DRAFT

INVENTORY OF MATERIALS WITH VERY LOW LEVELS OF RADIOACTIVITY POTENTIALLY CLEARABLE FROM VARIOUS TYPES OF FACILITIES

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TABLE OF CONTENTS

TABLE OF CONTENTS
(continued)

TABLE OF CONTENTS
(continued)

TABLE OF CONTENTS
(continued)

TABLE OF CONTENTS
(continued)

APPENDICES

LIST OF TABLES

LIST OF TABLES
(continued)

LIST OF TABLES
(continued)

LIST OF TABLES
(continued)

LIST OF TABLES
(continued)

LIST OF TABLES
(continued)

LIST OF FIGURES

LIST OF FIGURES
(continued)

CONVERSION FACTORS

To Convert From	To	Multiply By	To Convert From	To	Multiply By
acre	hectare	0.405	meter (m)	inch	39.4
	sq. meter (m^2)	4,050		mile	0.000621
	sq. feet (ft^2)	43,600	sq. meter (m^2)	acre	0.000247
becquerel (Bq)	curie (Ci)	2.7×10^{-11}		hectare	0.0001
	dps	1		sq. feet (ft^2)	10.8
	pCi	27		sq. mile	3.86×10^{-7}
Bq/kg	pCi/g	0.027	m^3	liter	1,000
Bq/m^2	dpm/100 cm^2	1.67	mrem	mSv	0.01
Bq/m^3	Bq/L	0.001	mrem/y	mSv/y	0.01
	pCi/L	0.027	mSv	mrem	100
centimeter (cm)	inch	0.394	mSv/y	mrem/y	100
Ci	Bq	3.70×10^{10}	ounce (oz)	liter (L)	0.0296
	pCi	1×10^{12}	pCi	Bq	0.037
				dpm	0.45
dps	dpm	0.0167	pCi/g	Bq/kg	37
	pCi	27	pCi/L	Bq/m^3	37
dpm	dps	60	rad	Gy	0.01
	pCi	2.22	rem	mrem	1,000
gray (Gy)	rad	100		mSv	10
hectare	acre	2.47		Sv	0.01
liter (L)	cm^3	1000	seivert (Sv)	mrem	100,000
	m^3	0.001		mSv	1,000
	ounce (fluid)	33.8		rem	100

EXECUTIVE SUMMARY

DRAFT

INTRODUCTION

On June 30, 1999, the U.S. Nuclear Regulatory Commission (NRC) published an Issues Paper indicating that the NRC was examining its approach for "release of solid material" (64 FR 35090). The NRC does not have a regulation that uniformly addresses the release of solid materials. The term "release of solid material," as used in the Issues Paper, is a general term applied to solid materials and equipment that leave the radiological control of the licensee. It includes releases with and without further restrictions or controls. A subset of all kinds of releases are those releases of materials and equipment with no subsequent radiological control. This latter kind of release is internationally called, "clearance." Solid materials and equipment that undergo clearance are said to be "cleared." Thus, from a regulatory point of view, cleared materials and equipment may be treated with respect to their radiological properties as ordinary materials and equipment.

Following a number of related activities, on August 18, 2000, the Commission directed the staff to:

(1) Defer a final decision on whether to proceed with rulemaking,

(2) Proceed with a National Academies study on possible alternatives for release of slightly contaminated materials,

(3) Continue the development of technical information base necessary to support a Commission policy decision in this area, and

(4) Stay informed of international initiatives in this area, related U.S. Environmental Protection Agency (EPA) and Department of State (DOS) activities, and potential for import and trade issues.

The regulatory alternatives in the Issues Paper provide benchmarks for the continued development of the technical information. They are:

- No action
- Prohibiting release of material that had been in an area in a licensed facility where radioactive material was used or stored

- Restricting release to only certain authorized uses
- Permitting material to be released from license control if it meets certain dose-based criteria; i.e., 100 μSv/a (10 mrem/yr) clearance level, 10 μSv/a (1 mrem/yr) clearance level, 1 μSv/a (0.1 mrem/yr) clearance level, or zero above background

This report addresses a portion of the continued development of technical information. It is a review of available published sources of information on the amounts and kinds of radionuclides associated with potentially clearable materials and equipment. The report seeks to identify the form and composition of scrap, its tonnage, and the kinds and amounts of radioactivity that are associated with each category of material as appropriate for the dose assessments and costs estimates. The report focuses on sources of information regarding materials and equipment with the potential for clearance by NRC and Agreement State licensees. Broader considerations are also recognized. Generators of materials and equipment with very low levels of associated radioactivity, such as the U.S. Department of Energy (DOE), other industries, and the U.S. Department of Defense (DOD), also may be affected by the proposed NRC actions. These sources are also considered. This report highlights four categories of materials -- ferrous metals (carbon steel, stainless steel, cast iron), aluminum and its alloys, copper and its alloys, and concrete. Other materials are addressed in a more limited way.

This report is the first step in an iterative process of developing a database for use in collective dose and cost/benefit analyses associated with the various alternatives. The next step is to identify data gaps in the information available in the literature and propose means to fill the gaps, and the final step in the process is to take appropriate actions to address the gaps and complete the inventory.

Chapter 1 provides background information and other introductory material.

Chapter 2 of this report discusses inventories associated with NRC and Agreement State Licensees. Inventories expected to be generated from commercial nuclear power reactors, non-power reactors, nuclear fuel cycle facilities (e.g., fuel fabrication plants, UF_6 conversion plants, independent spent fuel storage facilities, and uranium mills), and non-fuel-cycle materials licensees (e.g., sealed source manufacturers, R&D laboratories, hospitals) are developed to the extent that information on quantities of materials and kinds and amounts of radioactivity is available in the literature. Detailed information is not available for each licensee. However, for some types of facilities, the NRC has previously prepared decommissioning studies on generic or reference facilities which facilitate the inventory analyses. To the degree possible, quantities of material associated with specific radioactivity levels are estimated for each reference facility. In the case of nuclear power reactors, which are the source of the majority of the NRC-licensee inventories of potentially clearable materials, scaling factors are developed to account for differences in reactor size.

Chapter 3 provides information on DOE inventories. DOE has estimated that current metals in scrap yards within the DOE complex plus metals expected from dismantling of obsolete facilities will amount to about one million tons. Approximately 60 percent of this metal is associated with the gaseous diffusion plants at Oak Ridge (K-25 Site), Portsmouth, and Paducah. Decontamination and decommissioning (D&D) is in progress at the K-25 Site. Based on limited frequency distributions of metal surface radioactivity, estimates are made in this chapter of the incremental quantities of ferrous metals associated with specific surface radioactivity levels. Insufficient information was uncovered in the published literature to quantitatively characterize DOE facilities other than the gaseous diffusion plants.

Chapter 4 provides information on the DOD facilities, both licensed and unlicensed. Most DOD facilities using potentially clearable materials are licensed by the NRC and cover the same spectrum of operations as other non-fuel-cycle materials licensees, e.g., hospitals, research laboratories, users of sealed source gauges, and irradiators. Naval nuclear reactor propulsion facilities are not licensed by the NRC. When nuclear ships are decommissioned, the reactor compartments are cut from the hull, sealed, and shipped for burial at Hanford. The ship hulls are scrapped. Depleted uranium armaments used by the military as armor-piercing ammunition and as tank armor are also not licensed by the NRC.

Chapter 5 discusses those unlicensed commercial industries using or processing materials that contain naturally occurring radioactivity (NORM), which because of their operations, create higher concentrations of radioactivity than associated with an undisturbed natural setting. This material is defined as technically enhanced, naturally occurring radioactive material (TENORM). Radioactive species associated with TENORM are typically uranium, thorium, and their decay products.

INVENTORIES FROM NRC AND AGREEMENT STATE LICENSEES

Facilities currently licensed to operate by NRC include:

- Commercial nuclear power reactors - 104
- Non-nuclear power reactors - 37
- Fuel cycle facilities
  - Uranium fuel fabrication plants - 7
  - Uranium hexafluoride production plant - 1
  - Gaseous diffusion plants - 2
  - Uranium mills and leaching operations - 25
  - Spent fuel storage facilities - 14
- Materials licensees - 5,288

In addition, the Agreement States have issued 15,512 materials licenses. A number of formerly licensed facilities are also significant inventory sources.

These broad groupings of licensees were retained for the analyses presented here since the groupings represent facilities with common characteristics, such as materials of construction and radioactivity levels. That is not to say that inventory-related parameters do not range significantly within each category, but use of the reference facility approach makes the inventory development a more tractable problem. The focus of the analysis is on materials generated during the D&D of obsolete facilities since this involves the greatest quantity of materials and therefore the greatest impacts on collective doses, costs, and benefits. However, for several of the facility groupings (e.g., commercial nuclear power plants and non-fuel-cycle licensees), information was also developed on the inventory generated by ongoing operations. The general approach taken to develop the inventory for each group or subgroup of facilities is as follows:

- Define the number of facilities within a group or subgroup of licensees

- Determine the inventory characteristics of a reference facility for the group including quantities and types of materials and quantities and types of radionuclides

- Develop scaling factors to expand the reference facility inventories to the entire group of licensees

- Augment and validate inventory characteristics for the reference facility with actual decommissioning experience from facilities currently undergoing D&D

Commercial Nuclear Power Reactors

The decommissioning of commercial nuclear power reactors is the largest source of solid materials potentially available for clearance from licensed facilities. Materials generated during D&D include the following:

- Neutron-activated metals and concrete, which contains the bulk of the residual radioactivity at shutdown and are disposed of as low-level waste
- Materials with associated surficial radioactivity, which are unlikely to be cleaned to levels that would permit clearance and are also disposed of as low-level waste
- Materials that have low levels of radioactivity, which might be clearable after cleaning
- Materials that have very low levels of radioactivity, which might be clearable under a clearance standard
- Materials that were outside the radiation control area at the reactor facility and have no detectable radioactivity, designated as clean material

The reference Boiling Water Reactor is a 1,100 MWe plant (WPPSS No. 2) containing about 34,000 metric tons (t) of ferrous metals (of which about 18,000 t are rebar embedded in concrete) and about 355,000 t of concrete. Materials estimates for the reference Pressurized Water Reactor are based on a 1,000 MWe PWR plant containing about 36,000 t of ferrous metals (including 9,600 t of rebar) and 180,000 t of concrete. Radioactivity levels estimated for the reference PWR are based on the 1,175 MWe Trojan Nuclear Plant. Sufficient data are available from reactor measurements to adequately characterize the mix of radionuclides contributing to the radioactivity levels of various reactor components both from neutron-activation and surface deposition in the reference BWR and PWR. The masses of materials in the reference BWR with associated qualitative radioactivity levels are summarized in Table ES-1. No information was available on copper.

The lowest radioactivity level of any of the ferrous metal items in this table is 1,000 pCi/g (37 Bq/g) for the main turbine.

Table ES-1. Mass Summary for Reference BWR

Radioactivity Level	Material Mass (t)		
	Concrete	Ferrous Metals	Aluminum
Activated	180	330	N/A
Surfical	800	8,000	58
Clean	354,000	26,000	N/A
Total	355,000	34,000	N/A

N/A - not available

Similar estimates for the reference PWR are summarized in Table ES-2.

Table ES-2. Mass Summary for Reference PWR

Radioactivity Level	Material Mass (t)			
	Concrete	Ferrous Metals	Copper	Aluminum
Activated/ Surficial	284[a]	3,978	50	5
Clean	179,000[b]	32,091	644	13
Total	179,681	36,069	694	18

a Excluding activated volume of primary shield wall

b Rounded value

The radioactivity concentrations of ferrous metal components/systems generally exceed about 1,000 pCi/g (37 Bq/g).

A scaling factor, based on the electrical power output of each nuclear power plant, has been developed to adjust the amount of material in the reference BWR and PWR to units of various sizes.

Information about the quantities of materials and the kinds and amounts of radioactivity expected to be generated during the decommissioning of the reference BWR and PWR has been supplemented from experience at several large commercial reactors currently undergoing decommissioning including Haddam Neck, Maine Yankee, Yankee Rowe, Trojan, and Rancho Seco.

In general, concrete is either used as onsite fill after removal of radioactivity, shipped for disposal as low-level waste, or shipped for disposal in a commercial landfill. Although the construction industry recycles some concrete, no instances of concrete recycle by NRC licensees were identified. Only limited examples of equipment reuse by licensees were identified. More commonly, equipment is disposed of as scrap metal, buried as low-level waste, buried in a commercial landfill, or melted into shielding blocks for use by the Department of Energy.

In addition to solid materials expected from decommissioning, a variety of materials are removed from each operating site on an ongoing basis. Such movement of solid materials may include disposal as low-level waste, clearance for reuse, recycle, or commercial landfill disposal, processing by a nuclear waste broker, or case-by-case disposal under the provisions of 10 CFR Part 20.2002. The NRC has authorized about 35 case-by-case disposals under 10 CFR 20.2002. These case-by-case authorizations have included sand, soils, roofing materials, wood, sewage sludges, resins, pond sediments, and the like. The radioactivity varies from a fraction of a millicurie to tens of millicuries of nuclides such as Co-58, Co-60, Cs-134, Cs-137, and Mn-54.

Fuel Cycle Facilities

Fuel cycle facilities that may be sources of potentially clearable materials include uranium mills, uranium hexafluoride conversion plants, fuel fabrication facilities, and independent spent fuel storage installations. Uranium mills include both conventional mills and in situ leaching operations licensed by either the NRC or specific Agreement States. The major radioactive isotopes are U-235 (and daughters) and U-238 (and daughters). Most conventional uranium mills have been shutdown and are undergoing decommissioning. These mills are not likely to be significantly affected by any future NRC regulations relating to the clearance of solid materials from regulatory control since dismantlement of most will likely be well advanced or completed prior to any rulemaking. Four conventional mills are either operating or on standby status. Since there is little or no salvageable equipment and most materials are inexpensively buried in onsite tailings piles or at other approved sites, the quantities of potentially clearable materials from uranium mills are expected to be quite small.

Similar to the situation with conventional uranium mills, many in situ leach facilities have been shut down and are undergoing decommissioning. These shutdown facilities are unlikely to be affected by an NRC clearance rule since dismantlement is expected to be largely completed prior to issuance of any final rule. Seven in situ leach facilities are operating, in standby status, or not yet built. Large quantities of materials and equipment are not expected to be available when these facilities are ultimately decommissioned. Equipment and plastic piping with associated radioactivity are likely to be disposed of in tailings piles or at other licensed disposal sites.

There are only two UF_6 production facilities: the shutdown Sequoyah Fuels Corporation facility with a capacity of 5,000 MTU/yr and the ConverDyn facility with a capacity of 14,000 MTU/yr. The reference facility developed in a 1981 NRC study was assumed to have an annual processing rate of 10,000 MTU/yr. The principal radionuclides of concern are U-235 and U-238 and progeny. The mass of steel in the reference facility is 1,300 Mg in equipment, 517 Mg in structural steel, and 229 Mg in rebar. In addition, the reference uranium hexafluoride production facility was estimated to have about 1800 m^3 of concrete in its floor. Approximately 13.9 percent of the equipment is anticipated to be free of radioactivity, 28.7 percent would have an average radioactivity level (after cleaning) of about 0.02 Bq/g (0.6 pCi/g), and 45.2 percent would have an average radioactivity level (after cleaning) of about 181 Bq/g (4,890 pCi/g).

Seven uranium fuel fabrication facilities, with annual capacities ranging from 400 to 1,200 MTU/yr, are currently licensed to operate by the NRC, and authorization to construct a mixed oxide (MOX) fuel fabrication facility has been requested of the NRC. The mass of steel in the reference facility (1,200 MTU/yr capacity) includes 2,430 Mg in equipment, 1,710 Mg in structural steel, and 2,490 Mg in rebar. The facility is estimated to have approximately 5,940 m^3 of concrete in its floor. Scaling relationships were developed to adjust actual fabrication plant inventories from those of the reference facility. The principal radionuclides of concern are uranium isotopes and certain daughter products. Approximately 46.7 percent of the equipment is anticipated to be free of radioactivity, and 41.1 percent would have an average radioactivity level (after cleaning) of about 2.57 Bq/g (69.4 pCi/g).

Independent spent fuel storage installations (ISFSI) are complexes designed and constructed for the interim storage of spent nuclear fuel. Currently, the NRC licenses the operation of 14 dry spent fuel storage facilities. In addition, there is a single wet storage facility operated by the General Electric Company in Morris, Illinois. A variety of units have been approved for spent fuel storage including concrete casks, horizontal storage modules (NUHOMS), metal casks, pool (wet) storage, and modular vault dry storage. Concrete casks contain a steel mass of 3.65 Mg per fuel assembly and a rebar mass of 0.35 Mg per PWR fuel assembly. NUHOMS contain a steel mass of 2.1/1.0 Mg per PWR/BWR assembly and a rebar mass of 0.06/0.03 Mg per PWR/BWR assembly. Metal casks contain a steel mass of 3.0 Mg per PWR assembly and a rebar mass (in the basepad) of 0.07 Mg per assembly.

The steel mass contained in the modular vault dry storage facility is estimated to be 468 Mg of rebar and 6.2 Mg of equipment with associated radioactivity. The steel mass in the pool (wet) storage facility is estimated to include 451 Mg of equipment and 77 Mg of rebar.

Concrete casks contain a concrete mass of 8.0 Mg per PWR assembly, including the basepad. NUHOMS contain a concrete mass of 5.7/2.8 Mg per PWR/BWR assembly, including the basepad. Metal casks contain a concrete mass in the basepad of 1.6 Mg per PWR assembly. The mass of concrete contained in the modular vault dry storage facility is estimated to be 11,800 Mg, while the concrete mass in the pool (wet) storage facility is estimated as 1,940 Mg. For both concrete casks and metal casks, Fe-55 and Co-60 are the radionuclides expected to be the nuclides of primary concern.

Non-Power Reactors

The NRC currently licenses the operation of 37 non-power reactors. Non-power reactors (NPR) come in many varieties and forms, with most being either pool-type or tank-type. Non-power reactors are also categorized by fuel type: plate-type fuel, TRIGA (Training, Research, Isotopes, General Atomics), or AGN (Aerojet General Nucleonics). The masses of structural steel and rebar in the reference 1,100 kW non-power reactor are 25.2 and 88 Mg, respectively. The total mass of activated steel and aluminum is 1.6 Mg, while the aluminum reactor vessel weighs 0.9 Mg and steel components with associated radioactivity weigh about 45 Mg. The mass of concrete in the reference non-power reactor is 1,925 Mg, of which about 11 Mg has associated radioactivity. An approach was developed for scaling the reference NPR characteristics to other non-power reactors. This scaling approach is based on the quantities of waste generated during the actual decommissioning of four non-power reactors, ranging in size from 10 to 5,000 kilowatts.

Non-Fuel-Cycle Facilities

The NRC currently has in place about 2,997 licenses for users of nuclear materials other than sealed source users, who are eliminated from consideration here because they are not a significant source of potentially clearable materials. This total includes uranium fuel fabrication facilities and independent spent fuel storage installations which were discussed above.

The remainder of the non-fuel-cycle facilities include primarily medical and medical research facilities, research and development laboratories, nuclear pharmacies, and manufacturers fo sealed sources and radio-labeled compounds. In general, the medical laboratories and nuclear pharmacies handle relatively short-lived radionuclides. The radionuclides used in nuclear medicine (e.g., Tc-99m, I-123, Tl-201, Ga-67, Xe-133, In-111, Rb-82, O-15, C-11, F-18, and N-13) have half-lives on the order of minutes to days. As a result, they do not represent sources of contamination that have implications related to clearance of materials. Medical research facilities and research and development laboratories handle C-14 (5,730 years), H-3 (12.35 years), I-131 (8 days), P-32 (14.29 days), and S-35 (87.44 days). Of these, materials contaminated with the shorter lived radionuclides (I-131, P-32, and S-35) generally are not an issue with respect to clearance. However, material contaminated with C-14 and H-3 are of concern with respect to clearance. Manufacturers of sealed sources handle relatively long-lived radionuclides (e.g., Co-60 (5 years) and Cs-137 (30 years)) and generate materials that are pertinent to clearance.

Large medical facilities account for 1224 of these licensees. The inventory of materials in the reference room under regulatory control is about 5,000 pounds. A large hospital may contain several of these rooms. Scaling this quantity nationally and including Agreement States, it is estimated that from 9,600 to 24,000 tons per year could be subject to clearance. This is an estimate of the total quantity of material that may require licensed disposal from hospitals in the United States <u>if the prohibition option were implemented</u>. Of course, only a very small fraction of this material would actually require licensed disposal under most options because the vast majority of this material is clean.

Research and development laboratories account for 566 NRC licensees. Principal radionuclides include H-3, C-14, Co-60, and Cs-137. A reference laboratory is estimated to contain about 1,000 kg of material that may be subject to clearance. Insufficient information was found in the available literature to extrapolate this information to all 566 NRC licensees (and an unknown number of Agreement State licensees). The NRC currently licenses 56 nuclear pharmacies. Since these facilities typically use short half-life radionuclides (e.g., less than 10 days), cleanup can readily be achieved by allowing decay to occur for about 10 half-lives. Consequently, this sector is not expected to generate significant inventories of potentially clearable materials. Sixty-three facilities are licensed by the NRC to manufacture sealed sources and radio-labeled compounds. Not including the hot cell and fume hoods, and other areas of the facility with prohibitively high radioactivity levels dictating disposal as LLW, the quantity of material in the reference hot cell lab that may be a candidate for clearance is estimated to be about 1700 kg, consisting of a mixture of metal, concrete, and asphalt tile, with an inventory of about $1x10^6$ Bq of Co-60, or an estimated 1 Bq /g (27 pCi/g). As is the case for R&D labs, the individual facilities that make up this category are very diverse.

Site Decommissioning Management Plan Inventories

The NRC is currently remediating or planning to remediate 28 locations under the aegis of its Site Decommissioning Management Plan (SDMP). Much of the material with associated radioactivity is soils, which are not addressed in this report. A separate NRC study on soils is underway. Almost half the sites (12 out of 28) are (or were) involved in metal alloy production with large volumes of soil and slag waste. Seven of the sites are former nuclear fuel production facilities. Reference facilities were used to quantify the potentially clearable materials from these sites. Using the reference facility concept, it is estimated that about 4,100 m^3 of building material will be generated from all 19 facilities, and 84,000 m^3 of slag will be generated from the 12 metal processing sites. Since the building material is assumed to be removed (e.g., scabbled) from structures as part of the cleanup process, this material would probably be disposed of as LLW under most regulatory options. The slag may be reprocessed for metal recovery or used for other commercial applications.

DEPARTMENT OF ENERGY INVENTORIES

Based on a complex-wide data call made in the fall of 2000, DOE estimated that about one million tons of potentially clearable scrap metal was in inventory or expected to be generated from D&D activities through the year 2035. A summary is provided in Table ES-3. The data call included only the listed metals since it was intended to support a DOE feasibility study on building a captive steel melting facility.

Table ES-3. Recent DOE Estimates of Potentially Clearable Scrap Metals Through 2035

Material	Minimum Estimate (tons)
Carbon Steel & Iron	827,000
Stainless Steel	153,000
Nickel	31,200
Totals	1,010,000

About 57 percent of this scrap metal will be generated from D&D of the gaseous diffusion plants at Oak Ridge (K-25), Portsmouth, and Paducah. Using limited surface radioactivity data from the Small Scale Recycle Project at K-25, the authors of this report estimate ferrous metals clearable at various surface concentration limits or total radioactivity limits, with the following results:

Cumulative Tons	Cumulative Curies
124,000	0.1
330,000	0.6
422,000	3

Similar estimates were also made for copper and aluminum from the diffusion plants.

Insufficient information was available on the amounts and kinds of radioactivity at other DOE installations to quantitatively characterize radioactivity levels of the inventory.

DEPARTMENT OF DEFENSE INVENTORIES

Most Department of Defense (DOD) users of nuclear materials are licensed by the U.S. Nuclear Regulatory Commission. DOD licensees vary widely in function and size and include hospitals, laboratories, R&D facilities, proving grounds, bombing and gunnery practice ranges, nuclear reactors, weapons manufacturing, and storage facilities. The U.S. Army holds 96 non-fuel-cycle materials licenses from the NRC. The U.S. Navy and the U.S. Air Force each hold Master Materials Licenses from the NRC, and each organization issues permits under its master license. Currently, the Navy has about 139 permits in place, with about 60 percent involving sealed source users. The Air Force has 376 permits in place, with about 89 percent involving sealed source users. Details are provided in Table ES-4.

Table ES-4. Distribution of NRC Licenses/Permits Within DOD

Organization	Total NRC Licenses/ Permits	Sealed Source Users	Medical	R&D	Source Material	Other
U.S. Army	96	35	14	20	20	7
U.S. Navy	139	84	17	7	3	28
U.S. Air Force	376	335	12	6	12	11

Potentially clearable inventories for those applications involving non-sealed sources have been adequately captured by the NRC non-fuel-cycle materials licensees discussed above.

The Naval Nuclear Propulsion Program, which is not licensed by the NRC, currently disposes of large quantities of scrap metal from decommissioned nuclear-powered submarines and cruisers. By 1994, the Navy had more than 100 nuclear-powered vessels in operation. Recycling a typical submarine generates about 2,500,000 pounds of HY-80 steel, 600,000 pounds of carbon steel, 20,000 pounds of sheet metal, 110,000 pounds of stainless steel, 8,000 pounds of galvanized steel, 85,000 pounds of aluminum, 250,000 pounds of brass/bronze, 150,000 pounds of monel, 90,000 pounds of copper, 6,500 pounds of zinc, and up to 1,800,000 pounds of lead as saleable scrap metal.

All three military services have used depleted uranium (DU) ammunition, but the Navy has dropped its use and the Air Force is phasing it out. The Abrams Main Battle Tank uses DU penetrators as one type of ammunition for its 120-mm gun. This use can create low levels of surface radioactivity which are readily removable. These tanks also use steel- encased depleted uranium armor. When the armor becomes defective or damaged, it is disposed of at the Nevada Test Site due to its classified characteristics.

GENERATORS OF TECHNOLOGICALLY ENHANCED NATURALLY OCCURRING RADIOACTIVE MATERIALS

Equipment and scrap metals with associated TENORM may be generated when handling or processing mineral and metallic ores or performing industrial processes using feedstocks containing natural radioactivity. Depending on the process, the radioactivity present in initial material or feedstock can become concentrated, thereby resulting in radioactivity levels that may be orders of magnitude higher than those of the source material. The following industry sectors that handle or process TENORM and are believed to generate scrap metals and other materials with associated radioactivity were evaluated:

- Petroleum production
- Uranium mining
- Phosphate and phosphate fertilizers
- Ash from fossil fuel combustion
- Drinking water treatment
- Metal mining and processing
- Geothermal energy production

NORM and TENORM are not regulated by the NRC, but may be regulated by certain Agreement States.

Petroleum Production

Scale with associated TENORM is frequently found in oil and gas production equipment. Estimates of the amount of equipment and piping with associated radioactivity generated annually by this industry sector and the associated radioactivity levels are summarized in Table ES-5.

In this table, the residue concentration is the concentration of the radionuclide in the scale on the equipment and the effective steel concentration is the concentration of the radionuclide that would result in the steel if the readionuclide in the scale was distributed throughout the mass of the steel.

Table ES-5. Radioactivity Concentrations in Steel Scrap from the Oil and Gas Sector

Metal//Tonnage/ Radionuclide	Residue Conc. (pCi/g)	Surface Conc. (pCi/cm^2)	Inventory (mCi)	Effective Steel Conc. (pCi/g)	Effective Steel Conc. (Bq/g)
Piping/Tubulars 100,000 t Ra-226					
Low:	260	28	290	2.9	0.11
High:	640	68	710	7.1	0.26
Average:	420	44	470	4.7	0.17
Tanks/Vessels 20,000 t Ra-226					
Low:	13	1.4	0.57	0.028	0.0010
High:	32	3.4	1.40	0.069	0.0026
Average:	21	2.2	0.90	0.045	0.0017
Gas Lines 10,000 t Pb-210/Po-210					
Low:	27	1.0	64	6.4	0.24
High:	240	9.1	570	57	2.1
Average:	80	3.0	190	19	0.70

Other Industry Sectors

While information on the kinds of radioactivity and, in some cases, the levels of radioactivity were developed during the literature search of the other industry sectors listed above, no information was found on the types and quantities of materials affected by TENORM radioactivity.

CONCLUSIONS

The literature offers substantial information on inventories of kinds and quantities of potentially clearable materials and associated kinds and quantities of radioactivity at NRC-licensed facilities. Inventory information from facilities that are not licensed by the NRC, but may release materials and equipment with very low levels of radioactivity into general commerce, is much sparser than the information for licensed facilities. The information contained in this report will form the basis for the continuing development of a technical information base necessary to support a Commission policy decision in this area.

1.0 INTRODUCTION

1.1 Background

On June 30, 1999, the U.S. Nuclear Regulatory Commission (NRC) published an Issues Paper indicating that the NRC was examining its approach for "release of solid material" (64 FR 35090). The NRC does not have a regulation that uniformly addresses the release of solid materials. The term "release of solid material," as used in the Issues Paper, is a general term applied to solid materials and equipment that leave the radiological control of the licensee. It includes releases with and without further restrictions or controls. A subset of all kinds of releases are those releases of materials and equipment with no subsequent radiological control. This latter kind of release is internationally called, "clearance." Solid materials and equipment that undergo clearance are said to be "cleared." Thus, from a regulatory point of view, cleared materials and equipment may be treated with respect to their radiological properties as ordinary materials and equipment. A detailed analysis of individual doses expected from the release (clearance) of ferrous metals, aluminum, copper, and concrete was presented in Draft NUREG-1640, *Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities* (NRC 1999). Draft NUREG-1640 did not consider collective doses or costs and benefits associated with alternative approaches to clearance.

Subsequently, on August 18, 2000, the Commission directed the staff to:

- Defer a final decision on whether to proceed with rulemaking,
- Proceed with a National Academies study on possible alternatives for release of slightly contaminated materials,
- Continue the development of technical information base necessary to support a Commission policy decision in this area, and
- Stay informed of international initiatives in this area, related U.S. Environmental Protection Agency (EPA) and Department of State (DOS) activities, and potential for import and trade issues.

The regulatory alternatives in the Issues Paper provide benchmarks for the continued development of the technical information. They are:

- No action
- Prohibiting release of material that had been in an area in a licensed facility where radioactive material was used or stored
- Restricting release to only certain authorized uses
- Permitting material to be released from license control if it meets certain dose-based criteria; i.e., 100 μSv/a (10 mrem/yr) clearance level, 10 μSv/a (1 mrem/yr) clearance level, 1 μSv/a (0.1 mrem/yr) clearance level, or zero above background

This report addresses a portion of the process of developing additional technical information to support any Commission policy decisions regarding release of solid materials. It is a review of available published sources of information on the amounts and kinds of radionuclides associated with potentially clearable materials and equipment. The report seeks to identify the form and composition of solid materials generated at nuclear facilities both during normal operations and during decommissioning, the tonnages generated, and the kinds and amounts of radioactivity that are associated with each category of material as appropriate for the dose assessments and costs estimates. The report focuses on sources of information regarding materials and equipment with the potential for clearance by NRC and Agreement State licensees. Broader considerations are also recognized. Since generators of materials and equipment with very low levels of associated radioactivity, such as the U.S. Department of Energy (DOE), other industries, and the U.S. Department of Defense (DOD), also may be affected by the proposed NRC actions, these sources are also considered. This report highlights four categories of materials -- ferrous metals (carbon steel, stainless steel, cast iron), aluminum and its alloys, copper and its alloys, and concrete. Other materials are addressed in a more limited way.

This report is the first step in an iterative process of developing a database for use in collective dose and cost/benefit analyses associated with the various alternatives. The next step is to identify data gaps in the information available in the literature, which can impede the required analyses, and propose means to fill the gaps. The final step in the process is to take appropriate actions to address the gaps and complete the inventory documentation.

To support the analyses, information is needed regarding:

- The types of equipment and volumes of the various types of materials that may be affected by the alternatives,
- The quantities of materials: carbon steel, stainless steel, copper, aluminium, concrete, and other materials,
- The radionuclide composition and concentration distributions of the potentially affected equipment and material, and
- The time when disposition decisions will be made and implemented.

This study does not include soils which are covered under a separate NRC investigation.

The NRC has established regulations at 10 CFR 20 (Subpart E) for restricted and unrestricted use of structures and lands associated with decommissioned facilities after license termination. However, the NRC currently addresses requests by licensees for release of solid materials on a case-by-case basis, using existing regulatory guidance, license conditions, NRC Branch Technical Positions, and the like, to make a decision.

Some solid materials considered for release from NRC-licensed facilities may have no radioactive contamination, some may have surface contamination, and some may have volumetric contamination. Volumetric contamination can occur when radioactive material is mixed into soils, when concrete or metals are exposed to neutron bombardment which causes activation below the surface, when surface-contaminated concrete is "rubble-ized," or when surface contaminated metals are remelted. The radioactivity levels will depend on the type of the licensed facility and the location of the solid material within the facility. Many facilities are licensed for the use of sealed sources where no contamination of solid materials is expected. Included in this category of licensees are various small research and development laboratories and industrial users of special nuclear measuring devices. A large group of non-fuel cycle licensees use short-lived isotopes with half lives ranging from minutes to days. Because of the nature of the short half-life radionuclides, these licensees do not represent sources of materials that would have significant implications in the analysis of clearance alternatives. A smaller, but very significant, segment of the licensed community, including nuclear power plants, nuclear fuel cycle facilities, certain medical institutions and research facilities, and manufacturers of sealed sources, will have a range of radioactivity levels within the facility depending on the nature of the operations.

1.2 Report Outline

The report is a review of available published sources of information about the amount and kinds of radionuclides associated with potentially clearable materials and equipment. The report seeks to identify the quality and type of solid materials, the relevant tonnages, and the kinds and amounts of radioactivity that are associated with each category of material as appropriate for the dose assessments and costs estimates.

Chapter 2 of this report discusses inventories associated with NRC and Agreement State Licensees. Inventories expected to be generated from commercial nuclear power reactors, non-power reactors, nuclear fuel cycle facilities (e.g., fuel fabrication plants, UF_6 conversion plants, independent spent fuel storage facilities, uranium mills), and materials licensees (e.g., sealed source manufacturers, R&D laboratories, hospitals) are developed to the extent that information on quantities of materials and kinds and amounts of radioactivity are available in the literature. Detailed information is not available for each licensee. However, for some types of facilities, the NRC has previously prepared decommissioning studies on generic or reference facilities which facilitate the inventory analyses. To the degree possible, quantities of material associated with specific radioactivity levels are estimated for each reference facility. In the case of nuclear power reactors, which are the source of the majority of the NRC-licensee inventories of potentially clearable materials, scaling factors are used to account for differences in reactor size.

Year-by-year estimates of the potentially clearable masses and curies can be developed for each material based on current stockpiles, assumed shutdown schedules, and time required for decontamination and decommissioning (D&D). The information included here can be used to develop annual rollouts of potentially clearable materials.

Chapter 3 provides information on DOE inventories. DOE has estimated that current metals in scrap yards within the DOE complex plus metals expected from D&D will amount to about one million tons. Approximately 60 percent of this metal is associated with the gaseous diffusion plants at Oak Ridge (K-25 Site), Portsmouth, and Paducah. D&D is in progress at the K-25 Site. The other two plants have been leased to the U.S. Enrichment Corporation for operation. Although these leased facilities are now under NRC regulation (NRC 2000a), they are included with the other DOE facilities since DOE continues to have management and cleanup responsibilities. Based on limited frequency distributions of surface contamination, estimates are made in Chapter 3 of the incremental quantities of ferrous metals associated with specific surface radioactivity levels. Information in the published literature was insufficient to quantitatively characterize inventories at DOE facilities other than the gaseous diffusion plants.

In January 2000, the Secretary of the Department of Energy placed a moratorium on release into commerce of volumetrically contaminated metals from DOE facilities (DOE 2000). The moratorium was intended to provide time to obtain public input on policy development, to allow the NRC to develop standards, and to allow DOE to examine alternatives to free release. Subsequently, in July 2000, the Secretary suspended release of potentially contaminated scrap metals for recycling from all DOE nuclear facilities (DOE 2000c). The suspension is intended to remain in effect until DOE facilities can demonstrate that materials from recycling contain no detectable contamination. DOE is also exploring recycling of steel into waste containers for use within the complex.

Chapter 4 provides information on the DOD facilities. Most DOD facilities are licensed by the NRC and cover the same spectrum of operations as other materials licensees, e.g., hospitals, research laboratories, sealed source gauges, and irradiators. Naval nuclear reactor facilities are not licensed by the NRC. When nuclear ships are decommissioned, the reactor compartments are cut from the hull, sealed, and shipped for burial at Hanford. The ship hulls are scrapped.

Chapter 5 discusses those commercial industries using or processing materials which contain naturally occurring radioactivity (NORM) and which, because of their operations, create higher concentrations of radioactivity than that associated with an undisturbed natural setting. Concentrated NORM associated with such human activities is described as technically enhanced, naturally occurring radioactivity (TENORM). Radioactive species associated with TENORM are typically uranium, thorium and their equilibrium decay products. Chapter 5 examines the following industry sectors:

- Petroleum production
- Uranium mining
- Phosphate and phosphate fertilizers
- Coal ash from combustion
- Drinking water treatment
- Metal mining and processing
- Geothermal energy production

Data on radioactivity levels and quantities of materials are lacking for virtually all industry sectors. The data quality is of somewhat better for petroleum production. Based on limited information, it is estimated that about 130,000 metric tons of ferrous metals contaminated with TENORM are annually generated by the oil and gas industry.

1.3 Note on Units of Measurement

Since this report is primarily a literature search, units of measurement used by the cited authors are generally retained to insure traceability and transparency of the data. When cited data are manipulated in this report, they are converted to SI units. A common metric for mass is metric tons. Metric tons, tonnes, and megagrams are synonymous. In this report, the abbreviation *t* is used for metric tons. Shorts tons are not abbreviated; rather they are characterized as *tons*. A list of useful conversion factors is included at the front of this report.

2.0 INVENTORY ASSOCIATED WITH NRC AND AGREEMENT STATE LICENSEES

This chapter discusses NRC and Agreement State Licensees. Inventories associated with commercial nuclear power reactors, non-power reactors, nuclear fuel cycle facilities (e.g., fuel fabrication plants, UF_6 conversion plants, independent spent fuel storage facilities, and uranium mills), and non-fuel cycle materials licensees (e.g. sealed source manufacturers, R&D laboratories, hospitals) are developed to the extent that information on quantities of materials and kinds and amounts of radioactivity is available in the literature. Detailed information is not available for each licensee. However, for some types of facilities, the NRC has previously prepared decommissioning studies on generic or reference facilities which facilitate the inventory analyses. To the degree possible, quantities of material associated with specific radioactivity levels are developed for each reference facility. In the case of nuclear power reactors, which are the source of the majority of the NRC-licensee inventories of potentially-clearable materials, scaling factors are developed to account for differences in reactor size. Scaling approaches are also developed for some other categories of licensees.

Decommissioned nuclear power reactors are a major source of potentially clearable materials. A total of 69 PWRs and 35 BWRS are currently licensed to operate by the NRC. A typical PWR contains about 36,000 t of ferrous metals, 700 t of copper, 20 t of aluminum and 180,000 t of concrete. About 10 percent of the ferrous metals is activated or has surficial radioactivity and the balance is clean. About one third of the clean material is rebar in the concrete structures. BWRs typically generate more materials with associated radioactivity because of the nature of the reactor design. It is estimated that about 25 percent of the ferrous metals in a BWR are activated or have surficial activity.

In addition to solid materials expected from decommissioning of nuclear reactors, a variety of materials are removed from each operating site on an on-going basis. Such movement of solid materials may include disposal as low-level waste, clearance for reuse, recycle, or commercial landfill disposal, processing by a nuclear waste broker, or case-by-case disposal under the provisions of 10 CFR Part 20.2002. The NRC has authorized about 35 case-by-case disposals under 10 CFR 20.2002. These case-by-case authorizations have included sand, soils, roofing materials, wood, sewage sludges, resins, pond sediments and the like. The radioactivity varies from a fraction of a millicurie to tens of millicuries of nuclides such as Co-58, Co-60, Cs-134, Cs-137, and Mn-54.

Information is also developed here on the masses of ferrous metals and concrete expected from the decommissioning of fuel cycle facilities including two uranium hexafluoride conversion plants, seven fuel fabrication plants, and 15 independent spent fuel storage installations. Uranium milling operations including both NRC and Agreement State licensees are qualitatively described but these facilities are not expected to be a major source of potentially clearable materials since most are currently undergoing decommissioning and dismantlement will be completed prior to development of any new regulatory approach by the Commission. Information on copper and aluminum from fuel cycle facilities is lacking but quantities of these metals are expected to be small relative to quantities of ferrous metals.

Thirty-seven non-power reactors are currently licensed to operate by the NRC. The masses of structural steel and rebar in the reference 1,100 kW non-power reactor are 25.2 and 88 t, respectively. The total mass of activated steel and aluminum is 1.6 t. Components with surficial radioactivity include the aluminum reactor vessel weighing 0.9 t and various steel components weighing about 45 t. The mass of concrete in the reference non-power reactor is 1,925 t, of which more than 99 percent is clean.

The NRC currently has in place about 2,997 licenses for users of nuclear materials other than sealed source users who are eliminated from consideration here because they are not a significant source of potentially clearable materials. Large medical facilities account for 1,224 of these licensees. The inventory of materials in the reference room under regulatory control in a large hospital is about 2.3 t. Scaling this quantity nationally and including Agreement States, it is estimated that from 8,700 to 22,000 t per year could be subject to clearance. Only a very small fraction of this material would actually require licensed disposal under most options, because the vast majority of this material is clean. Research and development laboratories account for 566 NRC licensees. Principle radionuclides include H-3, C-14, Co-60, and Cs-137. A reference laboratory is estimated to contain about 1 t of material that may be subject to clearance.

Twenty-eight sites are being remediated under the NRC's Site Decommissioning Management Plan (SDMP). At many of these locations, the only issue is soil with elevated levels of radioactivity. In addition, it is estimated that 4,100 m^3 of building materials and 84,000 m^3 of slags will also be generated during clean up operations. Most of the building materials will probably be buried as low-level waste. Some of the slags may be processed for metal recovery.

This chapter reviews the available written sources of information on the amounts and kinds of radionuclides associated with materials and equipment used by NRC and Agreement State licensees.

2.1 Scope of Licensed Community

As reported in the June 2000 *Information Digest* (NRC 2000a), facilities licensed by NRC include:

- Operating commercial nuclear power reactors (as of December 1999) - 104
- Operating non-nuclear power reactors[1]: 37
- Fuel cycle facilities
  - Uranium fuel fabrication plants: 7
  - Uranium hexafluoride production plant: 1

[1] Of these, eight reactors are being decommissioned, and seven reactors have possession-only licenses. Since 1958, 73 licensed non-power reactors have been decommissioned.

- Gaseous diffusion plants: 2
- Uranium mills and leaching operations[2]: 25
- Dry spent fuel storage facilities: 14

- Materials licensees: 5,288

In addition, there are 15,512 materials licenses issued by the Agreement States (NRC 2000a). Some of the NRC materials licenses issued by the NRC to other government agencies (i.e., Department of the Air Force, Department of the Navy, and Department of Agriculture) are Master Materials Licenses. Numerous permits or sub-licenses are issued under these Master Licenses.

