from Filter, 1977.

The volume increase factors and packaging efficiencies are in Table 3.4-17.

3.4.6 Dow Binder

3.4.6.1 Process and Material Description

Dow Industrial Services (DIS), a division of Dow Chemical, is marketing a proprietary vinyl esterstyrene polymer system for the solidification of radioactive wastes. This process uses a combination of the binder (vinyl ester resin) with a small amount of a catalyst and a promoter. The process encapsulates the waste into a stable, solid matrix. The Dow process has the capability of encapsulating wastes with a pH range of 2.5 to 10.5. The resultant product is devoid of free liquid in the waste package, according to DIS.

3.4.6.2 Waste Characteristics

The properties of the Dow system simulated waste forms were investigated by the Dow Chemical Company (Filter, 1977). Preliminary tests on various simulated waste types indicated that the solidified waste forms were without free water and were stable after a 6 to 12 month observation.

Samples of simulated waste, solidified by the Dow system, were subjected to a test published by the DOT. The samples were placed in a muffle furnace preheated at 1,000°F where they remained for 10 minutes. The surface formed a porous char layer that protected the body of the sample from the heat. The samples did not melt, sublime, or ignite. This effect would occur even if the material were broken into pieces.

Compression tests showed the samples to have satisfactory mechanical properties. A summary of the Dow tests are presented in Table 3.4-18. Figure 3.4-9 shows the results of leach tests performed on simulated PWR evaporator bottoms and mixed bed resins.

3.4.6.3 Volume Increase Factors for Solidification With Dow Binder

The Dow process is relatively new. Most of the development work is testing. Only limited data on the optimum ratios of waste to binder are currently available. Based on this limited information the volume increase factors in Table 3.4-19 have been calculated.

3.4.7 Summary of the Volumetric Effects of Waste Solidification

Tables 3.4-7, 3.4-12, 3.4-15, 3.4-17, and 3.4-19 give the volume increase factors and packaging efficiencies for the five solidification agents identified in Section 3.4.1. Figures for bitumen also include the volume reduction effects resulting from the evaporation of the water contained in the waste as it passes through the extruder evaporator. Data on the Dow Binder are too sketchy to use in comparison with the other materials. Table 3.4-20 is a

Table 3.4-17 Volume Increase Factors for Waste Solidified in Polyester Resin

Waste Type	Volume Increase Factor[1]	Packaging Efficiency[2]
Spent Resin (50 wt% solids)	1.67	.60
24 wt% $NaSO_4$	1.54	.65
50 wt% $NaSO_4$	1.67	.60
12 wt% boric acid	1.67	.60
Filter sludge (50 wt% solids)	1.67	.60
Dry salt (calcined waste)	2.0	.50
Incinerator ash[3]	2.0	.50

1. Volume increase factor = $\frac{\text{Waste volume + binder volume}}{\text{Waste volume}}$

2. Packaging efficiency = $\frac{\text{Waste volume}}{\text{Waste volume + binder volume}}$

3. For 1,000 ft^3 of uncompacted combustible trash burned in an incinerator with a volume reduction factor of 80, a polyester resin system will generate 25 ft^3 of solidified waste.

Table 3.4-18 Physical Tests on Wastes Solidified With Dow Binder Non-Radioactive Simulated Waste - Ratio Waste/Binder[1]

Test Method	BWR Evaporator Bottoms[2] 1.75/1.0	PWR Evaporator Bottoms[3] 1.80/1.0	Mixed Bed Resin[4] 2.25/1.0	Filter Sludge[5] 1.50/1.0	DS[6] 1.50/1.0
Free Liquid	None	None	None	None	None
Heat Exposure 10 min. @ 1000°F	Darkened, surface checked, 27.1% loss	Darkened, surface checked, 27.8% loss	Darkened, surface cracked, 28.8% loss	Darkened, no checks or cracks, 27.3% loss	Surface blackened, some cracks, 27.5% loss
Percussion Test	No damage - rod rebounded	No damage - rod rebounded	No damage - rod rebounded	---	No damage - rod rebounded
Compressive Strength, psi	3,952	2,790	1,761	4,210	3,312

1. Taken from Filter, 1977.
2. 24.5 pounds sodium sulfate, 4 pounds trisodium phosphate, 1 pound motor oil, pH adjusted to 10.6 with sodium hydroxide in 50 gallons of water.
3. 20 pounds of boric acid, 2.5 pounds sulfuric acid, 1 pound trisodium phosphate, 1 pound calcium hydroxide, pH adjusted to 2.8 with sulfuric acid in 50 gallons of water.
4. Mixed-bed ion exchange resin slurry with 10% free water.
5. Diatomaceous earth slurry with 10% free water.
6. Waste type DS is not defined by reference document.

Table 3.4-19 Volume Increase Factors for Solidification With Dow Binder

Waste Type	Volume Increase Factors[1]	Packaging Efficiency[2]
Spent resin (33 wt% water)	1.50	.67
7 wt% $NaSO_4$	1.80-1.57	.55-.64
6 wt% $NaSO_4$	1.80-1.57	.55-.64

1. Volume increase factor = $\frac{\text{Waste volume + binder volume}}{\text{Waste volume}}$

2. Packaging efficiency = $\frac{\text{Waste volume}}{\text{Waste volume + binder volume}}$

Table 3.4-20 Comparative Effects of Solidification on 1,000 Gallons of Radioactive Waste

Waste Type	Cement		Urea Formaldehyde		Polyester Resin		Bituminization	
	Number of 55-Gal Drums	Gallons[1] per Drum	Number of 55-Gal Drums	Gallons[1] per Drum	Number of 55-Gal Drums	Gallons[1] per Drum	Number of 55-Gal Drums	Gallons[1,2] per Drum
Spent resin								
33 wt% solids	27	37.8	30[3]	33.7	30	33.2	20	50
50 wt% solids	25	40.0	-[4]	-	-	-	30	33.3
Evaporator bottoms								
Sodium sulfate 25 wt% solids	30	32.8	29	34	31	32.2	10	100
Boric acid 12.5 wt% solids	49	20.2	29	34	34	29.7	4.3	232
Crystallizer bottoms								
Sodium sulfate 50 wt% solids	27	37.8	-	-	34	29.7	23	43
Boric acid 50 wt% solids	-	-	-	-	-	-	23	43
Filter sludge 50 wt% solids	25	40.0	29	34	34	29.7	20	50
Dry salt 100 wt% solids 100 ft^3	26	28.4	-	-	30	24.8	35.8	27.9
Incinerator ash 100 wt% solids 1,000 ft^3 uncompacted combustible trash	3.3	3.8[5]	-	-	3.8	3.3[5]	4.4	2.8[5]

1. Drums are 90% full.
2. Equivalent gallons per drum prior to evaporation of water.
3. Each drum also contains 12.3 gallons of evaporator bottoms.
4. Not available.
5. Ft^3 of ash per drum.

comparison of the various solidification agents showing the gallons of waste that can be solidified in a 55-gallon drum assuming the drum is filled to 90% of capacity. The table also shows the number of drums needed to solidify 1,000 gallons of waste. The number of drums needed to solidify dry salts is based on 100 ft^3 of salts. Dry salts refers to the product of a fluidized-bed dryer, or calciner, discussed in Section 3.5. The number of drums needed to solidify incinerator ash is based on 1,000 ft^3 of uncompacted combustible trash burned in an incinerator. Section 3.5 provides a discussion of two commercially available incinerators.

3.5 Near-Term Volume-Reduction Processes

3.5.1 Introduction

The several volume-reduction processes now available in the United States can be broken down into three main categories: incinerators, fluidized-bed dryers, and bituminization. Bituminization is a process in which the binder is added prior to the volume reduction process and is discussed in a limited context in the previous section. This section addresses the volume-reduction phase of each of these processes and the subsequent effect of solidification on the waste volumes. Trash compactors and evaporative crystallizers are also discussed.

3.5.2 Incinerators

As discussed in Section 4.4 of this report, incinerators are currently used at several fuel-fabrication facilities. These incinerators concentrate the uranium contamination in combustible trash so that recovery of the uranium is economical. In addition to fuel-fabrication facilities incinerators are used at a number of government and privately owned laboratories. In all, there are 17 radwaste incinerators in operation in the United States, none of them at a nuclear power plant. Radwaste incinerators are used in the Soviet Union, United Kingdom, Canada, France, Belgium, Portugal, the Federal Republic of Germany, India, and Japan. The materials that would be incinerated at an LWR are the combustible items listed in Tables 4.2-16, 4.2-39, and 4.2-40. The two commercially available incinerator systems are manufactured by Trecan Limited of Canada and Wellman Incandescent Ltd., a British firm.

3.5.2.1 Trecan Batch-Type Incinerator (Choi, 1977)

The Trecan incinerator is a batch-type, controlled-air incinerator. Waste is loaded into the unit through a 60 ft^3 loading chamber at the top of the unit. Normal capacity of the unit is 300 ft^3, or 5 loads. In controlled-air incineration the primary combustion chamber is starved of air to obtain a partially oxidized effluent. This effluent consists of carbon monoxide, carbon dioxide, hydrogen, nitrogen, and water vapor. Starving the combustion chamber of air is accomplished by limiting the air flow to only 30% of that needed for complete oxidation. The waste is then pyrolyzed so that the ash remains in the primary chamber. Complete oxidation does occur by the end of the cycle.

The pyrolysis reaction proceeds gradually throughout the waste bed. First, the moisture and volatile matter (partially oxidized) are driven off, leaving the fixed carbon in the waste for complete oxidization in the later stages of the burning cycle.

The greatest volume change occurs in the initial stages, when the volatiles are driven off and partially oxidized. When incinerating the general waste, a 90% volume reduction will occur within

2 hours. The balance of the burn cycle is used to oxidize the fixed carbon content of the waste, leaving radioactive ash.

The partially oxidized effluent with small amounts of particulates is completely burned in a refractory-lined afterburner where fuel and additional air (in excess of 100%) are introduced to complete the oxidation. The afterburner section is designed to ignite and burn the partially oxidized effluent from the primary chamber at a temperature of 1,800°F for 0.5 second.

A Trecan batch-type controlled-air incinerator has been installed at the Bruce Nuclear Power Development Site in Canada. Waste volume reduction ratios as high as 80 to 1 have been reported for this incinerator. A schematic of a Trecan incinerator system is given in Figure 3.5-1.

3.5.2.2 Wellman Incandescent Ltd. Incinerator (Yapp, 1977)

A Wellman Incandescent Ltd. incinerator system is presently in operation in the United Kingdom with a second unit scheduled for installation in the near future.

A typical system uses propane as fuel but other gases or oil can be used. Conveyors are used to transport loaded waste bins to the operating station and return empty bins for reuse. All waste is passed through a metal detector before being charged into the incinerator which employs sealed ash extraction. While ash is being removed from the incinerator a vacuum system is used to avoid spillage of radioactive dust.

The system is designed to handle a wide variety of wastes such as contaminated paper, rubber, plastics (including PVC), used lubricating oil, and other waste oils. The system can handle large amounts of PVC since the exhaust gases leaving the incinerator are thoroughly cleaned of acids and particulates. The effluent-treatment system employs a wet gas scrubber in which the liquor is maintained in the pH range 6 to 8 in order to minimize corrosion and allow a recirculating system to be used. Residual particulates in the scrubbed exhaust gases are removed in a high-efficiency filter train with a reported efficiency of 99.9%.

Exhaust gases from the incinerator are cooled in a heat exchanger, which operates under partial vacuum to preclude the leakage of exhaust gases to the atmosphere except through the high-efficiency filter system. More recent designs use a direct water quench in place of the intermediate heat exchanger. This eliminates possible corrosion problems.

Volume reduction factors for this unit are expected to be the same as those for the Trecan unit, approximately 80 to 1 for uncompacted combustible trash.

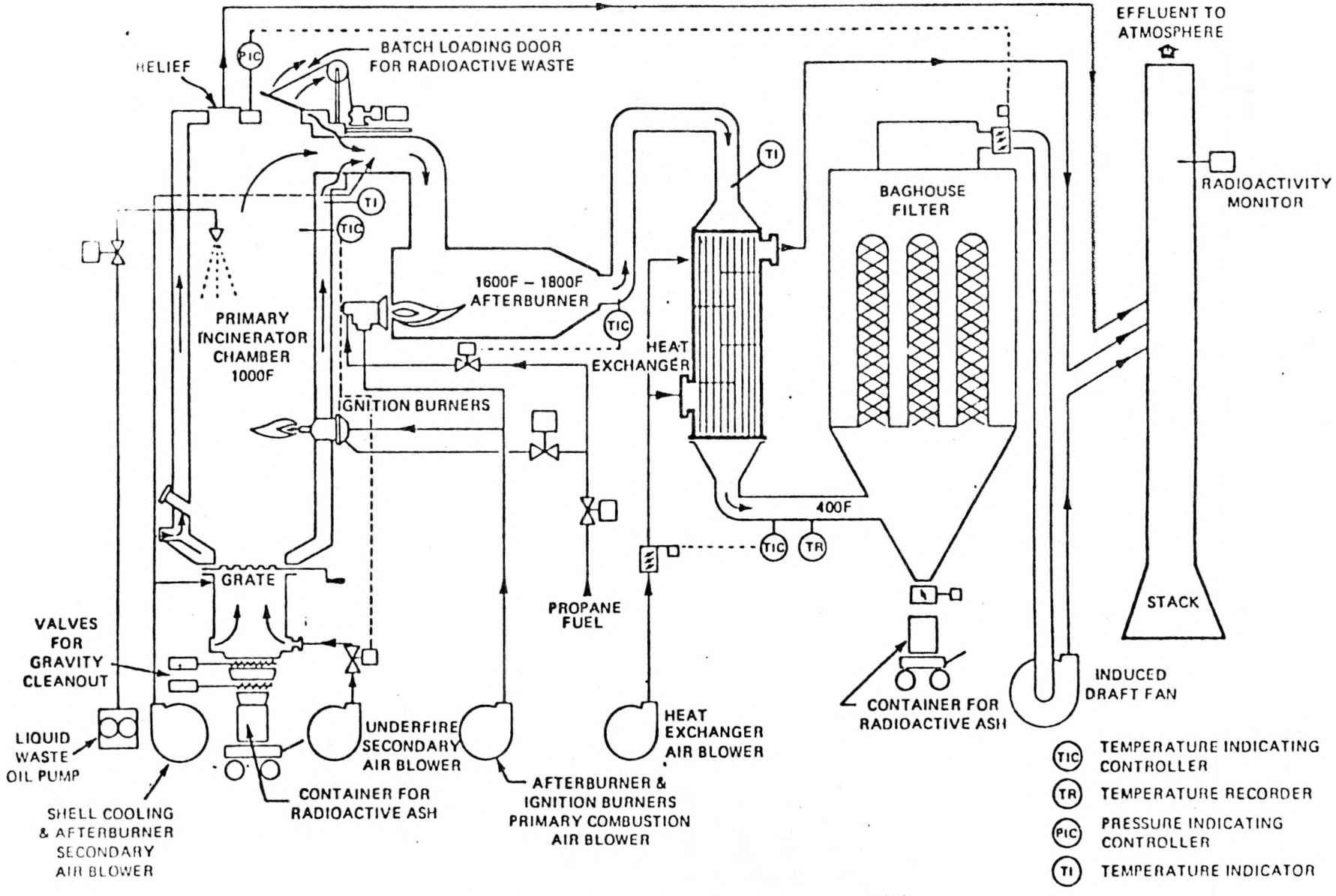


Figure 3.5-1 Schematic of Trecan Incinerator(1)

1. Taken from Choi, 1977.

3.5.3 Fluidized-Bed Dryers

Waste calcination is the process of drying liquid waste to remove 100% of the water, thus producing dry salts. In a fluidized-bed calcination process the waste is sprayed into a continuously agitated bed. As the bed depth increases, a portion of the material is drawn off. The bed is held in suspension by a stream of hot air. Initial startup of the fluidized-bed dryer is accomplished by using sand as bed material. As more waste is processed, the sand and the waste in the bed are eventually removed from the unit until the bed is only dry salts. At the end of a batch cycle the airstream is stopped and the bed literally drops to the bottom of the unit. When more waste is available for processing, the previously processed waste (dry salt) is used as the initial bed.

The first fluidized-bed dryer facility in the United States that processed radioactive waste was at the Idaho Chemical Waste Facility Plant in 1963. A conceptual design for the commercial use of fluidized-bed dryers has been available since 1972.

Commonwealth Edison Co. and Carolina Power and Light Co. will be the first utilities to use fluidized-bed dryers commercially. Commonwealth Edison Co. will use fluidized-bed dryers at Byron 1 & 2 and Braidwood 1 & 2. Carolina Power and Light Co. will use fluidized-bed dryers at Harris Units 1, 2, 3, and 4. This system, which includes an incinerator, is being provided by Aerojet Energy Conversion Company. The incinerator is a standard component in the Aerojet system. The only other commercially available system is manufactured by the Newport News Industrial Corporation.

3.5.3.1 Aerojet Energy Conversion Company

3.5.3.1.1 System Description

The Aerojet system combines a fluidized-bed dryer for processing evaporator concentrates and an optional fluidized-bed incinerator used to handle wet solid wastes and dry active wastes.

The incinerator and the fluidized-bed dryer both use the same offgas cleanup system.

The basic system consists of a fluidized-bed dryer, fluidized-bed incinerator, and offgas cleanup system. Figure 3.5-2 is a simplified process flow schematic.

Evaporator bottoms are preconcentrated in a high-energy venturi scrubber by heat recovered from the exhaust gases from the dryer. The preconcentrated feed, containing 25 to 28 wt% dissolved solids, is pumped from the scrubbed sump to the dryer where it is atomized with air and injected into the dryer. The water is flash evaporated as the liquid droplets contact the particles in the bed and the dissolved solids are deposited on the hot bed particles. The

VOLUME REDUCTION FLOW DIAGRAM

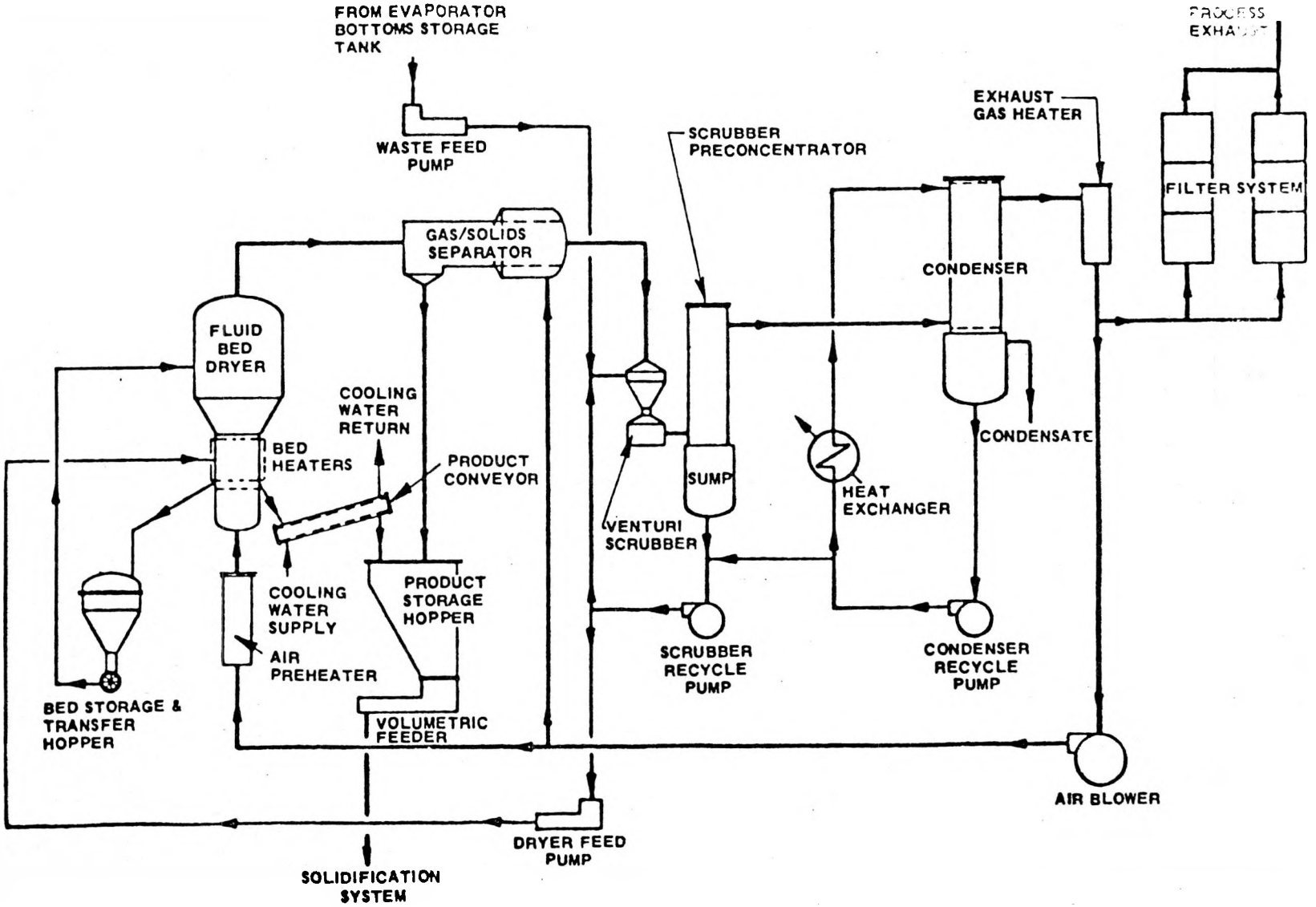


Figure 3.5-2 Schematic of Aerojet Fluidized-Bed Dryer[1]

1. Courtesy of Aerojet Energy Conversion Company

fluidizing air is electrically heated and passed upward through the dryer vessel, fluidizing the bed. The air is supplied by an air blower operating in a semiclosed-loop mode. The dry particles are discharged from the fluidized bed by a product conveyor and transported to the product storage hopper.

Dry active wastes consisting of paper, plastic bags, plastic or rubber gloves, boots, laboratory clothing, and rags are shredded and metered into a pneumatic transport feed system leading to the incinerator. A detector is provided to keep large metal objects out of the shredder. Wet solid wastes are pump fed into the incinerator in slurry form simultaneously with the dry wastes.

The offgas from the incinerator is quenched with condensate from the dryer system. The overhead gas stream from the dryer and the exhaust from the incinerator are both ducted through a gas-solids separator.

3.5.3.1.2 Volume-Reduction Factors

Based on data from the Aerojet Energy Conversion Company, expected volume-reduction factors for a combined fluidized-bed calciner and incinerator are as given in Table 3.5-1.

Aerojet does not recommend the drying of resin or filter sludge because there could be problems involving offgas filter handling if too much of the radioactivity in the resin or filter media becomes dislodged during the drying process. This dislodged radioactivity, most of which would be particulate, would subsequently be trapped on the offgas filters, causing additional replacement problems. Another potential problem is that resins could form clumps of liquid material in the bed, thus causing the bed to collapse. Aerojet takes the position that the volume reduction achieved for resins and filter sludge is not worth the problems that could result.

3.5.3.2 Newport News Industrial Corporation System Description

The Newport News system, designated the RWR-1, is designed to process and remove all moisture from the concentrated liquid waste, to process and incinerate spent resin and filter sludge slurries, and to incinerate other combustible bulk solids. It can reduce both liquid and solid radwaste to an anhydrous granular solid. It resembles the Aerojet system in that it uses both calcination and incineration for volume reduction. They differ in that the RWR-1 system uses a single process vessel for both operations. Newport News has installed a pilot system at Niagara Mohawk's Nine Mile Point Station.

The RWR-1 system is shown in Figure 3.5-3. The heart of the process is the process vessel and the dry cyclone, which accomplish the primary volume-reduction and collection functions.

Table 3.5-1 Volume-Reduction Factors for Aerojet Fluidized-Bed Dryer/Incinerator

Waste Type	Volume-Reduction Factor
Resin	NR[1]
Filter sludge 50 wt% solids	5
Evaporator bottoms 12.5 wt% boric acid 25 wt% $NaSO_4$	 9.3[2] 4.6[3]
Crystallizer bottoms 50 wt% boric acid 50 wt% $NaSO_4$	 2.9[2] 2.9[3]
Combustible trash uncompacted	 80

1. Not recommended.
2. Based on a volume-reduction factor of 11.6 for 10 wt% solution of boric acid.
3. Based on a volume-reduction factor of 5.8 for a 20 wt% solution of $NaSO_4$.

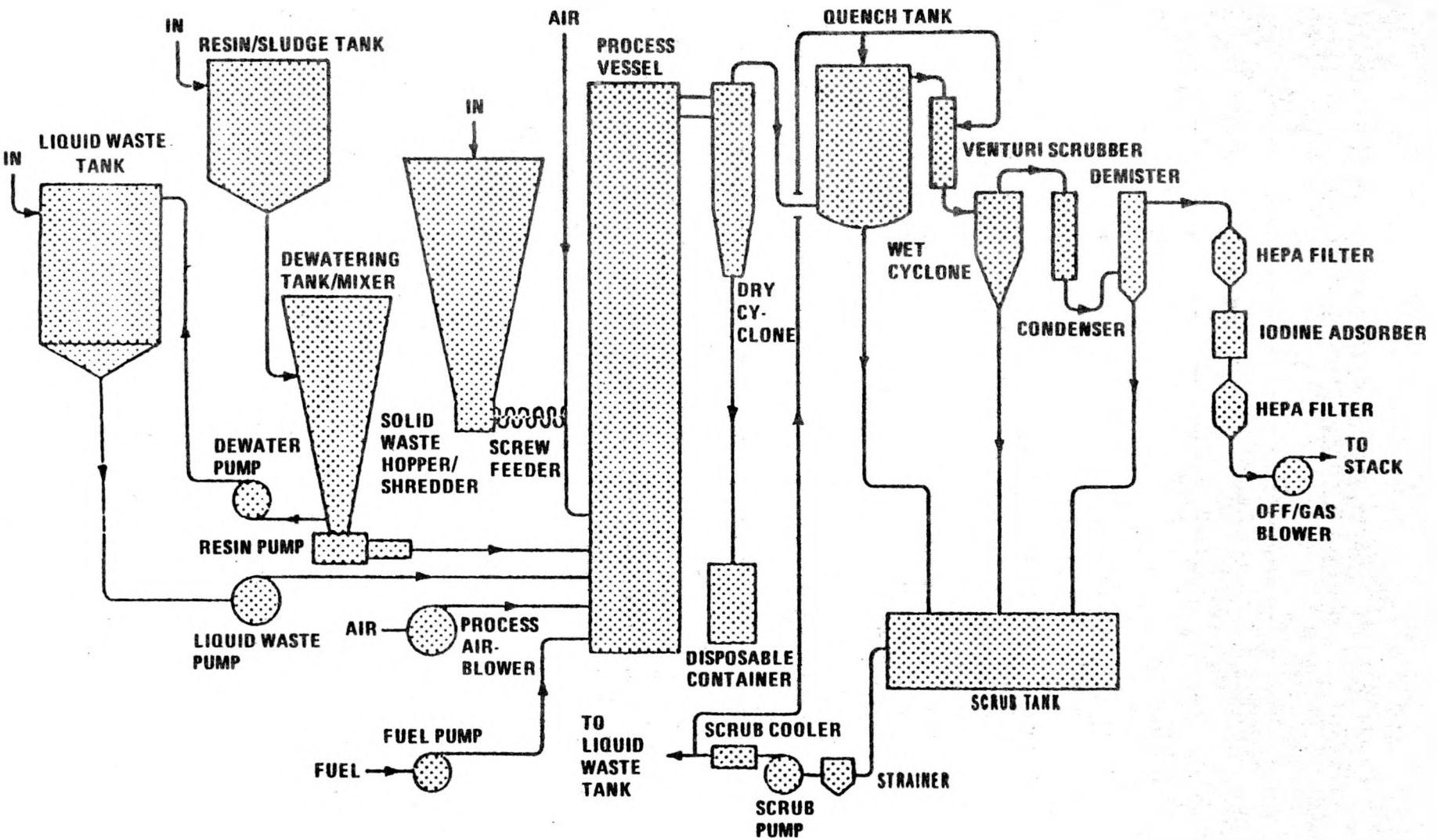


Figure 3.5-3 Flow Diagram of Newport News Calciner/Incinerator(1)

1. Taken from Feizollahi, 1978 (to be published).

The fluidized-bed dryer/incinerator has three different operating modes. For purposes of better process control, the various types of radwaste are separated into three feed systems--concentrated liquids such as sodium sulfate, boric acid, and decontamination solutions.

Two incineration modes are used. One mode is used for spent resins and/or filter sludges. The other mode is used for miscellaneous combustible solids.

Steady-state operating parameters are preprogrammed into the control system. These parameters are selected automatically depending on the waste to be processed. A process vessel temperature is selected to be either 400, 800, or 1,000°C based on the feed stream to be processed. Control interlocking prevents feeding more than one waste type at a time. The lower temperatures are used for concentrated liquid wastes to avoid melting the dried residue. Higher temperatures for the combustible wastes assure efficient combustion.

Most of the particulate matter is removed in the dry cyclone. The offgas is quenched, which removes more fine particles. The offgas then proceeds through a venturi scrubber, condenser, demister, iodine adsorber, and several HEPA filters before proceeding to the stack.

3.5.3.2.1 Volume-Reduction Factors

Based on data presented by Newport News Industrial Corp., volume reduction factors for the RWR-1 system are expected to be as listed in Table 3.5-2.

3.5.4 Bituminization

Bituminization of waste has already been discussed as a solidification method in Section 5.5.2. Because bituminization evaporates all of the free water, it is also a volume-reduction process. Several European companies manufacture bitumen systems but only one has had any commercial success in the United States. The Werner & Pfleiderer Corp. has sold bituminization systems to Consumers Power Company for Midland Units 1 and 2 and to Puget Sound Power & Light Company for the Skagit Nuclear Power Project Units 1 and 2.

3.5.4.1 Werner & Pfleiderer Corporation

The Werner & Pfleiderer Corporation process uses a twin-screw extruder in a multisection housing. Each section is heated separately with steam. Liquid bitumen and waste are fed into the first section of the unit and mixed. Heat from the steam heating causes the moisture in the waste to evaporate. This vapor is exhausted through the steam domes, condensed, filtered, and discharged to a condensate tank for recycle to the plant waste evaporator. By the time the waste completes the process it has been heated three to five times and all the moisture has been

Table 3.5-2 Volume-Reduction Factors for Newport News Fluidized-Bed Dryer/Incinerator

Waste Type	Volume-Reduction Factor
Spent resin 33 wt% solids	18
Filter sludge 50 wt% solids	5
Evaporator bottoms 12.5 wt% boric acid 25 wt% $NaSO_4$	 8 (1) 6.4 (2)
Crystallizer bottoms 50 wt% boric acid 50 wt%	 2 (1) 3.2 (2)
Combustible trash Uncompacted	80

1. Based on a range of 8 to 11 for 8 wt% boric acid.
2. Based on a volume reduction of 8 for 20 wt% $NaSO_4$.

removed. The mixture of dried waste and bitumen is extruded as a viscous stream into a shipping container. The shipping container is located in a shielded area on a rotating platform. This allows continuous processing of enough waste to fill several drums. The system need only be shut down long enough to rotate the platform.

Wastes that can be processed include evaporator bottoms, resins, filter sludge, and even calcined salts and incinerator ash. For calcined salts and incinerator ash no additional moisture would be removed; but the process, operating at a temperature sufficient to keep the bitumen liquid, could be used to immobilize these wastes. Figure 3.5-4 shows the basic system flow patterns. Table 3.5-3 lists approximate volume-reduction factors. The first column of numbers is the absolute volume reduction with no bitumen added. The next column shows the effective volume reduction (numbers less than 1.0), or volume increase (numbers greater than 1.0) with the bitumen added. The last column is the packaging efficiency of the final product with the bitumen added.

3.5.5 Trash Compactors

The trash compactor is the only system that is widely used to achieve volume reduction in nuclear facilities.

Trash compactors are manufactured by several companies, including Consolidated Baling Machine Co., Nuclear Packaging Co., RAM Corporation, and Stock Equipment Co.

The basic function of a trash compactor is to compress low-level trash, such as paper, rags, glass, non-reusable clothing, low-activity filters, and other dry wastes into 55-gallon drums. The typical compactor holds a single 55-gallon drum in an enclosed housing with a hydraulic ram operating downward. Nuclear Packaging Co. uses a mechanical ram. Full drums are removed, and empty drums are placed in the unit through the front doors. These doors must be closed for the ram to operate.

The trash compactor includes a ventilation system that operates during compaction. The fan exhausts through a HEPA filter to collect any dust or other contaminants released during compaction. The hydraulic systems have a force of approximately 30,000 pounds.

Compactors currently in use have effective compaction ratios of about two. In this study this factor of two will be used although newer machines with greater pressures are rated at compaction ratios up to four.

3.5.6 Evaporative Crystallizer

An evaporative crystallizer manufactured by Horton Process Development, Inc. (HPD) is a modification of their standard radwaste evaporator. These modifications, which control nucleation rate, crystal growth rate, heat balance, and material balance make it possible to concentrate solutions up to 50% total solids by weight.

Table 3.5-3 Volume-Reduction Factors for WPC Bituminization Process

Waste Type	Absolute Volume-Reduction Factor(1), no Bitumen	With Bitumen Added	
		Volume Reduction Volume Increase(2)	Packaging Efficiency(3)
Spent Resin			
33 wt% solids	.23	.64	1.56
50 wt% solids	.34	.97	1.03
Filter Sludge			
50 wt% solids	.34	.97	1.03
Evaporator Bottoms			
12.5 wt% Boric Acid	.08	.21	4.7
25 wt% $NaSO_4$	.17	.47	2.1
Crystallizer Bottoms			
50 wt% Boric Acid	(4)	(4)	(4)
50 wt% $NaSO_4$	.41	1.13	.88
Calcined Salts	NA(5)	2.4	.42
Incinerator Ash	NA	2.3	.43

1. Volume reduction factor = $\frac{\text{Waste volume + binder volume}}{\text{Waste volume}}$; less than one.
2. Volume increase factor = $\frac{\text{Waste volume + binder volume}}{\text{Waste volume}}$; more than one.
3. Packaging efficiency = $\frac{\text{Waste volume}}{\text{Waste volume + binder volume}}$
4. Not available.
5. Not applicable.

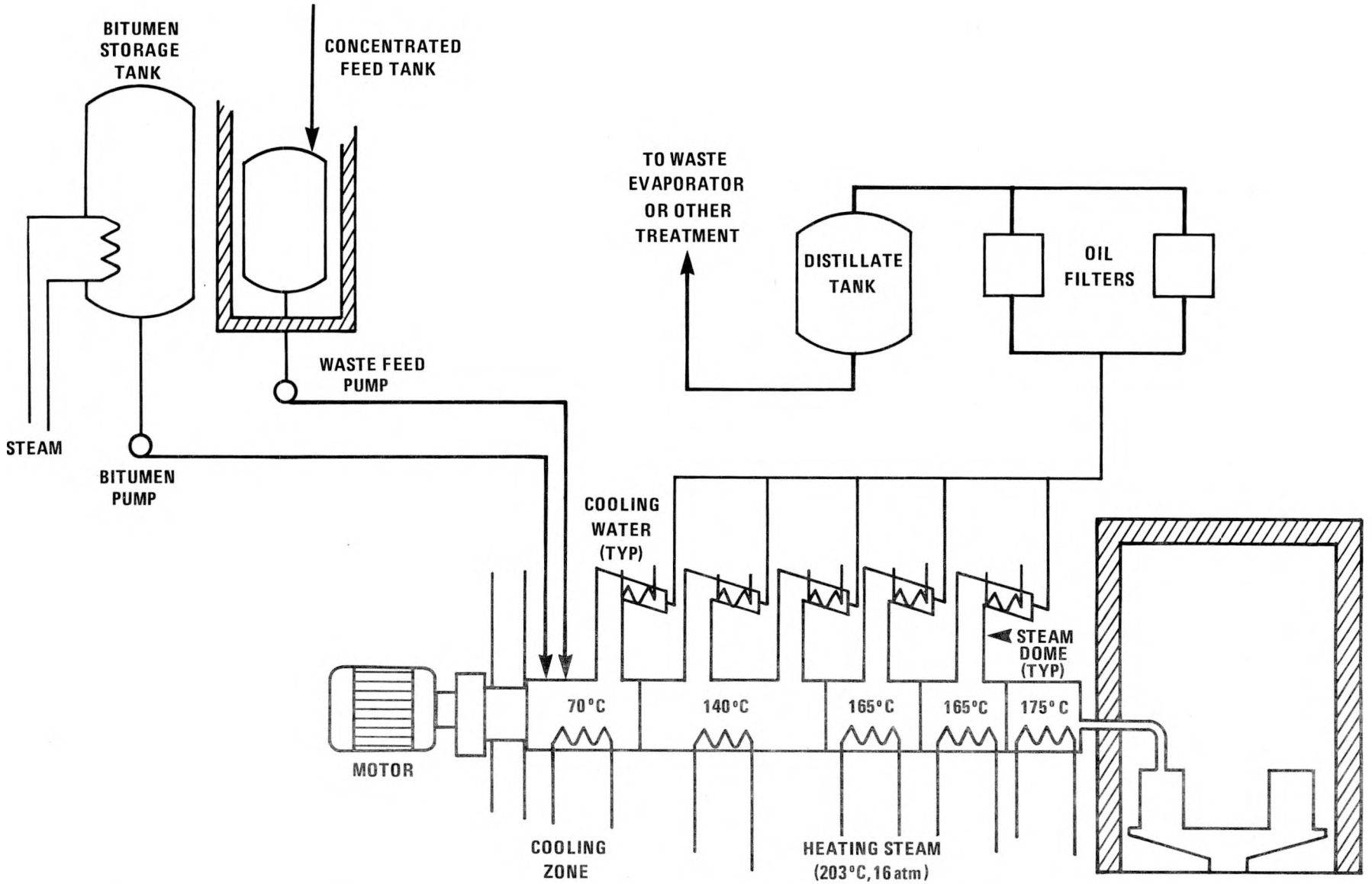


Figure 3.5-4 Flow Diagram of WPC Bituminization System

Crystallizers can be used in both PWRs and BWRs on solutions of sodium sulfate, ammonium sulfate and boric acid. The HPD crystallizer is a vertical-tube, forced-circulation type unit consisting of a vapor body, recirculation pipe, a large recirculation pump, and a heater. Figure 3.5-5 is a flow diagram of an HPD evaporative crystallizer.

Conventional evaporators are capable of concentrating boric acid solutions to approximately 12.5 wt% solids and sulfate compounds to 25 wt% solids. Crystallizers provide an additional volume-reduction factor of approximately 6 for boric acid, and approximately 2.4 for sulfates. This is achieved by concentrating these wastes to 50 wt% solids. Figure 3.5-6 is a diagram of an evaporator/crystallizer which can be compared to the diagrams of evaporators in Section 3.3.1.3.

3.6 Radwaste Shipping Containers

As shown by many of the tables in Sections 4.2.1 and 4.2.2 of this report, containers for the shipment of low-level and intermediate-level waste come in almost any size and shape. The most widely used container for compactible trash and process wastes is the standard DOT Spec 17-H 55-gallon carbon steel drum. For noncompactible trash, plywood boxes ranging from 1 ft^3 up to 128 ft^3 are used. Many of these larger boxes are 7 feet long so that they fit across the bed of a flatbed trailer. Metal containers other than 55-gallon drums get as large as 300 ft^3. Most of these containers and casks are specifically manufactured to be used together. Because few utilities own their own shipping casks they rent the casks from their contracted shipper. Each of these transportation companies owns several differently sized casks to accommodate almost any size container used in various radwaste systems. Figures 3.6-1 through 3.6-6 show the physical dimensions of the most widely used containers. Other pertinent data are also given in these figures.

3.7 Radwaste Shipping Casks

As mentioned in Section 3.6, radwaste shipping containers (liners) are loaded into casks. For instance, a 50-ft^3 container fits into a slightly larger companion cask, which is also called 50 ft^3 in size. Some casks may be used to ship containers of varying sizes. For example, the HN-100 holds three 55-gallon drums, eight 30-gallon drums, or one 75-ft^3 container.

The primary purpose of any shielded cask is to ensure that the dose rates at specified distances from the transport vehicle do not exceed regulatory limits. In order to determine which casks are suitable for each category of waste, they are rated in accordance with the type of radioactive material they are designed to handle. These categories are:

- Low specific activity (LSA)
- Type A

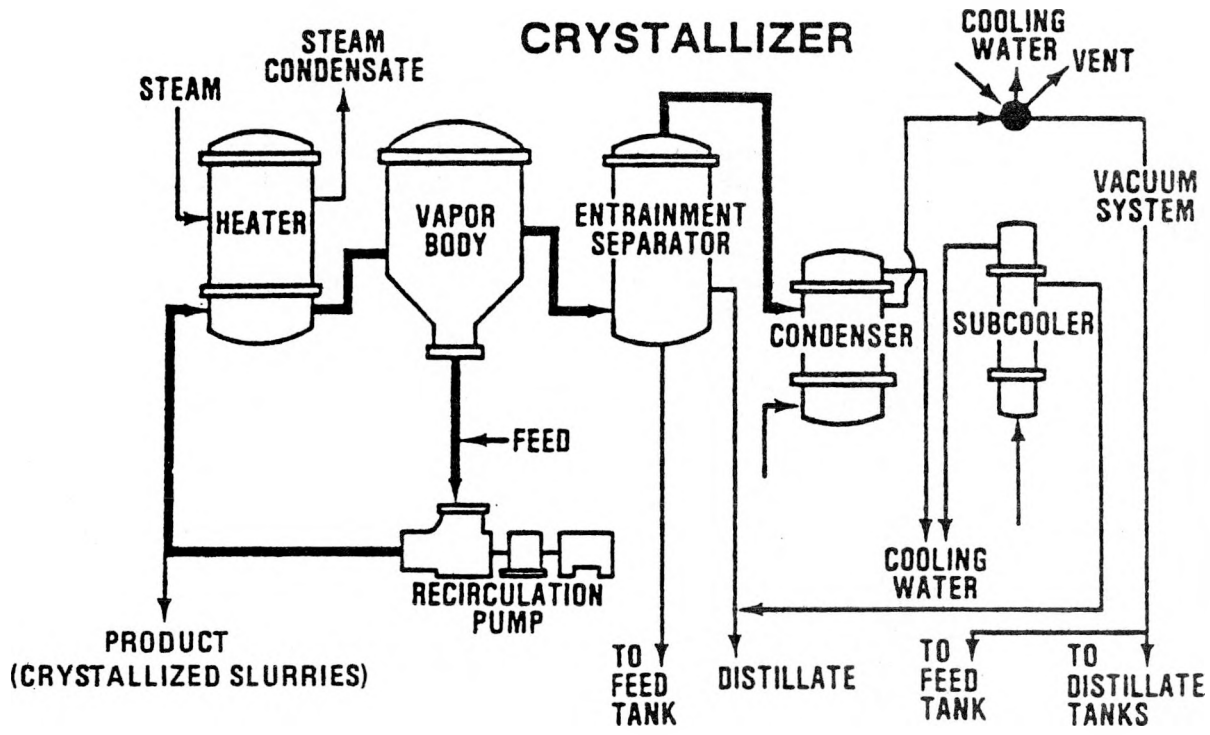


Figure 3.5-5 Flow Diagram of HPD Evaporative Crystallizer[1]

1. Courtesy of HDP Inc.

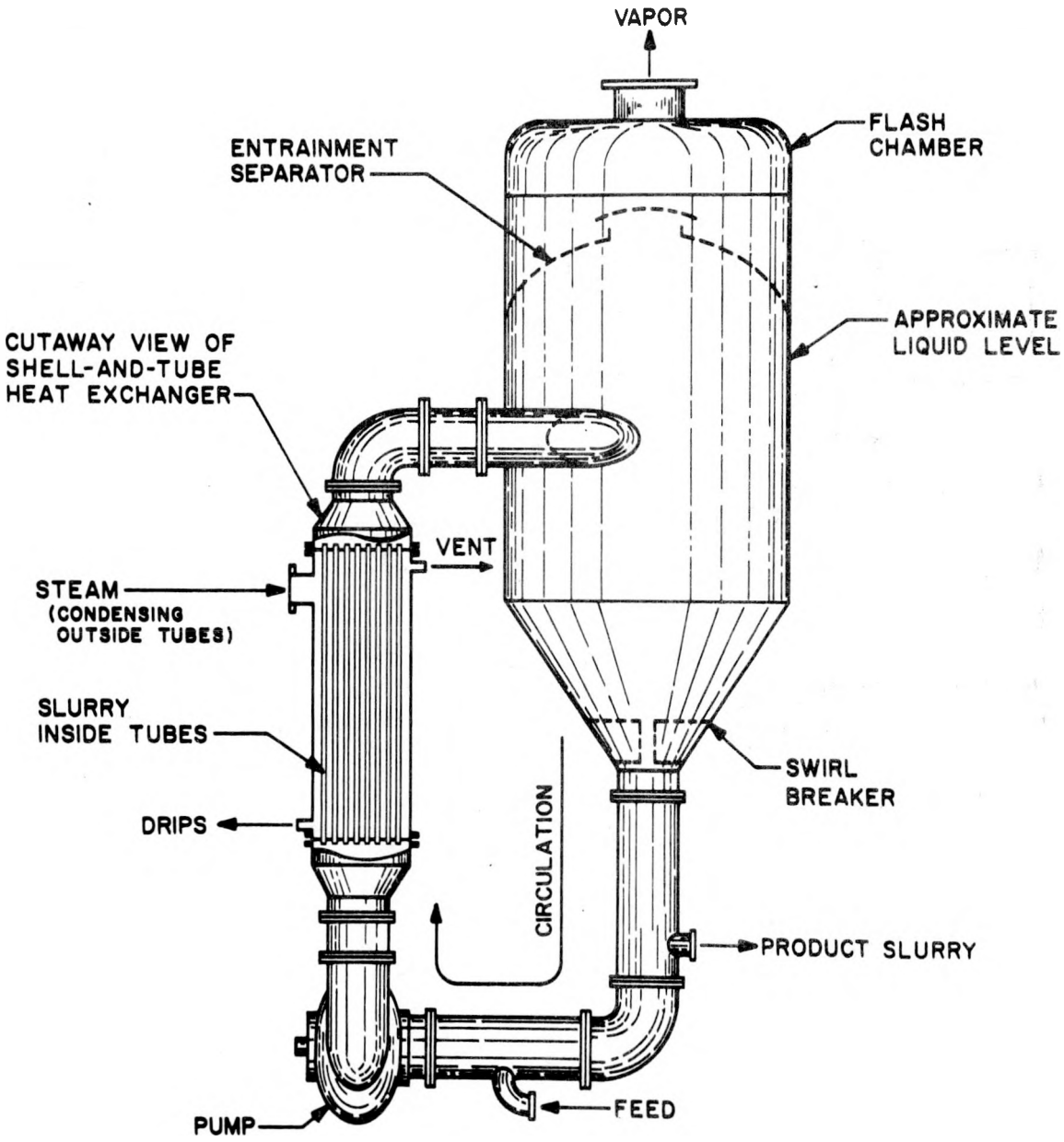


Figure 3.5-6 Evaporator/Crystallizer [1]

1. Taken from Godbee, 1978.

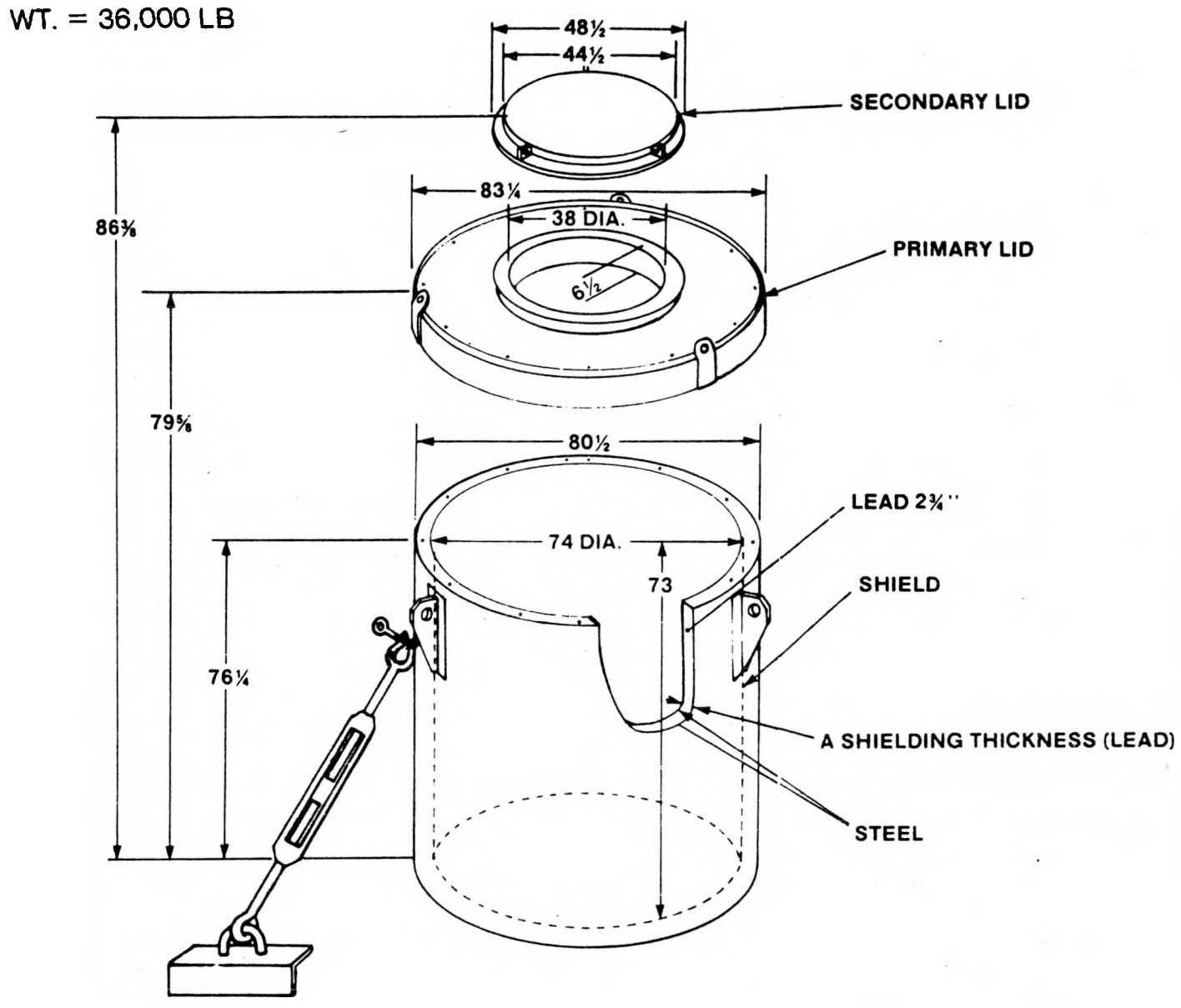
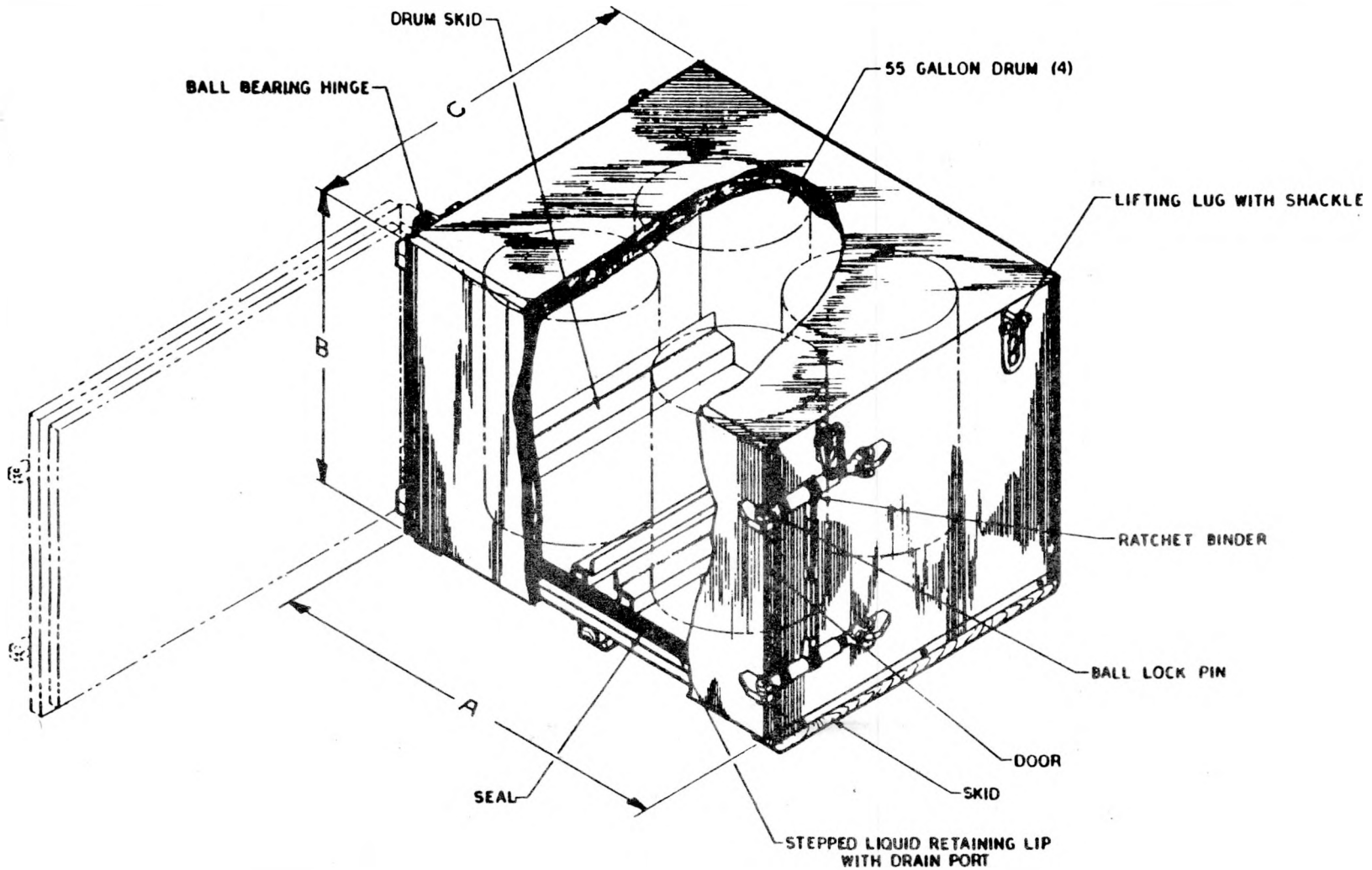


Figure 3.6-1 L3-181 Transport Cask[1]

1. Courtesy of Nuclear Engineering Company, Inc.



MODEL	DIM. A	DIM. B	DIM. C	STEEL THK.	EQUIV. LEAD THK	WEIGHT
4D-3S/2L-E	66	45	54	3	~2	13,800 LBS.
4D-4S/3L-E	68	47	56	4	~3	19,200 LBS.

Figure 3.6-2 4D-3S/2L-E and 4D-4S/3L-E Transport Cask[1]

1. Courtesy of Nuclear Packaging, Inc.

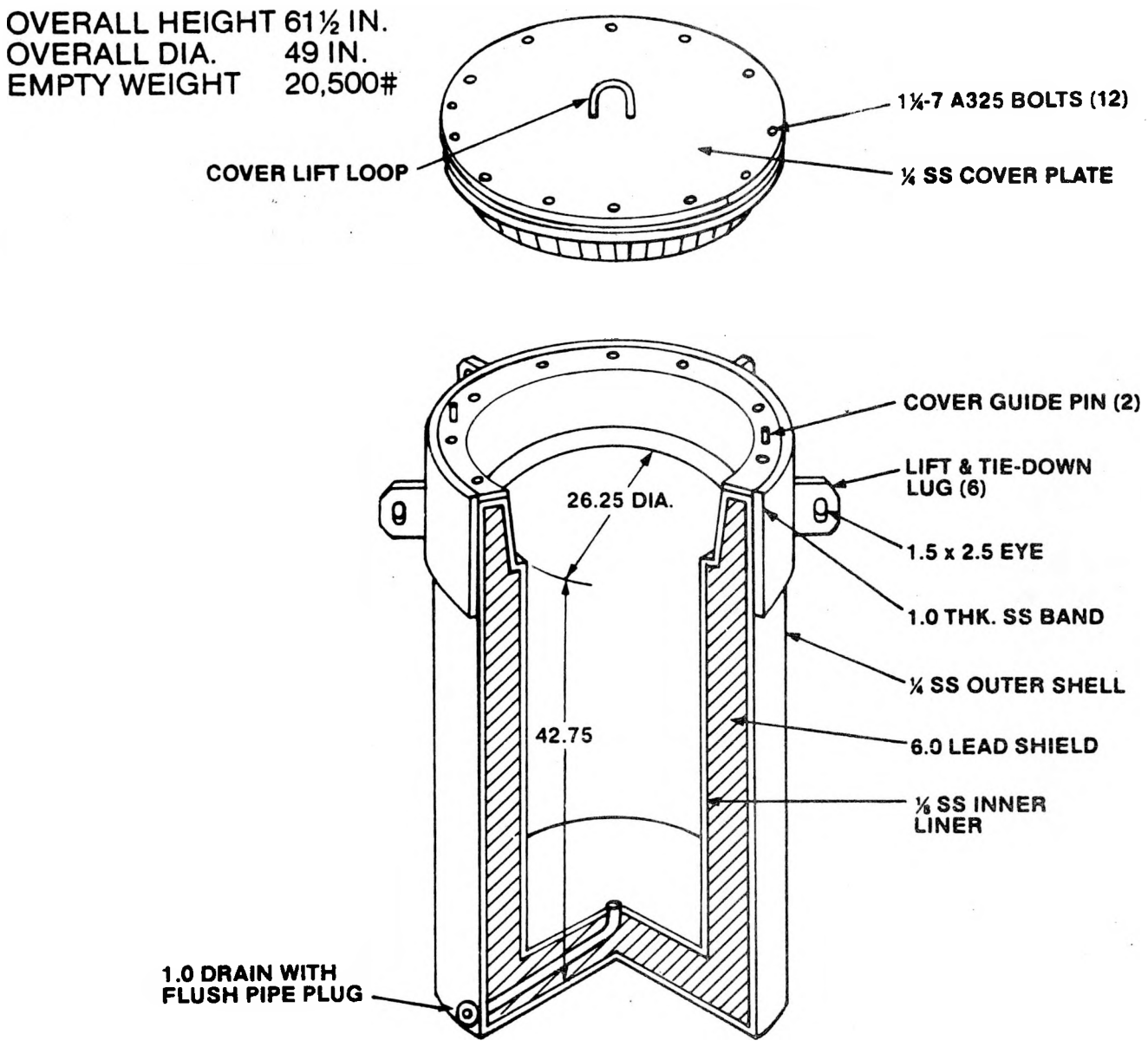


Figure 3.6-3 NECO B3 Transport Cask(1)

1. Courtesy of Nuclear Engineering Company, Inc.

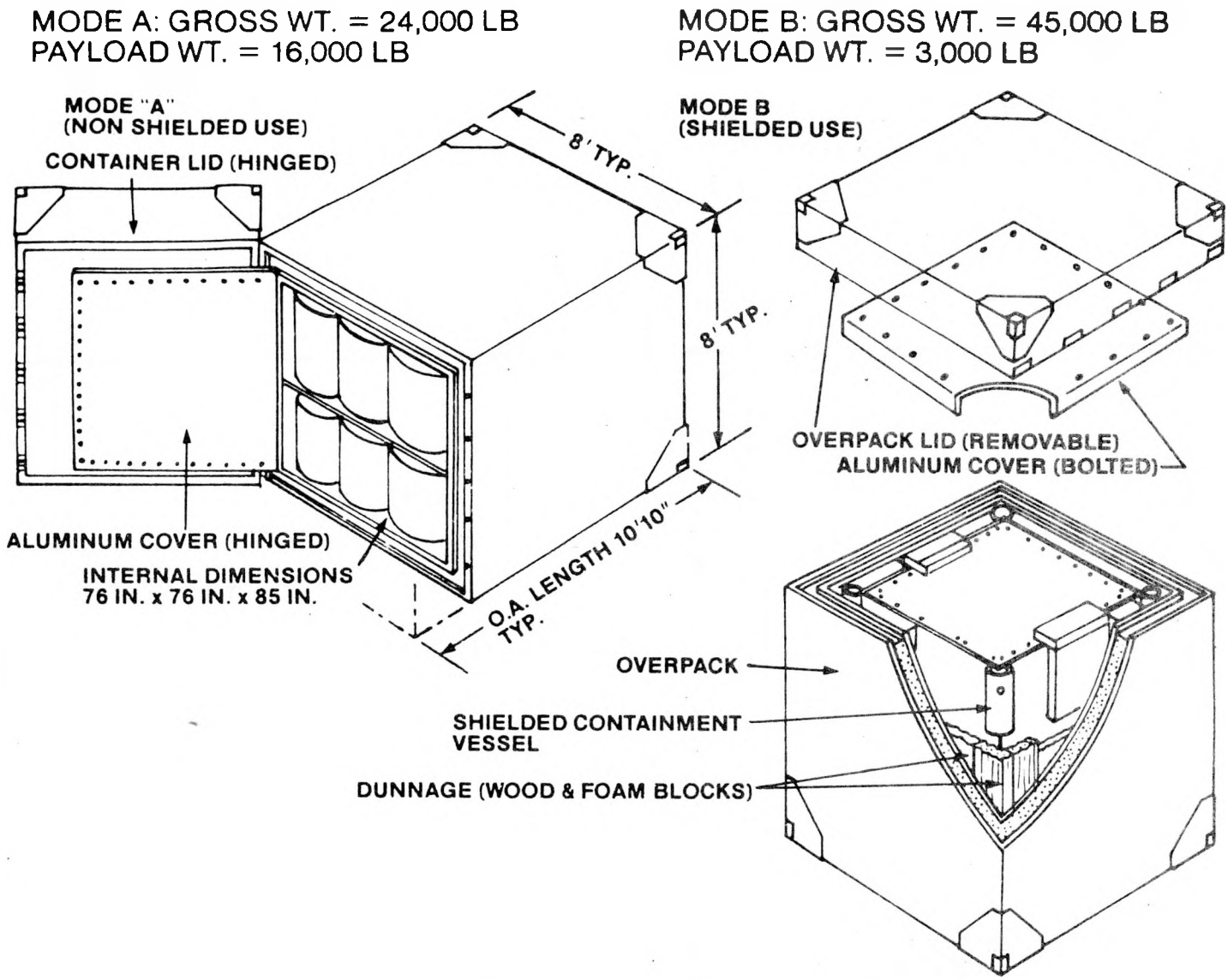


Figure 3.6-4 Half Super Tiger Transport Cask[1]
(DOT Permit No. 6679)

1. Courtesy of Nuclear Engineering Company, Inc.

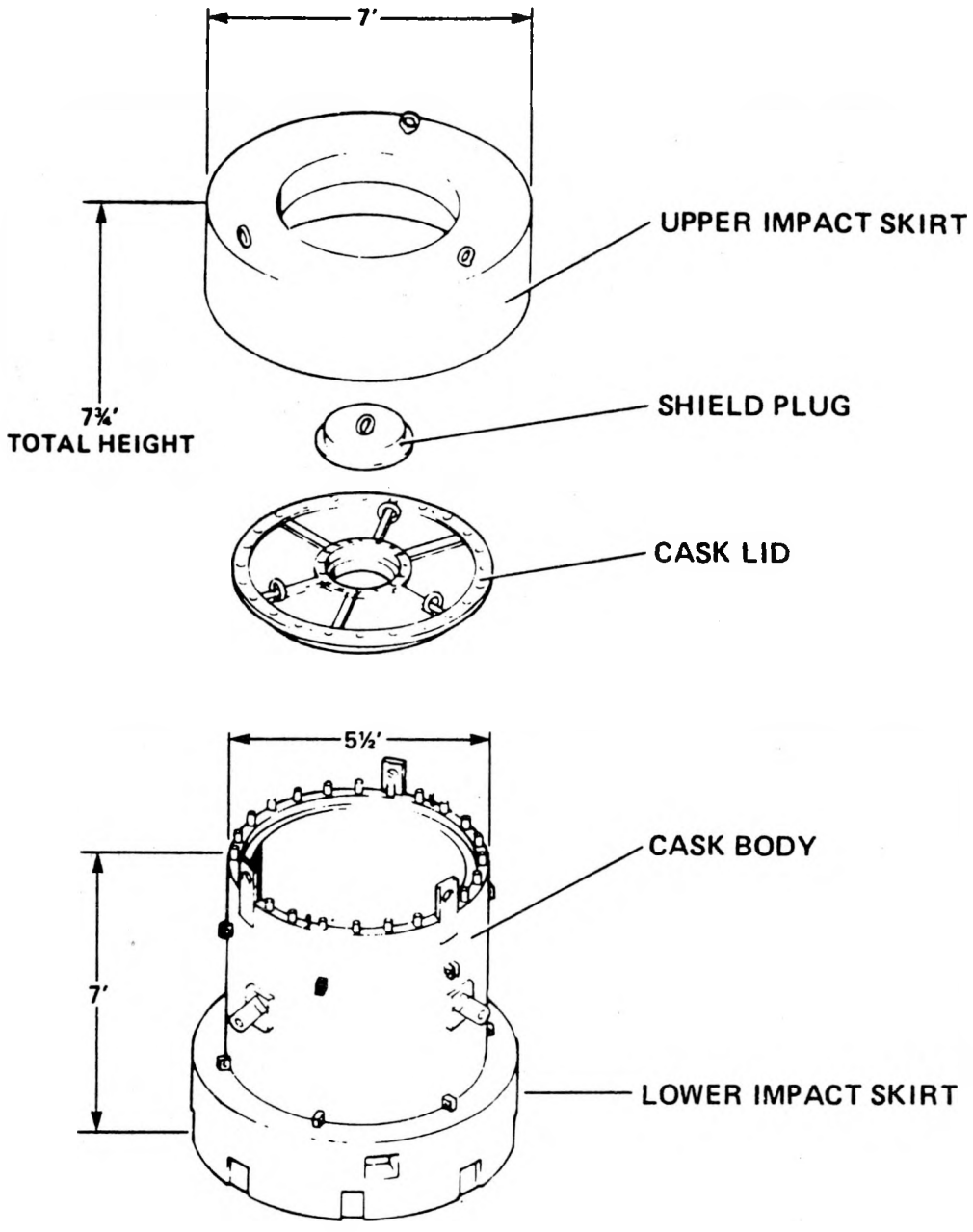


Figure 3.6-5 HN-200 Series Transport Cask(1)

1. Courtesy of Hittman Nuclear & Development Corporation

PAYLOAD LIMIT: 30,000 LB.
EMPTY WEIGHT: 15,000 LB.

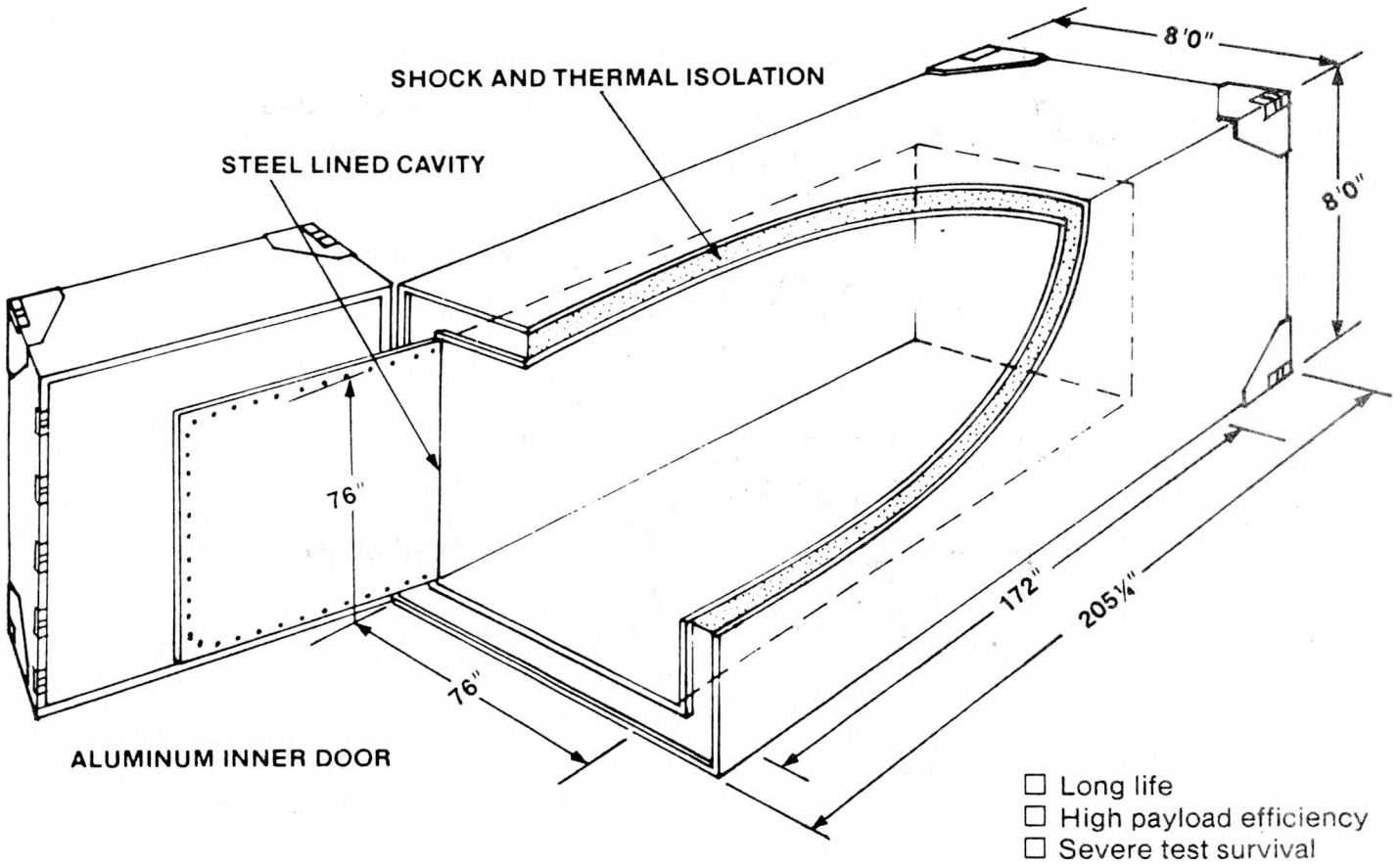


Figure 3.6-6 Super Tiger Transport Cask[1]
(DOT Permit No. 6400)

1. Courtesy of Nuclear Engineering Company, Inc.

- Type B
- Large quantities.

Each of these waste types has been described in Chapter 2. Table 3.7-1 lists general data regarding most of the commercial shipping casks used in the United States today. The following subsections describe the casks by category.

3.7.1 Low Specific Activity (LSA) Casks

LSA materials transported in sole-use vehicles usually only have to be shipped in strong tight containers as defined by the packager and shipper. Shipments of LSA material must still conform, however, to the radiation dose requirements of 49 CFR 173.393. One example of a shielded container for LSA material is the BB-1 (Big Bertha) (Figure 3.7-1) owned by Chem Nuclear Systems. The BB-1's usable volume is 12.8 m^3, with an empty cask weight of about 15 tonnes. The shielding can be varied from 2.5 to 5.0 cm of lead. Another type of container for LSA material is the HN-300 series radwaste shipping cask (Figure 3.7-2) offered by the Hittman Nuclear & Development Corporation. This cask is a box-shaped container weighing 19 tonnes, including the internal conveyor system. It is used for shipping fourteen 55-gallon drums with a total payload not exceeding 9,000 pounds. Similar shielded containers for LSA material are available from other companies. The wastes with lowest activity are usually shipped in a van truck. If shielding is needed, steel sheet metal is welded to the van.

3.7.2 Type A Quantity Casks

Type A quantities of low-level reactor waste must be shipped in packaging that meets the requirements of DOT Specification 7A as found in 49 CFR 178.350. In addition to meeting the general packaging requirements of 49 CFR 173.24 and 49 CFR 173.393, Specification 7A packaging must be capable of maintaining its shielding integrity and preventing the dispersal of its contents while the package is subject to the defined normal conditions of transport. The regulations, which prescribe a series of tests designed to simulate these normal conditions, are found in 49 CFR 173.398(b). These tests are designed to simulate the severity of an accident under conditions normally incident to transportation.

Each shipper of a DOT Specification A package is required to maintain on file for a period of at least a year after the latest shipment a complete certification and safety analysis report demonstrating that the packaging is in compliance with Specification A. Most reactor wastes are packaged in steel cylinders (liners) ranging in volume from 50 to 200 ft^3 and shipped in shielded casks that meet Type A packaging requirements. In this case the liner does not have to meet DOT specifications. For radioactive waste shipments that do not require radiation shielding, metal drums that meet DOT specifications must be used. Wooden boxes and fiberboard drums meeting DOT specifications are examples of unshielded Type A

Table 3.7-1 Radioactive Waste Transport Shield Casks

Cask Identification	Type of Material	Approximate Empty Weight (lb)	Payload Weight (lb)	Capacity
Nuclear Engineering Company				
L3-181 Spec. 55	LSA Type A	38,000	7,000	14 55-gal drums 1 100-ft^3 liner 169 ft^3
L2-181 Spec. 55	LSA Type A	26,000	8,000	14 55-gal drums 1 100-ft^3 liner 169 ft^3
A-4 L2-252 Spec. 55	LSA Type A	42,000	8,000	18 55-gal drum 1 100-ft^3 liner 252 ft^3
S2-5-208 7A	Type A	25,000	15,000	15 55-gal drums 208 ft^3
B-2(1) S3-208-IL DOT 6144	LSA Type A Type B	45,000	7,000	15 55-gal drums 208 ft^3
B-2 S3-208 DOT 6144	Type B	42,000	8,000	15 55-gal drums 208 ft^3
B-3 L6-13 DOT 6058	LSA Type A Type B	20,000	500	1 55-gal drum 10 ft^3
L6-1(2) (Stainless St.) Spec. 55	Type A Type B	4,500	500	1 ft^3
L4-50 4" Pb	LSA Type A	24,100	14,500	1 50-ft^3 liner
DOT 6400(1) Super Tiger	Type B	18,000	27,000	42 55-gal drums
DOT 6679 1/2 S-T	Type B	Unshielded 24,000 Shielded 42,000	16,000 3,000	18 55-gal drums

Table 3.7-1 Radioactive Waste Transport Shield Casks (Cont'd)

Cask Identification	Type of Material	Approximate Empty Weight (lb)	Payload Weight (lb)	Capacity
Nuclear Engineering Company (cont'd)				
DOT 6272 Poly Panther	Type B	3,500	3,000	120 ft^3
DOT 5800	Type B (Non-fissile)	700	3,300	5 ft^3
DOT 6008	Type B (Non-fissile)	880	5,120	5 ft^3
DOT 6744(1) Poly Tiger	Type B Large quantities	---Built to Customer Specifications---		
L4-50 3 in. Pb	LSA	18,000	4,000	1 50-ft^3 liner
L4-50 2 in. Pb	LSA	13,500	15,000	1 50-ft^3 liner
L4-50 1.5 in. Pb	LSA	9,600	13,200	1 50-ft^3 liner
Atcor Casks				
LL-57-65 Vandenburgh	Type B Large quantity	57,000	NA	3 55-gal drums 65-ft^3 liner
LL-28-4	Type B Large quantity	28,000	NA	4 ft^3
BC-48-220	Type B	48,000	NA	200 ft^3 14 55-gal drums
AL-33-40	LSA	32,800	NA	6 55-gal drums 1 75-ft^3 liner
LL-50-100	Type B	50,000	NA	8 55-gal drums 1 100-ft^3 liner
AL-31-12D	LSA	31,000	NA	12 55-gal drums

Table 3.7-1 Radioactive Waste Transport Shield Casks (Cont'd)

Cask Identification	Type of Material	Approximate Empty Weight (lb)	Payload Weight (lb)	Capacity
Atcor Casks (cont'd)				
AC-10-14	LSA	10,000	NA	1 55-gal drum
AC-27-240	LSA	27,300	NA	14 55-gal drums
Nuclear Packaging				
14D-2L	Type A	35,400	NA	191 ft^3
50CF-1S	Type A	3,800	NA	56 ft^3
50CF-1.5L	Type A	9,960	NA	56 ft^3
50CF-2.5L	Type A	16,000	NA	56 ft^3
50CF-4L	Type A	26,500	NA	56 ft^3
14D-IL	Type A	34,000	NA	191 ft^3
4D-3S/2L-E	Type A	13,800	NA	4 55-gal drums
4D-4S/3L-E	Type A	19,200	NA	4 55-gal drums
N-55	Type B	NA	NA	1 55-gal drum
7D-1.5L	Type A	16,250	NA	106 ft^3
7D-3L	Type A	15,000	NA	106 ft^3
Hittman Casks				
HN-100[3] Series	LSA Type B	33,500	14,000	170-ft^3 liner 14 55-gal drums 18 30-gal drums
HN-200 Series	Type B	37,400	10,500	3 55-gal drums 8 30-gal drums 75-ft^3 liner
HN-300 Series	LSA Type A	38,000	9,000	12 55-gal drums 3 50-ft^3 liners

Table 3.7-1 Radioactive Waste Transport Shield Casks (Cont'd)

Cask Identification	Type of Material	Approximate Empty Weight (lb)	Payload Weight (lb)	Capacity
Hittman Casks (cont'd)				
HN-400	LSA	32,000		18 55-gal drums
Shielded Van(4)	LSA	18,000		24-36 55-gal drums 6 70-ft^3 liners
Chem. Nuclear				
BB-1 Big Bertha	LSA	37,000		450 ft^3
CNS-101	LSA			6 55-gal drums
DOT 6144	Type B Large quantity	42,000	5,000	15 55-gal drums
MODAL 1600	Type D	23,050	6,950	1 55-gal drum or liner
CNS-1	LSA			6 55-gal drums
CNS-14-175	LSA	38,650	3,400	14 55-gal drums
NUS Corp.				
SN-1	Type B	34,000	8,000	14 55-gal drums

1. Authorized for liquids.

2. Type B in Suitable Overpak.

3. Impact skirts are required for Type B quantities.

4. Also handles any non-standard approved packages up to 800 lb. and compatible with shielded envelope.

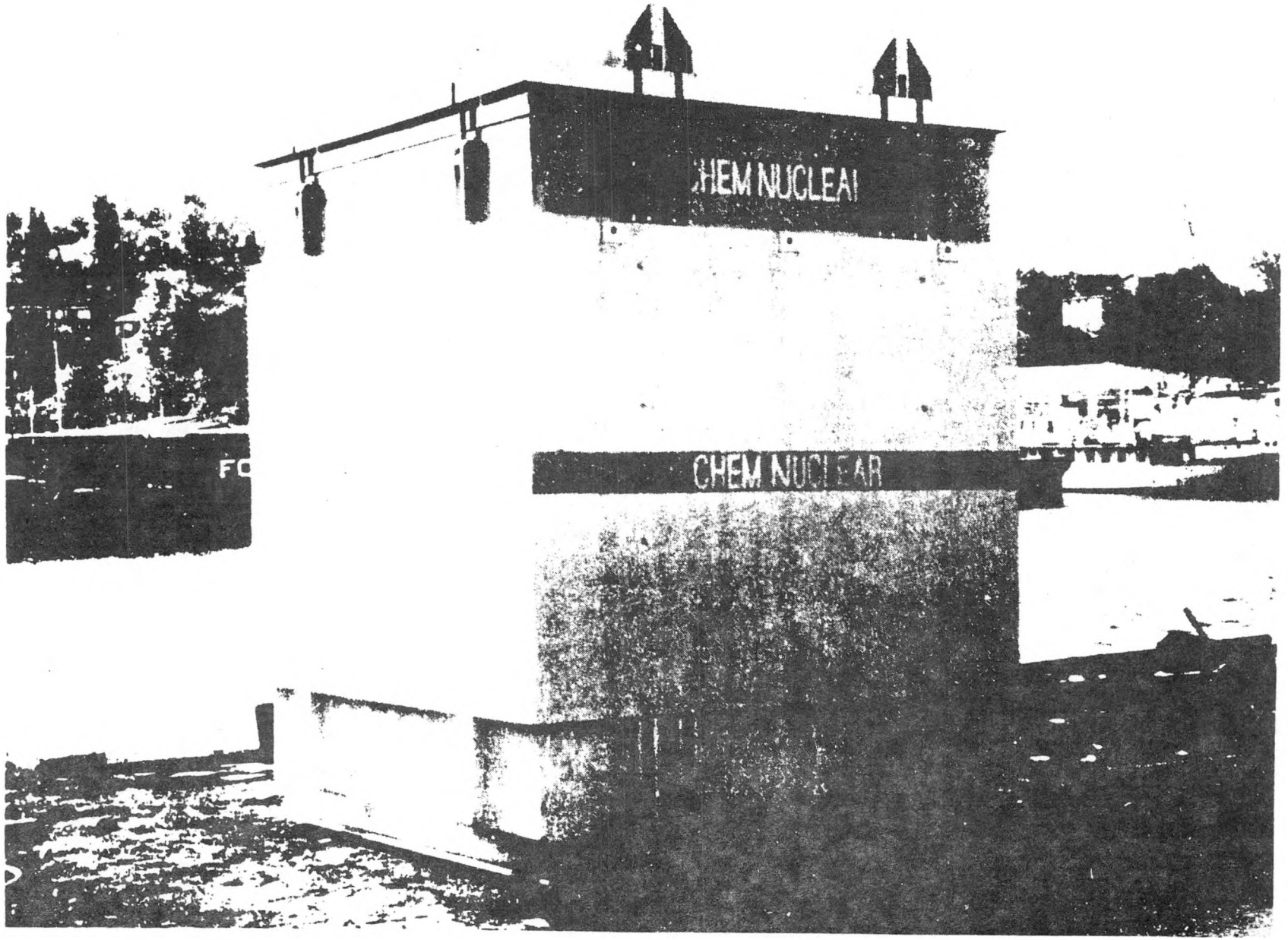


Figure 3.7-1 BB-1 (Big Bertha) Transport Cask(1)

1. Courtesty of Chem Nuclear Systems, Inc.

containers. A list of DOT specification containers meeting Specification 7A requirements is presented in Table 3.7-2.

An example of a radwaste shipping shielded cask employing lead and steel shielding for Type A materials is the L3-181 (Figure 3.6-1) offered by the Nuclear Engineering Company. This is a top-loading cylindrically shaped cask weighing about 18 tonnes with the capacity to hold fourteen 17-H drums.

Other examples of commercially available shipping casks are the 4D-3S/2L-E and 4D-4S/3L-E (Figure 3.6-2) offered by Nuclear Packaging, Inc. These are box-shaped steel containers weighing 13,800 and 19,200 lb, respectively. Each has the capacity to hold four steel drums. A variety of other packaging is also available, including steel bins in a variety of sizes, shielding casks of various volumes and shielding thicknesses, banded wooden boxes, and concrete bins.

3.7.3 Type B Quantity Casks

Type B quantities are also divided into two basic categories, those that do require shielding and those that do not. In almost all cases the reactor wastes falling in Type B category will require shielding. Type B packages are subjected to a much more severe testing environment than Type A packaging since Type B packages are required to carry larger quantities of radioactive materials. In addition to the general packaging requirements and the performance standards for normal conditions of transport, certain accident-damage test conditions, with resulting limited loss of shielding capability and essentially no loss of containment, must be satisfied. The performance criteria that the package designer must use to assess Type B packaging against these accident-damage test conditions of transport are prescribed in 49 CFR 173.398(c) and 10 CFR 71.

Two examples of Type B shield casks are Nuclear Engineering's NECO B3 and their Half Super Tiger. The NECO B3 (Figure 3.6-3) is a cylindrical top-loading lead and steel shielded cask weighing about 20,500 pounds and holding one steel drum. The Half Super Tiger (Figure 3.6-4) is a box-shaped container weighing 24,000 pounds and having a payload of 18 drums weighing up to a total of 16,000 pounds in the unshielded mode. In the shielded mode the empty weight is 45,000 pounds with a payload weight of 3,000 pounds. All Type B packaging designs require the prior approval of the U.S. Nuclear Regulatory Commission in the form of a license or certificate.

3.7.4 Large Quantity Casks

Large quantities of material are defined as quantities greater than a Type B quantity and require special packaging. The most common materials involved as large quantities are the high-curie irradiation sources, plutonium-bearing unirradiated nuclear fuels, irradiated fuel materials, and many forms of nuclear wastes.

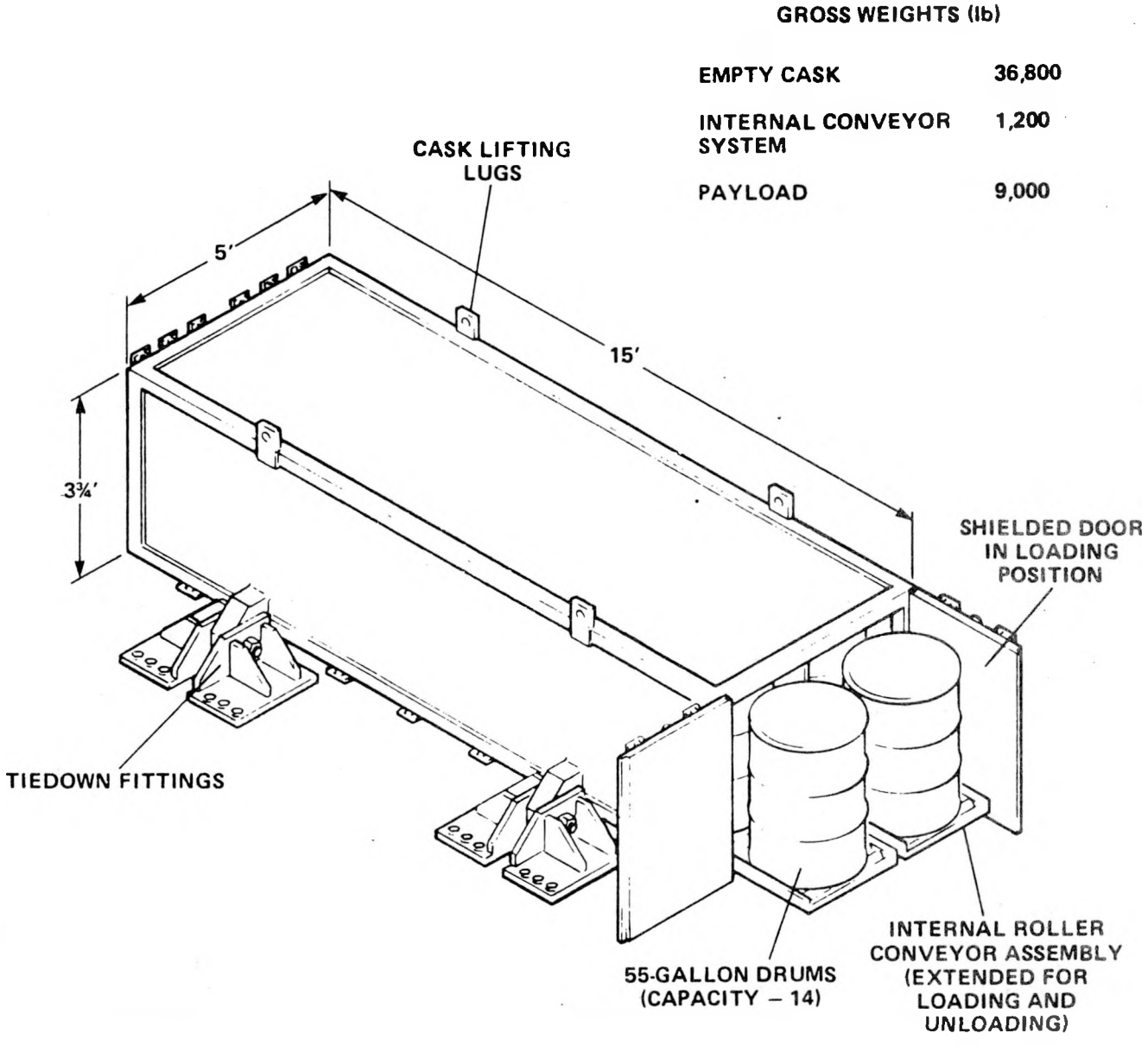


Figure 3.7-2 HN-300 Series Transport Cask[1]

1. Courtesy of Hittman Nuclear & Development Corporation

Table 3.7-2 DOT Specification Containers Found to Meet Specification 7A

Container	DOT Specification Number	Container Capacity	
		Volume (liters)	Weight (kg)
Steel drums	6B	113	272
	6C	19	36
		38	72
	6J	210	400
	6L	Various	Various
	6M	Various	Various
	17C	19	45
		210	380
	17H	113	227
		210	380
Aluminum drums	42B	210	250
Wooden boxes	15A	4-230	9.5-152
	19A	43-230	66-181
	19B	43-230	34-68
Fiberboard drums	21C	57-210	27-181

Packaging requirements for large-quantity materials involve all of the type B packaging requirements plus other provisions for such things as decay heat dissipation, potential leakage of contaminated heat-transfer medium, heavier shielding, and the like, which in some cases may involve a requirement for certain administrative controls during shipment. Low-level waste from power reactors is seldom shipped in Type B large-quantity casks.

One example of a radwaste shipping cask employing lead and steel shielding for Type B and large-quantity materials is the Hittman HN-200 series (Figure 3.6-5). This cask has an empty weight of 37,400 pounds and a maximum payload of 10,500 pounds. It will carry either a single disposable liner, or three 55-gallon drums, or eight 30-gallon drums. Another example of a transport container capable of carrying large quantities is Nuclear Engineering's Super Tiger (Figure 3.6-6). This is a box-shaped container with an empty weight of 15,000 pounds and a maximum payload of 30,000 pounds. The Super Tiger carries 42 drums.

3.8 Advanced Volume Reduction Techniques

Volume-reduction techniques that may provide volume-reduction factors larger than those for the systems discussed in Section 3.5 are under development. These systems are being adapted either from other industries or were developed originally for high-level waste treatment and are now being modified for low-level waste. Three of these techniques are discussed.

3.8.1 Molten-Salt Combustion Process

The molten-salt combustion process being developed by the Atomics International (AI) Division of Rockwell International involves reacting the waste in a molten pool of inorganic sodium salts, at temperatures ranging from 1,500 to 1,800°F. Sodium carbonate, which is a liquid at these temperatures, is the primary salt used.

The salts are fed into the molten-salt vessel through the carbonate feeder. Combustible materials are transferred directly from a hammermill in which they are crushed to the required size, into a feed hopper provided with a variable-speed auger, and then introduced into the airstream for transport into the vessel.

Exhaust gases generated in the vessel leave through refractory-lined tubes to a refractory-lined mist separator. The separator traps entrained melt droplets on a baffle assembly. The gases are then transferred through a duct to a high-energy venturi scrubber to remove any particulate matter before release to the atmosphere. A flow diagram of the molten-salt process is shown in Figure 3.8-1.

The combined combustion process and chemical reaction completely destroy the waste. The combustion gases are scrubbed by the molten carbonate pool, which absorbs any sulfur, halogens, and phosphorus

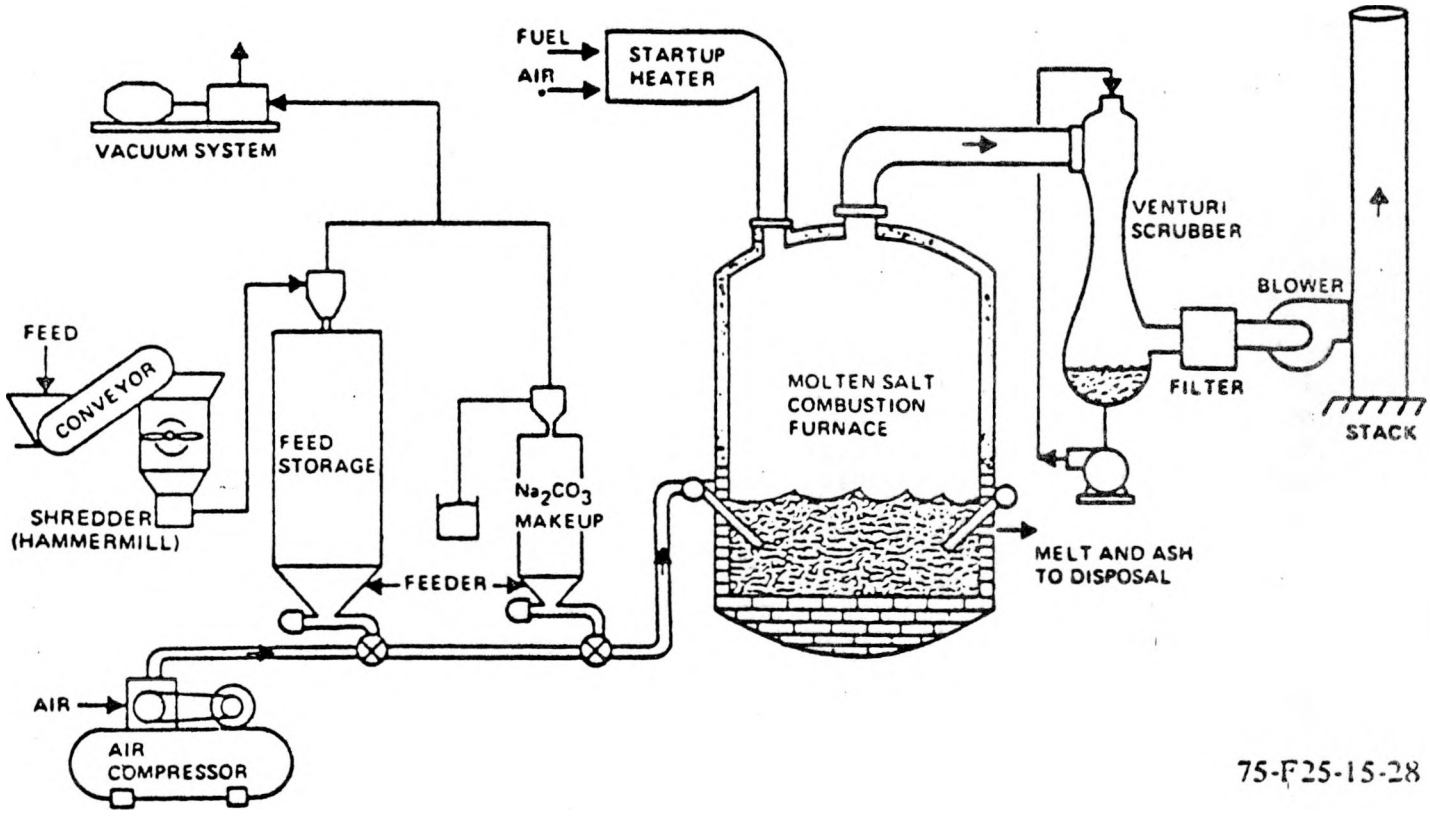


Figure 3.8-1 Molten Salt Flow Diagram[1]

1. Taken from Feizollahi, 1978 (to be published).

as the corresponding sodium salt. The only effluents are carbon dioxide and water vapor. Significant quantities of nitrogen oxides are not created by fixation of nitrogen in the air because the temperatures are too low. Noncombustibles such as glass, metal, or ash are allowed to build up in the pool until their presence begins to alter the pool's chemical and physical properties. This limit is 20 wt% noncombustibles. At this time the bed is dumped and a new bed of fresh sodium carbonate is started. An alternate method used is to move a small side stream of the melt, quench it in an aqueous solution, filter it to remove the noncombustibles and return the sodium carbonate to the vessel. Systems that are operated on a batch basis, such as for a nuclear power plant, would use the first method whereas large operations operating continuously would use the second method.

Retention of radionuclides in the bed ranges from 99% for iodine to 99.9% for transuranics and activated corrosion products. Fission products have a retention factor of 99.5%. It is also possible to glassify the waste salts in the combustor by adding a glassifying matrix such as borosilicate glass to the combustor after the temperature has been raised to 2000°F or more. This would increase the viscosity of the melt and would result in a slower operation than draining the melt into a separate remelter. It has the advantage of casting the waste material directly into disposal containers without further treatment.

This process can give a volume-reduction factor of 46 without salt glassification. This volume-reduction factor results from casting the waste salt directly into the waste container. The final density is approximately that of the liquid rather than of a bulk powder. This volume-reduction factor is for the process only and does not include HEPA filters or other wastes produced by the process. A volume-reduction factor of 10 to 20 is projected for the process if glassification is used.

3.8.2 Inert-Carrier Radwaste Process (ICRP)

The inert-carrier radwaste process concept is being developed by United Technologies Corporation. This process uses a large vessel containing high-temperature inert fluid as a heat exchanger. It is claimed that an efficient evaporation can be accomplished by introducing liquid waste into a hot bath of the inert fluid while creating an extreme turbulence by high-velocity recirculation. The water in the radwaste is driven off and the dissolved solids are converted into granular particles suspended in the inert fluid bath. The suspended solids are removed from the inert fluid in a settling column. A proprietary solidification agent is used to cast the solids discharged from the settling column. A flow diagram of the system is shown in Figure 3.8-2.

The ICRP system is an extension of the well-established ICP. No new basic technology is involved in its application to radwaste volume reduction. This process has been in use for 20 years and was the basis for the design and operation of three production

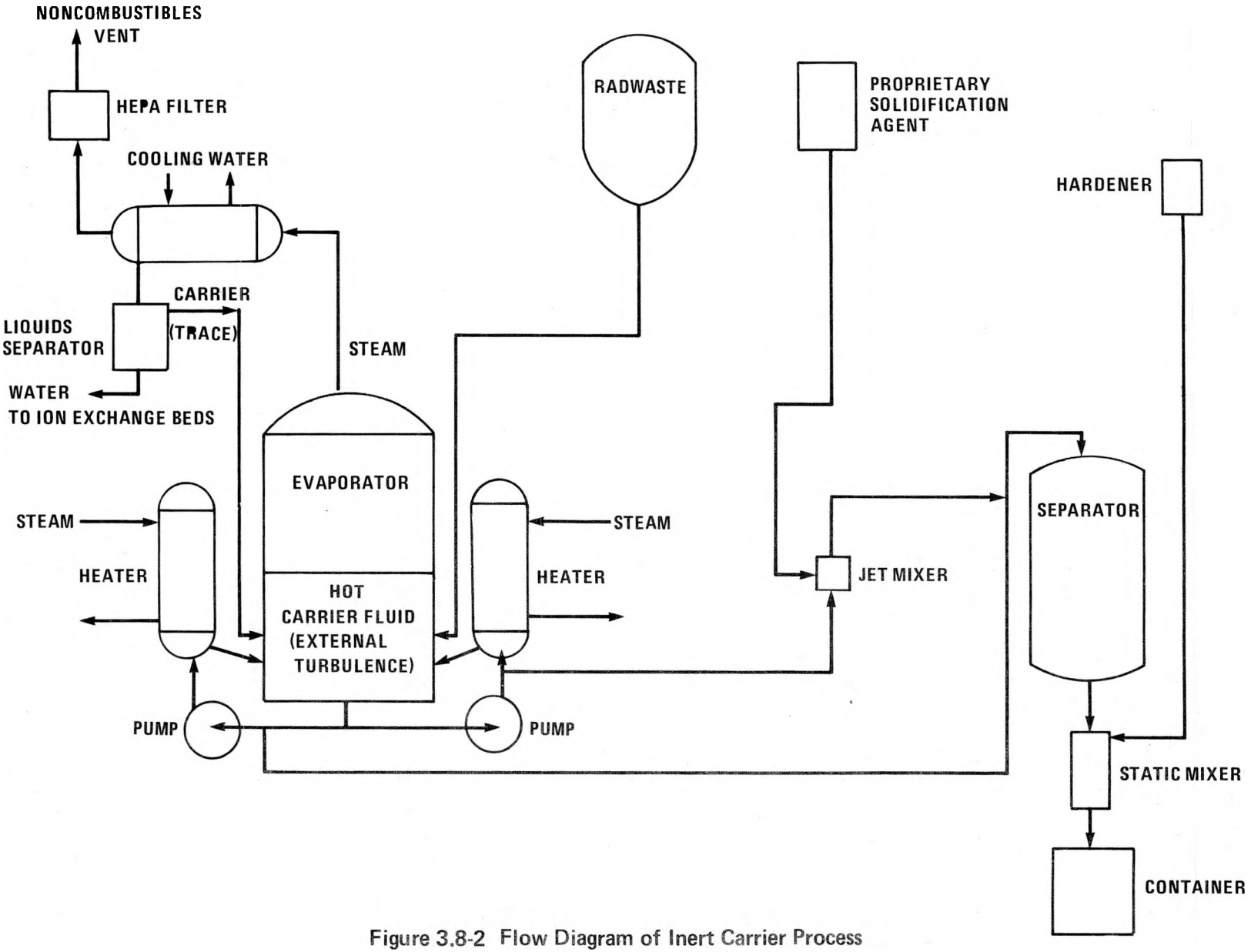


Figure 3.8-2 Flow Diagram of Inert Carrier Process

plants. One plant, at the Naval Ordnance Station, Indian Head, Maryland, is used for producing high-energy solid propellants for rockets at 2,500 lb/hr. A second plant, at Pine Bluff Arsenal, Arkansas, was used for manufacturing smoke compositions at 1,200 lb/hr. A third plant, at the Bermite Division of the Whittaker Corporation, Saugus, California, has been used for producing a flare composition at 5,000 lb/hr. The technology developed for processing hazardous materials in these plants can be directly applied to plants handling other hazardous materials, including radioactive wastes.

In addition to three production plant installations, successful ICRP pilot plant programs have been conducted for the Atlantic Richfield Hanford Company on its aqueous silicate process for high-level radwaste disposal and for the Power Reactor Development Corporation in Michigan (a group of 11 power companies) for the conversion of radioactive metallic sodium (from the Enrico Fermi reactor) to a solid hydroxide for burial. A small jet mixer (3 by 6 in.) was used in conjunction with a 5-gallon basic reactor for these two programs. Semi-solid aqueous silicate waste was produced at a rate of 12 lb/min (720 lb/hr) with this small equipment.

3.8.3 Acid-Digestion Process

The acid-digestion process is under development at the Hanford Engineering Development Laboratory (HEDL). The process is being developed to reduce the volume of combustible waste by converting it into a noncombustible residue. Various waste materials such as polyvinylchloride (PVC), polyethylene, paper and other cellulose materials, ion-exchange resins, and various types of rubber are digested in hot (230 to 270°C) concentrated sulfuric acid containing nitric acid as oxidant.

The following four reactions represent the major chemical reactions involved in the acid-digestion process:

a. $C_mH_n + n/2H_2SO_4 \longrightarrow nH_2O + n/2SO_2 + mC$

b. $C + 2H_2SO_4 \longrightarrow 2H_2O + 2SO_2 + CO_2$

c. $3C + 4HNO_3 \longrightarrow 4NO + 2H_2O + 3CO_2$

d. $5C + 4HNO_3 \longrightarrow 2N_2 + 2H_2O + 5CO_2$

The purpose of the sulfuric acid in reactions (a.) and (b.) is to carbonize the waste and to oxidize it to carbon dioxide. The oxidation reaction shown in reaction (b.) is somewhat slow, however, and nitric acid serves as a better oxidant. In many respects the sulfuric acid serves primarily as a high temperature reaction medium. This is particularly necessary for digestion of plastics such as PVC and polyethylene where temperatures near 250°C are required for complete oxidation of the waste. Even at these temperatures nitric acid alone will not destroy most of the waste

materials. A flow diagram of the Acid Digestion Test Unit is shown in Figure 3.8-3.

The offgas passes from the digester to the oxidation-absorption tower. The offgas consists primarily of carbon dioxide, carbon monoxide, sulfur dioxide, water vapor, oxides of nitrogen, hydrogen chloride, and chlorine when chlorine-containing species are present.

The overall volume reductions for the acid-digestion process depend on the final method for disposing of the residue. Also the wastes generated by the acid-digestion process itself, such as rubber gloves, HEPA filters, and solids from offgas treatment operations must be factored into the overall waste volume-reduction calculations. Overall projected volume-reduction factors are anticipated to be 10 to 30 for a complete acid-digestion process.

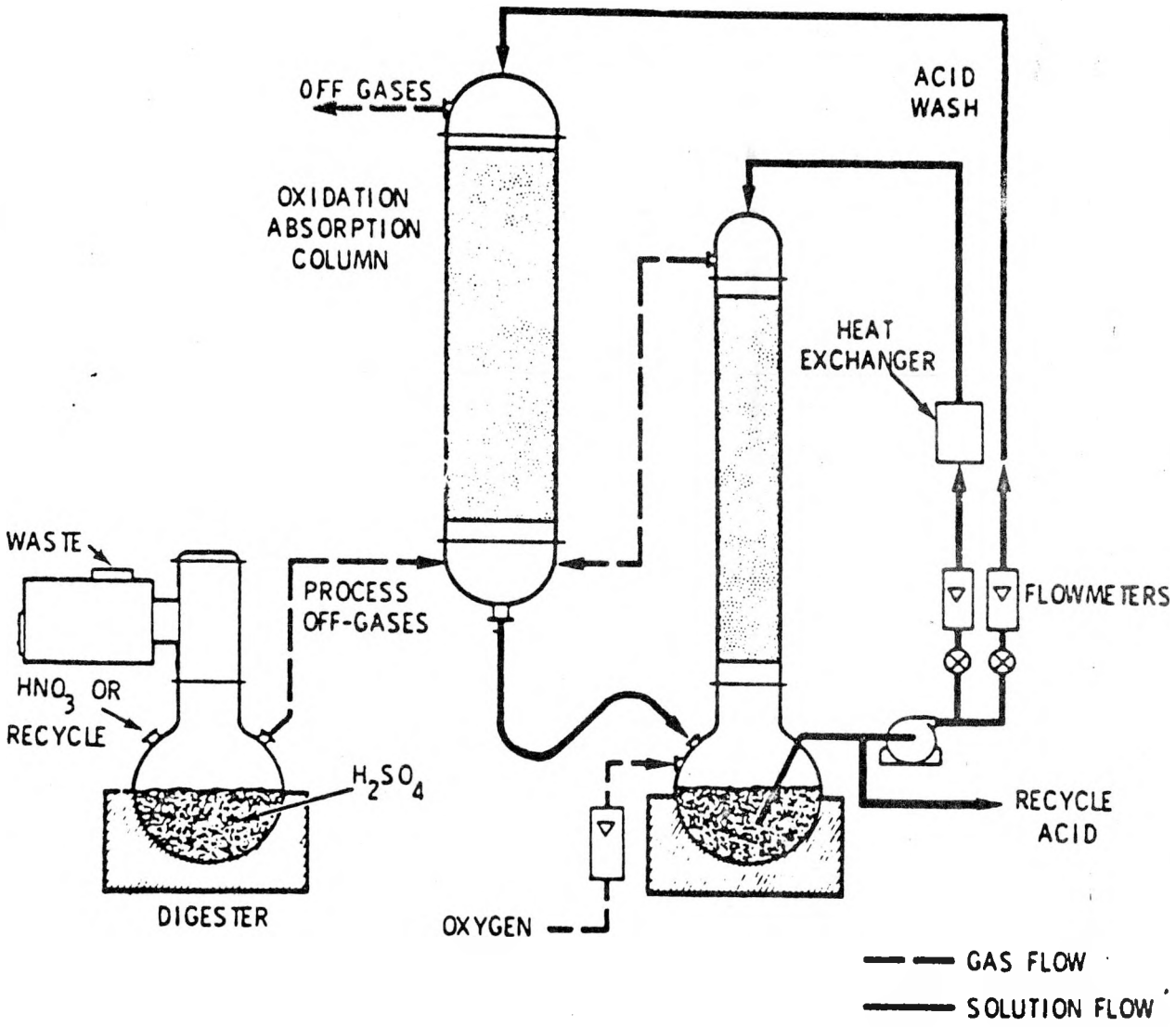


Figure 3.8-3 Flow Diagram of Acid-Digestion System[1]

1. Taken from Feizollahi, 1978 (to be published).

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4. SURVEY OF LWRS AND FUEL-FABRICATION PLANTS

4.1 Introduction

Aside from restricting the survey to LWRs in the United States only two other criteria were used to qualify plants for the survey:

1. The plant must have been operating commercially for 2 years prior to December 31, 1977.

2. The plant was required to have a maximum dependable capacity greater than or equal to 400 MWe.

A list of 36 plants meeting these criteria (Table 4.1-1) was made, and survey efforts proceeded from that point.

Of the 36 eligible plants, 14 were BWRs consisting of 17 units with 77 years of operating experience, and 22 were PWRs consisting of 29 units with 127 years of operating experience. Each of the facilities was sent a letter that described the purpose and scope of the survey. Soon after, officials at each plant were telephoned to make arrangements for an NUS Corp. employee to visit plant personnel onsite to complete the survey form. Of the 36 plants contacted, Oyster Creek, Peach Bottom, Fort Calhoun, Rancho Seco, and Surry could not participate, either because of the timing of the survey or other reasons beyond the control of plant personnel. The survey was conducted from April through June 1978.

Plants responding to the survey represent 63 years of operating data for BWRs and 112 years of operating data for PWRs.

These figures may be impressive but they may also be deceptive. The intent of this study was to predict the annual quantity of low-level waste from LWRs and fuel fabrication facilities in the United States. For both facilities, annual estimates are made for specific types of wastes generated, such as concentrated liquids and spent resins. In order for the data collected to be useful, it too must be broken down in this manner. Before 1975 the only regulatory requirement was to report the total annual volume of solid waste and the total curies associated with it. In June 1974, Revision 1 to the Nuclear Regulatory Commission (NRC) Regulatory Guide 1.21 (R.G. 1.21), "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, 1974," was issued. It requires that waste data be broken down into four categories: (a) spent resins, filter sludges, and evaporator bottoms; (b) dry compressible waste and contaminated equipment; (c) irradiated components and

Table 4.1-1 Plants Selected for Survey

Plant	Date of Commercial Operation	Design Power (MWe)
Boiling Water Reactors		
Brunswick 2	11-3-75	821
Cooper	7-1-74	778
Dresden 2&3	6-9-72; 11-16-71	809 each
Duane Arnold 1	7-28-75	569
FitzPatrick 2	7-28-75	821
Hatch 1	12-31-75	786
Millstone Point 1	3-71	652
Monticello	6-30-71	545
Nine Mile Point 1	12-69	625
Oyster Creek 1	12-69	640
Peach Bottom 2&3	7-5-74; 12-23-74	1,065 each
Pilgrim 1	12-72	664
Quad Cities 1&2	2-18-73; 3-10-73	800 each
Vermont Yankee	12-29-72	514
Pressurized Water Reactors		
Arkansas Nuclear 1	12-19-74	902
Calvert Cliffs 1	5-8-75	880
Donald C. Cook	8-27-75	1,090
Fort Calhoun	6-20-74	501
R. E. Ginna	3-70	517
Haddam Neck	1-1-68	600
Indian Point 2	8-73	906
Kewaunee	6-74	563
Maine Yankee	12-28-72	830
Millstone Point 2	12-26-75	860
Oconee Units 1,2&3	7-15-73; 9-9-74; 12-16-74	922 each
Palisades	12-31-71	811
Point Beach 1&2	12-21-70; 4-20-73	497 each
Prairie Island 1&2	12-16-73; 12-21-74	538 each
Rancho Seco 1	4-17-75	889
H. B. Robinson	3-7-71	772
San Onofre 1	1-1-68	450
Surry 1&2	12-22-72; 5-1-73	788 each
Trojan	12-24-75	1,130
Three Mile Island 1	9-2-74	870
Turkey Point 3&4	12-14-72; 9-7-73	760 each
Zion 1&2	12-31-73; 9-17-74	1,050 each

control rods; and (d) other. R.G. 1.21 requires that both volumes and gross activity be reported. Most plants now submit their semiannual reports according to this format, but a few plants have not adopted it, and some plants have records that are more detailed. The data from plants with detailed records are used to develop projections of annual waste volumes and activities for specific waste types.