The number of licenses cited here was the number as of June 2000; however, this number is in a constant state of flux. In addition, the dates of license termination are changeable, as licenses are terminated early or renewed for an additional period. For example, some commercial nuclear power reactors have received license extensions for an additional 20 years beyond the 40-year span of the initial operating license. Other extensions are pending.

2.2 Commercial Nuclear Power Reactors

This section summarizes current commercial nuclear power reactor decommissioning practices and develops inventories of materials and associated amounts of radioactivity for a reference boiling water reactor (BWR) and a reference pressurized water reactor (PWR). Section 2.2.1 summarizes regulatory decommissioning options. Sections 2.2.2 and 2.2.3 describe a reference PWR and a reference BWR in terms of materials inventory, expected contamination levels, and mix of radionuclides present. Section 2.2.4 provides a list of commercial nuclear power plants, their scheduled shutdown dates, and scaling factors to adjust the materials inventories of the reference reactors described in Sections 2.2.2 and 2.2.3 to other reactor sizes. By combining the information on quantities of materials in the reference reactors with the scaling factors and shutdown dates, one can develop a temporal schedule of materials availability. Section 2.2.5 provides examples of actual inventories of materials at nuclear power plants currently undergoing decommissioning. Section 2.2.6 discusses materials that are cleared from operating nuclear reactors, often on a routine basis, but sometimes based on case-by-case approvals.

[2] Does not include numerous Agreement State licensees.

2.2.1 Decommissioning Policies

With the publication of the NRC's Decommissioning Rule in June 1988 (NRC 1988), owners and/or operators of licensed nuclear power plants were required to prepare and submit plans and cost estimates for decommissioning their facilities to the NRC for review. Decommissioning, as defined in the rule, means to remove nuclear facilities safely from service and to reduce radioactive contamination to a level that permits release of the property for restricted and unrestricted use and termination of the license. The decommissioning rule applies to the site, buildings, and contents and equipment. Several utilities have submitted decommissioning plans to the NRC for review. Selected plans will be discussed in subsequent sections.

Historically, the NRC has established three classifications for the decommissioning of nuclear facilities (NRC 2000c):

- **DECON** is the alternative in which "the equipment, structures, and portions of a facility and site that contain radioactive contaminants are removed or decontaminated to a level that permits termination of the license after cessation of operations."

- **SAFSTOR** is the alternative in which "the facility is placed in a safe, stable condition and maintained in that state until it is subsequently decontaminated and dismantled to levels that permit license termination. During SAFSTOR, a facility is left intact, but the fuel has been removed from the reactor vessel and radioactive liquids are drained from systems and components and then processed. Radioactive decay occurs during the SAFSTOR period, thus reducing the levels of radioactivity in and on the material, and potentially the quantity of material that must be disposed of during decommissioning and dismantlement."

- **ENTOMB** is the alternative in which radioactive structures, systems, and components are encased in a structurally long-lived substance, such as concrete. "The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level that permits termination of the license."

The decommissioning rules were revised in July 1997 to include radiological criteria for license termination. As specified in 10 CFR 20, Subpart E, *Radiological Criteria for License Termination*, a site is considered acceptable for unrestricted use if the residual radioactivity above background results in a total effective dose equivalent (TEDE) to the average member of the critical group that is less than 25 mrem per year. In addition, the residual radioactivity must be reduced to levels that are as low as reasonably achievable (ALARA).

The DECON alternative cannot be implemented immediately after reactor shutdown, because the U.S. Department of Energy has promulgated a requirement that spent nuclear fuel (SNF) must be cooled in the reactor pool for at least five years before it can be placed in dry storage (NRC 1995b). Consequently, DECON cannot be completed until after the pool storage requirement has been met. Activities under the SAFSTOR alternative must be completed within 60 years after shutdown.

Even though the NRC has primary regulatory authority for nuclear power reactor decommissioning, decisions by State agencies may influence the actual disposition of materials. Examples of this will be presented in later sections of this chapter.

2.2.2 Reference Boiling Water Reactor

As of May 2000, there were 35 boiling water nuclear power reactors (BWRs) with NRC operating licenses (NRC 2000a). These reactors are listed in Section 2.2.4 (Table 2-26). In addition, the NRC lists 10 BWRs formerly licensed to operate (Table 2-27); many of these are low power units (i.e., < 200 Mwt).

2.2.2.1 General Description

In the late 1970s and early 1980s, the NRC commissioned a series of studies on the technology and costs of decommissioning several types of nuclear facilities. A generic or reference design was selected for each facility studied. Designs for both a reference PWR and a reference BWR were developed. The reference BWR is based primarily on the description of the 3,320 MWt (1,155 MWe) Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 presented in two NRC reports: *Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station,* Volume 1 and Volume 2 (NRC 1980 and 1980a), and *Revised Analyses of Decommissioning for the Reference Boiling Water Reactor Power Station,* Volume 1 and Volume 2 (NRC 1994a and 1994b).

The primary purpose of the revised analysis published in 1994 was to reevaluate estimated costs and schedules for license termination of the reference BWR. This reevaluation was designed, among other reasons, to reflect the DOE requirement that spent fuel must be cooled in the reactor pool for five years before being placed in dry storage (NRC 1994a). For reactors where DECON was the selected decommissioning alternative, the following schedule was assumed:

- Pre-shutdown planning/engineering and regulatory reviews: 2.5 years
- Plant deactivation, chemical decontamination, removal of reactor pressure vessel internals, and systems layup: 1.2 years
- Safe storage of laid-up plant, spent fuel pool storage operations, preparation for active decommissioning activities: 3.4 years
- Plant dismantlement: 1.7 years

Thus, dismantlement of the reference plant is scheduled to be completed 6.3 years after shutdown.

The reference BWR consists of three principal buildings where radioactive contamination may be present: the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building (NRC 1980, pp. 7-6 to 7-9).

The Reactor Building contains the nuclear steam supply system and its supporting systems. It is constructed of reinforced concrete capped by metal siding and roofing supported by structural steel. The building surrounds the primary containment vessel, which is a free-standing steel pressure vessel. The exterior dimensions of the Reactor Building are approximately 42 m by 53 m in plan, with 70 m above grade and 10.6 m below grade to the bottom of the foundation.

The Turbine Generator Building, which contains the power conversion system equipment and supporting systems, is constructed of reinforced concrete capped by steel-supported metal siding and roofing. This structure is approximately 60 m by 90 m in plan and 42.5 m high.

The Radwaste and Control Building houses, among other systems: the condenser off-gas treatment system, the radioactive liquid and solid waste systems, the condensate demineralizer system, the reactor water cleanup demineralizer system, and the fuel-pool cooling and cleanup demineralizer system. The building is constructed of reinforced concrete, structural steel, and metal siding and roofing. This structure is approximately 64 by 49 m in plan and 32 m in overall height.

Several additional buildings make up the reference BWR complex. These include the Diesel Generator Building, Service Building, Circulating Water Pump House, Spray Pond Complex, Makeup Water Pump House, the yard, and other buildings (i.e., Office Building, Warehouse, Guard House, and Gas Bottle Storage Building). These building are assumed not to be radioactively contaminated.

The authors of NRC 1980 and 1980a assume that the reference BWR is decontaminated prior to dismantlement, but the study gives no credit for any additional potentially clearable materials, because availability of low-level waste disposal site and disposal costs were not significant issues. The authors assumed that decontamination would not permit additional materials to be cleared from regulatory control. All major components, including the turbines and condenser, would be buried as LLW.

The reports on reference BWR do not provide a summary of all the materials in the plant. Consequently, it was necessary for this report to construct such a summary from information scattered throughout the four documents addressing the reference reactor (i.e., NRC 1980 and 1980a, and NRC 1994a and 1994b). The documents contain considerable information on ferrous metals, some information on concrete, limited information on aluminum, and no specific information on copper.

Summary information on the structural materials, including concrete, structural steel, and rebar used in the reference BWR are provided in NRC 1980a (Table C.2-1) and are presented here in Table 2-1. Appendix H of NRC 1994b contains related information. While the data are similar to those in NRC 1980a, significant differences exist. The data from NRC 1980a are used here, since the data set is more complete.

Table 2-1. Estimated Quantities of Structural Materials in the Reference BWR Facilities

Structure	Concrete (m^3)	Rebar (Metric tons)	Structural Steel (Metric tons)
Reactor Building	42,804	8,608	902
Primary Containment	1,225	187	693
Turbine Generator Building	46,672	4,717	742
Radwaste and Control Building	26,697	2,746	372
Diesel Generator Building	2,964	408	0
Service Building	2,359	151	395
Circulating Water Pump House	2,963	321	112
Spray Pond Complex	8,321	789	0
Makeup Water Pump House	1,278	163	0
Yard and Other Buildings	12,825	247	91
Total	147,880	18,351	3,307

Radioactive contamination is assumed to be associated only with the Reactor Building (including primary containment), the Turbine Generator Building, and the Radwaste and Control Building (NRC 1980, Table 7.3-3).

2.2.2.2 Radionuclide Distributions in Reference BWR

Neutron-Activated Components

Estimates of the fractional radioactivity at shutdown associated with neutron-activated stainless steel, carbon steel, and concrete were developed for the reference BWR. These estimates are presented in Table 2-2 (NRC 1980a, Tables 7.4-1, 7.4-2, and 7.4-3). Table 2-2 includes only nuclides contributing >$1x10^{-4}$ to the fractional activity. (The source document provides a more complete listing of all nuclides assumed to be present.)

Table 2-2. Radionuclide Mix in Neutron-Activated Components of Reference BWR at Shutdown

	Fractional Radioactivity		
Radionuclide	**Stainless Steel**	**Carbon Steel**	**Concrete**
P-32		2.51E-3	1.13E-2
S-35			8.46E-4
Cr-51	5.09E-1	4.77E-3	2.89E-3[a]
Ar-37			8.02E-3
Ar-39			1.48E-3
Ca-41			2.06E-4
Ca-45			1.02E-1
Mn-54	2.98E-3	2.89E-2	2.50E-3[a]
Fe-55	3.24E-1	9.13E-1	8.20e-1[a]
Fe-59	9.62E-3	2.57E-2	2.63E-2[a]
Co-58	7.37E-3	4.06E-3	7.81E–4[a]
Co-60	1.18E-1	1.77E-2	1.68E-2[a]
Ni-59	2.23E-4		
Ni-63	3.07E-2	4.71E-3	3.83E-3[a]
Ag-110			4.53E-4
Sm-151			3.39E-4
Eu-152			2.60E-3
Eu-154			3.41E-4

a Due largely to structural steel in the sacrificial shield

Internal Surface Contamination of Equipment and Piping

Activated corrosion products from structural materials in contact with the reactor coolant and fission products from leaking fuel contribute to the presence of radioactivity in reactor coolant streams during plant operation. Although most of these radionuclides are removed through filtration and demineralization by a plant's chemical and volume control systems (letdown cleanup system), a smaller component escapes removal. With time, some of the radionuclides, principally the neutron-activated insoluble corrosion products, tend to deposit on inner surfaces of equipment and piping systems. The metal oxide layer consists primarily of iron, chromium, and nickel, with smaller but radiologically significant quantities of cobalt, manganese, and zinc.

The estimated radionuclide mix for internal surface contamination in a reference BWR is included in Table 2-3 (NRC 1980, Table 7.4-6).

Table 2-3. Radionuclide Inventory for Internal Surface Contamination of Reference BWR at Shutdown

Radionuclide	Fractional Radioactivity
Cr-51	2.1e-2
Mn-54	3.9e-1
Fe-59	2.5e-2
Co-58	9.3e-3
Co-60	4.7e-1
Zn-65	6.1e-3
Zr-95	4.0e-3
Nb-95	4.0e-3
Ru-103	2.3e-3
Ru-106	2.8e-3
Cs-134	1.9e-2
Cs-137	3.4e-2
Ce-141	3.0e-3
Ce-144	8.1e-3

Table 2-4 presents the total fractional radionuclide inventory for three BWR plants decay-corrected to the date the plant was shut down. The average for the three plants is also included, as is the estimated fractional activity for the reference BWR. Considerable variability appears among the three operating reactors and the reference BWR. The reference BWR internal surface contamination measurements were based on a sludge sample analysis given in a 1975 EPRI report (EPRI 404-2) as referenced in NRC 1980. The operating BWR results are for reactors with electrical ratings of 63 to 550 MW(e) operating for 10 to 18 years, as described in NRC 1986. The operating BWR results include only nuclides with half-lives greater than 245 days and do not include highly activated components of the reactor pressure vessel.

Table 2-4. Fractional Radionuclide Inventory for Three BWRs and the Reference BWR for Internal Surfaces at Shutdown

Nuclide	Fractional Activity[a]				
	Humboldt Bay	Dresden 1	Monticello	BWR Average	Reference BWR
Cr-51	---	---	---	---	0.021
Mn-54	0.03	0.009	0.01	0.016	0.39
Fe-55	0.90	0.28	0.01	0.39	0.025
Co-58	---	---	---	---	0.0093
Co-60	0.06	0.46	0.11	0.21	0.47
Ni-59	---	0.0009	---	0.0003	--
Ni-63	0.002	0.05	0.0004	0.017	--
Zn-65	---	0.19	0.84	0.34	0.0061
Sr-90	0.00004	0.00007	0.00002	0.00004	--
Nb-94	<0.00004	<0.00003	<0.001	<0.0004	
Nb-95	---	---	---	---	0.004
Zr-95	---	---	---	---	0.004
Tc-99	3E-06	4E-07	8E-07	<1.4E-06	---
Ru-103	---	---	---	---	0.0023
Ru-106	---	---	---	---	0.0028
I-129	<3E-08	<1E-07	<1E-08	<4E-08	--
Cs-134	---	---	---	---	0.019
Cs-137	0.005	0.0004	0.02	0.008	0.034
Ce-141	---	---	---	---	0.003
Ce-144	---	0.01	---	0.003	0.0081
TRU[b]	0.00005	0.001	0.00008	0.0004	--
Total Ci	596	2350	448		8500

a Source: Abel et al. 1986 for Operating BWRs and NRC 1980 (Table 7.4-6) for Reference BWR

b TRU: Transuranic alpha-emitting radionuclides with half-lives greater than five years, including Pu-238, Pu-239, Pu-240, Am-241, Am-243, and Cm-244

2.2.2.3 Ferrous Metals in Reference BWR

The constructed summary of all ferrous metals in the reference BWR is exhibited in Table 2-5. The information from the reference BWR documentation (NRC 1980 and 1980a, and NRC 1994a and 1994b) was used to develop estimates of the surface area and weights for the individual systems. While the table title indicates it is a summary of ferrous metals (carbon and stainless steels), that description is a simplification. In fact, the masses cited include components such as motors, so that some other metals are implicitly included. However, the fraction of non-ferrous metals in the reference BWR is estimated to be only about 2 percent, so the effect is minor.

Table 2-5. Summary of Ferrous Metals in Reference BWR

Location	System	System Mass (kg)
Reactor Bldg.	Containment Instrument Air	6,996
Reactor Bldg.	Control Rod Drive	219,442
Reactor Bldg.	Equipment Drain Processing	52,631
Reactor Bldg.	Fuel Pool Cooling & Cleanup	343,121
Reactor Bldg.	High Pressure Core Spray	46,295
Reactor Bldg.	HVAC Components	58,162
Reactor Bldg.	Low Pressure Core Spray	20,799
Reactor Bldg.	Main Steam[a]	3,894,143
Reactor Bldg.	Main Steam Leakage Control	2,125
Reactor Bldg.	Misc. Items from Partial System	67,399
Reactor Bldg.	Closed Cooling Water	42,321
Reactor Bldg.	Equipment & Floor Drains	9,943
Reactor Bldg.	Reactor Core Isolation Cooling	21,554
Reactor Bldg.	Reactor Water Cleanup	47,520
Reactor Bldg.	Residual Heat Removal	306,401
Reactor Bldg.	Misc. Drains	7,650
Reactor Bldg.	Piping (non-primary containment)	348,122
Reactor Bldg.	Piping (primary containment)	838,180
Reactor Bldg.	Primary Containment	693,000
Reactor Bldg.	Steam Dryer	41,277
Reactor Bldg.	Steam Separator	33,929
Reactor Bldg.	Steam Separator	15,436
Reactor Bldg.	Core Shroud	37,694
Reactor Bldg.	Top Fuel Guide	5,429
Reactor Bldg.	Jet Pumps	5,683
Reactor Bldg.	Support Ring	2,359

Table 2-5. Summary of Ferrous Metals in Reference BWR (continued)

Location	System	System Mass (kg)
Reactor Bldg.	Core Support Plate	9,298
Reactor Bldg.	Orificed Fuel Supports	5,366
Reactor Bldg.	CR Guides	9,549
Reactor Bldg.	Limiters, Housings	43,967
Reactor Bldg.	Shroud Support	44,906
Reactor Bldg.	RPV Upper Flange	55,221
Reactor Bldg.	RPV Upper Head	29,602
Reactor Bldg.	RPV Lower Flange	54,514
Reactor Bldg.	Non-Activated RPV Wall	255,111
Reactor Bldg.	Activated RPV Wall	161,210
Reactor Bldg.	Lower Head	110,922
Reactor Bldg.	Nozzles	59,139
Reactor Bldg.	Studs & Nuts	31,481
Reactor Bldg.	Skirt, Base Ring, & Collar	61,350
Reactor Bldg.	Sac. Shield, steel only	77,364
Radwaste Bldg.	Chemical Waste Processing	59,803
Radwaste Bldg.	Condensate Demineralizers	89,783
Radwaste Bldg.	HVAC Components	102,751
Radwaste Bldg.	Radioactive Floor Drain Processing	31,237
Radwaste Bldg.	Building Drains	3,897
Radwaste Bldg.	Standby Gas Treatment	40,588
Radwaste Bldg.	Sample Systems	1,148
Radwaste Bldg.	Piping	165,927
Turbine Bldg.	Feed & Condensate	1,592,118
Turbine Bldg.	Extraction Steam	115,710
Turbine Bldg.	Heater Vents & Drains	198,775
Turbine Bldg.	HVAC Components	76,393
Turbine Bldg.	Offgas (Augmented)	64,483
Turbine Bldg.	Recirculation	120,536
Turbine Bldg.	Building Drains	4,730
Turbine Bldg.	Piping	1,176,074
Total Plant	Small Pipe Hangers[b]	282,366
Total Plant	Large Pipe Hangers[b]	802,872
Total Plant	Rebar	18,351,000
Total Plant	Structural Steel	2,614,000
GRAND TOTAL		34,070,549

a Some Main Steam System components in Turbine Building
b Source: NRC 1996

Masses for most of the systems in Table 2-5 were obtained from Appendix A (Section 5.1.1) of EPA's *Technical Support Document on Scrap Metal Recycling* (EPA 1997a). Tables A5-2A, A5-2B, and A5-2C in EPA 1997a were developed primarily from Table C.5 of NRC 1994b by converting the mass data to kilograms and summing for each major subsystem. Some of the tables in EPA 1997a have been adjusted based on the current review. Details are included in Appendix A of this report.

Table 2-5 shows that about 34,000 metric tons of components and systems composed of ferrous metals have been identified in the reference BWR.

Neutron-Activated Ferrous Metals

The masses of the neutron-activated ferrous metal (carbon and stainless steel) components and the associated radioactivity are summarized in Table 2-6.

Table 2-6. Neutron-Activated Ferrous Metal Components in Reference BWR

Component	Mass (kg)[a]	Radioactivity	
		(Ci)[b]	(Bq)
Steam Separator	15,436	9.60e+3	3.56e+14
Core Shroud	37,694	6.30e+6	2.33e+17
Top Fuel Guide	5,429	3.0e+04	1.11e+15
Jet Pumps	5,683	2.00e+4	7.41e+14
Core Support Plate	9,298	6.50e+2	2.41e+13
Orifice Fuel Supports	5,366	7.01e+2	2.60e+13
Control Rod Guides	9,549	9.47e+1	3.51e+12
Activated RPV Wall/Cladding	161,210	2.16e+3	8.00e+13
Sacrificial Shield (steel only)	77,364	1.66e+2	6.15e+12
TOTALS	327,028	6.36e+6	2.36e+17

a Source: NRC 1994b, Table E.4
b Source: NRC 1980, Table 7.4-4

Table 2-6 excludes those portions of the reactor pressure vessel and internals that are assumed not to be activated, but to have surface contamination.

Ferrous Metal Equipment and Piping with Internal Contamination

Internal surface contamination will be present on piping and components in the reactor system, particularly those exposed to reactor cooling water. The methodology used to develop the contamination estimates for these items is as follows:

- The internal surface area of components expected to have internal surface contamination was estimated.
- Radioactive deposition levels based on operational exposure were developed.
- Five deposition levels, ranging from 5 Ci/m^2 for tanks and equipment containing concentrated waste to 5x10-4 Ci/m^2 for the turbine, were selected (NRC 1980a, Vol. 2, Table E.2-6).
- Total deposited radioactivity was calculated as the product of the surface area and the assumed deposition rate.

A summary of identified components with internal surface contamination is included as Table 2-7. The total mass of the components in Table 2-7 is about 3,000 metric tons less than the amount in Table 2-5 for the systems with identified internal contamination due to the inability to associate mass with activity for all listed components and systems.

Table 2-7. Mass and Activity of Internally Contaminated Components in Reference BWR

System	Component	Component Mass (kg)	Activity (Ci)*
Equipment Drain Processing			
	Spent Resin Tank (in Radwaste Bldg.)	657	65
	Waste Collector Tank (in Radwaste Bldg.)	10,229	5
	Waste Surge Tank (in Radwaste Bldg.)	18,282	950
	Waste Sample Tanks (in Radwaste Bldg.)	13,920	8
Fuel Pool Cooling & Cleanup			
	Fuel Pool HX	4,076	40
	Skimmer Surge Tanks	10,708	50
	Fuel pool, reactor well, dryer & separator pool		70
Main Steam			
	Turbine Bypass Valve Assembly	5,266	0.75
	Moisture Separator Reheaters	416,772	90
	Steam Evaporator	26,944	10
	Gland Steam Condenser	1,816	17
	Main Condenser	1,570,000	390
	Main Turbine	1,338,372	1.3
Closed Cooling Water			
	RBCCW Heat Exchangers	22,380	90
Reactor Water Cleanup			
	RWCU Non-regenerative HX	8,172	60
	RWCU Regenerative HX	12,394	90
	Cleanup Phase Separator Tanks (in Radwaste Bldg.)	4,086	340
Residual Heat Removal			
	RHR HX	58,380	540
Pressure Vessel and Internals			
	Steam Dryer	41,277	25
	Steam Separator	33,929	34
	Limiters, Housings	43,967	3
	Shroud Support	44,906	76
	RPV Upper Flange	55,221	11
	RPV Upper Head	29,602	8
	RPV Lower Flange	54,514	11
	Non-Act. RPV Wall	255,111	39
	Lower Head	110,922	16

Table 2-7. Mass and Activity of Internally Contaminated Components in Reference BWR (continued)

System	Component	Component Mass (kg)	Activity (Ci)*
	Nozzles	59,139	5
	Studs & Nuts	31,481	7
	Skirt, Base Ring, & Collar	61,350	5
	Support Ring	2,359	700
Chemical Waste Processing			
	Distillate Tanks	10,048	7.5
	Decon Solution Concentrator Tanks	1,422	120
	Detergent Drain Tanks	3,668	16
	Chemical Waste Tanks	10,048	7.5
	Decon Solution Concentrators	6,810	95
Condensate Demineralizers			
	Condensate Backwash Receiver Tank	6,912	420
	Condensate Phase Separator Tanks	6,356	900
Radioactive Floor Drain Processing			
	Floor Drain Collector Tank	10,229	5.5
	Floor Drain Sample Tank	6,960	3.9
	Waste Sludge Phase Separation Tank	5,490	300
Feed & Condensate			
	Condensate Storage Tanks	100,950	80
	High & Low Pressure Feed Water Heaters	883,896	455
	Air Ejector Condensers	13,228	80
Heater Vents & Drains			
	Moisture Separator Drain Tanks	3,430	0.15
	Reheater Drain Tanks	29,632	42
	Steam Evaporator Drain Tanks	1,796	0.5
Piping			
	Piping <60 mm o.d	68,289	220
	Piping >60 mm o.d	2,460,014	1,980
SUMMARY		7,975,410	8,490

* See Table 2-3 for radionuclide mix.

Ferrous Metals with External Contamination

Some radionuclide releases associated with normal plant operation are expected over the operating life of the reference BWR. Total estimated contamination on external surfaces at shutdown is about 114 Ci distributed as follows:

- Reactor Building: 74 Ci
- Turbine Generator Building: 4.4 Ci
- Radwaste and Control Building: 36 Ci

Details on the distribution of the external contamination are presented in Table 2-8. The table shows that most of the systems with internal contamination (see Table 2-7) also have external contamination, and that the magnitude of the external contamination is small compared with that of the internal contamination (i.e., about 1.3 percent of the internal).

Table 2-8. Deposited External Radioactivity on Reference BWR

System	Deposited External Radioactivity (Ci)
Control Rod Drive	0.45
Equipment Drain Processing	0.45
Fuel Pool Cooling & Cleanup	11.4
High Pressure Core Spray	0.27
HVAC Components	0.04
Low Pressure Core Spray	0.035
Main Steam	1.17
Closed Cooling Water	0.03
Equipment & Floor Drains	1.8
Reactor Core Isolation Cooling	0.038
Reactor Water Cleanup	7.78
Residual Heat Removal	0.42
Primary Containment	55
Chemical Waste Processing	12.01
Condensate Demineralizers	9
Radioactive Floor Drain Processing	4.83
Building Drains	4.61
Standby Gas Treatment	0.1
Feed & Condensate	1.86
Heater Vents & Drains	0.047

Table 2-8. Deposited External Radioactivity on Reference BWR (continued)

System	Deposited External Radioactivity (Ci)
HVAC Components	0.097
Offgas (Augmented)	0.45
Building Drains	1.47
Total	113.36

Ferrous Metals Assumed to Have No Significant Contamination

A large fraction of the total ferrous metals in the reference BWR are assumed to have no detectable levels of contamination. These metals are classified according to MARSSIM screening classifications (NRC 1997a) in Table 2-9 based on the authors' engineering judgment.

Table 2-9. MARSSIM Categorization of Rebar and Structural Steel in Reference BWR

Facility	Rebar		Structural Steel	
	Quantity (Mg)	Condition*	Quantity (Mg)	Condition*
Reactor Bldg.	8,608	Class 3	902	Class 2
Primary Containment	187	Class 2	693	Class 2
Turbine Generator Bldg.	4,717	Class 3	742	Class 3
Radwaste & Control Bldg.	2,746	Class 3	372	Class 2
Diesel Generator Bldg.	408	Nonimpacted	0	Nonimpacted
Service Bldg.	151	Nonimpacted	395	Nonimpacted
Circ. Water Pump House	321	Nonimpacted	112	Nonimpacted
Spray Pond Complex	789	Nonimpacted	0	Nonimpacted
Makeup Water Pump House	163	Nonimpacted	0	Nonimpacted
Yard & Other Bldg.	247	Nonimpacted	91	Nonimpacted
TOTAL	18,351		3,307	

* Assumed MARSSIM Categorization

2.2.2.4 Concrete in the Reference BWR

As indicated in Table 2-1, the reference BWR contains about 148,000 m^3 of concrete. Assuming a density of 2400 kg/m^3, the concrete mass would be 355,000 t. Of this material, 79.3 percent is associated with the Reactor Building (including primary containment), the Turbine Building, and the Radwaste and Control Building, with the balance being distributed over support buildings with no assumed radioactive contamination.

After shutdown, the reference BWR studies assume that all concrete surfaces (except those expected to be neutron-activated) will be vacuumed to remove any loose radioactive debris. The surfaces will surveyed for significant radioactivity levels. Contaminated areas will then be washed using high-pressure water and resurveyed to identify areas still contaminated. Contaminated areas will then be scabbled to a nominal depth of one inch to remove the remaining surface contamination. Estimated concrete volumes removed by scabbling are as follows (NRC 1994a, Table 3.20):

- Reactor Building: 36.9 m^3
- Turbine Generator Building: 3.50 m^3
- Radwaste and Control Building: 11.0 m^3

In the original analysis of the reference BWR, it was assumed that about 700 m^3 of concrete rubble would be generated (NRC 1980, Table 7.3-3). Detailed support for the original estimates is included in Appendix D of NRC 1980a. The original study assumed that concrete would be scabbled to a depth of two inches (0.051 m). However, this factor alone does not account for the large difference between the two studies. The contaminated surface areas assumed in NRC 1994a are substantially lower than those in NRC 1980. Comparisons of the original study and the revised study are summarized in Table 2-10.

Table 2-10. Comparison of Estimates of Surface-Contaminated Concrete

Location	Scabbled Volume (m^3)	
	Scabble Depth = 51 mm (NRC 1980)	Scabble Depth = 25.4 mm (NRC 1994a)
Reactor Building*	360.3	36.9
Turbine Generator Building	105.8	3.50
Radwaste & Control Building	203.4	11.0
Total	699.5	82.9

* Including primary containment

The neutron-activated reinforced concrete in the sacrificial shield (biological shield) has an estimated volume of 73.3 m^3 (NRC 1980a, Table E.1-6). This is a small fraction of the total concrete in the Primary Containment (1,225 m^3, see Table 2-1). The radioactivity associated with this activated mass has been accounted for in Table 2-6 above.

Because of the nature and the levels of contamination, all of these scabbled or neutron-activated materials will be buried as LLW. However, most of the concrete is potentially clearable for recycle or burial as ordinary waste.

2.2.2.5 Aluminum in the Reference BWR

The only aluminum components specifically identified in the reference BWR are piping (NRC 1980a, Tables C.3-7, C.3-9, and C.3-10). The mass and associated radioactivity for this piping are summarized in Table 2-11.

Table 2-11. Estimated Mass and Radioactivity Levels Associated with Aluminum Piping in Reference BWR

Quantity	Outside Diameter (mm)		
	<60	73 to 254	305 to 406
Mass in Reactor Building (kg)	1,020	--	--
Mass in Turbine Building (kg)	1,712	23,449	12,716
Mass in Radwaste Building (kg)	288	17,841	930
Total Mass (kg)	3,020	41,290	13,646
Total Radioactivity (Ci)[a]	0.4	3.54	0.7

a Source: NRC 1980a, Table E.2-5

This table indicates that about 4.6 Ci of radioactivity are associated with about 58,000 kg of aluminum piping.

2.2.2.6 Contamination Summary

Information on masses of materials and their contamination status are summarized in Table 2-12. The bulk of the materials are described as “clean.” Within the context of the reference BWR, the term “clean” carries with it a certain subjectivity. Materials designated as “clean” have no radioactivity specifically ascribed to them in the cited NUREG documents. The values in Table 2-12 were obtained by subtracting activated and contaminated masses from the total masses to estimate the quantities of clean materials. The clean estimates may be overstated if some systems with contamination were not specifically listed as such in the source reports. In addition, “clean” is not precisely defined; presumably, it could range from zero up to the surface contamination levels defined in Regulatory Guide 1.86.

Table 2-12. Mass Summary for Reference BWR

Contamination	Material Mass (t)		
	Concrete[a]	**Ferrous Metals**	**Aluminum**
Activated	180	330	N/A
Contaminated	200	8,000	58
Clean	354,600	26,000	N/A
Total	355,000	34,300	N/A

N/A - not available
a Based on concrete density of 2,400Kg/m^3

The lowest contamination level of any of the contaminated ferrous metal items in Table 2-12 is 1,000 pCi/g (37 Bq/g) for the main turbine (see Table 2-7).

2.2.3 Reference Pressurized Water Reactor

This section develops estimates of masses of materials and associated quantities for a generic (i.e., reference) pressurized water reactor (PWR). As of December 1999, 69 PWRs had NRC operating licenses (NRC 2000a). These reactors are listed in Section 2.2.4 (Table 2-26). In addition, the NRC lists 12 PWRs formerly licensed to operate (see Section 2.2.4, Table 2-27).

This section provides a general physical description of the reference PWR, including a materials breakdown by power plant system, a discussion of the mix of radionuclides contributing to reactor contamination, and a summary of the contamination levels throughout the reference facility.

2.2.3.1 General Description

The reference PWR is based on a combination of literature sources. The quantities of metals are taken from *Estimated Quantities of Materials Contained in a 1000-MW(e) PWR Power Plant* (Bryan and Dudley 1974). This plant used run-of-river cooling, and the design features were those that prevailed in 1971. More detailed information on the plant systems and residual levels of radioactivity are based primarily on description of the 3,500 MWt (1,175 MWe) Trojan Nuclear Plant (TNP) at Rainier, Oregon, operated by Portland General Electric Company (PGE) in two NRC reports: *Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station,* Volume 1 (NRC 1978a) and Volume 2 (NRC 1978b), and *Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station,* Volume 1 (NRC 1995b) and Volume 2 (NRC 1995c).

The principal structures at the reference PWR power station are the Reactor Building, Fuel Building, Auxiliary Building, Control Building, and Turbine Building (NRC 1978b, Appendix A).

The Reactor Building houses the nuclear steam supply system. Since its primary purpose is to provide a leak-tight enclosure for normal as well as accident conditions, it is frequently referred to as the containment building. Major interior structures include the biological shield, pressurizer cubicles, and a steel-lined refueling cavity. Supports for equipment, operating decks, access stairways, grates, and platforms are also part of the containment structure internals. The Reactor Building is in the shape of a right circular cylinder approximately 64 m tall and 22.5 m in diameter. It has a hemispherical dome; a flat base slab with a central cavity and instrumentation tunnel.

The Fuel Building, approximately 27 m tall and 19 by 54 m wide, is a steel-reinforced concrete structure with four floors. This building contains the spent-fuel storage pool and its cooling system, much of the chemical and volume control system, and the solid radioactive waste handling equipment. Major steel structural components include fuel storage racks and liner, support structures for fuel handling, and components, ducts, and piping associated with air conditioning, heating, cooling, and ventilation.

The Auxiliary Building, approximately 30 m tall and lateral dimensions of 19 by 35 m, is a steel and reinforced concrete structure with two floors below grade and four floors above grade. Principal systems contained in the Auxiliary Building include the liquid radioactive waste treatment systems, the filter and ion exchanger vaults, waste gas treatment system, and the ventilation equipment for the Reactor, Fuel, and Auxiliary Buildings.

Other major building structures with substantial inventories of metals include the Control Building and Turbine Building. The principal contents of the Control Building are the reactor control room, as well as process and personnel facilities. The principal systems in the Turbine Building are the turbine generator, condensers, associated power production equipment, steam generator auxiliary pumps, and emergency diesel generator units.

Table 2-13 lists the estimated quantities of the principal metals used to construct a 1,000-MWe PWR facility. According to Bryan and Dudley (1974), this facility also contained 179,681 t (74,970 m^3) of concrete.

Table 2-13. Inventory Estimates of Metals Used to Construct a 1,000-MWe Pressurized Water Reactor Facility

Material	Total Quantity (Metric tons)
Carbon Steel	3.3×10^4
(Rebar)	(9.6×10^3)
(All Other)	(2.3×10^4)
Stainless Steel	2.1×10^3
Galvanized Iron	1.3×10^3
Copper	6.9×10^2
Bronze	2.5×10^1
Brass	1.0×10^1
Aluminum	1.8×10^1

Source: Bryan and Dudley 1974, except rebar which is from UE&C 1972.

A detailed description of the location of the various materials within the 1000 MWe PWR is included in Table 2-14.

Other materials identified in constructing the 1000 MWe PWR included 962 t of insulation and 17,681 m^3 of paint.

As noted, the 1000 MWe PWR analyzed by Bryan and Dudley (1974) contained 74,970 m^3 of concrete. This quantity is similar to that for the reference PWR detailed by the NRC (NRC 1995c, Appendix L). Information from the two sources is compared in Table 2-15. The data in column 3 were obtained from mass estimates in Table 2-14 assuming a density of 2.4 Mg/m^3.

Table 2-14. Breakdown of Materials Used in PWR Plant Structures
and Reactor Systems (Metric Tons)

System	Carbon Steel	Stainless Steel	Galvanized Iron	Copper	Inconel	Lead	Bronze	Aluminum	Brass	Nickel	Concrete	Silver
Structures/Site	**16519.3**	**28.6**	**814.2**	**33.1**	**0**	**33.1**	**0.2**	**1.2**	**2.9**	**0.1**	**146472.**	**0.1**
Site Improvements	1692.9	0.0	17.9	1.5	0.0	0.7	0.0	0.1	0.0	0.0	4887	0.0
Reactor Building	7264.2	5.7	301.2	9.3	0.0	0.0	0.0	0.1	0.3	0.0	54329	0.0
Turbine Building	3641.2	0.0	196.4	1.6	0.0	0.0	0.1	0.8	1.4	0.0	15931	0.0
Intake/Discharge	333.7	0.0	3.6	0.2	0.0	0.0	0.0	0.0	0.0	0.0	13215	0.0
Reactor Auxiliaries*	1358.7	0.0	109.8	0.8	0.0	0.0	0.0	0.0	0.2	0.0	33876	0.0
Fuel Storage	364.6	21.1	43.4	0.3	0.0	0.0	0.0	0.1	0.1	0.0	7163	0.0
Miscellaneous Bldgs.	1864	1.8	141.9	19.4	0.0	32.4	0.1	0.1	0.9	0.1	17071	0.1
Reactor Plant Equipment	**3444.9**	**1154.6**	**5.5**	**50.4**	**124.1**	**4.5**	**0.5**	**5.2**	**0**	**0**	**981**	**0**
Reactor Equipment	430.0	275.1	0.0	6.8	124.1	0.0	0.0	0.0	0.0	0.0	136	0.0
Main Heat Trans. System	1686.5	202.5	1.6	9.8	0.0	0.0	0.0	0.0	0.0	0.0	732	0.0
Safeguards Cool. System	274.2	199.1	1.1	2.9	0.0	0.0	0.1	0.0	0.0	0.0	0.0	0.0
Radwaste System	35.2	31.9	0.8	0.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Fuel Handling System	82.0	67.0	0.3	0.2	0.0	0.0	0.0	0.0	0.0	0.0	13	0.0
Other Reactor Equipment	823.5	230.3	1.7	1.5	0.0	4.5	0.4	0.0	0.0	0.0	134.0	0.0
Instrumentation & Control	113.5	148.7	0.0	29.0	0.0	0.0	0.0	5.2	0.0	0.0	0.0	0.0
Turbine Plant Equipment	**10958.3**	**883.2**	**4.7**	**51.4**	**0.0**	**0.0**	**21.5**	**1.2**	**6.9**	**0.0**	**30506.0**	**0.0**
Turbine-Generator	4138.6	129.9	0.5	35.2	0.0	0.0	19.7	0.0	0.0	0.0	11353	0.0
Heat Rejection Systems	2501.1	9.1	2.2	3.0	0.0	0.0	0.7	0.0	0.4	0.0	15143	0.0
Condensing Systems	1359.8	392.3	0.6	1.3	0.0	0.0	0.3	0.0	1.5	0.0	1282	0.0
Feed-Heating System	1367.7	221.2	0.5	1.2	0.0	0.0	0.3	0.0	3.9	0.0	110	0.0
Other Equipment	1541.3	89.4	0.9	0.7	0.0	0.0	0.5	0.0	1.1	0.0	2618	0.0
Instrumentation & Control	49.8	41.3	0.0	10.0	0.0	0.0	0.0	1.2	0.0	0.0	0.0	0.0

Table 2-14. Breakdown of Materials Used in PWR Plant Structures
and Reactor Systems (Metric Tons) (continued)

System	Carbon Steel	Stainless Steel	Galvanized Iron	Copper	Inconel	Lead	Bronze	Aluminum	Brass	Nickel	Concrete	Silver
Electric Plant Equipment	**965.5**	**0.0**	**431**	**556.5**	**0.0**	**6.8**	**2.5**	**4.1**	**0.0**	**0.6**	**1263**	**0.4**
Switchgear	30.4	0.0	1.4	2.8	0.0	0.0	0.7	0.0	0.0	0.0	0.0	0.3
Station Service Equip.	654.1	0.0	8.5	19.0	0.0	6.8	0.7	0.0	0.0	0.0	128	0.1
Switchboards	87.0	0.0	0.0	13.5	0.0	0.0	0.1	4.1	0.0	0.0	0.0	0.0
Protective Equipment	5.9	0.0	0.0	39.0	0.0	0.0	0.5	0.0	0.0	0.0	0.0	0.0
Structures & Enclosure	112.5	0.0	421.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1135	0.0
Power & Control Wiring	75.6	0.0	0.0	482.2	0.0	0.0	0.5	0.0	0.0	0.6	0.0	0.0
Miscellaneous Equipment	**843.2**	**13.7**	**2**	**2.6**	**0**	**2**	**0.4**	**6.5**	**0.3**	**0**	**458**	**0**
Transportation & Lifting	529.3	0.0	0.0	0.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Air & Water Service Sys.	232.5	6.0	0.0	1.1	0.0	0.0	0.0	0.0	0.3	0.0	458	0.0
Communications Equip.	4.7	0.0	0.6	1.0	0.0	0.0	0.0	0.4	0.0	0.0	0.0	0.0
Furnishings & Fixtures	76.7	7.7	1.4	0.0	0.0	2.0	0.4	6.1	0.0	0.0	0.0	0.0
Entire Plant	**32731.2**	**2080.1**	**1257.4**	**694**	**124.1**	**46.4**	**25.1**	**18.2**	**10.1**	**0.7**	**179681**	**0.5**

Source: Bryan and Dudley 1974

* Underlined text identifies equipment/systems expected to have significant amounts of radioactive contamination

Table 2-15. Concrete Volumes in Pressurized Water Reactors

Structure	Concrete Volume (m^3) (NRC 1995b)	Concrete Volume (m^3) (Bryan and Dudley 1974)
Cooling Tower	30,963	5,510*
Containment Shell	7,645	
Containment Internals	2,450	
Containment Base	3,050	
Reactor Building		23,100
Turbine Generator Building	12,000	19,400
Reactor Auxiliary Building	7,620	14,100
Fuel Building	4,330	2,980
Control Building	4,890	
Turbine Auxiliary Building	1,330	
Miscellaneous Buildings		9,880
Total	74,278	74,970

* Water intake and discharge - no cooling tower

2.2.3.2 Radionuclide Distributions in Reference PWR

This section describes the mix or fractional activity of radionuclides associated with various components or systems in the reference PWR.

Neutron-Activated Components

Estimates of the fractional radioactivity at shutdown associated with neutron-activated stainless steel, carbon steel, and concrete were developed for the reference PWR. These estimates are presented in Table 2-16 (NRC 1978a, Tables 7.3-3, 7.3-4, and 7.3-6). The table includes only nuclides contributing > 1×10^{-4} to the fractional activity. (The source document provides a more complete listing of all nuclides assumed to be present.)

Table 2-16. Radionuclide Mix in Neutron-Activated Components of Reference PWR at Shutdown

Radionuclide	Fractional Radioactivity (%)		
	Stainless Steel	Carbon Steel	Concrete
Ar-39			0.11
Ca-41			0.020
Ca-45			10
Mn-54	2.6	5.3	0.48
Fe-55	49	82	87
Fe-59	1.7	3.1	
Co-58	5.7	0.75	
Co-60	36	8.5	1.9
Ni-59	0.028		
Ni-63	4.5	0.43	0.40

Internal Surface Contamination

Internal surface contamination of reactor components results primarily from the deposition of neutron-activated corrosion products throughout the system. The estimated fractional distribution of the radionuclides contributing to this radiation source is summarized for the reference PWR in Table 2-17 (NRC 1978a, Table 7.3-7). These data are based on measurements taken from the Turkey Point Reactors during steam generator repairs.