The analysis of the data yielded two results. The first result is the annual generation of a specific waste type for all plants in a given category (for example, the annual generation of spent resin for BWRs with deep bed condensate polishing systems). In developing this number, virtually all of the data collected in the survey are considered. A reason is given for any data not used. The second result is the annual generation of this waste type for a typical plant. In this case some of the data may be excluded from the analysis. For instance, data on large quantities of spent resin would be excluded when the resin resulted from inleakage of seawater to the condenser in plants with a deep bed condensate polishing system. If the leak were sufficiently large that the resin beds could not be regenerated fast enough, several hundred cubic feet of resin might be dumped into the spent resin tank in an effort to keep the plant online. In using the data to determine annual waste volumes, as seen by the burial facility, these atypical values are important and with few exceptions have been included in the analysis. When using the data to describe a typical plant, these data are not appropriate to the analysis and have been excluded.

The same consideration is given to waste activities. An even more graphic example is the shipment of chopped spent fuel channels listed as noncompactible trash. Even though the volume is not a major contribution to overall annual waste volume, it may increase the year's activity by tens of thousands of curies.

In addition to information on volumes and activity levels of LWR waste, the survey collected data regarding the following items:

1. Radionuclides present
2. Nature of waste (e.g. type of resin and filter material)
3. Whether waste is solidified
4. Fraction of shipped volume which is waste instead of solidification agent
5. Density
6. Container used to ship waste and type of shielding used
7. Percent void in container when full
8. Surface dose and dose at 3 feet.

With the exception of item 4, each item listed is discussed, and the survey information pertaining to it is tabulated in the following sections. The tables in the following sections that show the volume of waste shipped offsite have been adjusted so that the reported volume is the waste volume. The reported volume does not include solidification agent. In cases where the waste was not

solidified, obviously no adjustment was made. Tables 4.1-2 and 4.1-3 show what wastes different BWRs and PWRs solidify. The tables also show the solidification agent used and the percent of the final waste product that is radioactive waste.

4.2 Light-Water Reactors

4.2.1 Boiling Water Reactors

4.2.1.1 BWR Spent Resin

Table 4.2-1 lists spent resin data collected from plants with deep bed condensate polishing systems for 14 plant-years. The spent resin ranged from 0 to 9,420 ft^3/yr. Two plant-years of data for facility B4 did not include data on the total activity but the remaining 12 plant-years of data ranged from 0 to 2,098 Ci/yr. The annual waste generation rate is 3,200 ft^3/yr with 900 curies. When weighted to consider plant size, the generation rate is 4.6 ft^3/MWe-yr at 1.9 Ci/MWe-yr based on an average concentration of 0.42 Ci/ft^3. Based on the data, this information appears to result from normal plant operations. Therefore, these results are also indicative of a typical plant.

Of the six reactor sites that have deep bed condensate polishing systems, three use saltwater for condenser cooling. The other three use freshwater. The average waste generation rate for the saltwater sited plants is 5.8 ft^3/MWe-yr with 1.8 Ci/MWe-yr based on an average concentration of 0.32 Ci/ft^3. The rate for freshwater sited plants is 0.64 ft^3/MWe-yr with 0.35 Ci/MWe-yr based on an average concentration of 0.55 Ci/ft^3.

Only two of the six plants that use precoat filtration for condensate water quality control were able to supply specific data regarding the quantity of deep bed resin used in the plant.

As shown in Table 4.2-1 the annual shipments of deep bed resin ranged from 72 ft^3/yr to 188 ft^3/yr. When these data are weighted in terms of the gross electrical generating capacity of these plants, an average value of 0.23 ft^3/MWe-yr is derived. Of the two plants supplying data on the quantity of deep bed resin used, only facility B12 had data on the waste activity levels. Using these data as a guide, ranging from 0.11 Ci/yr to 1.86 Ci/yr, the estimated average activity collected on the resins is 0.0014 Ci/MWe-yr based on an average concentration of 0.006 Ci/ft^3. With such little data available and none of it exhibiting a significant difference from the rest of the data, it is expected that the typical plant has the same characteristics that the average plant has.

Table 4.2-2 lists the radionuclides present in BWR resins. There is no difference in the spectrum of radionuclides as a function of type of condensate system. The list includes both fission products and activated corrosion products, and soluble nuclides and insoluble nuclides. Of the 20 nuclides listed by at least

Table 4.1-2 Percentage of Waste in Solidified Product for BWRs

Plant	Solidification Agent	Liquid Waste	Filter Precoat	Deep Bed Resin
Deep Bed Condensate Polishing System				
B1	Cement	50	(1)	(2)
B2	Cement (until 3-76)	NA(3)	NA	NA
	UF (since 3-76)	60-65	NA	NA
B3	Microcell Process (1970-1973)	50	(4)	(2)
	UF (1974 - present)	66	60(5)	(2)
B4	UF	55	(5)	(2)
B5	Cement	55	55	55
B6	Cement (1975 and 1976)	67	(4)	(2)
	UF (1977 - present)	67	(4)	NA
Precoat Filter Condensate Polishing System				
B7	(6)	NCL(7)	(4)	(2)
B8	(8)	NCL	(5)	(2)
B9	Cement	NCL	55	45-55
B10	Cement (1971-1973)	50	50	63
	UF (1974 - present)	63	63	63
B11	(9)	NCL	(4)	(2)
B12	Cement	NCL	67	80
% of Plants Solidifying Waste Type		100	50	43

1. Radwaste filter precoat not solidified, RWCU precoat solidified.
2. Resin dewatered but not solidified.
3. Not available.
4. Filter sludge not solidified.
5. Solidification started in 1977.
6. Installed UF system not used.
7. No concentration of liquid wastes.
8. No solidification system installed.
9. Portable UF system installed but not used.

Average % Waste in Solidified Product

Waste Type	UF Systems	Cement Systems
Liquid waste	63%	52%
Filter sludge	62%	57%
Resin	63%	62%

Table 4.1-3 Percentage of Waste in Solidified Product for PWRs

Plant	Solidification Agent	Liquid Waste	Precoat Type Filter	Cartridge or Bag Filters	Deep Bed Resin
With Condensate Polishing Systems					
P1	(1)	NCL[2]	(3)	NA[4]	(5)
P2	1973 to present, cement and vermiculite	33	(3)	NA	(5)
P3	Vermiculite (until 1977)	55	(3)	(6)	(5)
	Cement (1977 to present)	68	(3)	(6)	(5)
P4	Cement (until 1977)		(3)	NA	NA
	UF (1977 to present)	60	(3)	NA	60
P5	Cement	36	(3)	32	(4)
P6	Plaster of paris (until 1977)	NA	(3)	NA	NA
	Cement and vermiculite (1977 to present)	33	(3)	NA	(5)
P7	Cement	50[7]	(3)	68	67
P8	UF	60	(3)	NA	(8)
P9	Cement	45	(3)	66	55
P10	Cement	58	NA	(9)	(5)
Without Condensate Polishing Systems					
P11	Cement and vermiculite	55	(3)	26	(5)
P12	Cement[10]	NCL	NA	NA	NA
P13	UF[10]	NCL	(3)	NA	NA
P14	UF	75	(3)	NA	(5)
P15	UF	66	-	NA	(5)
P16	UF[11]	30-40	NA	NA	(5)
P17	UF	70	NA	NA	(5)
P18	UF	60	60	20	(9)
% of Plants Solidifying Waste Type		100	-	100	22

1. No solidification system installed.
2. No concentration of liquid wastes.
3. No precoat type filters.
4. Not available.
5. Resin dewatered only.
6. Cartridge filters not encapsulated.
7. Stopped using evaporator in 1976.
8. Plant has never shipped resin.
9. Shipped in cemented evaporator bottoms.
10. Installed systems not used.
11. By outside contractor.

Average % Waste in Solidified Product

Waste Type	UF Systems	Cement Systems
Liquid waste	55%	48%
Filter sludge	60%	60%
Resin	60%	59%
Cartridge filters	20%	52%

Table 4.2-1 BWR Deep Bed Resin Annual Wastes

Plant	1971 Volume (ft^3)	1971 Activity (Ci)	1972 Volume (ft^3)	1972 Activity (Ci)	1973 Volume (ft^3)	1973 Activity (Ci)	1974 Volume (ft^3)	1974 Activity (Ci)	1975 Volume (ft^3)	1975 Activity (Ci)	1976 Volume (ft^3)	1976 Activity (Ci)	1977 Volume (ft^3)	1977 Activity (Ci)
Deep Bed Condensate Polishing System														
B1					1,026.	9.06	0	0	5,551.	1,467.	8,975.	2,626.	5,337.	2,098.
B2	NA[1]		NA		NA		NA		NA		NA		1,767.	1,038.
B3	NA		NA		NA		NA		490.	253.	989.	933.	743.	536.
B4									NA		1,020.	NA	9,420.	NA
B5	NA		NA		NA		NA		NA		NA		NA	
B6									190.	<5.	0.0	0.0	0.0	0.0
Precoat Filter Condensate Polishing System														
B7									NA		NA		NA	
B8			NA		105.	NA	105.	NA	105.	NA	105.	NA	105.	NA
B9					NA		NA		NA		NA		NA	
B10	NA		NA		NA		NA		NA		NA		NA	
B11							NA		NA		NA		NA	
B12							188.	1.46	318.	1.86	78.	0.11	71.5	0.61

1. Not available.

Table 4.2-2 Radionuclides Present in BWR Spent Resins

Plant	Cr-51	Mn-54	Fe-59	Co-58	Co-60	Zn-65	Sr-91	Zr-95	Nb-95	Tc-99m	Ag-110m	Cs-134	Cs-137	Ce-141	I-131	Ce-144	Ba-140	La-140	I-133	Sb-124
Deep Bed Condensate Polishing System																				
B1	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	
B2		X		X	X							X	X							
B3		X			X							X	X							
B4	X	X		X	X	X														
B5		X			X							X	X							
B6	X	X	X	X	X	X						X	X							X
Plants With Precoat Filter Condensate Polishing System																				
B7	X			X	X	X							X							
B8					X	X							X	X	X		X	X		
B9		X			X							X	X							
B10	X	X		X	X	X			X			X	X	X						
B11	X	X	X	X	X	X							X	X						
B12	X	X	X	X	X	X		X	X		X	X	X		X					

one of the facilities, 7 were listed by more than half of the plants. These seven are Cr-51, Mn-54, Co-58, Co-60, Zn-65, Cs-134, and Cs-137.

The remaining data on BWR spent resins are given in Table 4.2-3. The various resins used in BWRs are identified and discussed in detail in Section 2.2.1. The average density of the solidified waste product is 1.39 g/cc (87 lb/ft^3). The density of unsolidified resin is 0.81 g/cc (50 lb/ft^3). The density reported by B3 is not used because it includes the weight of the 80-ft^3 liner and shielding. Solidification of the resins increases the density by 72%.

The most common size of container used for shipping resins is the standard 55-gallon drum. Other container sizes range from 75 ft^3 to 300 ft^3. There is no direct correlation between container size and the amount of free volume at the top of the container, called percent void. Two plants reported the percent void to be zero, one plant reported a range of 2 to 20 with most falling into the range of less than 1 to 12.

Personnel from 7 of the 12 facilities reported that they shipped spent resin unshielded. Representatives of the remaining five plants listed the casks which they use, and the list is in Table 4.2-3. A separate discussion of shipping casks is provided in Chapter 3.

In order to allow the comparison of dose rates from various process waste types, doses are tabulated separately at the end of Section 4.2.1.4 in Table 4.2-15. Contact doses range from 10 mr/hr to 40 R/hr for unshielded wastes with a majority of the doses between 5 R/hr to 40 R/hr. When shielded (as noted in Table 4.2-3), the dose rates drop to between 8 mr/hr and 300 mr/hr. At 3 feet the doses for unshielded wastes drop to between 0.5 mr/hr and 5 R/hr. At 3 feet only one plant reported a dose rate over 300 mr/hr. The shielded dose rates range between 2 mr/hr and 30 mr/hr.

4.2.1.2 BWR Concentrated Liquids

The data collected for plants with a deep bed condensate polishing system represent 9 plant-years of operating data ranging from 100 ft^3/yr to 24,900 ft^3/yr and 0.001 Ci/yr to 470 Ci/yr. The average annual volume of waste shipped offsite is approximately 9,100 ft^3 containing 267 curies. When weighted in terms of each plant's rated electrical capacity the average annual volume becomes 12.7 ft^3/MWe-yr. The average activity is 0.58 Ci/MWe-yr based on a concentration of 0.046 Ci/ft^3. Annual volumes and activities are given in Table 4.2-4.

Two of the 9 plant-years of data account for 54% of the total waste volume reported but they account for only 22% of the activity. Thus, while the activity associated with these two plant-years of data are not disproportionate, the volumes are. One of

Table 4.2-3 Characteristics of BWR Spent Resin Wastes

Plant	Types of Resin Used	Solidified	Density gm/cc	Container Size, ft^3 7.35	75	80	170	182	195	216	300	%Void at Top	Shield Material
Deep Bed Condensate Polishing System													
B1	Unidentified[1]	No	.84-.96		X			X		X		0	(2)
B2	Unidentified	No	.65	X								0	None
B3	2:1 cation (R&H 200c) to anion (R&H 900c)[3]	No	8.9[4]	X		X			X			<1	HN-100; 150, 60,200
B4	Epicore cation HCRW-2, anion AP-100	No	1.0	X							X	10	21-300 cask[5]
B5	Unidentified	Yes	1.5	X								5	None
B6	Unidentified	No	[6]						X			<1	None
Precoat Filter Condensate Polishing System													
B7	Epicore APCW-21 mixed bed	No	0.8-1.1	X								<1	.5-in. lead[7]
B8	Rohm & Haas IRN-150 mixed bed	No					X					12	2-in. lead[8]
B9	No data available	Yes	1.25	X								5	None
B10	(9)	Yes	1.5						X			<5	(10)
B11	Duolite: C-20H cation; GPA-316 anion	No	0.5-0.6	X								<5	None
B12	Rohm & Haas IRN-150	Yes	~1.3	X								2-20	None

1. Over 2 years: 90% condensate; 7% radwaste; 3% fuel pit.
2. Container size selected based on specific activity of waste such that shielding is not required.
3. 33% condensate polishing, 33% high purity radwaste; 33% reactor coolant.
4. Includes 80 ft^3 liner and shield (HN-200).
5. Chem Nuclear Cask.
6. Unobtainable.
7. Also Chem Nuclear Cask 15-160-B.
8. Hittman HN-100 Cask.
9. Amberlite (R&H) IRN-78 & Graver NR-1 (Anion), Amberlite IRN-77 & Graver NR-2 (Cation).
10. Chem Nuclear Cask CNSI-195-14 with liner.

Table 4.2-4 BWR Concentrated Liquid Wastes

Plant	1971 Volume (ft^3)	1971 Activity (Ci)	1972 Volume (ft^3)	1972 Activity (Ci)	1973 Volume (ft^3)	1973 Activity (Ci)	1974 Volume (ft^3)	1974 Activity (Ci)	1975 Volume (ft^3)	1975 Activity (Ci)	1976 Volume (ft^3)	1976 Activity (Ci)	1977 Volume (ft^3)	1977 Activity (Ci)
Deep Bed Condensate Polishing System														
B1					NCL[1]		NCL		NCL		100.2	1E-3	NCL	
B2		NA[2]	NA		NA		NA		NA		NA		22,800	412
B3		NA	NA		NA		3,260	287	3,906	470	7,224	660	7,338	348
B4									NA		NA		NA	
B5		NA	NA		NA		NA		NA		NA		NA	
B6									5,328	112.4	24,900	172	12,730	211.5
Precoat Filter Condensate Polishing System														
B7									NCL		NCL		NCL	
B8			NCL		NCL		NCL		NCL		NCL		NCL	
B9					NCL		NCL		NCL		NCL		NCL	
B10		NA	NA		NA		640	34	NA		NA		NA	
B11							NCL		NCL		NCL		NCL	
B12							NCL		NCL		NCL		NCL	

1. No concentrated liquids.
2. Not available.

two data points is the only usable data from facility B2 resulting from its seventh year of operation. Plant records prior to this are not broken down by waste type. All process waste (evaporator bottoms, resin, and filter sludge) are lumped together. The other data point is 1 to 3 plant-years of data from plant B6 from its second year of operation. The first year's volume is a factor of 5 lower than that and the third year's volume is a factor of 2 lower than the second year's volume.

Because of the disproportionate weight these two volume data points have, and with no supporting information that they are typical, they have not been included in defining the typical plant. The activity associated with these 2 plant-years of data is also dropped in computing the average radioactive concentration.

The typical plant generates 8.1 ft^3/MWe-yr and 0.45 Ci/MWe-yr with a concentration of 0.055 Ci/ft^3. Only one plant that uses precoat filters for condensate polishing has consistently generated concentrated liquids that were subsequently solidified and shipped offsite. Even for this plant there is only 1 year for which specific data on the volume and activity of the waste were available. Assuming that these values are typical of the other years of operation the estimated annual waste volume prorated to plant size is 0.6 ft^3/MWe-yr at an average concentration of 0.026 Ci/ft^3 for an annual generation of .016 Ci/MWe-yr. These values have been reduced by a factor of 2 based on an estimate that half of all future plants using precoat filters for condensate cleanup will have evaporators for the treatment of decontamination fluids and other chemical wastes. With the limited data available the estimates above are used for both the average plant and the typical plant.

Data on the radionuclides in concentrated liquids are given in Table 4.2-5. To a large extent the same nuclides are present in concentrated liquids that are present in BWR spent resin. The predominant nuclides are Mn-54, Co-58, Co-60, Cs-134, and Cs-137. A detailed summary by facility B3 showed that 47.7% of the activity in one particular sample was Cs-137 and 40.2% was Cs-134.

The physical and chemical characteristics of concentrated liquids are given in Table 4.2-6.

In all the plants for which data were available, the major contaminants concentrated were cleaning solutions, chemicals from water chemistry control, and antifoam compounds. Plants B4 and B5 reported sulfuric acid (H_2SO_4) and sodium hydroxide (NaOH). These are the primary chemicals used in the regeneration of deep bed demineralizer resins. These chemicals are used in many plants. When they are combined in chemical waste tanks they form sodium sulfate and water. Thus, while some excess sodium hydroxide or sulfuric acid is present (in solution), it is the sodium sulfate that is concentrated in the evaporator as an undissolved solid.

Table 4.2-5 Radionuclides Present in BWR Concentrated Liquid Wastes

Plant	Na-24	Cr-51	Mn-54	Fe-59	Co-58	Co-60	Zn-65	Sr-89	Sr-90	Nb-95	Zr-95	I-131	Sb-124	Cs-134	Cs-137	La-140	Ce-141	W-187
Deep Bed Condensate Polishing System																		
B1						X												
B2			X		X	X								X	X			
B3[1]	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X
B4		X	X		X	X	X											
B5			X			X								X	X			
B6		X	X	X	X	X	X						X	X	X			
Precoat Filter Condensate Polishing System																		
B7	NCL[2]																	
B8	NCL																	
B9	NCL																	
B10			X			X				X				X	X			
B11	NCL																	
B12	NCL																	

1. 47.7% Cs-137, 40.2% Cs-134.
2. NCL - No concentrated liquids.

Table 4.2-6 Characteristics of BWR Concentrated Liquid Wastes

Plant	Significant Chemicals in Waste	pH	Source or Means of Production	Weight % Solids	Solidified	Density gm/cc	% Void at Top	Container Size, ft^3 7.35	85	130	200	195	300	Shielding
Deep Bed Condensate Polishing System														
B1	Borated water	NA[1]			Yes	4.7	0				X			None
B2	None identified	>12	Concentrator bottoms	~22	Yes	1.18-1.25[2]	0		X		X		X	Atcor casks
B3	Unknown[3]	6-9	Concentrator bottoms	20-50	Yes	NA[1]	< 1		X	X		X	X	Steel cask
B4	H_2SO_4, NaOH, Betz #1185 polymer[4], solution A[5], PO_4[6], antifoam compound	4.5-6.0	Concentrator bottoms	~7	Yes	~.95[2]	~ 5	X					X	Chem Nuclear 21-300 Cask
B5	H_2SO_4, NaOH, PO_4 (Amway) Antifoam compound (Dow Corning)	9	Concentrator bottoms	25-30	Yes	1.5	~ 5	X						None
B6	None reported	8-10	Concentrator bottoms	25	Yes	1.33	< 1						X	Steel and concrete
Precoat Filter Condensate Polishing System														
B7	NCL[7]													
B8	NCL													
B9	NCL													
B10	Cleaning solutions, water chemistry chemicals	8-10	Concentrator bottoms	25	Yes	1.53	2	X						None
B11	NCL													
B12	NCL													

1. Not available.
2. Unsolidified.
3. Evaporator used to concentrate laundry waste and floor drain waste water.
4. Aids settling of colloidal Fe in phase separators.
5. Filter Precoat Coagulant (Graver & Epicor).
6. Tri- and disodium phosphate.
7. No concentrated liquids.

The reported pH of the concentrated liquids ranges from 4.5 to 12 or greater with most plants falling in a range of 8 to 10. The plant with the lowest reported pH (4.5-6.0) also reported low weight percent solids in the concentrate (approximately 7 weight percent). The remaining plants report concentrate running mostly between 20 to 30 weight percent. Plant B3 reports obtaining concentrate as high as 50 weight percent. An average for standard evaporators of 25 weight percent is typical with crystallizers reaching as high as 50 weight percent.

Prior to solidification, the average density is 1.2 g/cc (75 lb/ft^3) increasing to 1.45 g/cc (90 lb/ft^3) when solidified. The percent void at the top of the containers averaged only 2% based on a range of 0% to 5%. Container sizes range from 7.35 ft^3 (55-gallon drum) to 300 ft^3 steel liners with larger containers appearing to be used more often than drums. The larger containers are shipped in their own cask or shield and drums are shipped unshielded (in the case of B10) or shipped several at a time in a special shield pack.

As stated previously only one BWR using precoat filters in the condensate system regularly ships solidified concentrated liquids. These wastes are low in activity and result in an unshielded contact dose rate of 20 to 50 mr/hr. At 3 feet the dose rates were still reported to be between 30 to 50 mr/hr. Of the six plants with deep bed demineralizers in their condensate systems, one does not regenerate its resin. Officials at that plant report an unshielded dose rate from concentrated liquids of 1 mr/hr with a 3-foot dose rate of 0.5 mr/hr. The 5 remaining plants regenerate their resins and report an unshielded contact dose rate of 35 R/hr and shielded contact dose rates ranging from 1 mr/hr to 200 mr/hr. At 3 feet these drop to 350 mr/hr and from <1 mr/hr to 30 mr/hr respectively. The dose rates for individual plants are tabulated in Table 4.2-15.

4.2.1.3 BWR Filter/Demineralizer Sludge and Filter Precoat

Data on filter precoat and sludge for plants using deep bed condersate polishing systems, given in Table 4.2-7, cover 14 plant-years for volumes and 12 plant-years for activity. Volumes range from 412 ft^3/yr to 7,460 ft^3/yr. Activities range from 4.2 Ci/yr to 2,540 Ci/yr. The annual waste generation rate is 3,650 ft^3/yr containing approximately 1,350 curies. When weighted to consider plant size the generation rate is 5.4 ft^3/MWe-yr at 2.0 Ci/MWe-yr. The average concentration is 0.37 Ci/ft^3. As with spent resin, the data for filter precoat and sludge do not include any unusually high values indicative of abnormal plant operations. Thus, the results given in this section are used for both the average plant and the typical plant.

For plants using precoat filters in their condensate systems, 8 plant-years of data, also given in Table 4.2-7, on annual waste volume are available ranging from 3,651 ft^3/yr to 5,313 ft^3/yr. Data on total radioactivity were available from facility B12 only,

Table 4.2-7 BWR Filter Sludge Volume and Activity

Plant	1971 Volume (ft^3)	1971 Activity (Ci)	1972 Volume (ft^3)	1972 Activity (Ci)	1973 Volume (ft^3)	1973 Activity (Ci)	1974 Volume (ft^3)	1974 Activity (Ci)	1975 Volume (ft^3)	1975 Activity (Ci)	1976 Volume (ft^3)	1976 Activity (Ci)	1977 Volume (ft^3)	1977 Activity (Ci)
Deep Bed Condensate Polishing System														
B1					4,224	494	7,458	1,503	4,390	2,261	7,015	2,540	6,660	1,812
B2	NA[1]		NA		NA		NA		NA		NA		750	66
B3	NA		NA		NA		NA		1,876	2,497	2,320	912	412	690
B4									NA		5,400	NA	2,250	NA
B5	NA		NA		NA		NA		NA		NA		NA	
B6									970	4.2	3,586	165	4,225	1,098
Precoat Filter Condensate Polishing System														
B7									NA		NA		NA	
B8			NA		5,145	NA	4,454	NA	4,185	NA	3,779	NA	4,605	NA
B9					NA		NA		NA		NA		NA	
B10	NA		NA		NA		NA		NA		NA		NA	
B11							NA		NA		NA		NA	
B12							0	0	4,900	264	5,313	319	3,651	280

1. Not available.

ranging from 264 Ci/yr to 319 Ci/yr. Based on plant size, the average generation rates are an estimated 7.7 ft^3/MWe-yr and 0.50 Ci/MWe-yr based on an average concentration of 0.065 Ci/ft^3. Again there is no specific datum which reflects high unusual operating circumstances. Therefore, the average values are also used for the typical plant.

The list of radionuclides present in precoat filter wastes is essentially the same as the list mentioned previously for resin and concentrated liquids. The most commonly listed isotopes are Mn-54, Co-58, Co-60, Zn-65, Cs-134, and Cs-137. Representatives at plant B1 provided two separate listings of radionuclides: one for radwaste filter sludge, and one for the filter/demineralizer in the reactor water cleanup system. The complete listing is given in Table 4.2-8.

The various precoat materials used are listed in Tables 4.2-9 and 4.2-10 with a general discussion of these materials found in Section 2.2.2. Table 4.2-10 was added because of the details these plants gave regarding the material used in various systems.

Filter wastes that are not solidified have an average density of 0.86 g/cc (54 lb/ft^3) and can be as high as 1.69 g/cc (100 lb/ft^3) when solidified in cement. The average density of solidified filter/sludge is 1.5 g/cc.

Nine of the 12 plants surveyed use 55-gallon drums for waste shipment. Other containers range from 30-ft^3 solidpacks to 300-ft^3 liners. The various container sizes used by each plant are shown in Table 4.2-9. Officials at most plants report that the waste containers were shipped with less than a 5% void. Employees at two plants reported filling the containers 90%, for a 10% void, and one plant reported a void of 33%.

The majority of plants do not provide separate shielding for the waste containers, especially for 55-gallon drums. Table 4.2-9 is a listing of various shield casks used for larger containers. Section 3.7 provides a general description of individual shielding casks.

For those containers that are shielded, the contact dose rates range from 3 to 300 mr/hr. When these wastes are unshielded the dose rates are predominantly between 1 R/hr and 20 R/hr but may also be as low as 2 mr/hr. At 3 feet the shielded dose rates range from 2 mr/hr to 30 mr/hr, but the unshielded dose rates cover a range of 1 mr/hr to 2.5 R/hr primarily grouped between 50 mr/hr to 500 mr/hr. Table 4.2-15 provides a listing of dose rates for each of the 12 facilities surveyed.

4.2.1.4 BWR Cartridge Filters

Ten of the 12 plants in the survey use cartridge filters in either their fuel pool cleanup systems, control rod drive system, laundry system, or radwaste system. Because these systems are not directly affected by the type of condensate polishing system used, this

Table 4.2-8 Radionuclides Present in BWR Precoat Filter Wastes

Plant	Cr-51	Mn-54	Fe-59	Co-58	Co-60	Zn-65	Zr-95	Nb-95	Tc-99m	Ru-103	Ru-106	Ag-110m	I-131	I-133	Cs-134	Cs-136	Cs-137	Ba-140	La-140	Ce-141	Ce-144	Hg-203	Bi-207	Np-239	Sb-124
B1[1]	X	X	X	X	X	X		X				X		X		X		X	X	X	X	X			
B1[2]	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X
B2		X		X	X											X		X							
B3		X			X											X		X							
B4	X	X	X	X	X	X																			
B5		X			X											X		X							
B6	X	X	X	X	X	X							X			X		X							
B7	X			X	X	X												X							
B8					X	X								X				X	X	X	X				
B9		X			X	X										X		X							
B10	X	X		X	X	X										X		X			X	X			
B11	X	X	X	X	X	X		X				X				X		X							
B12	X	X	X	X	X	X	X	X				X		X		X		X							

1. Reactor water cleanup system filter/demineralizer wastes.
2. Radwaste system precoat filter.

Table 4.2-9 Characteristics of BWR Precoat Filter Wastes

Plant	Type of Precoat Material	Waste Solidified	Density g/cm^3	Container Size, ft^3 7.35	30	80	170	182	216	300	% Void at Top	Shield Material
B1[1]	Powdex[1,2]	No	0.615-1.07		X						0	Concrete
B1[3]	Diatomaceous earth[4]	No	0.84-1.03					X	X		0	None
B2	Diatomaceous earth	No	0.588	X							0	None
B3	Solka Floc; Selite and crushed resin	Yes (UF)	(5)	X							<1	None
B4	Ecodex, Ecocote (Graver)[6]	No	~0.95	X						X	10	21-300 Cask
B5	Diatomaceous earth, Powdex	Yes (C)	0.95	X							≤5	None
B6	Powdex, Solka Floc[4]	No	0.74							X	<1	(7)
B7	See Table 4.2-10.	No	0.8-1.1	X							<1	None
B8	See Table 4.2-10.	No		X		X	X				10-12	HN-100 HN-200
B9	Ecodex and Ecocote, Etched Disk[8,4]	Yes (C)	0.95	X							≤5	None
B10	See Table 4.2-10.	Yes (C)	1.4-1.6	X							<5	None
B11	See Table 4.2-10.	No	0.8-0.9	X							<5	None
B12	Powdex, Ecocote, Ecodex	Yes (C)	~0.9	X							33	None

1. Reactor water cleanup system filter/demineralizer waste.
2. Shipped in Solidpaks, 75 ft^2 in 2 years.
3. Radwaste filter waste.
4. 13,600 ft^3 in 2 years.
5. Not available.
6. Chem Nuclear cask, Polymer-Betz solution A found in waste.
7. Cask used, but no specific information on size or material.
8. No precoat.

Table 4.2-10 Precoat Material Used in BWR Precoat Filters

Plant	Radwaste System	Reactor Water Cleanup System	Condensate Polishing System	Fuel Pool Cleanup System
Plant B7				
Graver Powdex PAO-anion and PCH cation			X	
Epicor PD-1 anion		X	X	X
Epicor PD-3 cation		X	X	X
Plant B8 (0.2#/ft^2 surface area)				
Ecodex	11 lb			
Graver PAO-anion and PCH cation (1:1)		11 lb	92 lb	53 lb
Ecocote			92 lb	
Plant B10				
Ecodex	X			X
Ecocote (used as overlay)			X	
Solka Floc		X		
Graver Powdex 2:1 cation to anion		X	X	
Plant B11				
Epicore Epifloc or Ecodex	X			
Epicore Powdex - anion (PD-1), cation (PD-3)	X	X	X	X
Solka Floc (BW-40)				X

distinction is dropped. The discussions in this section center on data from 8 of the 10 plants shown in Table 4.2-11. Officials at the other two plants were unable to provide specific data because one has not yet shipped any cartridge filters and the other ships its cartridge filters with its trash and separate records for the two are not kept.

Based on data supplied by employees from five of the facilities, the annual contingency of filter cartridges shipped offsite is 25 to 50 standard 55-gallon drums, approximately 150 to 370 ft^3/yr. Employees from only three of these plants ship cartridge filters separately. At plants B1 and B6, cartridge filters are shipped with compactible or noncompactible trash. When the data are divided by the plant size and averaged out for all the plants, the resulting generation rate is less than 0.09 ft^3/MWe-yr. Based on such a low generation rate, and on the fact that some data on cartridge filters have already been considered (since they are included in the data on compactible and noncompactible trash), a specific contribution to the overall waste generation rate is included.

No data were collected with respect to the total activity contained in the cartridge filters either on a per cartridge basis or an annual basis. Six of the facilities had data on radionuclides present in the filters. These data are given in Table 4.2-12. As was expected, the insoluble activated corrosion products dominate the list: Cr-51, Mn-54, Fe-59, Co-58, Co-60, and Zn-65. Cs-134 and Cs-137 also were reported by most plants.

Table 4.2-11 also lists details of specific filters, applications, and containers used to ship spent filters offsite. It also lists the shielding used. Details on specific filter cartridges are given in Section 2.2.3. Cartridge filters are used by facilities B1 and B8 in the spent fuel pool cleanup system, by facilities B4, B5, and B9 for treatment of laundry waste water, by B6 and B10 for filtration of control rod drive water and by B11 as prefilters (two in series) to the filter/demineralizers in the equipment drain and floor drain systems. Micron ratings range from 1 to 100 microns for the prefilter application at B11.

All of the plants surveyed use 55-gallon drums for shipping filter cartridges. The only exception to this is one plant which encapsulates 10 fuel pool cleanup system cartridges in an 8-inch-diameter concrete stove pipe drum. The remaining plants report placing between 40 and 100 cartridges in a single 55-gallon drum.

Dose rates based on the application of the filter are given in Table 4.2-14. Both the contact dose rates and the dose rates at 3 feet are given. The highest dose rates are from the spent fuel pool cleanup filters with contact dose rates of 40 R/hr at plant B1 and ranging from 2 to 3 R/hr at plant B8. Plant B1 gave a 3-foot dose rate of 4 R/hr, one-tenth of the contact dose. Laundry filters are reported to result in contact dose rates ranging from 50 mr/hr to 15 R/hr with dose rates at 3 feet, lower by a factor

Table 4.2-11 Characteristics of BWR Cartridge Filter Wastes

Plant	Filter Type, Manufacturer Size, Rating	System	Density gm/cc	Number of Elements per Filter	Number of Filters Shipped (Annual/Avg)	Number of Elements per Container	Shipping Containers ft^3 7.35	Integral Shielding
Deep Bed Condensate System	Polishing							
B1	Not identified	Fuel pool	0.3	--	1[1]	--	X	None
B2					[2]			
B3	[3]							
B4[4]	50 micron cotton cellulose	Detergent w/PO_4	1	8	416	65-70	X	None
B5	Cuno microklean II, 1 micron	Laundry	1	12	486	50	X	50%[5]
B6	Cuno cloth, 1 ft x 5 ft, 50 micron[6]	Control rod drive	0.22	20	50	40	X	None
Precoat Filter Condensate Polishing System								
B7[7]								
B8[8]								
B9	Clarite Model IL-36-135 Cuno AMF #CG4DB2 51040-03	Laundry[9] Chemical drain	0.85	41	-	35-45	X	None
B10	Cuno wound cotton 5 micron	Control rod drive	.435	50	50	100	X	
B11	See Table 4.2-13	Table 4.2-13	0.3-0.4		24	81	X	
B12	[10]							

1. Normally shipped with trash data for 1 filter 1976 only.
2. Shipped with trash, no records.
3. None used.
4. 8 Filter replacements per week.
5. Concrete shields
6. Mixed with compactible waste.
7. None used.
8. 120 3-in. ϕ x 10-in. control rod drive suction filter cartridges per yr, low activity level shipped w/compacted trash; 20 3-in. ϕ x 10-in. fuel pump skimmer filters per yr. 2 8-in. concrete stove pipe drums per yr.
9. Both solidified; both 5 micron.
10. Cartridge filters in radwaste system; none ever shipped.

Table 4.2-12 Radionuclides Present on BWR Cartridge Filters

Plant	Cr-51	Mn-54	Fe-59	Co-58	Co-60	Zn-65	Zr-95	Nb-95	Sb-124	Cs-134	Cs-137	MFP[1]	MCP[2]
Deep Bed Condensate Polishing System													
B1					X						X		
B2	(3)												
B3	(4)												
B4	X	X	X	X	X	X							
B5		X			X					X	X		
B6	X	X	X	X	X	X			X	X	X		
Precoat Filter Condensate Polishing System													
B7	(4)												
B8	(3)												
B9		X			X					X	X		
B10												X	X
B11	X	X	X	X	X	X	X	X		X	X		
B12	(3)												

1. Mixed fission products.
2. Mixed corrosion products.
3. Not available.
4. None used.

Table 4.2-13 Cartridge Filter Applications in Plant B11

Radwaste System Surge Tank: 2 Cuno Filters in series

- First filter: 81 elements/filter
  25, 50, or 100 micron
  changed 12 times/hr
- Second filter: 81 elements/filter
  3 or 5 micron
  changed 6 times/yr

Chemical Waste System: turbine and radwaste building floor drains and chem lab sinks

- 81 elements/filter
- 5 or 25 micron
- changed 3 times in 2 years (no longer used)

Portable Filter: used to clean up turbine lube oil

- 1 filter with 27 elements
- changed 6 times/yr

Table 4.2-14 BWR Cartridge Filter Dose Rates

Process System	Plant	Dose Rates	
		Contact	3 feet
Spent fuel pool cleanup	B1	40 R/hr	4 R/hr
	B8	2-3 R/hr	-
Laundry	B4	50-100 mr/hr	5-10 mr/hr
	B5	5-10 R/hr	50-80 mr/hr
	B9	10-15 R/hr	100-150 mr/hr
Control rod drive	B6	100 mr/hr	10 mr/hr
	B10	<5 mr/hr	<1 mr/hr
Radwaste	B11	1 mr/hr-2 R/hr	1-200 mr/hr

of 10. Contact dose rates from control rod drive filters are between 5 and 100 mr/hr, and from 1 to 10 mr/hr at 3 feet. Radwaste filter dose rates are 1 mr/hr to 2 R/hr on contact, and 1 mr/hr to 200 mr/hr at 3 feet. (See Table 4.2-15.)

4.2.1.5 BWR Compactible and Noncompactible Trash

NUS has endeavored to use all of the data collected in calculating average energy-specific annual waste volumes. By not excluding data points that have been influenced by abnormal plant occurrences or operations, the effects of these situations are factored into the projections in Chapter 5. Unfortunately, when evaluating the data on compactible and noncompactible trash a few data points were found to have an overwhelming effect on annual averages. In cases with values resulting from problems that are not likely to occur again[1] at any facility, the data were disregarded. In such cases the reason for excluding the data is explained.

Compactible and noncompactible radwaste is not a process waste. In many cases it is a result of maintenance work and facility practices that enhance or minimize the production of compactible and noncompactible radwaste. This waste will have a certain baseline source consisting of consumable material such as lab equipment, plastic shoe covers, step-off pads, cotton gloves, system components with a limited useful life, mop heads and blotter paper used to clean up spills, and anti-contaminant clothing damaged or contaminated to the point that it is not useful. These sources of compactible and noncompactible radwaste are augmented by periodic large maintenance and/or backfit jobs or large spills. Since these occur randomly throughout the operating lifetime of the station, a true pattern to compactible and noncompactible radwaste generation rates should not be expected.

A similar situation exists regarding the total activity associated with compactible and noncompactible radwaste. Therefore, it is futile to attempt to find a pattern with respect to compactible and noncompactible radwaste radioactivity inventory.

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1. While these problems are not likely to occur the possibility of recurrences at the same plant or other plants does exist. However, as additional plants become operational the quantity of waste resulting from such occurrences becomes less and less significant in comparison to the total quantity of waste from all plants. While operating problems, and their resulting high-waste volumes, should be considered during plant design they are by no means representative of typical or average plant operation.

Table 4.2-15 Contact and 3-Foot Dose Rate From BWR Wastes (mrem/hr)

Plant	Concentrated Liquid Waste		Resin		Filter Sludge		Cartridge Filters	
	Contact	3 ft	Contact	3 ft	Contact	3 ft	Contact	3ft
Deep Bed Condensate Polishing System								
B1	1	0.5	10mr/hr-40 R/hr	.5-100	10-25	5-10	40 R/hr	4 R/hr
B2	22	6	40 R/hr	5 R/hr	3.8 R/hr	580	-	-
B3	<1-20[1]	<1-10[1]	~10[1]	~2[1]	3-150[2]	<1-10[2]	-	-
B4	~200	~10-30	100-300	30	100-300	10-30	50-100	5-10
B5	~35 R/hr	350	~35 R/hr	300	(10-15) R/hr	150-300	(5-10) R/hr	50-80
B6	<1-60[3]	<1-10[3]	NA[4]	NA	(1-2) R/hr	50-250	100	10
Precoat Filter Condensate Polishing System								
B7	-	-	(0-15)R/hr	0-500	(0-15) R/hr	0-500	-	-
B8	-	-	8-10[5]	2-4[5]	8-10[5]	2-4[5]	(2-3) R/hr[6]	-
B9	-	-	(5-10) R/hr[7]	50-100[8]	(5-10) R/hr[7]	50-100[8]	(10-15) R/hr	100-150
B10	20-50	30-50	100[9]	10-15[9]	(1-5) R/hr	100-300	<5	<1
B11	-	-	100-150	10-20	(1-2) R/hr	100-300	1mr/hr-2R/hr[10]	<1-200[10]
B12	-	-	20-300	NA	20-100	10		

1. On contact and 3 ft from shielding. Various Hittman casks.
2. On contact and 3 ft from shielding. Various Chem Nuclear casks.
3. Shielded.
4. Not available.
5. Outside HN-100 cask.
6. Contact dose to cartridge.
7. (5-10) R/hr normal, maximum 30-35 R/hr.
8. 50-100 mr/hr normal, maximum 300-350 mr/hr.
9. Measured on contact to, and 3 ft from, CNSI-195-14 cask.
10. Radwaste filters.

4.2.1.5.1 Composition of Compactible and Noncompactible Radwaste

It is generally thought that compactible radwaste consists of paper, plastic, and cloth, and that noncompactible radwaste consists of equipment parts, piping, miscellaneous wood, and miscellaneous metal pieces. This survey confirms that these materials are the basic components of compactible and noncompactible radwaste.

On an individual plant basis the survey typically provided little information beyond that which was already known. However, when compiling all the data, a reasonable breakdown of the compactible and noncompactible radwaste forms was obtained and is listed in Table 4.2-16.

Basically there is little difference between PWR and BWR compactible and noncompactible radwaste forms. (Table 4.2-39 lists PWR waste composition.) The only significant difference is that BWR noncompactible radwaste includes more reactor internal components; for instance, fuel channels that are not a component of PWR reactor internals.

4.2.1.5.2 Volumes of Compactible and Noncompactible Radwaste

Although the composition of compactible and noncompactible radwaste forms can be reasonably defined, an annual average volume of both waste types cannot be as well defined for the following reasons.

1. Records do not identify the volume shipped as compactible and noncompactible.

2. Valves and small pieces of metal and wood are placed in 55-gallon drums with compactible waste, or bags of compactible waste are placed in larger containers of noncompactible waste.

3. One utility cuts up its noncompactible waste so that it can be shipped with its compactible waste. This utility reports no noncompactible waste shipments.

Volumes reported as noncompactible radwaste usually represent waste shipped in containers other than 55-gallon drums, and do not reflect the total volume of noncompactible radwaste.

4.2.1.5.3 Evaluation of Data Collected on BWR Compactible and Noncompactible Radwaste Volumes

Data collected in the annual volumes of compactible and noncompactible radwaste shipped from BWR nuclear power generating facilities surveyed for this study are presented in Table 4.2-17.

Table 4.2-16 Material Shipped as BWR Compactible and Noncompactible Radwaste

Radwaste Material	B1		B2[3]		B3		B4		B5		B6		B7		B8		B9		B10		B11		B12	
	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N
Anti-Contaminant Clothing	X				X						X		X		X				X		X		X	
Cloth					X		X		X															
Rags											X		X		X				X		X			
Cotton Gloves															X				X					
Dirt									X			X												
Drilling Filters																								
Filters											X													
HEPA		X														X				X				
Movable																X								
Respirator Cartridges															X									
Filter Cartridges									X						X									
Flow Channels																								X
Fuel Channels (may be crushed or cut up)		X								X		X												X
Laboratory Trash																							X	
Ladders																X								
Light Bulbs																		X						
Local Power Range Monitors																								X
Miscellaneous Metal		X							X	X														
Pipes						X						X		X		X				X				
Fittings						X																		
Equipment								X																
Components						X		X						X		X				X		X		
Hand Tools									X											X				
Non-aerosol																	X							
Aerosol																		X						
Crushed 55-gal Drums																X		X						
Valves												X				X								
Miscellaneous Wood		X				X		X		X		X		X		X	X	X		X		X		
Paper						X		X		X		X		X			X		X		X			
Blotter															X									
Kraft																						X		
Tissue	X																							
Towels															X									
Plastic Shoe Covers															X				X					
Poly Wastes	X						X		X				X				X							
Bags																			X		X	X		
Sheeting																								
Poison Channels										X														
Rubber					X				X															
Sample Bottles					X																			
Scaffolding								X		X								X						
Sweeping Compounds																		X						

1. Compactible.
2. Noncompactible.
3. Not available.

Table 4.2-17 BWR Compactible and Noncompactible Trash

Plant	1971 Volume (ft^3)	1971 Activity (Ci)	1972 Volume (ft^3)	1972 Activity (Ci)	1973 Volume (ft^3)	1973 Activity (Ci)	1974 Volume (ft^3)	1974 Activity (Ci)	1975 Volume (ft^3)	1975 Activity (Ci)	1976 Volume (ft^3)	1976 Activity (Ci)	1977 Volume (ft^3)	1977 Activity (Ci)
Deep Bed Plants Polishing System														
B1					2,848	18.5	6,809	42.4	6,221	74.1	16,570	31,710	15,593	2,101
B2	NA(1)		NA		NA		NA		NA		NA		NA	
B3	10,61[illegible]	54.9	11,206	57.5	9,880	50.1	6,330	7.25	5,572	19.3	4,749	4.54	10,137	13.3
B4									0	0	19,547	19.8	22,352	20.0
B5			35,900	123	59,000	133	19,103	4,340	113,700	2,810	173,000	37.4	NA	
B6									721	0.7	5,295	8.3	20,283	42.2
Precoat Filter Condensate Polishing System														
B7									NA		3,358	2.1	8,718	5.8
B8			NA		1,339	.98	2,426	1.69	6,727	3.96	4,506	13.0	4,315	12.6
B9					NA		NA		NA		NA		NA	
B10			NA		NA		NA		NA		NA		NA	
B11							1,024	NA	2,672	79	15,232	179	12,559	251
B12									2,543	.66	2,613	.24	4,829	3,469.73

1. Not available.

The data were obtained from nine stations with 12 individual units and 39 reactor-years of operation. The evaluation was made using the total of the reported compactible and noncompactible radwaste volume.

The BWR data had a wide range in the annual production rate of compactible and noncompactible radwaste. However, the 1975 or 1976 data for facility B5 were not used for two reasons. First, using these 2 years of data would result in 44% of the total volume from only 13% of the reactor years of data. Second, these 2 years of data reflect primarily contaminated soil resulting from an accidental spill. Even though there is no guarantee that a similar situation will not occur in the future, it is doubtful that cleanup operations resulting in hundreds of thousands of cubic feet of waste will occur every 7 to 8 reactor-years of operation.

Evaluation of the data provided the following information:

1. The annual average compactible and noncompactible radwaste production rate is 9,270 ft^3.

2. The annual average compactible and noncompactible radwaste production rate based on reactor size for

   a. Plants with less than 750 MWe generating capacity (21 reactor-years) is 7,016 ft^3.

   b. Plants with a generating capacity greater than or equal to 750 MWe (18 reactor-years) is 8,487 ft^3.

   This is a greater difference between large and small facilities than the difference which occurred with PWRs, (Section 4.2.2.5); however this is still only a 21% difference. The median value for the data points 6,330 ft^3 is 83% of the average value.

As with the data for other waste forms the data for BWR compactible and noncompactible waste were analyzed to determine the quantity of waste based on the installed electrical capacity of each unit. The average generation rate is 11.5 ft^3/MWe for the average operating plant. The data were also examined to determine an appropriate value for a typical plant. In doing so all of the data from plants B5 and B4 were dropped as were the 1977 data for B6. The data from B5 represent 31% of the reactor-years considered and a disproportionate 62% of the waste volume. Adding the data for B5 to the data for B4 and B6 equals 71% of the total waste volume from all 12 plants. The data from plants B4, B5, and B6 represent 38% of the reactor-years available for analysis. The average, in terms of annual volume per megawatt, is not much different from the 11.5 ft^3/MWe for the average plant. The typical plants ship approximately 10.6 ft^3/MWe-yr, while the average annual shipment from these plants is 6,600 ft^3/yr.

The BWR data provide information from only 11 reactor-years when specific comparative data on the volume of compactible and noncompactible waste are available. These data are in Table 4.2-18. These 11 reactor-years account for 59,420 ft^3 of compactible waste and 28,240 ft^3 of noncompactible waste, or 68% and 32% respectively.

4.2.1.5.4 Radioactivity in Compactible and Noncompactible Radwaste

Table 4.2-17 gives the total activity associated with each yearly shipment of compactible and noncompactible trash. The limited data comparing the amounts of radioactivity in compactible waste compared to noncompactible waste are in Table 4.2-19. The overall concentration of radioactivity using all available data, (34 reactor-years) is 0.035 Ci/ft^3. Of these 34 reactor-years of data, 3 reactor-years of data are dominated by activity resulting from the shipment of fuel channels containing between 0.14 Ci/ft^3 to 0.72 Ci/ft^3. Without these 3 reactor-years, the remaining 31 reactor-years average 0.0048 Ci/ft^3. Based on 11 reactor-years of data, only an estimated 1.3% of the total activity is associated with compactible waste and 98.7% with noncompactible waste. Thus, the compactible trash will contain 5.2 Ci/yr at a concentration of 670 $\mu Ci/ft^3$ and the noncompactible trash will contain 397 Ci/yr at a concentration of 0.11 Ci/ft^3.

There appear to be no significant abnormalities indicative of typical plant operation. Therefore, the same activity concentrations are used for both the average plant and the typical plant.

4.2.1.5.5 Radionuclides Present in BWR Trash

Typically the radionuclides found in compactible radwaste should be a representative mix of the longer-lived radionuclides found in the reactor coolant. The radionuclide inventory in noncompactible radwaste would depend on how the material was contaminated. If the noncompactible radwaste has an activated corrosion film or was part of the reactor internals, then radionuclides of activated corrosion products should be predominant. However, if the contamination of the material was caused by contact with reactor coolant during a maintenance or backfit job then the radionuclides present should represent the mix found in the reactor coolant. Table 4.2-20 presents the data collected on the various radionuclides in compactible and noncompactible trash.

4.2.1.5.6 Containers in Which Compactible and Noncompactible Radwaste Is Shipped and Disposed

Compactible radwaste is typically packaged in 55-gallon drums (Spec 17-C or 17-H). Personnel at BWRs reported a large selection of container sizes used for compactible radwaste. Use of container sizes other than 55-gallon drums would imply that compaction is not performed. This could explain the higher volume of compactible and noncompactible radwaste shipped from BWR sites.

Table 4.2-18 Reported BWR Compactible and Noncompactible Radwaste Volumes (ft^3) Generated per Calendar year

Plant	Radwaste	1971	1972	1973	1974	1975	1976	1977
Deep Bed Condensate Polishing System								
B1	Compactible		441[1]	2,483	6,809	6,005	14,106	10,791
	Noncompactible		NA[2]	365	IWC[3]	216	2,464	4,802
B2	(4)							
B3	Compactible	10,617	11,206	9,880	6,330	5,572	4,749	4,980
	Noncompactible	IWC	IWC	IWC	IWC	IWC	IWC	5,157
B4	Compactible					0.0	19,550	22,352[5]
	Noncompactible					IWC	IWC	IWC
B5	Compactible		35,900[5]	59,000[6]	19,100[6]	113,700[6,7]	173,000[6,7]	NA
	Noncompactible		IWC	IWC	IWC	IWC	IWC	NA
B6	Compactible					603	2,625	8,168[8]
	Noncompactible					118	2,670	12,115[8]
Precoat Filter Condensate Polishing System								
B7	Compactible					NA	3,358	8,719
	Noncompactible					NA	IWC	IWC
B8	Compactible		NA	1,339	2,426	6,727	4,506	4,315
	Noncompactible		NA	(9)	(9)	(9)	(9)	(9)
B9	(10)							
B10	(10)							
B11	Compactible				(11)	(11)	(11)	14.7
	Noncompactible				1,024	2,672	15,232	12,544
B12	Compactible				NA	2,543	2,613	4,499
	Noncompactible				NA	0	0	3

NOTE: Footnotes for Tables 4.2-18 through 4.2-23 appear on the following page.

Footnotes for Tables 4.2-18 through 4.2-23.

1. Year facility went on line. Data are not for a full operating year and will not be included in this evaluation.

2. Not available.

3. Included with compactible; volume of noncompactible waste included with reported compactible volume.

4. This facility shares a site with a PWR. Radwaste from both units is shipped from the BWR unit. No attempt was made to log separately the waste shipped for each unit. Therefore, the data obtained cannot be used in any projection since they are not representative of either a BWR or PWR.

5. Two units.

6. Three units.

7. In 1975 a spill occurred requiring the removal of significant quantities of contaminated dirt from the site. Removal of dirt from the site continued through mid 1976. These data are being presented but not included in projections.

8. An extensive plant cleanup was performed in this year.

9. The noncompactible trash volume was included with the compactible trash. However, it was estimated that for any given year, the volume of noncompactible trash was 640 to 768 cubic feet.

10. This facility could only provide the total quantity of radwaste shipped annually. A breakdown by the type of radwaste was not available. The same is true of data on activity.

11. The trash compactor is rarely used. Most waste which could be compressed is shipped along with noncompactible waste. Plant records do not indicate the volume of compactible waste shipped from 1974 to 1976. Only two drums were shipped in 1977.

12. Essentially all the activity was associated with fuel channels.

13. Compactible.

14. Noncompactible.

15. Mixed fission products.

16. Mixed corrosion products.

Table 4.2-19 Reported Activity (Curies) Shipped per Calendar Year With BWR Compactible and Noncompactible Radwaste

Plant	Radwaste	1971	1972	1973	1974	1975	1976	1977
Deep Bed Condensate Polishing System								
B1	Compactible		0.131[1]	18.44	42.37	70.46	252.7	119.7
	Noncompactible		NA[2]	0.013	IWC[3]	3.68	31,457[12]	1.9
B2	(4)							
B3	Compactible	54.93	57.54	50.14	7.25	19.3	4.54	11.59
	Noncompactible	IWC	IWC	IWC	IWC	IWC	IWC	1.74
B4	Compactible					0.0	19.76	20.01[5]
	Noncompactible					IWC	IWC	IWC
B5	Compactible		119[5]	133[6]	4,340[6]	2,810[6,7]	37.4[6,7]	
	Noncompactible		IWC	IWC	IWC	IWC	IWC	
B6	Compactible					0.6	7.34	22.4[8]
	Noncompactible					0.1	1	19.8[8]
Precoat Filter Condensate Polishing System								
B7	Compactible					1.1	2.1	5.84
	Noncompactible						IWC	IWC
B8	Compactible		NA	0.98	1.69	3.96	13.03	12.59
	Noncompactible		NA	IWC	IWC	IWC	IWC	IWC
B9	(10)							
B10	(10)							
B11	Compactible				NA	(11)	(11)	(11)
	Noncompactible				NA	79.03	179.5	251.2
B12	Compactible				.00008	0.66	0.24	0.73
	Noncompactible				NA	0	0	3,46

Table 4.2-20 Radionuclides Identified as Being Present in BWR Compactible and Noncompactible Radwaste

Radionuclides	B1		B2[2]		B3		B4		B5		B6		B7		B8		B9		B10		B11		B12	
	C[13]	N[14]	C	N	C	N	C	N	C	N[2]	C	N	C	N	C	N	C	N	C	N	C	N	C	N
Cr-51							X	X			X	X	X	X									X	X
Mn-54	X	X			X	X	X	X	X		X	X					X	X					X	X
Fe-59							X	X			X	X											X	
Co-58	X	X						X			X	X	X	X		X							X	X
Co-60	X	X			X	X		X	X		X	X	X	X	X	X	X	X					X	X
Zn-65							X	X			X	X	X		X	X							X	
Zr-95																							X	X
Nb-95																							X	X
Ag-110m																							X	
Sb-124											X	X												
Sb-125																								X
I-131															X								X	
Cs-134	X	X			X	X	X		X		X						X	X					X	
Cs-137	X	X			X	X	X	X	X		X	X	X	X	X	X	X	X					X	
Ba/La-140															X									
Ce-141															X									
MFP[15]													X	X							X	X		
MCP[16]																					X	X		

Occasionally a 55-gallon drum is damaged. When this happens the 55-gallon drum is placed in an 83-gallon drum for shipment and disposal.

Noncompactible radwaste is most frequently shipped in 55-gallon drums and 4 ft x 4 ft x 8 ft or 4 ft x 4 ft x 7 ft wooden boxes. However, the range of container sizes reported as being used to ship noncompactible radwaste is extensive.

Presented in Table 4.2-21 are the containers reported to be used at BWRs for the shipment of both compactible and noncompactible radwaste.

4.2.1.5.7 Reported Density of Compactible and Noncompactible Radwaste

The reported densities of BWR compactible trash covers a wide range from a lower limit of 22 lb/ft^3 (160 pounds per 55-gallon drum) to a high of 75 lb/ft^3 (530 pounds per 55-gallon drum). There were actually five different densities reported from 40 lb/ft^3 to 75 lb/ft^3.

Data on the densities of noncompactible radwaste are somewhat misleading. The density of noncompactible radwaste is dependent on the packaging efficiency or percent void volume remaining after packaging the waste. Typically the density of noncompactible waste, after packaging, is lower than the density for compactible waste. Because of the nature of the waste the possibility exists for very high densities.

Table 4.2-22 lists the reported densities for compactible and noncompactible wastes for BWRs.

4.2.1.5.8 Radiation Levels Associated with Compactible and Noncompactible Radwaste

Data were collected on radiation levels on contact with, and at 3 feet from, containers of compactible and noncompactible radwaste. Based on the data collected, a generalized statement can be made: The dose rates at 3 feet are usually a factor of 4 (for lower dose rates) to 10 (for higher dose rates) lower than the contact dose rate.

For BWRs the upper limit of the contact dose rate on compactible and noncompactible radwaste is normally less than 200 mrem/hr.

Of interest however, is that contact dose rates ranging from 1,000 mrem/hour to 10,000 mrem/hour were also reported for compactible radwaste. It is expected that such dose rates would be associated with noncompactible radwaste originating from the reactor system

Table 4.2-21 Containers Used for Compactible and Noncompactible Radwaste at BWR Facilities

Plant	Containers	
	Compactible	Noncompactible
Deep Bed Condensate Polishing System		
B1	55-gal drums, cardboard boxes, 216 ft^3 liner	4 ft^3 boxes, 216 ft^3 liner, 4 ft x 4 ft x 8 ft plywood boxes, 17 ft^3 liners
B2	NA(2)	NA
B3	55-gal drums	Cardboard boxes, cardboard drums, wooden boxes, 4 ft x 4 ft x 8 ft
B4	Mini dumpster - 6 ft x 4 ft x 4 ft carbon steel	Same as for compactible and 55-gal drums
B5	55-gal drums, wooden crates - 7 ft x 4 ft x 4 ft, 83-gal drums	Wooden crates, 7 ft x 4 ft x 4ft
B6	55-gal drums	Wooden box, 7.3 ft x 4 ft x 4 ft
Precoat Filter Condensate Polishing System		
B7	55-gal drums	55-gal drums
B8	55-gal drums	4 ft x 4 ft x 8 ft plywood boxes
B9	Muncher bins - 184 ft^3, 55-gal drums, Argon bins - 123 ft^3, wooden crate - 69 in. x 45 3/4 in. x 39 in.	Same as for compactible
B-10	55-gal drums	55-gal drums, 4 ft x 4 ft x 8 ft plywood boxes
B-11	55-gal drums	4 ft x 4 ft x 7 ft & 4 ft x 4 ft x 8 ft plywood boxes
B-12	55-gal drums	Atcor cask #5805/B - type B - 55 ft^3, 4 ft 4 ft x 7 ft wooden crate

Table 4.2-22 Reported Density (lbs/ft^3) of BWR Compactible and Noncompactible Radwaste

Plant	Compactible	Noncompactible
Deep Bed Condensate Polishing System		
B1	21.8 - 32.1	8.4 - 46.2 crushed fuel channels up to 59 lb/ft^3
B2	NA	NA
B3	25 - 75	3 - 20
B4	49.9	49.9
B5	<62.4	<62.4
B6	34	NA
Precoat Filter Condensate Polishing System		
B7	12.5 - 25	12.5 - 25
B8	34 - 40.8	NA
B9	NA	NA
B10	27.1 - 54.3	12.5 - 18.7
B11	25 - 31.2	12.5 - 31.2
B12	NA	NA

or reactor internals. The dose rates may be attributed to compactible radwaste for the following reasons:

- Noncompactible radwaste is also packaged in 55-gallon drums. Once capped, a definite identification of radwaste types cannot be made.

- Compactible and noncompactible radwaste are commonly packaged together and reported as compactible radwaste

Data collected on radiation levels are presented in Table 4.2-23 for BWRs.

4.2.2 Pressurized Water Reactors

4.2.2.1 PWR Spent Resin

Data on the annual volume of deep bed resins shipped from PWRs without condensate polishing systems are available from 9 of the 10 facilities surveyed. For facility P3 the data represented an estimated annual average, according to the plant's radwaste supervisor, whereas Plant 8 has not shipped any of its spent resins from the site. The 9 plants that supplied data represent 50 reactor-years of operation. These plants have shipped an average of 540 ft^3/yr of resin to burial sites. When weighted by plant size the rate of waste generation is 0.94 ft^3/MWe-yr, and the waste has an average radioactive concentration of 0.65 Ci/ft^3, or approximately 0.61 Ci/MWe-yr. In PWRs that process secondary system condensate through either deep bed demineralizers or precoat type filters or filter/demineralizers the annual generation rate of spent resin is 0.32 ft^3/MWe-yr. Of the 27 reactor-years for which data on the volume and activity of the shipped resins are available, 6 are years in which no resin was shipped. The average activity concentration for the remaining years is 0.62 Ci/ft^3, which results in an annual generation rate of 0.20 Ci/MWe-yr.

There is no readily available explanation for the fact that plants without a condensate polishing system generate more waste in the form of spent resin than plants with a condensate polishing system. For plants without a condensate polishing system there were 50 reactor-years of data ranging from years when no resin was shipped to years when 5,000 ft^3 of resin were shipped. When converted to ft^3/MWe-yr there were 11 reactor-years in which the specific volume was greater than 1.0 ft^3/MWe-yr with an actual range of 1.10 ft^3/MWe-yr to 8.88 ft^3/MWe-yr. For plants with condensate polishing systems, 35 reactor-years of data are available. The highest specific volume reported was 0.97 ft^3/MWe-yr. The volumes actually shipped ranged from 0 to 765 ft^3/yr for a 2 unit plant. The volume of over 10,000 ft^3 reported by plant P17 was not used in the analysis. These resins are associated with plant startup and are not a result of actual plant operations. If this 1 year of data were used it would increase the average generation rate by 78%. (See Table 4.2-24.)

Table 4.2-23 Reported Radiation Levels (mrem/hr) From BWR Compactible and Noncompactible Radwaste

Plant	Compactible		Noncompactible	
	3 ft	Contact	3 ft	Contact
Deep Bed Condensate Polishing System				
B1	1 - 800	10 - 10,000	0.5 - 10	2 - 100
B2	NA(2)	NA	NA	NA
B3	1 - 20	5 - 170	1 - 5	1 - 25
B4	10	100	5 - 10	40 - 60
B5	6 - 7	60 - 70	10	100
B6	<1 - 200	1 - 2,200	<1 - 8	<1 - 250
Precoat Filter Condensate Polishing System				
B7	0 - 10	5 - 50	0 - 10	5 - 50
B8	1 - 3	20 - 25	2	10
B9	5 - 10	50 - 150	5 - 15	50 - 150
B10	<5	10 - 30	<1	<5
B11	<1	5	1 - 3	7 - 15
B12	10 - 20	100 - 130	12 - 30	60 - 100

Table 4.2-24 Volumes and Activities of PWR Deep Bed Resin

Plants Without Condensate Polishing Systems	1971 Volume (ft^3)	1971 Activity (Ci)	1972 Volume (ft^3)	1972 Activity (Ci)	1973 Volume (ft^3)	1973 Activity (Ci)	1974 Volume (ft^3)	1974 Activity (Ci)	1975 Volume (ft^3)	1975 Activity (Ci)	1976 Volume (ft^3)	1976 Activity (Ci)	1977 Volume (ft^3)	1977 Activity (Ci)
P1	NA[1]		860.	72.	225.	79.5	504.	51.	336.	8.1	922.	5.1	1,245.	51.9
P2	NA		86.5	-	356.	-	393.	141.	1,080.	1,245.	237.	8.8	557.	1,195.
P3					675.	134.	675.	134.	675.	134.	675.	134.	675.	134.
P4					NA	NA	43.7	384.	121.	1,117.	112.	462.	35.	64.2
P5	1,046.	261.	1,495.	4,752.	225.	138.	680.	800.	665.	1,309.	820.	678.	820.	755.
P6							NA	NA	776.	41.2	0	0	0	0
P7							0	0	99.6	0.054	5,000.	1.735	450.	328.
P8									0	0	0	0	0	0
P9						NA		NA		NA		NA		NA
P10							105.	7.61	346.	34.6	645.	44.6	460.	200.
Plants With Condensate Polishing Systems														
P11	0	0	500.	NA	200.	NA	200.	NA	200.	1.17	100.	65.5	300.	553.
P12									NA	NA	765.	121.	765.	46.9
P13									0	0	363.	177.	484.	72.
P14	0	0	340.	1.8	295.	27.3	190.	20.9	NA	NA	350.	40.3	310.	66.
P15											150.	2.	322.	64.4
P16					0	0	0	0	275.	931.	0	0	336.	1,005.
P17													0	0
P18							300.	NA	300	NA	300.	NA	300.	NA

1. Not available.

With the exception of the 1 year of data from P17 none of the remaining data were considered to be so inconsistent with the remaining data that they should be excluded from the data base in determining the generation rates for a typical plant. Thus the typical plant generation rates are identical to the average plant generation rates.

A tabulation of radionuclides found on PWR spent resins is given in Table 4.2-25. Plants with condensate polishing systems and plants without them have the same radionuclides. Also, there is no perceptible difference between the radionuclides found in PWR spent resins and those found in BWR spent resins. The predominant nuclides are Mn-54, Co-58, Co-60, Cs-134, and Cs-137.

Table 4.2-26 lists the information on the chemical and physical properties of PWR spent resins. Specific resins are listed except when too numerous to present a complete list. Complete lists are given in Table 4.2-27. The resins used are predominantly mixed bed type resins produced by Rohm & Haas, Graver, Illinois Water Treatment, or Diamond Shamrock. Section 2.2.1.1 is a detailed description of the chemical and physical properties of resins.

Reported densities average approximately 0.91 g/cc (56.5 lb/ft^3) for unsolidified resins and, based solely on information from plant P7, roughly 1.4 g/cc (87 lb/ft^3) when solidified. The increase in density is 55%. Containers used to ship spent resin offsite for burial range from standard 55-gallon drums to 300-ft^3 containers. Employees at most facilities said they are able to fill the container to the point where the void at the top is less than 5%. Personnel at two plants said they left a 10% void. One plant listed a 25% void on a 170-ft^3 cask. Most plants ship resins in shielded casks with the shipping container as a liner. Fifty-five gallon drums are shipped in shielded overpacks that hold several drums.

Of the 18 plants surveyed, 15 supplied data on the dose rates at 3 feet and on contact for spent resin waste. Based on the data from 7 of these 15 plants the contact doses for unshielded resins range from 10 mr/hr to 200 R/hr. When shielded the dose rates range from 1.5 mr/hr to 400 mr/hr. When measured at 3 feet the unshielded dose rates are 1.0 mr/hr to 50 R/hr. The shielded dose rates range from 0.5 mr/hr to 30 mr/hr. The individual reported dose rates are in Table 4.2-38 at the end of Section 4.2.2.5.

4.2.2.2 PWR Concentrated Liquids

PWRs without condensate polishing systems supplied 38 reactor-years of data on concentrated liquids. The annual volumes shipped ranged from 0 to almost 19,000 ft^3 for a two-unit plant. The average volume, shipped over a time span that includes one plant with 10 years of operating experience, is 3,700 ft^3/yr.

Table 4.2-25 Radionuclides Present on PWR Spent Resins

Plants Without Condensate Polishing Systems	H-3	Cr-51	Mn-54	Fe-59	Co-57	Co-58	Co-60	Sr-90	Zr-95	Nb-95	Mo-99	Ag-110m	Sb-124	I-131	I-133	Cs-134	Cs-136	Cs-137	Ba-140	La-140	MFP(1)	MCP(2)
P1			X	X		X	X									X		X				
P2			X		X	X	X			X	X	X		X	X	X	X	X				
P3		X	X			X	X		X	X	X					X		X				
P4			X			X	X									X		X				
P5																					X	X
P6						X	X						X			X		X				
P7			X			X	X									X		X				
P8	NA(3)																					
P9			X				X									X		X				
P10			X			X	X									X		X				
Plants With Condensate Polishing Systems																						
P11						X	X									X		X				
P12			X	X		X	X	X										X		X		
P13	X	X	X		X	X	X			X						X	X	X	X			
P14							X									X		X				
P15						X	X															
P16			X		X	X	X					X				X		X				
P17																					X	X
P18	NA																					

1. Mixed fission products.
2. Mixed corrosion products.
3. Not available.

Table 4.2-26 Characteristics of PWR Spent Resin Wastes

Plants Without Condensate Polishing Systems	Types of Resin Used	Solidified	Density g/cm^3	Container Size (ft^3) 7.35	50	60	85	88	90	100	170	180	195	200	% Void at Top	Shield Material
P1[1]	Radwaste-IRN-300; R.C.S.-IRN-217	No	0.88								X				2	NECO L3-181
P2[2]	IRN-77, 78, 150, 217	No	-				X	X					X		-	CNS 4-85, CNS-14-195 DOT-6144
P3[3]	See Table 4.2-27	No	0.9-1.1			X			X						<5	
P4	Not available	Yes	0.69-0.75	X											0	HN-200
P5	See Table 4.2-27	No	0.85-1.0							X					5-10	Atcor LL-50
P6	Not available	No	-							X					<1	Atcor LL-50
P7[4]	See Table 4.2-27	Yes	1.3-1.5	X											<1	Atcor BC-48-220
P8[5]																
P9	Not available	Yes	0.9	X											<5	None
P10[6]	See Table 4.2-27	No	0.8-1.1	X								X			<5	DOT-6144 CNS-15-160
Plants With Condensate Polishing Systems																
P11	See Table 4.2-27	No	.96							X					<1	Atcor LL-50
P12	See Table 4.2-27	No	-								X				25	HN170 and 200
P13[7]	See Table 4.2-27	No	0.89-1.09		X										10	PPI Cask-50 ft
P14[8]	See Table 4.2-27	No	0.60		X					X					10	None
P15[9]	Not available	No	1.28		X										-	Lead and steel (New)
P16	Not available	No	-				X								-	
P17	Not available	No	-		X										-	
P18[5]	See Table 4.2-27	No	-											X	-	

1. Boric acid found in waste.
2. Primary casks CNS4-85.
3. Boric acid found in waste.
4. Oxalic acid, citric acid, boron, turbine lube oil, chem lab wastes found in resin.
5. No resins shipped as of 12/77.
6. Boric acid and lithium (0.75-2.2 ppm) found in waste.
7. Boric acid, sodium thyosulfate, sodium nitrate found in waste.
8. Boron in waste.
9. Boron and nitrates in waste.

Table 4.2-27 Resins Used in PWR Deep Bed Demineralizers

Plant 3

Radwaste system: HOH Types, mixed bed

Dowex: MR-3
Rohm & Haas: IRN 150
Diamond Shamrock: ARM-381

Chemical volume control system; spent fuel pool and deborating demineralizers:

Dowex: MR-5, SBR, HCR-S-H
Rohm & Haas: IRN-217, IRN-78, IRN-77
Diamond Shamrock: Duolite 386, ARA-366, ARC-351

Plant 5

Boron recovery (115 ft^3/yr); aerated waste (200 ft^3/yr); spent fuel cleanup (90 ft^3/yr)

Rohm & Haas: IRN 150 or,
Ionac: NM-60

(Volumes are average annual usage.)

Primary letdown (CVCS) (180 ft^3/yr)

Rohm & Haas: IRN 150 or Ionac NM-60
Rohm & Haas: IRN 217
Rohm & Haas: IRN 77 or Ionac NC-10
Rohm & Haas: IRN 78 or Ionac NA-38

Plant 7

Liquid radwaste system

Illinois Water Treatment: TM-1, TC-1 and NR-1
Rohm & Haas: IRN-150 and IRN-77
Diamond Shamrock: ARC-351

Chemical and volume control system

Illinois Water Treatment: NR-6, NR-1
Rohm & Haas: IRN-150AC, IRN-78, IRN-77
Diamond Shamrock: ARM-386, ARM-381, ARC-351

Table 4.2-27 Resins Used in PWR Deep Bed Demineralizers (Cont'd)

Boron recycle

Illinois Water Treatment: NR-6, NR-1
Rohm & Haas: IRN-15AC, IRN-78AB, IRN-77
Diamond Shamrock: ARM-381, ARA-366W, ARC-351

Chem lab

Illinois Water Treatment: TMD-12

Plant 10

Chemical and volume control system (200 ft^3/yr)

Rohm & Haas: IRN-77, IRN-76, IRN-150 and 150T, IRN-217
Diamond Shamrock: ARA-366

Boron recycle

Rohm & Haas: IRN-150 and 150T, IRN-77

Fuel pool cleanup and radwaste

Rohm & Haas: IRN-150 and 150T

Steam generator blowdown

Illinois Water Treatment: NR-2 WS

Plant 11

Chemical and volume control system

Rohm & Haas: IRN-217, IRN-218

Boron recovery system

Rohm & Haas: IRN-78

Spent fuel pool cleanup, liquid radwaste and condensate polishing systems

Rohm & Haas: IRN-217

Table 4.2-27 Resins Used in PWR Deep Bed Demineralizers (Cont'd)

Plant 12

Boron recovery system

Graver: GR-1 and GR-2

Chemical and volume control system, spent fuel pool cleanup system, steam generator blowdown system and liquid radwaste system

Graver: GR-3

Plant 13

Radwaste system (2,700 ft^3 over 3 yr)

Diamond Shamrock: ARA-371 and ARC-368

Spent fuel pool cleanup system (126 ft^3 over 3 yr),
Chemical and volume control system (150 ft^3 over 3 yr)

Rohm & Haas: IRN-150

Plant 14

Dirty radwaste (120 ft^3/ft)(1), condensate polishing (35 ft^3/yr), CVCS (50 ft^3/yr), spent fuel pool cleanup (33 ft^3/yr), clean radwaste (45 ft^3/yr), boron recovery (35 ft^3/yr)

Dow: MR-3

Plant 18

Radwaste system, chemical and volume control system, boron recovery system

Rohm & Haas: IRN-150

1. Average annual use over 6 years.

Based on individual plant size the weighted average is 3.9 ft^3/MWe-yr. The annual activity shipped ranged from 0 to 1,941 curies with an average of 190 curies for a concentration of 0.05 Ci/ft^3 or 0.20 Ci/MWe-yr.

Ten of the 38 reactor-years of data are from plant P3. Both the reported volumes and activities are estimated and are substantially higher than the numbers reported by most of the other plants. Although this plant represents slightly less than 25% of the data it accounts for 57% of the total reported volume and 35% of the activity. Excluding this plant from the calculations for the typical plant results in the following tabulations. The average annual waste volume is 2,100 ft^3/yr containing an average of 170 curies for a concentration of 0.081 Ci/ft^3. Annually, the energy-weighted shipment is 2.6 ft^3/MWe, resulting in 0.21 Ci/MWe-yr.

Plants with a condensate polishing system provided 28 reactor-years of data ranging from 0 to 12,300 ft^3 for a three-unit site. The annual average shipment of 3,200 ft^3 contains 15 curies for an average concentration of 0.005 Ci/ft^3. Total activity levels range from 0.0 to 60 curies. Based on plant size, the average volume of concentrated liquids shipped offsite is 4.8 ft^3/MWe-yr. The activity shipped is 0.024 Ci/MWe-yr.

There are 1 or 2 years of data for which the volume or activity of the waste is many times greater than the average, but no plant stands out as abnormally high throughout its operating life. Therefore, it appears that all of these facilities are operating within the definition of typical plants. As such, the average plant and typical plant share the same statistics. (See Table 4.2-28.)

The radionuclides reported to be in concentrated liquids are in Table 4.2-29. Employees of all 12 facilities who provided detailed isotopic breakdowns reported Co-60. Personnel at 11 plants reported Co-58. Employees at a majority of the plants reported Mn-54, Cs-134, and Cs-137. A scattering of other radionuclides were reported by officials at one or more of the plants.

Table 4.2-30 provides the remaining data collected on PWR concentrated liquid wastes. The most commonly reported chemical in the concentrated wastes is boron or boric acid. This is to be expected because most of the plants also reported boron or boric acid on the resins. Considering that many of the resins are regenerative, the boron will, along with the regeneration chemicals, be transferred to the concentrator feed tank and become part of the concentrator bottoms. Other chemicals reported include soap, used in the plant laundry; anti-foaming agents, used to minimize soap foaming in the concentrator; sodium hydroxide and sodium sulfate from the resin regeneration process; potassium chromate and other chemical inhibitors; and numerous other chemicals used in primary water chemistry control. For most plants the pH was approximately 6.5. The range was 4 to 9. The average weight percent solids was 11.4 over a range of 2 to 20. Personnel from a few plants reported

Table 4.2-28 Volumes and Activities of PWR Concentrated Liquid Wastes

Plants Without Condensate Polishing Systems	1971		1972		1973		1974		1975		1976		1977	
	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)
P1	NS[1]		NS		NS		NS		NS		NS		NS	
P2					3,460.		2,670.	26.6	2,020.	55.	2,000.	51.6	2,000.	44.
P3			0	0	15,300.	478.	15,300.	478.	15,300.	478.	15,300.	478.	18,970.	478.
P4					696.	2.2	1,274.	4.1	2,398.	109.	917.	35.6	978.	15.5
P5	NS		NS		NS		NS		NA	NA	2,584.		2,488.	
P6							0	0	NA	1,941.	NA	789.	7,380.	1,188.
P7							0	0	74	.05	0	0	0	0
P8									NA	NA	1,968.	3.9	8,280.	40.3
P9					NA[2]			NA		NA		NA		NA
P10							1,884	7.61	2,090.	34.6	1,121.	44.6	12,910.	45.8
Plants With Condensate Polishing Systems														
P11	10,126.	NA	2,773.	NA	2,167.	NA	2,191.		3,144.	34.8	2,563.	23.1	1,338.	11.8
P12									NCL[3]		NCL		NCL	
P13									NCL		NCL		NCL	
P14	0	0	0	0	0	0	4,350.	3.13	8,363.	5.02	13,725.	4.59	8,775.	2.0
P15												0	779.	12.9
P16					NCL		NCL		221.	21.1	11,585.	45.5	12,285.	59.1
P17											0	0	376.	57.5
P18							NA		NA		6,640.	69.2	3,360.	26.4

1. No solidification.
2. Not available.
3. No concentrated liquids.

Table 4.2-29 Radionuclides Present in PWR Concentrated Liquid Waste

Plant	Cr-51	Mn-54	Fe-59	Co-57	Co-58	Co-60	Zr-95	Zr-97	Nb-95	Ku-103	Ag-110m	Sb-124	Sb-125	I-131	I-133	Cs-134	Cs-136	Cs-137	La-140	Ba-140	Ce-141	Ce-144	MFP	MCP
Plants Without Condensate Polishing Systems																								
P1	NCL[1]																							
P2	X	X		X	X	X		X	X		X	X	X	X		X		X						
P3	NI[2]																							
P4		X			X	X										X		X			X	X		
P5	X	X	X		X	X	X		X	X														
P6	U[3]																							
P7																							X	X
P8		X			X	X			X					X		X	X	X						
P9		X				X										X		X						
P10		X			X	X										X		X						
Plants With Condensate Polishing Systems																								
P11					X	X										X		X						
P12	NCL																							
P13	NCL																							
P14		X			X	X																		
P15		X			X	X								X										
P16					X	X								X				X						
P17		X			X	X										X		X						
P18		X			X	X												X						

1. No concentrated liquid.
2. Not identified.
3. Unknown.

Table 4.2-30 Characteristics of PWR Concentrated Liquid Wastes

Plants Without Condensate Polishing Systems	Significant Chemicals in Waste	pH	Source or Means of Production	Weight % Solids	Solidified	Density gm/cc	% Void at Top	Container Size (ft^3) 7.35	11.1	40	50	133	150	195	200	224	Shielding
P1	No concentrated liquid wastes																
P2	Boric acid, silica, oil, sulfates, defoamer	6	(1)	15-19 (H_3BO_3)	Yes	1.3	5	X									NA[2]
P3	Sears Laundry Soap, chemicals from primary chemistry lab, boron	<6.5	(1)	2-4	Yes	1.74	1						X				NA
P4	Unidentified	>7.5	(1)	12 (H_3BO_3)	Yes	0.98-1.97	0	X									NA
P5	Boric acid	4-6	(1)	-	Yes	1.6-1.7	5-10	X	X								NA
P6	None	-	(1)	-	Yes	0.64	50	X									NA
P7	Unknown	<6.5	(1)	-	Yes	1.5	<1	X									NU[3]
P8	Sodium tetraborate, 22,000 ppm boron	8.5-9.0	(1)	5-10	Yes	0.8	5					X			X		NU
P9	Boric acid, anti-foam compound (Dow-Corning emulsion), organic and inorganic laboratory chemicals	4.5-6.5	(1)	~12	Yes	0.99	5	X									NA
P10	$NaOH, K_2CrO_4, H_3BO_3$ (12%)	-	(1)	-	Yes	1.4-1.5	5	X									NU
Plants With Condensate Polishing Systems																	
P11	Anti-foam agents (infrequent)	8.0-8.5	(1)	11	Yes	1.53	1	X									2 in. lead
P12[4]																	
P13[4]																	
P14	Boron, lithium hydroxide (LiOH)	4-8	(1)	~20	Yes	1.06	20-25			X							None-Neco liner
P15	Potassium permanganates, phosphates, detergents, boric acid, sodium silicate, sodium nitrate, lithium hydroxide	-	(1)		Yes	-	-				X			X			lead
P16	Anti-foam agents	6.5-7.5	(1)		Yes	-	10			X							NA
P17	Boric acid	<6.5	(1)	10[5]	Yes	-	0	X				X				X	NA
P18	Boric acid, Na_2SO_4, particulates <1% oil, 13,000 ppm boron	4.8-9	(1)	10	Yes	1.22	5				X						NA

1. Concentrator bottoms.
2. Not applicable.
3. Not used.
4. No concentrated liquids.
5. Boron recovery evaporation 10% solids in bottoms, 20% solids in floor drains.

specific boric acid weight percent solids of 12, and one plant reported 19 weight percent solids. As identified in Table 4.1-3, all of the plants solidify their concentrated liquid wastes prior to shipment. Unsolidified densities averaged 1.00 g/cc, while the solidified densities, irrespective of solidification agent, averaged 1.64 g/cc, a 64% increase in density.

Container sizes ranged from the standard 55-gallon drum (7.35 ft^3) to 224-ft^3 casks or liners.

It appears that most plants use 90 to 95% of the container leaving less than a 10% void at the top. Two plants reported unusually high free space in the containers: 20 to 25% for one plant, and 50% for the other. Neither of these plants supplied additional information regarding these percentages of free space.

Of the 15 plants that ship concentrated solidified liquid wastes, only two reported dose rates with shielding and many indicated that no shielding was used. For unshielded wastes the contact dose rates ranged from 7 mr/hr to 50 R/hr with most of the dose rates between 50 and 200 mr/hr. Employees of the two plants that indicated shielding was used in the shipment of this waste reported contact dose rates of 1 to 2 mr/hr and 200 to 800 mr/hr. The dose rates at 3 feet for unshielded wastes range from 1 mr/hr to 1 R/hr with most of the data ranging from 5 to 50 mr/hr. Personnel from the two plants that use shielding reported dose rates of 0.2 to 1.5 mr/hr and 5 to 75 mr/hr. In both cases these dose rates are more compatible with unshielded dose rates and may be the dose rates prior to shielding for shipment. The ranges 1 to 2 mr/hr on contact and 0.2 to 1.5 mr/hr at 3 feet are the lowest reported dose rates and definitely indicate dose rates with shielding. The data for each individual plant are in Table 4.2-38.

4.2.2.3 PWR Filter/Demineralizer and Precoat Filter Sludge

Precoat type filters are used in three of the 18 PWRs surveyed. They are P15, P16, and P18. At the time of the survey P15 had not produced a significant quantity of precoat sludge, and none had been shipped. Facility P16 uses partial flow filter/demineralizers for condensate polishing but personnel from that plant did not supply any information regarding the operation of the units. Facility P18 has filter demineralizers in the boron recovery system, spent fuel pool cleanup system, the miscellaneous radwaste system, and the condensate polishing system. Data on the volumes and radioactivity levels associated with the sludge from these filters are available for 1976 and 1977 only. These data are given in Table 4.2-31. For plants without a condensate polishing system the waste volumes and activities from precoat filters are zero. For plants with condensate polishing systems the average, annual, normalized generation rate is 0.15 ft^3/MWe-yr. At an average radionuclide concentration of 0.083 Ci/ft^3 the activity generation rate is 0.012 Ci/MWe-yr. These figures are for both the average plant and the typical plant.

Table 4.2-31 Volumes and Activities of PWR Precoat Filter Waste

Plant	1971 Volume (ft^3)	1971 Activity (Ci)	1972 Volume (ft^3)	1972 Activity (Ci)	1973 Volume (ft^3)	1973 Activity (Ci)	1974 Volume (ft^3)	1974 Activity (Ci)	1975 Volume (ft^3)	1975 Activity (Ci)	1976 Volume (ft^3)	1976 Activity (Ci)	1977 Volume (ft^3)	1977 Activity (Ci)
Plants with Condensate Polishing Systems														
P15											0		0	
P16[1]						NA[2]	NA		NA		NA		NA	
P18							NA		NA		362	20.9	180	23.6

1. Plant has not yet shipped any filter sludge.
2. Not applicable.

The only radionuclides reported are Mn-54, Co-58, Co-60, I-131, and Cs-137. Solidified filter sludge is shipped in 55-gallon drums and 50-ft^3 liners filled to approximately 94% capacity. The contact dose rates are 1.2 to 2.0 R/hr unshielded and drop to about 10 mr/hr with 2.5 inches of lead shielding. (See Table 4.2-32) At 3 feet the unshielded dose rate is 120 to 250 mr/hr and it drops to 3 mr/hr when shielded. (These data are given in Table 4.2-38)

4.2.2.4 PWR Cartridge Filter

Cartridge filters are used extensively in PWRs. Although they are applicable to most systems they are not used with the condensate polishing system. The flow rates (several thousand gallons per minute) are too high for typical cartridge filters. Because cartridge filters are not used with condensate polishing systems, the quantity and activity of cartridge filter wastes are not affected by the presence or absence of condensate polishing systems. Furthermore, cartridge filters only remove insoluble contaminants whereas demineralizers and precoat filters using ground-ion exchange resins remove both soluble and insoluble contaminants.

The volumes and activities of cartridge filter wastes collected from the 18 PWRs in the survey are given in Table 4.2-33. Unlike the volume data for other waste types the data in Table 4.2-33 include the material used to solidify the cartridges. Volumes range from 14 ft^3/yr to 1,040 ft^3/yr and average 260 ft^3/yr over the 52 reactor-years of data. The radioactivity levels associated with these filters is available for 48 of those 52 years. The average radioactivity content is 77 Ci/yr. The lowest, non-zero activity was 0.6 Ci/yr and the highest reported activity level was 1,081 Ci/yr. The average activity concentration is 0.30 Ci/ft^3. When weighted according to plant size, the volume generation rate is 0.39 ft^3/MWe-yr; the activity-generation rate is 0.12 Ci/MWe-yr. None of the data collected appear to be so far out of line with the rest of the data that it should be excluded from the calculations of the typical plant. Therefore, the average plant and the typical plant have the same volume and activity generation rates.

As with other forms of waste, cartridge filters show the same basic radionuclides: Co-58, Co-60, Cs-134, and Cs-137. These data, as supplied by individual plants, are given in Table 4.2-34. Tables 4.2-35 and 4.2-36 contain data on specific cartridge filters used in the surveyed PWRs. The vast majority of cartridge filters are manufactured by AMF Cuno or Filterite and are made of wound cotton. Specific manufacturers' data on representative cartridges identified in the survey are given in Section 2.2.3. Micron ratings range from 0.1 to 100. Most of the filters are rated between 5 micron and 25 micron. Each filter housing contains up to 48 cartridges, whereas the most popular size contains only 8. A single element filter is the smallest. Packaged densities range from 0.6 g/cc to 2.4 g/cc, averaging 1.35 g/cc (84.2 lb/ft^3). Twelve of the 18 facilities use the standard 55-gallon drum to package and ship cartridge filters. Three of these

Table 4.2-32 Characteristics of PWR Precoat Filter Sludge

Plant[1]	Precoat Type	Solidified	Density g/cm^3	Container Size (ft^3) 7.35	50	% Void at Top	Shield Material
Plants With Condensate Polishing Systems							
P15[2] P16[3] P18[4]	Powdered Resin Epicore HOH for Reactor Water Cleanup, Spent Fuel Pool Cleanup and Miscellaneous Waste.	Yes	-	X	X	~ 6	2-1/2 in. lead for 50 ft^3 liner

1. Plants P1 through P14 and P17 do not use precoat type filters anywhere within the plant.
2. Filter demineralizers in condensate polishing system; no waste shipped as of 12/77.
3. Filter demineralizers in condensate polishing system; no specific data available.
4. Radionuclides (Mn-54, Co-58, Co-60, I-131 & Cs-137). Neco casks, one for 14 - 55-gal drums, one for 50 ft^3 liner.

Table 4.2-33 Volumes and Activities of PWR Cartridge Filter Waste - As Solidified

Plant	1971		1972		1973		1974		1975		1976		1977	
	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)
Plants Without Polishing System														
P1														
P2				NA[1]		NA	265	23.7		NA		NA		NA
P3					739	7.93	739	7.93	739	7.93	739	7.93	739	7.93
P4							56	130	14	224		NA		NA
P5	72	11.3	504	13.3	204	85.9	0	0		NA		NA		NA
P6														
P7									52.15	.55	252	24.6	17.1	45.8
P8											186	35.9	120.2	485.9
P9							7,926	NA	7,926	NA	7,926	NA	7,926	NA
P10														
Plants With Condensate Polishing Systems														
P11	294	NA	294	NA	294	NA	294	NA	294	.6	294	3.5	294	10.4
P12											720	NA	345	2.35
P13									400	14	1,040	315	450	1,081
P14					290	9.2	100	1.2			107	1.5	200	14
P15											0	0	100	9.23
P16					0	0	22	55	242	140	29.4	155	66.2	628
P17														
P18														

1. Not available.

Table 4.2-34 Radionuclides Present in PWR Cartridge Filters

Plants Without Condensate Polishing Systems	Cr-51	Mn-54	Fe-59	Co-58	Co-60	Zr-95	Nb-95	Ag-110m	Sb-124	Cs-134	Cs-137	MFP	MCP
P1		X	X	X	X					X	X		
P2	U(1)												
P3												X	X
P4				X	X						X	X	
P5												X	X
P6		X		X	X				X	X	X		
P7	X	X	X	X	X	X	X						
P8	X			X	X					X	X		
P9	X				X					X	X		
P10												X	X
Plants With Condensate Polishing Systems													
P11				X	X					X	X		
P12												X	X
P13	X	X		X	X			X					
P14				X	X					X	X		
P15				X	X								
P16				X	X					X	X		
P17	NA(2)												
P18		X		X	X					X	X		

1. Unknown.
2. Not available.

Table 4.2-35 Characteristics of PWR Cartridge Filter Wastes

Plant	Process System	Type of Filter	Micron Rating	Elements per Filter	Density g/cm^3	Shipping Container (ft^3) 4.0	7.35	10	28	50	90	100	150	170	195	Elements per Container	Shield Material
Plants Without Condensate Polishing Systems																	
P1	Seal water	Filterite-nylon	5	8			X									6	6-in. cement
	Radwaste	Fram-paper	25	8 to 18			X									6	6-in. cement
P2	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	1.3		X									(1)	
P3	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	0.8-1.1			X			X		X		X	60-in 90 ft^3	Chem. Nuc. Casks
P4[2]	C.V.C.S.	--	--	—	.6		X		X							2	
P5[3]	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	0.8-1.2		X									12	6-in. cement
P6	NA[4]	NA	0.1-25	8	1.6		X									8-10	Lead
P7	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	2.1-2.4		X[5]									10	6-in. cement
P8	NG[6]	NG	5, 25	1-8	—	X										1	None
P9[7]	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	1.5		X									NA	None
P10[8]	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	See Table 4.2-36	1.4-1.5		X									2-10	As used for evaporator bottoms
Plants With Condensate Polishing Systems																	
P11	NG	NG	0.5	3-8	1.5		X									5	None
P12	NG	NG	0.5	1	—	X[9]	X							X		1	None
P13[10]	NG	See Table 4.2-36	0.43 and others	1	—					X						30-50	None
P14[11]	NG	Filterite	1-25	1-8	—								X			25	None
P15	NG	AMF, Pall, Petters	10	—	—					X						—	2-in. lead

Table 4.2-35 Characteristics of PWR Cartridge Filter Wastes (Cont'd)

Plant	Process System	Type of Filter	Micron Rating	Elements per Filter	Density g/cm^3	Shipping Container (ft^3) 4.0	7.35	10	28	50	90	100	150	170	195	Elements per Container	Shield Material
P16	NG	Cuno	1	20	--		X									20	(12)
P17	ND(13)																
P18	NG	NG	NG	NG	--					X						--	None

1. Varies.
2. Density and doses for 28-ft^3 containers.
3. Additional shielding using Atcor AL-31 and LL-50 used if necessary.
4. Not available.
5. Approximately 3 ft^3 usable drum volume.
6. Not given.
7. Filters contain boron.
8. Shipped with solidified evaporator bottoms.
9. Basket filters, packed in 4-ft^3 box, with trash in 55-gal. drum and with resin in 170-ft^3 liner.
10. Filters contain boric acid.
11. Filters contain boron and lithium doses with shielding.
12. Not available.
13. No data; treated as compactible or noncompactible trash.

Table 4.2-36 Cartridge Filters Used in PWR

Plant	Filter Service	Micron Size	Number of Elements Per Filter	Manufacturer and Part Number
P2	Reactor coolant	25	8	Tate Engineering - C9627-6366
	Boric acid filter	25	8	Tate Engineering - C9627-6366
	Spent fuel pool	5	8	Tate Engineering - C9627-6742
	Seal water injection	5	6	Tate Engineering - C9627-6742
	S.F.P. skimmer	25	8	Tate Engineering - N23R 305V
	Waste holdup tank	25	2	Pall Trinity - 5ESC107702EGJ-007
	Polishing demineralizer	5	8	AMF Cuno 89338-32
	Condensate	5	3	AMF Cuno 89338-32
	Seal water return	25	48	AMF Cuno 89338-33
	Concentrates	5	3	AMF Cuno 89338-33
	Ion exchange gas stripper	25	3	AMF Cuno 89338-33
P3	Reactor coolant	25	1	AMF Cuno
	Seal water injection	5	1	AMF Cuno
	Seal water return	5	1	AMF Cuno
	Waste evaporator feed	25	3	AMF Cuno, cotton wound
	Recycle evaporator condensate	-	1	-
	Recycle evaporator concentrate	-	1	-
	Spent fuel pit	5	1	AMF Cuno
	Spent fuel pit skimmer	5	1	AMF Cuno
	Boric acid	-	5	-
P5	Cavity purification	10,5 & 1	-	AMF Cuno
	Safety injection	1	-	AMF Cuno
	Aerated waste tank	25	-	AMF Cuno

Table 4.2-36 Cartridge Filters Used in PWF (Cont'd)

Plant	Filter Service	Micron Size	Number of Elements Per Filter	Manufacturer and Part Number
P5 (cont'd)	Reactor coolant pump seal water	1	-	AMF Cuno
	R.C. purification pre-filter	5	-	AMF Cuno
	R.C. purification post-filter	5 & 25	-	AMF Cuno
	Spent fuel pool	5	-	AMF Cuno
	Boric acid mixing tank	20	-	AMF Cuno
	Liquid radwaste	3	-	AMF Cuno
P7	Reactor coolant	15	8	AMF Cuno, cotton wound
	Seal water injection	5	1	AMF Cuno, cotton wound
	Seal water return	25	6	AMF Cuno, cotton wound
	Recycle evaporator feed	25	3	AMF Cuno, cotton wound
	Waste evaporator feed	15	3	AMF Cuno, cotton wound
	Recycle evaporator concentrate	25	4	AMF Cuno, cotton wound
	Spent fuel pit	25	1	AMF Cuno, cotton wound
	Boric acid	25	-	AMF Cuno, cotton wound
	Letdown demineralizer filter	2	-	AMF Cuno, cotton wound
P9	Reactor coolant	25	8	AMF Cuno CG8DB3
	Boric acid	25	8	AMF Cuno CG8DB3
	Spent fuel pit	5	8	Commercial Filter Co. - 1732-6TSSCN-2735

Table 4.2-36 Cartridge Filters Used in PWR (Cont'd)

Plant	Filter Service	Micron Size	Number of Elements Per Filter	Manufacturer and Part Number
P9 (cont'd)	Seal water injection	5	6	Commercial Filter Co. - 1732-6TSSCN-2735
	Seal water return	25	26	AMF Cuno CG13DB4
	Ion exchanger	25	4	AMF Cuno CG4DB2
	Evaporator condensate	25	4	AMF Cuno CG4DB2
	Boric acid concentrator	25	4	AMF Cuno CG4DB2
	Spent fuel skimmer	5	3	NG(1)
P10	Reactor coolant drain filter	20	8	Filterite - cotton wound
	Nonaerated drain filter	15	1	AMF Cuno - stainless steel
	Aerated drain filter	15	1	AMF Cuno - stainless steel
	Waste disposal sump A	15	1	AMF Cuno - stainless steel
	Waste disposal	100	3	Filterite - cotton wound
	Reactor coolant	15	1	AMF Cuno - stainless steel
	Steam generator blowdown holdup tank	15	1	AMF Cuno - stainless steel
	Seal water filter	15	1	AMF Cuno - stainless steel
	Letdown filter	15	1	AMF Cuno - stainless steel
	Spent fuel pit	25	8	Filterite - cotton wound
	Seal water filter	20	8	Filterite - cotton wound
	Ion exchange filter	30	3	Filterite - cotton wound
	Seal water injection filter	5	6	Filterite - cotton wound
	Concentrates filter	5	4	Filterite - cotton wound
	Waste evaporator feed	10	19	Filterite - cotton wound
	Spent fuel pit skimmer	25	8	Filterite - cotton wound
	Reactor cavity cleanup	1	6	Filterite - cotton wound

Table 4.2-36 Cartridge Filters Used in PWR (Cont'd)

Plant	Filter Service	Micron Size	Number of Elements Per Filter	Manufacturer and Part Number
P10 (cont'd)	Steam generator blowdown ion exchange reclaim	25	6	Filterite - cotton wound
	Boric acid filters	20	8	Filterite - cotton wound
P13	Uses cartridge filter manufactured by Pall Trinity Micro Corp. Model Numbers - SESC107703-ECJ004, SESC107705-ECJ004, SESC107705-ECJ004 and SESC100703-EG5004.			

1. Not given.

12 place a 12-inch diameter pipe sleeve in the center of the drum and pour cement in the outer annulus. This results in approximately 5 to 6 inches of cement shielding in the drum. Cement or a lead plate is put in the top and bottom of the sleeve with the cartridges in the sleeve. Depending on the size of the cartridge, 6 to 12 cartridges can be placed in the 3 ft^3 of space in the sleeve. If the entire volume of the drum is used, 20 cartridges can be disposed of in a 55-gallon drum. Larger containers will obviously hold more cartridges. Facility P3 reports disposing of 60 cartridges in a 90-ft^3 liner and facility P13 reports getting 30 to 50 cartridges into a 50-ft^3 liner.

Detailed data supplied by plants P2, P7, and P10 are in Table 4.2-37. Average contact and 3-foot dose rates are given in Table 4.2-38. Because of the diverse application of cartridge filters, the contact doses for unshielded filters cover a range from 5 mr/hr to 100 R/hr. Shielded dose rates, for those filters that require additional shielding for shipment, are several orders of magnitude lower. Dose rates at 3 feet tend to be a factor of 4 to 10 less than contact doses.

4.2.2.5 PWR Compactible and Noncompactible Waste

As with BWRs, PWR compactible and noncompactible wastes do not occur as a direct result of waste treatment processes. They are the byproducts of maintenance and laboratory work, or they are disposable clothing, step-off pads, mop heads, and/or broken tools. The volume of compactible and noncompactible waste varies significantly with the maximum volumes generated during major equipment maintenance periods and refueling outages. The activity levels of these wastes will also vary widely, with high activity not necessarily associated with high volume.

Table 4.2-39 lists the various items reported to compose compactible and noncompactible wastes from PWRs. The list includes items that are typically associated with compactible and noncompactible wastes, such as wood, paper, contaminated tools, glassware, contaminated clothing, rags, and various plastics. The list of compactible and noncompactible waste from facility P2 was unusually detailed and is given separately in Table 4.2-40.

The total volume of compactible and noncompactible radwaste produced as a result of 89 reactor-years of operation was 838,000 ft^3. However, 547,000 ft^3 (65%) of this radwaste was generated at just three reactor sites (P5, P9 and P16). These three sites represent 17% of the sites surveyed and 28 reactor-years of operation (31% of the reactor-years under evaluation). The situations at these three facilities are significantly different from the rest of the PWR facilities surveyed, and therefore it is questionable that they represent a typical PWR. Table 4.2-41 lists the volume and activities of the compactible and noncompactible wastes reported in the survey.

Table 4.2-37 Average Radiation Levels for Specific PWR Cartridge Filters

Plant	Filter Service	Dose Rate (R/hr) Contact	1.5 ft
P2	Reactor coolant	30.	15.
	Spent fuel pit	6.	2.5
	Waste filters	0.160	30 mr/hr
	Seal water injection	3.5	0.650
	Seal water filters	2.5	0.450
	Polishing demineralizer filters	0.140	35 mr/hr
	Spent fuel pit skimmer filters	7 mr/hr	3 mr/hr
	Boric acid filter	0.225	80 mr/hr
	Condensate filters	90 mr/hr	15 mr/hr
	Concentrates filter	1.500	0.250
	Ion exchange filter	9 mr/hr	4 mr/hr
P7	Reactor coolant	13.	NA(1)
	Seal water injection	3.	NA
	Seal water return	2.	NA
	Recycle evaporator feed	0.6	NA
	Waste evaporator feed	0.5	NA
	Recycle evaporator concentrate	10.	NA
	Spent fuel pit	1	NA
P10	Reactor coolant drain filter	15-20	NA
	Non-aerated drain filter	50 mr/hr	NA
	Waste disposal sump 'A' filter	10 mr/hr	NA
	Reactor coolant filter	10-15	NA
	Seal water filter	1-10	NA
	Spent fuel pit filter	5	NA
	Seal water injection filter	5-7	NA
	Waste evaporator feed filter	1-2	NA
	Spent fuel pit skimmer filter	4-5	NA
	Reactor cavity cleanup filter	4-5	NA
	Boric acid filters	2-3	NA

1. Not available.

Table 4.2-38 Contact and 3-Foot Dose Rates (mrem/hr) From PWR Wastes as Shipped

Plant	Concentrated Liquids		Resin		Filter Sludge		Cartridge Filters	
	Contact	3 ft	Contact	3 ft	Contact	3 ft	Contact	3 ft
Plants Without Condensate Polishing Systems								
P1	-	-	< 50	-	-	-	~ 1 R/hr	
P2	50-150	5-15	8-75	3-10	-	-	10 mr/hr-45 R/hr	3 mr/hr-2.5 R/hr
P3	60-100	3-10	10	2-3	-	-	5-20 (1)	1-5 (1)
P4	30 mr/hr-50 R/hr	1 mr/hr-1 R/hr	(10-200) R/hr	(1-50) R/hr	-	-	100 R	10 R
P5	50-150	10-30	20-50 (2)	5-20 (2)	-	-	2-4 R	50-200
P6	150-200	10-100	NA	NA	-	-	NA	NA
P7	100-300	15-50	18-25 R (3)	1-4 R (3)	-	-	20-1500 (3)	4-150 (3)
P8	7-10	1.5-3	(4)	(4)	-	-	600-30 R	50 mr/hr-2 R/hr
P9	200-300	20-30	.1 R-2 R	10-200	-	-	(5-10) R/hr	300
P10	100	5-10	15-70 (5)	1-10 (5)	-	-	10 mr-20 R (5)	< 5 mr-3 R (5)
Plants With Condensate Polishing System								
P11	200-800	5-75	100	10	-	-	1 R-2 R	200
P12	-	-	1.5-400	.5-30	-	-	10-120	5-10
P13	50-100 (6)	4 (at 6 ft) (6)	20-25	5-6	-	-	.5 R-1.8 R	50-300
P14	25	3.5	(5-10) R/hr	(.5-1) R/hr	-	-	20 R-40 R	(.5-5) R/hr
P15	1-2 (7)	.2-1.5 (7)	6-35 (8)	3-4 (8)	-	-	11 (8)	4 (8)
P16	10-50	< 5	10-200	< 10	-	-	200	50
P17	7.3 (9)	2.4 (9)	NR (10)	NR (10)	-	-	NA	NA
P18	100-650	25-180	(4)	(4)	10 (11)	3 (11)	10	7

1. Shielded with Chem Nuclear Cask, unshielded doses (55-gal drum) are 1-80 R/hr contact and 50mr-15 R/hr at 3 ft.
2. Shielded with Atcor LL-50 Shield, unshielded doses liner (100 ft^3) are 10-100 R/hr contact and 1.5-12 R/hr at 3 ft.
3. Unshielded doses. Shipped in Atcor BC-48-220 Cask, holds 14-55 gal drums.
4. Plant has not yet shipped this type of waste.
5. Unshielded doses, shielding is Chem Nuclear CNS-15-160B when needed.
6. Dose rates for unsolidified liquid waste shipped to burial sites.
7. Shielded with Chem Nuclear M-189 Cask, unshielded doses (50 ft^3 liner) are 45 mr/hr contact and 14 mr/hr at 3 ft.
8. Shielded with 2-in. lead.
9. Average dose rates, unshielded.
10. Not recorded.
11. Shielded with 2 1/2-in. lead. Unshielded doses are 1.2 R/hr to 2.0 R/hr contact and 120 mr/hr to 250 mr/hr at 3 feet.

Table 4.2-39 Material Shipped as PWR Compactible and Noncompactible Radwaste[1]

Radwaste Material	P1		P3		P4		P5		P6		P7		P8		P9		P10		P11		P12		P13		P14		P15		P16	
	C[2]	N[3]	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N
ANTI Cs	X		X		NA[5]		X		X		(4)						X		X		X		X		X		X			
Conduit														X																
Contaminated tools and Equipment				X				X										X				X								
Valves																						X								X
Pipes																		X		X		X								
Spent fuel racks																								X						
Lighting equipment														X																
Glassware																X														
High density block														X																
Hoses														X																
Instrument channels														X																
Insulation																X		X												
Irradiated components								X		X																				
In-core detectors																										X				
Fuel assembly post						X																				X				
Shim rods						X																								
Flux wires																														X
Laboratory equipment													X										X	X						
Ladders														X						X										
Low level air filters				X															X		X				X	X				X
Miscellaneous metals		X						X		X						X				X								X		
Miscellaneous woods		X		X				X	X	X	X					X		X		X								X		
Mop heads												X					X													
Paper	X		X						X				X			X				X		X		X		X		X		X
Blotters								X																						
Suits								X																						
Kraft																	X													
Plastic	X		X										X			X									X					
Gloves								X																			X			
Bags								X				X					X													
Shoe covers								X				X																		

Table 4.2-39 Material Shipped as PWR Compactible and Noncompactible Radwaste[1] (Cont'd)

Radwaste Material	P1		P3		P4		P5		P6		P7		P8		P9		P10		P11		P12		P13		P14		P15		P16	
	C[2]	N[3]	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N
Rags	X		X				X		X		X				X		X		X		X				X				X	
Scaffolding															X															
Test equipment																														
Vessel inspection equipment																									X					
Eddy current equipment																									X					

1. Plant P2 reported in Table 4.2-40; Plants P17 and P18 did not have a data breakdown available.
2. Compactible.
3. Noncompactible.
4. Material normally considered noncompactible is cut and packaged with compactible material.
5. Not available.

Table 4.2-40 Material Shipped as Compactible and Noncompactible Radwaste From PWR Facility P2

Compactible[1]	Noncompactible[1]
Paper[2]	Piping[3]
Plastic bags[2]	Scaffolding[3]
Respirator cartridge[2]	Ladders[3]
Protective clothing[2]	Valves
Cotton	Seals
Nylon	Packing Matter
Tyvec	Old Motors
Rubbers	Pumps
Cement bags	Vacuum cleaner
Scintillation vials	Electrical cable
Glass	Welding leads
Plastic	Hoses (rubber)
Chemical laboratory equipment	Water
Glassware	Air
Plastic bottles	
Rope	
Polyethylene	
Hemp	
Nylon	
Tape	

1. The utility estimates that 80% of the waste is compactible and that the remaining is noncompactible.

2. The utility estimates that this material is 80% of the compactible waste.

3. The utility estimates that this material is 65 to 70% of the noncompactible waste.

Table 4.2-41 PWR Trash - Compactible and Noncompactible Volumes and Activity

Plant	1971		1972		1973		1974		1975		1976		1977	
	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)	Activity (Ci)
Plants Without Condensate Polishing Systems														
P1			2,920		3,770		1,910		2,474		4,092			
P2							2,888	4.49	5,402	51.9	4,911	2.47	2,952	3.18
P3					13,200	201.	13,200	201.	13,200	201.	13,200	201.	13,200	201.
P4					972	1.04	2,940	8.1	2,998	10.7	2,154	4.9	4,757	
P5	2,075	1.6	4,849	2.8	5,266	347.	6,435	73.	21,175	15.4	30,111	46.7	50,155	46.
P6									6,730		11,965	50.	23,239	73.7
P7									357	1.51	9,898	12.6	431	2.6
P8											3,245	23.9	11,009	59.8
P9					14,706	.16	57,026	4.6	55,613	15.4	71,326	68.	69,702	225.
P10							1,430		2,013		3,035		22,260	458.
Plants With Condensate Polishing Systems														
P11	2,312		5,743		1,938		4,855		4,756	1.09	3,636	5.93	3,431	25.4
P12											2,337	.14	10,209	64.6
P13									1,066	.43	1,927	.27	10,039	8.4
P14					1,830	1.3	7,370	3.94	1,900		7,338	47.6	4,515	6.17
P15											1,073	.207	2,540	5.89
P16					9,360	32.	20,519	221.5	47,560	405.	36,128	582.	44,910	
P17														
P18											3,514	2.34	2,330	1.16

From the data obtained, reason for the significantly different annual production rates of compactible and noncompactible radwaste reported at facilities P5, P9, and P16 cannot be identified. Limited information about waste volumes at facilities P5 and P9 is as follows:

1. Facility P5 - Of the 7 years of data received from this facility, 3 years of the data are included in the 28 reactor-years being questioned as being unrepresentative of the PWR power plants. With respect to these three years of data, officials of the utility specifically pointed out that the waste consisted mostly of noncompactible material.

2. Facility P9 - This facility reports that the density of the compactible radwaste is 12 lb/ft^3. It appears that at this facility very little compaction of the compactible radwaste is performed.

A detailed evaluation of 88 reactor-years of data provided the following information:

1. The annual average compactible and noncompactible radwaste production rate for all plants is 8,800 ft^3/yr.

2. The annual average compactible and noncompactible radwaste production rate for

   a. Plants with less than 750 MWe generating capacity (28 reactor years of data)[1] is 5,730 ft^3.

   b. Plants with greater than or equal to 750 MWe generating capacity (60 reactor years of data) is 8,700 ft^3.