Table 2-17. Radioactivity Distribution at Shutdown for Reference PWR

Radionuclide	Fractional Radioactivity (%)
Cr-51	2.4
Mn-54	3.6
Fe-59	0.82
Co-58	46
Co-60	32
Zr-95	5.6
Nb-95	5.6
Ru-103	2.6
Cs-137	0.12
Ce-141	6.6

For comparison, data taken from other pressurized water reactors are summarized in Table 2-18. Substantial differences in the mix of radioactivity exist among the various PWRs.

Table 2-18. Relative Activities of Long-Lived Radionuclides at Three PWR Nuclear Power Plants* †

Radionuclide	Fractional Activity, Decayed to Shutdown Date (%)		
	Indian Point-1	Turkey Point-3	Rancho Seco
Mn-54	4	0.4	4
Fe-55	67	31	28
Co-57	—	43	24
Co-60	15	24	18
Ni-59	0.02	0.004	0.1
Ni-63	2	0.1	19
Zn-65	11	1	0.09
Sr-90	0.0007	0.0008	< 0.01
Nb-94	0.0008	< 0.004	< 0.004
Tc-99	0.00008	0.008	< 0.005
Ag-110m	—	—	4
I-129	0.00002	< 0.003	< 0.00005
Cs-137	0.5	—	0.4
Ce-144	—	0.2	< 0.04
TRU**	0.002	0.006	0.001
Total (Ci)	1,070	2,580	4,460

Source: Abel et al. 1986

* Excludes activated metal components of the reactor pressure vessel and internals and activated concrete.

† Relative activity of each nuclide as a percentage of total activity at each power plant

** Transuranic alpha-emitting radionuclides with half-lives greater than 5 years, including Pu-238, Pu-239, Pu-240, Am-241, Am-243 and Cm-244.

2.2.3.3 Contamination in Reference PWR

This section describes the amounts of radioactivity associated with various systems and components in the reference PWR.

Volumetric Contamination

Volumetric contamination is distributed throughout the thickness of a component and cannot be removed by surface decontamination methods. For the purposes of this investigation, volumetric contamination is restricted to neutron-activated reactor components and structural materials. The reactor vessel and various internal components for the reference PWR are considered to be volumetrically contaminated. In addition, concrete, rebar, and structural steel in the reactor building can become activated through interaction with neutrons. For example, it has been estimated that the concrete bio-shield will contain 1,200 Ci of radioactivity at shutdown (NRC 1978a, p. 7-23).

Table 2-19 lists the neutron-activated reactor components, their masses, and estimated residual radioactivity levels for the reference PWR.

Table 2-19. Volumetrically Contaminated Reactor Components in Reference PWR

Component	Activated Mass (kg)	Shutdown Activity (Bq/g)
Shroud	12,312	1.03×10^{10}
Lower 4.7 m of Core Barrel	26,783	9.00×10^{8}
Thermal Shield	10,413	5.20×10^{8}
Vessel Inner Cladding	2,074	2.68×10^{7}
Lower 5.02 m of Vessel Wall	245,582	2.65×10^{6}
Upper Grid Plate	4,627	1.94×10^{8}
Lower Grid Plate	3,946	5.20×10^{9}
Total	305,737	1.71×10^{10}

Source: NRC 1978a, Table 7.3-2

Internal Surface Contamination

The distribution of surface radioactivity among the principal components and systems in the reference PWR and four actual PWRs is summarized in Table 2-20.

The steam generators are the single largest repository of internally deposited radionuclides. This is because the steam generators provide the largest portion of the internal surface area in the primary loop, and the heat transfer process occurring within steam generators tends to increase the corrosion product deposition process (Abel et al. 1986, p. 28).

Since the reference PWR studies did not provide correlated data on the masses and activity of systems and components, the Trojan Radiological Site Characterization Report was used as a source for the data (TNP 1995). Details are included in Table 2-21. Data in this table are based on actual samples. In some cases, measurements were not made because of radiation levels or because of continued operation of certain systems. Of the 58 systems sampled, 23 systems showed levels of radioactivity greater than 1000 dpm/100 cm^2 and two showed low, but detectable levels (less than 1000 dpm per 100 cm^2). The table also indicates the assumed contamination condition prior to survey.

Table 2-20. Distribution of the Radionuclide Inventory Estimates for Four Pressurized Water Reactors (% of total)

Component/System	Trojan	Turkey Point-2	Indian Point-1	Rancho Seco	Reference PWR
Steam Generators	57	89	77	94	92
Pressurizer	2.1	0.5	0.5	0.33	0.08
RCS Piping	8.9	0.9	2.6	0.71	3.3
Piping (Except RCS)	21	< 0.01	14	< 0.01	1.2
Secondary Systems		0.1	0.2	0.05	--
Radwaste	5.1[a]	9.2	7	5	--
Reactor Vessel/Internals[b]	14				2.7

Source: Abel et al. 1986; NRC 1978a, Table 7.3-8; and TNP 1995

a Including spent fuel pool

b Excludes neutron-activated components

Table 2-21. Trojan PWR System Characterization Data

Class	No. of Samples	System #	System Name	System Weight (lbs)	Activity[(a)] (Ci)
C2	6	16	Component Cooling Water	475,874	GeLi[(c)]
C1	18	32	HVAC-Fuel & Auxiliary Buildings	45,800	>1K[(c)]
C1	NS	35	Spent Fuel Pool Cooling & Demin.	57,281	5.6
C1	NS	36	Spent Fuel Pool	628,378	100
C2	5	39	Condensate Demineralizers	18,000	Clean[(d)]
		42D	Discharge & Dilution	63,505	<1
C1	NS	49	Residual Heat Removal	183,855	36
C1	NS	50	Chemical & Volume Control	534,034	25
C1	NS	52	Safety Injection & Accumulators	493,765	<1
C1	NS	55	Control Rod Drive Mechanisms	106,318	83
C2	8	60	HVAC-Containment	407,328	>1K
C2	2	61	Containment Spray	75,252	>1K
C1	NS	63A	Steam Generators	2,650,448	1416
C2	6	63B	SG Blowdown	39,449	>1K
C1	NS	64A	Reactor Coolant Pumps	768,400	134
C1	NS	64B	Reactor Coolant System Piping	294,460	221
C1	NS	64C	Pressurizer	195,508	52.1
C1	NS	64D	Reactor Vessel/Internals (surface only)	1,286,000[(b)]	357.9
C1	2	66	Hydrogen Recombiners	12,600	>1K
C2	2	67AB	Primary Water Makeup System	90,006	>1K
C1	1	67D	Refueling Water Storage Tank	97,928	7

Table 2-21. Trojan PWR System Characterization Data (continued)

Class	No. of Samples	System #	System Name	System Weight (lbs)	Activity[a] (Ci)
C1	NS	68	Solid Radwaste	10,341	<1
C1	NS	69	Clean Radwaste	110,634	14
C1	NS	71	Dirty Radwaste	24,116	<1
C1	NS	72	Gaseous Radwaste	77,261	<1
C1	NS	76	Process Sampling	3,093	4
C2	21	99A	Miscellaneous Sumps	19,136	>1K
C2	2	25	Startup Boiler		Clean
C2	3	37	Condensate Storage/Transfer		Clean
C2	2	43	Condenser & Air Removal		Clean
C2	9	44	Condensate		Clean
C2	5	45	Feedwater System (& AFW)		Clean
C2	2	46	Extraction Steam		Clean
C2	4	47	Feedwater Heaters, Vents & Drains		Clean
C2	4	48	SG Feed Pump Turbine Drivers		Clean
C2	11	65	Oily Waste & Storm Drains	(1,882 ft^3)	Clean[d]
C2	1	67C	Degassifier		Clean
C2	13	83	Main Steam		Clean
C2	9	84	Reheat & Moisture Separators		Clean
I	2	11	Service Water		Clean
I	2	15	Turbine Bldg. Cooling Water		Clean
I	1	28	Process & Aux. Steam		Clean
I	4	33	HVAC - Turbine		Clean
I	2	42A	Circulating Water Pumps & Aux.		Clean
I	2	74	Misc. Gas Supply		GeLi
I	4	82	Chemical Injection		Clean
I	13	93	Main Turbine		Clean
N	2	2	125 V dc	(175 ft^3)	Clean[d]
N	2	5	480 V ac Aux. Load Centers	(5,080 ft^3)	Clean[d]
N	2	6	480 V Motor Control Centers	(8,426 ft^3)	Clean[d]
N	1	7	Lighting Panel Power Supply	(007 ft^3)	Clean[d]
N	5	8	Domestic Water		Clean
N	2	22	Makeup Demin. Water		Clean
N	6	30	HVAC - Control Panel		Clean
N	1	31A	P-250 Computer		Clean
N	1	57	125 V ac Preferred Instrument	(1,400 ft^3)	Clean[c]
N	2	90	Communications		Clean

Table 2-21. Trojan PWR System Characterization Data (continued)

Class	No. of Samples	System #	System Name	System Weight (lbs)	Activity[a] (Ci)
N	1	91	Annunciators		Clean
N	1	97	Stator Cooling		Clean
N	2	98	Main Generator & Excitation		Clean
N	4	99G	Fish Rearing Facility		Clean

Table Notes:

C1 - contaminated
C2 - potentially contaminated
I - indeterminant
N - not contaminated (clean)

>1K - greater than 1000 dpm/100 cm^2 above background
NS - not sampled due to radiation levels or continued operation
GeLi - less than 1000 dpm/100 cm^2 above background but showing activity with highly sensitive GeLi detector

a Does not include activation
b Includes reactor vessel - 308.4 Mg, vessel head - 88.5 Mg, lower core support structure - 127 Mg, upper core support structure - 54.4 Mg, in-core instrumentation support structure - 5 Mg (total = 583.3 Mg or 1,286,000 lb) (NRC 1978b, Appendix A).
c GeLi and >K subsequently reclassified as <1 Ci (Trojan 2001)
d Subsequently reclassified as < 1 Ci (Trojan 2001)

The total mass of systems with internal surface contamination is 8,769,000 lb (3,978 t).

In addition, the Trojan staff characterized the contamination on other systems/components which were not internally sampled because they were included in other assessments (A), were entirely electrical (B), or had a low probability of contamination (C). These are listed in Table 2-22.

Radioactive material on the contaminated systems listed in Table 2-21 consists of both fixed and removable material. According to TNP 1995, Section 6.1.2, "the total radioactivity is not expected to be substantially reduced by nonaggressive decontamination methods. Operational experience during activities such as steam generator primary bowl hydrolasing indicated that the radioactivity is tightly adherent to surfaces and will require disposal of the entire component."

Table 2-22. Expected Contamination of Trojan Systems/Components Not Sampled

Class	Description	Reason Not Sampled*
C1	Refueling Equipment	A
C1	Containment Bldg.	A
C1	In-Core Neutron Flux Monitors	A
C1	In-Core Temperature Monitors	A
C1	Fuel Handling & Refueling Cavity	A
C1	Vacuum Cleaners	A
C2	Misc. Bldgs/Structures	A
C2	Turbine & Turbine Aux. Bldg.	A
C2	Condensate Demin. Bldg.	A
C2	Seismic Monitor	A
C2	Fuel & Aux. Bldg.	A
C2	Radiation Shielding	A
C2	Containment & Misc. Cranes	A
I	Intake/Discharge & Chlorine Bldgs.	A
I	Cooling Tower Structure	A
I	Elevators	A
I	Fuel Bldg. Crane	A
N	Tech. Support Center Bldg.	A
N	Turbine Bldg. Crane	A
N	Condensate Demin. Crane	A
C2	Electric Heat Tracing Power	B
C2	Vibration & Loose Parts Monitor	B
C2	Nuclear Instrumentation	B
C2	Radiation Monitors	B
N	230 kV Switchyard	B
N	12.47 kV Startup Transformer	B
N	4.17 kV Aux. Power	B
N	120 V Non-preferred Instrument ac	B
N	Computers (other than P-250)	B
N	Feedwater Flow & Level Control	B
N	Feed Line Isolation Actuation	B
N	Auxiliary Feedwater Autostart	B
N	Engineered Safeguards Actuation	B
N	Reactor Control and Protection	B
N	Reactor Non-nuclear Instrumentation	B
N	ATWS Mitigation & Actuation	B
N	Transformers & Auxiliaries	B
N	250 V dc	B
N	Meteorological Equipment	B
N	Welding Receptacles	B

Table 2-22. Expected Contamination of Trojan Systems/Components Not Sampled (continued)

Class	Description	Reason Not Sampled*
N	Cathodic Protection	B
N	Vehicle Battery Charger	B
N	Motor Operated Doors	B
N	Security System	B
C2	Fire Protection	C
C2	DD&DS, Dechlorination System	C
I	Instrument & Service Air	C
I	Emergency Diesel Generators	C
I	HVAC - Misc. Buildings	C
I	Sewage Treatment	C
I	Turbine Steam Seal & Drain	C
N	Traveling Water Screens & Screen Wash	C
N	Chlorination	C
N	Bearing Cooling Water	C
N	Water Pretreatment	C
N	Diesel Fuel Oil	C
N	HVAC - Admin. Bldg. & Gatehouse	C
N	Lube Oil Storage & Filtration	C
N	Cooling Tower Makeup & Discharge	C
N	Cooling Tower Acid Pump	C
N	Primary Containment Testing	C
N	Chilled Water	C
N	Generator & Hydrogen Seal Oil	C

* A - included in other assessments
B - entirely electrical
C - low probability of contamination

Concrete Contamination in Reference PWR

Remediation of contaminated structural surfaces is expected to be accomplished more easily than remediation of reactor piping surfaces with internal contamination. Removable contamination can be addressed by mopping or wiping the surfaces. Fixed contamination on concrete can be removed by scabbling about 1 cm from the exposed surfaces. The contamination associated with concrete surfaces is described in Table 2-23 (TNP 1995, Table 6.1).

Table 2-23. Contamination of Structural Concrete in the Trojan PWR

Building	Contaminated Volume (ft^3)	Activity (mCi)
Containment (floors)	668	20.4
Containment (walls)	2,262	2.71
Auxiliary	234	2.31
Fuel	176	1.13
MSSS/EP*	43	1.36
Turbine	75	2.39
Total	3,458	30.3

MSSS/EP - main steam support structure/electrical penetrations

The average concrete contamination level from Table 2-23 is 130 pCi/g (4.8 Bq/g). This material would be disposed of as LLW.

The Trojan staff estimates that the volumes in Table 2-23 should be increased by an additional 10 percent to account for contaminated ceilings and other non-floor surfaces, and the volumes should be increased by another 10 percent to account for the possibility that some areas must have more than 1 cm of the surface removed. Thus, the net contaminated volume removed is 4,184 ft^3. This volume is exclusive of the activated volume in the primary shield wall.

Reference PWR Contamination Summary

As noted in Table 2-14, the reactor plant equipment category is expected to be a large source of contaminated materials. This category for the generic 1000-MWe PWR involves a total of 4,605 t of ferrous metals (carbon steel, stainless steel, and galvanized iron) as compared to 3,978 t based on actual measurements at the Trojan Nuclear Plant as shown in Table 2-21. Predicated on the systems in Table 2-21 designated as clean, it is reasonable to assume that the mass of ferrous metals in the Turbine Plant Equipment, Electric Plant Equipment, and Miscellaneous Equipment categories of Table 2-14 would fall within the clean designation. This amounts to 14,102 t Further, as noted in Table 2-23, some contamination exists on concrete building walls, but this contamination can be removed by scabbbling about 1 cm of material from the contaminated areas. No information is available as to the extent of contamination on structural steel and rebar in those areas where concrete surface contamination is present. In the absence of any specific information, the ferrous metals in these structures are assumed to be clean or easily decontaminated by washing/wiping. Thus, the assumed ferrous metals breakdown for a generic 1000 MWe PWR and the assumed MARSSIM Class for each are presented in Table 2-24.

Table 2-24. Summary of Ferrous Metals Radioactivity Levels in 1,000-MWe Generic PWR

System	Mass (t)	MARSSIM Category	Contamination Level
Reactor Plant Equipment	3,978[a]	Class 1	See Table 7 for contamination details
Reactor Plant Equipment	627	Class 3	Clean
Turbine Plant Equipment	11,846	Class 3	Clean
Electrical Plant Equipment	1,396	Nonimpacted	Clean
Miscellaneous Equipment	859	Nonimpacted	Clean
Site Improvements	1,711	Nonimpacted	Clean
Reactor Building	7,571	Class 2	Clean*
Turbine Building	3,838	Class 2	Clean*
Intake/Discharge	337	Nonimpacted	Clean
Reactor Auxiliaries	1,468	Class 2	Clean*
Fuel Storage	429	Class 2	Clean*
Miscellaneous Buildings	2,008	Class 2	Clean*

a Includes 306 t of activated steel per Table 2-19

* Status after scabbling 1 cm of concrete from contaminated areas and washing/wiping exposed steel.

If one assumes that copper and aluminum contamination generally parallels that of ferrous metals, but that 100 percent of these metals associated with reactor equipment are contaminated, then from Table 2-14, one can estimate that the reference PWR contains 50 t of contaminated copper and 644 t of clean copper, and 5 t of contaminated aluminum and 13 t of clean aluminum.

In an earlier study, SAIC (1998) used the material masses from Brian and Dudley (1974) (see Table 2-14) and estimated the fractions that were activated, contaminated, and clean. According to the SAIC authors, the estimations were based on contamination data contained in PNL 1985 and Charles and Smith 1992. These estimates are presented in Table 2-25. Quantities of contaminated and activated materials are generally higher than those developed in this report (e.g., 6,000 t of contaminated and activated steel in SAIC 1998 as compared to 4,000 t in this report). However, this variation may well be within the range expected for different nuclear power plants based on operating experience and approaches to decommissioning.

Table 2-25. Reference Waste Masses Produced During Decommissioning of a Typical PWR

Material Category	Concrete (t)	Steel (t)	Copper (t)	Aluminum (t)
Activated	1,000 (0.5%)	1,000 (3%)	0	0
Contaminated	14,000 (7.5%)	5,000 (14%)	300 (43%)	4 (22%)
Clean	165,000 (92%)	29,000 (83%)	400 (57%)	14 (78%)
Total	180,000	35,000	700	18

Source: SAIC 1998

2.2.4 Inventory from Commercial Reactors

A complete listing of U.S. commercial nuclear power reactors, together with the shutdown date established by their operating license, is included as Table 2-26 (NRC 2000a). The normal duration of the license is 40 years. Several reactor operators have applied to the NRC for a license extension of an additional 20 years. Shutdown dates have been adjusted in Table 2-26 to reflect those plants granted license extensions. Table 2-26 also contains scaling factors used to adjust the quantities of materials for various design power levels. By this means, data developed for reference PWRs and BWRs can be utilized to estimate inventories for the industry at large. In reports prepared for the DOE, Argonne National Laboratory (ANL) employed a scaling method based on the mass of PWR and BWR pressure vessels described in Nuclear Engineering International (NEI 1991, 1992, and 1993). ANL assumed that all metal inventories for both PWRs and BWRs can be calculated from those at the corresponding reference plant based on the design power rating as follows (Nieves et al. 1995):

$$M = M_r \left(\frac{P}{P_r} \right)^{2/3}$$

M = mass of metal (e.g., carbon steel) in actual reactor
M_r = mass of same metal in reference reactor
P = power rating of actual reactor (MWe)
P_r = power rating of reference reactor

The quantity $\left(\frac{P}{P_r} \right)^{2/3}$ is referred to as the scaling factor.

Table 2-26. Nuclear Power Reactors Currently Licensed to Operate

Electric Utility	Reactor	Type	Power Rating (MWe)[a]	Scaling Factor[b] PWR	Scaling Factor[b] BWR	Year of Projected Shutdown
Arizona Public Service	Palo Verde 1	PWR	1,227	1.146	—	2024
Arizona Public Service	Palo Verde 2	PWR	1,227	1.146	—	2025
Arizona Public Service	Palo Verde 3	PWR	1,230	1.148	—	2027
Baltimore Gas & Electric	Calvert Cliffs 1	PWR	835	0.887	—	2034
Baltimore Gas & Electric	Calvert Cliffs 2	PWR	840	0.890	—	2036
Boston Edison	Pilgrim 1	BWR	670	—	0.766	2012
Carolina Power & Light	Brunswick 1	BWR	767	—	0.838	2016
Carolina Power & Light	Brunswick 2	BWR	754	—	0.828	2014
Carolina Power & Light	H. B. Robinson 2	PWR	683	0.776	—	2010
Carolina Power & Light	Shearon Harris 1	PWR	860	0.904	—	2026
Centerior Energy	Davis-Besse	PWR	873	0.913	—	2017
Cleveland Electric	Perry 1	BWR	1,160	—	1.104	2026
Commonwealth Edison	Braidwood 1	PWR	1,100	1.066	—	2026
Commonwealth Edison	Braidwood 2	PWR	1,100	1.066	—	2027
Commonwealth Edison	Byron 1	PWR	1,105	1.069	—	2024
Commonwealth Edison	Byron 2	PWR	1,105	1.069	—	2026
Commonwealth Edison	Dresden 2	BWR	772	—	0.842	2006
Commonwealth Edison	Dresden 3	BWR	773	—	0.842	2011
Commonwealth Edison	LaSalle 1	BWR	1,036	—	1.024	2022
Commonwealth Edison	LaSalle 2	BWR	1,036	—	1.024	2023
Commonwealth Edison	Quad Cities 1	BWR	769	—	0.839	2012
Commonwealth Edison	Quad Cities 2	BWR	769	—	0.839	2012
Consolidated Edison	Indian Point 2	PWR	951	0.967	—	2013
Consumers Energy	Palisades 1	PWR	730	0.811	—	2011[c]
Detroit Edison	Fermi 2	BWR	876	—	0.916	2025
Duke Power	Catawba 1	PWR	1,129	1.084	—	2024
Duke Power	Catawba 2	PWR	1,129	1.084	—	2026
Duke Power	McGuire 1	PWR	1,129	1.084	—	2021
Duke Power	McGuire 2	PWR	1,129	1.084	—	2023
Duke Power	Oconee 1	PWR	846	0.895	—	2033
Duke Power	Oconee 2	PWR	846	0.895	—	2033
Duke Power	Oconee 3	PWR	846	0.895	—	2034
Duquesne Light	Beaver Valley 1	PWR	810	0.869	—	2016
Duquesne Light	Beaver Valley 2	PWR	820	0.876	—	2027
Entergy Operations, Inc.	Arkansas Nuclear 1	PWR	836	0.887	—	2014
Entergy Operations, Inc.	Arkansas Nuclear 2	PWR	858	0.903	—	2018
Entergy Operations, Inc.	Grand Gulf 1	BWR	1,179	—	1.116	2022

Table 2-26. Nuclear Power Reactors Currently Licensed to Operate (continued)

Electric Utility	Reactor	Type	Power Rating (MWe)[a]	Scaling Factor[b] PWR	Scaling Factor[b] BWR	Year of Projected Shutdown
Entergy Operations, Inc.	River Bend 1	BWR	936	—	0.957	2025
Entergy Operations, Inc.	Waterford 3	PWR	1,104	1.068	—	2024
Florida Power Corp.	Crystal River 3	PWR	818	0.875	—	2016
Florida Power & Light	St. Lucie 1	PWR	839	0.890	—	2016
Florida Power & Light	St. Lucie 2	PWR	839	0.890	—	2023
Florida Power & Light	Turkey Point 3	PWR	693	0.783	—	2012
Florida Power & Light	Turkey Point 4	PWR	693	0.783	—	2013
GPU Nuclear	Oyster Creek	BWR	619	—	0.726	2009
GPU Nuclear	Three Mile Island 1	PWR	786	0.852	—	2014
Illinois Power	Clinton	BWR	930	—	0.953	2026
Indiana/Michigan Power	D. C. Cook 1	PWR	1,000	1.000	—	2014
Indiana/Michigan Power	D. C. Cook 2	PWR	1,060	1.040	—	2017
IES Utilities	Duane Arnold	BWR	520	—	0.647	2014
Nebraska Public Power	Cooper	BWR	764	—	0.836	2014
New York Power Authority	James A. Fitzpatrick	BWR	762	—	0.834	2014
New York Power Authority	Indian Point 3	PWR	965	0.977	—	2015
Niagara Mohawk	Nine Mile Point 1	BWR	565	—	0.683	2009
Niagara Mohawk	Nine Mile Point 2	BWR	1,105	—	1.069	2026
North Atlantic Energy	Seabrook 1	PWR	1,158	1.103	—	2026
Northeast Nuclear Energy	Millstone 2	PWR	871	0.912	—	2015
Northeast Nuclear Energy	Millstone 3	PWR	1,137	1.089	—	2025
Northern States Power	Monticello	BWR	544	—	0.666	2010
Northern States Power	Prairie Island 1	PWR	513	0.641	—	2013
Northern States Power	Prairie Island 2	PWR	512	0.640	—	2014
Omaha Public Power	Fort Calhoun	PWR	478	0.611	—	2013
Pacific Gas & Electric	Diablo Canyon 1	PWR	1,073	1.048	—	2021
Pacific Gas & Electric	Diablo Canyon 2	PWR	1,087	1.057	—	2025
PECO Energy	Peach Bottom 2	BWR	1,093	—	1.061	2013
PECO Energy	Peach Bottom 3	BWR	1,093	—	1.061	2014
Pennsylvania Power	Susquehanna 1	BWR	1,090	—	1.059	2022
Pennsylvania Power	Susquehanna 2	BWR	1,094	—	1.062	2024
Philadephia Electric	Limerick 1	BWR	1,105	—	1.069	2024
Philadelphia Electric	Limerick 2	BWR	1,115	—	1.075	2029
Public Service E & G	Hope Creek 1	BWR	1,031	—	1.021	2026
Public Service E & G	Salem 1	PWR	1,115	1.075	—	2016
Public Service E & G	Salem 2	PWR	1,115	1.075	—	2020
Rochester Gas & Electric	Ginna 3	PWR	470	0.605	—	2009

Table 2-26. Nuclear Power Reactors Currently Licensed to Operate (continued)

Electric Utility	Reactor	Type	Power Rating (MWe)[a]	Scaling Factor[b] PWR	Scaling Factor[b] BWR	Year of Projected Shutdown
South Carolina E & G	Summer	PWR	945	0.963	—	2022
Southern California Edison	San Onofre 2	PWR	1,070	1.046	—	2022[c]
Southern California Edison	San Onofre 3	PWR	1,080	1.053	—	2022[c]
Southern Nuclear	Edwin I. Hatch 1	BWR	805	—	0.865	2014
Southern Nuclear	Edwin I. Hatch 2	BWR	809	—	0.868	2018
Southern Nuclear	Joseph M. Farley 1	PWR	812	0.870	—	2017
Southern Nuclear	Joseph M. Farley 2	PWR	822	0.878	—	2021
Southern Nuclear	Vogtle 1	PWR	1,162	1.105	—	2027
Southern Nuclear	Vogtle 2	PWR	1,162	1.105	—	2029
STP Nuclear	South Texas 1	PWR	1,251	1.161	—	2027
STP Nuclear	South Texas 2	PWR	1,251	1.161	—	2028
Tennessee Valley Authority	Browns Ferry 1	BWR	1,065[d]	—	1.043	2013
Tennessee Valley Authority	Browns Ferry 2	BWR	1,065	—	1.043	2014
Tennessee Valley Authority	Browns Ferry 3	BWR	1,065	—	1.043	2016
Tennessee Valley Authority	Sequoya 1	PWR	1,117	1.077	—	2020
Tennessee Valley Authority	Sequoya 2	PWR	1,117	1.077	—	2021
Tennessee Valley Authority	Watts Bar 1	PWR	1,117	1.077	—	2035
Texas Utilities Electric	Comanche Peak 1	PWR	1,150	1.098	—	2030
Texas Utilities Electric	Comanche Peak 2	PWR	1,150	1.098	—	2033
Union Electric	Callaway	PWR	1,171	1.111	—	2024
Vermont Yankee Nuclear	Vermont Yankee	BWR	510	—	0.638	2012
Virginia Electric & Power	North Anna 1	PWR	893	0.927	—	2018
Virginia Electric & Power	North Anna 2	PWR	897	0.930	—	2020
Virginia Electric & Power	Surry 1	PWR	801	0.862	—	2012
Virginia Electric & Power	Surry 2	PWR	801	0.862	—	2013
Washington Public Power	Washington Nuclear 2	BWR	1,107	—	1.070	2023
Wisconsin Electric Power	Point Beach 1	PWR	485	0.617	—	2010
Wisconsin Electric Power	Point Beach 2	PWR	485	0.617	—	2013
Wisconsin Public Service	Kewaunee	PWR	511	0.639	—	2013
Wolf Creek Nuclear	Wolf Creek 1	PWR	1,163	1.106	—	2025
Total				65.866	32.327	

a Net maximum dependable capacity
b Scaling factor = (power rating/1000)$^{2/3}$ (see text)
c Assuming construction recapture
d Based on design characteristics—reactor has no fuel loaded and requires NRC approval to restart

Table 2-27 lists the commercial nuclear power reactors that were formerly licensed but have been shut down. The list excludes reactors whose owners have chosen the ENTOMB decommissioning alternative and those with the DECON alternative that have begun or already completed decommissioning. For the purpose of the present analysis, the three non-light-water reactors are treated as if they were BWRs.

Table 2-27. Formerly Licensed to Operate Nuclear Power Reactors

Reactor	Type	Power Rating (MWe)[a]	Scaling Factor[b]		Alternative[c]	Year	
			PWR	BWR		Shutdown	Release[d]
Big Rock Point	BWR	72	—	0.173	DECON	1997	2007
CVTR	PTHW[e]	20	—	0.074	SAFSTOR	1967	2027
Dresden 1	BWR	210	—	0.353	SAFSTOR	1978	2038
Fermi 1	SCF[e]	60	—	0.153	SAFSTOR	1972	2032
GE VBWR	BWR	15	—	0.061	SAFSTOR	1963	2023
Haddam Neck	PWR	548	0.670	—	DECON	1996	2006
Humboldt Bay	BWR	60	—	0.153	SAFSTOR	1976	2036
Indian Point 1	PWR	185	0.325	—	SAFSTOR	1974	2034
La Crosse	BWR	50	—	0.136	SAFSTOR	1987	2047
Maine Yankee	PWR	732	0.812	—	DECON	1996	2006
Millstone 1	BWR	603	—	0.714	SAFSTOR	1998	2058
Peach Bottom 1	HTGR[e]	34	—	0.105	SAFSTOR	1974	2034
Rancho Seco	PWR	832	0.885	—	SAFSTOR[f]	1989	2049
San Onofre 1	PWR	404	0.547	—	SAFSTOR	1992	2052
Three Mile Island 2	PWR	831	0.884	—	[g]	1979	2039
Zion 1	PWR	975	0.983	—	SAFSTOR	1997	2057
Zion 2	PWR	975	0.983	—	SAFSTOR	1996	2056
Total	shutdown reactors[h]		6.088	1.922			
Total	including currently licensed reactors		71.954	34.249			

Source: NRC 2000a

a Licensed thermal capacity × 0.3
b Scaling factor = (power rating/1000)$^{2/3}$ (see text)
c Selected decommissioning alternative
d Year that significant quantities of scrap metal will be released—10 years after shutdown for the DECON alternative, 60 years for SAFSTOR
e Metals inventory and contamination levels assumed same as for BWR
f Dismantlement of radioactive secondary piping and components is ongoing
g In monitored storage until TMI-1 is shut down, then both will be decommissioned
h Excludes reactors at which DECON has started or been completed and those in ENTOMB status

The last column lists the date that significant quantities of scrap metal would be released from these reactors. For reactors in SAFSTOR, this is assumed to be 60 years after the shutdown date, while for those with the DECON alternative, it is 10 years after shutdown.

2.2.5 Specific Reactor Decommissioning Plans and Experience

Decommissioning plans and/or current activities for several reactors are described below with particular reference to the kinds and quantities of materials and associated radioactivity.
A common practice in reactor decommissioning is to employ a waste broker or processor to manage the wastes, including decisions as to the disposition of the waste, how to process the waste, and where to dispose of the waste. The decision-making model used by one waste broker (Duratek) to determine the most cost-effective waste dispositioning options is shown in Figure 2-1 (Radwaste 1999). Duratek processed 62 million pounds of waste in 2000 (Johnson 2001).

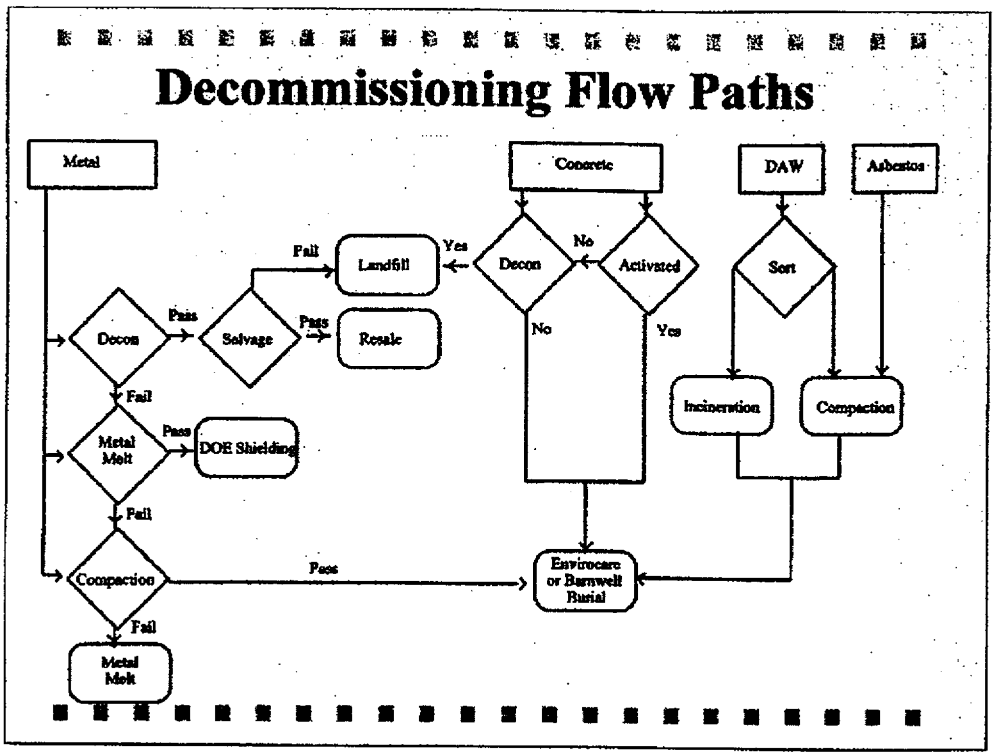


Figure 2-1. Alternative Flow Paths for Dispositioning Waste from Nuclear Reactor Decommissioning

In theory, a utility will choose to decontaminate a component or structure if the cost of decontamination and the economic value of the decontaminated object exceed the cost of disposal as LLW. In practice, the situation is more complicated. For example:

- Decontamination is not always as effective as planned, and the costs of failed decommissioning must be added to the costs of disposal as LLW.
- Contamination surveys can be misleading. Contamination is frequently concentrated in cracks and crevices, rather than distributed uniformly over a surface. Consequently, the utility may opt for disposal as LLW rather than conduct exhaustive radiation surveys followed by attempted decontamination of difficult geometries.
- The costs of maintaining the project decommissioning staff for longer periods may outweigh the cost savings from a comprehensive decontamination program.
- A strategy of risk avoidance may dominate over one of cost avoidance. LLW disposal is a lower risk strategy.
- A nuclear facility may choose to decontaminate certain components to take advantage of lower waste disposal costs associated with LLW of low activity. Such costs may vary significantly among disposal sites.

2.2.5.1 Haddam Neck

The Haddam Neck pressurized water reactor, located in Meriden, Connecticut, and owned by the Connecticut Yankee Atomic Power Company, was shut down on December 4, 1996, after about 28 years of operation. This four-loop PWR was designed to produce 1,825 MW of thermal power and 590 MW of gross electrical power. Based on ALARA considerations, the site management team decided to perform a full-system chemical decontamination operation prior to removal of any primary system components (Szymczak et al., 1998). The reactor pressure vessel was bypassed during the decontamination process to minimize removal of activated materials with high concentrations of radioactivity. Decontaminated components and systems included primary system piping, the residual heat removal system, and the chemical volume control system. A decontamination factor of 15 was achieved. A total of 131 Ci of Co-60 activity was removed during this process (Haddam LTP 2000, p. 3-4).

The reactor site was characterized using MARSSIM guidelines, and it was determined that 93 acres were "nonimpacted," and that 53 percent of the survey areas were initially identified as Class 1, 27 percent as Class 2, and 20 percent as Class 3. To the extent possible, impacted facility materials and surfaces will be decontaminated to allow for beneficial reuse. Buildings may be demolished and the concrete debris used as backfill on site (Haddam LTP 2000, Section 1). The buildings would be decontaminated to meet the requirements of 10 CFR 20.1402 prior to demolition. To insure that possible reuse scenarios were adequately bounded, two post-demolition scenarios were evaluated. One involved a resident farmer whose drinking water originates from a well located in the buried debris, and the other considers excavation and reuse of the concrete debris in a structure. "The results of these two additional surveys have been analyzed to insure that the most limiting radionuclide-specific derived concentration guideline levels (DCGLs) are used to calculate operational DCGLs for building surface surveys" (Haddam LTP 2000).

During decommissioning, the low-pressure turbine rotors have been removed from the system and transported to the Palisades Nuclear Plant in Michigan. The high-pressure turbine rotor remains onsite for eventual disposal.

Potential quantities of radioactive waste from the decommissioning of the Haddam Neck Plant are provided in Table 2-28 (Haddam LTP 2000, Section 3, Table 3-3). As of the spring of 2001, dismantlement was 40 percent completed, and the project is scheduled for completion in 2004 (Cavanaugh 2001).

Table 2-28. Projected Quantities of Radioactive Waste from the Decommissioning of the Haddam Neck Plant

Item Description	Waste Classification per 10 CFR Part 61	Volume (ft^3)
Reactor Vessel	A and B	7,892
Reactor Vessel Internals	A, B, C, and GTCC	2,444
Pressurizer	A	2,083
Reactor Coolant Pumps & Motors	A	5,592
Steam Generators (4)	A	20,772
Balance of NSSS Equipment	A	6,441
Primary System Decon. Resin	C	1,000
Fuel Racks	A	17,398
Balance of Non-NSSS Equipment	A	75,534
Concrete & Structural Steel	A	7,344
Soil & Pavement	A	93,749
Dry Active Waste	A	12,696
Contaminated Tools	A	10,000
Processed Liquids	B, some A	6,406
Contaminated Asbestos	A	21,266
Total Volume Generated		290,617
Estimated Volume Reduction		-7,500
Burial Volume		283,117

2.2.5.2 Maine Yankee

Maine Yankee is a three-loop pressurized water reactor with a power rating of 2,700 MWt on an 820-acre site in Wiscasset, Maine. The plant terminated operations in December 1996 and is currently undergoing decommissioning. The License Termination Plan (available at http://www.maineyankee.com/public/pdfs/) describes the planned D&D activities (Maine LTP 2000).

In 1990, a leak in one of the steam generators caused some transport of radioactivity to the secondary system. The contamination was limited to volatile radionuclides. A subsequent survey did not reveal the presence of any residual radioactivity. When the plant was permanently shut down in December 1996, the primary system was chemically decontaminated, and decontamination factors of 5 to 10 were obtained (Maine LTP 2000, Section 2). Initial site characterization work indicated that the principal contaminants on interior piping surfaces were Fe-55, Ni-63, Co-60, and Cs-137, which accounted for 97 percent of the system activity.

Cobalt-60 was the main beta/gamma-emitter. Transuranic radionuclides contributed less than 1 percent of the total activity. External surfaces show the same contamination pattern.

Site characterization activities to support the license termination plan divided the facility and its environs into various groupings with similar characteristics to consolidate survey results. Survey results are summarized below (Maine LTP 2000, Section 2):

- Group A "Affected Structures and Surfaces" - This group consists of buildings and surfaces within the RCA (Radiation Control Area) including Reactor Containment, Fuel and Primary Auxiliary Buildings as well as tanks containing radioactive liquids, electrical/mechanical penetrations, and concrete surfaces. "Maximum surface activities ranged from greater than 100,000 dpm/100 cm^2 in the RCA Storage, Containment and Spray Building to less than 1,000 dpm/100 cm^2 in auxiliary support areas (e.g., electrical/mechanical penetrations)."

- Group B "Unaffected Structures and Surfaces" - This group comprises buildings and surfaces outside the RCA including the Turbine Hall, sections of the Service Building, the Control Room, office spaces, and various out buildings. "Survey results ranged from a high of 8,600 dpm/100 cm^2 (in the chemical addition tank from naturally occurring radioactivity) to a low of 275 dpm/100 cm^2 in the cable vault."

- Group C "Affected Plant Systems" - Systems included the RCS, CVCS, ECCS, liquid and solid waste, containment ventilation, and primary vents and drains. These are the highest activity systems and components in Maine Yankee and will be disposed of as radioactive waste during decommissioning. Group C systems had internal surface contamination of up to 500,000 dpm/100 cm^2 of removable beta activity.

- Group D "Unaffected Plant Systems (Including the Sewerage Treatment System)" - Systems in this group are secondary side systems designed to be free of contamination. Examples are the main steam, feedwater, compressed air, and potable water systems. However, some minor contamination was expected due to steam generator leakage and other operational issues. For example, the auxiliary condensate system had a maximum removable beta activity of 115 dpm/100 cm^2 and the Turbine Hall sumps had a maximum direct beta activity of 5,800 dpm/100 cm^2. Of the 34 systems in Group D, 9 had detectable levels of plant-derived radioactive material.

- Group R "Environs Affected and Unaffected" - This group consisted of seven areas assumed to be affected and 18 areas assumed to be unaffected. The seven areas assumed to be affected (all within the protected area) and three of the areas assumed to be unaffected (outside the protected area) had elevated readings and will require further assessment.

- Ventilation Ducts and Drains (Included in Groups C, D, and R) - Affected System Vents and Drains showed mean removable contamination values ranging from 2,000 to 400,000 dpm/100 cm^2 (with maximum values ranging from 6,000 to 500,000 dpm/100 cm^2). Unaffected System Vents and Drains (Group D) had one system with significant activity above the MDA and three other systems with positive activity. Activity was also reported for two Turbine Hall sumps and a sump oil collection tank.

The license termination plan called for concrete structures to be decontaminated to levels permitting unrestricted use per 10 CFR Part 20 and then demolished to a height of 3 feet below grade (Maine LTP 2000, Section 2). The rubble would then be used as fill within lower building elevations or elsewhere on the site. This rubble is expected to contain a total of 0.3 Ci of radioactivity. To satisfy concerns raised by various stakeholders, Maine Yankee proposed to inject the decontaminated concrete rubble with flowable fill to create a consolidated mass. This should reduce the exposure from all pathways to less than10 mrem per year and the water exposure pathway to less than 4 mrem per year.