These are weighted averages based on the number of years the plant has operated. Facility P5 is the only plant for which tenth-year data are available. These data, already identified as being unusually high, would increase the average generation rate for plants with a generating capacity less than 750 MWe by 77% if included in the analysis. The difference between the two size classifications of facilities is approximately 53%, more than double the 21% difference of BWRs.

The average generation rate is 8,800 ft^3 and the median value is only 4,849 ft^3/yr. Again this indicates that the distribution is skewed to the right and that a few high numbers have a dominating effect on the average.

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1. This is not the same 28 reactor-years of data discussed in the beginning of this section as coming from facilities P5, P9 and P16.

Removing the data from plants P5, P9, and P16 from consideration of a typical plant results in the following evaluation:

1. The annual average compactible and noncompactible radwaste production rate for all plants is 4,400 ft^3.

2. The annual average compactible and noncompactible radwaste production rate for

   a. Plants with less than 750 MWe generating capacity (22 reactor-years of data) is 3500 ft^3.

   b. Plants with a generating capacity greater than or equal to 750 MWe (37 reactor years of data) is 5,200 ft^3.

Again the median generation rate is less than the average but much closer than in the case of the average plant analysis. For the typical plant analysis the median radwaste production rate is 3,470 ft^3/yr which is only 21% less than the average radwaste production rate, which is 4,400 ft^3/yr. For the average plant the median quantity of radwaste generated is 45% less than the average waste production.

When the data for the average plant are evaluated in terms of the annual generation per MWe of installed capacity, the average plant produces 11.5 ft^3/MWe-yr. This is the same quantity of waste produced by an average BWR. The typical plant generation rate is 6.5 ft^3/MWe-yr.

Out of the 89 reactor-years of data on the total volume of compactible and noncompactible waste, 48 of those years provided separate data for each waste type. During these 48 reactor-years a total of 230,000 cubic feet of compactible and 116,000 cubic feet of noncompactible waste was reported. Thus, the compactible trash accounts for 66% of the total volume of trash. The reported data are given in Table 4.2-42.

Specific data on the quantity of combustible as opposed to noncombustible trash were not available. Based on the data on compactible versus noncompactible waste, it appears that almost all of the compactible waste is combustible with the exception of spray cans and small tools. Only a small fraction of noncompactible waste is combustible, such as wood. Based on these observations it is assumed that 66% of the total trash volume is also combustible and 34% is not combustible.

The total activity associated with each year's volume of compactible and noncompactible waste is given in Table 4.2-41 with the distribution of this activity between the compactible and the noncompactible waste given in Table 4.2-43. Based on the 52 years of data available in this form the average concentration of activity is 0.0055 Ci/ft^3. For BWRs the average concentration 0.0048 Ci/ft^3. Furthermore, 7.7% of the annual activity is associated

Table 4.2-42 Reported PWR Compactible and Noncompactible Radwaste Volumes (ft^3) Generated per Calendar Year

Facility		1971	1972	1973	1974	1975	1976	1977
P1	Compactible[1]	ND[2]	2,920	3,770	1,910	2,474	4,092	(3)
	Noncompactible[4]	ND	IWC[5]	IWC	IWC	IWC	IWC	
P2	Compactible	ND	ND	ND	2,888	5,402	4,911	2,952
	Noncompactible	ND	ND	ND	IWC	IWC	IWC	IWC
P3	Compactible			8,085[6,7]	8,085[6,7]	8,085[6,7]	8,085[6,7]	8,085[6,7]
	Noncompactible			5,083	5,083	5,083	5,083	5,083
P4	Compactible			972[8]	2,940	2,998	2,154	2,893
	Noncompactible			IWC	IWC	IWC	IWC	1,864
P5	Compactible	2,075	4,849	5,266	4,093	21,175	30,111	50,155
	Noncompactible	IWC	IWC	IWC	2,342	IWC	IWC	IWC
P6[9]	Compactible	ND	ND	ND	ND	6,730	4,813	8,799
	Noncompactible	ND	ND	ND	ND	IWC	7,152	14,440
P7	Compactible	ND	ND	ND	ND	357	9,898	431
	Noncompactible	ND	ND	ND	ND	IWC	IWC	IWC
P8	Compactible					ND	2,485	9,607
	Noncompactible					ND	760	1,402

Table 4.2-42 Reported PWR Compactible and Noncompactible Radwaste Volumes (ft^3) Generated per Calendar Year (Cont'd)

Facility		1971	1972	1973	1974	1975	1976	1977
P9	Compactible	ND	ND	14,706	57,025 (6)	55,613 (6)	71,326 (6)	69,702 (6)
	Noncompactible	ND	ND	IWC	IWC	IWC	IWC	IWC
P10	Compactible	ND	ND	ND	1,154	2,639 (6)	2,367 (6)	22,260 (6)
	Noncompactible	ND	ND	ND	276	366	668	IWC
P11	Compactible	1,544	3,183	978	2,359	2,132	2,036	1,131
	Noncompactible	768	2,560	960	2,496	2,624	1,600	2,304
P12	Compactible	ND	ND	ND	ND	ND	2,049	6,648 (6)
	Noncompactible	ND	ND	ND	ND	ND	288	3,561
P13	Compactible	ND	ND	ND	ND	1,066	647	4,806
	Noncompactible	ND	ND	ND	ND	IWC	1,280	5,233
P14	Compactible	ND	ND	1,830	7,200	1,350	3,138	3,515
	Noncompactible	ND	ND	IWC	170	550	4,200	1,000
P15	Compactible	ND	ND	ND	ND	ND	1,073	2,540
	Noncompactible	ND	ND	ND	ND	ND	IWC	IWC

Table 4.2-42 Reported PWR Compactible and Noncompactible Radwaste Volumes (ft^3) Generated per Calendar Year (Cont'd)

Facility		1971	1972	1973	1974	1975	1976	1977
P16	Compactible	ND	ND	9,360	20,169[6]	45,000[10]	28,878[10]	22,910[10]
	Noncompactible	ND	ND	IWC	350	2,560	7,250	22,000
P17[11]								
P18	Compactible	ND	ND	ND	ND	ND	1,300	2,330
	Noncompactible	ND	ND	ND	ND	ND	2,214	0

Footnotes for Tables 4.2-42 Through 4.2-47

1. Compactible waste.
2. No data; facility either did not have data or it had not commenced commercial operation.
3. A full year's data was not available at the time of this survey.
4. Noncompactible waste.
5. The volume of or the radionuclides in noncompactible waste was included with reported compactible wastes.
6. Data for two units.
7. Data were prepared for the manufacturer of the facility's nuclear steam supply system. They are claimed to be representative of a typical year's volume. For this study, only one year's data are used.

Footnotes for Tables 4.2-42 Through 4.2-47 (Cont'd)

8. Year facility went on line; data are not for a full operating year and are not included in this evaluation.

9. This unit shares a site with a unit that has not operated for a few years. Work continued within the shutdown unit, and waste generated within the shutdown unit is shipped out of the operating unit. Since the situation with the shutdown unit is not typical and these volumes cannot be separated from the wastes of the operating unit, these data are not included in this evaluation.

10. Data for three units.

11. This unit's waste is shipped from another unit at the same site. The other unit is a BWR. No attempt is made at the facility from which the waste is shipped to monitor separately the waste shipped from each unit. No useful data exist for this facility.

12. Facility personnel believe that this value includes activity associated with liquid radwaste filters and does not represent just trash activity.

13. Shim rods account for 10,250 Ci.

14. Utility only kept records on total activity shipped (process waste plus trash). Data obtained are not useful for this phase of the study.

15. Flux wires account for a large portion of total activity.

16. Mixed fission products.

17. Mixed corrosion products.

18. Material normally considered noncompactible is cut up and packaged with compactible material.

19. Values assumed to be the same as those for noncompacted radwaste.

Table 4.2-43 Reported Activity (Ci) Shipped with PWR Compactible and Noncompactible Radwaste Per Calendar Year

Facility		1971	1972	1973	1974	1975	1976	1977
P1	Compactible	ND[2]	ND	ND	ND	ND	ND	ND
	Noncompactible	ND	ND	ND	ND	ND	ND	ND
P2	Compactible[1]	ND	ND	ND	4.5	51.9[12]	2.47	3.18
	Noncompactible[4]	ND	ND	ND	IWC[5]	IWC	IWC	IWC
P3	Compactible	ND	ND	ND	ND	ND	0.22[6,7]	ND
	Noncompactible	ND	ND	ND	ND	ND	201	ND
P4	Compactible	ND	ND	1.04[8]	8.09	10.7	4.9	7.69
	Noncompactible	ND	ND	IWC	IWC	IWC	IWC	10,255[13]
P5	Compactible	1.6	2.83	347	50.4	15.4	46.7	46.0
	Noncompactible	IWC	IWC	IWC	22.6	IWC	IWC	IWC
P6	Compactible	ND	ND	ND	ND	219	49.1	72.2
	Noncompactible	ND	ND	ND	ND	IWC	0.87	1.49
P7	Compactible	ND	ND	ND	ND	1.51	12.6	2.64
	Noncompactible	ND	ND	ND	ND	IWC	IWC	IWC
P8	Compactible	ND	ND	ND	ND	ND	22.3	42.5
	Noncompactible	ND	ND	ND	ND	ND	1.60	17.3
P9	Compactible	ND	ND	0.16[8]	4.6	15.4[6]	68.0[6]	225[6]
	Noncompactible	ND	ND	IWC	IWC	IWC	IWC	IWC

Table 4.2-43 Reported Activity (Ci) Shipped with PWR Compactible and Noncompactible Radwaste Per Calendar Year (Cont'd)

Facility		1971	1972	1973	1974	1975	1976	1977
P10	Compactible	ND(14)	ND	ND	ND	ND	ND	ND
	Noncompactible	ND	ND	ND	ND	ND	ND	ND
P11	Compactible	ND	ND	ND	ND	1	1.00	25.00
	Noncompactible	ND	ND	ND	ND	0.09	4.93	0.35
P12	Compactible	ND	ND	ND	ND	ND	0.14	1.01
	Noncompactible	ND	ND	ND	ND	ND	0	63.6
P13	Compactible	ND	ND	ND	ND	0.43	0.22	1.20
	Noncompactible	ND	ND	ND	ND	IWC	0.05	7.23
P14	Compactible	ND	ND	1.30	3.74	0.90	4.59	5.17
	Noncompactible	ND	ND	IWC	0.20	0.10	43.0	1
P15	Compactible	ND	ND	ND	ND	ND	0.21(8)	5.89
	Noncompactible	ND	ND	ND	ND	ND	IWC	IWC
P16	Compactible	ND	ND	32.0(8)	218	378(10)	300(10)	222(10)
	Noncompactible	ND	ND	IWC	3.50	26.2	282	5,466(15)
P17(11)								
P18	Compactible	ND	ND	ND	ND	ND	1.96	1.16
	Noncompactible	ND	ND	ND	ND	ND	0.38	0

with the compactible waste and the remaining 92.3% with the non-compactible waste. For the generation rate of 11.5%/MWe-yr the compactible waste accounts for 66% of the volume and 7.7% of the activity at a concentration of 640 Ci/ft^3, whereas the non-compactible waste, containing 92.3% of the activity and 34% of the volume, has a concentration of 0.015 Ci/ft^3.

Because of the large difference between the volume-generation rates of the average plant and the typical plant, the activity level and concentrations were calculated again excluding the data from plants P5, P9, and P16. The average concentration is slightly less (0.0045 Ci/ft^3); but with almost half the volume of waste being produced, the total number of curies produced is also reduced by half, to 0.029 Ci/MWe-yr. The reevaluation of the relative composition of the waste reveals that 57% of the waste is compactible containing 2.2 of the activity at a concentration of 170 Ci/ft^3. The reevaluation also reveals that 43% of the waste is noncompactible containing 97.8% of the activity at a concentration of 0.010 Ci/ft^3. The corresponding activity generation rates are 0.00064 Ci/MWe-yr for compactible waste and 0.028 Ci/MWe-yr for noncompactible waste. These values are summarized in Section 4.2.3.

The radionuclides found in PWR trash should be representative of the longer-lived isotopes including activated corrosion products and fission products. Noncompactible trash such as shim rods, flux wires, and other items from the reactor core are expected to be predominantly activated corrosion products. Materials contaminated as a result of contact with primary coolant during maintenance or refuelings would show a representative mix of nuclides found in the reactor coolant. The data as collected in the survey are given in Table 4.2-44.

The containers that PWR plant officials reported being used for the shipment of noncompactible radwaste are listed in Table 4.2-45.

Container sizes for compactible radwaste were not listed because 55-gallon and 83-gallon drums were the only sizes of containers used for compactible waste.

The density of PWR compacted radwaste was usually between 20 lb/ft^3 and 40.8 lb/ft^3 (150 to 300 pounds per 55-gallon drum). There were few exceptions to this range.

Densities of noncompactible radwaste are somewhat misleading. The density of noncompactible radwaste is dependent on the packing efficiency or percent void volume remaining after packaging. Typically, the density of noncompactible radwaste after packaging is lower than the density for compactible radwaste after packaging. However, the possibility exists for compactible radwaste to have very high densities. The reported densities of PWR compactible and noncompactible radwaste are in Table 4.2-46.

Table 4.2-44 Radionuclides Identified as Being Present in PWR Compactible and Noncompactible Radwastes

Radionuclides	P1		P2		P3		P4		P5		P6		P7		P8		P9		P10		P11		P12		P13		P14		P15		P16		P17		P18	
	C[1]	N[4]	C	N[5]	C	N	C	N	C	N	C	N	C	N[5]	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N	C	N
Na-24															X																					
Cr-51			X					X																	X	X							NA[3]		NA[3]	
Mn-54	X	X	X					X	X	X	X	X			X		X	X		X					X	X										
Fe-55								X																												
Fe-59	X	X						X																												
Co-57			X																																	
Co-58	X	X	X				X	X	X	X	X	X			X				X	X	X	X			X				X							
Co-60	X	X	X				X	X	X	X	X	X			X		X	X	X	X	X	X	X	X	X	X	X	X	X	X						
Zr-95								X																												
Zr-97			X																																	
No-95			X																						X											
Nb-97			X																																	
Ag-110m																										X										
Sn-113																										X										
Sb-124											X	X																								
Sb-125			X																																	
I-131															X																					
Cs-134	X	X	X				X	X			X	X			X		X	X		X	X	X									X	X				
Cs-137	X	X	X				X	X	X	X	X	X			X		X	X	X	X	X	X			X	X	X	X			X	X				
MFP[16]					X	X							X			X			X				X	X				X								
MCP[17]					X	X							X						X				X	X												

Table 4.2-45 Containers Used in the Shipment and Burial of Noncompactible Radwaste

Facility	Container Description
P1	7 ft x 4 ft x 6 ft plywood boxes
P2	55-gal drums
P3	55-gal drums
P4	CNSI-13 #1600, NFS-4 cask, 112-ft^3 crates
P5	4 ft x 4 ft x 8 ft plywood boxes
P6	4 ft x 4 ft x 8 ft plywood boxes
P7	(18)
P8	55-gal drum
P9	7 ft x 4 ft x 4 ft wooden crates
P10	4 ft x 4 ft x 8 ft wood boxes, 4-ft^3 cardboard boxes
P11	Plywood box (DOT), double banded, 4 ft x 4 ft x 4 ft
P12	DOT wooden boxes, 2 ft x 2 ft x 2 ft, 4 ft x 4 ft x 4 ft, 2 ft x 2 ft x 8 ft
P13	Wooden boxes, 8 ft x 8 ft x 20 ft, 4 ft x 4 ft x 5 ft, and several others of varying sizes
P14	4 ft x 4 ft x 8 ft, with a reported range in other containers of 0.1-ft^3 - 4,000-ft^3
P15	55-gal drums
P16	DOT wooden boxes 4 ft x 4 ft x 8 ft
P17	ND(2)
P18	55-gal drums, 4 ft x 4 ft x 8 ft wooden boxes and others of varying sizes

Table 4.2-46 Reported Density (lb/ft^3) of Compactible and Noncompactible PWR Radwaste

Facility	Compactible	Noncompactible
P1	27.2 - 40.8	Varies because of different waste forms
P2	40.8	(19)
P3	18.7 - 31.2	13.6 - 27.2
P4	19.7 - 43.4	ND(2)
P5	20.4 - 40.8	15.6 - 19.5
P6	ND	15
P7	25 - 37.4	(18)
P8	27.2	ND
P9	11.9	11.9
P10	40.8	12.5 - 31.2
P11	27 - 40.8	25
P12	10 - 40	Varies
P13	6.2	5 - 31.2
P14	27.2	27.2 - 272
P15	20 - 34.5 One value as high as 298	(19)
P16	ND	ND
P17	ND	ND
P18	20.4 - 27.2	ND

Data were collected on radiation levels in contact with and at 3 feet from containers of compactible and noncompactible radwaste. Based on the data collected, a general statement can be made that the dose at 3 feet is usually a factor of 4 (for lower dose rates) to 10 (for higher dose rates) lower than the contact dose rate.

The data collected in the survey did not reveal any specific reason for this dependence on source strength, however, a number of possible answers do exist. One possibility is that the higher dose rates are due to local hot spots in the waste which appear as point sources for contact dose rate measurements. When the detector is moved back 3 feet from the container the waste no longer appears as a point source but as a cylindrical source. The average concentration of the cylindrical source is much less than the point-source hot spot, and therefore, the dose rate drops off faster than it would for a container without a hot spot. The other possibility is that the background radiation levels in the area where the measurements are taken are of the same order of magnitude as the dose rates associated with the low-activity wastes. In this case, when the detector is moved away from the low-dose-rate containers, what is actually being picked up by the detector is to a large degree the general background radiation and not due to the drum itself.

For PWRs the upper limit of the contact dose rate on compactible and noncompactible radwaste is usually less than 200 mrem. However, contact dose rates ranging from 1,000 mrem/hr to 10,000 mrem/hr were also reported for compactible radwaste. It is expected that such dose rates would be associated with noncompactible radwaste originating from the reactor system or reactor internals. The dose rates may be attributed to compactible radwaste for the following reasons:

- Noncompactible radwaste is also packaged in 55-gallon drums. Once capped, a definite identification of radwaste type cannot be made.

- Compactible and noncompactible radwaste are commonly packaged together and reported as compactible radwaste.

Data collected on radiation levels are presented in Table 4.2-47.

4.2.3 LWR Summary

This section is a summary of the topics discussed for all waste types in Sections 4.2.1 and 4.2.2. Topics discussed include volumes of waste, total radioactivity in the waste, and individual radionuclides reported in the waste. Other topics such as resins used in deep bed demineralizers, are applicable to a discussion of only one waste type and are not summarized here.

Table 4.2-47 Reported Radiation Levels (mrem/hr) from Compactible and Noncompactible Radwaste

Facility	Compactible		Noncompactible	
	3 ft	Contact	3 ft	Contact
P1	ND[2]	200 (max.)	ND	200 (max.)
P2	<1	<15	(19)	(19)
P3	5 - 50	30 - 200	ND	ND
P4	1 - 20	4 - 100	2	10
P5	1 - 5	10 - 20	5 - 20	10 - 30
P6	1 - 10	1 - 50	<1 - 25	1 - 200
P7	<1 - 10	20 - 50	(18)	(18)
P8	<0.1	0.1 - 20	1	10
P9	10	100	1 - 2	10
P10	5 - 10	20 - 30	5	20
P11	1 - 25	1 - 300	1 - 3	1 - 10
P12	0.05 - 50	0.05 - 2000	0.05 - 10	0.05 - 90
P13	0.1 - 15	0.1 - 100	2	10
P14	0.1 - 2	10	1 - 20	10 - 100
P15	0.05 - 100	0.05 - 1000	(19)	(19)
P16	0.1 - 10	0.5 - 150	<10	<200
P17	ND	ND	ND	ND
P18	0.2 - 5	0.2 - 140	0.15 - 1	1 - 50

4.2.3.1 Waste Volumes and Activities

Tables 4.2-48 through 4.2-51 summarize the volume and activity generation rates for the average facility surveyed and the typical plant. Throughout the analysis, waste volumes and activities that were determined to be unrepresentative of typical plant operations were without exception higher than average waste volumes and activities. This results in the typical plant generating less waste by volume in every case and less total activity in all but one case. The volume increase due to solidification and the effects of various volume reduction techniques are discussed in Chapter 5.

4.2.3.2 Radionuclides Reported in LWR Wastes

Previous studies (Phillips, 1977; Bell, 1977), based on operating plants' semiannual effluent release reports, reported the radionuclides that are predominant in Category A waste as defined by Regulatory Guide 1.2.1. As reported, the volume of Category A waste is the total volume of the spent resin, filter wastes, and evaporator bottoms shipped offsite during the reporting period. The same waste sources are used to give the total reported activity. At the onset of this study, it was assumed that the radionuclides reported in resin wastes would be predominantly fission products, whereas cartridge filters and filter sludge would be predominantly the activated corrosion products. Radionuclides in evaporator bottoms were expected to be either the same as those on resins for plants that regenerate resins or essentially the same as those found in compactible trash for the remainder of the plants.

The five predominant radionuclides, in virtually every type of waste for BWRs and PWRs, were Mn-54, Co-58, and Co-60, which are insoluble activated corrosion products, and Cs-134 and Cs-137, which are soluble fission products. A composite list of the various radionuclides found in LWR wastes is given in Table 4.2-52. This table also indicates which radionuclides were most frequently reported.

Viewing the data presented in Table 4.2-52 the following additional observations can be made:

1. Three of the five most commonly reported radionuclides (Co-60, Cs-134, Cs-137) have half-lives greater than 1 year. The half-life of Mn-54 is 313 days, and the half-life of Co-58 is 71.4 days.

2. The radionuclide, Zn-65, is found in all types of BWR waste. It is one of the most predominant isotopes in spent resins, precoat filter sludge, and trash, but it is not found in any of the PWR waste. The half-life of this activated corrosion product is 245 days.

Table 4.2-48 Average Plant Untreated Waste Volumes

Waste Type	Waste Volumes (ft^3/MWe-yr) Boiling Water Reactors Deep Bed CPS[1]	Boiling Water Reactors Precoat CPS	Pressurized Water Reactors Without CPS	Pressurized Water Reactors With CPS
Deep bed resin	4.6	0.23	0.94	0.32
Concentrated liquids	12.7	0.6	3.9	4.8
Filter sludge	5.4	7.7	-	.015
Cartridge filters	-	-	0.39	0.39
Trash				
Total	11.5	11.5	11.5	11.5
Compactible	7.8	7.8	7.6	7.6
Noncompactible	3.7	3.7	3.9	3.9
Total	34.2	20.0	16.7	17.2
Annual volume (ft^3/yr) for a 1,000 MWe plant	34,200	20,000	16,700	17,200

1. Condensate polishing system.

Table 4.2-49 Average Plant Waste Activity

Waste Type	Waste Activity (Ci/MWe-yr)			
	Boiling Water Reactors		Pressurized Water Reactors	
	Deep Bed CPS(1)	Precoat CPS	Without CPS	With CPS
Deep bed resin	1.9	.0014	0.61	0.2
Concentrated liquids	0.58	0.016	0.20	0.024
Filter sludge	2.0	0.5	-	0.012
Cartridge filters	-	-	0.12	0.12
Trash				
Total	0.402	0.402	0.063	0.063
Compactible	0.0052	0.0052	0.0049	0.0049
Noncompactible	0.397	0.397	0.058	0.058
Total	4.88	0.92	1.00	0.42
Annual activity (Ci/yr) for a 1,000 MWe plant	4,880	920	1,000	420

1. Condensate polishing system.

Table 4.2-50 Typical Plant Untreated Waste Volumes

Waste Type	Untreated Waste Volumes (ft^3/MWe-yr)			
	Boiling Water Reactors		Pressurized Water Reactors	
	Deep Bed CPS[1]	Precoat CPS	Without CPS	With CPS
Deep bed resin	4.6	0.23	0.94	0.32
Concentrated liquids	8.1	0.6	2.6	4.8
Filter sludge	5.4	7.7	-	0.15
Cartridge filters	-	-	0.39	0.39
Trash				
Total	10.6	10.6	6.5	6.5
Compactible	7.2	7.2	3.7	3.7
Noncompactible	3.4	3.4	2.8	2.8
Total	28.7	19.1	10.4	12.2
Annual volume (ft^3/yr) for a 1,000 MWe plant	28,700	19,100	10,400	12,200

1. Condensate polishing system.

Table 4.2-51 Typical Plant Waste Activity

Waste Type	Waste Activity (Ci/MWe-yr) Boiling Water Reactors Deep Bed CPS[1]	Boiling Water Reactors Precoat CPS	Pressurized Water Reactors Without CPS	Pressurized Water Reactors With CPS
Deep bed resin	1.9	0.0014	0.61	0.2
Concentrated liquids	0.45	0.016	0.21	0.024
Filter sludge	2.0	0.5	-	0.012
Cartridge filters	-	-	0.12	0.12
Trash				
Total	0.402	0.402	0.029	0.029
Compactible	0.0052	0.0052	0.00064	0.00064
Noncompactible	0.0397	0.397	0.0284	0.0284
Total	4.75	0.92	0.97	0.385
Annual activity (Ci/yr) for a 1,000 MWe plant	4,750	920.	970.	385.

1. Condensate polishing system.

Table 4.2-52 Radionuclides Reported in LWR Wastes by Plant With Most Detail

	BWRs					PWRs				
	Resins	Precoat Filter	Cartridge Filter	Evaporator Bottoms	Trash	Resins	Precoat Filter	Cartridge Filter	Evaporator Bottoms	Trash
Na-24				X						
Cr-51	X	X	X	C	X			C	X	C
Mn-54	C	C	C	C	C	C	C	C	C	C
Fe-59	X	X	X	X	X			X		
Co-57						X			X	X
Co-58	C	C	X	C	C	C	C	C	C	C
Co-60	C	C	C	C	C	C	C	C	C	C
Zn-65	C	C	X	X	C					
Sr-89				X						
Sr-90				X						
Sr-91	X									
Zr-95	X	X	X	X	X			X		
Zr-97									X	X
Nb-95	X	X	X	X	X	X		X	X	X
Nb-97										X
Mo-99						X				X
Tc-99m	X	X								
Ru-103		X								
Ru-106		X								
Ag-110m		X			X	X			X	
Sb-124				X					X	
Sb-125									X	X
I-131	X	X		X	X	X	C		C	
I-133	X	X				X				
Cs-134	C	C	C	C	C	C		C	C	C
Cs-136		X				X				
Cs-137	C	C	C	C	C	C	C	C	C	C
La-140	X	X		X						
Ba-140	X	X								
Ce-141	X	X		X						
Ce-144	X	X								

Key

X = Radionuclide reported.
C = Most commonly reported radionuclide by all plants.

3. In BWRs, four other nuclides are found in all types of waste although they are not generally found in PWRs. They are Cr-51 ($t_{1/2}$ = 28 days), Fe-59 ($t_{1/2}$ = 45 days), Zr-95 ($t_{1/2}$ = 65.5 days), and Nb-95 ($t_{1/2}$ = 35.1 days).

4. Co-57, with a half-life of 271 days, is found in PWR spent resins, evaporator bottoms, and trash, but it was not reported by personnel at any of the BWRs surveyed.

5. Except for the presence of Zn-65 in BWRs, the same radionuclides are found on cartridge filters, in BWRS and PWRs. All the radionuclides reported are activated corrosion products except Cs-134 and Cs-137.

6. The radionuclides reported in the compactible and noncompactible wastes are generally the same as those in the cartridge filters but with a few more fission products. This is true for both PWRs and BWRs.

7. In PWRs, where there is no concentration of regeneration chemicals, from regenerative demineralizers, the radionuclides in the concentrator bottoms are basically the same as those in the compactible and noncompactible trash. These radionuclides are from surface contamination of materials that end up in the trash or from surface contamination that was removed during equipment decontamination and is subsequently concentrated.

8. In BWRs, where demineralizer resins are regenerated, the radionuclides in the concentrator bottoms are primarily the same as those reported in the resins.

9. Data from PWR precoat filters are too limited to draw any conclusions regarding the nuclides typically found in precoat filter media.

10. BWR precoat filters, many of which use ground ion-exchange resin as a filter media, produce wastes that basically contain the same radionuclides as BWR resins. Most plants surveyed reported that precoat filter wastes and spent resins contain Mn-54, Co-58, Co-60, Zn-65, Cs-134, and Cs-137. Other nuclides that have been reported are given in Table 4.2-52.

11. The list of radionuclides found or PWR resins is not as extensive as the one for BWR resins, but does show the same basic mixture of activated corrosion products and fission products.

With respect to the radionuclides reported to be found in the various types of waste the following must be borne in mind: The listing in Table 4.2-52 only indicates which radionuclides were reported. Those radionuclides that are indicated to be predominant are predominant only with respect to the number of plants that reported those radionuclides. Although it can be assumed that these radionuclides will also be predominant in terms of their isotopic concentration in the waste (Ci/ft^3), the data do not provide specific concentrations. Also, survey results show that few actual samples are taken to determine the isotopic distribution in the waste or even the total activity contained within a single container. At most plants the total activity within a container is determined by measuring the dose rate at some predetermined distance. Using this dose rate, the equivalent curies of one specific isotope is calculated. The remaining isotopes are calculated based on some standard isotopic mix that may have been calculated for an actual sample taken previously.

4.3 Transuranic Radionuclides in LWR Wastes

4.3.1 Concentration of Transuranic Radionuclides in LWR Wastes

As part of the survey process, officials at each facility were asked two questions concerning transuranic radionuclides in plant low-level wastes:

- Are wastes generally checked for alpha contamination?
- If wastes are generally checked for alpha contamination, what is the typical reading, and what percentage of the waste volume exhibits concentration in excess of 10 nCi/g.

Representatives of seven facilities responded affirmatively to the first question. Answers to the second question were diverse. Personnel from two plants did not respond. Officials from three plants said activity was undetectable. A spokesman from one plant said the typical reading and percentage of waste volumes with concentrations in excess of 10 nCi/g was zero. At another plant an employee said the typical reading is less than 10 nanocuries per gram and that zero percent of the waste volume exceeds 10 nCi/g. These results are in Table 4.3-1.

A 1977 report by Dames and Moore (1977) for the U.S. Environmental Protection Agency (EPA) found alpha contamination in LWRs ranging from 2×10^{-7} $\mu Ci/ml$ to 2×10^{-3} $\mu Ci/g$ (note difference in units). This report covered four reactors: two BWRs and two PWRs in New York State.

A 1978 report by Science Application Incorporated (SAI) (Cline, 1978) for the Electric Power Research Institute (EPRI) focused on transuranics in solid wastes from LWRs. The EPRI study consisted of four BWRs and three PWRs. Samples were taken at each of the seven facilities. Major systems sampled included fuel pool filter sludge,

Table 4.3-1 Responses to Survey Questions on Alpha Contamination in Low-Level Wastes

Plant	Wastes Checked for Contamination		Contamination (nCi/g)	
	No	Yes	Typical Reading nCi/g	% Exceeding 10 nCi/g
Boiling Water Reactors				
B1		X	--	--
B2		X	NDA[1]	--
B3	X		--	--
B4	X		--	--
B5	X		--	--
B6		X	10	0
B7	X		--	--
B8	X		--	--
B9	X		--	--
B10	X		--	--
B11	X		--	--
B12	X		--	--
Pressurized Water Reactors				
P1	X[2]		0	--
P2	NR[3]			
P3	X		--	--
P4		X	NDA	--
P5	X[4]		--	--
P6	X		--	--
P7	X		--	--
P8		X	--	--
P9	X		--	--
P10	X		--	--
Pressurized Water Reactors (Cont'd)				
P11	X		--	--
P12		X[5]	0	0
P13	X		--	--
P14	X		--	--
P15	X		--	--

Table 4.3-1 Responses to Survey Questions on Alpha Contamination in Low-Level Wastes (Cont'd)

Plant	Wastes Checked for Contamination		Contamination (nCi/g)	
	No	Yes	Typical Reading nCi/g	% Exceeding 10 nCi/g
P16		X	0	0
P17		X	NDA	--
P18	X		--	--

1. No detectable activity.

2. Reactor coolant samples checked for contamination monthly.

3. No response.

4. Contamination found in smears taken throughout plant; average readings are 194 to 726 dpm/smear.

5. Only primary system equipment that has been shipped offsite for shallow land burial has ever shown contamination.

reactor water cleanup sludge, condensate system sludge or resin, radwaste sludge, resin and evaporator bottoms, and special crud and smear samples. Table 4.3-2 lists the size, type of condensate polishing system, the wastes sampled, and the number of samples taken from the seven plants in the EPRI study. The same data for the facilities in the EPA study also are included in Table 4.3-2. Using information in this table, specific samples were identified that represent the waste types into which this study has categorized BWR and PWR wastes. Neither the EPA study nor the EPRI study analyzed cartridge filters. Neither study analyzed compactible and noncompactible trash, which is very low in contamination.

For BWRs with deep bed condensate polishing systems the following correlations between waste type, data source, and sample were made:

- Spent resin - EPRI study, plant 2, spent resin tank sample
- Filter sludge - EPRI study, plant 4, centrifuge solids sample
- Evaporator bottoms - EPA study, Nine Mile Point, radwaste evaporator bottoms sample.

For BWRs with precoat filter condensate polishing systems the following correlations between waste type, data source, and sample were made:

- Spent resin - included with filter sludge
- Filter sludge - EPRI study, plant 1, cleanup and condensate phase separator samples
- Evaporator bottoms - none.

The unavailability of data for evaporator bottoms is not a serious loss of information. As seen from Table 4.2-48 the filter sludge and spent resins account for 93% of the process waste volume and almost 97% of the process waste activity. Evaporator bottoms in this type of plant are basically concentrated decontamination solutions that are expected to contain very low concentrations of contamination, especially transuranic radionuclides.

For PWRs without condensate polishing systems the following correlations between waste type, data source, and sample were made:

- Spent resin - EPA study, Ginna, spent resin tank sample
- Filter sludge - negligible waste volume
- Evaporator bottoms - EPRI study, plant 7, floor drain and chemical waste evaporator bottoms sample.

Table 4.3-2 Samples Analyzed in the EPRI and EPA Studies

Plant	Size (MWe)	Condensate Polishing System	Samples Obtained
BWRs			
1	500	Powdex	1 Radwaste sludge; 4 clean-up phase separators; 4 condensate phase separators; 4 reactor coolant filters
2	650	NRD[1]	1 Condensate resin; 1 radwaste resin; 1 spent resin tank; 2 flatbed filters; 6 reactor water insolubles
3	100	NRD	1 Cleanup resin; 1 radwaste resin; 4 reactor coolant insolubles; 4 special crud and smears
4	650	NRD	5 Concentrated wastes (floor drains); 6 centrifuge solids (filter sludge); 1 cleanup resin; 1 condensate resin; 4 RWCU filters; 3 reactor coolant liquid; 1 special crud sample (PBF)
PWRs			
5	500	None	1 Evaporator bottoms (chem-waste); 3 leakoff demineralizer; 1 fuel pool filter; 2 waste holdup tank filters; 1 reactor water insoluble; 2 fuel pool resins
6	850	PP[2]	3 Spent resin tank; 1 evaporator bottoms; 1 reactor coolant insoluble; 4 special crud smears
7	200	None	2 Evaporator bottoms (floor drains and chem); 1 letdown filter; 1 letdown resin; 1 fuel pool filter; 1 reactor coolant insoluble; 1 reactor coolant ionics; 2 special crud samples

Table 4.3-2 Samples Analyzed in the EPRI and EPA Studies (Cont'd)

Plant	Size (MWe)	Condensate Polishing System	Samples Obtained
PWRs (cont'd)			
Ginna	500	None[3]	1 Unspecified evaporator bottoms; 3 smears primary coolant filter; 1 spent resin tank
Indian Pt. 2	900	None	1 Unit 1 composite evaporator bottoms; 1 primary coolant tapline filter
Nine Mile Point	620	RD	1 Radwaste evaporator bottom; 1 sludge storage tank; 1 spent resin tank
Fitz Patrick	820	PD	1 Chem waste evaporator; 1 dry centrifuge filter waste

1. Nonregenerative demineralizer.

2. Regenerative demineralizer.

3. At time of study deep bed regenerative demineralizers installed for condensate polishing as of January 1978.

For PWRs with condensate polishing systems the following correlations among waste types, data sources, and samples were made:

- Spent resin - EPRI study, plant 6, spent resin tank sample
- Filter sludge - negligible waste volume
- Evaporator bottoms - EPRI study, plant 6, evaporator bottoms sample.

Based on the EPRI and EPA studies, average concentrations of long-lived (greater than 30 years) transuranic radionuclides were calculated from the samples identified in the previous listing. These concentrations, for the four reactor types, are given in Tables 4.3-3 through 4.3-6. The total activity levels in the samples are summarized, by waste type and reactor type, in Table 4.3-7. The method of production of the transuranic radionuclides is shown in Figure 4.3-1 (Cline, 1978). The shaded areas are the nuclides usually found in LWR wastes.

4.3.2 Total Activity of Transuranic Radionuclides in LWR Low Level Wastes

The estimated total annual activity of the transuranic radionuclides was calculated by using the concentrations of transuranic radionuclides listed in Table 4.3-7, the densities (calculated from current survey data) of the various waste types prior to solidification, and the estimated untreated waste volumes in Table 4.2-48. Estimates of the total annual activity of transuranic radionuclides are in Table 4.3-8. The range in total activity for a 1,000 MWe reactor is three orders of magnitude, from 0.058 Ci/yr to 1.6 Ci/yr. When viewed in terms of average concentration per unit volume over all of the process-related waste volumes (the total annual waste volume excluding trash), the range of values is much smaller. The concentrations range from 3.6×10^{-4} μCi/cc to 6.7×10^{-3} μCi/cc.

The estimates given here for each waste type are based on data from a single plant, and in many cases only one sample from that plant. Even though the accuracy of the numbers may be open to refinement, the relative concentrations reveal which wastes have higher concentrations of transuranic radionuclides. For BWRs the majority of the activity is concentrated in the precoat filter sludge, especially in plants that use precoat filters in the condensate polishing system.

Other sources of filter sludge are the radwaste system, the fuel pool cleanup system, and the reactor water cleanup system. The fact that most of the transuranic activity is accumulated on the plant filters is borne out by other samples in the EPRI study. These samples indicate that precoat filter wastes from the reactor water cleanup system, condensate polishing system, and radwaste systems have the highest concentration of transuranic

Table 4.3-3 Concentrations of Transuranic Radionuclides in BWRs With Deep Bed Condensate Polishing Systems

Nuclide	Radionuclide Concentration (μCi/g)		
	Spent Resin	Filter Sludge	Evaporator Bottoms
U-234	--	--	3.0 (-6)
U-235	2.0 (-7) [1]	5.1 (-5)	2.0 (-6)
U-238	2.0 (-7)	5.1 (-5)	1.5 (-6)
Pu-238	5.7 (-6)	7.9 (-4)	1.3 (-5)
Pu-239	5.4 (-6)	4.5 (-4)	8.0 (-6)
Pu-240	0	0	
Am-241	7.4 (-6)	2.2 (-4)	5.0 (-6)
Am-243	0	0	
Total	1.9 (-5)	1.6 (-3)	3.3 (-5)

1. 2.0 (-7) = 2.0×10^{-7}.

Table 4.3-4 Concentrations of Transuranic Radionuclides in BWRs With Precoat Filter Condensate Polishing Systems

Nuclide	Radionuclide Concentration (μCi/g)	
	Spent Resin and Filter Sludge	Concentrator Bottoms
U-235	2.5 (-6) [1]	
U-238	4.0 (-6)	
Pu-238	4.7 (-3)	
Pu-239	3.0 (-3)	
Pu-242	0	
Am-241	7.7 (-4)	
Am-243	0	
Total	8.5 (-3)	NA [2]

1. 2.5(-6) = 2.5×10^{-6}.
2. Not available.

Table 4.3-5 Concentrations of Transuranic Radionuclides in PWRs Without Condensate Polishing Systems

Nuclide	Radionuclide Concentration (μCi/g)	
	Spent Resin	Evaporator Bottoms
U-234	2.3 (-5) (1)	--
U-235	1.2 (-5)	8.0 (-7)
U-238	4.5 (-5)	8.0 (-7)
Pu-238	4 (-5)	5.1 (-5)
Pu-239	8 (-4) (2)	1.3 (-5)
Pu-242		0
Am-241	7 (-4)	2.4 (-5)
Am-243		0
Total	2.0 (-3)	9.0 (-5)

1. 2.3(-5) = 2.3 x 10^{-5}.
2. Includes Pu-240.

Table 4.3-6 Concentrations of Transuranic Radionuclides in PWRs With Condensate Polishing Systems

Nuclide	Radionuclide Concentration (μCi/g)	
	Spent Resin	Evaporator Bottoms
U-234	--	--
U-235	2.7 (-7)[1]	2.0 (-4)
U-238	5.0 (-7)	2.0 (-4)
Pu-238	1.1 (-5)	3.0 (-4)
Pu-239	1.1 (-4)	3.0 (-4)
Pu-242	0	0
Am-241	1.2 (-5)	4.0 (-4)
Am-243	0	0
Total	1.3 (-4)	1.4 (-3)

1. $2.7(-7) = 2.7 \times 10^{-7}$

Table 4.3-7 Summary of Transuranic Radionuclides in Unsolidified LWR Low Level Wastes

Waste Type	Radionuclides (μCi/g) BWRs Deep Bed CPS[1]	BWRs Precoat CPS	PWRs Without CPS	PWRs With CPS
Deep bed resin	1.9 (-5)[2]	[3]	2.0 (-3)	1.3 (-4)
Concentrator bottoms	3.3 (-5)	ND[4]	9.0 (-5)	1.4 (-3)
Filter sludge	1.6 (-3)	8.5 (-3)	--	--
Cartridge filters	--	--	ND	ND
Trash	ND	ND	ND	ND

1. Condensate polishing system.
2. $1.9(-5) = 1.9 \times 10^{-5}$
3. Included in filter sludge.
4. No data.

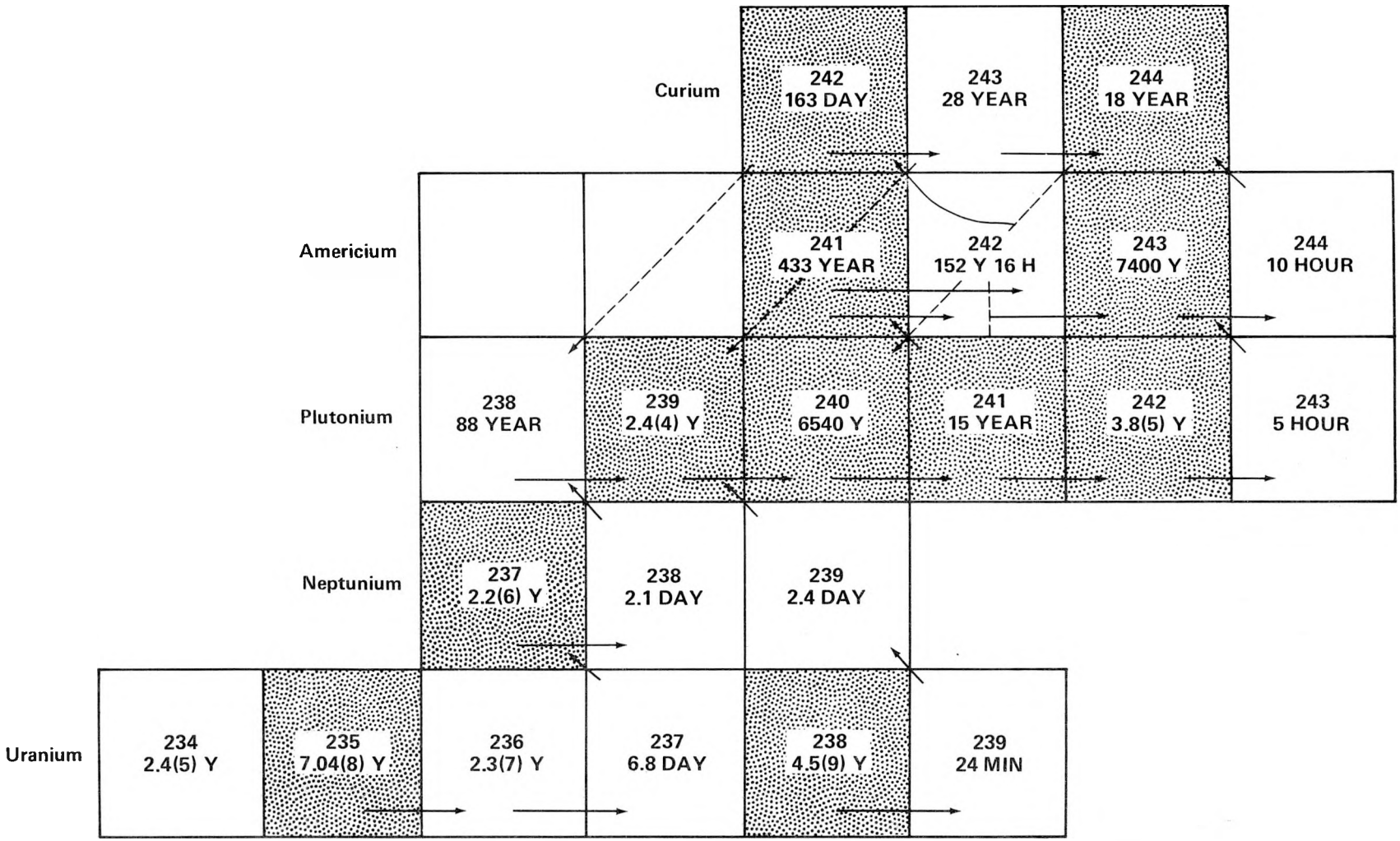

Figure 4.3-1 Chart of Nuclides Showing Production of Transuranics.

Table 4.3-8 Annual Activity of Transuranic Radionuclides Shipped From LWRs per MWe of Installed Capacity

Waste type	Activity (Ci/MWe-gr) BWRs Deep Bed CPS	BWRs Precoat CPS	PWRs Without CPS	PWRs With CPS
Deep bed resin	2.0	(1)	48	1.1
Concentrator bottoms	14.	(2)	9.9	190.
Filter sludge	210.	1600.(3)	(2)	(4)
Cartridge filters	(4)	(4)	NA(5)	NA
Trash	(6)	(6)	(6)	(6)
Total	226.	1600.	57.9	191.
Total activity for a 1,000 MWe plant	0.23 Ci/yr	1.6 Ci/yr	0.058 Ci/yr	0.19 Ci/yr
Average concentration for process waste (not including trash)	3.6×10^{-4} Ci/cc	6.7×10^{-3} Ci/cc	3.6×10^{-4} Ci/cc	1.3×10^{-3} Ci/cc

1. Included with filter sludge.
2. No data available - volume insignificant.
3. Includes deep bed resin.
4. Not applicable.
5. Not available.
6. Negligible.

radionuclides. Spent resins have the second highest concentration of transuranic radionuclides among the resins. Reactor water cleanup system resins and condensate cleanup system resins have the highest concentration, and condensate polishing system resins the lowest. Evaporator bottoms have the same range of activity concentration as resins. The lowest activities were found in the reactor coolant insolubles. For PWRs the highest concentration of transuranic radionuclides was again found in precoat filter sludge, this time from the spent fuel pool cleanup system. The remaining PWR wastes, resins, evaporator bottoms, and reactor coolant insolubles have concentrations approximately equal to their BWR counterparts. Generally it appears that the concentration levels are as follows:

Waste Type	Transuranic Concentration (μCi/g)
Precoat filter sludge	10^{-4} - 10^{-1}
Spent resins	10^{-5} - 10^{-3}
Evaporator bottoms	10^{-6} - 10^{-3}
Reactor coolant insolubles	10^{-8} - 10^{-6}

The concentrations in Table 4.3-6 can be explained. In BWRs with deep bed condensate polishing systems the resins exhibit the lowest concentration of transuranics because they are mainly from the radwaste system or from the condensate polishing system. The concentrated bottoms tend to have higher concentrations of transuranics since many of them are concentrated chemicals from the regeneration of the condensate polishing system resins. Therefore whatever transuranic activity is collected by the resins is concentrated by the evaporator. The filter sludge in these plants comes from the reactor water cleanup system, the spent fuel pool cleanup system, and the radwaste system. The first of these systems removes the transuranics directly from the reactor coolant system while the second system removes transuranics that either leak out of the fuel or sluff off the exterior of the fuel elements in the spent fuel pool. Analysis of the samples in the EPRI study shows that spent fuel pool filter wastes have the highest concentration of transuranics of all LWR wastes, and that BWR reactor water cleanup system wastes rank second in concentration of transuranics. The radwaste system consists of collected system leakage and equipment drainage.

Other systems and equipment are in turn contaminated by leakage from the primary and secondary systems. A more complete description of the relationship between systems is given in Section 3.2. Therefore the concentration of activity, when wastes from the three systems are averaged together, should still be significantly higher than the concentration of activity for resins and evaporator bottoms. The available data for BWRs with precoat filters for

condensate polishing was a composite sample of both precoat filter sludge and resin. The volume of resin is small (approximately 1.0%) compared to the quantity of filter sludge. In view of the fact that resin activities are generally lower than filter sludge activities, its overall contribution is minimal. The concentration of transuranic activity in the concentrator bottoms is also expected to be relatively low because this waste is mostly from chemical laboratory drains and decontamination solutions. The activity concentration on the filter sludge is well within the range established previously, but it is a factor of 5 higher than the activity concentration of BWRs with deep bed condensate polishing systems. This may result from more fuel in the spent fuel pool of the plant from which these data were taken. Thus there is more activity to be removed. Or, this plant could have had an unusually bad fuel failure resulting in higher pool activity levels. A second possibility is that the data used for BWRs with deep bed condensate polishing systems may have had exceptionally good fuel. But this is doubtful because the sample from that plant showed relatively high concentrations of short-lived nuclides which would indicate high fuel leakage. A third possibility is that precoat type filters are better suited (on a pound for pound basis) for the removal of transuranics than deep bed resins are. If this is true, the concentration of transuranics in precoat filter sludge would be higher than the transuranics in deep bed resin, and the total activity might be the same or even less. The fourth and last possibility is that the sample used for this estimate was not truly representative of an average activity concentration for precoat filter sludge. Instead the sample may have been solely from the fuel pool cleanup system which would yield a higher than average concentration.

In the PWRs without condensate polishing systems the concentrator bottoms are an order of magnitude lower in concentration than the resins. Many of the resins may be from the chemical and volume control systems or the boron control systems. These systems are like the BWR water cleanup systems in that they treat primary coolant. Therefore, it is not unrealistic to see concentrations that might be roughly equivalent. On the other hand, one would expect the evaporator bottoms to be lower since there is no regeneration of resins with the possible exception of deborating demineralizers. These are typically located downstream of other process equipment which would remove much of the transuranics prior to their arrival to the deborating demineralizers. Most of the waste processed through the radwaste concentrator is chemical laboratory drains and decontamination solutions.

The concentration of radionuclides in the spent resin of a PWR with a condensate polishing system has a lower concentration of transuranic radionuclides than a PWR without a condensate polishing system. This can be due to either the moderating effect of condensate resins or to better fuel performance. The estimates of higher radionuclide activity concentration in the concentrator bottoms compared to estimates for the PWRs without condensate polishing systems could be due to better concentrator efficiency.

Or, the datum for this estimate could have come from a plant that does not use its evaporator for the concentration of demineralizer regeneration solutions. The datum could have come from a plant that uses the evaporator to concentrate primary coolant for boron recovery. In this case the boron, and therefore the transuranics, are mostly recycled to the plant.

Obviously, much more information is needed on this subject before any realistic final projections are made on the total quantity of transuranics which are shipped offsite from LWRs on an annual basis.

4.3.3 Ranking of Contaminated Transuranic LWR Wastes as Candidates for Storage in Federal Repositories

Ranking LWR waste for storage in a federal repository is based on the estimated concentrations of long-lived (half-life greater than 30 years) transuranic radionuclides found in the samples used in the EPA and EPRI studies. The ranking in this report is based solely on these concentrations. Few plants are designed to segregate specific wastes from specific systems without costly modifications. Based on their application, individual cartridge filters are expected to fall throughout the entire range. Assigning specific concentrations or a range of concentrations to each waste type is not possible with current limited data. The following list is in the order of the highest estimated concentration to the lowest estimated concentration:

1. Spent fuel pool and primary system sludge
2. Condensate and radwaste sludge
3. Spent fuel pool, radwaste, and BWR reactor water cleanup system resin
4. Evaporator bottoms
5. Condensate polishing system resins
6. Reactor coolant insolubles.

4.4 Fuel-Fabrication Facilities

4.4.1 Introduction

There are seven facilities in the United States that produce fuel for light-water-cooled nuclear power plants. Three of the facilities convert UF_6 to UO_2 pellets and manufacture fuel assemblies.

Two of the facilities convert the UF_6 to UO_2, which is shipped to two plants that manufacture the final product. The plants are

1. Babcock & Wilcox

   a. Apollo, PA: Converts UF_6 to UO_2 (to Lynchburg, VA).

   b. Lynchburg, VA: UO_2 powder (from Apollo) manufactured into fuel assemblies.

2. Combustion Engineering

   a. Hematite, MO: Converts UF_6 to UO_2 (to Windsor CT).

   b. Windsor, CT: UO_2 powder (from Hematite) manufactured into fuel assemblies.

3. Exxon

   a. Richland, WA: Converts UF_6 to UO_2 pellets and manufactures fuel assemblies.

4. General Electric

   a. Wilmington, NC: Converts UF_6 to UO_2 pellets and manufactures fuel assemblies.

5. Westinghouse

   a. Columbia, SC: Converts UF_6 to UO_2 pellets and manufactures fuel assemblies.

4.4.2 Process Description

Babcock & Wilcox (B&W), Exxon, and General Electric (GE) use an ammonium-diuranate (ADU) process for the conversion of UF_6 to UO_2. Combustion Engineering uses a dry direct conversion (DDC) process and the Westinghouse facility uses both. While each facility is different from the other, the basic ADU or DDC process is the same. In the ADU process the UF_6 is dissolved in deionized water to which ammonium hydroxide is added. The uranium then precipitates as ADU. The ADU slurry is dewatered using either filtration or centrifugation, dried and calcined to decompose the ADU to U_3O_8, and then reduced to UO_2 powder. The liquids from the precipitation operation contain ammonium fluoride or ammonium nitrate, ammonium hydroxide and approximately 10 to 15 ppm residual uranium.

In the direct dry process, UF_6 is processed through a series of retorts which result in UO_4. This is then reacted with H_2 to form UO_2 powder and water vapor. The details of these systems are proprietary. The UF_6 to UO_2 conversion processes do not directly result in solid wastes. Most liquid wastes are pumped to a settling pond, or lagoon, or are recycled for reuse.

The process employed to convert the UO_2 powder into pellets is basically the same at all of the facilities.

The following is a description of one of the plants surveyed. The procedures at other plants varied only slightly:

Pellets are produced on a mechanical rotary press. They are loaded vertically in one layer on a steel tray about 10 in. by 10 in.

When each tray is loaded, it is moved into a ventilated hood enclosure and the pellets are vacuum cleaned to remove loose powder. Vacuum dust is considered dirty scrap and is sent to recovery. A molybdenum boat approximately 10 in. by 10 in. by 1/4 in. deep is placed over the pellets, top side down, and the two trays are inverted. The steel tray is removed and the pellets revacuumed to remove remaining powder. Four loaded moly-boats are stacked and placed in the automatic feed position of the sintering furnace. When a stack exits the furnace, the procedure is reversed to place the pellets in steel trays for transfer to the safe geometry storage racks and/or to the grinding operation. The moly-boats are recycled to the head-end of the furnace. The pellet transfer is carried out in a hood. Trays of sintered pellets are loaded onto the rotary feed table of the centerless grinder. A diamond wheel is used on the grinder together with a water lubricant. The water, containing finely divided UO_2, is passed through a centrifuge to remove most of the UO_2 and recirculated. About once a month, the water is collected in a plastic container, recentrifuged, and allowed to stand for several days to permit the ultra-fine UO_2 to settle out. The liquids are separated from the UO_2 by decantation. The dewatered UO_2 is dried, oxidized, and blended in limited quantities in subsequent blend lots. The liquids are solidified and shipped to a commercial burial facility. The ground pellets are loaded onto corrugated stainless steel trays, dried, and transferred to the rod loading area. Clean scrap (reject pellets) are oxidized to U_3O_8, reduced to UO_2, and blended into the virgin UO_2 in limited quantities. Dirty scrap is sent to scrap recovery.

The unit operations in the pellet manufacture area are

- Powder transfer
- Blending and precompaction
- Granulation
- Pressing
- Green pellet vacuum cleaning and tray transfer
- Oxidizing
- Sintering furnace (entrance and exit)
- Sintered pellet tray transfer
- Sampling hoods
- Centerless grinding.

Each of the unit operations has a ventilation exhaust that is double prefiltered with fiberglass filters as close to the operation as is feasible. The prefilters are 40% and 70% rating

respectively. The sintering furnace is on a separate 8,000 cfm exhaust system. All process steps exhaust to a 16,000 cfm system. The prefilters are backed up with dual fire retardant High Efficiency Particulate (HEPA) filters before the monitored ventilation system is exhausted to the environment. The 8,000 cfm system (from furnace) can be recirculated to the other plant areas to provide heat if necessary in winter months. The 16,000 cfm system is recirculated and treated as controlled air. Both systems are continuously monitored. The manufacturer has found no evidence of contamination on the HEPA filters or in the ductwork. The manufacturer has found contamination on some of the prefilters on gamma-scanning. Contaminated prefilters are disassembled and the filter media are sent to scrap recovery where they are incinerated, and the glass is processed by wet chemistry. Uncontaminated filters are discarded as nonradioactive waste. HEPA filters have always been uncontaminated.

Trays of ground pellets are then transferred into the rod loading room and placed on the conveyor. Each tray contains one rod load or stack of pellets. Trays are about 9 in. with 16 rows of pellets. An automatic loader moves one row at a time into a loading trough and after all 16 rows are in, the trough load is automatically inserted into a rod with one end plug already welded in place and with the vent hole already cut by laser. The loaded rod is moved out of the loading room and into a welder where the spring and retainer are inserted and the second end plug is welded in place.

The rod loading area has ventilation hoods over the automatic indexing mechanism and the pellet insertion fixture. These hoods are equipped with dual prefilters of 40% and 70% efficiency respectively before the ventilation air enters the 16,000 cfm process equipment ventilation system.

4.4.3 Waste Characterization

As with the UF_6 to UO_2 conversion process, the pelleting and fuel assembly fabrication process does not produce any solid waste directly other than recoverable UO_2. The primary sources of solid wastes are noncombustible and combustible trash, filters from ventilation systems, and filter cakes from some liquid waste treatment systems.

4.4.3.1 Combustible and Noncombustible Trash

The items that commonly make up combustible and noncombustible trash are listed in Table 4.4-1. These items are generally found at all plants. Data indicates that 85% of the total trash volume is combustible. Several of the facilities incinerate their combustible trash in order to concentrate any uranium contamination. Once concentrated, the uranium recovery is economically feasible. Still, much of the waste contains uranium that is not economically recoverable. These wastes are packaged and shipped offsite for shallow land burial.

Table 4.4-1 Items Generally Found in Combustible and Noncombustible Trash

Combustible Trash	Noncombustible Trash
Shoe covers	Worn-out equipment
Paper wipes	Piping
Plastic gloves	Fire brick
Coveralls	Wire
Smocks	Metal
Waste paper	Ceramic scrap
Filter components	Glassware
Plastic bags	Discarded jigs and fixtures
Wood	Damaged metal pails
Oil	Insulation
	Tools

4.4.3.2 Filter Sludges

Filter sludges are the result of the filtration of various liquid waste streams such as floor drains, wash basin drains, and floor scrubber solutions. If the uranium concentration is too low for economical recovery, the sludge is packaged for burial. If sufficient uranium is present, the sludge is processed through scrap recovery prior to disposal.

4.4.3.3 Prefilters and HEPA Filters

The same basic procedures are followed for prefilters and HEPA filters as for liquid filter sludges. Some plants with incinerators disassemble filters and burn the combustible components. Table 4.4-2 lists the procedures followed at most facilities for the disposition of most prefilters and HEPA filters.

4.4.3.4 Oil

As a result of the rotary press operation oil is present in those facilities that manufacture UO_2 pellets. Currently, this oil is being shipped offsite for burial or being stored onsite while an economical method of UO_2 recovery is found. Burning the oil in the incinerators is a possibility but would require modifications to the equipment. Table 4.4-3 shows the disposition of oil at each facility.

4.4.4 Uranium Production

The annual fuel requirement of the typical 1000 MWe plant is between 30 and 33 metric tons of uranium per year, or 30 to 33 MTU/yr/GWe. An average value of 31 MTU/GWe/yr is used in this

study and is based on a 2:1 mix of PWRs to BWRs, with PWRs being slightly more efficient. This average value is used to estimate the required uranium fuel production that will be needed in the year 2000 to support 380 GWe of generating capacity. The estimate is 11,780 MTU/yr.

The annual throughput for four of the five facilities surveyed is given in Table 4.4-4. The estimated production of 3,185 MTU, in 1980, is sufficient to support approximately 100 GWe of total generating capacity. According to Table 5.2-1 the total nuclear generating capacity in the United States in 1980 will only be 66 GWe, with the 100 GWe mark being reached in 1984. Two reasons explain the differences in the numbers for 1980. First, the uranium production and fuel fabrication must be maintained ahead of the demand for the fuel in the plant. This time delay is needed to allow for delivery to the plant, to accumulate the required quantity of fuel for either initial loading or refueling, and for new plant low power testing and startup. Second, many of the United States fabricators sell fuel overseas.

Table 4.4-5 and Figure 4.4-1 give the forecasted demand for uranium based on the generating capacities in Table 5.2-1. Initial core loadings are assumed to be 120 MTU for each 1,000 MWe of BWR capacity and 100 MTU for each 1,000 MWe of PWR capacity. Second-year fuel requirements are zero with an average fuel requirement of 31 MTU per year for each 1,000 MWe of installed capacity thereafter. Also given is the estimated industry production to meet the necessary lead times discussed and foreign sales. Industry production is assumed to be 1.5 times domestic demand based on the 1980 projections previously quoted.

Table 4.4-2 Disposition of Prefilters and HEPA Filters

Facility	Disposition
F1	Prefilters are incinerated; others are boxed for burial.
F2	Contamination is too low for economical recovery. Filters are packaged and sent offsite for burial.
F3	Combustible portions of prefilters and HEPA filters are incinerated.
F4	Filters from the conversion process are burned. Filters from the assembly process are boxed for burial.
F5	If the uranium contamination is less than 150 grams, the filter is repackaged in the original box which has been lined with plastic and sent offsite for burial.

Table 4.4-3 Disposition of Oil

Facility	Current Method of Disposition	Future Method of Disposition
F1	Collected and stored	Looking for recovery method; may solidify in concrete, 10% liquid and 90% concrete in 30-gal drum.
F2	Collected and held	Looking at filtration or centrifugation to recover UO_2, expects incineration would be best recovery method but would require modification to incinerator.
F3	Packaged in 55-gal drums and sent to shallow land burial. (6-8 drums/yr).	
F4	Packaged and stored for future disposal.	
F5	Absorbed on vermiculite and sent to shallow land burial.	

Table 4.4-4 Uranium Production Throughput, 1974-1980

Facility	Uranium Production Throughput (MTU/yr) 1974	1975	1976	1977	1978	1979	1980
F1		275	220	230	100	(275) [1]	(275)
F2		500	650	750	750	(750)	(750)
F3		(1,000)	(1,000)	(1,000)	(1,000)	(1,000)	(1,000)
F4		130	130	130	130	(130)	(130)
F5				515	665	925	1,030
Annual capacity		1,905	2,000	2,625	2,645	3,080	3,185

1. Numbers in parentheses are estimates.

2. Facility F3 declined to give uranium production throughput. These estimates have been made by NUS Corp.

4.4.5 Solid Waste Generation

4.4.5.1 Volume

As seen from the discussion on waste characteristics, very little, if any, of the waste from fuel fabrication facilities is solidified prior to shipment for burial. In contrast to LWRs the waste from fuel fabrication facilities does not include any concentrated liquid wastes other than those that are pumped into evaporation ponds onsite. Data on solid wastes shipped offsite for burial consist of 15 plant-years of data, 3 of which are estimates for 1978. The range of data are extremely large, ranging from slightly less than 3,000 ft^3/yr to over 100,000 ft^3/yr. Table 4.4-6 provides information on annual waste volumes from the five full process facilities. When viewed in this manner it seems obvious that the facilities with the higher annual waste volumes are those without the incinerators. Unfortunately that is not necessarily true, especially when it is noted that F3 uses an incinerator for almost all contaminated combustibles.

Another problem is that in the two-part facilities both plants do not always have an incinerator. Furthermore, the data on annual shipment are not broken down into sufficient detail to determine what percentage of the waste was ash, filter sludge, filters, or noncombustible trash. Therefore it became necessary to proceed with the analysis based on the assumption that the practices being

Table 4.4-5 Projected United States Fuel Production

Year	Domestic Demand	Industry Production
	MTU/yr	MTU/yr
1980	2,700	4,000
1985	4,700	7,000
1990	6,800	10,000
1995	9,700	14,600
2000	12,500	18,800

followed today will continue through the year 2000. Also since some of the facilities do incinerate it is probable that any further reduction in the total waste shipped offsite from a number of facilities would be offset by the volume increase caused by solidification should it become necessary within the next 20 years.

Based on the data presented in Table 4.4-6, and giving each year of data equal weight, it is estimated that a typical facility will ship approximately 40,000 ft^3/yr of waste offsite for burial.

The data can also be viewed in terms of cubic feet of waste per MTU processed. These are shown in Table 4.4-7. This results in an average estimated annual production rate of 80 ft^3/MTU.

4.4.5.2 Activity

The only nuclides present in wastes from fuel fabrication facilities are U-235 and U-238. If the concentration of activity is high enough to warrant economical recovery, then the waste is processed to recover as much of the uranium as possible. Trash is incinerated to increase the concentration of the uranium in the ash to a level that makes recovery economical. The data collected in the survey represent the nonrecoverable uranium shipped offsite with the waste for burial. The available data in kilograms per year and curies per year are given in Table 4.4-8. In terms of the plant size this represents approximately 2.8 kilograms of uranium per MTU throughput containing 940 μCi/MTU. The concentration of uranium in the waste, at 80 ft^3/MTU, is about 12 $\mu Ci/ft^3$.

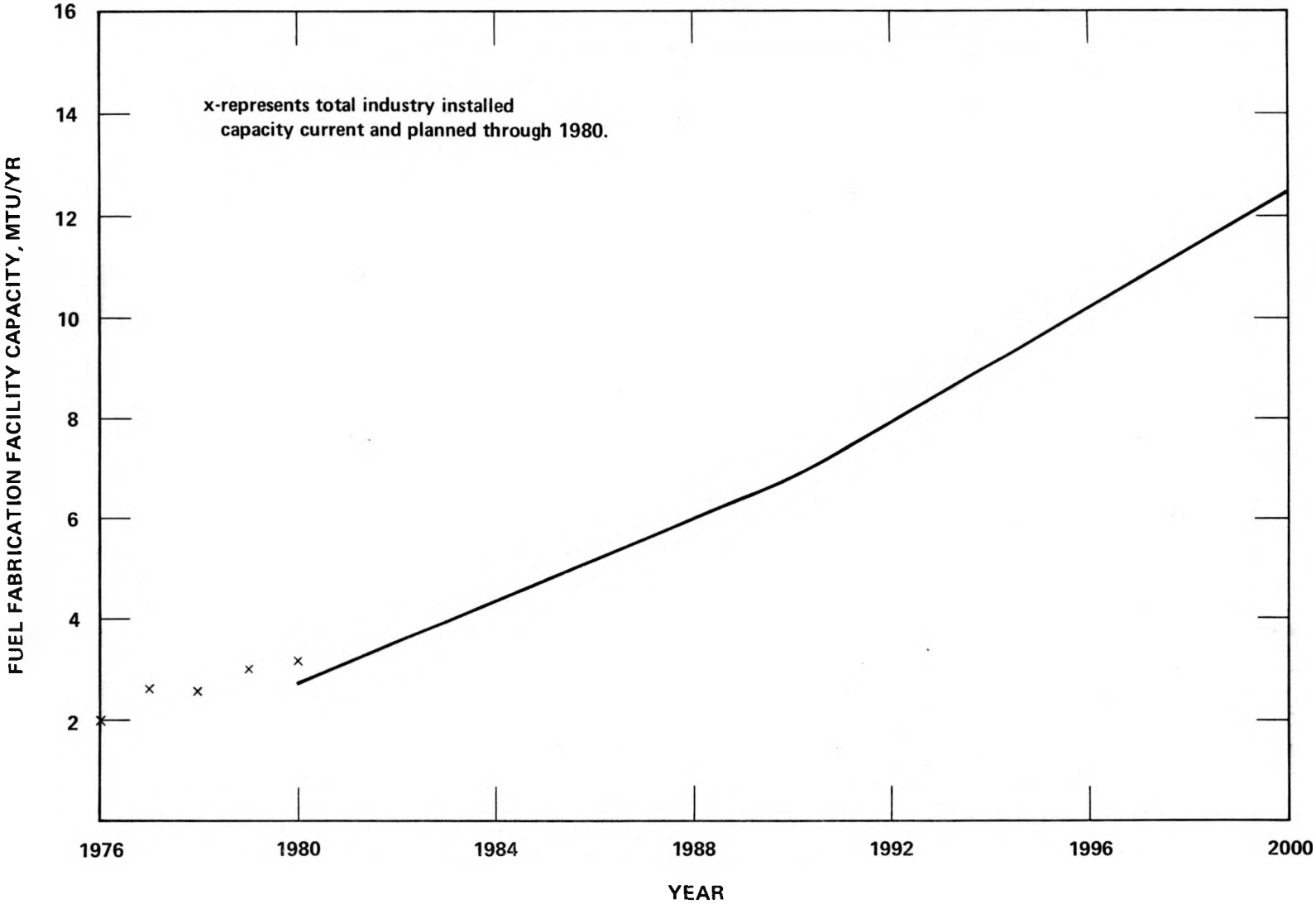

Figure 4.4-1 Total Required Throughput to Support U.S. LWRs (10^3MTU).

Table 4.4-6 Volume of Solid Waste Shipped Offsite for Burial, 1974-1978

Facility	Solid Waste Shipped Offsite for Burial (ft^3/yr)				
	1974	1975	1976	1977	1978
F1			33,665	27,191	27,400[1]
F2		48,600	43,800	46,200	
F3	104,000	55,000	101,000	62,000	
F4			2,674	4,214	7,500[1]
F5				15,170	19,700[1]

1. Estimated.

Table 4.4-7 Volume of Waste Shipped Offsite for Burial, 1974-1978

	Waste Shipped Offsite for Burial (ft^3/MTU/yr)[1]				
Facility	1974	1975	1976	1977	1978
F1			153	118	274[2]
F2		97	67	62	
F3	104	55	101	62	
F4			21	32	58[2]
F5				29	30[2]

1. Average = 80 ft^3/MTU.
2. Estimated.

Table 4.4-8 U-238 Content of Waste Shipped to Burial

(3.33×10^{-4} Ci/kgU-238 or 3,000 kgU-238/Ci)
(2.14×15^{3} Ci/kgU-235 or 467 kgU-235/Ci)

Facility	1973	1974	1975	1976	1977	1978
			U-238 Content of Waste (kg/yr)			
F1(1)						
F2			4,600	3,400	2,300	1,200
F3	1,369	1,456	1,231	1,841	493	
F4(1)						
F5(2)						
			U-238 Content of Waste (Ci/yr)			
F1(1)						
F2			1.53	1.13	.766	.400
F3	.456	.485	.410	.613	.164	
F4(1)						
F5(2)						

1. Insufficient data.
2. Not given.

CHAPTER 4 REFERENCES

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5. PROJECTIONS THROUGH 2000

5.1 Introduction

This section of the report provides an estimate of the amount of electrical power to be derived from light-water-cooled nuclear power plants through 2000. The projection will be used to estimate the quantity of solid wastes that will be generated assuming three different cases: first, that no solidification of any wastes occurs; second, that solidification of some waste will continue but there will not be any substantive change from current practices; and third, that all waste, with the exception of trash, will be solidified. This analysis is followed by an investigation of what effect each case has on the radionuclide concentration in the waste. The remainder of the section is concerned with the effects, both on volume and activity concentration, that various volume-reduction techniques will have.

5.2 Electrical Power Generated by Nuclear Power Plants Through 2000

For the purpose of this study, it was determined that only one energy generation projection would be used. The projection selected as most representative of the future growth of nuclear power in the United States is the 1977 projection by Blomeke (1977). The projections presented by Blomeke are reproduced in Table 5.2-1 for the period 1980 to 2000.

In order to translate these generation projections into waste volumes and activities, it is necessary to determine what portion of the BWRs employ deep bed condensate polishing systems and what portion uses precoat filters. Similarly, it is necessary to know what portion of PWRs have condensate polishing and what portion does not.

Table 5.2-2 shows the installed capacity of BWRs and PWRs by type of condensate polishing system through 2000. Values given through 1977 are based on actual plant installed capacities. Actual plant data were used for those plants expected to go on line from 1978 through 1980. The installed capacities for the 5-year intervals from 1981 through 2000 were calculated from the data in Table 5.2-1 with 75% of the BWR generating capacity to be from deep bed plants and 25% to be from plants with precoat condensate polishing systems. PWRs are equally split between plants with condensate polishing systems and without condensate polishing systems, as determined from Tables C-3 and C-4, in Appendix C. Figure 5.2-1 shows the installed generating capacity projected from 1977 to 2000.

Table 5.2-1 Forecast of U.S. Nuclear Electrical Power Generating Capacity (GWe)

Calendar Year	Installed Capacity		
	BWR	PWR	Total
1980	22.0	43.8	65.8
1981	23.3	47.5	70.8
1982	25.5	56.8	82.3
1983	28.1	65.4	93.5
1984	32.7	76.5	109.2
1985	37.7	88.9	126.6
1986	42.3	98.3	140.6
1987	46.5	106.7	153.2
1988	50.7	115.1	165.8
1989	55.2	124.1	179.3
1990	60.2	134.2	194.4
1991	65.4	144.8	210.2
1992	70.9	156.1	227.0
1993	76.4	167.4	243.8
1994	82.6	180.1	262.7
1995	89.1	193.4	282.5
1996	95.6	206.7	302.3
1997	102.1	220.0	322.1
1998	108.7	233.2	341.9
1999	115.2	246.5	361.7
2000	121.1	258.6	379.7

Table 5.2-2 Projected U.S. Nuclear Electrical Power Generating Capacity by Reactor Type (GWe)

	Boiling Water Reactors				Pressurized Water Reactors					
	Deep Bed	Precoat	Total	Cumulative	With CPS(1)	Without CPS	Total	Cumulative	LWR Total	Cumulative
Current through 1977	5.4	8.7	14.1	14.1	9.3	18.7	28.0	28.0	42.1	42.1
Capacity added in:										
1978-1980	5.5	2.4	7.9	22.0	7.9	7.9	15.8	43.8	23.7	65.8
1981-1985	11.8	3.9	15.7	37.7	22.5	22.6	45.1	88.9	60.8	126.6
1986-1990	16.9	5.6	22.5	60.2	22.7	22.6	45.3	134.2	67.8	194.4
1991-1995	21.7	7.2	28.9	89.1	29.6	29.6	59.2	193.4	88.1	282.5
1996-2000	24.0	8.0	32.0	121.1	32.6	32.6	65.2	258.6	97.2	379.7
Cumulative through 2000	85.3	35.8	121.1		124.6	134.0	258.6			379.7

1. Condensate polishing system.

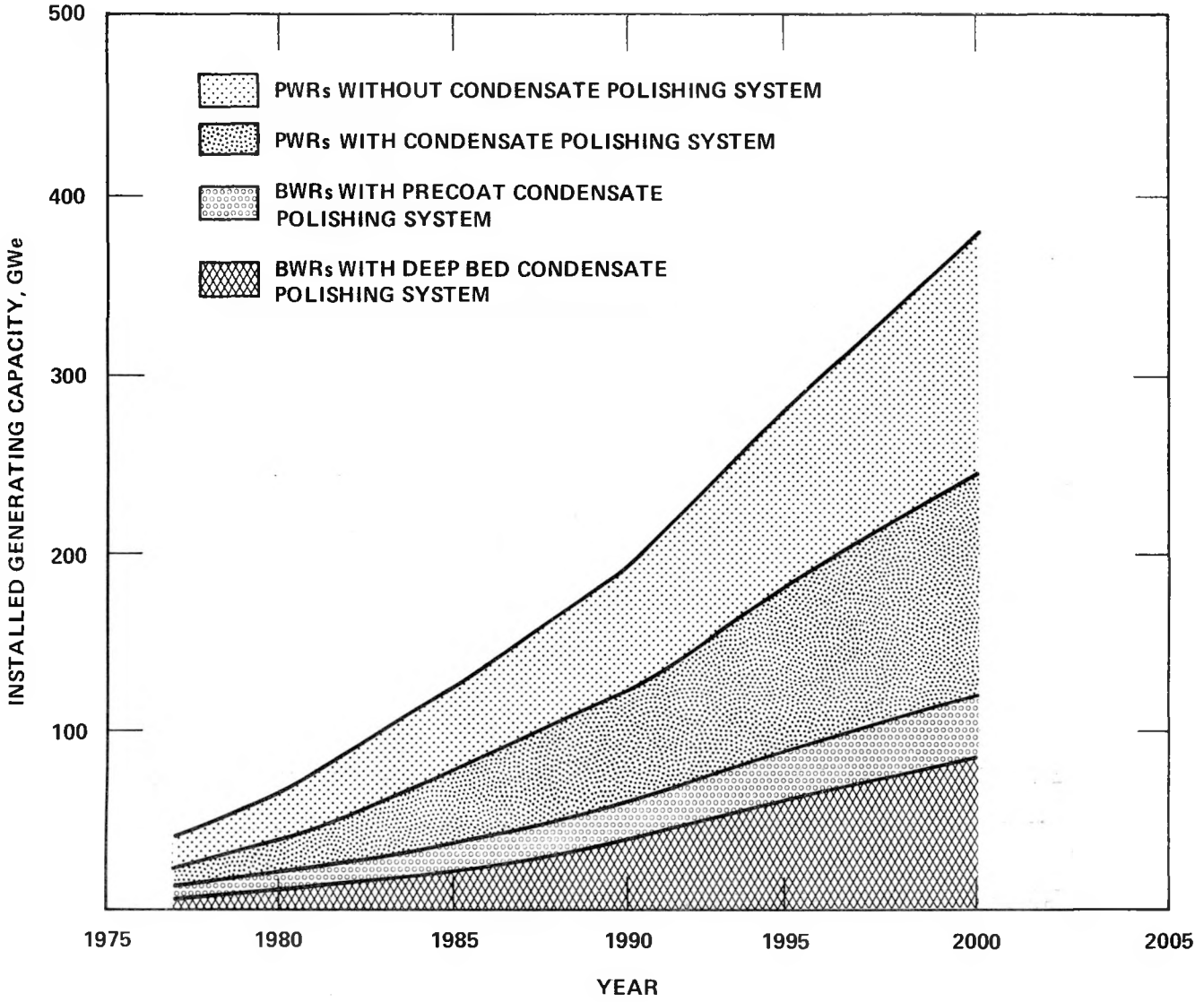

Figure 5.2-1 Total Installed Generating Capacity (GWe).

5.3 Solid Waste Volumes Through 2000

5.3.1 Unsolidified Wastes

Table 5.3-1 lists the annual volume of solid waste that is expected to be generated between 1977 and 2000. The estimated volume of unsolidified waste in 1977 is 830,000 ft^3 from all reactor types and is projected to increase to 8,020,000 ft^3 by 2000.

For comparative purposes, the volumes of waste shipped for burial from 1972 through 1976 are listed in Table 5.3-2. These volume amounts, originally taken from the facilities' semiannual effluent release reports, include the volume increase effect of solidification.

The data in Tables 5.3-1 and 5.3-2 are plotted in Figure 5.3-1. The apparent sharp increase in 1975 in both BWR and PWR wastes were due primarily to two plants. Of the 645,000 ft^3 shipped from PWRs, 325,000 ft^3 was from a two-unit plant not included in this survey. The BWR plant in 1975 and 1976 shipped contaminated soil totaling 113,700 ft^3 and 173,000 ft^3, respectively. If these 3 years of data are excluded from the graph, then the data would be plotted as shown by the o for BWRs and by the x for the total, with the difference between the two being the PWR contribution.

5.3.2 Solidified Wastes

In order to determine the quantities of solid waste that could be expected, it is necessary to consider each type of waste separately and to calculate the volume increase based on such factors as the number of plants that solidify this type of waste and what solidification agents are used. With proper consideration to these factors, the values in Table 4.2-48 are modified as shown in Table 5.3-3. As practiced over the last few years, solidification of process wastes in BWRs will result in a 36% increase in waste volumes from deep bed plants and a 22% increase from precoat plants. For PWRs, the increase is 27% for plants with a condensate polishing system and 30% for plants without a condensate polishing system. The overall effect in the projected annual waste volumes shipped for burial is shown in Table 5.3-4 and plotted in Figure 5.3-2.

5.3.3 Waste Projections With All Wastes Solidified

NRC Standard Review Plan 11.4, Solid Waste Management Systems, which contains Branch Technical Position ESTB 11.3, Rev. 1, Design Guidance for Solid Radioactive Waste Management Systems Installed

Table 5.3-1 Annual Volume of Unsolidified Wastes

Calendar Year	Volume of Unsolidified Waste ($10^3 ft^3$)						
	Boiling Water Reactors			Pressurized Water Reactors			
	Deep Bed CPS[1]	Precoat CPS	BWR Total	With CPS	Without CPS	PWR Total	LWR Total
1977	184	174	358	160	312	472	830
1980	378	222	595	296	444	740	1,340
1985	776	300	1,080	722	822	1,540	2,620
1990	1,354	412	1,770	1,070	1,200	2,270	4,040
1995	2,100	556	2,660	1,580	1,690	3,270	5,930
2000	2,920	716	3,640	2,140	2,240	4,380	8,020

1. Condensate polishing system.

Table 5.3-2 Reported Volumes of Solidified Waste (10^3 ft^3)

Calendar Year	Boiling Water Reactors	Pressurized Water Reactors	Total
1972	151	53	204
1973	208	109	317
1974	286	234	570
1975	554 (440) [1]	645 (320)	1,200 (760)
1976	645 (472)	412	1,060 (880)

1. Numbers in parentheses are excluding the three plant-years of data discussed in Section 5.3.1.

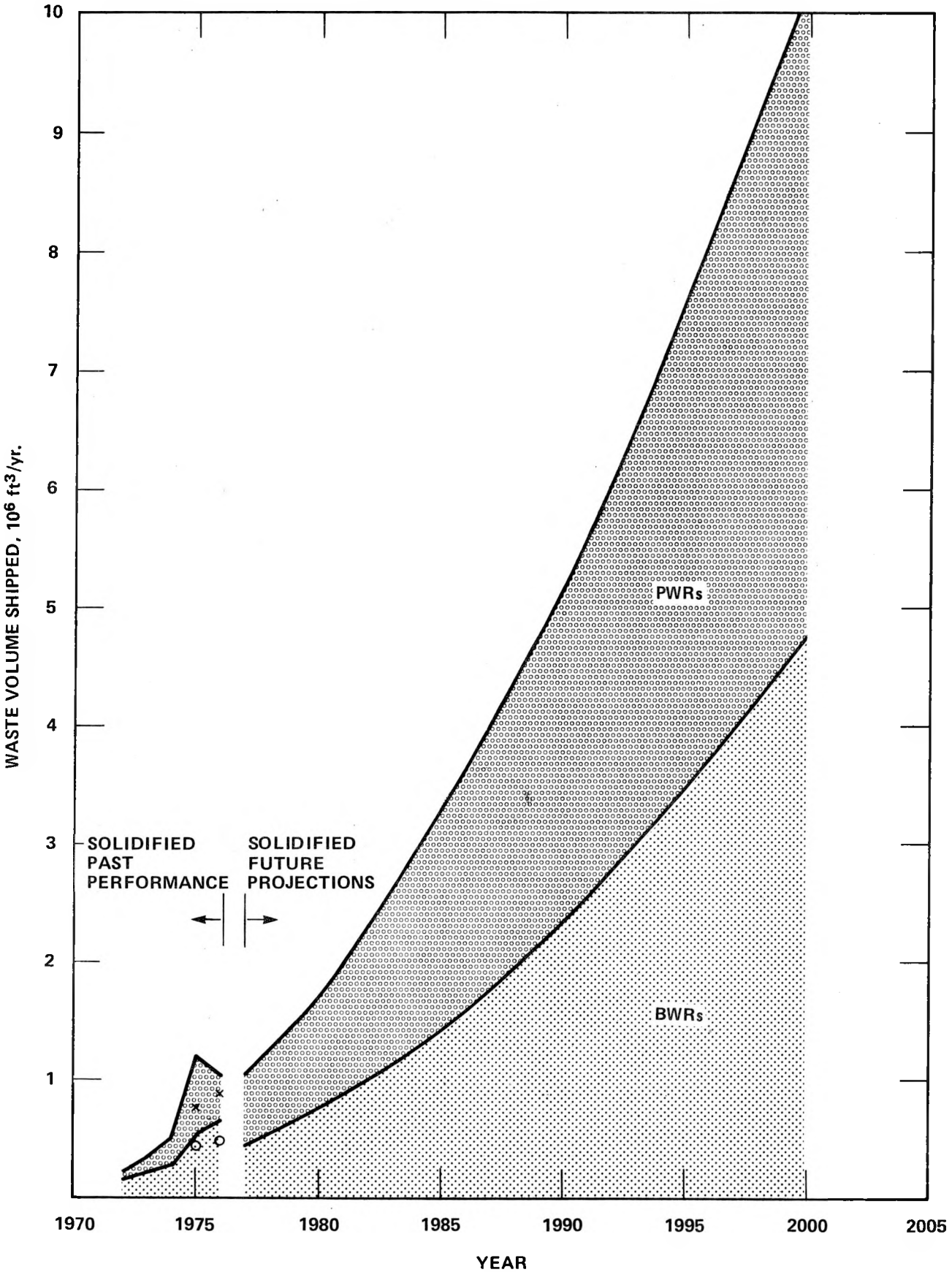

Figure 5.3-1 Annual Waste Shipped With Current Practices .

Table 5.3-3 Solidified Waste Volumes With Current Practices Ft^3/MWe Installed

Waste Type	Boiling Water Reactors		Pressurized Water Reactors	
	Deep Bed	Precoat	With CPS[1]	Without CPS
Deep bed resin	5.4	0.31	.32	1.2
Concentrated liquids	22.0	1.1	8.8	8.0
Filter sludge	7.0	10.8	0.25	-
Cartridge filters	-	-	0.39	0.39
Trash (all)	11.5	11.5	11.5	11.5
Total	45.9	23.7	21.3	21.1
1,000-MWe plant	45,900 ft^3/yr	23,700 ft^3/yr	21,300 ft^3/yr	21,100

1. Condensate polishing system.

Table 5.3-4 Estimated Annual Solidified Waste Volumes With Current Practices

Calendar Year	Solidified Waste Volume ($10^3 ft^3$)						
	Boiling Water Reactor			Pressurized Water Reactors, LWR			
	Deep Bed CPS[1]	Precoat CPS	BWR Total	With CPS	Without CPS	PWR Total	LWR Total
1977	248	206	454	198	394	592	1,050
1980	500	263	763	370	560	930	1,690
1985	1,040	360	1,400	850	1,040	1,890	3,290
1990	1,820	490	2,310	1,330	1,510	2,840	5,150
1995	2,820	660	3,480	1,960	2,140	4,100	7,580
2000	3,920	850	4,770	2,650	2,830	5,480	10,300

1. Condensate polishing system.

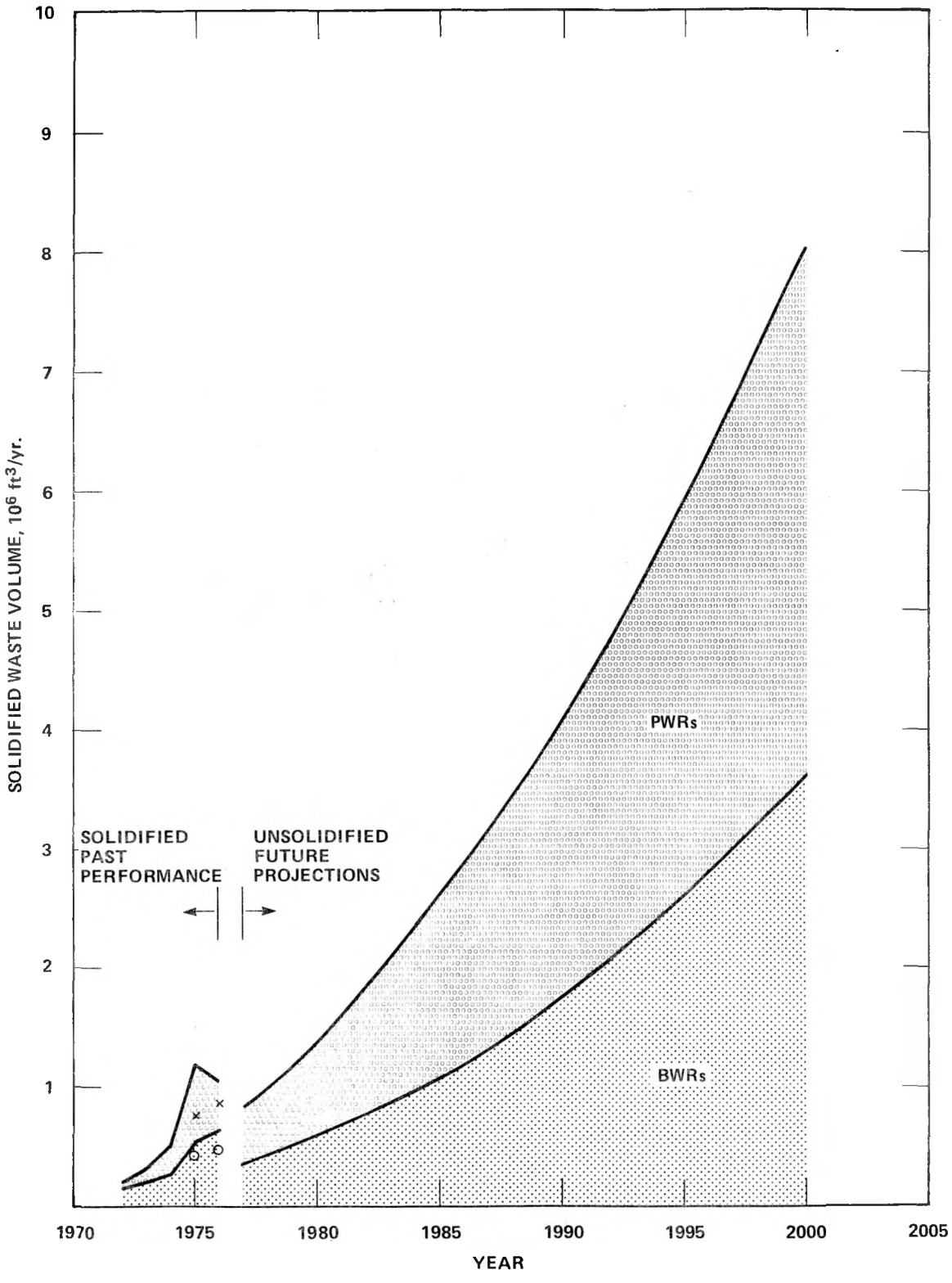

Figure 5.3-2 Annual Waste Generation , No Solidification.

in Light-Water-Cooled Nuclear Power Reactor Plants, contains the acceptance criteria which are quoted as follows:

A. Processing Requirements

1. Dry Wastes

a. Compaction devices for compressible dry wastes (rags, paper, and clothing) should include a ventilated shroud around the waste container to control the release of airborne dusts generated during the compaction process.

b. Activated charcoal, HEPA filters, and other dry wastes which do not normally require solidification processing should be treated as radioactively contaminated solids and packaged for disposal in accordance with applicable Federal regulations.

2. Wet Wastes

a. Wet wastes such as spent bed and powdered resins, filter sludge, and evaporator and reverse osmosis concentrates should be rendered immobile by combining with a suitable binding agency (cement, urea formaldehyde, asphalt, etc.) to form a homogeneous solid matrix (absent of free water) prior to off-site shipment. Absorbents such as vermiculite are not acceptable substitutes for binding agents.

b. Spent cartridge filter elements may be packaged in a shielded container with a suitable absorber such as vermiculite, although it would be desirable to solidify the elements in a suitable binder.

Should this position be implemented as a Regulatory Guide, all process wastes would require solidification. Assuming that these criteria will eventually be equally applied to both existing plants and plants under construction as well as new plants yet to go through the licensing process, annual projections can be made as shown in Table 5.3-5. These projections are based on waste generation rates, by type of waste, as given in Table 5.3-6. The projections in Table 5.3-5 are plotted in Figure 5.3-3.

Table 5.3-5 Estimated Annual Waste Volumes, All Wastes Solidified

Calendar Year	Waste Volumes ($10^3 ft^3$)						
	Boiling Water Reactor			Pressurized Water Reactors, LWR			
	Deep Bed CPS[1]	Precoat CPS	BWR Total	With CPS	Without CPS	PWR Total	LWR Total
1977	270	230	500	200	400	600	1,100
1980	550	290	840	370	570	940	1,780
1985	1,150	390	1,540	850	1,060	1,910	3,450
1990	1,990	540	2,530	1,340	1,540	2,880	5,410
1995	3,080	730	3,810	1,980	2,180	4,160	7,970
2000	4,290	930	5,220	2,680	2,880	5,560	10,800

1. Condensate polishing system.

Table 5.3-6 Solidified Waste Volumes With All Waste Solidified Ft^3/MWe Installed

Waste Type	Boiling Water Reactors		Pressurized Water Reactors	
	Deep Bed CPS(1)	Precoat CPS	With CPS(1)	Without CPS
Deep bed resin	7.4	0.37	0.54	1.6
Concentrated liquids	22.0	1.1	8.8	8.0
Filter sludge	9.3	13.2	0.25	-
Cartridge filters	-	-	0.39	0.39
Trash (all)	11.5	11.5	11.5	11.5
Total	50.2	26.2	21.5	21.5
1,000 MWe plant	50,200 ft^3/yr	26,200 ft^3/yr	21,500 ft^3/yr	21,500 ft^3/yr

1. Condensate polishing system.

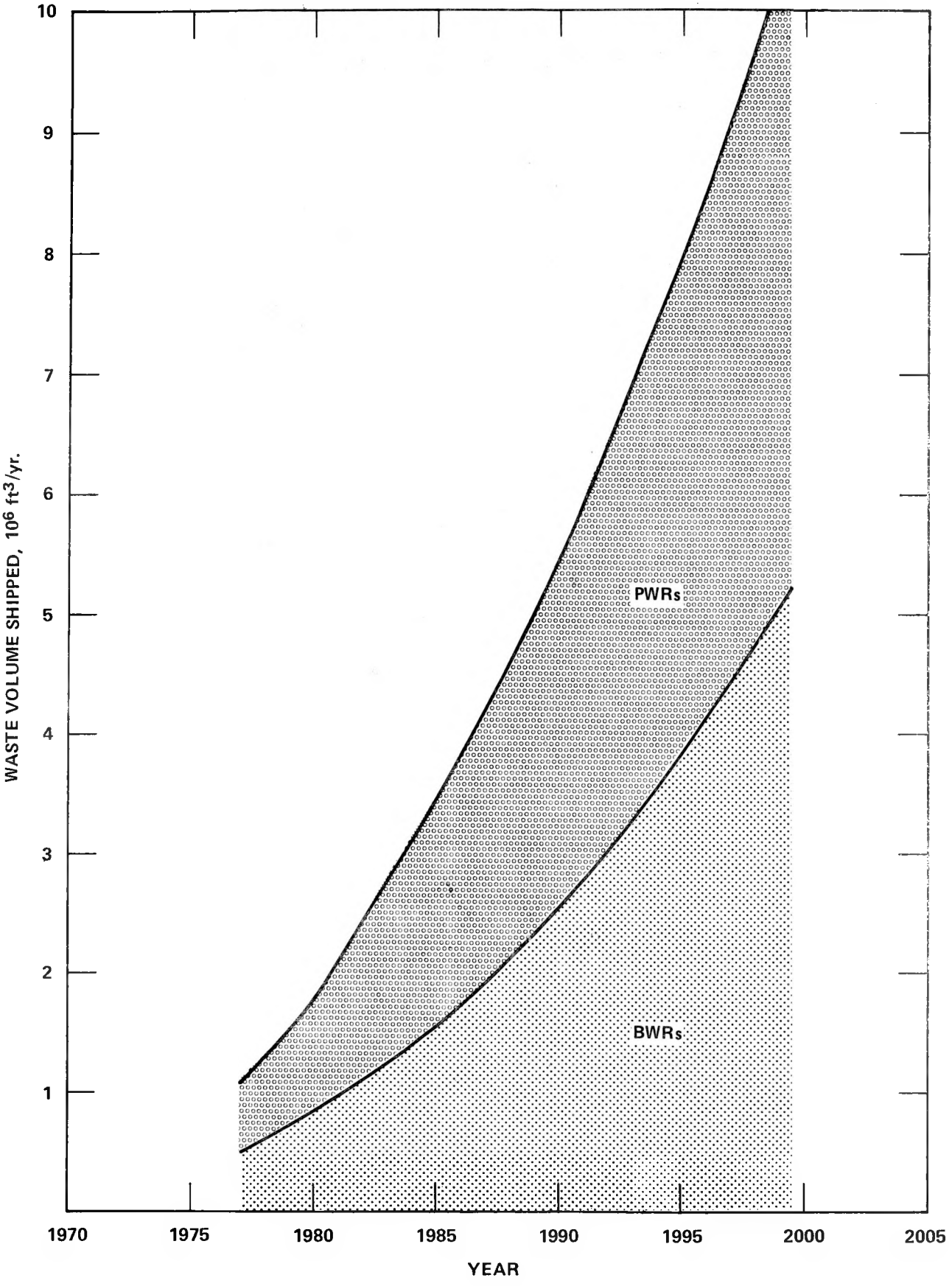

Figure 5.3-3 Annual Waste Volume Shipped With All Waste Solidified.

5.4 LWR Solid Waste Activities Through 2000

5.4.1 Total Activity Generated

5.4.1.1 Fission Products and Activated Corrosion Products

The total curies of radioactivity to be generated by LWR through 2000 are based entirely on the estimated curies produced of each reactor type and the projected installed nuclear generating capacity. The activity generation rates are given in Table 4.2-49 and the projected increases in generating capacity are given in Table 5.2-2. Table 5.4-1 and Figure 5.4-1 show the total curies estimated to be shipped from LWRs for burial through 2000.

5.4.1.2 Transuranic Radionuclides

Calculations using the estimated annual generation rates in Table 4.3-8 and the projected installed nuclear generating capacity in Table 5.2-2 yield the overall generation of transuranic radionuclides as shown in Table 5.4-2.

5.4.2 Concentrations in Unsolidified Wastes

5.4.2.1 Fission Products and Activated Corrosion Products

The gross activity concentrations of fission products and activated corrosion products have been discussed in Sections 4.2.1.1 through 4.2.1.5 and 4.2.2.1 through 4.2.2.5 and have been summarized in Tables 4.2-49 and 4.2-51. Concentrations of specific radionuclides for an average plant are given in Table 5.4-3.

5.4.2.2 Transuranic Radionuclides

The concentrations of transuranics are discussed in Section 4.3.1 and tabulated in Tables 4.3-3 through 4.3-6. These concentrations are summarized in Table 4.3-7.

5.4.3 Concentrations in Solidified Wastes

5.4.3.1 Fission Products, Activated Corrosion Products, and Transuranics

The extent to which the concentration of radionuclides change when the waste is solidified depends upon the solidification agent used. If the solidification agent doubles the volume then the concentration would be reduced by a factor of two. To determine the activity concentration of solidified wastes the values in Table 5.4-3 should be multiplied by the volume increase factor (i.e., packaging factor) listed in Section 3.4 for the specific solidification agent used.

Table 5.4-1 Total Activity Shipped From LWRs (10^3 Ci)

Year	BWR		PWR		Total
	Deep Bed CPS(1)	Precoat CPS	Without CPS	With CPS	
1977	26.5	8.0	18.7	3.9	57.1
1980	53.4	10.2	26.6	7.2	97.4
1985	111.0	13.8	49.2	16.7	191.0
1990	194.0	19.0	71.8	26.2	311.0
1995	300.0	25.6	101.0	38.5	465.0
2000	418.0	32.9	134.0	52.2	637.0

1. Condensate polishing system.

Table 5.4-2 Estimated Generation of Transuranic Radionuclides From LWRs Through 2000

Year	Radionuclide Generation (Ci/yr)				Total
	BWRs		PWRs		
	Deep Bed CPS(1)	Precoat CPS	Without CPS	With CPS	
1977	1.2	13.9	1.1	1.8	18.0
1980	2.5	17.8	1.5	3.3	25.1
1985	5.2	24.0	2.9	7.5	39.6
1990	9.1	33.0	4.2	11.9	58.2
1995	14.0	44.5	5.9	17.5	81.9
2000	19.6	57.3	7.8	23.7	108.0

1. Condensate polishing system.

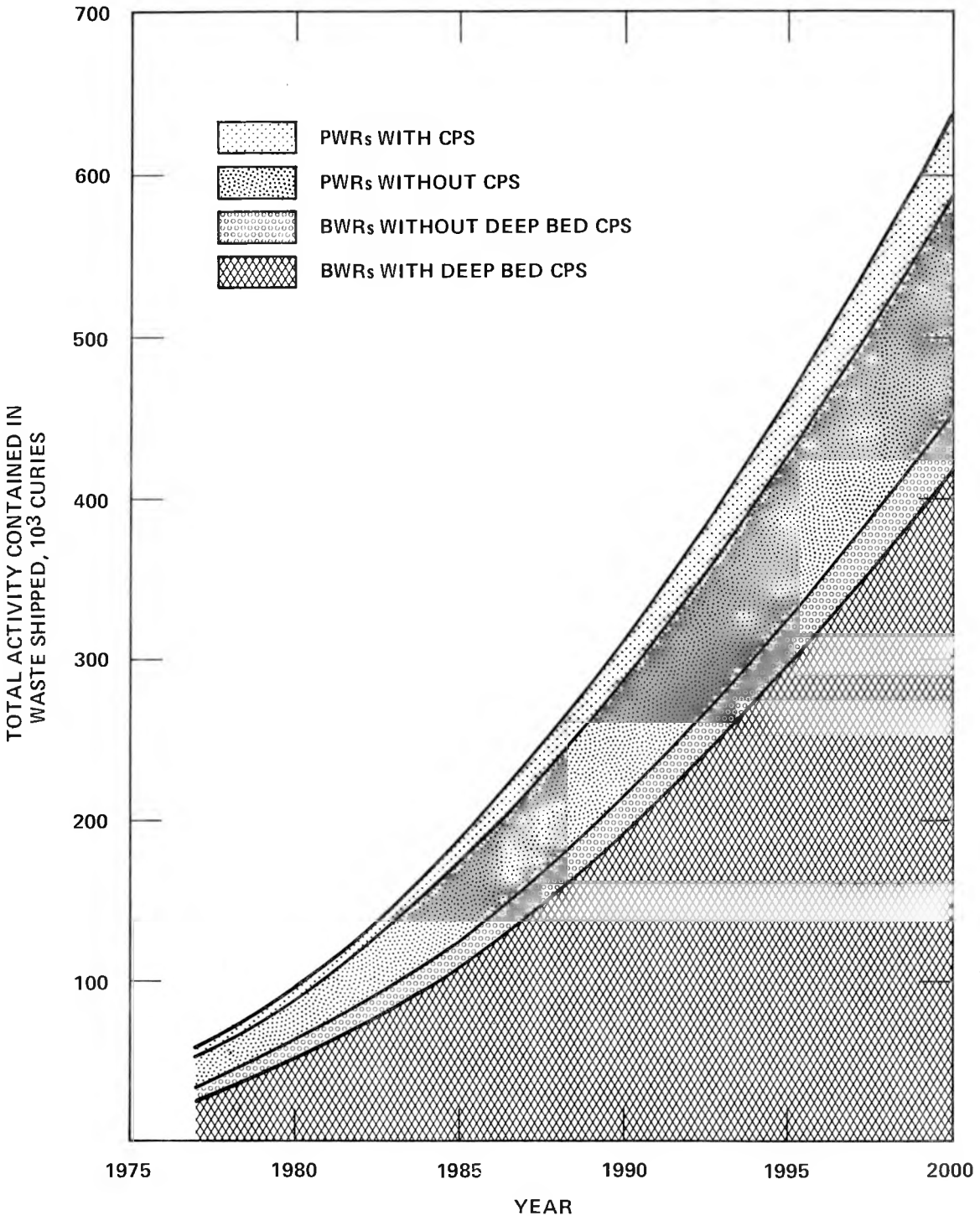

Figure 5.4-1 Total Activity Shipped From LWRs Through 2000.

Table 5.4-3 Radionuclide Concentrations in Unsolidified Wastes

Nuclide	BWRs With Deep Bed CPS[1]			BWRs With Precoat Filter CPS		PWRs Without CPS			PWRs With CPS	
	Spent Resin	Precoat Filter Sludge	Concentrator Bottoms	Compactible[2] Trash	Combined[3] Waste	Spent Resin	Concentrator Bottoms	Compactible[4] Trash	Spent Resin	Concentrator Bottoms
Fission products and activated corrosion products (Ci/ft^3)										
Cr-51		.0063			.00012	.070	.023			.0029
Mn-54	.0028	.085	.0012	.00014	.00082	.065	.00051	.000026	.0031	.00020
Fe-59		.0052			.00012	.0053	.0021			.00020
Co-58	.00043	.0022		<.000013	.0018	.11	.013	.00017	.0062	.00062
Co-60	.018	.19	.0052	.00028	.022	.081	<.0003	<.00013	.031	.00062
Zn-65	.00087	.0015	.00042		.019	.0021	<.0003		.0031	<.0002
Zr-95		.0011			.00047	.0032	.0054			<.0002
Cs-134	.11	.023	.0099	.000039	.0056	.052	<.0003	.000066	.19	<.0002
Cs-137	.28	.051	.012	.000077	.0093	.23	<.0003	.00013	.33	<.0002
Other	.0017	.0048	.017	.00013	.0018	.026	.0067	.00012	.059	<.0002
Total	.41	.37	.046	.00067	.061	.65	.051	.00064	.62	.005
Transuranics ($\mu Ci/ft^3$)										
U-234	--	--	.099	(5)		.54	--	(5)	--	--
U-235	.0048	1.2	.066		.061	.28	.022		<.0065	<5.7
U-238	.0048	1.2	.049		.097	1.0	.022		<.012	<5.7
Pu-235	.14	19.	.44		118.	9.3	1.4		.26	<8.6
Pu-239	.13	11.	.27		77.	19.	.36		2.6	<8.6
Am-241	.18	5.2	.17		18.	16.2	.66		.28	<12.
Total	.45	39.	1.1	(5)	210.	47.	2.5	(5)	3.0	<40.

1. Condensate polishing system.
2. Applicable to both types of BWRs.
3. Composite of spent resin, filter sludge, and concentrator bottoms.
4. Applicable to both types of PWRs.
5. Not Available.

5.5 Effects of Volume Reduction on Annual Waste Volumes

5.5.1 Introduction

In order to determine the effects of volume reduction processes on annual waste volumes, it is necessary to evaluate each waste type for each reactor system. The two alternatives examined are as follows:

1. A single-step process using a thin film extruder evaporator to effect the volume reduction on process wastes while simultaneously mixing the dried waste with bitumen binder; this alternative also includes an incinerator for burning the combustible trash. The dry ash from the incinerator is processed through the extruder to be mixed with the bitumen.

2. A two-step process using a calciner/incinerator for volume reduction followed by solidification with cement.

The effects of these processes on the volume of waste requiring disposal on an annual basis is given in Tables 5.5-1 and 5.5-2.

These values are then compared to the annual waste volumes generated assuming solidification of all wastes without prior volume reduction, referred to as the base case. In both cases noncombustible wastes are packaged in accordance with current practice.

The effect of the combined volume-reduction solidification process on the concentration of transuranic radionuclides in the waste is examined in Section 5.6.

5.5.2 Single-Step Volume Reduction/Solidification Process With Incineration

5.5.2.1 Boiling Water Reactors With A Deep Bed Condensate Polishing System

Use of the bitumen system results in a 60% reduction in annual waste volume. The largest decrease is in concentrator bottoms, from 22.0 ft^3/MWe-yr to 6.0 ft^3/MWe-yr, a reduction of 73%. Filter sludge volumes are dropped by 42% from 9.3 ft^3/MWe-yr to 5.4 ft^3/MWe-yr, and spent resins experience a 38% volume reduction to 4.6 ft^3/MWe-yr from 7.4 ft^3/MWe. The trash at a BWR is roughly 68% combustible. When incinerated the BWR trash drops from 11.5 ft^3/MWe-yr to 4.1 ft^3/MWe-yr (a drop of 64%), of which 3.7 ft^3/ MWe-yr is noncombustible.

5.5.2.2 Boiling Water Reactors With A Precoat Condensate Polishing System

In BWRs with a precoat condensate polishing system a majority of the waste volume is from the solidified filter precoat. By processing the waste through a volume-reduction system prior to

Table 5.5-1 Single-Step Volume-Reduction Process With Incineration and Immobilization in Bitumen Ft^3/MWe Installed Per Year

Waste type	Boiling Water Reactors		Pressurized Waste Reactors	
	Deep Bed CPS(1)	Precoat CPS	With CPS	Without CPS
Deep bed resin	4.6	.23	.32	.94
Concentrator bottoms	6.0(2)	.29(2)	.37(3)	.30(3)
Filter sludge	5.4	7.7	.15	--
Cartridge filters	--	--	.39	.39
Trash (with incinerator)	4.1	4.1	4.3	4.3
Total	20.1	12.3	5.53	5.93
1,000-MWe plant ft^3/yr	20,100.	12,300.	5,500.	5,900.

1. Condensate polishing system.
2. Based on 25 wt% solution of sodium sulfate.
3. Based on 12.5 wt% solution of boric acid.

Table 5.5-2 Two-Step Volume-Reduction Process With Immobilization in Cement, Ft^3/MWe Installed Per Year

Waste type	Boiling Water Reactors		Pressurized Waste Reactors	
	Deep Bed CPS(1)	Precoat CPS	With CPS	Without CPS
Deep bed resin	6.0	0.30	0.42	1.2
Concentrator bottoms	4.8	.23	1.1	.83
Filter sludge	6.6	9.5	.18	--
Cartridge filters	--	--	.39	.39
Trash (with incinerator)	4.0	4.0	4.2	4.2
Total	21.4	14.0	6.29	6.62
1,000-MWe plant ft^3/yr	21,400.	14,000.	6,300.	6,600.

1. Condensate polishing system.

solidification, the final volume is reduced from 13.2 ft^3/MWe-yr to 7.7 ft^3/MWe-yr. The next most abundant form of waste is reduced from 11.5 ft^3/MWe-yr to 4.1 ft^3/MWe-yr, of which 3.7 ft^3/MWe-yr is noncombustible. Concentrator bottoms and spent resins represent a minimal contribution to the total waste volume at 0.29 ft^3/MWe-yr and 0.37 ft^3/MWe-yr, respectively. The percent reductions are basically the same as for BWRs with a deep bed condensate polishing system, varying at most by one percent.

5.5.2.3 Pressurized Water Reactors With A Condensate Polishing System

Unlike BWRs in which the type of condensate polishing system predicts the major waste type, PWR waste is dominated by incinerated and noncombustible trash. Combined they form 78% of the solidified waste from a PWR with a condensate polishing system. The 4.3 ft^3/MWe-yr of solidified ash and noncombustibles (3.9 ft^3/MWe-yr) is a reduction of 63% from the 11.5 ft^3/MWe-yr in the base case. Cartridge filters at 0.39 ft^3/MWe-yr, slightly over 9% of the annual volume of trash, are the second largest contributors. This is unchanged from the base case because cartridge filters are loaded into drums that are then filled with other waste that is being solidified. In this way they only occupy their own space and do not cause a net increase in disposable volume. Concentrator bottoms average 0.37 ft^3/MWe-yr, a 96% reduction over the base case volume of 8.8 ft^3/MWe-yr. For deep bed resins and filter sludge the volume reduction is approximately 40%, from 0.54 ft^3/MWe-yr to 0.32 ft^3/MWe-yr and from 0.25 ft^3/MWe-yr to 0.15 ft^3/MWe-yr, respectively.