This approach was unacceptable to authorities in the State of Maine who were concerned with the amount of radioactivity that would be immobilized onsite and the possible creation of a pH plume in the groundwater. Consequently, Maine Yankee decided to dispose of about 60 percent of the concrete (45,000 tons) at a licensed LLW disposal facility (Envirocare) and about 40 percent (30,000 tons) at commercial landfills in the Northeast (Odell 2001).

Characterization has determined that concrete within the Radiologically Controlled Area (RCA) of the site shows the following (Maine LTP 200, Section 3):

- Painted concrete has surface contamination up to 1 million dpm/100 cm^2 (worst case), which is amenable to surface remediation techniques such as wiping, washing, power washing, or abrasive surface removal.

- Bare concrete has surface contamination, absorbed contamination, and activation products within the concrete mix. Absorbed activity has been found to penetrate to a depth of approximately 0.5 inches.

- Concrete structures adjacent to the reactor vessel also showed activation products at levels of a few pCi/g except for the In Core Instrumentation (ICI) sump, where levels were as high as 600 to 800 pCi/g to depths of several inches. These levels of radioactivity are amenable to remediation by surface removal techniques except for the deeply deposited activation products.

The radionuclide mix for surface-contaminated and activated concrete is presented in Table 2-29. Co-60, Ni-63, and Cs-137 constitute approximately 98 percent of the beta/gamma activity of the surface contaminated concrete. Eu-152 and Eu-154 contribute 71 percent of the activity of the activated concrete, with Co-60 adding another 27 percent.

About 5 percent of the concrete in the containment structure is activated and will be buried offsite as low-level waste (LLW) (11,400 ft^3 per Maine LTP 2000, Section 3, Table 3). About 1,200,000 ft^3 of concrete will be decontaminated to meet the unrestricted use criteria and used as onsite fill (ibid.).

Table 2-29. Radionuclide Mix in Contaminated Concrete at Maine Yankee

Radionuclides	Activity Fraction	Depth of Activity
Surface Contaminated		
H-3	0.002	99 percent of activity is in first 1 mm
Fe-55	0.009	
C0-60	0.071	
Ni-63	0.601	
Sr-90	0.0007	
Cs-134	0.007	
Cs-137	0.310	
Np-237	<0.00008	
Activated		
Co-60	0.27	Approximately evenly distributed to a depth of 3 inches
Cs-134	0.02	
Eu-152	0.65	
Eu-154	0.06	
Cs-137	none	

The projected volumes and activities for all radioactive wastes expected from decommissioning Maine Yankee are summarized in Table 2-30 (Maine LTP 2000, Section 3). Some of these materials have been or will be shipped to an approved processor for handling and/or treatment and disposal, and some will be shipped for direct offsite burial. For example, the reactor vessel and reactor coolant pumps will be shipped to Barnwell, South Carolina, for disposal, while the reactor coolant pump motors will be shipped to Envirocare in Utah. The three steam generators and the pressurizer will be shipped to Duratek, where parts will be melted, cast into shield blocks, and sold to DOE.

A list of all wastes expected to be generated during the Maine Yankee decommissioning, both radioactive and nonradioactive, is presented in Table 2-31 (Maine LTP 2000, Section 3, Table 3). In this table, direct burial offsite is presumed to mean that Maine Yankee plans to send the waste directly to a LLW waste facility rather than to a waste broker.

An updated summary of wastes expected to be generated during dismantlement of Maine Yankee is included in Table 2-32. These revised estimates are taken from Odell's March 2001 presentation to the National Academy of Sciences/National Research Council Committee on Solid Materials from NRC-Licensed Facilities (Odell 2001). The basis for changes in concrete disposition were discussed above. Distributables in the table refer to items such as asbestos and asphalt, while commodities refer to items such as pumps, valves, and ductwork.

The proposed sequence for the dismantlement and demolition of a building with high levels of contamination (Area #1) is as follows (Maine LTP 2000, Section 3):

- Strip, package, and ship commodities from the buildings (piping, steel, steel components, etc.). Commodities determined to be clean, including building steel, may be released to the demolition contractor.
- Perform decontamination of building concrete surfaces to levels below the unrestricted use criteria. Package the debris from decontamination and ship for LLW processing and/or disposal.
- Perform a final survey.
- Release building for demolition.
- Demolish the building structure to three feet below grade. Separate the clean rebar from the concrete.
- "Rubble-ize" the concrete as necessary and use as onsite fill.
- Release the rebar using operational release procedures and ship to demolition contractor.

Obviously, this approach has now been modified to preclude the use of concrete as onsite fill.

Table 2-30. Projected Volumes and Activities of Radioactive Waste from Decommissioning of Maine Yankee

Item	Activity (Ci)	Volume (ft^3)
Reactor Vessel & Internals	2,600,000	11,527
Components	1,600	27,000
Activated Concrete	388	23,000[a]
Debris	0.10	163,000
Radioactive Water	0.03	113,500
Soil	0.01	25,000

a Maine LTP 2000, Section 3, Table 3 quotes a value of 11,400 ft^3. The reason for the difference is not obvious, but may be due to the fact that some activated material which meets volumetric DCGL values will be deposited within lower levels of plant structures as fill.

Table 2-31. Maine Yankee Decommissioning Waste Volume Estimates Per January 2000 License Termination Plan

Waste	Source	Amount	Disposition Path
High-level waste	Spent fuel	1434 fuel assemblies	ISFSI (for interim storage) US DOE (for disposal)
Greater than Class C (GTCC)	Reactor vessel internals Segmentation filters Pre-existing filter	227 ft^3 TBD TBD	ISFSI/USDOE
Large nuclear steam supply components	Reactor pressure vessel (RPV) Non-GTCC RPV hardware RPV head Pressurizer Reactor coolant pumps, motors, & assemblies Steam generators (3)	9,500 ft^3 1,500 ft^3 300 ft^3 2,200 ft^3 4,800 ft^3 20,000ft^3	Direct offsite burial
Scrap metal (not included in above categories)	Radioactive-contaminated metal Non-radioactive contaminated metal	3,100 tons (150,000 ft^3) 6,200 tons (300,000 ft^3)	Approved processor for offsite disposal
Dry active waste (DAW)	Non-metallic trash	13,000 ft^3 (100 tons)	Approved processor
Resin	Liquid radioactive waste processing Spent fuel purification Pre-existing	400 ft^3 150 ft^3 200 ft^3	Approved processor for volume reduction and disposal
Soil	Radioactive areas	25,000 ft^3	Direct burial offsite
Concrete	Scabble residue, activated concrete (some activated concrete will remain) Concrete below unrestricted use criteria	11,400 ft^3 1,200,000 ft^3	Direct burial offsite Onsite fill

Source: Maine LTP 2000, Section 3, Table 3.

Table 2-32. Projected Shipments of Wastes from Dismantlement of Maine Yankee (March 2001 Estimate)

Category	Projected Waste Mass (lb)	Completion (%)
Radioactive Waste		
Concrete	90,000,000	3.0
Commodities	9,679,264	40.0
Distributables	3,000,000	40.3
Large Components	4,586,250	70.1
Non-Radioactive Waste		
Asbestos	400,000	55.2
Other	30,000	36.0
Hazardous	100,000	18.9
Oil	24,000	51.7
Paper/Cardboard	500,000	14.9
Trash	1,250,000	41.9
Concrete	60,000,000	0.1
Soil	5,500,000	71.8
Demolition Debris	7,000,000	31.1
Metal	15,000,000	39.0

The demolition process for buildings on the cold side of the plant (Area #2), which have generally been maintained as radiologically clean (with the exception of a few systems and equipment that may have internal contamination), is as follows:

- Remediate, package, and ship systems, components, and commodities identified with the site characterization report, and assessed and bounded by the site charaterization team. Structural steel in plant buildings will either be surveyed and released for demolition or dismantled for packaging and shipment as material.

- Decontaminate building, as required, to meet established release criteria.

- Perform building radiation surveys and release to contractor for demolition.

- Demolish foundations and structures to specified depth.

- Handle subsurface piping according to final environmental plan.

- Perform final grade.

Buildings, structures, and facilities within Area #3 have no history of contamination and are "probably clean." These will be processed as follows:

- Remove ancillary equipment required for asset recovery (e.g., furniture)
- Perform survey in accordance with established procedures and criteria
- Release to demolition contractor

The demolition contractor will dispose of reinforcing steel and structural steel that have been released as scrap and/or in landfills.

2.2.5.3 Trojan Nuclear Plant

The Trojan Nuclear Plant, located near Rainier, Oregon, is a 3,500 MWt (1,175 MWe), four-loop pressurized water reactor. The reactor was shut down on November 9, 1992, and decommissioning is in progress. Major structures within the protected area at the reactor site include (TNP 1995):

- Containment Building
- Auxiliary Building
- Turbine Building
- Condensate Demineralizer Building
- Maintenance Building
- Control Building
- WSH Warehouse
- Intake Structure
- Central Building
- Main Warehouse
- Technical Support Center
- Security Building
- Radiowaste Storage Building
- Plant Mods Shops
- Main Steam Support Structure
- Steam Generator Blowdown Structure

All of these buildings and structures were included in the site characterization activities needed to support Trojan decommissioning decisions. Buildings and structures outside the protected area were excluded from site characterization activities because:

- No potentially contaminated systems pass through these outside structures

- No significant releases of airborne materials occurred that could have resulted in the deposition of radioactive material on these structures

- There was no indication that radioactive material was transported to these buildings from the protected area

The validity of these suppositions was verified by walk-through surveys with gamma sensitive instruments.

Decommissioning release criteria specified in the Trojan Decommissioning Plan include the following:

- <15 mrem/yr from all pathways for the applicable exposure scenario (per NUREG/CR-5512, *Residual Radioactive Contamination from Decommissioning*, draft report, January 1990)
- <5 μR/hr at 3 feet from any surface of structures or equipment
- Surface contamination levels as described in Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*
- EPA regulation 40 CFR 141, *National Primary Drinking Water Standards*

The first major step in dismantling Trojan was removal of the four steam generators and the pressurizer. These components were removed through an opening cut in the south face of the containment building and shipped to U.S. Ecology near Richland, Washington, for disposal. In 1999, the reactor vessel and internals were also removed and shipped to the same disposal destination. The reactor vessel and internals contained approximately two million curies of radioactivity. This operation resulted in the elimination of more than 99 percent of the activity remaining at the site (Trojan 2001).

Activity levels for Trojan are summarized in Table 2-33 (Trojan 2001). It is clear that very little activity is contained on structures (which include external system surfaces).

Table 2-33. Activity Levels at Trojan Nuclear Plant

Section	Activity (Ci)	Burial Volume (ft^3)
Structures	0.031	59,945
Systems	1070.5[a]	215,789
Activation	4.2×10^{6} [b]	N/A
TOTAL	4.2×10^{6}	--

a Not including steam generators, pressurizer, or activation.

b Most activity is contained in the vessel internals. Activation curies for reactor vessel, clad, insulation, and concrete are approximately 3.1x104 Ci.

N/A - not available

Additional information about Trojan decommissioning is presented in Section 2.2.2.

2.2.5.4 Yankee Nuclear Power Station (Yankee Rowe)

Yankee Rowe, a 600 MWt PWR located in Franklin County, Massachusetts, was shut down on October 1, 1991, and is currently undergoing decommissioning. According to the NRC Order approving Yankee's Decommissioning Plan, "Within the plant, fixed quantities of radioactivity are contained in neutron-activated structures. In addition, within system piping and components, radioactive material is contained in the corrosion film. As a result of leaks and spills that occurred during the operating life of the plant, contamination exists on surfaces of buildings and structures consisting of both fixed and loose contamination products" (NRC 1995).

Table 2-34 summarizes the radioactivity associated with highly contaminated components removed from the site during the Component Removal Program completed early in the decommissioning process. Table 2-35 summarizes the radioactivity associated with other plant systems, structures, and components after completion of the Component Removal Program.

Surface contamination levels in systems and structures are summarized in Table 2-36. The range of contamination levels for buildings is for the contaminated areas only, not the entire building. The indicated system contamination levels are averaged over the entire system. From 20 to 50 percent of the total activity is loose and can be removed by wiping. Cobalt-60 is the most significant contributor to gamma activity (77 percent of total gamma) (YNPS 1993).

Table 2-34. Radioactive Contamination Eliminated from Yankee Rowe During Component Removal Program

Component Removed	Radioactivity (MBq)	(Ci)
Reactor Internals	4.9e+9	132,500
Miscellaneous Waste	1.15e+8	3,100
Steam Generators (4)	4.82e+7	1,302
Reactor Coolant Pumps (5)	4.44e+5	12
Pressurizer	1.11e+5	3

Table 2-35. Radioactive Inventory at Yankee Rowe

Location	Radioactivity (MBq)	(Ci)
Reactor Internals*	3.5e+10	943,500
Reactor Vessel	11.74e+8	4,700
System Components	6.48e+6	175
Neutron Shield Tank	6.7e+6	180
Misc. Drums & Solid Wastes	1.5e+6	41
Misc. Tools & Equipment	1.2e+6	39
Instrument Calibration Sources	1.18e+6	32
Water in SFP & IX	1.85e+5	5
Contaminated Concrete	<3.7e+4	<1
Contaminated Steel	<3.7e+4	<1

* Greater than Class C, stored in Spent Fuel Pool (SFP)

Table 2-36. Surface Contamination Levels for Yankee Rowe Buildings and Systems

Source	Surface Contamination (dpm/100 cm^2)
Main Coolant System	7.1e+9
Spent Fuel Cooling System	3.3e+8
Charging & Volume Control System	1.2e+7
Waste Disposal System	1.2e+7
Primary Auxiliary Bldg.	5e+2 to 5e+4
Vapor Container	1e+3 to 1e+4
Waste Disposal Bldg.	2e+2 to 9e+3

The estimated volumes of solid radioactive waste expected to be generated during dismantlement (not including items in Table 2-34) are listed in Table 2-37. The total waste volume is expected to be about 2,500 m^3.

Of the items in Table 2-37, 39 Ci of activity are associated with "miscellaneous tools, equipment, duct and conduit" and 175 Ci are associated with "contaminated system components" (YNPS 1993).

Table 2-38 summarizes system average contamination levels (YNPS 1993). For those systems with both internal and external contamination, the quantified contamination is for internal surfaces.

Table 2-37. Estimated Volume of Class A Radioactive Waste Generated During Yankee Rowe Dismantlement

Source	Activity (Ci)	Volume (m^3)
Reactor Vessel	4700	117.8
Fuel Racks		240.1
Neutron Shield Tank	180	42.9
Main Coolant Piping & Supports		69.9
Pipe, Tubing & Supports		389.2
Valves		114.3
Mechanical Equipment		212.7
Tanks		120.2
Duct & Supports		59.8
HVAC Equipment		260.2
Cable, Conduit, Cable Trays & Supports		188.2
Electrical Equipment		1.5
Concrete	0.43	261.4
Structural Steel[a]	0.03	37.5
Misc. Materials (e.g., tools, equipment)		202.5
Drums (solidified waste)	41	162.2
PCA Warehouse No. 1 Material		18.9

a Estimate 15 percent of total to be buried.

Table 2-38. System Average Contamination Levels for Yankee Rowe

System	dpm/100 cm^2	$\mu Ci/cm^2$	Internal Contamination	External Contamination
Steam Generator Blowdown	1.0e+3	4.5e-6	X	X
Compressed Air	1.0e+4	4.5e-5		X
Component Cooling	1.0e+3	4.5e-6	X	X
Charging and Volume Control	1.2e+7	5.3e-2	X	X
Containment Isolation	1.0e+3	4.5e-6		X
Primary Plant Corrosion Control	1.2e+4	5.4e-5		X
Chemical Shutdown	1.1e+4	5.0e-5	X	X
Demineralized Water	5.0e+3	2.3e-5		X
Emergency Feedwater	1.0e+3	4.5e-6		X
Fuel Handling	1.7e+6	7.8e-3	X	X
Spent Fuel Cooling	3.3e+8	1.5e+0	X	X
Fire Protection	1.0e+3	4.5e-6		X
Feedwater	1.0e+3	4.5e-6		X
Heating	1.0e+3	4.5e-6	X	X
Ventilation	5.0e+3	2.3e-5	X	X
Main Coolant	7.1e+9*	3.2e+1	X	X
Main Steam	1.0e+3	4.5e-6		X
Miscellaneous Vent and Drain	5.0e+3	2.3e-5	X	X
Primary Pump Seal Water	5.0e+3	2.3e-5		X
Pressure Control and Relief	1.0e+4	4.5e-5	X	X
Purification	1.4e+6	6.1e-3	X	X
Primary Plant Sampling	1.4e+6	6.1e-3	X	X
Shutdown Cooling	1.2e+7	5.3e-2	X	X
Safety Injection	1.4e+5	6.5e-4	X	X
Safe Shutdown	1.4e+5	6.5e-4	X	X
Service Water	5.0e+3	2.3e-5		X
Primary Plant Vent and Drain	1.2e+7	5.3e-2	X	X
Post Accident Hydrogen Control	1.2e+4	5.4e-5	X	X
VC Heating and Cooling	1.2e+4	5.4e-5	X	X
VC Ventilation and Purge	1.2e+4	5.4e-5	X	X
Water Cleanup	5.3e+5	2.4e-3		X
Waste Disposal	1.2e+7	5.3e-2	X	X

Data on concrete surface contamination are included in Table 2-39.

Table 2-39. Concrete Average Contamination Levels in Yankee Rowe

Building	Description	$dpm/100 cm^2$	Depth (mm)
Diesel Generator Building	SI Pump Room	200	5
Diesel Generator Building	SI Accumulator Room	200	5
Primary Auxiliary Building (PAB)	Gravity Drain Tank Cubicle	2500	5
PAB	Primary Building Sump Cubicle	2000	5
PAB	Primary Drain Collecting Tank Cubicle	50000	5
PAB	Lower Level	0	0
PAB	Cubicle Corridor	500	5
PAB	Low Pressure Surge Tank Pump Cubicle	2000	5
PAB	Shut Down Cooling Heat Exchanger Cubicle	2000	5
PAB	Shut Down Cooling Heat Exchanger Pump Cubicle	2000	5
PAB	Low Pressure Surge Tank Heat Exchanger Cubicle	2000	5
PAB	No. 1 Charging Pump Cubicle	2000	15
PAB	No. 2 Charging Pump Cubicle	2000	15
PAB	No. 3 Charging Pump Cubicle	2000	15
PAB	No. 1 Purification Pump Cubicle	2000	15
PAB	No. 2 Purification Pump Cubicle	2000	15
PAB	Pipe Trench	50000	15
PAB	Upper Level	0	0
PAB	Valve Room	500	5
PAB	Vertical Pipe Chase	5000	5
PAB	Lower Pipe Chase	500	5
PAB	Low Pressure Surge Tank Cubicle	2000	5
PAB	Chemistry Sample Room	1000	5
Potentially Contaminated Area Storage Building	Potentially Contaminated Area Storage Building 1	5000	15
Potentially Contaminated Area Storage Building	Potentially Contaminated Area Storage Building 2	500	5
Service Building (SB)	North Decon. Room	1000	5
SB	South Decon Room	5000	5
SB	Primary Side Machine Shop	0	0
SB	Welding Booth	0	0
SB	Tool Decontamination Room	0	0
SB	Primary Side Chemistry Lab	0	0
SB	Rad Protection Calibration Lab	0	0
Spent Fuel Building	Spent Fuel Pit	10000	102
Spent Fuel Building	New Fuel Vault	0	0
Vapor Container (VC)	Shield Tank Cavity	10000	5

Table 2-39. Concrete Average Contamination Levels in Yankee Rowe (continued)

Building	Description	dpm/100 cm²	Depth (mm)
VC	Charging Floor	1000	5
VC	Loop 1	10000	5
VC	Loop 2	10000	5
VC	Loop 3	10000	5
VC	Loop 4	10000	5
VC	Pressurizer Cubicle	10000	5
VC	Brass Drain Box	10000	15
VC	Feed and Bleed Heat Exchanger Cubicle	10000	5
VC	Equipment Hatch Area	10000	5
VC	Broadway	1000	5
VC	Shell Area	10000	5
Waste Disposal Building (WD)	Gas Compressor Room	0	0
WD	Drumming Pit	1000	15
WD	Corridor Area	200	5
WD	Liquid Water Transfer Pump Cubicle	200	5
WD	Evaporator Cubicle	4000	32
WD	Stripper Cubicle	9000	32
WD	Sump Room	500	15
Compactor Building	Compactor Building	500	5
Yard Area in RCA	Ion Exchange Pit	530000	152
Yard Area in RCA	IEP Pipe Tunnel	20000	5
Yard Area in RCA	Fuel Chute	20000	102
Yard Area in RCA	Tank Farm Area	5000	5

2.2.5.5 Rancho Seco

Rancho Seco is a 2,772 MWt PWR located in Herald, California. The reactor was shut down on June 7, 1989, and the SAFSTOR decommissioning alternative was selected. However, dismantlement of contaminated secondary system piping and components is ongoing (NRC 2000). The diesel generators have been removed and sold to a business in China (Susnjara 1997).

Some of the Rancho Seco decommissioning cost study assumptions relevant to clearance issues are (TLG 1991):

- The switchyard remains intact for use by the balance of the utility's electrical distribution system. Transmission towers remain in place.

- No credit is included for scrap generated during decommissioning because (1) the scrap value merely offsets the site removal and scrap processing costs, and (2) scrap has a relatively low value in the market.

- Decommissioning will take place sufficiently far in the future that no equipment is salvageable as used equipment.

As part of its ongoing dismantlement program, Rancho Seco decided to capitalize on the opportunity to dispose of low-activity waste at Envirocare's facility in Utah at a favorable cost (Gardiner and Newey 1999). The Turbine Building was selected for initial dismantlement because large volumes of low-activity waste were available. Some contamination existed in that building due to leakage from the primary coolant system. Areas of expected contamination in the Turbine Building included the turbine plant cooling water system, auxiliary steam, first- and second-point heaters, reheaters, and the turbine itself. The high-pressure turbine and the moisture separator/reheaters had fixed contamination of up to 50,000 counts per minute and loose contamination of up to 20,000 dpm/100 cm^2 in isolated locations. About 6 million pounds of scrap from the third, fourth, fifth, and sixth point heaters, a major portion of the condenser and most of the auxiliary boilers, and the outer turbine covers were released based on radiological survey data. During the course of the program, a few scrap bins containing material that had been monitored and shown to pass the free-release criteria exhibited detectable contamination levels above background when monitored in aggregate. The problem was remedied by improved surveying and training procedures, as well as use of a truck monitor as a final check on materials leaving the site.

In addition to material that was surveyed and released, approximately 2 million pounds of material was decontaminated in an onsite grit-blasting facility and then sent for recycling. Some components that were mostly clean, but not easily decontaminated or surveyed, were sent offsite for processing if the economics appeared to be favorable. Some components were transferred to other licensees thereby avoiding disposal costs. Components in this category included the high-pressure turbine rotor and two moisture separator/reheater tube bundles. In addition, pumps and motors from the Auxiliary Building are being sent to a vendor who will refurbish them and provide them to other plants. The balance of the waste (approximately 12,000 ft^3) generated during this phase of the dismantlement was shipped for disposal at a licensed facility (Gardiner and Newey 1999).

2.2.6 Clearance of Materials from Operating Nuclear Power Plants

NRC regulations under 10 CFR 20.2001 limit the manner in which a licensee can dispose of licensed material. All other disposal options are precluded. If material is to be cleared from licensed control, the licensee must be sure that the cleared material does not contain any licensed material as defined by the regulations.

In spite of the difficulty in demonstrating that material for clearance does not contain licensed material, many items are routinely cleared from nuclear power plants during the course of normal operations. In 1981, NRC issued IE Circular No. 81-07, *Control of Radioactively Contaminated Material*. This circular stated that items or material should not be removed from the restricted area at nuclear power facilities unless surveyed with a portable survey instrument or a laboratory instrument capable of detecting 5,000 dpm/100 cm^2 of total beta/gamma contamination or 1,000 dpm/100 cm^2 of removable beta/gamma. In 1985, the NRC issued IE Information Notice No. 85-92, *Survey of Wastes Before Disposal from Nuclear Reactor Facilities*. This information notice was designed to address concerns that hand-held pancake G.M. probes were not ideal for scanning large surface area items, such as paper and plastics. The Information Notice defined a good monitoring program, which would preclude unintentional release of radioactive materials, as follows:

> *1. Careful surveys, using methods (equipment and techniques) for detecting very low levels of radioactivity, are made of materials that may be contaminated and that are to be disposed of as clean waste. These surveys should provide licensees with reasonable assurance that licensed material is not being released from their control.*
> *2. Surveys conducted with portable survey instruments using pancake G.M. probes are generally more appropriate for small items and small areas because of the loss of detection sensitivity created by moving the probe and the difficulties in completely scanning large areas. This does not preclude their use for larger items and areas, if supplemented by other survey equipment or techniques.*
> *3. Final measurements of each package (e.g., bag or drum) of aggregated wastes are performed to ensure that there has not been an accumulation of licensed material resulting from a buildup of multiple, nondetectable quantities (i.e., final measurements using sensitive scintillation detectors in low background areas).*

2.2.6.1 Green Is Clean Program

Duratek, Inc., conducts various licensed processing operations for radioactive waste in Tennessee. One such licensed process is called the Green Is Clean (GIC) program. The objective of the GIC program is to monitor and potentially clear solid waste materials generated within radiologically controlled areas of facilities. Material that is cleared is buried in a permitted industrial landfill (Johnson 2000). Waste streams handled under the GIC program include trash, soils and sludges, junk metals, and concrete and demolition debris (Johnson 2000a). Under the GIC program, individual containers of waste are surveyed for gross radioactivity using handheld instruments. The waste is then assayed in bulk (drums or waste boxes) for all individual gamma-emitters potentially present. Any contamination-free waste containing radioactive symbols, signs, or labels is shredded to make the symbols unidentifiable. The waste is statistically sampled for QC purposes, and the samples are assayed with laboratory instruments. Acceptable waste is placed in roll-off containers and receives a final radioactivity check via a truck monitoring system prior to shipment to the landfill for disposal. The GIC program, which has been in operation since 1992, has processed 8 million pounds of waste from commercial plants, byproduct licensees, and DOE facilities.

2.2.6.2 Case-by-Case Disposal under 10 CFR Part 20.2002

The NRC currently permits disposal of slightly contaminated solids under 10 CFR Part 20.2002 (formerly 10 CFR 20.302) based on case-by-case review and approval of the proposed disposal by the Commission. As of the end of 1993, the NRC had approved 30 requests for case-by-case disposal from nuclear power plants (Minns 1994). The principal radionuclides involved were Co-58, Co-60, Cs-134, and Cs-137, with activities ranging from about 1 to 50 pCi/g. A summary of the approved case-by-case disposals is included in Table 2-40 The breakdown by types of waste for both onsite and offsite disposal is included in Figure 2-2 (reproduced from Minns 1994). The NRC is no longer involved in case-by-case disposal from reactors in Agreement States.

In the last five years, the NRC has granted approval for five additional instances of onsite disposal of soils and sludges from nuclear power plants (Klementowicz 2001).

Table 2-40. Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants under 10 CFR 20.2002

Reactor	Date Rec.	Date Completed	Lic. Tac.	Waste Character/ Volume (m^3)	Proposed Disposal	Nuclides Present	Total Activity
San Onofre	7/16/81	9/24/81	50206	Sand 300	Onsite	Cs-137	0.2 mCi
Oyster Creek	10/12/82	11/16/82	50219	Contaminated soil 480	Onsite	Co-60 Cs-137 Mn-54 Cs-134	5 mCi
D.C. Cook	2/29/88	8/30/88	67788	Contaminated concrete, steam generator replacement 653	Onsite	Co-60 Cs-134 Cs-137	0.1 mCi
Vermont Yankee	6/28/89	8/30/89	73766	Septic waste 262	Onsite	Co-60 Mn-54 Cs-137 Cs-134 Zn-65	0.2 mCi per acre
Yankee Rowe	4/11/90	5/17/90		Sewage 200 once every 1 to 2 years for 30 years	Onsite	Co-60 Mn-54 Cs-134 Cs-137	0.2 mCi
Big Rock Point	12/29/89	8/24/90	75589	Dredging spoils 15 yr	Onsite	Co-60 Mn-54 Cs-137 Cs-134 Sr-90	0.9 mCi
Palisades	11/12/87	3/21/97	67408	Soil 170	Onsite	Co-60 Cs-137	0.03 mCi
Maine Yankee	4/26/90	4/18/91	71250	Hazardous chemical solution 40	Offsite	Co-60 Zn-65 Cs-137	0.1 mCi
Sequoyah*	12/23/85	12/7/87	00179/80	Trash 750	Offsite		200 mCi

Table 2-40. Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants under 10 CFR 20.2002 (continued)

Reactor	Date Rec.	Date Completed	Lic. Tac.	Waste Character/ Volume (m^3)	Proposed Disposal	Nuclides Present	Total Activity
Fermi-2	5/26/87	3/14/88	65459	Contaminated soil 850	Onsite	Cr-51 Mn-54 Co-58 Co-60	0.3 mCi
Kewaunee	9/12/89	6/17/92	75047	Waste sludge 454	Onsite	Co-60 Cs-137	0.2 mCi
Brunswick	10/4/90	12/11/91	81827/25	Dredging sediments, sand	Onsite	Mn-54 Co-60 Cs-137	
Point Beach 1,2	10/8/87	1/13/88	65821/22	Sewage sludge 113	Onsite	Co-60 Cs-137	0.003 mCi
Surry 1,2	11/26/86	4/9/87	64191/92	Soil 300	Onsite	Co-60 Cs-134 Cs-137 Mn-54	72
H.B. Robinson	2/10/83		40447	Sediment 6,000	Fossil plant ash pond in licensee's controlled area	Co-58	75
H.B. Robinson	4/28/83		51347	Soil, 50 cu ft 1.5 cu meters	Onsite along the bottom of a drainage ditch	Co-58 Co-60 Co-134 (10%) Cs-137 (23%) Mn-54 all nuclides	0.014
Humboldt Bay 3	10/27/83		52637	Sludge 1,300 cu feet	Offsite RCRA chemical waste disposal landfill (Martinez, CA)	Co-60 Cs-134 Cs-137 Th-234	267.8 3.1 155 19.3
Oconee Units 1,2,3	7/19/84		55264 55265 55266	Sewage sludge 4,000 cu ft	Offsite sanitary landfill	Co-58 Co-60 (27%) Cs-134 Cs-137 (45%) all nuclides	0.07

Table 2-40. Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants under 10 CFR 20.2002 (continued)

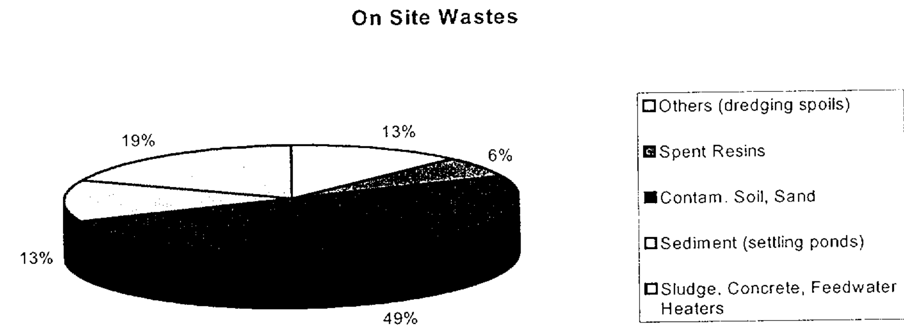
Reactor	Date Rec.	Date Completed	Lic. Tac.	Waste Character/ Volume (m^3)	Proposed Disposal	Nuclides Present	Total Activity
H.B. Robinson	10/17/84		54484	Setting pond sediment 60,000 cu meters	Onsite fossil ash pond	Co-60	1700 (over life of pond)
R.E. Ginna 1	10/17/84		54509	Roofing materials <100 tons	Offsite municipal landfill	Co-60 Cs-134 Cs-137	0.30 0.23 0.92
McGuire 1,2	10/18/84		55306	Wastewater residue sludge <13,000 cu ft	Onsite	Co-58 Co-60	0.05 0.05
Oconee 1,2,3			55832 55833 55834	Feedwater heater a. high activity tube bundles 160 tons b. very low activity heater shells 100 tons	Company controlled area (outside security fence) Onsite	Co-60 (79%) Cs-137 (15%) Co-60 (80%)	6.5
Oconee 1,2,3	1/31/85		55509 55510 55511	Sand 1500 cu ft 45 cu m	Onsite company controlled area (outside security fence)	Cs-134 Cs-137 Co-60 Mn-54 all nuclides	1.2 3 0.1 0.005 <12.3
Big Rock Point		5/8/86		Contaminated soil leak in condensate process monitor	Onsite retain soil in place		
Davis-Besse	3/11/85	10/15/85	52484	Secondary side resins 5,000 cu ft or 150 cu in every 5 years	Offsite company owned	Co-58 (34%) Co-60 (3%) Cs-134 (27%) Cs-137 (36%)	8.5/every 5 yrs
Oconee 1,2,3 McGuire 1,2 Catawba 1,2	2/7/85		55056 55057 55058 55058 55058	Wood 400-700 cu ft 12-21 cu m	Offsite sanitary landfill	Assume Cs-137 100%	0.4 to 0.7 per yr per station
Pilgrim	1/15/93	4/8/93	85501	Contaminated soil 2,238	Onsite in place	Co-60 Cs-137	0.19 mCi 0.4

Table 2-40. Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants under 10 CFR 20.2002 (continued)

Reactor	Date Rec.	Date Completed	Lic. Tac.	Waste Character/ Volume (m^3)	Proposed Disposal	Nuclides Present	Total Activity
D.C. Cook	10/9/91	12/16/93	81885 81886	Contaminated sludge	Onsite pre-burial	Cs-137 Cs-134 Co-60 I-131	8.89 (1982) 5.02 (1991)

* Rejected due to high specific activity and total activity

Source: Minns 1994



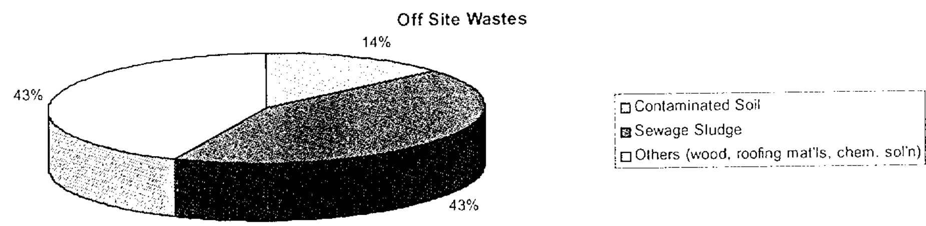


Figure 2-2. Types of Waste Approved for Disposal from Nuclear Power Plants under 10 CFR 20.2002

NRC guidelines for making case-by-case disposal decisions are as follows (quoted from Minns 1994):

1. *The radioactive material should be disposed of in a manner such that it is unlikely that the material would be recycled.*

2. *Doses to the whole body and any body organ of a maximally exposed individual (a member of the general public or a non-occupationally exposed worker) from the probable pathways of exposure to the disposed material should be less than 1 mrem/yr.*
3. *Doses to the whole body and any body organ of an inadvertent intruder from the probable pathways of exposure should be less than 5 mrem/yr.*
4. *For onsite disposal, the dose to the whole body and any body organ of an individual from assumed recycling of the disposed material at the time the disposal site is released from regulatory control from all pathways of exposure should be less than 1 mrem/yr.*
5. *For disposal in a sanitary landfill, the dose to the whole body and any body organ of an individual from assumed recycling of the disposed material at the time of disposal from all likely pathways of exposure should be less than 5 mrem/yr.*

Generally, case-by-case approvals are granted for a single disposal, but repetitive disposals are also allowed with certain limitations on cumulative activity in the disposed waste.

The ensuing paragraphs provide additional discussion of specific case-by-case approvals made by the NRC.

Sewage Sludges

In 1988, the NRC issued Information Notice No. 88-22 covering *Disposal of Sludge from Onsite Sewage Treatment Facilities at Nuclear Power Stations*. The Information Notice indicates that onsite sewage treatment plant sludges may be contaminated with licensed radioactive material which is concentrated by the treatment system. Contaminated sludges from onsite facilities are not covered under 10 CFR 20.303, which allows a licensee to discharge radioactive material into sanitary sewage systems under certain conditions. However, the licensee can apply for permission to deposit sludges onsite under the provisions of 10 CFR 20.302. (This section of the regulations is now designated 10 CFR Part 20.2002.)

Based on this guidance, the Point Beach Nuclear Power Plant obtained approval for onsite disposal of sanitary sewer sludges with very low levels of radioactivity. The primary source of the contamination was wash basins within the controlled area of the plant. About 30,000 gallons of sludge are disposed of annually by spreading on land in areas approved by the Wisconsin Department of Natural Resources and controlled by Wisconsin Electric Power Co.

Radionuclides identified in the sludges include Co-60 (0.233 pCi/cc) and Cs-137 (0.150 pCi/cc) (Fay 1987).

Vermont Yankee made a similar application requesting permission to dispose of slightly contaminated septic tank waste by land spreading onsite in designated areas complying with the State of Vermont health code requirements for septic waste (Capstick 1989). The disposal practice assumed by Vermont Yankee is that the septic tanks would be pumped out twice yearly for 20 years and the septage (700 kg per batch) would be deposited on a single 2-acre plot. The upper bound (three sigma) concentrations (pCi/dry kg) for various radionuclides are as follows:

Mn-54 - 1,348
Co-60 - 23,060
Zn-65 - 1,620
Cs-134 - 322
Cs-137 - 4,100

This proposed onsite land disposal was approved by the NRC on August 30, 1989 (Fairtile 1989).

Sand

Another example of permitted onsite disposal involved slightly contaminated sand at the Duke Power Company Oconee Nuclear Station (Muller 1985). From 300 to 1,500 ft^3 of sand are used annually in sand-blasting operations associated with decontamination. NRC authorized disposal of 1,500 ft^3 of sand per year containing no more than 150 pCi/g (from Cs-134, Cs-137, Co-58, Co-60, and Mn-54) in a trench 7 to 12 feet deep outside the security fence, but within the company controlled area. The sand would be covered with 3 feet of soil.

Concrete

Another example of case-by-case disposal permitted under 10 CFR 20.302 (now 10 CFR 20.2002) involved slightly contaminated concrete from the D.C. Cook Unit No. 2 in Berrien County, Michigan (Stang 1988). In order to obtain access through the containment structure to replace four steam generator lower assemblies, it was necessary to remove large sections of the reinforced concrete doghouses surrounding the steam generators. The surfaces were slightly contaminated with airborne Co-60, Cs-134, and Cs-137, which had been deposited on painted surfaces and had diffused through the paint and into the concrete. The utility operator proposed that the paint and a 1/16-in layer of underlying concrete be removed by a mechanical scarifying process and 24 to 30 large slabs, weighing 25 to 70 tons each, be disposed of with similar rubble onsite, but outside the protected area fence. The scarified material would be disposed of in a licensed LLW facility. The total material for disposal was estimated to weigh 920 tons, including 65 tons of rebar. The measured surface concentrations in a 1/16-in surface layer, after removal of the paint and the initial 1/16-in layer of concrete, are listed in Table 2-41.

Table 2-41. Analysis of Concrete Surfaces after Removal of Surface Layers

Sample Location	Radionuclides	Activity (pCi/g)
Top of Doghouse Enclosure		
Set #1	Co-60	1.90
	Cs-134	0.70
	Cs-137	7.70
Set #2	Co-60	0.20
	Cs-134	0.20
	Cs-137	7.70
Walls of Doghouse Enclosure		
Set #1	Co-60	2.70
	Cs-134	0.40
	Cs-137	0.90
Set #2	Co-60	0.50
	Cs-134	<LLD
	Cs-137	0.70

Using the surface concentrations, and assuming that diffusion can be described by an exponential (Fick's Law), the average activity in the concrete destined for onsite disposal was estimated to be 25.1 μCi. The concrete was determined to have little or no recycle value. The utility proposal was accepted by the NRC.

Cooling Tower Sludge

Cooling tower sludge is a mixture of aqueous and solid waste, a flowable solid, consisting of water, algae, dust and chemical precipitates which settle to the bottom of the cooling tower. This sludge is generally not considered radioactive but may exhibit detectable levels of radioactivity if radiation leaks into the cooling water system. Low levels of radioactivity may also result from background sources (i.e., fallout, naturally occurring radiation). The quantity and characteristics of the sludge deposited depend in part on the characteristics of the water source, including the contaminants it contains and the need for any additives to improve its quality (e.g., to prevent corrosion). The nature of the particular plant also affects sludge generation; not all nuclear power plants operate cooling towers, and radioactivity levels may be less likely at PWRs than BWRs due to plant configuration.

Sludge accumulation is a slow and variable process, and treatment may involve a variety of waste management techniques (HydroChem 2001). Sludge may be removed annually, every five to ten years, or even more infrequently. Additionally, some reactors filter their cooling water to remove some of the materials which eventually deposit themselves as sludge.

Prior to disposal, the sludge often undergoes several treatments. These treatments may include, for example, placing the sludge in a holding tank to precipitate the solids and consume the algae, then de-watering the material. The sludge may be land-applied on site or disposed of elsewhere depending on whether it is defined as radioactive or hazardous (e.g., due to the presence of toxic metals) waste.

Little information is available in the literature on the quantities of cooling tower sludge generated or its radiological characteristics, which (as noted above) may vary significantly. Information on sludge quantities and radiologic characteristics from two sources is reported below.

Rain for Rent. Rain for Rent is a for-profit company, specializing in the rental of equipment for water containment or storage. On its website, the company provides information on work performed for an environmental remediation company to provide equipment for the removal of cooling tower sludge from a nuclear power plant in Louisiana (Rain for Rent 2001). Rain for Rent estimated that the base of each of the four cooling towers was 250 feet in diameter and was covered in 3 to 4 feet of sludge. Therefore, these towers each contained approximately 172,000 cubic feet of wet sludge each, for a total of roughly 687,000 cubic feet. Rain for Rent staff indicate that the resulting wastes were not radioactive.

PECO Energy — Limerick Generating Station. In 1995, PECO Energy (PECO) submitted an application to the NRC for permission to dispose of slightly contaminated material on-site from Limerick Units 1 and 2 in accordance with 10 CFR 20, Subpart K, Section 2002 (Hunger 1995a, 1995b, Rinaldi 1997). PECO received permission to dispose of a maximum of 1.12 million cubic feet of soil, sediment, and sludge over a 16-year period. These wastes include materials from the site settling basin, emergency spray pond, and cooling tower basins; cooling tower sludge is expected to be the dominant source of materials. The material will be placed within a 1.5-acre plot (in the Site Restricted Area, but not in a sector posted as a radiation area), graded, and seeded with grass.

Dose assessment performed by the utility indicates that exposure will be well below the levels in 10 CFR Part 20 and 40 CFR Part 190. For example, the maximum dose to a member of the public was calculated as not exceeding 1.82×10^{-4} mrem per year total effective dose equivalent, based on conservative assumptions. Actual doses are expected to be lower. The material is expected to decay to nondetectable levels before the plant is decommissioned.

Limerick Unit 1 went online in 1986, and, in 1991, 68,000 cubic feet of de-watered cooling tower sludge were removed. The next cleaning was expected to occur in 1998. Limerick Unit 2 went on-line in 1991 and, as of November 1995, no sludge had been removed. According to PECO, the unit seems "less prone to flowable solids buildup and has never required cleaning" (Hunger 1995b, Attachment 2, p. 6).

PECO notes that "[l]ittle or no radioactivity has been found in these solids in the past, and they have been disposed of as non-radioactive, non-hazardous wastes" (Hunger 1995b, Attachment 2, p. 7). To support its application, PECO analyzed gamma emissions from 11 samples of holding pond material identified for disposal (the total amount of material was 8,000 cubic feet) (Hunger 1995a, p. 2). In addition to naturally occurring radionuclides (Be-7, K-40, Ra-226, and Th-228), statistically positive results were found for the presence of Mn-54, Co-60, and Cs-137. PECO notes that the Cs-137 levels are consistent with background levels in the area from previous weapons testing. Table 2-42 shows the range and the average activity for each of these latter nuclides in microcuries/gram (dry) for the holding pond; similar information specifically on cooling tower sludge is not reported.

Table 2-42. Radioactive Isotopes in Limerick Holding Pond Sludge (microcuries/gram (dry))

Nuclide	Average	Minimum	Maximum
Mn-54	1.8e-08	5.0e-09	4.0e-08
Co-60	1.14e-07	4.0e-08	2.2e-08
Cs-137	3.7e-08	3.8e-09	6.0e-08

Source: Hunger 1995a, p. 5, Table 1.

2.2.6.3 Low-Level Waste (LLW) Disposal

In addition to specifically permitted onsite disposals, large quantities of materials are shipped to licensed LLW facilities for disposal. For example, in 2000, according to the Manifest Information Management System at INEEL, utilities shipped 276,368 ft^3 of LLW containing 733,720 Ci of activity to licensed disposal facilities (http://mims.inel.gov/web/owa/gentype.report). Of this total, 8,896 ft^3 of waste containing 357,523 Ci was shipped from Pennsylvania. Most of the activity was attributable to the waste of two utilities. Based on 1998 data, the five principal radionuclides in order of decreasing activity were Co-60, Fe-55, Ni-63, Mn-54, and Cs-137 (Fuchs 1999). Typical waste from the nuclear utilities included spent resins, evaporator bottoms and concentrated waste, filter sludge, dry compressible waste, irradiated components, and contaminated plant hardware. If the clearance criteria were changed, these quantities could be significantly altered.

In support of the current study, the Manifest Information Management System staff kindly provided an Excel spreadsheet summarizing all shipments from nuclear utilities by weight for the year 2000 (Fuchs 2001). Figure 2-3 shows the distribution of total activity, Co-60 activity, and Cs-137 activity versus cumulative mass. From this figure, it can be seen that, at the low end, about 10 percent of the mass has a specific activity of less than 7.6×10^{-10} Ci/g. In the case of Co-60, on the low end, 10 percent of the mass has a specific activity of 41 pCi/g or less. The dose conversion factor developed in Draft NUREG-1640 for Co-60 in steel is 250 µSv/a per Bq/g (0.925 mrem/y per pCi/g) and an exposure of 1 mrem/y is equivalent to 316 dpm/100 cm^2, based on an average metal thickness of 0.17 cm (NRC 1999, Table 4.11). Thus, exposure limits for Co-60 based on Regulatory Guide 1.86 (i.e., 5,000 dpm/100 cm^2) are equivalent to a dose rate of about 16 mrem/y (160 µSv/a) for thin gauge steel. This comparison suggests that the possible regulatory options being evaluated by the NRC (e.g., dose rates of 1, 10, and 100 µSv/a) may be more restrictive than Regulatory Guide 1.86 and could increase the amounts of materials from nuclear power plants required to be disposed of as LLW.

Unfortunately, Envirocare does not include shipment weights on its manifests, so the data used for Figure 2-3 do not include disposal at that facility. Since shipments to Envirocare have historically been low-activity waste, the contribution of the low end of the waste spectrum may be understated in the figures.

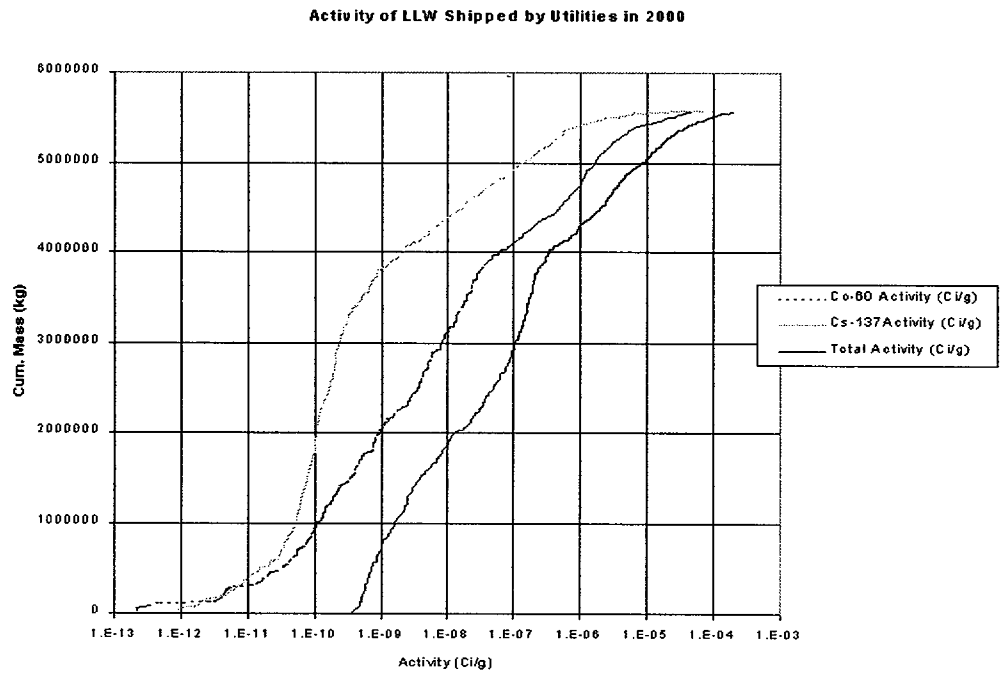


Figure 2-3. Activity of LLW for Disposal by Nuclear Utilities in 2000

The Electric Power Research Institute (EPRI) conducted an extensive analysis of low-level wastes with very low levels of contained radioactivity (EPRI 1989). The estimated mass and volumes of these wastes are summarized in Table 2-43. This table shows that Dry Active Waste (DAW) constitutes about 84 percent of this low activity waste.

Table 2-43. Estimated Annual Mass and Volumes of Radioactive Waste from Nuclear Power Reactors with Low Levels of Contamination

Waste Type	Volume (ft^3/yr)	Mass (lb/yr)
Dry Active Waste	617,300	22,800,000
Oil	50,000	800,000
PWR Secondary Side Resins	26,000	1,000,000
Soil (includes pond and sewage sludge & grit blast material)	25,000	1,900,000
Resin	18,000	720,000
TOTAL	736,300	27,220,000

Source: EPRI 1989, Table 2-6

DAW consists of paper, plastics, rubber cloth, etc. The results of a 1986 EPRI survey of 84 reactors to determine the physical nature of compactible DAW are presented in Table 2-44.

Table 2-44. Industry Average Composition of Compactible DAW Based on 1986 EPRI Survey (Volume %)*

Material	PWR	BWR
Plastic	39	51
Paper	17	19
PVC	12	4
Cloth	10	6
Rubber	6	4
Absorbent Material	4	4
Wood	3	3
Metal	3	2
Filters	2	1
Noncompactible Waste	1	2
Glass	<1	1
Miscellaneous	4	4

* May not sum to 100% due to round-off.
Source: EPRI 1989, Table 2-4

To more fully characterize DAW, EPRI conducted a survey of waste from 10 reactors including 6 PWRs and 4 BWRs. Plants were selected to insure that waste was from plants with both extensive quantities and minimal quantities of tramp fuel from failed fuel elements in the primary reactor systems. The distribution of gamma-emitting nuclides from each reactor, based on gamma spectroscopy, is summarized in Table 2-45.

To characterize the contamination distribution in the DAW, bags of DAW from each of the 10 plants were opened, and each piece of waste was surveyed with a handheld detector. The surveyed pieces of waste were placed into separate bags, based on the following contamination levels:

category	dpm per 100 cm^2	category	dpm per 100 cm^2
1	0 - 1000	6	25,000 - 37,500
2	1000 - 5000	7	37,500 - 50,000
3	5000 - 10,000	8	50,000 - 62,000
4	10,000 - 20,000	9	>62,000
5	20,000 - 25,000		

Table 2-45. Fractional Distribution of Gamma-Emitting Radionuclides in DAW (%)

Nuclide	PWR Reactors						PWR Average
	A	B	D	AAA	BBB	L	
Co-60	8.5	42	21	97	83	44	49
Co-58	7.9	10	29	0.4	3.9	7.8	9.8
Cs-134	16	8.7	17	0.3	0.2	9.0	8.6
Cs-137	62	26	24	1.0	9.6	38	27
Mn-54	0.7	1.3	0.3	0.1	1.1	0.8	0.7
Ru-106	2.2	4.7	--	0.5	--	--	2.4
Sb-125	0.7	0.8	--	0.3	0.9	0.1	0.6
Nb-95	0.9	2.8	7.9	--	0.5	--	2.4
Zr-95	0.2	0.7	1.2	--	--	--	0.7
Ag-110m	--	0.2	0.1	0.1	1.3	0.1	0.3

Nuclide	BWR Reactors				BWR Average	All Plants Average
	B	C	D	H		
Co-60	88	75	80	80	78	61
Co-58	0.6	0.2	--	1.1	0.6	6.7
Cs-134	1.3	2.7	4.1	1.9	2.5	6.1
Cs-137	11.8	11.8	11.1	7.2	10	20
Mn-54	7.7	9.2	1.2	7.3	6.4	3.0
Ru-106	--	1.5	2.3	1.6	1.8	2.1
Sb-125	0.3	--	1.2	0.6	0.7	0.6
Nb-95	0.3	--	--	--	0.2	1.8
Zr-65	--	--	--	--	--	0.7
Ag-110m	--	--	0.1	--	0.1	0.3

Source: EPRI 1989, Table 2-12

Since the activity levels in the waste were low, several bags of waste in the lower contamination categories would be filled before much waste had accumulated in the higher contamination categories. The survey at each reactor plant was terminated when 5 to 10 lb of waste was accumulated in the higher category bags. The survey encompassed a total of 487 bags containing 61,426 pieces of DAW weighing 6,628 lb. The gamma activity distribution for the bags of DAW is illustrated in Figure 2-4 (EPRI 1989, Figure 2-16).

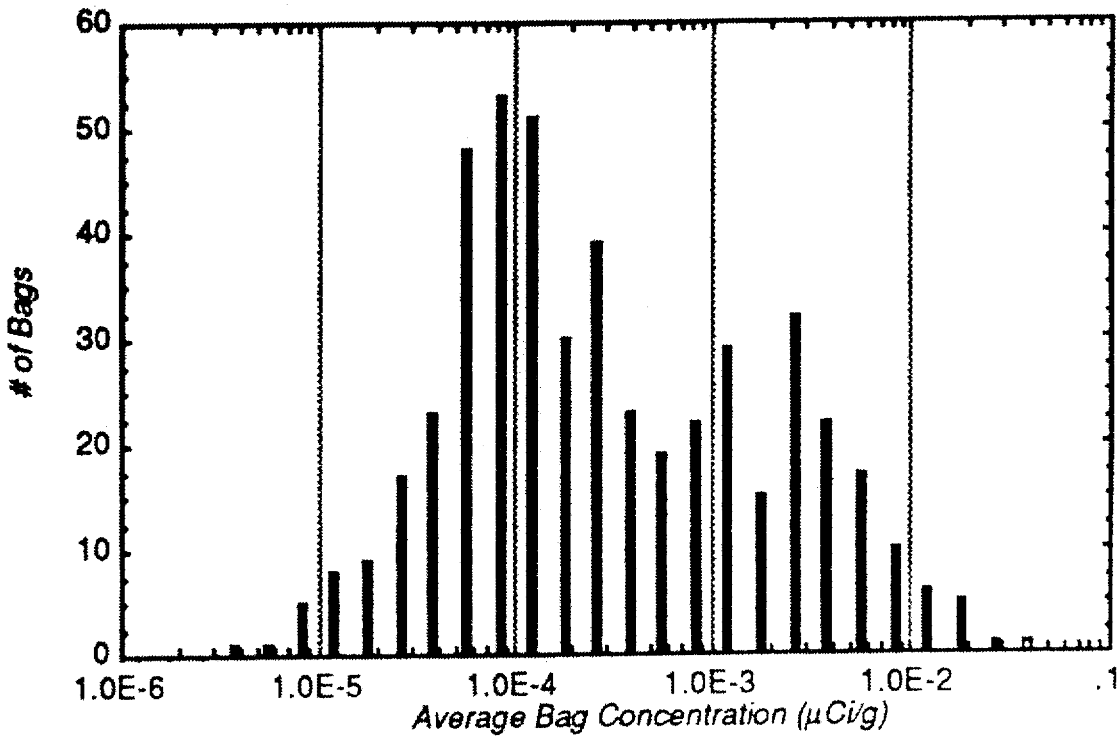


Figure 2-4. DAW Bag Concentration Distribution from Gamma Radiation

The statistics associated with the DAW gamma radiation distribution in Figure 2-4 are as follows:

Sample Size - 487
Weighted Arithmetic Average - 1.00e-3 μCi/g
Geometric Mean - 2.92e-4 μCi/g
Median - 2.09e-4 μCi/g
Geometric Standard Deviation - 6.1
Minimum - 4.06e-6 μCi/g
Maximum - 4.27e-2 μCi/g
95th Percentile Concentration - 5.85e-3 μCi/g

Similar distributions for other solid materials including soils and resins are available in the original EPRI reference (EPRI 1989).

In addition to gamma spectroscopy measurements, radiochemical measurements were made on non-gamma-emitting nuclides in the waste samples, and scaling factors were developed between gamma-emitters and non-gamma-emitters. To be conservative (i.e., to overestimate exposures to the public), EPRI selected the "BWR Average" radionuclide distribution from Table 2-45 since the Co-60 contribution was higher than for PWRs or for the all reactor composite. This distribution was coupled with the scaling factors based on radiochemical analyses from samples of DAW, oil, resin, and soil to obtain a conservative estimate of the radionuclide distribution for all low-activity reactor wastes. However, since one of the goals of the current study is to develop realistic estimates and appropriate probability distributions for the estimates, the data in EPRI 1989 have been reworked using the "All Plants Average" in Table 2-45 and scaling factors based on radiochemical assays of DAW only. The constructed distribution for the radionuclides in DAW from all reactors is presented in Table 2-46.

Table 2-46. Fractional Distribution for Radionuclides in DAW for All Reactors*

Radionuclide	Contribution to DAW Activity (%)
Mn-54	2.2
Co-58	5.0
Co-60	46
Nb-95	1.3
Ru-106	1.6
Ag-110m	0.22
Sb-125	0.45
Cs-134	4.6
Cs-137	15
C-14	0.0096
Fe-55	19
Ni-63	5.2
Sr-90	0.035
I-129	0.0000045
Pu-239	0.0013
Pu-238	0.0018

* May not sum to 100 percent due to round-off.

2.2.6.4 Recycling/Reuse of Materials and Equipment

At the end of 1996, Commonwealth Edison and the National Waste Management Corporation surveyed of nuclear utilities to assess generation and management of radioactive scrap metal (Lorenz and Gardner 1997). Responses were received from 19 BWRs and 26 PWRs. Based on these responses, the surveyors concluded that the then-current inventory of large components (e.g., steam generators and turbine rotors) was 5,732 tons, and an additional 1,080 tons would be generated in 1997 from the replacement of steam generators at the McGuire and Salem nuclear power plants. In addition, routine generation of radioactive scrap metal from the respondents was expected to be about 2,000 tons in 1997, and the existing inventory of routine items was 7,955 tons. Approximately 90 percent of the routinely generated metal is free released after decontamination, and 10 percent is disposed of as LLW. Inferentially, there is a "clean" component of scrap metal, which was outside the scope of the survey.

The steam generators, which typically contain about 100 Ci after a few years of storage, are kept in onsite mausoleums. At the time of the survey, there were 28 PWR steam generators in storage. Thirty-five turbine rotors, each weighing 110 tons, were also in storage. The rotors have very low levels of contamination.

In another example of recycle/reuse, Duke Power has undertaken maintenance of eight primary coolant pumps from its Oconee nuclear station, primarily to repair cavitation damage (Shiel 2000). The procedure involves removing the pump motor weighing 92,000 lb, which sits over the pump and associated structural steel. The pumps (weighing either 53,000 lb or 23,000 lb, depending on the manufacturer) are unbolted from their casings and shipped offsite for refurbishment. At the maintenance facility, the pumps are decontaminated using a multiple-cycle chemical process, resulting in a decontamination factor of ten. The refurbished pumps are then returned to the Oconee station. The process costs less than half the replacement cost of a new pump.

2.3 NRC Fuel Cycle Facility Licensees

This section discusses the nuclear fuel cycle facilities, including uranium mills, UF_6 conversion plants, fuel fabrication facilities, and independent spent fuel storage installations. Uranium enrichment facilities are considered in Chapter 3. For all of these facilities, the key radionuclides are U-235 (and daughters) and U-238 (and daughters).

2.3.1 Uranium Mills

The NRC currently licenses 24 uranium recovery facilities[3] under 10 CFR Part 40, including 17 conventional uranium mills, 6 in situ leach facilities, and 1 ion-exchange facility (http://www.nrc.gov/OPA/gmo/tip/tip19.htm). In addition, several milling sites have been licensed by Agreement States (http://www.antenna.nl/wise/uranium/umopusa.html, DOE 1995). These facilities are listed in Table 2-47.

Table 2-47. Uranium Milling Facilities

Facility Name	Facility Type	Location	NRC License No.	Status
Irigaray/ Christensen Ranch	In Situ Leach	Cogema Mining Inc.; north of Casper, WY	SUA-1341	shutdown, starting decommissioning
Crownpoint Uranium Project	In Situ Leach	Hydro Resources Inc.; McKinley County, NM	SUA-1504	licensed but not constructed
Crow Butte	In Situ Leach	Crowe Butte Resources, Inc.; Dawes County, NE	SUA-1534	operating
Highlands	In Situ Leach	Power Resources Inc.; near Douglas, WY	SUA-1511	operating, license renewal requested
Ruth/North Butte	In Situ Leach	Pathfinder Mines Corp.; near Pine Tree Junction, WY	SUA-1540	licensed but not constructed
Smith Ranch	In Situ Leach	Rio Algom Mining Corp.; Converse County, WY	SUA-1548	operating
Green Mountain	Ion Exchange	U.S. Energy Corp.; south of Jeffery City, WY	SUA-1524	shutdown, starting decommissioning
Lisbon Mill	Conventional	Rio Algom Mining Corp.; San Juan County, UT	SUA-1119	shutdown
Ambrosia Lake	Conventional	Quivira Mining Co.; McKinley County, NM	SUA-1473	shutdown, decommissioning plan approved by NRC
Church Rock	Conventional	UNC Inc.; McKinley Couunty, NM	SUA-1475	shutdown, undergoing decommissioning
Shootering Canyon	Conventional	Plateau Resources Ltd.; Garfield County, UT	SUA-1371	standby
Sweetwater	Conventional	Kennecott Uranium Co.; 42 mi NW of Rawlins, WY	SUA-1350	standby, license renewed 8/99
Bear Creek	Conventional	Union Pacific Resources; Glenrock, WY	SUA-1310	shutdown, under reclam.

[3] A uranium mill tailings waste disposal facility in South Clive, Utah, is also licensed under the regulation.

Table 2-47. Uranium Milling Facilities (continued)

Facility Name	Facility Type	Location	NRC License No.	Status
Highland Uranium Reclamation Project	Conventional	Exxon Coal and Minerals Co.; Converse County, WY	SUA-1139	shutdown, under reclam.
Pathfinder Lucky Mc	Conventional	Cogema Resources Inc.; Gas Hills, WY	SUA-672	shutdown, has reclam. plan
Pathfinder Shirley Basin	Conventional	Cogema Resources Inc.; Shirley Basin, WY	SUA-442	shutdown, reclam. plan under review
Split Rock	Conventional	Western Nuclear Inc.; Gas Hills, WY	SUA-056	shutdown, reclam. in progress
Umetco Gas Hills	Conventional	Umetco Minerals Corp.; Natrona County, WY	SUA-648	shutdown, reclam. in progress
ANC Gas Hills	Conventional	American Nuclear Corp.; Fremont County, WY	SUA-667	shutdown; reclam. in progress
White Mesa	Conventional	International Uranium (USA) Corp.; San Juan County, UT	SUA-1358	operating
Moab Mill	Conventional	Atlas Corp.; Moab, UT	SUA-917	shutdown, dismantling in progress
Petrotomics Shirley Basin	Conventional	Petrotomics Co.; Shirley Basin, WY	SUA-551	shutdown, reclam. in progress
Sohio L-Bar	Conventional	Kennecott Corp./Sohio Western Mining Co.; Bibo, NM	SUA-1472	shutdown, reclam. in progress
Grants Mill	Conventional	Homestake Mining Co.; Cibola County, NM	SUA-1471	shutdown, reclam. in progress
Umetco Maybell	Heap Leach	Umetco Minerals Corp.; Moffat County, CO	AS*	shutdown, reclam. in progress
Umetco Uravan	Conventional	Umetco Minerals Corp.; Uravan, CO	AS	shutdown, reclam. in progress
Conquista Project	Conventional	Conoco; TX	AS	decomm. in progress
Panna Maria	Conventional	Chevron Resources Co.; Karnes County, TX	AS	decomm. in progress
West Cole Project	In Situ Leach	Cogema Mining Inc.; Bruni, TX	AS	restoration in progress
Dawn Mining	Conventional	Dawn Mining Co.; Ford, Stevens County, WA	AS	shutdown, reclam. in progress
Sherwood	Conventional	Western Nuclear Inc.; Stevens County, WA	AS	shutdown, decommissioning in progress
Canon City Mill	Conventional	Cotter Corp.; Canon City, CO	AS	standby, restart planned

Table 2-47. Uranium Milling Facilities (continued)

Facility Name	Facility Type	Location	NRC License No.	Status
Hecla Durita	Heap Leach	Hecla Mining Co.; Montrose County, CO	AS	remediation on-going
Kingsville Dome	In Situ Leach	URI Inc.; Kingsville, TX	AS	standby
Rosita Project	In Situ Leach	URI Inc.; San Diego, TX	AS	closed permanently
O'Hern Project	In Situ Leach	Cogema Mining Inc.; Bruni, TX	AS	restoration in progress
Holiday/El Mesquite Project	In Situ Leach	Cogema Mining Inc.; Bruni, TX	AS	restoration in progress
Hobson	In Situ Leach	Everest Exploration; Karnes County, TX	AS	standby
Ray Point	Conventional	Exxon; TX	AS	decomm. in progress

* AS - Agreement State Licensee

Uranium mills are excluded from the license termination requirements of 10 CFR Part 20.

2.3.1.1 Conventional Mills

Steps in cleanup, decommissioning, and dismantling of conventional uranium mills include the following (DOE 1995):

> *1. Cleanup and decontamination of equipment and building using spraying, steam cleaning, or other methods as needed for salvage ......*
> *2. Removal of equipment from buildings during the cleanup process. Equipment is segregated into the following categories: (a) that which is potentially salable for unrestricted use following radiation checks and necessary decontamination, (b) that which is possibly contaminated but salable to other uranium operations, and (c) that which is disposable. Little, if any, equipment is likely to be salvageable. The sales of salvaged equipment are unlikely to be significant, considering the limited market potential, other costs involved for the buyer and seller, and the potential liabilities. ...*
> *3. Dismantling of building and foundation structures. ...*
> *4. Cutting up larger pieces of equipment and building materials; cutting, crushing, and flattening pipes, tanks, and similar structures for ease of handling.*
> *5. Transporting materials and placing them in a burial site, usually a tailings pile...*
> *6. Cleanup of the mill site. Contaminated debris and soil are removed, as are roads and parking lots.*
> *7. Ripping, regrading, resoiling, liming, fertilizing, and reseeding as necessary to reestablish vegetation.*

NRC assumed in its draft generic environmental impact statement to support the decommissioning rule that all concrete floors and walls in a conventional uranium mill were contaminated and would be buried in the mill tailings pile (NRC 1994).

2.3.1.2 In Situ Leach Facilities

The steps in cleaning up the plant at an in situ leach facility are similar to those involved in the cleanup of a conventional uranium mill, except that a tailings pile is generally not available on site for disposal of contaminated equipment and piping. Consequently, these materials must be shipped to an approved disposal site. In addition, well-field equipment, including well casings and piping, may require removal and disposal.

The decommissioning plan for the Irigaray and Christensen Ranch Projects describes how these companion in situ leach facilities in Wyoming will be remediated (ERG 2000). The approach described here is assumed to be typical for NRC-licensed facilities.

> *Both sites generally consist of uranium extraction/groundwater restoration plants, wellfields, and evaporation ponds. The uranium extraction portion of the plants contain sand filter tanks for filtering unwanted solids from the wellfield groundwater, and ion exchange resin columns for removing the dissolved uranium from the filtered wellfield ground water. The groundwater restoration facilities portion of the plants contain reverse osmosis filtration units for removing dissolved solids from the wellfield ground. In addition, the plants contain chemical storage tanks for uranium extraction and/or restoration purposes, and various pumps and piping.*
>
> *The wellfields contain injection and recovery wells completed in the ore zone at depths down to 600 feet. Pipes from the injection and recovery wells are completed to the plants through connecting wellfield buildings and trunk lines. The wellfields also contain monitoring wells for sampling the groundwater around the perimeter of the mined ore zone and in the aquifers above and below it.*
>
> *All ponds, except the permeate storage pond at the Christensen Ranch, have a synthetic liner placed over leak detection piping. The permeate storage pond is unlined because it is used to store low-TDS permeate from the reverse osmosis filtration process, which meets NPDES water quality standards for surface discharge.*

... The total acreage disturbed by the Irigaray operations is approximately 133 acres. This estimate includes the plant with a dryer, a wellfield building, topsoil piles, eleven lined evaporation ponds, roads and wellfields, several small utility buildings and the peripheral disturbance.... The uranium recovery and packaging facilities are located in the plant and consist of an elution circuit for the ion exchange resin, a uranium precipitation circuit, a yellowcake filtering (dewatering) circuit, yellowcake storage tanks, and a yellowcake dryer and packaging circuit (ibid.).

The Christiansen Ranch facilities include a satellite plant, four evaporation ponds, two shop buildings, roads and well-fields, topsoil piles, numerous small utility buildings, and two disposal wells.

Gross alpha contamination measurements were taken on the floors of both the Irigaray and Christensen Ranch plants using a Ludlum Model 43-90 instrument (a 125-cm^2 large area alpha probe). The Irigaray gross alpha results, as determined from 11 measurements made by the environmental restoration contractor, showed a mean of 2,199 dpm/100 cm^2 and a standard deviation of 2,335 dpm/100 cm^2. A series of 12 measurements made by the facility operator showed a mean 1,667 dpm/100 cm^2 and a standard deviation of 1,019 dpm/100 cm^2. The removable alpha was less than 5 percent of the total. A series of 20 measurements made by the operator at the Christensen Ranch Plant showed a mean of 429 dpm/100 cm^2 and a standard deviation of 638 dpm/100 cm^2. For perspective, the acceptable alpha surface contamination levels for uranium specified in NRC Regulatory Guide 1.86 are 5,000 dpm/100 cm^2 total (averaged over no more than 1 m^2) and 1,000 dpm/100 cm^2 removable.

Well-field piping samples showed internal total alpha surface contamination levels ranging from 2,700 to 6,600 dpm/100 cm^2 with an average value of 4,400 dpm/100 cm^2. Four samples taken from 4-in diameter recovery trunk line piping averaged 6,600 dpm/100 cm^2 and ranged from 4,400 to 8,900 dpm/100 cm^2. The ability to decontaminate trunk line piping in place was evaluated by exposing these samples of trunk line to a 10 percent hydrochloric acid solution. Results are shown in Table 2-48.

After groundwater restoration is completed, all wells will be plugged and abandoned. Surface piping including injection and recovery well lines and trunk lines will be removed, along with meters and related equipment. Underground well lines and trunk lines will either be excavated and removed, or surveyed to insure that the release criteria are met and left in place.

Decontamination may be required to achieve this objective with buried piping. Any pipe buried at depths of less than 2 feet will be removed.

Table 2-48. Decontamination of Trunk Line with 10% HCl

Sample	Initial Survey Total Alpha (dpm/100cm2)	Decontamination Process	Post-Decontamination Survey ($dpm/100\ cm^2$)	
			Total Alpha	Removable Alpha
#1	6,400	30 min. in HCl	3,300	882
#2	6,700	30 min. in HCl	3,300	519
#3	8,900	120 min. in HCl	2,900	603
#4	4,400	120 min. in HCl	1,300	627

Disposal plans for various facilities include the following:

- Small portable structures, such as well-field buildings, may be transported whole to any location upon verification that the structures are releaseable for unrestricted use

- Large structures with concrete foundations and sumps will either be decontaminated and left in place for the property owner, or dismantled and transported in sections to an off-site location (either a licensed facility, if contaminated, or a conventional landfill site, if decontaminated)

- Nonrestricted area structures, such as the maintenance shop, warehouse, and office, may be left in place if desired by the landowner

- Salvageable contaminated equipment, such as tanks, pumps, and reverse osmosis filtration units, may be transferred to another licensed facility

- Byproduct materials will be shipped to an NRC-licensed facility (currently the Shirley Basin Tailings Facility of Pathfinder Mines Corp.) for disposal

2.3.1.3 Current Decommissioning of Uranium Mills

A few anecdotal observations as to the handling of decommissioning activities at selected uranium mills are noted below:

- One conventional uranium mill undergoing decommissioning expects to release three trailers (including two soil lab trailers) and five pieces of large construction from the site during decommissioning. No buildings are planned to be released (Brummett 2001).

- In a typical in-situ leach facility, most of the piping, trunk lines, and tubing in the well-field is plastic. One in-situ leach facility plans to chip the piping into small pieces and dispose of it in a licensed facility. Demolished building structures would be released to a landfill (Brummett 2001).

- During reclamation of the Hecla Durita site, demolition debris and contaminated soil were placed in the tailings closure cell.

2.3.1.4 Inventory Summary - Uranium Mills

Uranium mills are licensed either by the NRC or by the Agreement States. No studies were located during the preparation of this report that describe the kinds and quantities of materials and associated levels of radioactivity for uranium mills or in situ leaching facilities. The major radioactive contaminants are U-235 (and daughters) and U-238 (and daughters). Most conventional uranium mills have been shut down and are undergoing decommissioning. These mills are not likely to be significantly affected by any future NRC regulations relating to the clearance of solid materials from regulatory control, since dismantlement will likely be well advanced or completed prior to any rulemaking. Four conventional mills are either operating or on standby status. Based on the approach to decommissioning described above, where there is little or no salvageable equipment and most materials are buried in on-site tailings piles or at other approved sites, the quantities of potentially clearable materials from uranium mills are expected to be quite small.

Similar to the situation with conventional uranium mills, many in situ leach facilities have been shut down and are undergoing decommissioning. These shutdown facilities are unlikely to be affected by an NRC clearance rule, since dismantlement is expected to be largely completed prior to issuance of any final rule. Seven in situ leach facilities are operating, on standby status, or not yet built. Large quantities of materials and equipment are not expected to be available when these facilities are ultimately decommissioned. Contaminated equipment and plastic piping are likely to be disposed of in tailings piles or at other licensed disposal sites. Disposition of structures and clean equipment could be affected by the specifics of any clearance rule.

2.3.2 Uranium Hexafluoride Production Facilities

Most nuclear reactors require uranium to be enriched from its natural isotopic composition of approximately 0.7 percent U-235 (most of the rest being U-238) to 3.5-4 percent U-235. To enrich uranium, it must first be put in a gaseous form, and the most convenient way of achieving this is to convert the uranium oxides to uranium hexafluoride (UF_6).

As shown in Table 2-49, the only operating UF_6 conversion facility in the United States is operated by ConverDyn in Metropolis, Illinois. The Sequoyah Fuels Corporation facility in Gore, Oklahoma, was shut down in 1993 and is currently waiting decommissioning. The NRC license for the ConverDyn facility currently expires on June 30, 2005. Closure of that facility at that time would force U.S. utilities to rely on foreign sources of conversion capacity. To ensure continued domestic UF_6 production capability, the ConverDyn license would need to be renewed, or a new facility would need to be constructed and licensed to operate by June 2005. An additional complication concerning the longevity of the ConverDyn facility is the importation of highly enriched uranium (HEU) from Russia. A recent GAO report stated that because of the loss of revenue caused by the importation of HEU, "it is doubtful that ConverDyn can survive much longer" (GAO 2000). Thus, the ConverDyn facility could shut down before its license expires.

Table 2-49. Licensed Uranium Hexafluoride Production Facilities

Facility	Process	License/ Docket	Startup - Shutdown	Capacity (MTU/yr)	Status
Gore/Sequoyah	Yellowcake to UF_6	SUB-1010 40-8027	1970 - 1993	5,000	Shutdown
Metropolis/ConverDyn	Yellowcake to UF_6	SUB-526 40-3392	1959 - 6/30/05	14,000	In operation

For the purposes of this study, it is assumed that the ConverDyn facility will shut down when its current license expires on June 30, 2005, and will be dismantled five years later. It is also assumed that the Gore facility will be dismantled in 2003, or 10 years after it was shut down.

2.3.2.1 Reference Uranium Hexafluoride Production Facility

In *Technology, Safety and Costs of Decommissioning a Reference Uranium Hexafluoride Conversion Plant* (PNL 1981), a reference UF_6 production facility was developed. This reference facility was assumed to have an annual processing rate of 10,000 metric tons of natural uranium (MTU). The basis for the reference facility was a combination of then existing and retired facilities (including both the Gore and Metropolis facilities); no attempt was made to use a single existing facility as the study basis.

The reference UF_6 production facility consists of a main building, a solvent extraction facility, a warehouse, a cooling tower, retention lagoons, and other storage areas. The main building is a 55 m by 100 m steel frame structure with 38 mm insulated metal siding. The interior walls are constructed of concrete block and sheetrock. The floors are heavily reinforced concrete to support equipment. The roof is 35 mm insulated corrugated metal deck that is capped with asphalt and gravel. The solvent extraction facility has a steel frame with metal siding and sealed concrete floors.

Material Masses

The reference uranium hexafluoride production facility report provided considerable information on the quantity of ferrous metals and some information on concrete, but no specific information on the quantities of aluminum or copper within the facility. Table 2-50 presents the mass of steel contained in equipment within the reference uranium hexafluoride production facility. This information has been condensed from a series of tables provided in Appendix A of PNL 1981.

In addition to steel in equipment, structural steel was used in the construction of the reference uranium hexafluoride production facility. Table 2-51 gives an estimate of the mass of structural steel used in the reference uranium hexafluoride production facility based on building dimensions provided in PNL 1981 and "Structural Steel Weights per S.F. of Floor Area" provided in Means (2000) (Section R051-220) for steel frame, one-story manufacturing buildings (i.e., 18 lbs/ft^2).

The reference uranium hexafluoride production facility is assumed to have a 12" (0.3 m) thick reinforced concrete slab basemat. Thus, there would be approximately 1800 m^3 (4100 Mg) of concrete in the floor, with an estimated 252 Mg of reinforcing steel rebar (based on an assumed density of 2.4 Mg/m^3 for reinforced concrete, 2.3 Mg/m^3 for unreinforced aggregate, and 7.8 Mg/m^3 for steel).

Table 2-50. Mass of Steel in Conversion Plant Equipment

Area	Weight (Mg)
Sampling Station	38.2
Wet Yellowcake System	28.1
UF_4 Slurry Processing	11.9
Ore Concentrate Dissolution	22.0
Solvent Extraction	41.3
Uranyl Nitrate Concentration	33.0
Denitration	32.2
Uranium Trioxide to Uranium Dioxide Reduction	80.1
Hydrofluorination	255.7
Fluorination	99.5
Fluorine Generation	264.6
Nitric Acid Recovery Area	15.6
Instrument Repair Shop	4.3
Radwaste Room	22.7
Decontamination Facility	5.6
Laundry	4.6
Change Rooms	1.7
Incinerator Facility	16.1
Subtotal	977.2
Piping, Ductwork, Trays & Light Fixtures	317.8
TOTAL	1,295.0

Table 2-51. Mass of Structural Steel in Buildings

Building	Width (m)	Length (m)	Area (m^2)	Weight (Mg)
Main Building	55	100	5,500	483.8
Solvent Extraction Facility	—	—	377	33.2
TOTAL			5,877	517.0

It was determined in PNL 1981 that 1260 m^3 (40 percent) of materials and equipment from the main and solvent extraction buildings would be shipped for disposal as low-level radioactive waste, while 1870 m^3 (60 percent) would be excessed or sent for commercial disposal.

Radiological Contamination

The radionuclides that compose the surface contamination are shown in Table 2-52, which was taken from PNL 1981, Table 7.4-1.

Table 2-52. Contamination Radionuclide Mix

Radionuclide	μCi/g* of mixture	
	Shutdown	100 Years
Th-230	2.9×10^{-3}	1.7×10^{-2}
Th-231	2.2×10^{-2}	2.2×10^{-2}
Th-234	2.5×10^{-1}	2.5×10^{-1}
Pa-231	3.0×10^{-5}	1.8×10^{-2}
Pa-234m	3.3×10^{-1}	3.3×10^{-1}
Pa-234	3.3×10^{-4}	3.3×10^{-4}
Ra-226	3.0×10^{-4}	3.0×10^{-4}
U-234	3.3×10^{-1}	3.3×10^{-1}
U-235	1.5×10^{-2}	1.5×10^{-2}
U-238	3.3×10^{-1}	3.3×10^{-1}
Total	1.3	1.3

* Multiply by 3.7×10^4 to convert from μCi/g to Bq/g

Table 2-53 summarizes the surface contamination levels of the reference uranium hexafluoride production facility equipment after decontamination. Post-decontamination levels (for the total mixture) on equipment were provided in PNL 1981, Appendix C; equipment weights are provided in Appendix A of that document. The average contamination in pCi/g was calculated based on 1.3 μCi/g, as shown in Table 2-52.

Table 2-53. Equipment Contamination Levels (Piping Not Included)

Activity Range (kg/Mg)†	Average Contamination*		Mass (Mg)	Percent
	(kg/Mg)†	(pCi/g)		
Clean	—	—	135.6	13.9%
0 to 0.1	4.63e-04	0.60	280.7	28.7%
0.1 to 1.0	0.408	530.4	35.5	3.6%
1.0 to 10	3.76	4.89e+03	442.0	45.2%
10 to 100	32.3	4.20e+04	79.0	8.1%
100 to 1000	125.	1.63e+05	4.4	0.5%

* See Table 2.3.2-4 for contamination mix

† kg/Mg - kilograms of contamination per metric ton of equipment

Figure 2-5 is a graphical representation of the data summarized in Table 2-53.

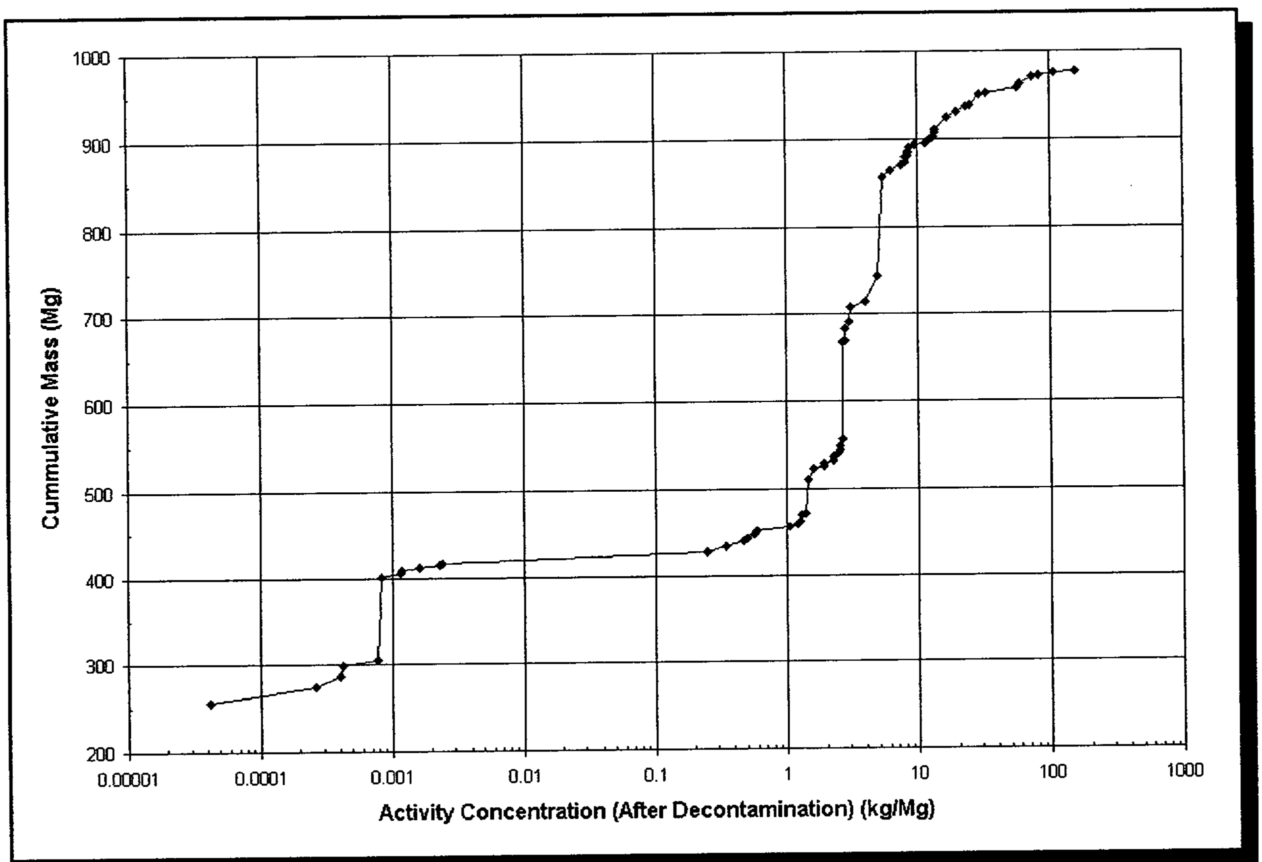


Figure 2-5. Activity Concentration (after Decontamination) of the Reference Uranium Hexafluoride Production Facility Equipment

The total and contaminated surface areas for the reference UF_6 facility are shown in Table 2-54 (NRC 1994, Table C.7.1.1).

Table 2-54. Surface Contamination in Reference UF_6 Facility

Uranium Activity (dpm/100 cm^2)	Surface Area (ft^2)		Percent Contaminated	
	Floor	Wall	Floor	Wall
1.10e+06	120,000	130,000	50	45

As shown in Figure 2-6, uranium does not penetrate very far into concrete surfaces. Consequently, removal of the outermost layer (e.g., 3.175 mm (1/8 inch)) would be sufficient to remove the contaminated portion. Thus, the total amount of contaminated concrete in the reference UF_6 conversion facility is estimated to be 35 m^3.