5.5.2.4 Pressurized Water Reactors Without A Condensate Polishing System

In PWRs with or without a condensate polishing system, the same situation is observed regarding trash and cartridge filters. Thus, 4.3 ft^3/MWe-yr of the total 5.9 ft^3/MWe-yr will be solidified ash and noncombustible trash along with 0.39 ft^3/MWe-yr of cartridge filters. In PWRs without a condensate polishing system, concentrator bottoms account for only 0.30 ft^3/MWe-yr, a reduction of 96%, down from 8.0 ft^3/MWe-yr in the base case. Spent resins account for 0.94 ft^3/MWe-yr down from 1.6 ft^3/MWe-yr, a 41% decrease. Filter sludge is not found in appreciable amounts in PWRs with a condensate polishing system.

5.5.3 Two-Step Volume Reduction/Solidification Process With Incineration

The systems assumed here for the processing of radwaste are the Aerojet Energy Conversion Company fluid bed dryer and incinerator coupled to a cement solidification system. Combining the gross volume-reduction factors for dried concentrator bottoms and incinerated combustible trash with the volume-increase factors for dried salts, ash, resin, and filter sludge solidified in cement

results in net volumes per MWe-yr for the various types of waste as shown in Table 5.5-2.

The packaging efficiencies used are taken from Table 3.4-7. These numbers result in lower annual waste volumes than would be calculated using packaging efficiencies indicative of current practices; see Tables 4.1-2 and 4.1-3. These lower numbers were used for two reasons: first, as more experience is gained in the operation of radwaste solidification systems and waste management, better packaging efficiencies will be experienced on a plant-by-plant basis. Second, many plants may decide to process resins and filter sludge through a fluid bed dryer thus resulting in further volume reduction as prescribed by Newport News Industrial Corp., (see Section 3.5.3.2).

5.5.3.1 Boiling Water Reactors With A Deep Bed Condensate Polishing System

The two-step volume-reduction system applied to a BWR employing a deep bed condensate polishing system will result in a 57% reduction in annual waste volume. Of the total 21.4 ft^3/MWe-yr, 19% is immobilized ash; and noncombustible trash is 4.0 ft^3/MWe-yr. Twenty-two percent of the total annual waste volume is solidified concentrator bottoms, down to 4.8 ft^3/MWe-yr with solidified filter sludge accounting for 31% of the total at 6.6 ft^3/MWe-yr. The remaining waste, immobilized resin, at 6.0 ft^3/MWe-yr accounts for 28% of the annual waste.

5.5.3.2 Boiling Water Reactors With A Precoat Condensate Polishing System

As with deep bed condensate polishing system plants, the largest contributor to the final waste volume is filter sludge which at 9.5 ft^3/MWe-yr constitutes 68% of the total annual waste of 14.0 ft^3/MWe-yr. Solidified ash and noncombustible trash amounts to 4.0 ft^3/MWe-yr, or approximately 28% of the annual total. Resin contributes 2%, at .30 ft^3/MWe-yr, and concentrator bottoms less than 2%, 0.23 ft^3/MWe-yr. The 14.0 ft^3/MWe-yr total represents a reduction of 47% over the base case of 26.2 ft^3/MWe-yr.

5.5.3.3 Pressurized Water Reactors With A Condensate Polishing System

After processing through a two-step volume-reduction system, the immobilized ash and noncombustible trash accounts for 67% of the total annual waste volume of 6.3 ft^3/MWe-yr, or approximately 4.2 ft^3/MWe-yr. Concentrator bottoms account for the second largest contribution at 17% of the total with 1.1 ft^3/MWe-yr. Cartridge filters are assumed to be solidified in containers filled with other wastes such that they take up only their own volume. Thus the 0.39 ft^3/ MWe-yr of untreated waste remains unchanged and contributes 6% of the total annual volume. The remainder of the waste is spent resin, at 0.42 ft^3/MWe-yr, and filter sludge, at 0.18 ft^3/MWe-yr, which when immobilized

in cement amounts to 7% and 3% of total annual volume respectively. The 6.3 ft^3//MWe-yr represents a 71% reduction in annual waste volume compared to the base case of 21.5 ft^3/MWe-yr.

5.5.3.4 Pressurized Water Reactors Without A Condensate Polishing System

As with PWRs with condensate polishing systems, the largest contribution to the total annual waste volume is still solidified ash and noncombustibles, amounting to 4.2 ft^3/MWe-yr, or 63% of a total of 6.6 ft^3/MWe-yr. Concentrator bottoms, once immobilized, constitute 13% of the total, at 0.8 ft^3/MWe-yr. Cartridge filters, for the reason discussed in Section 5.5.3.3, with 0.39 ft^3/MWe-yr contribute 6% of the total. Approximately 18% of the total annual volume, 1.2 ft^3/MWe-yr, is contributed by solidified spent resin.

5.5.4 Summary

Table 5.5-3 is a summary for both alternatives of the statistics in Sections 5.5.2 and 5.5.3 showing the annual waste volume of each waste type, what percentage that waste type is of the total, and what percentage it is of the base case volume.

Both of these systems may be purchased with or without the incinerator. If an incinerator is not used, the total annual waste volume would increase from 37 to 60% for a BWR, and from 111 to 131% for a PWR depending on the method of condensate cleanups and which volume reduction system is under consideration.

The use of solidification agents such as Dow's polyester resin or urea formaldehyde appears to result in slightly higher waste volumes for the second alternative than results from solidification with cement. Because the calciner/incinerator system has not yet been installed in an operating plant, information on the ratios of binder to waste is based solely on preliminary tests. Therefore, it is assumed for the purpose of this study that the final volumes of solidified waste are the same regardless of the solidification product used.

Comparing the projected annual waste-volume generation rates per megawatt (electric) of installed capacity as shown in Tables 5.5-1 and 5.5-2 indicates that the differences between using a bituminization system and using a fluid bed dryer with cement solidification is less than 15%. Table 5.5-4 shows these final totals, the percent difference, and the average of the two for BWRs and PWRs based on the type of condensate polishing system. The total annual waste shipped to shallow land burial sites is given in Table 5.5-5. These figures are based on the assumption that there is no appreciable volume reduction through 1980 and that all LWRs are practicing volume reduction with subsequent solidification of all waste (except noncombustible waste) by 1985. These estimates are given graphically in Figure 5.5-1.

Table 5.5-3 Summary of Solidified Waste Volumes Following Volume Reduction

Type of Plant, CPS[1], and Waste	Single-Step Volume Reduction Solidification		Two-Step Volume Reduction Solidification		Base Case (no volume reduction)	
	Immobilized Volume[2] (ft^3/MWe-yr)	% of Total	Immobilized Volume[3] (ft^3/MWe-yr)	% of Total	All Waste Solidified[4] (ft^3/MWe-yr)	No Solidification (ft^3/MWe-yr)
BWR deep bed CPS						
Total	20.1	100	21.4	100	50.2	34.2
Resin	4.6	23	6.0	28	7.4	4.6
Bottoms	6.0	30	4.8	22	22.0	12.7
Sludge	5.4	27	6.6	31	9.3	5.4
Cartridge	-	-	-	-	-	-
Trash	4.1	20	4.0	19	11.5	11.5
BWR precoat CPS						
Total	12.3	100	14.0	100	26.2	20.0
Resin	.23	2	.30	2	1.1	.23
Bottoms	.29	2	.23	2	1.1	.6
Sludge	7.7	63	9.5	68	13.2	7.7
Cartridges	-	0	-	0	-	-
Trash	4.1	33	4.0	28	11.5	11.5
PWR with CPS						
Total	5.53	100	6.29	100	21.5	16.7
Resin	.32	6	.42	7	.54	.32
Bottoms	.37	7	1.1	17	8.8	4.8
Sludge	.15	3	.18	3	.25	.15
Cartridge	.39	7	.39	6	.39	.39
Trash	4.3	78	4.2	67	11.5	11.5
PWR without CPS						
Total	5.93	100	6.62	100	21.5	17.2
Resin	.94	16	1.2	18.	1.6	.94
Bottoms	.30	5	0.83	13	8.0	3.9
Sludge	-	0	-	0	-	-
Cartridge	.39	7	.39	6	.39	.39
Trash	4.3	72	4.2	63	11.5	11.5

1. Condensate polishing system.
2. Based on packaging efficiencies in Table 3.4-15.
3. Based on packaging efficiencies in Table 3.4-7.
4. Based on current solidification practices as noted in Tables 4.1-2 and 4.1-3.

Table 5.5-4 Comparative Summary of Volume-Reduction/ Solidification Processes (ft^3/MWe)

	BWR		PWR	
	Deep Bed	Precoat	With CPS[1]	Without CPS
Single step with asphalt	20.1	12.3	5.5	5.9
Double step with cement	21.4	14.0	6.3	6.6
Percent difference	6.47	13.8	14.5	12.0
Average	20.8	13.2	5.9	6.2

1. Condensate polishing system.

Table 5.5-5 Estimated Annual Waste Volumes Shipped Through 2000 (10^3 ft^3/yr)

	BWRs			PWRs			
Year	Deep Bed	Precoat	Total	With CPS[1]	Without CPS	Total	Cumulative
1977[2]	248	206	454	198	394	592	1,050
1980[3]	550	290	840	370	570	940	1,780
1985[4]	470	200	670	230	310	540	1,210
1990	820	270	1,090	370	450	820	1,910
1995	1,280	370	1,650	540	630	1,170	2,820
2000	1,770	470	2,240	740	830	1,570	3,810

1. Condensate polishing system.
2. Based on current practices.
3. Minimal Volume-reduction, all process wastes solidified.
4. Volume-reduction techniques instituted at all plants, all wastes solidified.

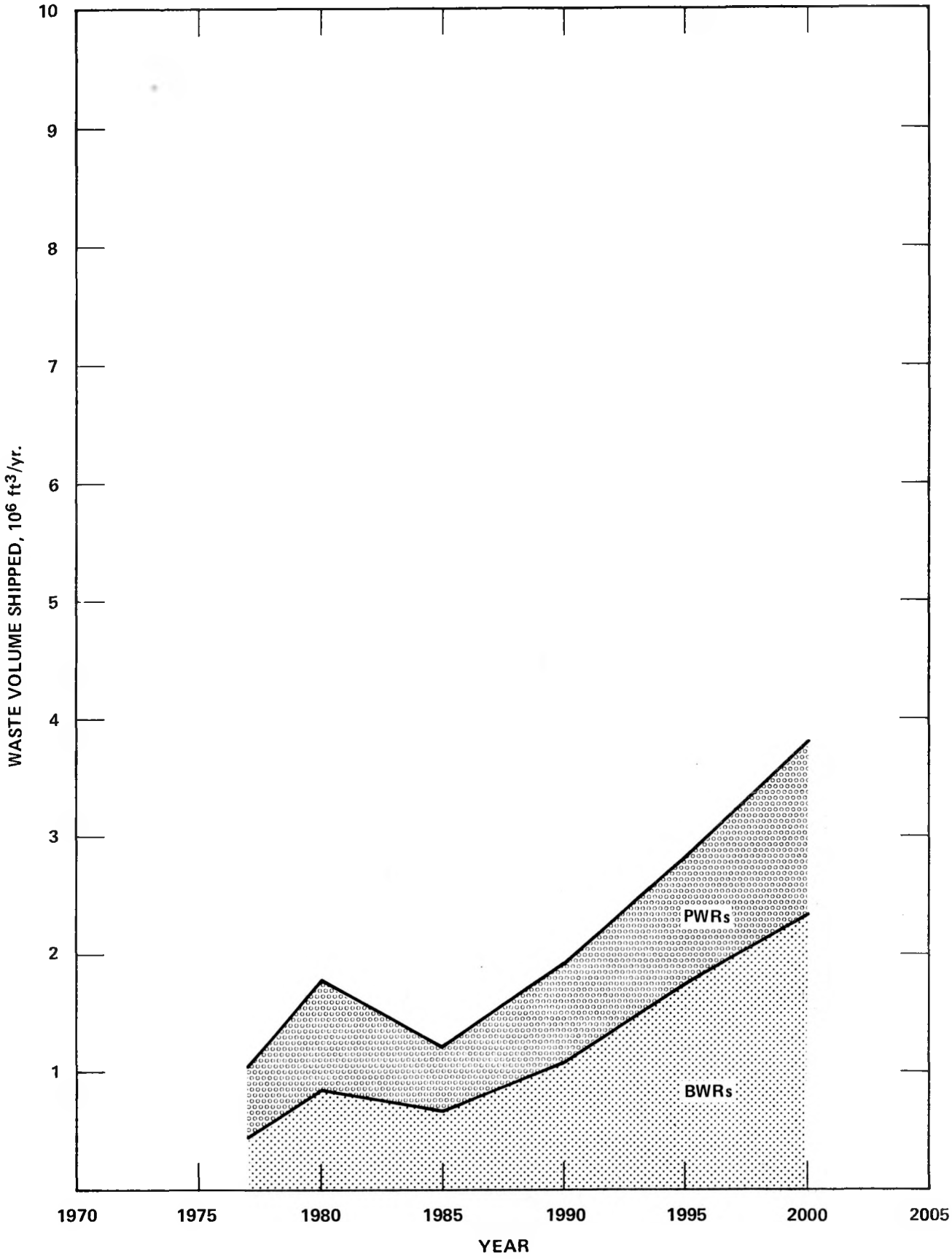

Figure 5.5-1 Annual Waste Volume Shipped With Phased-In Volume Reduction.

5.6 Effect of Waste Volume Reduction on Radionuclide Concentrations

As waste materials from LWRs are processed through volume-reduction systems, their radionuclide concentrations increase by the same factor that the waste volume decreases. Therefore, if the volume-reduction factor is two, the increase in radionuclide concentration is a factor of two.

The following sections show what happens as LWR wastes are treated for volume reduction and subsequent solidification. Only total activity levels are discussed. Isotopic distributions are listed in Tables 5.6-1 and 5.6-2. These tables also list the total and individual isotopic concentrations of the transuranic radionuclides based on the concentrations in Tables 4.3-3 through 4.3-6 for unsolidified waste. These concentrations include the effects of the volume-reduction process, volume increase from solidification, and the change in density caused by solidification. Concentrations are given in microcuries per gram.

5.6.1 Single-Step Volume Reduction and Solidification Process With Incineration

5.6.1.1 Boiling Water Reactors With A Deep Bed Condensate Polishing System

When applied to deep bed resin and filter sludge, the bituminization process drives off a quantity of water equal to the quantity of asphalt required for solidification. Therefore, the radionuclide concentrations in the solidified waste are the same as the concentrations in the unsolidified waste. The total activity in spent resins and filter sludge is 0.41 Ci/ft^3 and 0.37 Ci/ft^3, respectively. Concentrator bottoms lose three quarters of their weight as water vapors. When solidified in asphalt the net effect is to increase the activity concentration by a factor of 2.12, to 0.098 Ci/ft^3. Assuming that all of the combustible trash is contaminated with radioactivity at the levels identified for compactible trash, the gross concentration of the incinerator ash, mixed with asphalt, is 0.012 Ci/ft^3. The individual nuclide concentrations are given in Table 5.6-1.

5.6.1.2 Boiling Water Reactors With A Precoat Filter Condensate Polishing System

In plants of this type the precoat filter sludge accounts for 90% of the unsolidified waste volume. The only sample available from a BWR with a precoat filter condensate polishing system was a composite sample consisting of deep bed resin, filter sludge, and concentrator bottoms. Accounting for no net volume reduction for the deep bed resin or the filter sludge, and accounting for a volume-reduction factor of two for concentrator bottoms, the increase in the radionuclide concentration for a composite sample

Table 5.6-1 Radionuclide Concentrations, Single-Step Volume Reduction and Solidification System With Incinerator, Immobilization in Asphalt

Nuclide	BWRs With Deep Bed CPS[1]			BWRs With Precoat CPS		PWRs Without a CPS			PWRs With a CPS		
	Spent Resin	Precoat Filter Sludge	Concentrator Bottoms	Compactible Trash	Combined Waste	Spent Resin	Concentrator Bottoms	Compactible Trash	Spent Resin	Concentrator Bottoms	Precoat Filter Sludge[2]
Nontransuranics (Ci/ft^3)											
Cr-51		6.3 (-3)[3]			1.2 (-4)	7.0 (-2)	3.0 (-1)			3.8 (-2)	
Mn-54	2.8 (-3)	8.5 (-2)	2.5 (-3)	2.4 (-3)	8.2 (-4)	6.5 (-2)	6.6 (-3)	4.5 (-4)	3.1 (-3)	2.6 (-3)	
Fe-59		5.2 (-3)			1.2 (-4)	5.3 (-3)	2.7 (-2)			2.6 (-3)	
Co-58	4.3 (-4)	2.2 (-3)		2.2 (-4)	1.8 (-3)	1.1 (-1)	1.7 (-1)	2.9 (-3)	6.2 (-3)	8.1 (-3)	
Co-60	1.8 (-2)	1.9 (-1)	1.1 (-2)	4.8 (-3)	2.2 (-2)	8.1 (-2)	3.9 (-3)	2.2 (-3)	3.1 (-2)	8.1 (-3)	
Zn-65	8.7 (-4)	1.5 (-3)	8.9 (-4)		1.9 (-2)	2.1 (-3)	3.9 (-3)		3.1 (-3)	2.6 (-3)	
Zr-95		1.1 (-3)			4.7 (-4)	3.2 (-3)	7.0 (-2)			2.6 (-3)	
Cs-134	1.1 (-1)	2.3 (-2)	2.1 (-2)	6.7 (-4)	5.6 (-3)	5.2 (-2)	3.9 (-3)	1.1 (-3)	1.9 (-1)	2.6 (-3)	
Cs-137	2.8 (-1)	5.1 (-2)	2.5 (-2)	1.3 (-3)	9.3 (-3)	2.3 (-1)	3.9 (-3)	2.2 (-3)	3.3 (-1)	2.6 (-3)	
Other	1.7 (-3)	4.8 (-3)	3.6 (-2)	2.2 (-3)	1.8 (-3)	2.6 (-2)	8.7 (-2)	2.1 (-3)	5.9 (-2)	2.6 (-3)	
Total	4.1 (-1)	3.7 (-1)	9.8 (-2)	1.2 (-2)	6.1 (-2)	6.5 (-1)	6.6 (-1)	1.1 (-2)	6.2 (-1)	6.5 (-2)	8.0 (-2)
Transuranics (μCi/g)											
U-234	-	-	2.8 (-6)	(2)	-	1.7 (-5)	-	(2)	-		
U-235	1.4 (-7)	3.7 (-5)	1.8 (-6)		1.9 (-6)	8.6 (-6)	8.0 (-6)		1.9 (-7)	2.0 (-3)	(2)
U-238	1.4 (-7)	3.7 (-5)	1.4 (-6)		3.0 (-6)	3.2 (-5)	8.0 (-6)		3.6 (-7)	2.0 (-3)	
Pu-238	4.1 (-6)	5.7 (-4)	1.2 (-6)		3.5 (-3)	2.9 (-4)	5.1 (-4)		7.9 (-6)	3.0 (-3)	
Pu-239	3.9 (-6)	3.2 (-4)	7.4 (-6)[4]		2.2 (-3)	5.8 (-4)[4]	1.3 (-4)		7.9 (-5)	3.0 (-3)	
Am-241	5.3 (-6)	1.6 (-4)	4.6 (-6)		5.8 (-4)	5.0 (-4)	2.4 (-4)		8.6 (-6)	4.0 (-3)	
Am-243	-										
Total	1.4 (-5)	1.1 (-3)	3.0 (-5)	(2)	6.4 (-3)	1.4 (-3)	9.0 (-4)	(2)	9.4 (-5)	1.4 (-2)	(2)

1. Condensate polishing system.
2. Not available.
3. 6.3 (-3) means 6.3×10^{-3}.
4. Includes Pu-240.

Table 5.6-2 Radionuclide Concentrations, Two-Step Volume Reduction and Solidification System With Incinerator, Immobilization in Cement

Nuclide	BWRs With Deep Bed CPS[1] Spent Resin	BWRs With Deep Bed CPS[1] Precoat Filter Sludge	BWRs With Deep Bed CPS[1] Concentrator Bottoms	BWRs With Precoat CPS Compactible Trash	BWRs With Precoat CPS Combined Waste	PWRs Without A CPS Spent Resin	PWRs Without A CPS Concentrator Bottoms	PWRs Without A CPS Compactible Trash	PWRs with a CPS Spent Resin	PWRs with a CPS Concentrator Bottoms	PWRs with a CPS Precoat Filter Sludge[2]
Nontransuranics (Ci/ft^3)											
Cr-51		5.2 (-3)[3]			1.0 (-4)	5.5 (-2)	1.1 (-1)			1.3 (-2)	
Mn-54	2.2 (-3)	7.0 (-2)	3.2 (-3)	3.5 (-3)	7.0 (-4)	5.1 (-2)	2.4 (-3)	6.5 (-4)	2.4 (-3)	8.7 (-4)	
Fe-59		4.2 (-3)			1.0 (-4)	4.2 (-3)	9.9 (-3)			8.7 (-4)	
Co-58	3.3 (-4)	1.8 (-3)		3.2 (-4)	1.5 (-3)	8.6 (-2)	6.1 (-2)	4.2 (-3)	4.7 (-3)	2.7 (-3)	
Co-60	1.4 (-2)	1.6 (-1)	1.4 (-2)	7.0 (-3)	1.9 (-2)	6.3 (-2)	1.4 (-3)	3.2 (-3)	2.4 (-2)	2.7 (-3)	
Zn-65	6.7 (-4)	1.2 (-3)	1.1 (-3)		1.6 (-2)	1.6 (-3)	1.4 (-3)		2.4 (-3)	9.0 (-4)	
Zr-95		9.0 (-4)			4.0 (-4)	2.5 (-3)	2.5 (-2)			9.0 (-4)	
Cs-134	8.5 (-2)	1.9 (-2)	2.6 (-2)	9.8 (-4)	4.8 (-3)	4.1 (-2)	1.4 (-3)	1.6 (-3)	1.5 (-1)	9.0 (-4)	
Cs-137	2.2 (-1)	4.2 (-2)	3.2 (-2)	1.9 (-3)	7.9 (-3)	1.8 (-1)	1.4 (-3)	3.2 (-3)	2.5 (-1)	9.0 (-4)	
Other	1.3 (-3)	3.9 (-3)	4.5 (-2)	3.2 (-3)	1.5 (-3)	2.0 (-2)	3.2 (-2)	3.0 (-3)	4.5 (-2)	9.0 (-4)	
Total	3.2 (-1)	3.0 (-1)	1.2 (-1)	1.7 (-2)	5.2 (-2)	5.1 (-1)	2.4 (-1)	1.6 (-1)	4.7 (-1)	2.2 (-2)	6.7 (-2)
Transuranics (μCi/g)											
U-234	-	-	6.6 (-6)	[2]		1.1 (-5)	-	[2]	-	-	
U-235	9.4 (-8)	2.4 (-5)	4.4 (-6)		1.2 (-6)	5.7 (-6)	2.3 (-6)		1.3 (-7)	5.3 (-4)	[2]
U-238	9.4 (-8)	2.4 (-5)	3.3 (-6)		1.9 (-6)	2.1 (-5)	2.3 (-6)		2.3 (-7)	5.3 (-4)	
Pu-238	2.7 (-6)	3.7 (-4)	2.9 (-5)		2.3 (-3)	1.9 (-4)	1.5 (-4)		5.1 (-6)	8.0 (-4)	
Pu-239	2.5 (-6)	2.1 (-4)	1.8 (-5)[4]		1.5 (-3)	3.8 (-4)[4]	3.7 (-5)		5.1 (-5)	8.0 (-4)	
Am-241	3.5 (-6)	1.0 (-4)	1.1 (-5)		3.7 (-4)	3.3 (-4)	6.9 (-5)		5.6 (-6)	1.1 (-3)	
Am-243											
Total	8.9 (-6)	7.5 (-4)	7.3 (-5)	[2]	4.1 (-3)	9.6 (-4)	2.6 (-4)	[2]	6.0 (-5)	3.7 (-3)	[2]

1. Condensate polishing system.
2. Not available.
3. 5.2 (-3) means 5.2×10^{-3}.
4. Includes Pu-240.

is only 4%. Therefore, the total activity concentration of unsolidified waste and waste solidified with asphalt is 0.061 Ci/ft^3. Concentrations of individual isotopes are given in Table 5.6-1.

5.6.1.3 Pressurized Water Reactors With A Condensate Polishing System

As with BWRs the total activity concentration in spent resins remains unchanged when solidified in asphalt. In PWRs with a condensate polishing system the total concentration of radioactivity in spent resins is 0.62 Ci/ft^3. Concentrator bottoms, at 12.5 wt% boric acid initially, end up concentrating the activity by a factor of 13, from 0.005 Ci/ft^3 to 0.065 Ci/ft^3 (unsolidified). Using the same assumption for combustible trash in PWRs that was made for combustible trash in BWRs, the solidified ash will have a gross activity concentration of 0.011 Ci/ft^3.

The precoat filter sludge from precoat filters in the condensate polishing system will also have the same concentration solidified with asphalt, as unsolidified, that is 0.08 Ci/ft^3. With the exception of precoat filter sludge, for which no appropriate sample was available, the individual isotopic concentrations are given in Table 5.6-1.

5.6.1.4 Pressurized Water Reactors Without A Condensate Polishing System

The concentration of radioactivity in the resin is 0.65 Ci/ft^3 when solidified in asphalt, the same as the unsolidified concentration. The concentration of activity in concentrator bottoms will increase to 0.66 Ci/ft^3. Solidified incinerator ash immobilized in asphalt will have a gross activity concentration of 0.011 Ci/ft^3. Cartridge filters for PWRs with and without a condensate polishing system are not processed through volume-reduction systems. Therefore, the 0.12 Ci/ft^3 associated with cartridge filters is unchanged. Individual isotopic concentrations are given in Table 5.6-1.

5.6.2 Two-Step Volume Reduction Solidification Process With Incineration

This process involves the processing of concentrator bottoms through an Aerojet Energy Conversion Company fluid-bed dryer and the burning of all combustible trash in the accompanying incinerator. The salts from the dryer, the ash from the incinerator, and the resin and filter sludge are then solidified with cement.

5.6.2.1 Boiling Water Reactors With A Deep Bed Condensate Polishing System

Deep bed resins and filter sludges are not volume reduced. These wastes are collected and solidified with the resulting volume increase resulting in a decrease in the overall radioactive concentration. The total radioactivity concentration for resins is

0.32 Ci/ft^3 and for filter sludge it is 0.30 Ci/ft^3. Concentrator bottoms experience a net volume reduction of 62.4% which results in a concentration of 0.12 Ci/ft^3. For trash it is assumed that the combustible noncompactible trash has the same concentrations of radioactivity associated with it as does the compactible trash.

This waste when incinerated and solidified in cement will have a gross radioactivity concentration of 0.017 Ci/ft^3. Individual isotope concentrations are given in Table 5.6-2.

5.6.2.2 Boiling Water Reactors With A Precoat Condensate Polishing System

As discussed in Section 5.6.1.2 the only sample available for BWRs with a precoat condensate polishing system is a composite sample of spent resins, precoat filter sludge, and concentrator bottoms. Only the concentrator bottoms would be affected by the volume-reduction system, and since they only represent 7% of the total waste volume, they will have a minimal effect on the final activity concentration, which is 0.052 Ci/ft^3. The concentration of radioactivity in solidified incinerator ash (0.017 Ci/ft^3) is identical to the concentration in solidified incinerator ash for BWRs with a deep bed condensate polishing system. Concentrations of individual radionuclides are given in Table 5.6-2.

5.6.2.3 Pressurized Water Reactors With A Condensate Polishing System

The volume of deep bed resin is not affected by the fluid bed dryer because this type of waste is not processed through the volume-reduction system. The final waste represents only the solidified resin at 1.3 times the original unsolidified volume. The concentration of radioactivity is then reduced by a factor of 1.3, to 0.47 Ci/ft^3. Concentrator bottoms that are nearly 88% water have a net decrease in volume after solidification of the salt produced in the fluid-bed dryer. The total activity concentration, after solidification of the dried concentrator bottoms, is 0.022 Ci/ft^3. With the same assumptions used for PWR trash as for BWR trash the activity level in the solidified incinerator ash will be 0.16 Ci/ft^3. The total radioactivity concentration in the small amount of filter sludge will be about 0.67 Ci/ft^3 and cartridge filters will contain about 0.12 Ci/ft^3. Individual radionuclide concentrations for those wastes for which data are available are given in Table 5.6-2.

5.6.2.4 Pressurized Water Reactors Without A Condensate Polishing System

The concentration of radioactivity on the spent resins when solidified is 0.51 Ci/ft^3 and for the concentration bottoms is 0.24 Ci/ft^3. Cartridge filter concentrations and incinerator ash concentrations will be the same as for PWRs with a condensate polishing system, that is 0.12 Ci/ft^3 and 0.16 Ci/ft^3 respectively. Individual nuclide concentrations are given in Table 5.6-2.

5.7 Fuel Fabrication Facility Waste Volumes and Activities Through 2000

The volume of waste resulting from LWR fuel fabrication, and the associated activity, through 2000 is given in Table 5.7-1 and shown graphically in Figure 5.7-1. Waste volume is estimated from the average waste volume generation rate of 80 ft^3/MTU in conjunction with the projected United States fuel production figures in Table 4.4-5. Waste activities are based on an average generation rate of 12 $\mu Ci/ft^3$ of uranium.

5.8 Burial Site Capacity

5.8.1 Maxey Flats

Maxey Flats is a 252-acre site previously leased from the Commonwealth of Kentucky by Nuclear Engineering Company, Inc. Recently the state bought out the lease and closed the site. Discounting the buffer zone on the northern boundary of the site, that portion of the site which is unsuitable for burial due to the terrain, and the area where waste is already buried, there were approximately 87 acres available for radioactive waste burial at the time the site was closed. It is unlikely that the state will reopen the site or approve another site within the state.

5.8.2 Barnwell

Barnwell is a 256-acre site in North Carolina owned by Chem Nuclear Systems, Inc. Of the original acreage 235 acres have been deeded to the state. Discounting a 100-foot-wide buffer zone (that portion of the site which is unsuitable for burial due to the terrain), and the area where waste is already buried, there are approximately 160 acres still available for burial. Expansion of the site is planned by Chem Nuclear Systems, Inc., although the exact acreage available has not been determined.

5.8.3 Sheffield

Sheffield is a 22-acre site leased from the State of Illinois. The majority of the site has been utilized, and very little space is available. The site is not now accepting waste for burial. Nine acres may be made available if the NRC approves a proposed backfill plan. The site operator recently purchased 120 acres adjacent to the site, and 80 acres of that purchase may be suitable for burial. The NRC is reviewing the permit application to expand the site.

5.8.4 Richland

Richland is a 100-acre site subleased by the Nuclear Engineering Company, Inc., from the State of Washington. This 100 acres is part of a 1,000-acre tract of land on the Hanford Reservation which the state has leased from the federal government. The remaining 900 acres have not been subleased for other activities,

Table 5.7-1 Volumes and Activities of Low-Level Waste From United States Fuel-Fabrication Facilities

Year	In Support of Domestic Demand			Total Industry Production		
	10^3 ft^3	Ci/yr	10^3g/yr	10^3 ft^3/yr	Ci/yr	10^3g/yr
1980	216.	2.6	7,500.	320.	3.8	11,200.
1985	376.	4.5	13,100.	560.	6.7	19,600.
1990	544.	6.5	18,700.	800.	9.6	28,000.
1995	776.	9.3	27,300.	1,170.	14.0	40,900.
2000	1,000.	12.0	35,100.	1,500.	18.0	52,600.

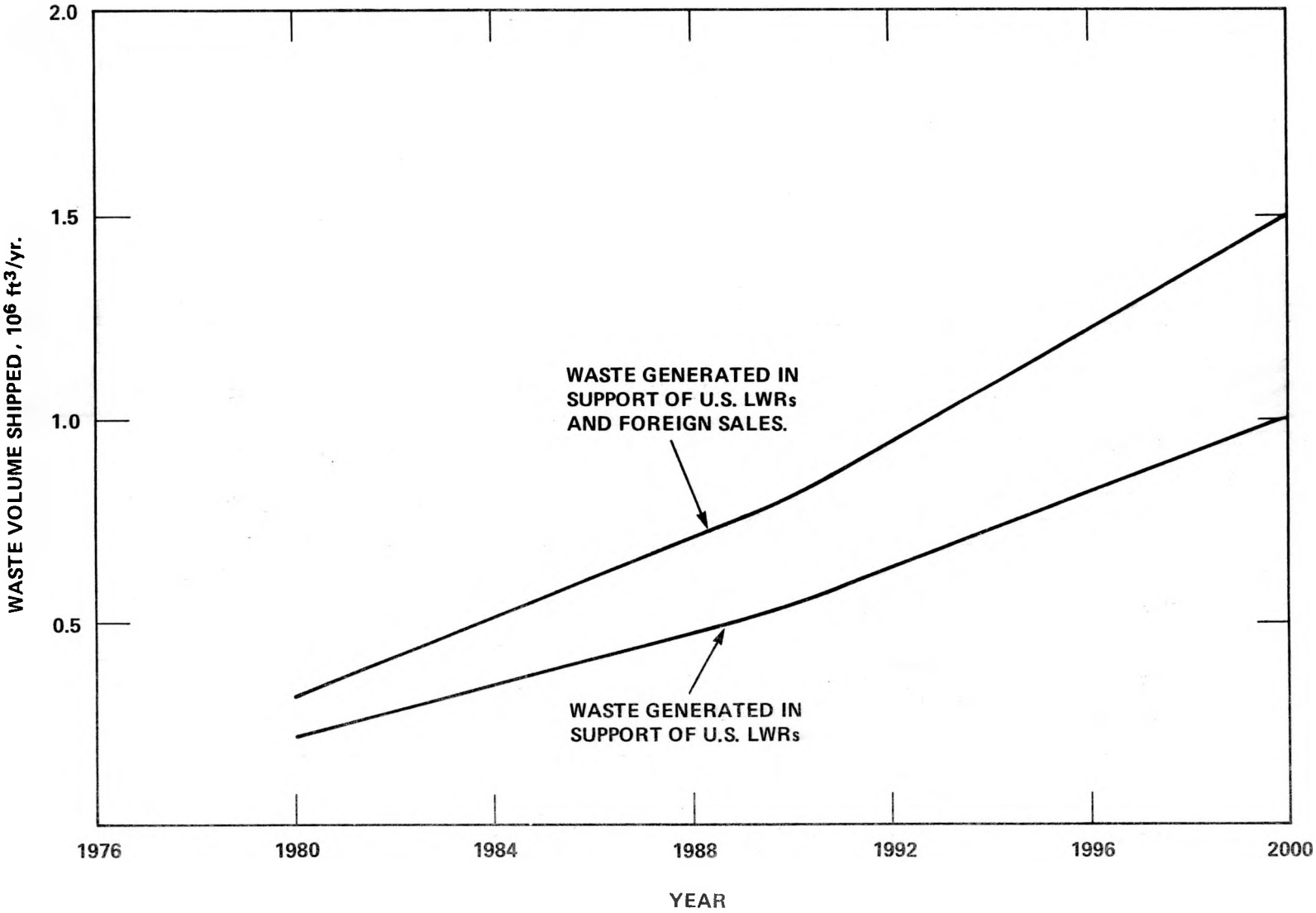

Figure 5.7-1 Annual Waste Volume from Fuel-Fabrication Facilities.

and it may be available for radioactive waste burial. Of the 100 acres now being used, 95 acres are still available.

5.8.5 Beatty

Beatty is an 80-acre site in Nevada with 46 acres available for radioactive waste burial. The remaining acreage consists of 34 acres for chemical waste disposal and a buffer zone between the two areas. Of the 46 acres set aside for radioactive waste burial, 28 acres are still available.

Expansion of the site may not be possible. The surrounding land is controlled by the U.S. Bureau of Land Management (BLM). The BLM is proposing regulations that would prohibit leasing federal land for purposes that require perpetual care. If expansion is permitted there are approximately 400 acres that could be purchased and added to the site.

5.8.6 West Valley

The West Valley site has been closed since 1973. There are no plans to reopen the site at any time in the future.

5.8.7 Summary

Of the six burial sites discussed (See Table 5.8-1), only three are still accepting waste for burial: Barnwell (NC), Beatty (NV), and Richland (WA). These three sites have a combined, total, licensed burial area of 283 acres, and they might be expanded to 1,476 acres. Present waste generation rates for LWRs are shown in Figure 5.3-1, and for fuel-fabrication facilities in Figure 5.7-1. Adding these generation rates and multiplying the total by two gives the total quantity of radioactive waste shipped to shallow land burial sites each year. The factor of two is used since it is estimated that only half of the waste buried at these sites originates at LWRs and fuel-fabrication facilities. Based on an average burial site capacity of 325,000 ft^3 per acre it is estimated that the currently licensed burial land will be exhausted in 1990 with an additional 600 acres needed through 2000.

If volume-reduction practices are initiated as projected in Section 5.5, then the total LWR-waste production follows the curve shown in Figure 5.5-1. With no change in the waste produced by fuel-fabrication facilities, the currently licensed burial land will be exhausted in 1992. An additional 366 acres are then needed for burial through 2000.

These estimates are shown graphically as the top and middle curve, respectively, in Figure 5.8-1. The bottom curve represents the burial site capacity needed based on volume-reduction techniques being instituted at non-LWR fuel-cycle facilities such that the

Table 5.8-1 Commercial Burial Site Disposal Capacity

Site	Licensed Area Left	Possible Additional Acreage
Maxey Flats, KY	0	Sixty-four acres if reopened following studies (in 2 years), and 23 acres if proposed backfill plan is approved.
Sheffield, IL	0	Nine acres if back-fill plan is approved by NRC; 80 acres if NRC adopts site expansion.
Barnwell, SC	160	Undetermined.
Beatty, NV	28	Approximately 400 acres if additional land can be purchased.
Richland, WA	95	Nine hundred acres leased by the State could be leased by NECO.
Total	283	1,476

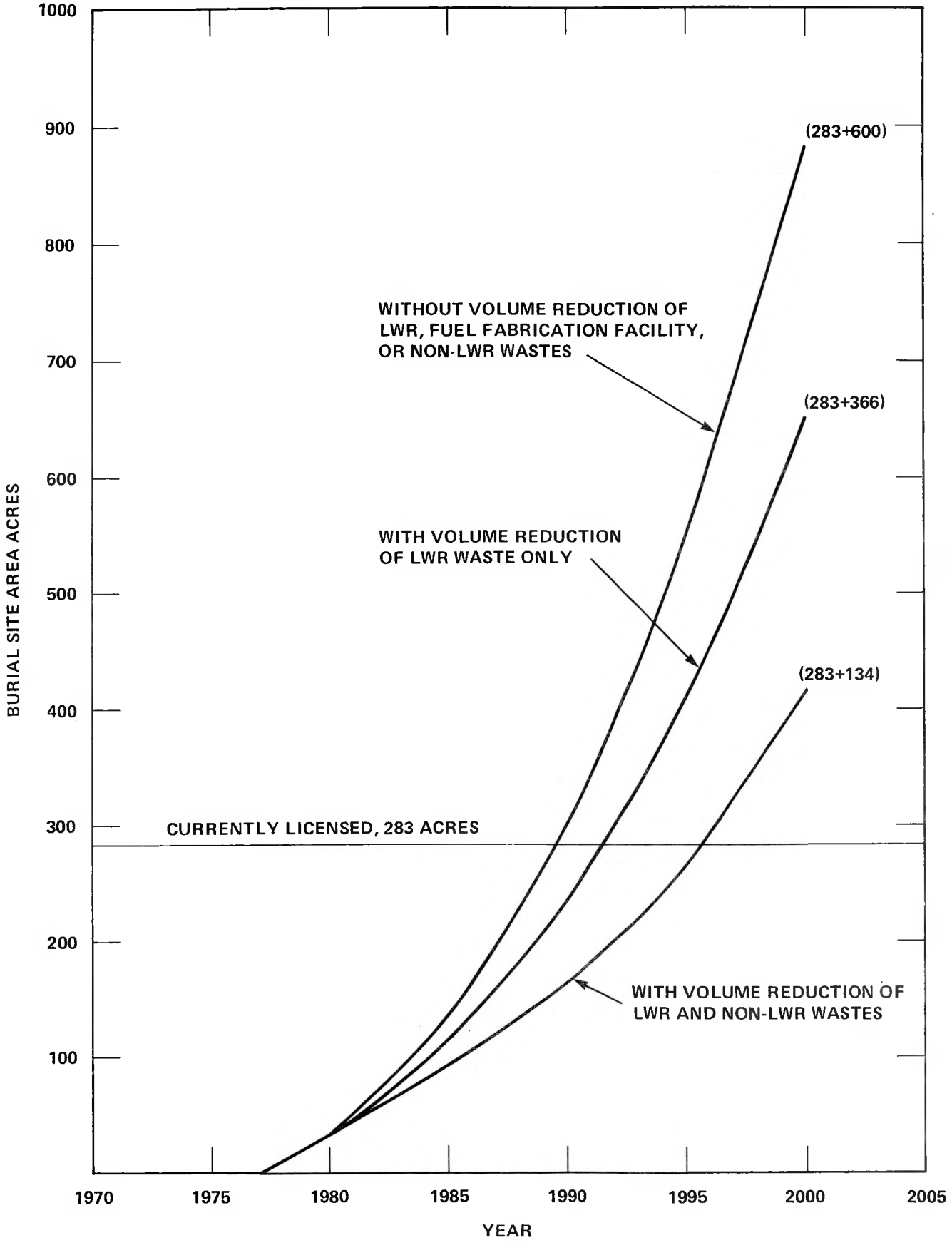

Figure 5.8-1 Burial Site Area Needed After 1977.

ratio between LWR-related wastes and non-LWR fuel-cycle-related wastes remains unchanged at 2 through 2000. Under these conditions the currently licensed burial land will last until 1996. Only an additional 134 acres would be needed through 2000.

Other factors that could affect these projections are

- Regional incineration facilities for waste from all sources, with minimal volume reduction of noncombustible wastes
- Retention of wastes at the source facility through onsite storage (temporary) or onsite burial (permanent)
- Acceptance of a definition for an innocuous level of radioactive contamination such that wastes below this level of contamination can be disposed of in sanitary landfills
- Unexpected large volumes of waste from decommissioning activities.

CHAPTER 5 REFERENCE

Blomeke, J. O., C. W. Alexander, C. W. Kee, and A. G. Croff, 1977. "Projections of Spent Fuel To Be Discharged by the U.S. Nuclear Power Industry," ORNL-TM-6008, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Appendix A

Glossary

Absorption	Incorporation of a substance within the physical, chemical, or molecular structure of another substance.
Actinides	Radioactive isotopes of elements with atomic numbers 89 to 103.
Adsorption	Adhesion of ions or molecules to the surface of liquid or solid bodies with which they come in contact. Adhering to a surface.
Alpha-contaminated wastes	Wastes containing alpha-emitting radionuclides, usually actinides.
Bitumen	A petroleum produce, specifically steep roofing asphalt, used by Warner & Pfliederer Corp. in conjunction with an extruder evaporator for the incorporation of radioactive waste in a solid matrix. The process is generally referred to as bituminization.
Cartridge filter	A filter that uses a replaceable element (usually made of cotton) for the removal of crud and dirt from the fluid processed.
Combustible trash	Articles of compactible and noncompactible trash that can be burned in an incinerator.
Compactible waste	Wastes with a level of radioactive contamination low enough to be handled manually without health risk and which can be effectively reduced in volume by mechanical means, for instance, rags, light bulbs, paper, and sample bottles.
Concentrator bottoms	The reduced liquid volume resulting from the evaporation of water in an evaporator or concentrator. This volume contains almost all of the dirt, chemicals and radioactivity originally in the feed stream.
Decontamination factor	Typically the factor by which a measured parameter, such as gross radionuclide activity, is reduced as the result of a given process (i.e., the inlet concentration divided by the outlet concentration).

Deep bed demineralizer	A vessel, filled with ion-exchange resin used for water purification, in which the depth of the resin is usualy 3 feet or more.
Filter sludge	The material discharged from a precoat filter at the end of filter life. The material consists of the original filter medium and any crud or dirt removed from the fluid processed.
Intermediate level waste	Wastes that require shielding and adequate confinement, but not heat dissipation, during their normal handling and transportation because of their specific activity and radionuclide content.
Ion exchange	The property of certain solids to adsorb ions from solution and at the same time release a different ion. Also called demineralization.
Ion-exchange resin	Any material that exhibits ion-exchange properties; also referred to as demineralizer resin.
Isotope	Any species of an element belonging to a set of atoms that have the same atomic number but a different atomic mass.
Light-water reactor	Any nuclear power fission reactor using slightly enriched uranium fuel and light water for cooling and neutron moderation.
Low-level waste	Those wastes that do not require shielding during normal handling and transportation because of their low radionuclie content.
Noncompactible waste	Wastes with a level of contamination low enough to be handled manually without health risk but which cannot be mechanically reduced in volume. Examples: tools, miscellaneous metal, poison channels, and fuel channels.
Precoat filter	Filters that are coated with a filtering medium prior to use. Common filter media are Solka Floc, Diatomaceous Earth, and Powdex.
Radionuclide	Any of the various species of a given element having the same atomic number, but different atomic weights.

Transuranics	Those elements with atomic numbers greater than that of uranium (92).
Trash	General term covering both compactible and noncompactible waste.

Appendix B

Acronyms and Abbreviations

Acronyms

ASTM	American Society for Testing Materials
CFR	Code of Federal Regulations
CPS	Condensate Polishing System
CRW	Clean Radwaste System
DIS	Dow Industrial Service
DOE	Department of Energy
DOT	Department of Transportation
DRW	Dirty Radwaste System
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
HEPA	High Efficiency Particulate Air
HNDC	Hittman Nuclear & Development Company
IAEA	International Atomic Energy Agency
LLS	Low Level Solids
LRMS	Liquid Radwaste Management System
LSA	Low Specific Activity
LWR	Light Water Reactor
NECO	Nuclear Engineering Company, Inc.
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PPI	Protective Packaging Incorporated
PVC	Polyvinylchloride
UNI	United Nuclear Incorporated

Abbreviations

mr	millirem
MTU	Metric tons of uranium
R	rem
rem	roentgen equivalent in man
SG	Steam generator
Wt%	Weight percent

Appendix C

LWR Systems and Components

Table C-1 CURRENT BOILING WATER REACTOR SYSTEMS AND COMPONENTS

LEGEND

NR - NONREGENERATIVE
R - REGENERATIVE
F - PLANNED FOR FUTURE INSTALLATION

SYSTEMS GENERATING WASTE

PLANT NAME	YEAR OF COMMERCIAL OPERATION	NET MWe OUTPUT	REACTOR CLEANUP SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	SPENT FUEL POOL CLEANUP SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	DISK FILTER
OPERATING PLANTS													
DRESDEN–1	1960	210				●			●				
BIG ROCK POINT	1962	75				●			●				
HUMBOLDT BAY	1963	65				●		●			●		
NINE MILE POINT–1	1969	620		●	●		●	●					
OYSTER CREEK	1969	640			●	●				●			
GENOA–2	1970	50	●			●		●			●		
MILLSTONE–1	1970	652		●			●		●			F	F
MONTICELLO	1971	545			●					●			
DRESDEN–2 & 3	71–72	1618				●			●			●	
QUAD CITIES–1 & 2	1973	1680	●		●					●			
PILGRIM–1	1972	655			●			●			●		
VERMONT YANKEE	1973	540			●			●		●			
COOPER	1974	778		●	●				●	●			
BROWNS FERRY–1, 2 & 3	74–77	2200		●	●				●	●			
ARNOLD	1975	550			●					●			
HATCH–1	1975	813		●	●				●	●			
BRUNSWICK–1 & 2*	75–77	821			●					●			

PLANT NAME	CONDENSATE POLISHING SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	ULTRASONIC CLEANER
DRESDEN–1				●		●
BIG ROCK POINT				●		
HUMBOLDT BAY					●	
NINE MILE POINT–1					●	●
OYSTER CREEK					●	
GENOA–2				●		
MILLSTONE–1					●	
MONTICELLO			●			
DRESDEN–2 & 3					●	
QUAD CITIES–1 & 2			●			
PILGRIM–1					●	●
VERMONT YANKEE			●			
COOPER		●	●			
BROWNS FERRY–1, 2 & 3	●	●	●			
ARNOLD			●			
HATCH–1			●			
BRUNSWICK–1 & 2*			●		●	

WASTE PROCESSING SYSTEMS

PLANT NAME	EQUIPMENT DRAINS WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	EVAPORATOR	FLOOR DRAINS WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	EVAPORATOR
DRESDEN–1		●		●							
BIG ROCK POINT	●			●			●			●	
HUMBOLDT BAY	●					●					
NINE MILE POINT–1					●			●	●		●
OYSTER CREEK		F	●		●			F	●		●
GENOA–2	F			●							
MILLSTONE–1		●			●			●			●
MONTICELLO			●	●							
DRESDEN–2 & 3		●			●						●
QUAD CITIES–1 & 2		●		●				●			
PILGRIM–1		●		●				●			●
VERMONT YANKEE		●	●	●				●	●		
COOPER		●	●	●				●		●	F
BROWNS FERRY–1, 2 & 3		●	●	●				●			F
ARNOLD			●	●					●	●	
HATCH–1		●	●	●					●		
BRUNSWICK–1 & 2*			●						●		●

PLANT NAME	CHEMICAL WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	ULTRA FILTRATION	DEEP BED DEMIN. (NR)	EVAPORATOR	LAUNDRY WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	EVAPORATOR	REVERSE OSMOSIS
DRESDEN–1							●				F
BIG ROCK POINT								●			
HUMBOLDT BAY											
NINE MILE POINT–1											
OYSTER CREEK											
GENOA–2									●		
MILLSTONE–1		●			●		●				
MONTICELLO							●				
DRESDEN–2 & 3					●						
QUAD CITIES–1 & 2	●						●				
PILGRIM–1					●						
VERMONT YANKEE											
COOPER							●				
BROWNS FERRY–1, 2 & 3							●			F	
ARNOLD			●	●							
HATCH–1					●		●				
BRUNSWICK–1 & 2*											

PLANT NAME	SOLID WASTE SYSTEM: TRASH COMPACTOR	CENTRIFUGE	PHASE SEPARATOR	DEWATERING FILTER/TANK	PARTIAL VOLUME RED.	FULL VOLUME RED.	SOLIDIFICATION EQUIP.	OFFGAS HOLDUP SYSTEM: DELAY PIPE/TANK	RECOMBINER	CHARCOAL ADSORPTION	COMPRESSED STORAGE	CRYOGENIC DISTILLATION
DRESDEN–1		●		●				●		F		
BIG ROCK POINT				●				●				
HUMBOLDT BAY							F	●	F	F		
NINE MILE POINT–1	●	●					●	●	F	F		
OYSTER CREEK	●	●					F	●	F	F		
GENOA–2	●			●				●	●	●	●	
MILLSTONE–1	●	●		●			F	●	F	F		
MONTICELLO	●	●	●				●	●	●		●	
DRESDEN–2 & 3	●	●	●					●	●	●		
QUAD CITIES–1 & 2	●	●	●				●	●	●	●		
PILGRIM–1	●			●				●	●	●		
VERMONT YANKEE	●	●						●	●	●		
COOPER	●	●					●	●	F	F		
BROWNS FERRY–1, 2 & 3	●			●			F	●	F	F		
ARNOLD	●	●						●	●	●		
HATCH–1	●	●		●				●	●	●		
BRUNSWICK–1 & 2*	●	●					F	●	F	●		

Table C-2 FUTURE BOILING WATER REACTOR SYSTEMS AND COMPONENTS

LEGEND

NR - NONREGENERATIVE
R - REGENERATIVE
F - PLANNED FOR FUTURE INSTALLATION

PLANT NAME	YEAR OF COMMERCIAL OPERATION	NET MWe OUTPUT
HATCH 2	1978	795
ZIMMER 1	1979	810
LA SALLE COUNTY 1 & 2	1979, 80	2,156
FERMI 2	1980	1,150
WPPSS 2	1980	1,103
SHOREHAM	1980	819
SUSQUEHANNA 1 & 2	1980, 82	2,100
PERRY 1 & 2	1981, 83	2,410
GRAND GULF 1 & 2	1981, 84	2,500
CLINTON 1 & 2	1981, 88	1,910
NINE MILE POINT 2	1982	1,100
BAILLY	1982	660
HARTVILLE 1, 3 & 2, 4	1983, 84	4,932
BLACK FOX 1 & 2	1983, 85	2,300
LIMERICK 1 & 2	1983, 85	2,110
RIVER BEND 1 & 2	1983, 85	1,880
PHIPPS BEND 1 & 2	1984, 85	2,466
HOPE CREEK 1 & 2	1984, 86	2,134
SKAGIT 1 & 2	1984, 86	2,576
ALLENS CREEK	1985	1,200
MONTAGUE 1 & 2	1986, 88	2,300

SYSTEMS GENERATING WASTE

PLANT NAME	REACTOR CLEANUP SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	SPENT FUEL POOL CLEANUP SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	DISK FILTER	CONDENSATE POLISHING SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	ULTRASONIC CLEANER
HATCH 2			●					●						●			
ZIMMER 1			●				●	●					●				
LA SALLE COUNTY 1 & 2			●					●								●	
FERMI 2			●					●						●			
WPPSS 2			●					●						●			
SHOREHAM			●			●			●							●	●
SUSQUEHANNA 1 & 2			●					●						●			
PERRY 1 & 2			●					●					●			●	
GRAND GULF 1 & 2			●					●								●	
CLINTON 1 & 2			●					●								●	●
NINE MILE POINT 2			●				●		●							●	●
BAILLY			●					●						●			
HARTVILLE 1, 3 & 2, 4			●					●								●	●
BLACK FOX 1 & 2			●					●								●	●
LIMERICK 1 & 2			●					●						●			
RIVER BEND 1 & 2			●				●		●							●	●
PHIPPS BEND 1 & 2			●					●								●	●
HOPE CREEK 1 & 2			●					●								●	
SKAGIT 1 & 2			●					●						●			
ALLENS CREEK			●					●								●	
MONTAGUE 1 & 2			●				●		●							●	

WASTE PROCESSING SYSTEMS

PLANT NAME	EQUIPMENT DRAINS WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	EVAPORATOR	FLOOR DRAINS WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	EVAPORATOR	CHEMICAL WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	ULTRA FILTRATION	DEEP BED DEMIN. (NR)	EVAPORATOR
HATCH 2		●		●				●		●			●				●
ZIMMER 1		●		●	●			●								●	●
LA SALLE COUNTY 1 & 2		●		●				●		●	●					●	●
FERMI 2		●		●				●	●				●				
WPPSS 2			●					●									●
SHOREHAM		●		●				●		●							●
SUSQUEHANNA 1 & 2		●		●				●		●							●
PERRY 1 & 2		●		●				●		●	●						●
GRAND GULF 1 & 2		●			●			●		●	●						●
CLINTON 1 & 2		●			●			●			●						●
NINE MILE POINT 2		●			●			●			●						●
BAILLY			●					●									●
HARTVILLE 1, 3 & 2, 4		●		●						●	●						●
BLACK FOX 1 & 2		●		●						●	●						●
LIMERICK 1 & 2			●	●					●	●							●
RIVER BEND 1 & 2		●		●				●									●
PHIPPS BEND 1 & 2		●		●						●	●					●	●
HOPE CREEK 1 & 2		●		●				●									●
SKAGIT 1 & 2		●		●													●
ALLENS CREEK		●			●			● (1)								●	●
MONTAGUE 1 & 2		●		●	●			●			●						●