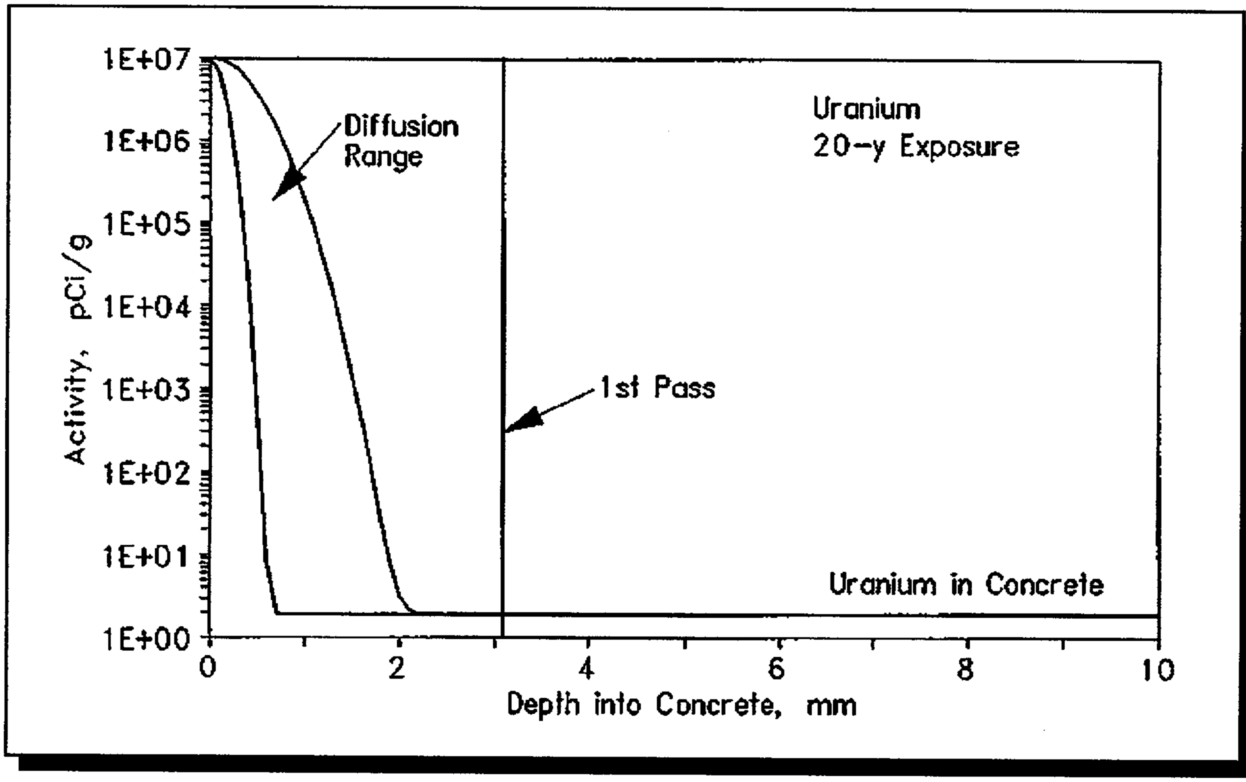


Figure 2-6. Predicted Range of Uranium Penetration into Concrete
(Source: NRC 1994, Figure C.4.8.10)

2.3.2.2 Inventory Summary - UF_6 Production Facilities

Mass of Steel

The mass of steel in the reference facility is 1,295 Mg in equipment (Table 2-50), 517 Mg in structural steel (Table 2-51), and 252 Mg in rebar.

Volume of Concrete

The reference uranium hexafluoride production facility was estimated to have approximately 1800 m^3 of concrete in its floor.

Masses of Copper and Aluminum

No studies were located during the preparation of this report that describe the quantity or associated levels of radioactivity of copper or aluminum contained in UF_6 production facilities.

Scaling from Reference Facility

There are only two UF_6 production facilities: the shutdown Sequoyah Fuels Corporation facility with a capacity of 5,000 MTU/yr, and the ConverDyn facility with a capacity of 14,000 MTU/yr. The PNL 1981 reference facility was assumed to have an annual processing rate of 10,000 MTU/yr. Two approaches to scaling are apparent: (1) scale based on the facilities' capacities, or (2) no scaling, an approach that simply uses the data from the reference facility for each actual facility, regardless of the actual facility's size. Since the sum of the capacities of the two actual facilities (19,000 MTU) is approximately the same as twice the reference facility's capacity (20,000 MTU), both approaches to scaling would give approximately the same total masses of material from UF_6 production facilities. Nonetheless, scaling by facility capacity is recommended, since it might affect the timing of the availability of some of the material.

Principal Radionuclides and Contamination Levels

Table 2-52 gives the principal radionuclides of concern for a UF_6 production facility, mainly U-235 and U-238 and equilibrium daughters. As shown in Table 2-53, approximately 13.9 percent of the equipment is anticipated to be clean of radioactivity, 28.7 percent would have an average contamination level (after decontamination) of about 0.02 Bq/g (0.6 pCi/g), and 45.2 percent would have an average contamination level (after decontamination) of about 181 Bq/g (4,890 pCi/g). From Draft NUREG-1640 (NRC 1999), Table 2.1, the mean dose factor for U-238 is 29 μSv/a per Bq/g (0.1 mrem/y per pCi/g). Thus, 45.2 percent of the equipment would not meet any of the clearance criteria under consideration and the balance of the equipment would meet the 1, 10 and 100 μSv/a regulatory options for clearance.

2.3.3 Fuel Fabrication Facilities

Fabrication is the final step in the process used to produce uranium fuel. This process converts enriched UF_6 into a solid form of uranium suitable for use in a nuclear reactor. Fabrication of reactor fuel consists of three basic steps: the chemical conversion of UF_6 to uranium dioxide (UO_2) powder; the ceramic process that converts UO_2 powder to pellets; and the mechanical process that loads the fuel pellets into rods and constructs finished fuel assemblies.

Table 2-55 lists the seven uranium fuel fabrication facilities currently licensed to operate by the NRC, and the single mixed oxide (MOX) fuel fabrication facility currently under development. DOE has requested authorization from the NRC to construct the MOX facility.

Table 2-55. Licensed Fuel Fabrication Facilities

Company	Location	Docket/ License	Expires	Capacity (MTU/yr)
CE Nuclear Power, LLC	Hematite, Missouri	70-36 SNM-33	June 1, 2001	450
Global Nuclear Fuel - America, L.L.C.	Wilmington, North Carolina	70-1113 SNM-1097	June 30, 2007	1200
Westinghouse Electric Company (BNFL)	Columbia, South Carolina	70-1151 SNM-1107	November 30, 2005	1150
Nuclear Fuel Services, Inc.	Erwin, Tennessee	70-143 SNM-124	July 31, 2009	Not Provided
Framatome Cogema Fuels	Lynchburg, Virginia	70-1201 SNM-1168	March 30, 2002	400
BWX Technologies Naval Nuclear Fuel Division	Lynchburg, Virginia	70-27 SNM-42	September 30, 2005	Not Provided
Framatome ANP Richland, Inc.	Richland, Washington	70-1257 SNM-1227	November 30, 2001	700
Duke, Cogema, Stone & Webster	Aiken, South Carolina	70-3098 Not Issued	anticipated 2026	Not Provided

Table 2-55 identifies the owner of each facility and provides location, docket and license number, current license expiration date, and each facility's processing capacity. As shown in Table 2-55, with the exception of the yet-to-be licensed MOX facility, all licensed fuel fabrication facilities are scheduled to have their licenses expire by July 31, 2009. Closure of all uranium fuel fabrication facilities by that time would force U.S. utilities to rely on foreign sources to fabricate their fuel. To ensure continued U.S. uranium fuel fabrication capability, the license of one or more facilities would need to be renewed, or a new facility would need to be constructed and licensed to operate by July 2009.

Nonetheless, this study uses only those fuel fabrication facilities listed in Table 2-55, and it is assumed that they will be dismantled 5 years after their current licenses expire.

2.3.3.1 Reference Fuel Fabrication Facility

In the late 1970s and early 1980s, the NRC commissioned a series of studies of the technology and costs of decommissioning several types of nuclear facilities. A generic or reference design was selected for each facility studied. The reference uranium fuel fabrication facility is based primarily on the description of the Global Nuclear Fuel facility in Wilmington, North Carolina, presented in *Technology, Safety and Costs of Decommissioning a Reference Uranium Fuel Fabrication Plant* (PNL 1980).

Material Masses

The reference uranium fuel fabrication facility report provides considerable information on the quantity of ferrous metals and some information on concrete, but no specific information on the quantities of aluminum or copper within the facility. Table 2-56 presents the mass of steel contained in equipment within the reference uranium fuel fabrication facility. This information has been condensed from a series of tables provided in Appendix A of PNL 1980.

In addition to steel in equipment, structural steel was used in the construction of the reference uranium fuel fabrication facility. Table 2-57 gives an estimate of the mass of structural steel used in the reference uranium fuel fabrication facility, based on building dimensions provided in PNL 1980 and "Structural Steel Weights per S.F. of Floor Area" provided in Means 2000 (Section R051-220) for steel frame, one-story manufacturing buildings (i.e., 18 lbs/ft^2).

Table 2-56. Mass of Steel in Fuel Fabrication Equipment

Area	Weight (Mg)
Powder Warehouse	75.4
UF_6 Cylinder Storage Room	97.0
UF_6 Vaporization Room	107.4
Chemical Areas	294.6
Powder Storage & Feed Room	60.3
Pelletizing Room	49.7
Sintering Room	241.0
Grinding Room	50.7
Rodding Room	310.6
Gadolinia Rod Fabrication	26.4
Uranium Scrap Recovery Room	13.2
Chemical & Metallurgical Analytical Lab	18.7
Process Development Laboratory	67.6
Hot Machine Shop	23.2
Hot Instrument Shop	5.4
Radwaste Room	22.7
Decontamination Facility	5.9
Laundry Room	4.9
Change Room	1.7
Incinerator Facility	16.1
Fluoride Waste Effluent Treatment System	129.4
Nitrate Waste Effluent Treatment System	44.4
Waste Treatment Building	15.6
Excess Equipment Storage Yard	100.2
Radwaste Effluent Treatment System	8.4
Subtotal Equipment	1,790.5
Piping, Ductwork, Trays & Light Fixtures	644.9
TOTAL	2,435.4

Table 2-57. Mass of Structural Steel in Buildings

Building	Width (m)	Length (m)	Area (m^2)	Weight (Mg)
Main Building	211.0	80.0	16,880	1,484.8
Uranium Scrap & Powder Storage Addition	47.0	27.0	1,269	111.6
Chemical Metallurgical Lab Addition	37.0	21.0	777	68.3
Fluoride Nitrate Waste Treatment Building	19.0	12.0	228	20.1
Incinerator Building	18.3	12.2	223	19.6
Boiler	14.0	7.0	98	8.6
TOTAL			19,475	1,713.1

The reference uranium fuel fabrication facility is assumed to have a 12" (0.3 m) thick reinforced concrete slab basemat. Thus, there would be approximately 5,940 m^3 of concrete in the floor, with 2,486 Mg of reinforcing steel rebar.

Table 2-58 gives the quantity of steel and concrete estimated to be contained in the MOX fuel fabrication facility (DCS 2001). As shown, substantially more concrete is used to construct a MOX facility than was estimated for the reference uranium fabrication facility, while the quantity of steel used in the MOX facility is less than half of the quantity of structural and equipment steel estimated for the uranium facility.

Table 2-58. MOX Fuel Fabrication Facility Materials

Material	Quantity	
Concrete	103,000 yd^3	26,250 m^3
Steel	2,000 tons	1,816 Mg

Source: DCS 2001

Radiological Contamination

Table 2-59 presents the specific radioactivity levels for the uranium fuel mixture at the time of shutdown and 100 years later. These mixtures were taken from PNL 1980 and are based on an average of 3 percent enriched uranium feed material.

Table 2-59. Radionuclide Mix in Fuel Fabrication Facility Contamination

Radionuclide	μCi/g* of mixture	
	Shutdown	100 Years
Th-230	3.10×10^{-4}	1.81×10^{-3}
Th-231	6.54×10^{-2}	6.54×10^{-2}
Th-234	3.36×10^{-1}	3.36×10^{-1}
Pa-231	2.93×10^{-5}	1.73×10^{-4}
Pa-234m	3.36×10^{-1}	3.36×10^{-1}
Pa-234	3.36×10^{-4}	3.28×10^{-6}
U-234	1.72	1.72
U-235	6.54×10^{-2}	6.54×10^{-2}
U-238	3.27×10^{-1}	3.27×10^{-1}
Total	2.85	2.85

* Multiply by 3.7×10^{4} to convert from μCi/g to Bq/g.

Table 2-60 summarizes the surface contamination levels of reference fuel fabrication facility equipment after decontamination. Post-decontamination equipment total uranium contamination was provided in PNL 1980, Appendix C, whereas equipment weights are provided in Appendix A of that document.

Table 2-60. Equipment Contamination Level

Activity Range (kg/Mg)*	Average Contamination		Mass (Mg)	Percent
	(kg/Mg)*	(pCi/g)		
Clean	—	—	1136.2	46.7%
0 to 0.1	0.024	69.4	1000.0	41.1%
0.1 to 1.0	0.23	653.0	274.2	11.3%
1.0 to 10	1.80	5117.7	18.6	0.8%
10 to 100	13.3	37810.0	1.5	0.1%

* kg/Mg = kilograms of contamination per metric ton of equipment

The total and contaminated surface areas for the reference fuel fabrication facility are shown in Table 2-61 (NRC 1994, Table C.7.1.1).

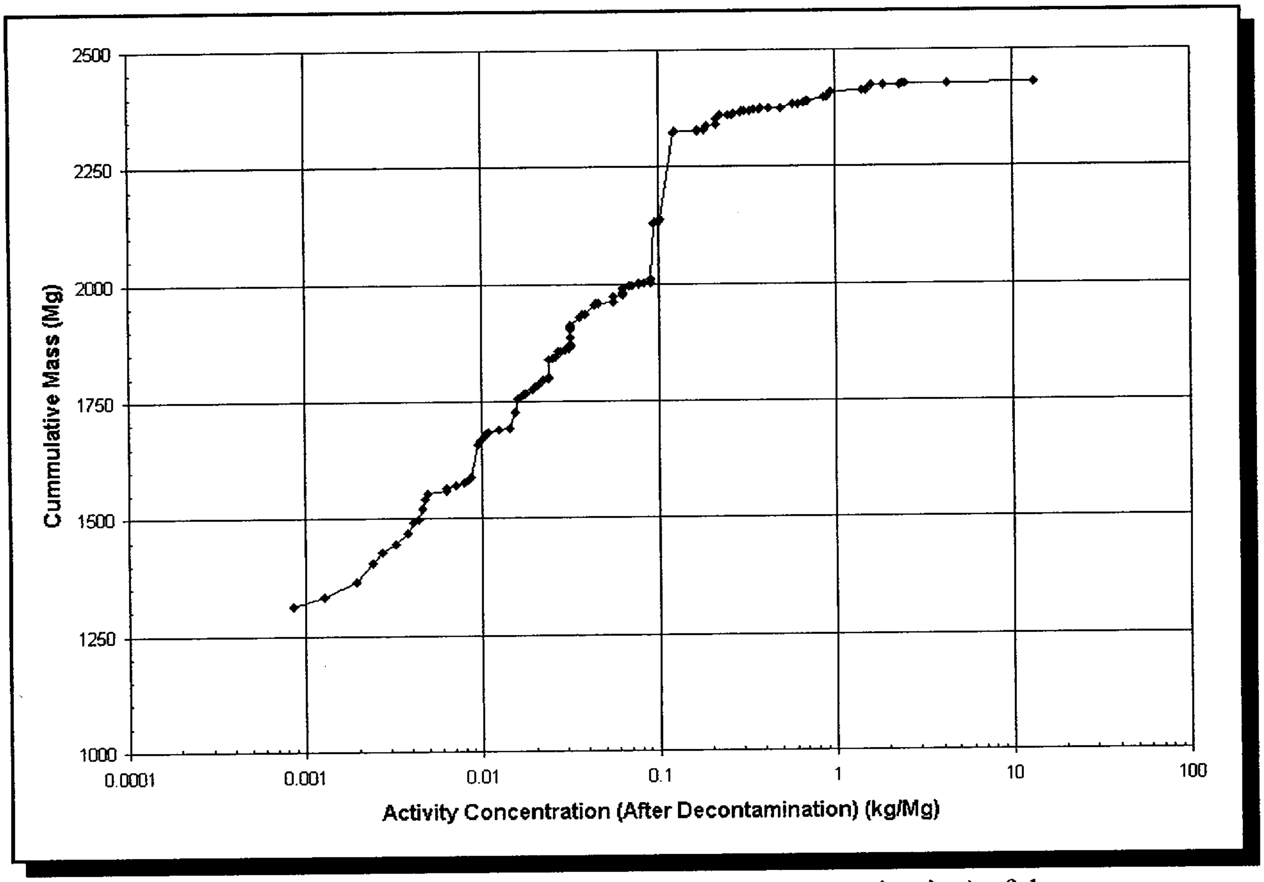


Figure 2-7. Activity Concentration (after Decontamination) of the Reference Uranium Fuel Fabrication Facility Equipment

Table 2-61. Surface Contamination Levels for Reference Fuel Fabrication Facility

Uranium Activity (dpm/100 cm^2)	Surface Area				Percent Contaminated	
	Floor		Wall		Floor	Wall
18,000	240,000 ft^2	22,300 m^2	240,000 ft^2	22,300 m^2	50	5

Source: NRC 1994, Table C.7.1.1

As shown previously in Figure 2-6, uranium does not penetrate very far into concrete surfaces. Consequently, removal of the outermost layer (e.g., 3.175 mm (1/8 inch)) would be sufficient to remove the contaminated portion. Thus, the total amount of contaminated concrete in the reference uranium fuel fabrication facility is estimated to be 39 m^3.

2.3.3.2 Inventory Summary - Fuel Fabrication Facilities

Mass of Steel

The mass of steel in the reference facility is 2,430 Mg in equipment (Table 2-56), 1,710 Mg in structural steel (Table 2-57), and 2,490 Mg in rebar.

Volume of Concrete

The reference uranium fuel fabrication facility was estimated to have approximately 5,940 m^3 of concrete in its floor.

Masses of Copper and Aluminum

No studies were located during preparation of this report that describe the quantity or associated levels of radioactivity of copper or aluminum contained in fuel fabrication facilities.

Scaling from Reference Facility

As shown in Table 2-55, the capacities of the seven operating fuel fabrication facilities range from 400 to 1200 MTU/yr. The PNL 1980 reference facility is based on the 1200 MTU/yr facility.

Means 2000, Section R17100-100, "Square Foot Project Size Modifier," states: "One factor that affects the S.F. [square foot] cost of a particular building is the size. In general, for buildings built to the same specification in the same locality, the larger building will have the lower S.F. cost. This is due mainly to the decreasing contribution of the exterior walls plus the economy of scale usually achievable in larger buildings." The source goes on to state that for facilities with a Size Factor (the actual facility size divided by the reference facility size) of 0.50 or less, the S.F. cost multiplier is 1.1, and for facilities with a Size Factor of 3.5 or more, the S.F. cost multiplier is 0.90. Means 2000 also provides a figure for the multiplier between 0.50 and 3.5, but for present purposes, linear interpolation between 0.50 and 1.0 and between 1.0 and 3.5 should be sufficient.

Scaling from the reference fuel fabrication facility to specific facilities will be based on the capacities of each facility, and will include the Means 2000 multiplier. For example, the capacity of the CE Nuclear Power facility is 450 MTU/yr (see Table 2-55); therefore, to adjust the material mass from the reference facility, the following equation will be used:

$$M = M_R \left(\frac{450}{1200} \right) 1.1$$

where:	M	=	mass of material in the CE Nuclear Power facility (kg)
	M_R	=	mass of material in the reference facility (kg)
	450	=	capacity of the CE Nuclear Power facility (MTU/yr)
	1200	=	capacity of the reference facility (MTU/yr)
	1.1	=	size adjustment multiplier (from Means 2000)

This equation is applicable to fuel fabrication facilities with capacities of less than 600 MTU/yr.

Principal Radionuclides and Contamination Levels

Table 2-59 gives the principal radionuclides of concern for a fuel fabrication facility, mainly uranium isotopes and certain daughter products. As shown in Table 2-60, approximately 46.7 percent of the equipment is anticipated to be free of radioactivity, and 41.1 percent would have an average contamination level (after decontamination) of about 2.57 Bq/g (69.4 pCi/g). From Draft NUREG-1640, Table 2.1 (NRC 1999), the mean dose factor for U-238 is 29 µSv/y per Bq/g (0.1 mrem/y per pCi/g).

2.3.4 Uranium Enrichment Facilities

DOE leases uranium enrichment facilities in Paducah, Kentucky, and Portsmouth, Ohio, to the U.S. Enrichment Corp. These facilities are administered under NRC regulations at 10 CFR Part 76 promulgated in September 1994. NRC regulation of the facilities commenced on March 3, 1997 (NRC 2000a). The K-25 enrichment facility in Oak Ridge, Tennessee, has been shut down and is undergoing decommissioning. Inventory information on these facilities is included with other DOE sites in Chapter 3 of this report.

2.3.5 Spent Fuel Storage Facilities

An independent spent fuel storage installation (ISFSI) is a complex designed and constructed for the interim storage of spent nuclear fuel. Table 2-62 lists the operating dry spent fuel storage facilities (NRC 2000a). In addition, there is a single wet storage facility in Morris, Illinois, operated by the General Electric Company.

ISFSIs may be initially licensed for a period up to 20 years. The license may also be renewed for an additional 20 years. Therefore, it is expected that the materials contained in the above ISFSIs would be available for clearance 20 or 40 years after the startup dates shown in Table 2-62. However, a recent study has determined that extending the storage period to 100 years would have no adverse impacts (PNL 1998). Conversely, fuel may be removed from storage prior to the end of the licensed lifetime of the ISFSI, if an HLW repository becomes available.

Spent fuel may be stored in either a wet or dry environment; the various techniques are as follows:

- Concrete Casks
- Horizontal Storage Modules (HSM)
- Metal Casks
- Pool (Wet) Storage
- Modular Vault Dry Storage (MVDS)

Table 2-63 lists the dry spent fuel storage system designs that have been approved by the U.S. NRC.

In *Technology, Safety and Costs of Decommissioning a Reference Independent Spent Fuel Storage Installations* (PNL 1984), five reference ISFSIs were developed of the following designs:

- Wet: The same as Pool (Wet) Storage listed above, also based on the Morris facility

- Silo: Similar to the Concrete Casks design listed above

- Cask: Similar to the Metal Casks design listed above, except that these casks are stored indoors

- Vault: Similar to the MVDS system listed above, but based on a facility at the Idaho National Engineering Lab, not a commercial design

- Drywell: In-ground storage, not currently being utilized

Table 2-62. Licensed Dry Spent Fuel Storage Facilities

Utility	Reactor Name	Vendor	Capacity (Fuel Assemblies)	Startup
Virginia Electric & Power Company	Surry 1, 2	General Nuclear Systems, and Others	926	1986
Carolina Power & Light Company	H. B. Robinson 2	Transnuclear West	56	1986
Duke Energy Company	Oconee 1, 2, 3	Transnuclear West	2112	1990
Public Service Company of Colorado	Fort St. Vrain*	FW Energy Applications	1464	1992
Baltimore Gas & Electric Company	Calvert Cliffs 1, 2	Transnuclear West	2880	1992
Consumers Energy	Palisades	BNFL Fuel Solutions	578	1993
Northern States Power Company	Prairie Island 1, 2	Transnuclear West	680	1994
Wisconsin Electric Power Company	Point Beach	BNFL Fuel Solutions	288	1995
Toledo Edison Company	Davis-Besse	Transnuclear West	720	1995
Entergy Operations	Arkansas Nuclear One	BNFL Fuel Solutions	336	1997
Virginia Electric & Power Company	North Anna	Transnuclear West	160	1998
Portland General Electric Corp	Trojan	BNFL Fuel Solutions	864	2000
Department of Energy	TMI-2 Fuel Debris	Transnuclear West	NA	1999
Pennsylvania Power & Light	Susquehanna	Transnuclear West	5460	2000

Source: NRC 2000a
*Plant undergoing decommissioning. Transferred to DOE 6/4/99.

Table 2-63. Spent Fuel Storage System Designs

Vendor	Storage Design	Model	Capacity*	Date of CofC**
General Nuclear Systems	Metal Cask	CASTOR V/21	21 PWR	08/17/1990
Westinghouse Electric	Metal Cask	MC-10	24 PWR	08/17/1990
NAC International, Inc	Metal Cask	NAC S/T	26 PWR	08/17/1990
Transnuclear, Inc	Metal Cask	TN-24	24 PWR	11/04/1993
BNFL Fuel Solutions (Sierra Nuclear Corp)	Ventilated Cask	VSC-24	24 PWR	05/03/1993
Transnuclear West	Concrete Module	NUHOMS-24P	24 PWR	01/18/1995
		NUHOMS-52B	52 BWR	
Holtec International	Concrete Cask	HI-STORM 100	24 PWR	10/04/1999
			68 BWR	

Source: NRC 2000a
* Number of PWR or BWR fuel assemblies.
** CofC - Certificate of Compliance

The design of commercial ISFSIs has evolved greatly since the time PNL 1984 was written. Therefore, this study uses only selected information (e.g., Morris facility data, and some Vault data) from PNL 1984. Data for the other ISFSI designs are taken from vendor-supplied documents (including Safety Analysis Reports, etc.) and other sources.

2.3.5.1 Concrete Casks

The ventilated storage cask system (VSC-24) (developed by Sierra Nuclear Corporation and currently owned by BNFL) is a typical concrete cask design, which vertically stores 24 PWR assemblies. The principal components of the system are a steel multi-assembly sealed basket (MSB), a ventilated concrete cask (VCC), and an MSB transfer cask (MTC). The following discussion was taken primarily from NRC 1996a, and supplemented with information and data from BNFL 2000, 2000a, 2000b, and 2000c. A diagram of the cask is shown in Figure 2-8. The weights of the principal components of the VSC-24 are given in Table 2-64.

Table 2-64. VSC-24 Concrete Cask Principal Component Weights

Component	Material	Weight (lbs)	Weight (Mg)
VCC Weather Cover Plate	Steel	1,110	0.504
MSB Structural Lid	Steel	2,384	1.082
MSB Shielding Lid – 2.5" Plate	Steel	2,003	0.909
MSB Shielding Lid – Sandwich Plate	Steel	4,368	1.983
MSB – Empty, w/o Lids	Steel	21,036	9.550
VCC – Empty, w/o Cover Plate	Total	204,875	93.013
	Concrete*	154,053	69.940
	Rebar*	6,676	3.031
MTC – Empty, w/o Lid	Steel	117,700	53.436
Basepad (per Cask)	Concrete*	270,000	122.580
	Rebar*	11,700	5.312

Source: BNFL 2000, Table 3.2-1

* Estimated for this study based on component dimensions, concrete density of 2.3 M/m³ and reinforcee concrete density of 2.4 Mg/m³.

The MSB consists of a steel cylindrical shell with a thick shield plug and steel cover plates welded at each end. The shell length is fuel-specific and varies from 4.2 to 4.9 m (164 to 192 in.), the outside diameter is 1.6 m (62.5 in.), and the shell thickness is 2.5 cm (1 in.); complete dimensions of the VSC-24 components are given in Table 2-65. The internal steel basket consists of a welded structure with 24 square storage locations. The basket aids in the insertion

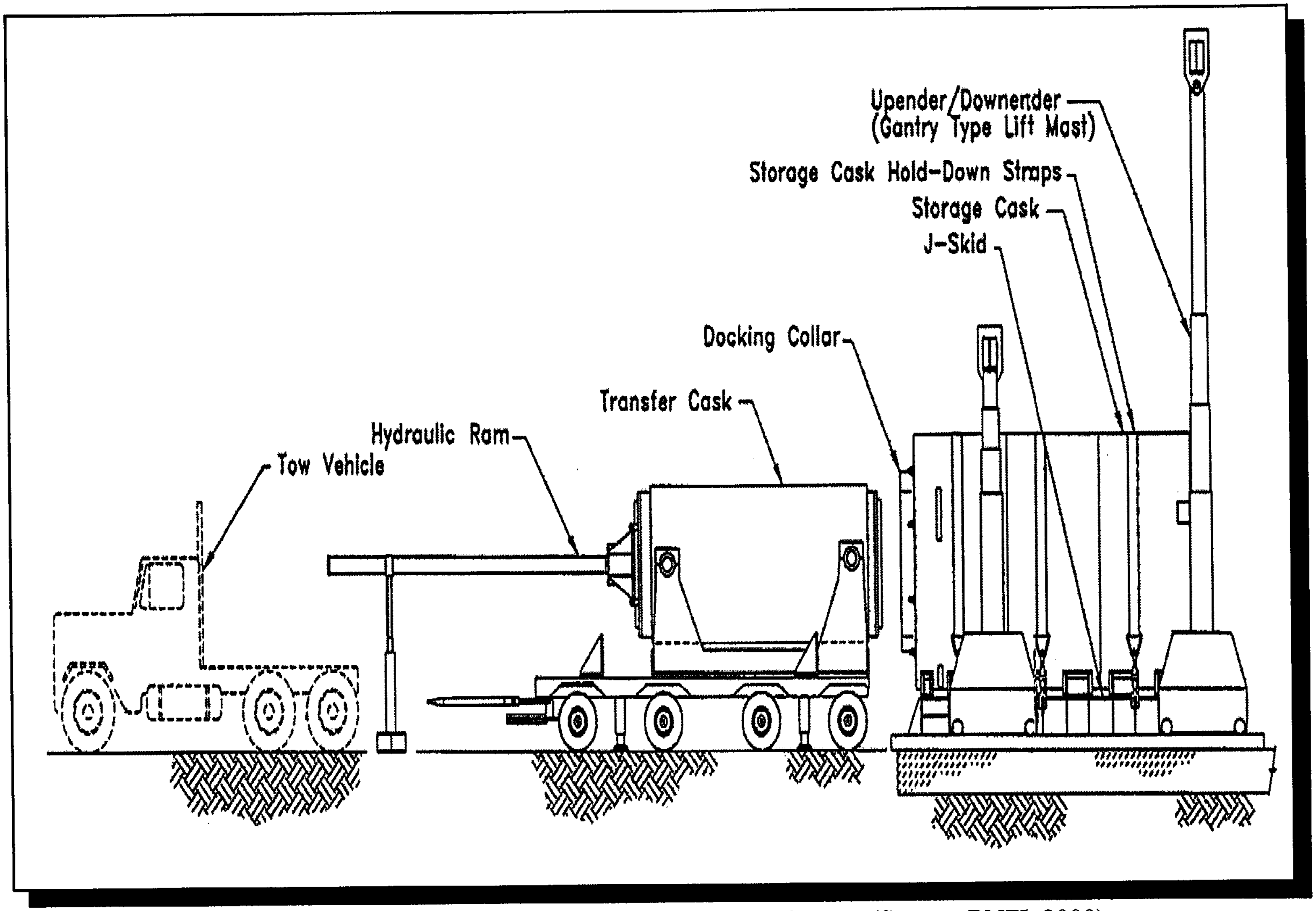


Figure 2-8. VSC-24 Concrete Storage Cask Storage System (Source: BNFL 2000)

of the fuel assemblies, enhances subcriticality during loading operations, and provides structural support during a potential drop accident. The basket is coated with Carbo Zinc 11 for corrosion protection.

Table 2-65. VSC-24 Concrete Cask Principal Component Dimensions

Component	Parameter	Dimension	
Ventilated Concrete Cask (VCC)	OD/ID	132/70.5 in	3.4/1.8 m
	Concrete Thickness	29 in	74. cm
	Height	197 to 225 in	5.0 to 5.7 m
	Steel Liner	1.75 in	4.4 cm
Multi-Assembly Sealed Basket (MSB)	OD	62.5 in	1.6 m
	Length	164 to 192 in	4.2 to 4.9 m
	Thickness	1 in	2.5 cm
	Shield Lid	9.5 in	24. cm
	Structural Lid	3 in	7.6 cm
MSB Transfer Cask	Length	Not Found	
	OD	Not Found	
	Thickness	Not Found	
Basepad (per Cask)	Length*	25 ft	7.6 m
	Width*	25 ft	7.6 m
	Thickness*	3 ft	0.9 m

Source: BNFL 2000
* Assumed for this study

The VCC is a reinforced-concrete cask in the shape of a hollow right circular cylinder. The VCC has four penetrations for air entry (located at the bottom of the VCC) and four outlets located at the top. The penetrations are protected from debris intrusion by wire mesh screens. The internal cavity of the VCC, as well as the inlets and outlets, are steel lined. After the MSB is inserted, a shield ring is placed over the MSB/VCC gap and the cask weather cover is installed. The VCC height is fuel-specific and varies from 5.0 to 5.7 m (197 to 225 in.). The outer diameter is approximately 3.4 m (132 in.), and the walls consist of 74 cm (29 in.) thick concrete and a 4.5 cm (1.75 in) thick steel liner.

The MTC is a shielded lifting device with inner and outer structural steel cylinders, which house lead and solid RX-277 neutron shield cylinders designed to reduce radiation from the fuel inside the MSB/MTC. The MTC functions to transfer the MSB from the spent fuel pool (SFP) to the VCC inside the fuel pool building.

Operations

The major operating systems are those required for handling and transferring the fuel from the spent fuel pool to the ISFSI for storage, and likewise for removing the fuel from the ISFSI. First, the MSB is placed in the MTC and is lowered into the SFP. After the fuel is loaded into the MSB, the MSB shield lid is placed on the MSB, and the MTC is raised out of the SFP. The MSB and MTC are then decontaminated and drained. The MSB is vacuum dried, pressurized with helium, and sealed by welding. The MSB is then transferred to the VCC. The VCC is then sealed and is transferred to the concrete pad. Unloading procedures are similar to the loading procedures (in reverse).

The decay heat is removed passively by natural draft convection. Air enters the lower part of the VCC, rises around the MSB, and exits through the top. The system is self-regulating, and the only required maintenance is the periodic inspection of the air inlet and outlet screens to ensure that they have not been blocked by debris. Normal radiation monitoring is also performed.

Contamination Levels

Calculated end-of-life contamination levels for a concrete cask storage system are presented in BNFL 2000a. The calculated volume of contaminated material is reproduced in Table 2-66, while Tables 2-67 and 2-68 give the calculated specific activities (μCi/g) in steel and concrete components, respectively.

Table 2-66. Activated Material in a Concrete Cask

Component	Volume
Metal components (guard rails, heat shield, steel liner, reinforcement, bottom plate)	58.1 ft^3
Reinforced concrete (wall segments)	<544 ft^3
Base pad (underneath storage cask)	<30 ft^3

Source: BNFL 2000a

Table 2-67. Specific Activities (μCi/g) of Steel Components in Concrete Cask

Radionuclide	Storage Cask		Transfer Cask	Canister
	Rails, Liners, & Bottom Plate	Heat Shield		
Nb-95	8.47e-11	1.75e-11	2.54e-11	2.54e-11
Fe-59	6.71e-06	1.39e-06	2.26e-06	2.27e-06
Co-58	1.35e-05	2.80e-06	5.98e-04	6.17e-04
Zr-95	1.01e-09	2.09e-10	2.51e-11	2.56e-11
Zn-65	8.59e-10	1.78e-10	9.66e-10	1.50e-09
Mn-54	1.39e-03	2.88e-04	1.85e-04	3.25e-04
Fe-55	3.84e-02	7.95e-03	3.86e-03	1.70e-02
Co-60	4.54e-03	9.38e-04	1.73e-03	1.38e-02
Ni-63	2.60e-04	5.39e-05	8.27e-05	2.25e-03
C-14	1.31e-06	2.71e-07	8.40e-08	2.52e-06
Nb-94	—	—	1.99e-09	5.97e-08
Ni-59	2.21e-06	4.57e-07	6.29e-07	1.89e-05

Source: BNFL 2000a

Multiply by 3.7×10^4 to convert from μCi/g to Bq/g.

Table 2-68. Radionuclide Concentrations (μCi/g) of Concrete Components in a VSC-24 Cask

Radionuclide	Concrete Components
Tm-170	1.72e-05
Ca-45	2.28e-04
Cs-134	1.18e-05
Fe-55	1.90e-04
Co-60	6.25e-05
Eu-154	2.15e-06
Eu-152	1.35e-05
Ca-41	5.33e-06

Source: BNFL 2000a

2.3.5.2 Horizontal Modular Storage (NUHOMS)

A popular version of a concrete ISFSI is the horizontal modular storage, or NUHOMS (NUtech HOrizontal Modular Storage) system, developed by the Vectra Company and now owned by Transnuclear West, Inc. The standardized NUHOMS-24P/52B is designed to store either PWR or BWR assemblies horizontally in a concrete structure, rather than vertically in a cask. The principal components of the standardized NUHOMS are: (1) a stainless steel, dry-shielded canister (DSC) with an internal fuel basket, (2) a concrete horizontal storage module (HSM) that protects the DSC and provides radiological shielding (overpack), (3) a transfer cask (TC) used to transfer the DSC from the spent fuel pool to the HSM, and (4) a hydraulic ram system (HRS) used to insert the DSC into the HSM and TC. The following discussion was taken primarily from NRC 1996b, and supplemented with information and data from TNW 2000. A horizontal modular storage system is illustrated in Figure 2-9. The weights of the principal NUHOMS components are shown in Table 2-69, with dimensions provided in Table 2-70.

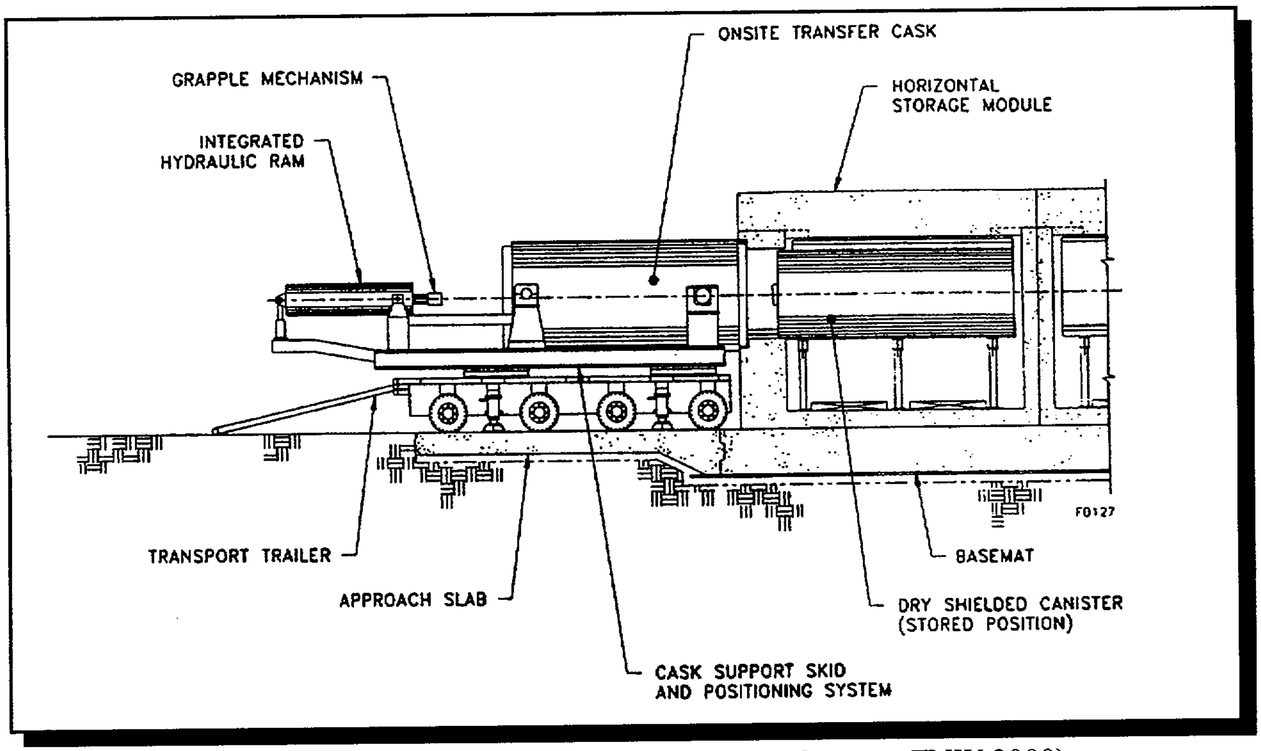


Figure 2-9. HSM Spent Fuel Storage System (Source: TNW 2000)

Table 2-69. HSM Principal Component Weights

Component	Material	Weight (lbs)		Weight (Mg)	
		PWR	BWR	PWR	BWR
DSC Shell Assembly	Steel	15,778	15,658	7.163	7.109
DSC Top Shield Plug	Lead	7,859	7,621	3.568	3.460
DSC Internal Basket	Steel	12,189	12,012	5.534	5.453
DSC Top Cover Plates	Steel	1,934	1,934	0.878	0.878
Horizontal Storage Module	Total	243,000	252,000	110.322	114.408
	Concrete*	226,600	234,900	102.876	106.645
	Rebar*	9,852	10,213	4.473	4.637
	Structural Steel*	6,548	6,887	2.973	3.127
Transfer Cask w/collar	Total	107,091	113,501	48.619	51.529
	Steel*	74,031	80,441	33.610	36.520
	Lead*	33,060	33,060	15.009	15.009
Basepad	Concrete*	77,100	80,500	35.003	36.547
	Rebar*	3,352	3,500	1.522	1.589

Source: TNW 2000, Tables 8.1-4 & 8.1-5.
* Estimated for this study

Table 2-70. HSM Component Dimensions

Component	Parameter		Dimensions	
DSC Shell Assembly	OD		67.25 in	1.7 m
	Thick		0.625 in	1.6 cm
	Length – PWR/BWR		186/196 in	4.7/5.0 m
	Cavity – PWR/BWR		167/177 in	4.2/ 4.5 m
Horizontal Storage Module	Height		15 ft	4.6 m
	Width		9 ft-8 in	2.9 m
	Length – PWR/BWR		19ft/19 ft-10 in	5.8/6.0 m
	Thicknesses	Roof	3 ft	0.9 m
		Front	2 ft-6 in	76. cm
		Side	1 ft-6 in	46. cm
		Rear	1 ft	30. cm
		Floor	1 ft	30. cm
		End	2 ft	61. cm
Transfer Cask w/collar	Overall Length		205.5 in	5.2 m
	Cavity Length		186.75 in	4.7 m
	OD/ID		85.25/68 in	2.2 m
	Thicknesses – S/L/S/N/S		0.5/3.5/1.5/3.0/0.125 in	1.3/8.9/3.8/7.6/0.3 cm
Basepad	Width		9 ft-8 in	2.9 m
	Length – PWR/BWR		19 ft/19 ft-10 in	5.8/6.0 m
	Thickness		3 ft	0.9 m

Source: TNW 2000

The DSC is designed to provide primary containment for 24 PWR or 52 BWR assemblies. The DSC is a stainless steel cylinder approximately 4.7 m (186 in.) long, with an outside diameter of 1.7 m (67.25 in.), and 0.016 m (0.625 in.) thick. Stainless steel end plates and steel end plugs filled with lead are welded to both the top and bottom of the DSC with redundant seal welds. The canister contains a basket assembly made of 24 or 52 guide sleeves consisting of stainless steel. The basket geometry and guide sleeves provide criticality control. The basket assembly for BWR assembly loading has additional neutron-absorbing plates. The lower end of the DSC is coated with a lubricant to reduce friction, when it is inserted and removed from the TC and HSM.