PLANT NAME	LAUNDRY WASTE SYSTEM: CARTRIDGE FILTER	PRECOAT FILTER	FILTER DEMIN.	EVAPORATOR	REVERSE OSMOSIS	SOLID WASTE SYSTEM	CENTRIFUGE	PHASE SEPARATOR	DEWATERING FILTER/TANK	PARTIAL VOLUME RED.	FULL VOLUME RED.	SOLIDIFICATION EQUIP.	OFFGAS HOLDUP SYSTEM: DELAY PIPE/TANK	RECOMBINER	CHARCOAL ADSORPTION	COMPRESSED STORAGE	CRYOGENIC DISTILLATION
HATCH 2							●	●					●	●	●		
ZIMMER 1		●		●		●		●					●	●	●		
LA SALLE COUNTY 1 & 2	●			●		●	●	●				●	●	●	●		
FERMI 2		●				●	●	●				●	●		●		
WPPSS 2		●				●	●	●	●			●	●	●	●		
SHOREHAM	●					●		●	●			●		●	●		
SUSQUEHANNA 1 & 2	●												●	●	●		
PERRY 1 & 2	●					●		●				●		●	●		
GRAND GULF 1 & 2						●		●	●			●	●	●	●		
CLINTON 1 & 2	●					●		●	●			●	●	●	●		
NINE MILE POINT 2						●		●	●	●			●	●	●		
BAILLY		●		●		●		●	●			●	●	●	●		
HARTVILLE 1, 3 & 2, 4	●			●		●		●	●			●	●	●	●		
BLACK FOX 1 & 2	●			●		●		●	●			●		●	●		
LIMERICK 1 & 2	●					●	●	●				●	●	●			
RIVER BEND 1 & 2						●		●	●			●		●	●		
PHIPPS BEND 1 & 2	●			●		●		●	●			●	●	●	●		
HOPE CREEK 1 & 2	●					●		●	●			●	●	●	●		
SKAGIT 1 & 2						●			●			●	●	●	●		
ALLENS CREEK	●				●	●	●	●	●			●	●	●	●		
MONTAGUE 1 & 2						●						●		●	●		

1. FLOOR DRAINS AND CHEM COMBINED.

Table C-3 CURRENT PRESSURIZED WATER REACTOR SYSTEMS AND COMPONENTS

LEGEND

NR - NON REGENERATIVE
R - REGENERATIVE
F - PLANNED FOR FUTURE INSTALLATION

SYSTEMS GENERATING WASTE

PLANT NAME	YEAR OF COMMERCIAL OPERATION	NET MWE OUTPUT	CHEMICAL VOLUME CONTROL SYSTEM (CVCS)				BORON RECOVERY SYSTEM						SPENT FUEL POOL CLEAN-UP SYSTEM			CONDENSATE POLISHING SYSTEM				STEAM GENERATOR BLOWDOWN				
			CARTRIDGE FILTER	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (CATION)	DEEP BED DEMIN. (ANION)	CARTRIDGE FILTER	FILTER-DEMIN.	DEBORATING RESIN BED	CESIUM BED DEMIN.	MIXED BED DEMIN.	EVAPORATOR	CARTRIDGE FILTER	FILTER DEMIN.	DEEP BED DEMIN. (NR)	CARTRIDGE FILTER	PRECOAT FILTER	FILTER-DEMIN.	DEEP BED DEMIN. (R)	CARTRIDGE FILTER	FILTER-DEMIN.	DEEP BED DEMIN. (NR)	DEEP BED DEMIN. (R)	EVAPORATOR
OPERATING PLANTS																								
YANKEE ROWE	1960	185	●	●									●		●									
INDIAN POINT–1	1962	285	●	●	●	●							●		●							●		F
SAN ONOFRE–1	1968	451	●	●									●		●									
CONNECTICUT YANKEE	1968	575	●	●	●		●				●	●	●		●									
GINNA	1970	500	●	●	●		●				●	●	●		●				●				F	
ROBINSON–2	1971	735	●	●	●				●		●	●	●		●									
PALISADES	1971	821	●	●											●			F						
SURRY–1 & 2	1972	1644	●	●	●		●			●		●	●		●	F			F				F	
MAIN YANKEE	1972	790	●	●			●		●			●	●		●									
TURKEY POINT–3 & 4	72-73	1332	●	●	●	●	●		●	●	●	●	●		●									
INDIAN POINT–2 & 3	73-76	900	●	●	●		●		●	●	●	●	●		●							●		F
FORT CALHOUN	1973	457	●	●	●	●							●		●									
OCONEE–1, 2 & 3	73-74	2667	●	●		●					●		●		●			●						
ZION–1 & 2	73-74	2170	●	●	●		●	●	●			●	●	●						●	●	●		
THREE MILE ISLAND–1	1974	800	●	●				●	●			●			●			●						
CALVERT CLIFFS–1 & 2	75-77	845	●	●			●		●		●	●	●		●		●		●		F			
RANCHO SECO	1975	964	●	●			●		●		●	●	●		●				●					

WASTE PROCESSING SYSTEMS

PLANT NAME	LIQUID WASTE SYSTEM				CHEMICAL WASTE SYSTEM			LAUNDRY WASTE SYSTEM				SOLID WASTE SYSTEM					GASEOUS WASTE SYSTEM				
	CARTRIDGE FILTER	FILTER-DEMIN.	DEEP BED DEMIN. (NR)	EVAPORATOR	CARTRIDGE FILTER	DEEP BED DEMIN. (NR)	EVAPORATOR	PRECOAT FILTER	CARTRIDGE FILTER	REVERSE OSMOSIS	EVAPORATOR	RESIN DEWATERING TANK	SOLIDIFICATION SYSTEM	TRASH COMPACTOR	PARTIAL VOLUME REDUCTION	FULL VOLUME REDUCTION	DELAY PIPE/TANK	RECOMBINER	COMPRESSED STORAGE	CHARCOAL ADSORPTION	CRYOGENIC DISTILLATION
OPERATING PLANTS																					
YANKEE ROWE	●			●									●						●		
INDIAN POINT–1	●		●	●									●	●			●	●			
SAN ONOFRE–1	●		●																●		●
CONNECTICUT YANKEE	●		●	●								●	●	●					●		
GINNA	●		●	●			●		●	●			●	●					●		
ROBINSON–2	●		●	●									●	●					●		
PALISADES	●		●	●			●		●				●	●					●		
SURRY–1 & 2	●			●	●		●					●	●	●				●	●		
MAIN YANKEE				●									F	●					●		
TURKEY POINT–3 & 4	●		●	●								●	F	●					●		
INDIAN POINT–2 & 3	●		●	●									●	●					●		
FORT CALHOUN	●		●	●								●	●	●					●		
OCONEE–1, 2 & 3			●	●					●			●		●			●		●	●	
ZION–1 & 2	●		●	●													●	●		●	
THREE MILE ISLAND–1		●	●	●				●			●		●						●		
CALVERT CLIFFS–1 & 2	●		●								●								●		
RANCHO SECO	●		●	●	●	●	●							●					●		

Table C-4 FUTURE PRESSURIZED WATER REACTOR SYSTEMS AND COMPONENTS

LEGEND

NR - NONREGENERATIVE
R - REGENERATIVE
F - PLANNED FOR FUTURE INSTALLATION

PLANT NAME	YEAR OF COMMERCIAL OPERATION	NET MWe OUTPUT	SYSTEMS GENERATING WASTE: CHEMICAL VOLUME CONTROL SYSTEM (CVCS) — CARTRIDGE FILTER	CVCS — DEEP BED DEMIN. (NR)	CVCS — DEEP BED DEMIN. (CATION)	CVCS — DEEP BED DEMIN. (ANION)	BORON RECOVERY SYSTEM — CARTRIDGE FILTER	BORON RECOVERY SYSTEM — FILTER-DEMIN.	BORON RECOVERY SYSTEM — DEBORATING RESIN BED	BORON RECOVERY SYSTEM — CESIUM BED DEMIN.	BORON RECOVERY SYSTEM — MIXED BED DEMIN.	BORON RECOVERY SYSTEM — EVAPORATOR	SPENT FUEL POOL CLEAN-UP SYSTEM — CARTRIDGE FILTER	SPENT FUEL POOL CLEAN-UP SYSTEM — FILTER DEMIN.	SPENT FUEL POOL CLEAN-UP SYSTEM — DEEP BED DEMIN. (NR)	CONDENSATE POLISHING SYSTEM — CARTRIDGE FILTER	CONDENSATE POLISHING SYSTEM — PRECOAT FILTER	CONDENSATE POLISHING SYSTEM — FILTER-DEMIN.	CONDENSATE POLISHING SYSTEM — DEEP BED DEMIN. (R)	STEAM GENERATOR BLOWDOWN — CARTRIDGE FILTER	STEAM GENERATOR BLOWDOWN — FILTER-DEMIN.	STEAM GENERATOR BLOWDOWN — DEEP BED DEMIN. (NR)	STEAM GENERATOR BLOWDOWN — EVAPORATOR	STEAM GENERATOR BLOWDOWN — DEEP BED DEMIN. (R)	WASTE PROCESSING SYSTEMS: LIQUID WASTE SYSTEM — CARTRIDGE FILTER	LIQUID WASTE SYSTEM — DEEP BED DEMIN. (NR)	LIQUID WASTE SYSTEM — EVAPORATOR	CHEMICAL WASTE SYSTEM — CARTRIDGE FILTER	CHEMICAL WASTE SYSTEM — DEEP BED DEMIN. (NR)	CHEMICAL WASTE SYSTEM — EVAPORATOR	LAUNDRY WASTE SYSTEM — PRECOAT FILTER	LAUNDRY WASTE SYSTEM — CARTRIDGE FILTER	LAUNDRY WASTE SYSTEM — REVERSE OSMOSIS	LAUNDRY WASTE SYSTEM — EVAPORATOR	LAUNDRY WASTE SYSTEM — RESIN DEWATERING TANK	SOLID WASTE SYSTEM — SOLIDIFICATION SYSTEM	SOLID WASTE SYSTEM — TRASH COMPACTOR	SOLID WASTE SYSTEM — PARTIAL VOLUME REDUCTION	SOLID WASTE SYSTEM — FULL VOLUME REDUCTION	SOLID WASTE SYSTEM — DELAY PIPE/TANK	SOLID WASTE SYSTEM — RECOMBINER	GASEOUS WASTE SYSTEM — COMPRESSED STORAGE	GASEOUS WASTE SYSTEM — CHARCOAL ADSORPTION	GASEOUS WASTE SYSTEM — CRYOGENIC DISTILLATION
ARKANSAS NUCLEAR ONE - 2	1978	912	●	●			●		●		●	●	●	●		●						●				●				●											
COOK 2[1]	1978	1,060	●		●	●	●		●	●	●	●	●	●	●		●				●																			
SEQUOYAH 1 & 2	1978, 79	2,296		●									●	●	●		●			●			●	●									(2)	●		●				
THREE MILE ISLAND 2	1978	880	●	●					●				●	●			●				●												(3)							
McGUIRE 1 & 2	1979, 81	2,360		●	●						●		●	●			●			●				●																
NORTH ANNA 2	1979	934									●		●	●					DD[4]					●																
SALEM 2	1979	1,115	●	●	●		●			●			●	●					DD					●																
FARLEY 2	1980	860	●	●	●		●		●		●		●	●							●			●																
BELLEFONTE 1 & 2	1980, 81	2,426	●	●			●		●		●		●	●						●	●			●																
SOUTH TEXAS 1 & 2	1980, 82	2,500	●	●			●	●			●		●	●			●					●		●	●	●														
COMANCHE PEAK 1 & 2	1981, 83	2,300			●	●		●	●		●		●	●						●		●		●	●	●														
BRAIDWOOD 1 & 2	1981, 82	2,240	●	●	●		●		●		●		●	●						●		●		●	●	●														
BYRON 1 & 2	1981, 82	2,240	●	●	●		●		●		●		●	●						●		●		●	●	●														
CATAWBA 1 & 2	1981, 83	2,306						●			●		●	●								●		●	●	●														
BEAVER VALLEY 2	1982	856	●	●	●			●	●		●		●	●						●		●		●	●															(5)
FORKED RIVER	1982	1,120	●					●			●		●	●			●			●		●		●																
MILLSTONE 3	1982	1,150			●		●		●	●	●		●	●								●		●																
SEABROOK 1 & 2	1982, 84	2,300			●	●	●	●	●		●		●	●	●	●						●		●	●															
NORTH ANNA 3 & 4	1983, 83	1,876	●				●		●	●	●		●	●								●		●	●															
ST. LUCIE	1983	810			●	●	●		●		●		●	●						●		●		●																
PILGRIM 2	1984	1,180	●	●			●		●		●		●	●								●		●	●															
CHEROKEE 1, 2 & 3	1984, 86, 89	3,840	●	●			●		●		●		●	●			●					●		●																
VOGTLE 1 & 2	1984, 85	2,226			●	●	●		●		●		●	●								●		●																
JAMESPORT 1 & 2	1984, 86	2,300	●	●	●		●		●		●		●	●								●		●	●															
TYRONE ENERGY PARK 1	1984	1,100	●		●		●		●		●		●	●								●		●																
GREEN COUNTY	1984	1,212	●	●		●	●		●		●		●	●								●		●																
DAVIS BESSE 2 & 3	1985, 87	1,812			●	●							●	●		●	●					●		●									(6)							
YELLOW CREEK 1 & 2	1985, 86	2,570	●		●	●	●		●		●		●	●	●		●					●		●																
ATLANTIC 1 & 2	1985, 87	2,300			●	●	●		●		●		●	●								●		●																
GREENWOOD 2 & 3	1987, 89	2,360	●	●			●		●		●		●	●			●					●		●																

1. DIRECT DRUMMING FROM CONDENSATE EVAPORATION.
2. DIRECT DRUMMING, NO SOLIDIFICATION EVAPORATOR IN RW EU SYSTEM.
3. IN UNIT ONE.
4. DIRECT DISCHARGE.
5. GASEOUS WASTE THROUGH UNIT 1.
6. LAUNDRY POLISHING DEMINERALIZER.

Appendix D

Source of Information

Appendix D

Source of Information

All of the data reported in Chapter 3 of this report were obtained in response to survey questionnaires developed by NUS Corporation in cooperation with the Office of Waste Isolation, Oak Ridge National Laboratory. NUS contacted 22 PWR and 14 BWR power plants and visited 18 PWR and 12 BWR plants to assist plant personnel in filling out the survey form. A separate survey form was developed for use with site visits to fuel-fabrication facilities. A total of seven facilities involved with the conversion of uranium hexafluoride to finished fuel elements were contacted and visited. Facilities that participated in these surveys are listed below.

NUCLEAR POWER PLANTS

Arkansas Nuclear One, Unit 1
Arkansas Power and Light
Russellville, Arkansas

Duane Arnold Nuclear Energy Center, Unit 1
Iowa Electric Light & Power Co.
Cedar Rapids, Iowa

Brunswick Steam Electric Plant, Unit 2
Carolina Power and Light Co.
Southport, North Carolina

Calvert Cliffs Nuclear Power Plant, Unit 1
Baltimore Gas & Electric Co.
Lusby, Maryland

Donald C. Cook Nuclear Station, Unit 1
Indiana and Michigan Power Co.
Bridgman, Michigan

Cooper Nuclear Station
Nebraska Public Power District
Brownville, Nebraska

Dresden Nuclear Power Station, Units 2 and 3
Commonwealth Edison Co.
Morris, Illinois

James A. FitzPatrick, Unit 2
Power Authority of the State of New York
Lycoming, New York

R. E. Ginna Nuclear Power Plant
Rochester Gas and Electric Corp.
Ontario, New York

Haddam Neck Atomic Plant
Connecticut Yankee Atomic Power Co.
Haddam Neck, Connecticut

Edwin I. Hatch, Unit 1
Georgia Power Co.
Baxley, Georgia

Indian Point Nuclear Power Plant, Unit 2
Consolidated Edison Co. of New York, Inc.
Buchanan, New York

Kewaunee Plant
Wisconsin Public Service Corp.
Kewaunee, Wisconsin

Maine Yankee Atomic Plant
Maine Yankee Atomic Power Co.
Wiscasset, Maine

Millstone Point Nuclear Power Station, Units 1 and 2
Northeast Utilities
Waterford, Connecticut

Monticello Nuclear Generating Plant
Northern States Power Co.
Monticello, Minnesota

Nine Mile Point, Unit 1
Niagara Mohawk Power Corp.
Lycoming, New York

Oconee Nuclear Plant, Units 1, 2, and 3
Duke Power Co.
Seneca, South Carolina

Palisades Nuclear Plant
Consumers Power Co.
Covert, Michigan

Pilgrim Nuclear Power Station, Unit 1
Boston Edison Co.
Plymouth, Massachusetts

Prairie Island Nuclear Plant, Units 1 and 2
Northern States Power Co.
Welch, Minnesota

Quad Cities Nuclear Power Station, Units 1 and 2
Commonwealth Edison Co.
Cordova, Illinois

H. B. Robinson Plant, Unit 2
Carolina Power & Light Co.
Hartsville, South Carolina

San Onofre Nuclear Generating Station, Unit 1
Southern California Edison Co.
San Clemente, California

Three Mile Island Nuclear Plant, Unit 1
Metropolitan Edison Co.
Middletown, Pennsylvania

Trojan Nuclear Power Plant
Portland General Electric Co.
Rainier, Oregon

Turkey Point Plant, Units 3 and 4
Florida Power & Light Co.
Miami, Florida

Vermont Yankee Nuclear Power Station
Vermont Yankee Nuclear Power Corp.
Vernon, Vermont

Zion Generating Station, Units 1 and 2
Commonwealth Edison Co.
Zion, Illinois

FUEL-FABRICATION PLANTS

Apollo Commercial Fuel Plant
Babcock & Wilcox
Apollo, Pennsylvania

C. E. Hematite Plant
Combustion Engineering
Hematite, Missouri

Lynchburg Commercial Fuel Plant
Babcock & Wilcox
Lynchburg, Virginia

Washington Fuel Fabrication Facility
Exxon Nuclear Company
Richland, Washington

Westinghouse Nuclear Fuel Plant
Columbia, South Carolina

Wilmington Fuel Manufacturing Plant
General Electric Company
Wilmington, North Carolina

C. E. Windsor Plant
Combustion Engineering
Windsor, Connecticut

Appendix E

Method of Analysis

Appendix E

Method of Analysis

Two steps were taken before an analysis of LWR waste volume data was performed. First, all of the waste volumes reported for spent resins, filter sludges, concentrated liquids, cartridge filters, and trash were multiplied by the reported packaging efficiency. The packaging efficiency is that fraction of the solidified waste product that is waste. The remaining volume is the solidification agent. For example, if the reported annual waste shipment of concentrated liquids is 1,000 ft^3 at a packaging efficiency of .60 then the analysis performed in this study used an annual waste generation of 600 ft^3. A packaging efficiency of 1.0 is used for wastes that are not solidified. Second, BWRs were divided into plants that use deep bed demineralizers for condensate polishing and those that use precoat filters. PWRs were divided into plants that process secondary site condensate and those that do not.(1)

The analysis proceeded from this point, with four separate cases being considered for spent resin, filter sludge and concentrated liquids. Data on cartridge filters were divided only on the basis of BWR and PWR plants because the amount of available data was limited. Preliminary analysis of the data on compactible and non-compactible trash indicated that it had very little dependence on the type of condensate polishing system. Therefore, these data also were analyzed only on the basis of BWR, PWR plants.

For each type of waste (e.g., filter sludge) and each case (e.g., BWRs with a deep bed resin condensate polishing system), the available annual data (for unsolidified waste) were divided by the maximum dependable electrical generating capacity. This provided normalized annual waste volumes in units of ft^3/MWe-yr.

The data, as ft^3/MWe-yr, were then tabulated by plant, and the number of years of commercial operation. Data from years in which little or no commercial power was generated were sometimes disregarded if they were not consistent with data for other years. For the examples cited above the tabulation of data is given in Table E-1.

Averages for each year of operation were then calculated as shown in Table E-1. In almost every case there was no discernable time-dependent trend in the average waste generation rates; however, it was not the purpose of this study to conduct a trend analysis. This study is concerned with estimating the annual quantity of waste that will be shipped from LWRs and fuel fabrication facilities to radioactive waste burial facilities. With new facilities

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1. There were not enough data to perform a separate analysis of PWRs with deep bed demineralizers versus precoat filters for secondary system condensate polishing.

Table E-1 Precoat Filter Waste Generation Rates in BWRs With a Deep Bed CPS[1] (ft^3/MWe-yr) Number of Years of Operation

Plant	1	2	3	4	5	6	7	8
B1	6.4	11.2	6.6	10.6	10.0			
B2							1.2	
B3						3.0	3.7	0.7
B4		6.6	2.7					
B5								
B6	1.2	4.4	5.1					
Average	3.8	7.4	4.8	10.6	10.0	3.0	2.5	0.7

1. Condensate polishing system.

expected to go on line every year between now and 2000, data from the first few years of operation were given the same weight in the final analysis as data from later years. Over half of the data in Table E-1 were from the first three years of plant operations.

The final value used in the report is the average of the annual averages, referred to in the text as the weighted average. For the example this is 5.4 ft^3/MWe-yr.

Activity generation rates are calculated by multiplying the volume-generation rates by the average-activity concentration, Ci/ft^3. This gives activity generation rates in units of Ci/MWe-yr. For the example the average concentration is 0.37 Ci/ft^3 resulting in 2.0 Ci/MWe-yr.

For some years plant operations resulted in waste volumes that were unusually high in comparison to the average for that plant, or to the computed weighted average for all plants. These data were included in the data base for the calculations detailed above for the average plant. Again, these values are used in calculating the annual volume of waste, from all operating plants, coming into the disposal site.

However, a second consideration is, what are the annual waste volumes and activity generation rates typical of a single plant? To answer this question any abnormally high values identified are excluded from the data base and the weighted-average waste volume, activity concentrations, and waste activities are recalculated. In the example given in this appendix none of the waste volumes were considered to be abnormally high. Therefore, the typical and average waste volume generation rates are identical.

Chapter 4 of this report lists both the typical and average waste volumes, and waste-activity generation rates. Chapter 5, however, is based solely on the average waste quantities.

Appendix F

LWR Survey Forms

The forms presented in this appendix were used during visits to operating U.S. LWRs to collect data for this report. There is a separate cover page for BWRs and PWRs. The subsequent pages presented here were used for both PWRs and BWRs.

A survey form for data from fuel-fabrication facilities was also prepared but could not be used due to the great dissimilarity between the facilities. The data collected from these facilities was tabulated as appropriate for each facility and the purpose of the study.

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4 RESEARCH PLACE
ROCKVILLE, MARYLAND 20850
(301) 948-7010

OWI RADWASTE QUESTIONNAIRE – BWR

STATION _____ **CAPACITY** _____ MWe Date of **COMMERCIAL OPERATION** _____

PERSON(S) TO CONTACT _____ TELEPHONE # AND EXTENSION _____

PART 1 RADWASTE DESIGN INFORMATION

Part 1 of the questionnaire identifies the design and operational characteristics of various process systems for the purpose of identifying each source of solid radioactive waste.

In the blank spaces provided, fill in the number of each type of equipment, and the appropriate letter from the following index which describes the availability and utilization of equipment in the facility

- **X – Installed and utilized**
- **N – Installed but not utilized**
- **F – Planned installation or use in the future**

Leave the space blank if the equipment is not installed and no plans have been made for future installation. If different equipment or methods are used than those specified, use the other space of describe this equipment.

1. SYSTEMS GENERATING OR PROCESSING WASTE

a. REACTOR CLEANUP SYSTEM

Filter-Demins._____ Regenerative Deep Bed Demin._____
Non-Regenerative Deep Bed Demin._____
Precoat_____ Cartridge Filters_____ Edge Type Filters_____
Other_____

b. SPENT FUEL POOL CLEANUP SYSTEM

Filter-Demins._____ Regenerative Deep Bed Demin._____
Non-Regenerative Deep Bed Demin._____
Precoat Filters_____ Cartridge Filters_____ Other_____

c. CONDENSATE POLISHING SYSTEM

Filter-Demins._____ Regenerative Deep Bed Demin._____ URC_____
Non-Regenerative Deep Bed Demin._____
Precoat Filters_____ Cartridge Filters_____ Other_____

d. EQUIPMENT DRAINS (LOW CONDUCTIVITY) RADWASTE SYSTEM

Filter-Demins._____ Regenerative Deep Bed Demin._____
Non-Regenerative Deep Bed Demin._____
Precoat Filters_____ Cartridge Filters_____ Other_____

e. FLOOR DRAINS (HIGH CONDUCTIVITY) RADWASTE SYSTEM

Precoat Filters_____ Cartridge Filters_____ Filter-Demins._____
Ultra Filtration_____ Evaporator_____ Reverse Osmosis_____
Other

f. LAUNDRY LIQUID RADWASTE SYSTEM

Precoat Filters_____ Cartridge Filters_____ Filter-Demins._____
Ultra Filtration_____ Evaporator_____ Reverse Osmosis_____
Other

g. CHEMICAL (NEUTRALIZATION) RADWASTE SYSTEM

Precoat Filters_____ Cartridge Filters_____ Filter-Demins._____
Ultra Filtration_____ Other_____

h. SOLID WASTE SYSTEM

Dewatering Tank_____ Centrifuge_____
Absorbant Material Mixing Equipment_____ Solidification Equipment_____
Trash Compactor_____ Other_____
Phase Separators: Fuel Pool_____ Reactor Cleanup_____ Condensate Cleanup_____

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OWI RADWASTE QUESTIONNAIRE – PWR

STATION _____ CAPACITY _____ MWe Date of COMMERCIAL OPERATION _____

PERSON(S) TO CONTACT _____ TELEPHONE # AND EXTENSION _____

_____ _____

PART 1 RADWASTE DESIGN INFORMATION

Part 1 of the questionnaire identifies the design and operational characteristics of various process systems for the purpose of identifying each source of solid radioactive waste.

In the blank spaces provided, fill in the number of each type of equipment, and the appropriate letter from the following index which describes the availability and utilization of equipment in the facility

- X – Installed and utilized
- N – Installed but not utilized
- F – Planned installation or use in the future

Leave the space blank if the equipment is not installed and no plans have been made for future installation. If different equipment or methods are used than those specified, use the other space of describe this equipment.

1. SYSTEMS GENERATING OR PROCESSING WASTE

a. REACTOR COOLANT CLEANUP SYSTEM (CVCS)

Mixed Deep Bed Demineralizers _____ Cartridge Filters _____ Edge Type Filters _____
Cation Deep Bed Demineralizers _____ Anion Deep Bed Demin. _____ Other _____

b. BORON RECYCLE SYSTEM

Evaporator _____ Deborating Deep Bed Demineralizer _____
Cesium Deep Bed Demineralizer _____ Mixed Bed Cleanup Demineralizer _____
Cartridge Filters _____ Filter-Demins. _____ Edge Type Filters _____ Other _____

c. SPENT FUEL POOL CLEANUP SYSTEM

Deep Bed Demineralizers _____ Filter-Demins. _____ Other _____
Cartridge Filters _____ Edge Type _____ Other _____

d. CONDENSATE POLISHING SYSTEM

Partial Condensate Flow _____ Full Condensate Flow _____ URC _____
Deep Bed Demineralizers _____ Regenerative _____ Non-Regenerative _____
Cartridge Filters _____ Filter-Demins. _____ Precoat Filters _____ Other _____

e. STEAM GENERATOR BLOWDOWN PURIFICATION SYSTEM

Deep Bed Demineralizers _____ Regenerative _____ Non-Regenerative _____
Filter-Demins. _____ Evaporator _____ Other _____

f. LIQUID RADWASTE SYSTEM

Evaporators _____ Deep Bed Demineralizers _____ Filter-Demins. _____
Cartridge Filters _____ Precoat Filters _____ Other _____

g. LAUNDRY LIQUID RADWASTE SYSTEM

Evaporators _____ Cartridge Filter _____ Precoat Filter _____ Other _____

h. CHEMICAL RADWASTE SYSTEM

Evaporators _____ Deep Bed Demineralizers _____ Filter-Demins. _____
Cartridge Filters _____ Precoat Filters _____ Other _____

i. SOLID WASTE SYSTEM

Dewatering Tank _____ Centrifuge _____
Absorbant Material Mixing Equipment _____ Solidification Equipment _____
Trash Compactor _____ Other _____
Phase Separators: Fuel Pool _____ Condensate Cleanup _____

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OWI RADWASTE QUESTIONNAIRE

PART II RADWASTE QUANTITIES AND CHARACTERISTICS

PART II OF THE QUESTIONNAIRE ESTABLISHES THE VOLUME, ACTIVITY, PRINCIPLE RADIONUCLIDES, AND PRINCIPLE CHEMICAL CHARACTERISTICS OF THE VARIOUS TYPES OF RADWASTE GENERATED AT A NUCLEAR POWER PLANT.

THE DATA OBTAINED FROM THESE QUESTIONNAIRES WILL BE USED TO PROJECT WASTE VOLUMES AND THE NEED FOR SPECIAL HANDLING SO THAT THE CONCEPTUAL DESIGN OF THE REPOSITORY WILL BE ADEQUATE FOR ITS PROJECTED USE.

THE QUESTIONNAIRE CONSISTS OF SIX FORMS:

FORM 1 FORM 1 COLLECTS DATA ON LIQUID RADWASTE. SEVERAL SHEETS ARE PROVIDED TO ALLOW FOR THE RECORDING OF DATA SPECIFIC TO CERTAIN LIQUID RADWASTE SYSTEMS OR SOURCES.

FORM 2 FORM 2 COLLECTS DATA ON DEEP BED RESIN WASTE. ONLY ONE SHEET OF FORM 2 IS ATTACHED, AS AT MOST FACILITIES, RESIN WASTES FROM VARIOUS PLANT SYSTEMS ARE NOT SEGREGATED. IF AT YOUR FACILITY RESINS FROM INDIVIDUAL SYSTEMS OR GROUPS OF SYSTEMS ARE SEGREGATED, FORM 2A CAN BE USED TO BREAKDOWN THE RESIN DATA FURTHER.

FORM 3 FORM 3 (AND FORM 3A) ARE USED FOR FILTER-DEMIN WASTE AND ARE USED IN A MANNER SIMILAR TO FORM 2.

FORM 4 FORM 4 IS USED FOR COLLECTING DATA ON FILTER CARTRIDGES. A SEPARATE SHEET IS PROVIDED FOR EACH PLANT SYSTEM.

FORM 5 FORM 5 IS USED FOR COLLECTING DATA ON COMPACTIBLE WASTE GENERATED AT YOUR FACILITY.

FORM 6 FORM 6 IS USED FOR COLLECTING DATA ON NON-COMPACTIBLE WASTE GENERATED AT YOUR FACILITY.

NOTES

1. IF THERE IS MORE THAN ONE ANSWER TO A QUESTION, ENTER ALL ANSWERS AND INDICATE THE RELATIVE % OF WHICH APPLIES. IF MORE ROOM IS NEEDED, USE PLAIN SHEETS OF PAPER AND CROSS REFERENCE.

2. IF YOUR PLANT DATA CANNOT BE BROKEN DOWN TO FIT THE CATAGORIES ASKED, ENTER THE TOTAL VALUES AND INDICATE TO WHICH CATAGORIES THE VALUE APPLIES.

3. IF FOR ANY GIVEN YEAR OR WASTE CATAGORY THE VOLUME SHIPPED DEVIATES SIGNIFICANTLY FROM PREVIOUS DATA, INDICATE THE REASON FOR THIS DEVIATION. AN EXAMPLE, WOULD BE THE SHIPMENT OF SPENT FUEL POOL RACKS WHICH OCCURS ONCE.

OWI-L FORM 1 (FRONT)

LIQUID WASTE

PLANT/UNIT: ____________________

PROCESS STREAM ________ SHEET ____ OF ______

		1971	1972	1973	1974	1975	1976	1977
1 TOTAL LIQUID WASTE VOLUME ①	PROCESSED							
	SHIPPED							
2 TOTAL ACTIVITY SHIPPED								
3 PRINCIPLE RADIONUCLIDES PRESENT. (TYPICAL) (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______
4 SIGNIFICANT CHEMICALS IN WASTE (TYPICAL) (IDENTIFY BY CHEMICAL SPECIES, MANUFACTURER, BRAND NAME, CATALOG NUMBER, OR OTHER DESCRIPTIVE CHARACTERISTIC)								
5 PHYSICAL NATURE OF SHIPPED WASTE (SEE NOTES)	A. REVERSE OSMOSIS SLUDGE ②	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	CONCENTRATOR BOTTOMS	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	CONCENTRATOR CONDENSATE	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	________ SLUDGE	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	________ LIQUID	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	B. SOLIDIFIED	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	SLURRY/SLUDGE	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	LIQUID	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	TURNED OVER TO VENDER FOR SUBSEQUENT PROCESSING	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
6 CONCENTRATOR	% SOLIDS IN CONCENTRATE (NOMINAL RANGE)							
pH IN CONCENTRATE (NOMINAL RANGE)	ACID <6.5 ②	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	NEUTRAL 6.5-7.5	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	BASE >7.5	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	ACTUAL RANGE IF KNOWN	______	______	______	______	______	______	______
7 SOLIDIFICATION	SOLIDIFICATION (IDENTIFY BY PROCESS, MANUFACTURER, BRAND NAME, CATALOG NUMBER, VENDOR, OR OTHER DESCRIPTIVE CHARACTERISTIC)							
	% WASTE TO SOLIDIFIED VOLUME (NOMINAL RANGE) ③							

		1971	1972	1973	1974	1975	1976	1977
8	CONTAINER DESCRIPTION (4) (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, LIC. #, DOT SPEC, MANUFACTURER, ETC.)							
9	DENSITY OF WASTE AS SHIPPED							
10	NOMINAL CONTAINER CAPACITY							
11	% CONTAINER CAPACITY USED (NOMINAL RANGE)							
	IS FILLING CONTROLLED ON THE BASIS OF RADIATION LEVEL? (2)	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%
12	INTEGRAL SHIELDING (5) — USED? (2)	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%
	WEIGHT OF SHIELDED PACKAGE, WEIGHT OF SHIELDING FOR ONE CONTAINER — NUMBER OF SHIELDED CONTAINERS SHIPPED							
	SHIELDING MATERIAL							
	TOTAL VOLUME OF SHIELDING SHIPPED							
13	RADIATION LEVELS MR/HR (NOMINAL RANGE) — CONTACT							
	@ 3 FEET							

NOTES

1 INCLUDE SOLIDIFICATION AND PACKAGING/SHIELDING VOLUMES IN TOTAL SHIPPED
2 CHECK THOSE BOXES WHICH APPLY. IF MORE THAN ONE BLOCK IS CHECKED PER SECTION INDICATE RELATIVE PERCENTAGES OF EACH.
3 FOR EXAMPLE: IF 25 GALS. OF WASTE IS MIXED WITH SOLIDIFICATION MEDIA TO FILL A 55 GAL. DRUM, PERCENTAGE WOULD BE 45%.
4 DESCRIBE THE CONTAINER WHICH IS ULTIMATELY BURIED. ALSO DESCRIBE THE SHIPPING OVERPACKS IF APPLICABLE.
5 IF THE SHIPPING CONTAINER INCORPORATES SHIELDING, AND THE SHIELDING WILL BE BURIED, CHECK ☑ AND ANSWER THE ADDITIONAL QUESTIONS. DO NOT INCLUDE CONCRETE USED AS SOLIDIFICATION MEDIA AS SHIELDING.

OWI-L FORM 2 (FRONT)

DEEP-BED RESIN WASTE
PLANT/UNIT ____________
CATEGORY ____________ ①
SHEET ______ OF ________

		1971	1972	1973	1974	1975	1976	1977
1 TOTAL WASTE VOLUME:	GENERATED							
	SHIPPED ②							
2 TOTAL ACTIVITY SHIPPED								
3 PRINCIPLE RADIONCLIDES (TYPICAL) (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ________
4 SIGNIFICANT CHEMICALS IN WASTE (IDENTIFY THOSE CHEMICALS WHICH THE RESIN IS KNOWN TO CONTAIN BY CHEMICAL NAME OR MANUFACTURER DATA)	SYSTEM ③							
5 PHYSICAL NATURE OF WASTE ④	RESIN SLURRY	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	DEWATERED RESIN SLURRY	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	SOLIDIFIED RESIN SLURRY	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	OTHER ____________	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	TURNED OVER TO VENDOR FOR SUBSEQUENT PROCESSING	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
6 SOLIDIFICATION	SOLIDIFICATION AGENT ⑤ (MANUFACTURER, BRAND NAME CATALOG NO., TYPE, PROCESS, ETC.)							
	% WASTE TO SOLIDIFIED WASTE (NOMINAL RANGE)							

	1971	1972	1973	1974	1975	1976	1977
7 CONTAINER DESCRIPTION (6) (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, LIC. #, DOT SPEC, MANUFACTURER, ETC.)							
8 DENSITY OF WASTE AS SHIPPED							
9 NOMINAL CONTAINER CAPACITY							
10 % VOID SPACE (NOMINAL RANGE)							
IS FILLING CONTROLLED ON THE BASIS OF RADIATION LEVEL? (4)	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
11 INTEGRAL SHIELDING (7) USED? (4)	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
WEIGHT OF SHIELDED PACKAGE, WEIGHT OF SHIELDING FOR ONE CONTAINER — SHIELD MATERIAL							
SHIELD VOLUME (TOTAL)							
NUMBER OF SHIELDED CONTAINERS SHIPPED							
12 RADIATION LEVELS (NOMINAL RANGE) — CONTACT							
@ 3 FEET							

	SYSTEM	RESIN USED	1971	1972	1973	1974	1975	1976	1977
(8) VOLUME EXPENDED OR DRAWN FROM INVENTORY									
RESIN DESCRIPTION									

(8) USING STOREROOM INVENTORY DATA OR OPERATIONS DATA COMPLETE THIS SUMMARY SECTION. DESCRIBE FILTER MEDIA BY MANUFACTURER, BRAND NAME, CATALOG NO, AND/OR OTHER DESCRIPTIVE CHARACTERISTICS. PLACE A CHECK (√) IN THE APPROPRIATE BLOCKS TO INDICATE WHICH FILTER MEDIAS ARE USED IN DIFFERENT SYSTEMS

1 IN MOST CASES A BREAKDOWN OF SLUDGE DATA SHIPPED WILL NOT BE POSSIBLE DUE TO COMMON COLLECTION, PROCESSING, AND STORAGE FACILITIES. IF A BREAKDOWN IS POSSIBLE, BECAUSE OF SEPARATE COLLECTION, PROCESSING, AND STORAGE FACILITIES, USE ADDITIONAL FORM 2A'S TO RECORD THIS DATA. IN EITHER EVENT, COMPLETE THE SUMMARY BLOCK. SEE NOTE 8.

2 INCLUDE SOLIDIFICATION, SHIELDING, AND PACKAGING VOLUMES.

3 TO THE EXTENT POSSIBLE, ANSWER BY SYSTEM.

4 CHECK THOSE BOXES WHICH APPLY. IF MORE THAN ONE BOX IS CHECKED PER SECTION, INDICATE RELATIVE % OF EACH.

5 FOR EXAMPLE, IF 25 GAL OF WASTE IS MIXED WITH SOLIDIFICATION MEDIA TO FILL A 55 GAL DRUM, PERCENTAGE WOULD BE 45%

6 DESCRIBE THE CONTAINER WHICH IS ULTIMATELY BURIED. ALSO DESCRIBE SHIPPING OVERPACKS IF APPLICABLE

7 IF THE SHIPPING CONTAINER INCORPORATES SHIELDING AND THE SHIELDING WILL BE BURIED, CHECK Y AND ANSWER THE ADDITIONAL QUESTIONS. DO NOT INCLUDE CONCRETE USED AS SOLIDIFICATION MEDIA AS SHIELDING.

OWI-L FORM 2A (FRONT)

DEEP-BED RESIN WASTE
PLANT/UNIT________
CATEGORY________
SHEET______ OF________

		1971	1972	1973	1974	1975	1976	1977
1 TOTAL WASTE VOLUME:	GENERATED							
	SHIPPED							
2 TOTAL ACTIVITY SHIPPED								
3 PRINCIPLE RADIONCLIDES (TYPICAL) (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____
4 SIGNIFICANT CHEMICALS IN WASTE (IDENTIFY THOSE CHEMICALS WHICH THE RESIN IS KNOWN TO CONTAIN BY CHEMICAL NAME OR MANUFACTURER DATA)	SYSTEM							
5 PHYSICAL NATURE OF WASTE	RESIN SLURRY	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%
	DEWATERED RESIN SLURRY	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%
	SOLIDIFIED RESIN SLURRY	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%
	OTHER ________	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%
	TURNED OVER TO VENDOR FOR SUBSEQUENT PROCESSING	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%	☐ ____%
6 SOLIDIFIED	SOLIDIFICATION AGENT (MANUFACTURER, BRAND NAME CATALOG NO., TYPE, PROCESS, ETC.)							
	% WASTE TO SOLIDIFIED WASTE (NOMINAL RANGE)							

		1971	1972	1973	1974	1975	1976	1977
7	CONTAINER DESCRIPTION (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, LIC. #, DOT SPEC, MANUFACTURER, ETC.)							
8	DENSITY OF WASTE AS SHIPPED							
9	NOMINAL CONTAINER CAPACITY							
10	% VOID SPACE (NOMINAL RANGE)							
	IS FILLING CONTROLLED ON THE BASIS OF RADIATION LEVEL?	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%
11	INTEGRAL SHIELDING — USED?	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%	☐__% ☐__%
	WEIGHT OF SHIELDED PACKAGE, WEIGHT OF SHIELDING FOR ONE CONTAINER — SHIELD MATERIAL							
	SHIELD VOLUME (TOTAL)							
	NUMBER OF SHIELDED CONTAINERS SHIPPED							
12	RADIATION LEVELS (NOMINAL RANGE) — CONTACT							
	@ 3 FEET							

OWI-L FORM 3 (FRONT)

FILTER-DEMIN AND PRECOAT SLUDGE

PLANT/UNIT__________

CATEGORY__________ ①

SHEET_____ OF_____

		1971	1972	1973	1974	1975	1976	1977
1 TOTAL WASTE VOLUME:	GENERATED							
	SHIPPED ②							
2 TOTAL ACTIVITY SHIPPED								
3 PRINCIPLE RADIONCLIDES (TYPICAL) (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other _____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other _____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other _____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other _____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other _____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other _____	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other _____
4 SIGNIFICANT CHEMICALS IN WASTE (IDENTIFY THOSE CHEMICALS WHICH THE RESIN IS KNOWN TO CONTAIN BY CHEMICAL NAME OR MANUFACTURER DATA)	SYSTEM ③							
5 PHYSICAL NATURE OF WASTE ④	A. POWDEX SLURRY	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %
	PRECOAT SLURRY	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %
	OTHER________	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %
	B. DEWATERED	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %
	SOLIDIFIED	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %
	TURNED OVER TO VENDOR FOR FURTHER PROCESSING							
	OTHER________	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %	☐ _____ %
6 SOLIDIFICATION	SOLIDIFICATION AGENT ⑤ (MANUFACTURER, BRAND NAME CATALOG NO., TYPE, PROCESS, ETC.)							
	% WASTE TO SOLIDIFIED WASTE (NOMINAL RANGE)							

		1971	1972	1973	1974	1975	1976	1977
7	CONTAINER DESCRIPTION (4) (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, LIC. #, DOT SPEC, MANUFACTURER, ETC.)							
8	DENSITY OF WASTE AS SHIPPED							
9	NOMINAL CONTAINER CAPACITY							
10	% VOID SPACE (NOMINAL RANGE)							
	IS FILLING CONTROLLED ON THE BASIS OF RADIATION LEVEL? (2)	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%
11	(5) INTEGRAL SHIELDING — USED? (2)	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%	YES ☐__% NO ☐__%
	WEIGHT OF SHIELDED PACKAGE, WEIGHT OF SHIELDING FOR ONE CONTAINER — SHIELD MATERIAL							
	SHIELD VOLUME (TOTAL)							
	NUMBER OF SHIELDED CONTAINERS SHIPPED							
12	RADIATION LEVELS (NOMINAL RANGE) — CONTACT							
	@ 3 FEET							

F-16

	SYSTEM	FILTER MEDIA	1971	1972	1973	1974	1975	1976	1977
(8) VOLUME EXPENDED OR DRAWN FROM INVENTORY									
DESCRIPTION OF FILTER MEDIA									

(8) USING STOREROOM INVENTORY DATA OR OPERATIONS DATA COMPLETE THIS SUMMARY SECTION. DESCRIBE FILTER MEDIA BY MANUFACTURER, BRAND NAME, CATALOG NO, AND/OR OTHER DESCRIPTIVE CHARACTERISTICS. PLACE A CHECK (√) IN THE APPROPRIATE BLOCKS TO INDICATE WHICH FILTER MEDIAS ARE USED IN DIFFERENT SYSTEMS

1 IN MOST CASES A BREAKDOWN OF SLUDGE DATA SHIPPED WILL NOT BE POSSIBLE DUE TO COMMON COLLECTION, PROCESSING, AND STORAGE FACILITIES. IF A BREAKDOWN IS POSSIBLE, BECAUSE OF SEPARATE COLLECTION, PROCESSING, AND STORAGE FACILITIES, USE ADDITIONAL FORM 3A'S TO RECORD DATA. IN EITHER EVENT, COMPLETE THE SUMMARY BLOCK. SEE NOTE 8.

2 INCLUDE SOLIDIFICATION, SHIELDING, AND PACKAGING VOLUMES.

3 TO THE EXTENT POSSIBLE, ANSWER BY SYSTEM.

4 CHECK THOSE BOXES WHICH APPLY. IF MORE THAN ONE BOX IS CHECKED PER SECTION, INDICATE RELATIVE % OF EACH.

5 FOR EXAMPLE, IF 25 GAL OF WASTE IS MIXED WITH SOLIDIFICATION MEDIA TO FILL A 55 GAL DRUM, PERCENTAGE WOULD BE 45%

6 DESCRIBE THE CONTAINER WHICH IS ULTIMATELY BURIED. ALSO DESCRIBE SHIPPING OVERPACKS IF APPLICABLE

7 IF THE SHIPPING CONTAINER INCORPORATES SHIELDING AND THE SHIELDING WILL BE BURIED, CHECK ☑ AND ANSWER THE ADDITIONAL QUESTIONS. DO NOT INCLUDE CONCRETE USED AS SOLIDIFICATION MEDIA AS SHIELDING.

OWI-L FORM 3A(FRONT)

FILTER-DEMIN AND PRECOAT SLUDGE

PLANT/UNIT____________

CATEGORY____________

SHEET______ OF________

		1971	1972	1973	1974	1975	1976	1977
1 TOTAL WASTE VOLUME:	GENERATED							
	SHIPPED							
2 TOTAL ACTIVITY SHIPPED								
3 PRINCIPLE RADIONCLIDES (TYPICAL) (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______
4 SIGNIFICANT CHEMICALS IN WASTE (IDENTIFY THOSE CHEMICALS WHICH THE RESIN IS KNOWN TO CONTAIN BY CHEMICAL NAME OR MANUFACTURER DATA)	SYSTEM							
5 PHYSICAL NATURE OF WASTE	A. POWDEX SLURRY	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	PRECOAT SLURRY	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	OTHER________	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	B. DEWATERED	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	SOLIDIFIED	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
	TURNED OVER TO VENDOR FOR FURTHER PROCESSING							
	OTHER________	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%	☐ ______%
6 SOLIDIFICATION	SOLIDIFICATION AGENT (MANUFACTURER, BRAND NAME CATALOG NO., TYPE, PROCESS, ETC.)							
	% WASTE TO SOLIDIFIED WASTE (NOMINAL RANGE)							

		1971	1972	1973	1974	1975	1976	1977
8 CONTAINER DESCRIPTION (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, LIC. #, DOT SPEC, MANUFACTURER, ETC.)								
9 DENSITY OF WASTE AS SHIPPED								
10 NOMINAL CONTAINER CAPACITY								
11 % VOID SPACE	(NOMINAL RANGE)							
IS FILLING CONTROLLED ON THE BASIS OF RADIATION LEVEL?		Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
12 INTEGRAL SHIELDING WEIGHT OF SHIELDED PACKAGE, WEIGHT OF SHIELDING FOR ONE CONTAINER	USED?	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
	SHIELD MATERIAL							
	SHIELD VOLUME (TOTAL)							
	NUMBER OF SHIELDED CONTAINERS SHIPPED							
13 RADIATION LEVELS MR/HR (NOMINAL RANGE)	CONTACT							
	@ 3 FEET							

OWI-L FORM 4 (FRONT)

FILTER CARTRIDGES

PLANT/UNIT ____________
SYSTEM ____________ ⑥
SHEET _____ OF ________

		1971	1972	1973	1974	1975	1976	1977
NUMBER OF (ASSEMBLIES) (CARTRIDGES) SHIPPED: ①								
TOTAL ACTIVITY SHIPPED								
FILTER DESCRIPTION (MANUFACTURER, TYPE, BRAND NAME, CATALOG NO. AND/OR OTHER DESCRIPTIVE CHARACTERISTICS)								
MICRON FILTER SIZE								
NUMBER OF ASSEMBLIES PER CARTRIDGE								
FILTER SHELL DISPOSED? ②		Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
PRINCIPLE RADIONUCLIDES (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ____________
SIGNIFICANT CHEMICALS IN WASTE (IDENTIFY BY CHEMICAL SPECIES, MANUFACTURER, BRAND NAME, CATALOG NO. AND/OR OTHER DESCRIPTIVE CHARACTERISTICS								
IS THE FILTER CHANGED DUE TO RADIATION LEVEL OR ΔP	RAD LEVEL	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
	ΔP ②	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
SOLIDIFICATION/ ENCAPSULATION	ARE FILTER (ASSEMBLIES) (CARTRIDGES) ENCAPSULATED PRIOR TO SHIPPING ① ②	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %	Y ____ % N ____ %
	ENCAPSULATION MATERIAL (MANUFACTURER, BRAND NAME, CATALOG NO., ETC.)							
	NOMINAL TOTAL VOLUME OF ENCAPSULATING MATERIAL ③							

		1971	1972	1973	1974	1975	1976	1977
SHIPPING CONTAINER DESCRIPTION (4) (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, DOT SPEC, LIC. #, MANUFACTURER, ETC.)								
DENSITY OF WASTE AS SHIPPED								
NOMINAL CONTAINER VOLUME								
NUMBER OF (ASSEMBLIES) (CARTRIDGES) PER SHIPPING CONTAINERS (1)								
(5) INTEGRAL SHIELDING	USED? (2)	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%
WEIGHT OF SHIELDED CONTAINER, WEIGHT OF SHIELDING FOR ONE CONTAINER	NUMBER OF SHIELDED CONTAINERS SHIPPED							
	SHIELD MATERIAL							
	TOTAL SHIELD VOLUME							
	IS SHIELD IN PLACE PRIOR TO OR AFTER USE? (2)	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%	Y ____% N ____%
RADIATION LEVEL (NOMINAL RANGE)	CONTACT							
	@ 3 FEET							

1 IF THE CARTRIDGES ARE REMOVED AND HANDLED SEPARATELY (NOT AS AN ASSEMBLY) CROSS OUT THE WORD ASSEMBLY IN ALL QUESTIONS AND BASE THE ANSWERS ON SINGLE CARTRIDGE DATA

2 CHECK THOSE BOXES WHICH APPLY. IF MORE THAN ONE BOX PER SECTION IS CHECKED, INDICATE RELATIVE PERCENTAGE OF EACH

3 INDICATE THE TOTAL VOLUME OF ENCAPSULATING MATERIAL USED FOR ALL SHIPMENTS FOR THE YEAR. THIS DATA MAY BE DERIVED FROM STOREROOM INVENTORY RECORDS.

4 DESCRIBE THE CONTAINER WHICH IS ULTIMATELY BURIED. ALSO DESCRIBE SHIPPING OVERPACKS, IF APPLICABLE.

5 IF THE SHIPPING CONTAINER INCORPORATES SHIELDING, AND THE SHIELDING WILL BE BURIED, CHECK Y AND ANSWER THE ADDITIONAL QUESTIONS. DO NOT INCLUDE CONCRETE USED AS SOLIDIFICATION MEDIA AS SHIELDING

6 IF NOT ALREADY COMPLETED, INDICATE SYSTEM IN WHICH FILTER IS USED.

OWI-L FORM 5 (FRONT)

COMPACTIBLE MATERIAL

PLANT/UNIT________________
CATEGORY________________
SHEET_______ OF__________

		1971	1972	1973	1974	1975	1976	1977
1 TOTAL VOLUME SHIPPED ①								
2 TOTAL ACTIVITY SHIPPED								
3 PRINCIPLE RADIONUCLIDES SHIPPED (TYPICAL) (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other________	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other________
4 DESCRIPTION OF REPRESENTATIVE RADIOACTIVE MATERIAL SHIPPED. ②								
5 CONTAINER DESCRIPTION (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, LIC. #, DOT SPEC.) MANUFACTURER, ETC.) ③								
6 DENSITY OF WASTE AS SHIPPED.								
7 NOMINAL CONTAINER VOLUME								
% VOID (NOMINAL RANGE)								
8 RADIATION LEVEL MR/HR (NOMINAL RANGE)	CONTACT							
	@ 3 FEET							

NOTES

1 INCLUDE SHIPPING CONTAINER VOLUME

2 AS DETAILED, AS POSSIBLE, DESCRIBE THE WASTE SHIPPED. IF A GIVEN YEARS VOLUME DEVIATES SIGNIFICANTLY FROM PREVIOUS OR SUBSEQUENT YEARS, INDICATE REASON.

3 DESCRIBE THE CONTAINER WHICH IS ULTIMATELY BURIED. ALSO, DESCRIBE THE SHIPPING OVERPACKS IF APPLICABLE.

4 IF CONTAINERS ARE NOT ROUTINELY COMPLETELY FILLED, INDICATE NOMINAL RANGE OF VOID SPACE

OWI-L FORM 6 (FRONT)

NON – COMPACTIBLE MATERIAL

PLANT/UNIT____________
CATEGORY____________
SHEET______ OF________

		1971	1972	1973	1974	1975	1976	1977
1 TOTAL VOLUME SHIPPED ①								
2 TOTAL ACTIVITY SHIPPED								
3 PRINCIPLE RADIONUCLIDES SHIPPED (TYPICAL) (CIRCLE THOSE NORMALLY PRESENT)		H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______	H3, Cr51, Mn54, Fe55, Fe59 Co58, Co60, Rb86, Sr89, Sr90, Sr91, Zr95, Nb95, Mo99, Tc99M, Ru103, Ru106, Ag110M, Cs134, Cs137, Ce141, Np239, MFP, MCP, other ______
4 DESCRIPTION OF REPRESENTATIVE RADIOACTIVE MATERIAL SHIPPED. ②								
5 CONTAINER DESCRIPTION (OVERALL DIMENSIONS, USEABLE VOLUME, # SHIPPED, MATERIAL, LIC. #, DOT SPEC.) MANUFACTURER, ETC.) ③								
6 DENSITY OF WASTE AS SHIPPED.								
7 NOMINAL CONTAINER VOLUME								
8 RADIATION LEVEL (NOMINAL RANGE)	CONTACT							
	@ 3 FEET							

NOTES

1 INCLUDE SHIPPING CONTAINER VOLUME

2 AS DETAILED, AS POSSIBLE, DESCRIBE THE WASTE SHIPPED. IF A GIVEN YEARS VOLUME DEVIATES SIGNIFICANTLY FROM PREVIOUS OR SUBSEQUENT YEARS, INDICATE REASON.

3 DESCRIBE THE CONTAINER WHICH IS ULTIMATELY BURIED. ALSO DESCRIBE THE SHIPPING OVERPACKS IF APPLICABLE.

GENERAL QUESTIONS	
ARE WASTES GENERALLY CHECKED FOR ALPHA CONTAMINATION?	YES ☐ NO ☐
IF SO, WHAT IS THE TYPICAL READING, AND WHAT % EXCEEDS 10 NANOCURIES/GM?	_____nCi/gm _____%
HOW LONG IS WASTE TYPICALLY STORED ON SITE PRIOR TO BEING SHIPPED FOR DISPOSAL?	_____ weeks
WHERE ARE YOUR WASTES BEING SHIPPED TODAY? BY WHAT MODE?	TRUCK ☐ RAIL ☐ _____ OTHER ☐
DO YOU OWN OR RENT TRANSPORTATION SHIELDS?	_____OWN _____RENT

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