The HSM is constructed of reinforced concrete, structural steel, and stainless steel. The HSM may be constructed as a single unit or as an array of modules (e.g., 2×20). A Standardized NUHOMS HSM is approximately 5.8 m (19 ft) long, 4.6 m (15 ft) high, and 2.9 m (9'-8") wide. The concrete walls and roof are 91 cm (3 ft) thick, and interior walls are 46 cm (18 in.) thick. The outside wall at the end of a row of HSMs has a thickness of 61 cm (2 ft). Gamma and neutron shielding are provided by the HSM structure.

A steel support rail structure anchored inside the HSM by the interior walls supports the DSC and extends to the access opening. Stoppers on the rails prevent horizontal movement of the DSC during a seismic event. A vertically sliding plate, consisting of thick steel and a neutron-absorbing material, covers the entrance to the HSM and is tack welded closed once the DSC is in place. Each HSM has two shielded air inlets on the front and two shielded air outlets on the roof.

The TC is used to transfer the DSC from the SFP to the HSM. The TC is approximately 5.2 m (205.5 in.) long with an inner diameter of 1.7 m (68 in.). The length can be extended to accommodate BWR assemblies. It consists of three concentric cylinders with shielding material in between, connected by top and integral end plates. The top and bottom end plates are made of steel and a solid neutron shield. The bottom end plate has a removable HRS access port plug. The TC wall consists of an inner stainless steel liner, a poured-lead shield, a structural carbon steel shell, a solid BISCO-N3 neutron shield, and an outer carbon steel shell. It is hoisted by the trunnions located on its sides, and mates (via the transfer trailer) with the access opening of the HSM for transfer of the DSC.

The HRS provides the motive force for transferring the DSC between the HSM and TC. The HRS consists of a single-stage hydraulic cylinder with a grapple assembly. The hydraulic cylinder is positioned by support frame and is designed to apply pushing or pulling forces of 90 kN (20,000-lb force) during normal operation.

Operations

The major operating systems are those required for handling and transferring the fuel from the SFP to the ISFSI for storage, and likewise for removing the fuel from the ISFSI. First, the DSC is placed in the TC, and is lowered into the SFP. After the fuel is loaded into the DSC, the DSC shield plug is placed on the DSC and the TC is raised out of the SFP. The DSC and TC are then decontaminated and drained. The DSC is vacuum dried, pressurized with helium, and sealed. The TC lid is then bolted to the cask, and the TC is lowered horizontally onto a transfer trailer. The TC is transferred to the ISFSI and the cask is mated (top side) to the HSM, as shown in Figure 2-9. The HRS arm is inserted through the TC rear access port and pushes the DSC into the HSM. The TC is then removed and the access steel cover plate is tack welded sealed to the HSM. Unloading procedures are similar to the loading procedures (in reverse).

The decay heat is removed by natural draft convection. Air enters the lower part of the HSM, rises around the DSC, and exits through the top shielded slab. The only required maintenance is the periodic inspection of the air inlet and outlet screens to ensure that they have not been blocked by debris.

Contamination Levels

Section 9.6, "Decommissioning Plan," of TNW 2000 states:

> *The NUHOMS system is a dry containment system that effectively confines all contamination within the DSC. When the DSC is removed from the HSM, the free standing HSM can be manually decontaminated for any trace activity, dismantled and removed from the site. It is possible that a thin layer of material comprising the inner wall of the HSM could become activated by the neutron flux from the fuel after an extended period of service. Estimates of the potential for activation are difficult due to the variability of rare earths which may be present in the local aggregate. The specific activity of the HSM inner wall surfaces may be measured at the time of decommissioning and compared with the existing guidelines to determine whether the values are below regulatory concern (BRC). Disposal procedures can then be developed, which comply with existing guidelines at the time of decommissioning.*

2.3.5.3 Metal Casks

Metal cask ISFSI designs are available from a number of vendors, as shown in Table 2-63. The CASTOR system, developed by General Nuclear Systems, Inc., and currently owned by Gesellschaft für Nuklear-Behälter, mbH (GNB), has been selected as the reference design for this study. The discussion below was taken primarily from NRC 1996b and supplemented with information and data from GNB 2000. Figure 2-10 provides a illustration of the CASTOR cask, while the cask's weights are shown in Table 2-71.

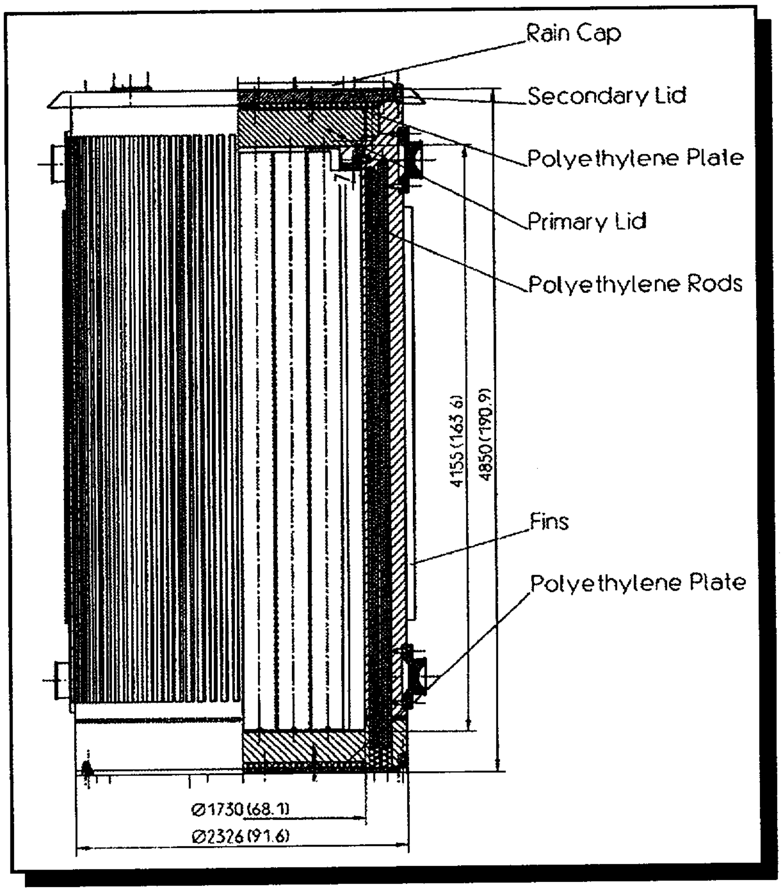


Figure 2-10. CASTOR Metal Cask Spent Fuel Storage System (Source: GNB 2000)

Table 2-71. CASTOR Metal Cask Weights

Component	Material	Weight (lbs)	Weight (Mg)
Cask Weight	Steel	180,557	81.973
Basket	Steel	32,052	14.552
Basepad (per Cask)	Concrete*	110,592	50.209
	Rebar*	4,792	2.176

Source: GNB 2000
* Estimated for this study from VEPCO 2000

The CASTOR X/32 has been designed to vertically store 32 PWR assemblies. The cask is approximately 4.8 m (190.9 in.) high and 2.3 m (91.6 in.) in (outside) diameter. It weighs approximately 96.5 metric tons (106.3 tons), without any fuel.

Table 2-72 shows the dimensions of the CASTOR metal cask's main components. The cask body consists of a ductile cast-iron material with a thickness of 30 cm (11.8 in.). The top of the cask is sealed with two stainless steel lids bolted onto the cask, using both metallic and elastomeric O-ring seals. The primary and secondary lids are 26 cm (10.2 in.) and 8 cm (3.15 in.) thick, respectively. Gamma shielding is provided by the wall of the cask, and neutron shielding is provided by a single row of polyethylene rods incorporated into the cask wall. The inside of the cask and sealing surfaces have a nickel coating for corrosion protection. The internal cavity is filled with helium for heat transfer and corrosion protection. An epoxy resin coating protects the outside surface of the cask. Four trunnions are connected to the cask body for lifting and rotating the cask. The inside of the cask contains a fuel basket structure comprising 32 square tubes of welded stainless steel and borated stainless steel plates (for criticality control).

Table 2-72. CASTOR Metal Cask Principal Component Dimensions

Component	Material	Parameter	Dimensions	
Cask Body	Carbon Steel	Overall Length	190.9 in	4.8 m
		Cavity Length	163.6 in	4.2 m
		Outside Diameter	91.6 in	2.3 m
		Inside Diameter	68.1 in	1.7 m
		Wall Thickness	11.8 in	30. cm
		Bottom Thickness	7.0 in	18. cm
Primary Lid	Stainless Steel	Thickness	10.2 in	26. cm
Secondary Lid		Thickness	3.15 in	8. cm
Bottom Plate		Thickness	1.38 in	4. cm
Moderator Rod	—	Thickness	2.76 in	7. cm

Source: GNB 2000

Operations

The major operating systems are those required for handling and transferring the fuel from the SFP to the ISFSI for storage, and likewise for removing the fuel from the ISFSI. The cask is loaded underwater in the SFP and the primary lid is placed on the cask. The cask is then lifted to the pool surface, and the seal of the primary lid is fastened and tested for tightness. After being lifted out of the SFP, the cask is pumped empty and vacuum dried. The secondary lid is fastened, the seals are tested, and the space between the lids is pressurized with helium. Next, a pressure-sensing device is mounted in the secondary lid, a protective transport impact limiter is installed, and the outside surface is decontaminated. The cask is then moved to the ISFSI site and is set in place on the concrete pad. The seal pressure-monitoring system is externally connected and will notify plant operators of a loss of seal integrity. Unloading procedures are similar to the loading procedures (in reverse). The cask is a totally passive system with natural cooling sufficient to maintain safe fuel cladding temperatures. The cask wall provides adequate shielding, and no radioactive products are released under any credible conditions. Normal radiation survey monitoring is also performed.

Contamination

Table 2-73 presents the end-of-life activation levels reported in GNB 2000.

Table 2-73. CASTOR Metal Cask Activation Levels

Component		Volumetric Activity (Ci/m³)*				
		C-14	Ni-59	H-3	Co-60	Ni-63
Cask Body		1.72e-11	1.22e-06	—	9.67e-04	1.40e-04
Restrainers		2.09e-11	9.64e-06	—	6.77e-03	1.11e-03
Lid	Primary	1.09e-12	5.03e-07	—	3.53e-04	5.77e-05
	Secondary	2.84e-14	1.31e-08	—	9.19e-06	1.50e-06
Trunnions		1.38e-14	6.34e-09	—	4.45e-06	7.27e-07
Fins		—	—	—	1.24e-07	3.14e-10
Moderator	Between Lids	6.41e-11	—	2.24e-10	—	—
	Sidewalls	1.86e-10	—	6.52e-10	—	—
	Bottom Side	1.59e-10	—	5.55e-10	—	—
Gusset Plates		—	—	—	3.13e-05	4.23e-11
Receptacles & Basket Bottom		—	2.10e-05	—	1.49e-02	2.41e-03
Heat Transfer Plate		4.42e-10	—	—	1.21e-04	—
Bottom Plate		1.31e-12	9.28e-08	—	7.35e-05	1.06e-05

Source: GNB 2000

* Multiply by 4700 to convert from Ci/m³ to Bq/g.

2.3.5.4 Modular Vault Dry Store

The Modular Vault Dry Store (MVDS) system, made by Foster Wheeler Energy Applications, Inc., is a concrete vault ISFSI designed to vertically store 1482 high-temperature gas-cooled (HTGC) spent fuel elements, 37 reflector elements, and 6 neutron source elements from the Fort St. Vrain power station, which has been decommissioned. A diagram of the MVDS system is shown in Figure 2-11. The MVDS system is unique among other licensed ISFSIs. It is the only installation that has the capability to handle and transfer HTGC fuel from 10 CFR Part 71 approved shipping casks at the actual ISFSI site. It is also equipped to conduct decontamination operations, and it uses a monitored nitrogen (rather than helium) cover gas.

The MVDS system consists of a foundation structure supporting a matrix of six concrete vault modules (VMs), one neutron source storage well, two standby storage wells, six charge face structures (CFSs) forming the roofs over the VMs, and a transfer cask reception bay, with a steel canopy above the structure. Overall dimensions of the MVDS facility are 44 m (143 ft) long, 22 m (72 ft) wide, and 25 m (81 ft) high. Each concrete VM contains a matrix of 45 storage positions capable of storing six fuel storage containers (FSCs) in each position. The neutron storage well, which is separated from the VMs, is capable of storing six neutron source elements and can be individually sealed. The two standby wells, which are separated from the VMs, are capable of storing failed FSCs and can be individually sealed. Neutron and gamma shielding is provided by the concrete mass of the ISFSI. The inherent geometry of the system maintains protection against criticality.

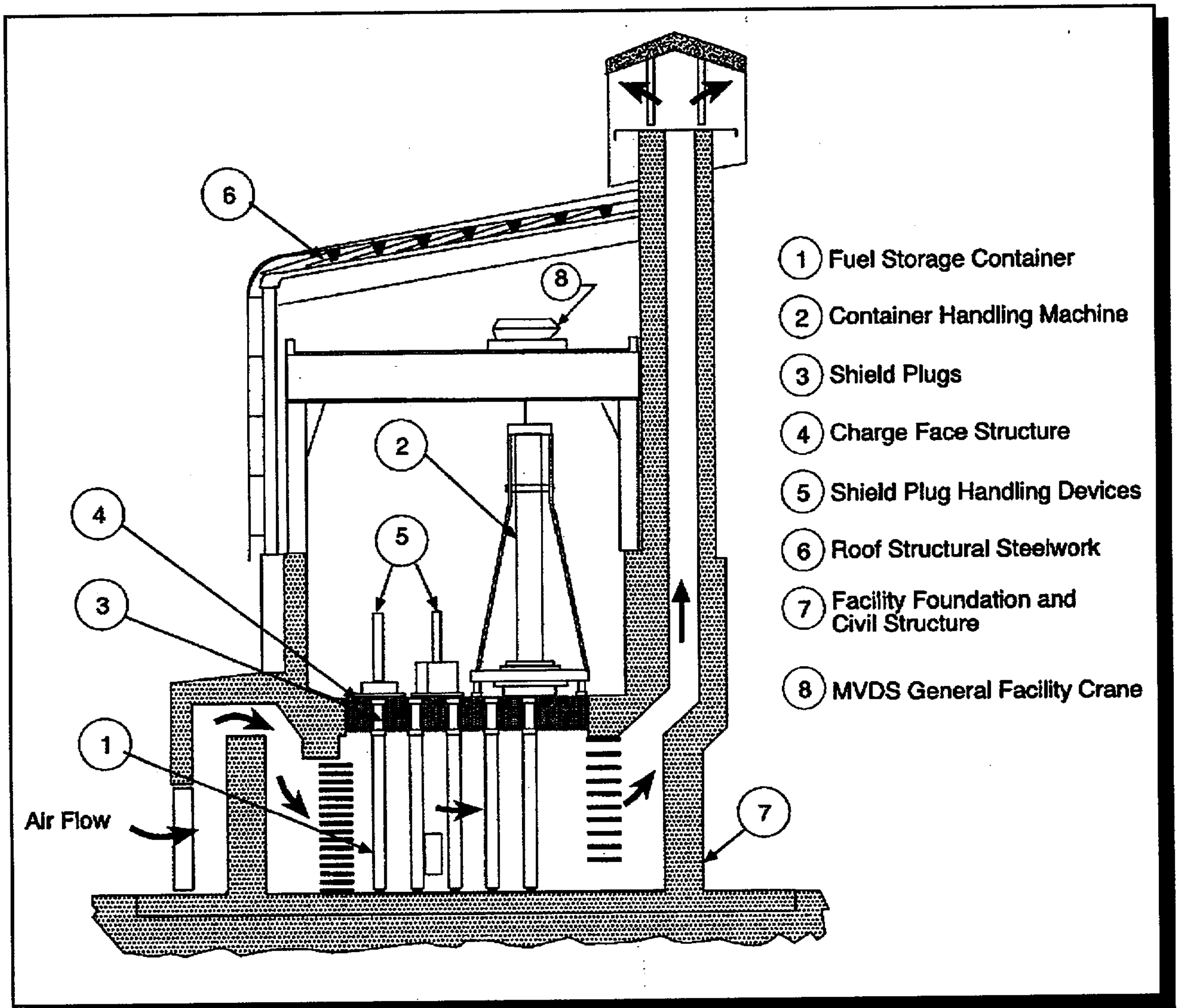


Figure 2-11. Diagram of a Modular Vault Dry Storage Facility (Source: NRC 1996a)

Based on the dimensions given above and a floor and wall thickness of 1.2 m (4 ft), the volume of concrete in the MVDS has been estimated at 5,125 m^3. The amount of reinforcing steel associated with this volume of concrete is estimated to be 511 Mg.

The FSC consists of a cylindrical carbon steel tube body with an exterior aluminum spray coating, O-ring seals, and a sealed lid, which is placed on the tube flange. The FSC can hold 6 fuel elements or 12 reflector elements. The top of the FSC is shielded by a plug placed in the CFS above the FSC during storage.

The Container Handling Machine (CHM) is a fully shielded machine used to raise and lower the FSCs to the TC and vault storage locations. It is moved over the CFS by the MVDS crane. The CHM consists of a main shield tube, a single-failure-proof raise/lower mechanism, and an FSC grapple by which an FSC can be raised or lowered. There is also an individual fuel element grapple, which can be used to lift fuel elements in the standby storage well during off-normal and accident operations.

No additional data concerning the weight of the equipment within the MVDS has been identified at this time. In PNL 1984, Table 9.5-2, the mass of contaminated material from the dismantlement of the Vault ISFSI is given as 37.7 Mg, with a contamination level of 15.2 Ci. The majority of this material came from the fuel storage room and consists of the fuel baskets, sleeve storage racks, and plenum. The PNL 1984 Vault ISFSI has a much larger "footprint" than the MVDS (5,900 m^2 versus 968 m^2). Adjusting the amount of contaminated material for this difference gives 6.2 Mg of equipment for the MVDS, with an average contamination level of 1.5×10^4 Bq/g (4×10^5 pCi/g).

Operations

The major operating systems are those required for handling and transferring the FSC from the TC into MVDS storage position, and those systems for removing the FSCs. In the reactor building, the fuel elements are loaded into the FSC, and the TC is lowered back onto the trailer. The TC is received in the MVDS in the transfer cask reception bay. The TC is lifted by the MVDS crane and positioned in the cask load/unload port. The FSC is then removed by the CHM, which, in turn, is lifted and moved by the MVDS crane over the CFS of the storage vault. Once the CHM reaches the desired position above the CFS, the crane lowers the CHM to the CFS, and the FSC can be inserted through the hole in the CFS into the VM. Unloading procedures are similar to the loading procedures (in reverse).

The decay heat is removed passively by ambient air flowing across the outside of the FSCs. The air flows into the vault module through a mesh-covered inlet duct and exits through a reinforced concrete exhaust stack covered by a steel canopy.

2.3.5.5 Pool (Wet) Storage

The only ISFSI to utilize pool or wet storage is General Electric's Morris Operations (GEMO). The GEMO facility was constructed in the late 1960s as a fuels reprocessing plant, but it never operated as such. For the last 18 years, the GEMO has functioned as an ISFSI under NRC License SNM-2500. In May 2000, GE applied to the NRC for an extension to SNM-2500 so that it could continue to store spent fuel at GEMO for 20 additional years (NRC 2000d).

The main building at GEMO is a massive structure of reinforced concrete, about 62 m (204 ft) by 24 m (78 ft), and about 27 m (88 ft) above grade. The western end of the building houses most of the fuel storage facilities. This portion of the building is of steel frame and insulated metal siding construction and is attached to the concrete main building. The fuel storage (western) portion of the main building is shown in Figure 2-12. Fuel storage operation areas include:

- Cask receiving area.
- Decontamination area.
- Cask unloading basin.
- Fuel storage Basins 1 and 2: Basin 1 has an area of about 84 m^2 (900 ft^2); Basin 2 has an area of about 139 m^2 (1,500 ft^2). There are a total of 414 fuel basket positions: 150 in Basin 1 and 264 in Basin 2.
- Low level waste evaporator (not shown on Figure 2-12).
- CAS/SAS (was Control Room) (not shown on Figure 2-12).
- Basin water cleanup and cooling system (not shown on Figure 2-12).

Fuel bundles are stored in stainless steel basket assemblies designed to protect fuel from physical damage and to maintain fuel in a subcritical configuration. Baskets are locked into grids in the fuel basins to provide seismic restraint. The basins are constructed below ground with stainless steel lined, reinforced concrete walls about 0.6 m (2) ft thick poured in contact with the sides of a

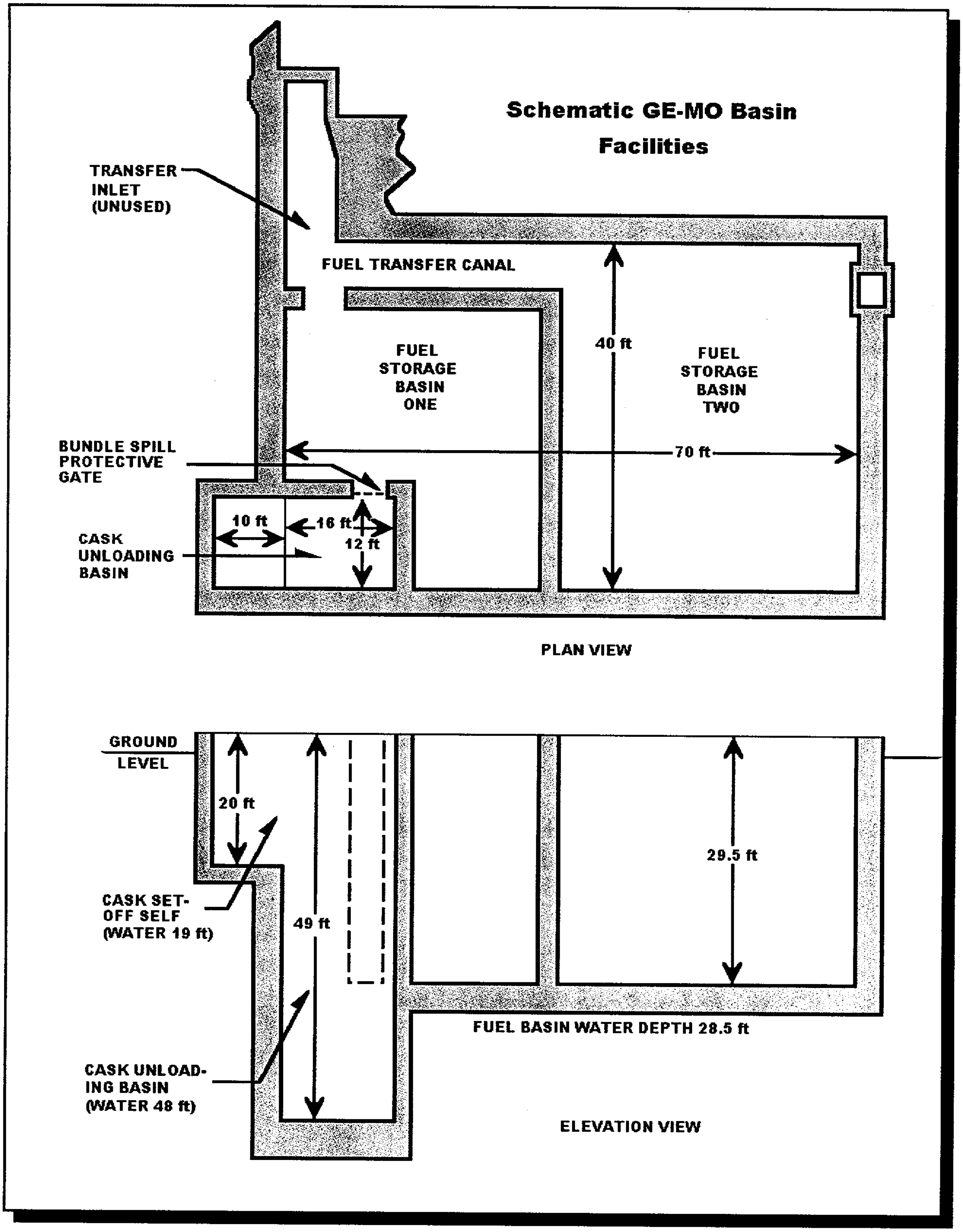


Figure 2-12. Pool (Wet) Storage Facility (Source: GE 2000)

bedrock excavation. The south wall of the basin is about 1.2 m (4 ft) thick, because it was intended to stand independent of the surrounding rock to facilitate possible future expansion. Storage basin floors were poured on bedrock and range in thickness from 76 cm to 1.4 m (30 to 54 in.). Reinforcing steel used in the basins has a 40,000 psi minimum yield strength.

The steel floor liner in the cask unloading pit is 0.6 cm (0.25 in.) thick and is placed over a 4.4 cm (1.75 in.) thick steel plate provided for distributing impact loads over the underlying concrete structure. The set off shelf liner, also 0.6 cm (0.25 in.) thick, is placed directly on the concrete structure with an energy absorbing assembly placed on top of the liner. For the remainder of the storage basin complex, the floor liner is 0.5 cm (0.187 in.) thick. Walls of the cask unloading pit, including shelf area, are lined with 11 gauge sheet steel.

Table 2-74 gives the estimated mass of ferrous metal in the Pool (Wet) Storage ISFSI. The mass of the storage baskets is based on the per assembly weight of PWR and BWR storage racks (and checked against baskets made from nine 8" (BWR) or four 12" (PWR) Schedule 5 pipe – the GEMO designs). The mass of the basin liners was also estimated, based on the dimensions of the GEMO storage basins. The other masses were taken from NRC 1996b for the Fuel Pool Cooling and Cleanup System.

Table 2-74. Pool Storage Ferrous Metal Weights*

Component	Mass (kg)
Fuel Storage Baskets	360,000
Basin Liners	56,000
FPCC pumps	1,054
FPCC Demineralizer	3,132
Skimmer Surge Tank	10,708
FPCC Heat Exchanger	4,076
Supp. Pool Cleanup Pump	527
Resin Tank Agitator	72
Fuel Pool Precoat Pump	284
(Precoat) Dust Evacuator	104
FPCC Hold Pump	390
FPCC Precoat Tank	227
FPCC Resin Tank	227
Valves (1 - 10" dia.) and Components	8,038
Turnbuckles	5,922
TOTAL	450,761

* Weight of racks and liners estimated for this study; other weights from NRC 1996a.

The pool storage area has been estimated to contain approximately 840 m^3 of concrete, with an estimated 111 Mg of rebar. The entire Main Building (including those portions intended for fuel reprocessing) would contain substantially more concrete and rebar.

Contamination

Principal radioactive contaminants in the storage basin water include fission products Cs-134 and Cs-137, with typical concentrations of 3.3×10^{-7} and 4.2×10^{-4} µCi/ml, respectively. Activation product Co-60 is present in a typical concentration of 2.6×10^{-5} µCi/ml. A maximum concentration of 5×10^{-3} µCi/ml was measured at the end of a 3-week period during which the filter was purposely not operated. Similar levels of contamination have occurred in recent years. Table 2-75 summarizes the typical radionuclide concentration in GEMO basin water.

Table 2-75. Typical Radionuclide Concentrations in Basin Water

Radionuclide	Concentration (µCi/ml)
Cs-134	3.3×10^{-7}
Cs-137	4.2×10^{-4}
Co-60	2.6×10^{-5}
H-3	1.1×10^{-4}

Source: GE 2000, Table 5-1

2.3.5.6 Material Projections

The Nuclear Energy Institute estimates that by the end of 2004, 30 nuclear power reactors will run out of spent fuel pool space, and 60 reactors will need additional storage space by the end of 2006, and 78 by the end of 2010 ((http://www.nei.org). Table 2-76 shows some of the ISFSIs planned to address this need.

Two of the proposed ISFSIs identified in Table 2-76 would be independent, privately owned and operated facilities: the Owl Creek Energy Project located in Shoshoni, Wyoming, and the Private Fuel Storage LLC located on the Reservation of the Skull Valley Band of Goshute Indians in Tooele County, Utah.

Table 2-76. Planned Dry Spent Fuel Storage Facilities

Facility	Process	Status
Big Rock Point	Cask	Planned
Dresden NPP	Cask	Under construction
McGuire	Cask	Planned
Owl Creek Energy Project	Dry	Planned
Oyster Creek NPP	Transnuclear West	Planned
Peach Bottom	Dry	Planned
Private Fuel Storage LLC	Cask	Planned
Rancho Seco ISFSI	Transnuclear West	Planned

Using the data from Table 2-62, Table 2-77 was constructed to show the percentage of usage of the various types of ISFSIs. Only PWR data are used, since only a single BWR (Susquehanna) facility is shown in Table 2-62. Also, Pool (Wet) Storage and the MVDS have not been included, since these concepts have not proven to be favored by the potential licensees (i.e., utilities)[4].

Table 2-77. Units of Each Type of Dry Storage Facility for PWRs

Type	Number of Assemblies	Percentage	Assemblies per Unit	Number of Units
NUHOMS	6448	67.2%	24	269
Metal Casks	1086	11.3%	21	52
Concrete Casks	2066	21.5%	24	86

Data from DOE 1996 were used to project the number of fuel assemblies expected to be discharged from nuclear power plants through 2040. Figure 2-13 shows the annual projected PWR and BWR discharges, while Figure 2-14 shows the cumulative discharges. Based on these data, Table 2-78 shows that current storage capacity should be sufficient until after the year 2010. However, if there is no disposition of spent fuel before 2020, additional ISFSIs will be required to store the fuel being discharged from reactors. Relicensing of nuclear reactors will increase the quantities of spent fuel for storage and eventual disposal.

[4] In May 2000, Foster Wheeler received an award from DOE INEEL to provide an interim storage facility for 55 metric tons of spent nuclear fuel. The primary functions of the MVDS are to receive three types of spent nuclear fuel from DOE including TRIGA, Peach Bottom, and Shippingport fuels; remove the individual fuel elements from the canisters in which they are currently housed; inspect and repackage the fuel into new DOE canisters; and transfer the canisters to the interim storage area. This facility will be licensed by the NRC (FW 2000).

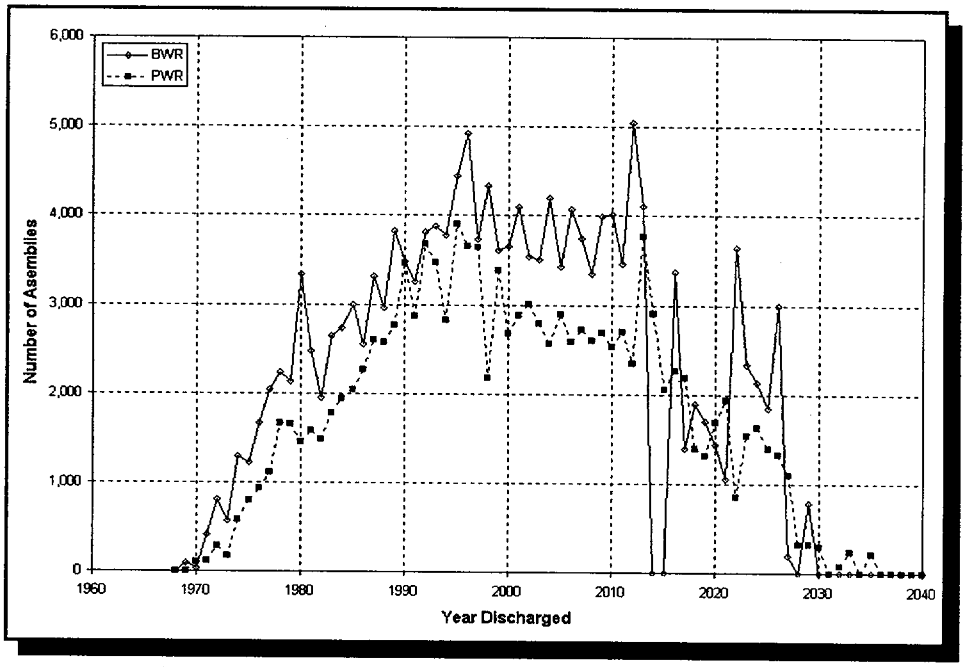


Figure 2-13. Projected Annual Spent Fuel Discharge
(From data provided in DOE 1996a)

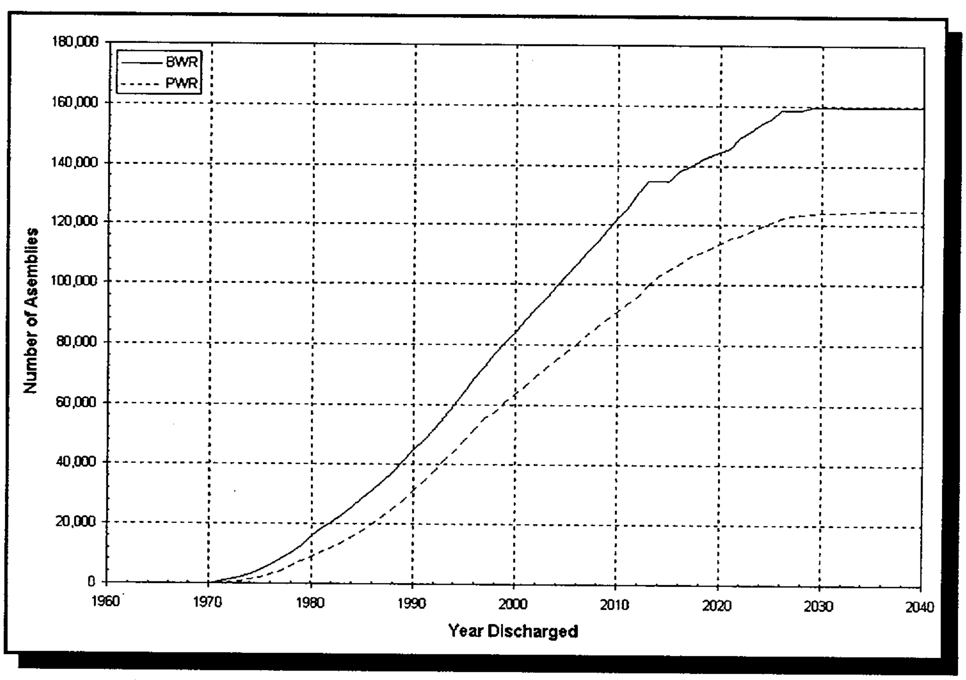


Figure 2-14. Projected Cumulative Spent Fuel Discharge
(From data provided in DOE 1996a)

Table 2-78. Projected ISFSI Requirements*

Assemblies		BWR	PWR
Current Capacity	In Pool	~118,000	~87,000
	Dry	5,460	9,600
	Total	~123,460	~96,600
Projected Inventory	2010	122,114	90,876
	2020	144,525	113,515
	2030	159,505	124,230
	2040	159,505	124,761

* Estimated based on data given in DOE 1996a.

As shown by Table 2-78, the quantity of fuel assemblies stored in ISFSIs could rise from its current levels of about 5,460 BWR and 9,600 PWR assemblies to about 26,500 assemblies of each type by 2020. The number of ISFSIs could be further increased by the decommissioning and dismantlement of power plants and their associated spent fuel pools, causing many of the spent fuel assemblies currently stored in the pools to be relocated to ISFSIs.

Contamination Levels

The volume of activated material for concrete cask type ISFSIs is given in Table 2-66. Combining these data with the cask weights given in Table 2-64 shows that about 96 percent of the MSB cask metal, about 50 percent of the cask concrete, and <2 percent of the basemat would become activated. The radionuclide specific activation levels are given in Tables 2-67 and 2-68 for metal and concrete components, respectively. The maximum Co-60 activation level would be about 511 Bq/g in the storage canister.

Although no contamination levels are presented for the NUHOMS, it would likely be activated to levels similar to those of the concrete cask.

Table 2-73 gives the activation levels for metal cask ISFSIs. The maximum Co-60 activation level would be about 70 Bq/g in the receptacles and bottom basket for the metal cask. The volume of activated material is not given for the metal cask, but it can be assumed to be similar to the volumes given for the concrete cask. Likewise, the activation of the basemat for the metal casks can be assumed to be activated to levels similar to the concrete cask's basemat.

Finally, the NRC assumes (NRC 1994) that 10 percent of the basemat of an ISFSI would become contaminated due to the "release of undetected contamination carried on the external surfaces of the storage cask to the storage pad." Table 7.1.1 of NRC 1994 gives the ISFSI contaminated surface activity as 980 dpm/100 cm^2 (0.16 Bq/cm^2) for Co-60 and 310 dpm/100 cm^2 (0.052 Bq/cm^2) for Cs-137.

2.3.5.7 Inventory Summary - Spent Fuel Storage Facilities

Mass of Steel

Concrete casks contain a steel mass of 3.65 Mg per assembly, and a rebar mass of 0.35 Mg per PWR assembly (Table 2-64). NUHOMS contain a steel mass of 2.1/1.0 Mg per PWR/BWR assembly, and a rebar mass of 0.06/0.03 Mg per PWR/BWR assembly (Table 2-69). Metal casks contain a steel mass of 3.0 Mg per PWR assembly, and a rebar mass (in the basepad) of 0.07 Mg per assembly (Table 2-71).

The steel mass contained in the modular vault dry storage facility was estimated as 468 Mg of rebar and 6.2 Mg of contaminated equipment. The steel mass in the pool (wet) storage facility was estimated as 451 Mg of equipment and 77 Mg of rebar.

Mass of Concrete

Concrete casks contain a concrete mass of 8.0 Mg per PWR assembly, including the basepad. NUHOMS contain a concrete mass of 5.7/2.8 Mg per PWR/BWR assembly, including the basepad. Metal casks contain a concrete mass in the basepad of 1.6 Mg per PWR assembly.

The mass of concrete contained in the modular vault dry storage facility was estimated to be 11,800 Mg, while the concrete mass in the pool (wet) storage facility was estimated as 1,940 Mg.

Masses of Copper and Aluminum

No studies were located during the preparation of this report that describe the quantity or associated levels of radioactivity of copper or aluminum contained in spent fuel facilities. However, for concrete casks, NUHOMS and metal casks, the mass of these metals is anticipated to be negligibly small. For modular vault dry store and pool (wet) storage, the mass of these metals is also likely to be quite small.

Scaling from Reference Facility

For concrete casks, NUHOMS and metal casks, the material masses will be scaled based on the calculated number of fuel assemblies stored in each type of storage facility. For modular vault dry store and pool (wet) storage, the material masses will be un-scaled, since there is only one of each type of these facilities.

Contamination Levels and Principal Radionuclides

As shown in Table 2-67 for concrete casks and Table 2-73 for metal casks, Fe-55 and Co-60 are two of the radionuclides expected to be of primary concern. From Draft NUREG-1640, Table 2.1, it is clear that of these two radionuclides, Co-60 with a mean dose factor of 250 μSv/y per Bq/g (0.925 mrem/y per pCi/g) is the most critical. The Co-60 specific activity ranges from 34.7 to 511 Bq/g (9.38×10^{-4} to 1.38×10^{-2} μCi/g, Table 2-67) for the steel components of the concrete cask, and 5.8×10^{-4} to 0.70 Bq/g (1.24×10^{-7} to 1.49×10^{-2} Ci/m^3, Table 2-73) for the steel cask. Although no activities were found for the NUHOMS system, it is anticipated that its contamination level would be similar to that of the concrete casks.

For the pool (wet) storage facility, Cs-137, in addition to Co-60, is likely to be a concern. The form of contamination in the pool (wet) storage facility is also likely to be different from the contamination expected in dry storage concepts. For the dry concepts, virtually all of the contamination would be from activation of the components from the stored spent fuel, while for wet storage, contamination would be transported through the pool water, and there would be surface as well as volumetric contamination.

2.4 Non-Power Reactors

Non-power reactors (NPR) come in many varieties and forms, with most being either pool-type or tank-type. Pool-type reactors have a core immersed in an open pool of water. The pools typically provide about 20 feet of water above the core to allow cooling and radiation shielding. At pool-type NPRs, the operating core and fuel can be observed through the pool water. Tank-type reactors have a core that is in a tank with water, sealed at the top, with the entire structure in the tank.

Non-power reactors are also categorized by fuel type: plate-type fuel, TRIGA (Training, Research, Isotopes, General Atomics), or AGN (Aerojet General Nucleonics). Plate-type fuel consists of several thin plates containing a uranium mixture clad with aluminum formed into an assembly. This geometry promotes efficient heat removal and the ability to provide a high-neutron density. TRIGA fuel is in the shape of rods and consists of a uranium and zirconium/hydride mixture. For some research, TRIGA fuel is used to generate large pulses of neutron energy which are self-limiting. These pulses are self-limiting because the associated fuel temperature increases quickly, reducing power and shutting down the reactor. This inherent design feature also acts as an additional safety margin. AGNs are compact, self-contained, low-power (<5 watts) tank-type reactors. The 10-inch diameter core consists of uranium oxide powder embedded in a polyethylene moderator.

Table 2-79 lists the currently licensed non-power reactors and other relevant information (including location, license and docket numbers, power level, etc.). Table 2-79 also provides the date of initial criticality for each licensed non-power reactor. It is assumed that each reactor will shut down 40 years after its initial criticality date. The date when its material will be available for release can be assumed to be approximately 5 years after shutdown. This assumption is highly speculative, since many non-power reactors have their licenses renewed for 20 years or more. For example, in May 2000, the University of Wisconsin requested that its TRIGA reactor license be renewed until June 30, 2020.

Table 2-79. Licensed Non-Power Reactors

Facility Name	Location	License Docket	Power (kW)	Type*	Initial Crit.**
POOL NON-POWER REACTORS — With plate-type fuel					
Ohio State University	Columbus, OH	R-75 50-150	500	LW Mod	1961 02/24
Purdue University	West Lafayette, IN	R-87 50-182	1	LW Mod Lockheed	1962 08/16
Rhode Island Nuclear Science Center	Narragansett, RI	R-95 50-193	2000	LW Mod GE	1964 07/21
University of Massachusetts	Lowell, MA	R-125 50-223	1000	LW Mod, GE	1974 12/24
University of Michigan - Ford Reactor	Ann Arbor, MI	R-28 50-2	2000	LW Mod	1957 09/13
University of Missouri	Rolla, MO	R-79 50-123	200	LW Mod	1961 11/21
Worcester Polytechnic Institute	Worcester, MA	R-61 50-134	10	LW Mod GE	1959 12/16
POOL NON-POWER REACTORS — With TRIGA fuel					
Aerotest Operations Inc.	San Ramon, CA	R-98 50-228	250	Conversion (Indus)	1965 07/02
Armed Forces Radiobiological Research Institute	Bethesda, MD	R-84 50-170	1100 +pulse	Mark F	1962 06/26
Cornell University	Ithaca, NY	R-80 50-157	500 +pulse	Mark II	1962 01/04
Dow Chemical Company	Midland, MI	R-108 50-264	300	Mark I	1967 07/03
Kansas State University	Manhattan, KS	R-88 50-188	250 +pulse	Mark II	1962 10/16
Oregon State University	Corvallis, OR	R-106 50-243	1100 +pulse	Mark II	1967 03/07
Pennsylvania State University	University Park, PA	R-2 50-5	1000 +pulse	Conversion Mark II	1955 07/08
Reed College	Portland, OR	R-112 50-288	250	Mark I	1968 07/02
Texas A&M	College Station, TX	R-83 50-128	1000 +pulse	Conversion	1961 12/07
U.S. Geological Survey	Denver, CO	R-113 50-274	1000 +pulse	Mark I	1969 02/04
U.S. Veterans Administration	Omaha, NE	R-57 50-131	18	Mark I	1959 06/26
University of Arizona	Tucson, AZ	R-52 50-113	100	Mark I	1958 12/05
McClellan Nuclear Radiation Center, UC-Davis	Sacramento, CA	R-130 50-607	2000 +pulse	Mark II	1998 08/13

Table 2-79. Licensed Non-Power Reactors (continued)

Facility Name	Location	License Docket	Power (kW)	Type*	Initial Crit.**
University of California	Irvine, CA	R-116 50-326	250	Mark I	1969 11/24
University of Maryland	College Park, MD	R-70 50-166	250	Modified	1960 10/14
University of Texas	Austin, TX	R-129 50-602	1100 +pulse	Mark II	1992 01/17
University of Utah	Salt Lake City, UT	R-126 50-407	100	Mark I	1975 09/30
University of Wisconsin	Madison, WI	R-74 50-156	1000 +pulse	Conversion	1960 11/23
Washington State University	Pullman, WA	R-76 50-27	1000 +pulse	Conversion	1961 03/06
PULSTAR (uranium dioxide pellets, zircalloy clad) fueled					
North Carolina State University	Raleigh, NC	R-120 50-297	1000	—	1972 08/25
CRITICAL EXPERIMENT FACILITY — With uranium dioxide, stainless steel clad fuel					
Rensselaer Polytechnic Institute	Schenectady, NY	CX-22 50-225	0.1	LW Mod	1964 03/07
TANK NON-POWER REACTORS — Plate-type fuel					
General Electric Company	Sunol, CA	R-33 50-73	100	Graphite Mod	1957 10/31
Massachusetts Institute of Technology	Boston, MA	R-37 50-20	4,900	LW Mod, HW Reflec	1958 06/09
National Institute of Standards and Technology	Gaithersburg, MD	TR-5 50-184	20,000	Heavy Water	1970 06/30
University of Florida	Gainesville, FL	R-56 50-83	100	Argonaut	1959 05/21
University of Missouri	Columbia, MO	R-103 50-186	10,000	LW Mod & Cooled	1966 10/14
TANK NON-POWER REACTORS — AGNs					
Idaho State University	Pocatello, ID	R-110 50-284	0.005	AGN-201 #103	1967 10/11
Texas A&M University	College Station, TX	R-23 50-59	0.005	AGN-201 #106	1957 08/26
University of New Mexico	Albuquerque, NM	R-102 50-252	0.005	AGN-201M #112	1966 09/17
NON-POWER REACTORS UNDER DECOMMISSION ORDERS OR AMENDMENTS:					
General Atomics	San Diego, CA	R-100 50-227	—	Mark II	—
		R-38 50-89	250	Mark I	—

Table 2-79. Licensed Non-Power Reactors (continued)

Facility Name	Location	License Docket	Power (kW)	Type*	Initial Crit.**
Georgia Institute of Technology	Atlanta, GA	R-97 50-160	5,000	HW Mod, Plate Fuel	—
Iowa State University	Ames, IA	R-59 50-116	10	Argonaut	—
Manhattan College	Riverdale, NY	R-94 50-199	0.1	Pool	—
University of Illinois	Urbana, IL	R-115 50-151	1	LW Mod, TRIGA	—
University of Washington	Seattle, WA	R-73 50-139	100	Argonaut	—
University of Virginia CAVALIER	Charlottesville, VA	R-123 50-396	0.1	LW Mod	—
CBS Corporation	Waltz Mill, PA	TR-2 50-22	—	—	—
NON-POWER REACTORS WITH POSSESSION ONLY AMENDMENTS					
Cornell University Zero Power Reactor	Ithaca, NY	R-89 50-97	0.1	Open Tank	—
General Electric Company	Sunol, CA	DPR-1 50-18	—	—	—
	Sunol, CA	DR-10 50-183	—	—	—
	Sunol, CA	TR-1 50-70	—	—	—
National Aeronautics and Space Administration (Plum Brook)	Sandusky, OH	TR-3 50-30	—	—	—
	Sandusky, OH	R-93 50-185	—	Mockup	—
State University of New York	Buffalo, NY	R-77 50-57	2000	PULSTAR	—
University of Virginia	Charlottesville, VA	R-66 50-62	2000	Pool	1960 06/27

* LW Mod is light water moderated, HW Mod is heavy water moderated, HW Refec is heavy water reflected
** Initial criticality - Year, month/day

2.4.1 Reference Non-Power Reactor

In the late 1970s and early 1980s, the NRC commissioned a series of studies on the technology and costs of decommissioning several types of nuclear facilities. A generic or reference design was selected for each facility studied. Two reference non-power reactors were presented in

Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors (PNL 1982).

10 CFR 50.2 defines a test reactor as a reactor with a thermal power level in excess of 10 MW, or a thermal power level in excess of 1 MW, if the reactor is to contain either a circulating loop through the core in which the applicant proposes to conduct fuel experiments, or a liquid fuel loading, or an experimental facility with the core in excess of 16 square inches in cross-section. 10 CFR 170.3 defines a research reactor to mean a reactor licensed for operation at a thermal power level of 10 MW or less, and which is not a testing facility. Since one reactor in Table 2-79 has a power level in excess of 10 MW, only the reference research reactor from PNL 1983 will be utilized in this study. The reference research reactor presented in PNL 1982 was based on Oregon State University's 1,100 kW TRIGA reactor.

Summary information on the structural materials including concrete, structural steel, and rebar used in the reference research reactor are provided in PNL 1982, Table B.2-1, and are presented here in Table 2-80.

Table 2-80. Estimated Quantities of Structural Material in the Reference Non-Power Reactor Facilities

Structure	Concrete (m^3)	Rebar (Mg)	Structural Steel (Mg)
Reactor Building	509	41.9	16.8
Reactor Structure	235	38.6	2.7
Radiation Center Annex	59	4.8	4.0
HX Addition	26	2.1	1.5
Pump House	8	0.6	0.2
TOTAL	837	88	25.2

Source: PNL 1982, Table B.2-1

As a point of comparison, the preconceptual design of a new research reactor was developed to meet the Department of Energy's missions of (1) producing medical and industrial radioisotopes, (2) producing Pu-238, and (3) supporting nuclear energy research and development. This design was presented in Appendix E of DOE/EIS-0310D (DOE 2000b). The resource requirements estimated for construction of a new 50 MW TRIGA reactor are given in Table 2-81. (Note, although DOE 2000 refers to this reactor as a "research" reactor, its 50 MW power level does not meet the 10 CFR170.3 definition of a "research" reactor).

Table 2-81. Non-Power Reactor Construction Resources

Construction Material	Quantity
Concrete volume	5,237 m^3
Mass of structural steel	50.122 Mg
Mass of stainless steel	3.468 Mg

Source: DOE 2000b

As expected from the high power level of the DOE 2000b research reactor, its material estimates are substantially greater than those of the PNL 1983 reference research reactor.

The U.S. Army Corp of Engineers prepared an Environmental Assessment for the decommissioning of the reactor facility at its Materials Technology Laboratory (MTL) in Watertown, Massachusetts. The MTL reactor was initially licensed by the NRC at 1 MW in 1960, and this level was increased to 5 MW in 1969. The MTL reactor was shut down in 1970 and placed on standby status. In 1971, the NRC approved a "possession-only" amendment to MTL's R-65 license. The Corps of Engineers provides a summary of the estimated amount of waste that would be generated during the complete dismantlement of the MTL reactor, which is reproduced as Table 2-82 (ACE 1991).

Table 2-82. Waste Volume Estimates for the Decommissioning of the Reactor Facility at the U.S. Army MTL

Material	Quantity			
	Contaminated		Clean	
Stainless Steel	3.7 m^3	130. ft^3	0. m^3	0. ft^3
Other Steel	5.7 m^3	200. ft^3	85.0 m^3	3,000. ft^3
Other Metals	4.2 m^3	150. ft^3	0. m^3	0. ft^3
Concrete	191.1 m^3	6,750. ft^3	2,208.7 m^3	78,000. ft^3
Other	0. m^3	0. ft^3	424.8 m^3	15,000. ft^3
TOTAL	204.7 m^3	7,230. ft^3	2,718.4 m^3	96,000. ft^3

Source: ACE 1991, Table A-3

The principal contaminant of stainless steel is Co-60 at 23 Ci; for other steel, the contaminant is Mn-54 at 0.15 mCi. The "Other" category of material includes galvanized sheet metal, motor control panels, control room equipment, metal piping, and substation transformers. The total weight of the 2,718 m^3 of clean material was estimated to be 4,540 Mg (5,000 tons) (ACE 1991).

The amount of radiologically contaminated material in the PNL 1983 reference research reactor is listed in Table 2-83. Table 2-84 gives a further breakdown of the 57.0 Mg of material identified in Table 2-83 as "Contaminated."

Table 2-83. Radioactive Materials in the Reference Non-Power Reactor

Radioactive Type	Mass (Mg)	Radioactivity (Ci)
Neutron Activated Metals (steel & aluminum)	1.6	1457
Activated Carbon	4.9	1.1
Contaminated Materials (see Table 2.4-6)	57.0	2
Radioactive Wastes	2.0	1

Source: PNL 1982, Table I.1-7

Table 2-84. Contaminated Materials in the Reference Non-Power Reactor

Location/Component		Estimated Mass (Mg)
Reactor Building	General Cleanup	10.2
	Beam Tube Caves	19.5
	Ion Exchanger Resins, Decontamination Eqpt	0.3
	Reactor Vessel	0.9
	Contaminated Concrete	10.9
	Reactor Building Equipment	4.3
	Piping, Drains & Sinks	2.9
Annex	Hot Cell	1.6
Heat Exchanger Building	Heat Exchanger	2.0
Pump House	Walls & Floor	0.05
	Retention Tank, Piping, and Equipment	3.3
Radiation Center Building	Piping & Equipment	1.0
TOTALS		56.95

Source: PNL 1982, Table I.1-8

The radionuclide composition of the neutron-activated material is given in Table 2-85 for the stainless steel, aluminum, and concrete components of the reference research reactor.

NRC 1994 provides additional information characterizing a reference research reactor. As with PNL 1982, NRC 1994 uses the Oregon State University (OSU) as its reference facility. In Table 7.1.1 of NRC 1994, it is estimated that 10 percent of the floor surface area and 2 percent of the wall surface area would be contaminated, and that contamination levels would be 102,000 dpm/ 100 cm^2 of Co-60 and 33,000 dpm/100 cm^2 of Cs-137.

Table 2-85. Radionuclide Inventory, Neutron Activated in the Reference Non-Power Reactor

Stainless Steel		Aluminum		Concrete	
Radionuclide	Concentration (Ci/m³)*	Radionuclide	Concentration (Ci/m³)*	Radionuclide	Concentration (Ci/m³)*
C-14	9.22e+00	Sc-46	9.60e-02	Ar-90	5.40e-04
Cr-51	1.27e+05	Mn-54	3.90e+00	Ca-41	9.80e-03
Mn-54	1.61e+04	Fe-55	2.77e+02	Ca-45	4.90e-02
Fe-55	2.52e+04	Co-60	1.36e-01	Mn-54	2.40e-03
Fe-59	2.41e+03	Ni-63	3.37e-02	Fe-55	4.20e-01
Co-58	1.03e+05	Zn-65	2.81e+02	Co-60	9.30e-03
Co-60	2.88e+05	TOTAL	5.62e+02	Ni-59	1.70e-05
Ni-59	5.59e+01			Ni-63	2.00e-03
Ni-63	6.40e+03			TOTAL	4.90e-01
Nb-93m	1.02e-02				
Nb-94	1.32e-01				
Nb-95	1.06e+01				
TOTAL	5.68e+05				

Source: PNL 1982, Tables E.1-5, E.1-6, and E.1-7
* Multiply by 3.7×10^{10} to convert from Ci/m³ to Bq/m³.

In ACE 1991, the Army Corp of Engineers reports the results of the radiological characterization of the shutdown MTL reactor; these results are shown in Table 2-86.

Table 2-86. Radiological Characterization of U.S. Army MTL Reactor Facility

General Location	Radiation Survey Location	mR/hr	Smear Sample Location	dpm/100 cm² Beta	dpm/100 cm² Alpha
Basement	Demineralizer	2 to 6	Inside Tubes of Storage Facility	293	<20
	Heat Exchangers	0.3			
	Fission-Product Monitor	0.05	All Other Smears	<200	<20
Main Floor	Californium-252 Source	16	All Main Floor Smears	<200	<20
	Mobile N-Ray	0.2			
First Platform	Reactor Keeper Side	0.2	All First Platform Smears	<200	<20
Second Platform	Reactor Top	0.7	All Second Platform Smears	<200	<20
	Magnets in Cabinets	0.4			
Reactor Vessel Internals	Blind Flanges	50 to 550	Floor by Access Ladder	204	<20
	Slant Tubes	8 to 30			
	Valves	10 to 60			
	Pedestal (Top)	550	All Other Reactor Vessel Internal Smears	<200	<20
	Pedestal (Bottom)	15			

Table 2-86. Radiological Characterization of U.S. Army MTL Reactor Facility (continued)

General Location	Radiation Survey Location	mR/hr	Smear Sample Location	dpm/100 cm²	
				Beta	Alpha
Reactor Annulus	Stainless Steel Racks & Pipe	55	Stainless Steel Racks	293 to 395	<20
	Below Reactor Grade	1,300	Reactor Annulus Floor	200 to 725	<20
	General Field by Stainless Steel Pipe at 3 ft	18	Stainless Steel Pipe Below Reactor Gate	749 to 5707	<20

Source: ACE 1991, Tables A-1 & A-2

2.4.2 Prorating to Other Sizes of Non-Power Reactors

As Table 2-79 shows, non-power reactors come in a wide range of designs and power levels. It is not practicable to attempt to uniquely estimate the amount of materials for each non-power licensee. Rather, a scaling factor will be developed here for estimating the reference non-power reactor material for any particular licensee.

In reports prepared for the DOE, Argonne National Laboratory (ANL) employed a scaling method based on the mass of PWR and BWR pressure vessels (Nieves et al. 1995). ANL assumes that all reactor metal inventories can be calculated from those at the corresponding reference plant based on the design power, as follows:

$$M = M_r \left(\frac{P}{P_r} \right)^{2/3} \qquad (1)$$

where:

	M	=	mass of material (e.g., carbon steel) in a reactor
	M_r	=	mass of material in the reference reactor
	P	=	power rating of a reactor
	P_r	=	power rating of the reference reactor

The quantity $\left(\frac{P}{P_r} \right)^{2/3}$ is referred to as the scaling factor. Since this factor was developed by ANL for large-scale commercial power reactors, it is unknown how well it applies to non-power reactors, which have a much larger, but substantially lower, power range, and very different design features.

Data from PNL 1983 were used to investigate the validity of the ANL scaling factor for NPRs. PNL 1983 presents the actual results of decommissioning five non-power reactors. Table 2-87 summarizes the results from PNL 1983 for the volume, mass, and activity level of the waste generated from each of the five decommissioning efforts.

Table 2-87. Non-Power Reactor Decommissioning Experience

Docket	Facility	Type	Power	m^3/Mg/Ci
NA	Diamond Ordnance Radiation Facility	Pool – TRIGA	250 kW	33/27.4/1.17E-04
NA	Ames Laboratory Research Reactor	Tank – MTR – D_2O	5 MW	1157/1224/6881
50-99	Lynchburg Pool Reactor	Pool – MTR	200 kW	20/14/<1
50-111	North Carolina State University Reactor-3	Pool – MTR	10 kW	10/1.5/unknown
50-106	Oregon State University*	Closed Vessel	0.1 w	<0.3/negligible/unknown

Source: PNL 1983
* Not to be confused with the 1.1 MW OSU TRIGA reactor in Table 2.4-1

The information regarding the mass of waste generated from each decommissioning effort is plotted against the power level of each reactor in Figure 2-15. A regression analysis was used to fit the power equation given below (with an R^2 of 0.97) to the data points:

$$W = 0.0835\, P^{1.0813} \tag{2}$$

where: W = waste generated (Mg)
P = reactor power level (kW)

This power relationship of equation 2 is also shown in Figure 2-15. Rearranging the above equation so that it relates the waste of one NPR to another (or reference NPR) gives:

$$W = W_r \left(\frac{P}{P_r} \right)^{1.0813} \tag{3}$$

If the assumption is made that the waste generated from the decommissioning of the NPRs listed in Table 2-87 is directly proportional to the mass of material used in the construction of the reactors, then mass can be substituted for waste in equation 3:

$$M = M_r \left(\frac{P}{P_r} \right)^{1.0813} \tag{4}$$

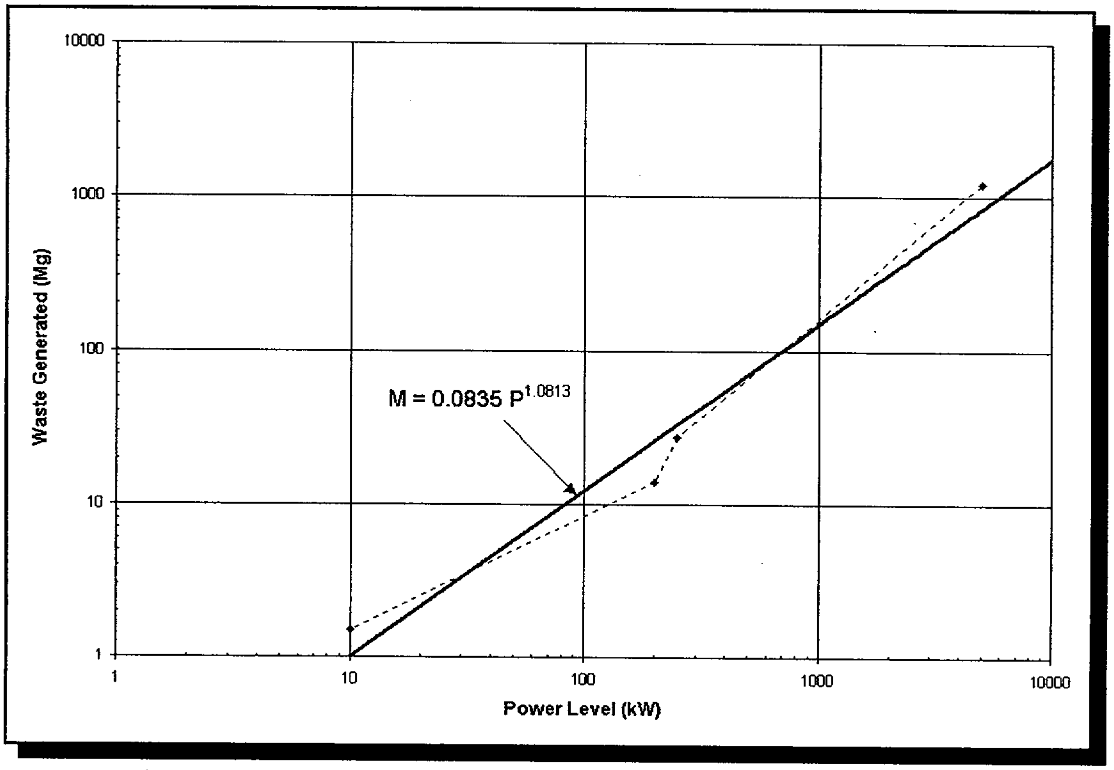


Figure 2-15: Mass of Decommissioning Radwaste Generated from Non-Power Reactors of Various Power Levels (PNL 1983)

This "scaling factor equation" is based on data obtained from NPRs, while the scaling factor from ANL discussed in equation 1 is based on commercial power reactors. An example of the use of equation 4 based on NPR data follows. Table 2-80 gives the structural steel content of the reference NPR as 25.2 Mg, and from Table 2-79 the power level of the reference NPR (i.e., the OSU TRIGA) is 1,100 kW. Therefore, using the above relationship, the mass of structural steel in the 250 kW Aerojet NPR is estimated to be 5.1 Mg. Equation 2 cannot be used for scaling; equation 4 must be used to scale from the reference NPR to other power levels.

2.4.3 Inventory Summary - Non-Power Reactors

Mass of Steel and Aluminum

Table 2-80 gives the mass of structural steel and rebar in the reference non-power reactor as 25.2 and 88 Mg, respectively. Table 2-83 shows that the total mass of activated steel and aluminum is 1.6 Mg, while Table 2-84 shows that the contaminated aluminum reactor vessel weighs 0.9 Mg and contaminated steel components weigh about 45 Mg.

Mass of Concrete

The mass of concrete in the reference non-power reactor is given in Table 2-80 as 837 m^3 or 1,925 Mg. Table 2-84 shows that there is about 11 Mg of contaminated concrete in the reference non-power reactor.

Mass of Copper

No studies were located during report preparation that describe the quantity or associated levels of radioactivity of copper contained in non-power reactors.

Scaling from Reference Facility

Section 2.4.2 describes the approach developed for scaling the reference NPR characteristics for other non-power reactors. This scaling approach is based on the quantities of waste generated during the actual decommissioning of four non-power reactors, ranging in size from 10 to 5000 kilowatts.

Contamination Levels and Principal Radionuclides

The calculated neutron activation levels in steel, aluminum, and concrete of the reference non-power reactor are 2.68×10^9, 7.70×10^6 and 7.55×10^3 Bq/g (5.68×10^5, 562 and 0.49 Ci/m^3 (Table 2-85), respectively. The principal radionuclides in activated steel are Co-60, Cr-51, and Co-58 at 50.7 percent, 22.4 percent and 18.1 percent, respectively. The principal radionuclides in activated aluminum are Zn-65 and Fe-55 at 50.0 percent and 49.3 percent, respectively. The principal radionuclides in activated reinforced concrete are Fe-55, Ca-45 and Co-60 at 85.7 percent, 10.0 percent and 1.2 percent, respectively. In addition, it is estimated that 10 percent of the floor area and 2 percent of the wall area of the reference non-power reactor would have surface contamination, and that the contamination levels would be 17 and 5.5 Bq/cm^2 (102,000 dpm/100 cm^2 and 33,000 dpm/100 cm^2) of Co-60 and Cs-137, respectively.

For perspective, the above values may be compared with mean mass dose factors and mean surface dose factors from Tables 2.1 and 2.2, respectively, in Draft NUREG-1640 (NRC 1999), as noted in Table 2-88:

Table 2-88. Limiting Mean Dose Factors for Individual Exposure

Nuclide	Mean Mass Dose Factor		Mean Surface Dose Factor	
	μSv/a per Bq/g	mrem/y per pCi/g	μSv/a per Bq/cm²	mrem/y per pCi/cm²
Co-60	250	0.92	190	0.70
Cs-137	--	--	200	0.74
Zn-65	210	0.78		
Co-58	95	0.35		
Cr-51	2.5	9.2×10^{-3}		
Ca-45	0.082	3.0×10^{-4}		
Fe-55	0.001	3.7×10^{-6}		

These are the highest dose factors from Draft NUREG-1640, Tables 2.1 and 2.2, regardless of material. it may be possible to lower these values by using material-specific dose factors.

2.5 Non-Fuel-Cycle Facilities

This section presents a review of the literature characterizing the quantities and the physical and radiological characteristics of equipment and material that may be candidates for clearance during routine operations and during decontamination and decommissioning in support of license termination of NRC and Agreement State non-fuel-cycle facilities. The objective of this section is to compile the information needed to support the evaluation of the costs and benefits of the seven alternatives for the different categories of non-fuel-cycle licensees.

Many simplifying assumptions were made in developing this characterization of potentially clearable material in each category of facility in the United States. The results are estimates of quantities of materials and inventories of radionuclides that, in the aggregate for some categories, could be perhaps 5 to 10 times higher or lower than the indicated values. In addition, the estimated values of types, quantities, and radiological characteristics of potentially clearable material may not be applicable to some individual facilities within a given category. Therefore, it may be more appropriate to refer to these characterizations as surrogates, which capture the potentially clearable material in a given category within an order of magnitude, rather than an accurate characterization of each facility within the category or the category as a whole.

An accurate characterization of each facility and each category would require a detailed investigation into each and every facility. This information likely exists in the file cabinets of the individual facilities. However, for the purposes of this investigation, a screening level characterization should suffice to gain insight into the potential impacts of the possible regulatory alternatives on each sector. As the analysis of the alternatives proceeds, and if it is determined that valid judgments cannot be made regarding the advantages and disadvantages of the alternatives without more precise characterizations, additional investigations may be needed.

2.5.1 Methodology and Organization

The methods used to accomplish the objectives of this section consisted of a seven-step process, which also corresponds to the major sections of this part of the report.

1. Development of a profile of potentially affected licensees: Starting with the NRC Information Digest (NRC 2000a), an overall count was obtained of the number of non-fuel-cycle licensees in the United States. An NRC relational database, which was developed for licensee tracking (as described in Appendix G, Program Codes Used in Materials and Fuel Licensing and Inspection Programs of *Consolidated Guidance about Materials Licenses*, NUREG-1556, Vol. 20, 2000) (NRC 2000g), was used to obtain additional information on non-fuel-cycle licensees. Using the NRC relational database, sealed source user licensees were excluded from further consideration, and the types and categories of non-fuel-cycle licensees requiring explicit consideration were identified.

2. Review of literature characterizing reference non-fuel-cycle licensee facilities: A review of the available literature identified the following two documents as directly relevant to the objectives of this section:

   - NUREG-1496, *Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities* (NRC 1994, NRC 1997)

   - NUREG/CR-1754, published in 1981, and its addendum, published in 1989, *Technology, Safety, and Costs of Decommissioning Reference Non-Fuel-Cycle Nuclear Facilities* (NRC 1981, NRC 1989)

   This material is summarized and then used as baseline information in the remainder of this section.

3. Characterization of a reference medical center: This section characterizes the types, quantities, and physical and radiological characteristics of candidate equipment and materials for clearance from the reference facilities selected to represent the majority of the materials licensees.

4. Characterization of a reference research and development laboratory.

5. Characterization of a reference nuclear pharmacy, including both manufacturers and regional and local distributors.

6. Characterization of manufacturers of source and radiolabeled compounds.

7. Evaluation of cumulative impacts of the alternatives: Using the information characterizing the reference facilities, along with appropriate scaling factors, the totality of the equipment and material that may be candidates for clearance at non-fuel-cycle facility licensees is characterized.

2.5.2 Profile of Potentially Affected Non-Fuel-Cycle Facility Licensees

The NRC Information Digest (NRC 2000a) indicates that there are a total of 20,800 medical, academic, and industrial materials licensees in the United States. Of these, about 5,288 are NRC licensees and 15,512 are administered by Agreement States. The NRC has developed a licensee tracking system, which assigns a five-digit program code number to each license to designate the major activity or principal use authorized in the license. Appendix G to NUREG-1556 (NRC 2000g) defines these codes. The NRC's licensee tracking system and associated program codes address only NRC licensees. Accordingly, information obtained from the NRC tracking system database addresses about one-third of the non-fuel-cycle licensees in the United States. The Agreement States have no requirement to use the same program codes.

A total of 114 program codes are assigned to facilities licensed for various activities and uses of byproduct, source, and special nuclear materials under Parts 30, 40, and 70, respectively, of Title 10 of the Code of Federal Regulations. Some of these program codes narrowly define an activity, such as radiography, while other program codes have a more extensive scope. More than one code may apply to a given license. However, the primary code indicates the licensee's principal use of material. Secondary codes may be used to indicate other significant uses. By the judicious selection of program codes, it is possible to use the tracking system, which operates on a relational database, to sort according to categories of facilities, such as hospitals, academic institutions, and specific industries, to obtain information about each category.

For each license, the database provides administrative details, such as the name and address of the facility, where the licensed material may be used, and the licensee's point of contact and/or Radiation Safety Officer. It also provides licensing data, including the NRC Program Code applicable to the license, the specific radionuclides and material forms covered by the license, and the possession limit for each radionuclide by material form.

The first step in the preparation of profiles of licensees potentially affected by the solid material clearance alternatives was to eliminate those licenses that authorize only possession and use of licensed materials in sealed sources or other non-disbursable forms, such as plated disks and foils. Termination of such licenses will normally entail no decontamination efforts; the licensed source will simply be removed from the facility and disposed of in accordance with NRC regulations. Appendix B presents an analysis of this category of licensee, along with the rationale for excluding it from explicit consideration in the evaluation of the costs and benefits of the clearance alternatives. Elimination of these licenses from the database resulted in a population of 2,997 non-reactor licenses that are potentially affected by the clearance alternatives.

The NRC profiles were augmented with additional data obtained from Dun and Bradstreet under another contract. These data include the SIC (Standard Industrial Classification) code for each facility, the number of employees, and annual revenues. Such data are useful for estimating the number of licensees that are small businesses and the relative sizes of the licensed facilities. Table 2-89 summarizes the categories of licenses.

Table 2-89. Categories of Non-Reactor NRC Licensees
(screened to eliminate sealed source licensees)

Category	NRC Program Code	Number
Research Facilities	1100 and 3600 series	566
Medical Facilities	2100 through 2500 series	1,923
Manufacturers and Distributors of Licensed Material and Services	3100 and 3200 series	212
Industrial Radiography and Irradiator Services	3300 and 3500 series	98
Miscellaneous Byproduct Licenses	3700 through 3900 or 6100 series	26
Source Material Licensees	11000 series	110
Uranium Fuel Fabrication and Critical Mass Assemblies	21000 series	17
Special Nuclear Material Licenses	22000, 23000, and 25000 series	45
Total		2,997

The following presents profiles of the major categories of facilities in the database.

Research Facilities: Licenses assigned Program Codes in the 1100 and 3600 series total 566, or almost 19 percent of all licensees. Of these, 60 are Academic Type A, B, or C Broad licenses and 112 are Research and Development Type A, B, or C Broad licenses. These licenses all allow possession of any radionuclide with an atomic weight of 83 or less in any form. Type A licenses, which allow possession of 1 curie for each radionuclide, account for 43 of the 60 Academic licenses and 86 of the 112 Research and Development licenses. Nine Academic and nineteen Research and Development licenses are Type B with possession limits of 0.1 curie, and there are eight Academic and seven Research and Development Type C licenses with possession limits of 0.001 curie. Finally, 394 Research & Development licenses are classified as other. These typically limit the radionuclides to be used and the quantities allowable to those expressly required by the licensees' specific research programs.

For the purpose of evaluating potentially clearable material, research facilities can be viewed as analogous to chemistry laboratories, with the typical equipment and fixtures that implies - laboratory benches, fume hoods and ductwork, and bench-scale processing equipment.

Medical Facilities: Licenses assigned Program Codes in the 2100 through 2500 series number 1,923, or slightly more than 64 percent of the total. Of these, 1,626 (85 percent) are for the diagnosis and/or treatment of humans using radiopharmaceuticals. Diagnostic and therapeutic medical licenses are held by 1,209 institutions and 447 private doctors. Also included in this classification are 46 licenses for teletherapy, 61 licenses for In-Vitro Testing laboratories, 52 nuclear pharmacies, 20 licenses for Veterinary services using radionuclides, and 7 licenses for the distribution of prepared radiopharmaceutical kits, sources, or devices.

While the database cannot distinguish quantitatively between hospitals with extensive nuclear medicine programs and those with limited programs, the distinction should be kept in mind when evaluating quantities and types of potentially clearable materials. Qualitatively, hospitals that are also teaching hospitals or research facilities will have far more extensive facilities than hospitals with programs limited to patient care. The facilities at a major medical center will be analogous to the research facilities identified above (as will the In-Vitro Testing Laboratories). The facilities with programs limited to patient care will be much smaller in terms of potentially contaminated areas and equipment. In fact, radiopharmaceutical use in these facilities will typically be limited to a central receiving, storage and dose preparation room, and specific and also limited patient dosing areas.

Manufacture and Distribution of Licensed Materials and Services: There are 212 licenses assigned to the 3100 and 3200 Program Code series, or about 7 percent of the total. Of these, 56 are licenses to manufacture and distribute licensed materials; 54 are licenses to provide well logging or other measuring system services; 11 licenses are for the distribution of exempt or general licensed materials; 10 are for waste disposal services; and the remaining 81 are for other services including nuclear laundry (3), decontamination (5), leak testing (7), and instrument calibration (14).

Of the diverse facilities in this category, the 56 facilities licensed to manufacture and distribute Licensed materials are the most important with respect to clearance. Included in this group are pharmaceutical and radio-labeled compound producers (e.g., Mallinkrodt and DuPont), and manufacturers of sealed sources, static eliminators, and industrial gauging systems.

Industrial Radiography and Irradiator Services: There are 98 licenses (just over 3 percent of the total) assigned 3300 series or 3500 series program codes. Of these, 76 licensees perform radiography at the customer's location, while 14 provide radiography services at their own facility. There are eight licensees who provide irradiator services.

Miscellaneous Byproduct Licenses: There are 26 licenses assigned Program Codes in the 3700 through 3900 or 6100 Program Code series. Representing less than 1 percent of the total, the activities covered include civil defense (2 licenses), possession only of byproduct material (17 licenses), decommissioning byproduct sites (6 licenses), and low-level waste storage (1 license). From the perspective of clearance, only the six decommissioning sites are of potential interest. The civil defense licenses are likely for calibration sources, the possession-only licenses do not permit activities that could result in contamination, and the waste storage license also does not permit any activities that would be expected to result in contamination.

Source Material Licensees: There 110 licenses (almost 4 percent of the total) assigned Program Codes in the 11000 series. Of these, 6 are for uranium mills, 1 is for solution mining, 1 is for uranium hexafluoride production, 6 are for rare earth extraction and processing facilities, and 1 is for decommissioning source material production facilities. There are 5 licenses for testing of military munitions (depleted uranium), 17 are for shielding (depleted uranium), 7 are for less than 150 kilograms of source material, 50 are for more than 150 kilograms of source material, and 6 allow only for the possession of source material.

For the purposes of assessing clearance issues, the uranium mills and uranium hexafluoride production plants are already included with the fuel-cycle facilities. The six Rare earth extraction and processing plants are assumed to be analogous to the uranium mills in terms of areas of contamination and processing equipment. The 17 licenses that permit use of source material for shielding and the 6 licenses that permit possession only can be excluded due to the low probability of residual contamination once the shielding or source material is removed. With the exception of the possible clearance of soil, the two outdoor munitions testing licenses can be dropped from further consideration. Finally, the 57 licenses that are simply for possession and use of source material are at this point undefined.

Uranium Fuel Fabrication and Critical Mass Assemblies: Seventeen licenses are assigned Program Codes in the 21000 series. Eight are for uranium fuel fabrication plants that are covered in the Fuel Cycle Category. The remaining nine are for critical mass assemblies. Seven of these are at universities, but no other information was obtained that characterizes these nine licenses at this time.

Special Nuclear Material (SNM) Licenses: Forty-five licenses (less than 2 percent of the total) are assigned Program Codes in the 22000, 23000, or 25000 Program code series. Six of these are for interim spent fuel storage, and are included in the Fuel Cycle Category. Twenty-two licenses are for plutonium in either neutron sources or sealed sources and can likely be eliminated from consideration due to the low probability of residual contamination. Fourteen licenses are for plutonium or uranium in quantities less than a critical mass. The final two licenses are for decommissioning of SNM facilities with less than a critical mass of SNM.

Based on this understanding of the universe of potentially affected non-reactor, non-sealed source NRC licensees, this section explicitly addresses four broad categories of licensees:

- Large medical centers (1,224 facilities)
- Research and development labs (566 facilities)

- Nuclear pharmacies, including both manufacturers and regional and local distributors (52 facilities)

- Manufacturers of source and radio-labeled compounds (63 facilities)

Appendix C presents a listing of the licensees under each category. The nature of the various categories results in some overlap. However, these categories and listings should capture the large majority of the non-fuel-cycle, NRC licensees that may be affected by the alternatives. Out of 2,997 non-sealed source material licensees in the NRC database, 2,005 non-fuel-cycle licensees were captured in this analysis. The major categories not captured include non-institutional medical users (i.e., doctor's offices and small medical providers, 447 licensees), 41 mobile nuclear medicine services, 111 high dose rate, remote after-loaders, 46 teletherapy units not screened by the sealed source screen, 61 in-vitro testing labs, 54 well logging and measuring system licenses not eliminated by the sealed source screening, 90 industrial radiographers not screened out by the sealed source screening, and 50 source material licenses. The licensees not captured are predominantly sealed-source licensees not eliminated by the initial screening and nuclear medicine providers that use short-lived radiopharmaceuticals.

2.5.3 Review of the Literature Characterizing Materials That May Be Candidates for Clearance During Routine Operations and During License Termination

A search of the available literature revealed two source documents, NUREG/CR-1754 (NRC 1981) and NUREG-1496 (NRC 1997), that provide information pertinent to characterizing reference non-fuel-cycle license facilities. The following section briefly summarizes the pertinent information.

2.5.3.1 NUREG/CR-1754, *Technology, Safety, and Costs of Decommissioning Reference Non-Fuel-Cycle Nuclear Facilities*

NUREG/CR-1754, published in 1981, and its addendum, published in 1989 (NRC 1989), are the primary sources of information pertinent to characterizing the types, quantities, and radiological characteristics of materials that may be cleared from non-fuel-cycle facilities, including medical, academic, and industrial facilities. The primary purpose of these studies was to provide information on the available technology, the safety considerations, and the possible costs of decommissioning those non-fuel-cycle facilities that represent a significant decommissioning task. In the course of addressing these issues, these reports also provide a great deal of information useful to the current investigation.

The approach used in NUREG/CR-1754 was to review the decommissioning experience of non-nuclear fuel cycle facilities that process radioactive material, manufacture radioactive sources, or use radioisotopes for medical or research applications. The facilities ranged in size from small rooms to large buildings, with activities ranging from the handling of millicurie amounts of radioactivity within the confines of hoods and glove boxes, to major installations where hot cells were used to process kilocurie amounts of radioactivity. Since the facilities included government, commercial, and institutional installations, these studies address the full range of facilities of interest to the current inquiry.

The reports used the information from decommissioning case studies to construct a set of reference laboratories designed to capture the full range of activities associated with non-fuel-cycle facilities. Tables 2-90 through 2-92, extracted from these reports, summarize the characteristics of the reference laboratories and the type, quantities, and radiological characteristics of the material and equipment generated during decommissioning.

In theory, these tables can be used to evaluate and characterize the types and amounts of material that may be candidates for clearance for a given category of non-fuel-cycle facility. This could be accomplished by estimating the total number of these reference labs in a reference facility of a given category, such as a hospital, and then multiplying by the number of such facilities in the United States.

Table 2-90. Characterization of Reference Laboratories

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Floor						
Quantity	$120m^2$	$80m^2$	$48m^2$	$48m^2$	$60m^2$	$80m^2$
Description	Asphalt tile over plywood	Asphalt tile over plywood	Concrete covered with asphalt tile	Concrete covered with asphalt tile	Concrete covered with linoleum	Concrete covered with asphalt tile
Contamination Level[a]	Generally less than $1x10^3$ d/m/100cm², with some spots as high as $5x10^4$ d/m/100cm² (usually at edges of benches) [p7-10]	Generally less than $1x10^3$ d/m/100cm², with some spots as high as $5x10^4$ d/m/100cm² [p7-13]	Generally less than 100 d/m/100cm², with some spots as high as $1x10^3$ d/m/100cm² [p7-16]	Generally less than $1x10^3$ d/m/100cm², with some spots as high as $5x10^4$ d/m/100cm² [p7-20]	Generally less than 20 d/m/100cm², with some spots as high as 1,000 d/m/100cm² [p7-23]	Generally less than 500 d/m/100cm², with some spots as high as $2x10^4$ d/m/100cm² at edges of benches [p7-33]
Walls						
Quantity	$132m^2$	$108m^2$	$84m^2$	$84m^2$	$168m^2$	$150m^2$
Description	Plasterboard painted with latex enamel paint	Plasterboard painted with latex enamel paint	Concrete sealed with epoxy paint	Concrete painted with latex enamel paint	Concrete sealed with acrylic paint	Plasterboard covered with latex enamel paint
Contamination Level[a]	Generally less than $1x10^3$ d/m/100cm², with some spots in the range of $1x10^4$ to $1x10^5$ d/m/100cm² [p7-10]	Generally less than $1x10^3$ d/m/100cm², with some spots in the range of $1x10^4$ to $1x10^5$ d/m/100cm² [p7-13]	Generally less than 100 d/m/100cm², with some spots as high as $1x10^3$ d/m/100cm² [p7-16]	Generally less than $1x10^3$ d/m/100cm², with some spots as high as $1x10^4$ d/m/100cm² [p7-20]	Less than 20 d/m/100cm² [p7-23]	Generally less than 500 d/m/100cm², with some spots as high as $1x10^4$ d/m/100cm² [p7-33]

Table 2-90. Characterization of Reference Laboratories (continued)

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Ceiling						
Quantity	120m^2	80m^2	48m^2	48m^2	60m^2	80m^2
Description	Acoustically treated fiberboard panels	Acoustically treated fiberboard panels	Concrete sealed with epoxy paint	Concrete painted with latex enamel paint	Concrete sealed with acrylic paint	Acoustically treated fiberboard panels
Contamination Level[a]	1x10^3 to 1x10^5 d/m/100cm^2	1x10^3 to 1x10^4 d/m/100cm^2	1x10^2 to 1x10^3 d/m/100cm^2	1x10^3 to 1x10^4 d/m/100cm^2	2x10^1 to 2x10^2 d/m/100cm^2	5x10^2 to 1x10^4 d/m/100cm^2
Fume Hoods						
Quantity	5	4	4	2	2	5
Description	Steel with interior surfaces of stainless steel	Steel with interior surfaces of stainless steel	Steel with interior surfaces of stainless steel	Steel with interior surfaces of stainless steel	Steel with interior surfaces of stainless steel	Steel with stainless steel and plastic laminate interiors
Contamination Level (inside)[a]	2x10^9 d/m/100cm^2, 1 mCi [p7-10]	Generally less than 1x10^5 d/m/100cm^2, with some spots in excess of 1x10^6 d/m/100cm^2 [p7-13]	Generally less than 1x10^3 d/m/100cm^2, with some spots as high as 1x10^5 d/m/100cm^2 [p7-16]	2x10^6 to 2x10^7 d/m/100cm^2, 1 to 10 mCi [p7-20]	Generally about 1x10^3 d/m/100cm^2, with some spots as high as 5x10^3 d/m/100cm^2 [7-23]	1x10^4 to 1x10^6 d/m/100cm^2

Table 2-90. Characterization of Reference Laboratories (continued)

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Glove Boxes						
Quantity	6	4	4	–	7	1
Description	Steel with acrylic plastic viewing panel	Steel with acrylic plastic viewing panel	Located inside fume hoods. Fabricated from acrylic plastic	–	Steel with acrylic plastic viewing panel	Steel with acrylic plastic viewing panel
Contamination Level (inside)[a]	$2x10^{10}$ to $2x10^{11}$ d/m/100cm^2, 10 to 100 mCi [p7-10]	Generally about $1x10^6$ d/m/100cm^2, with some spots in excess of this value [p7-13]	$2x10^6$ d/m/100cm^2, mCi levels [p7-16]	–	$2x10^7$ to $2x10^9$ d/m/100cm^2, 10mCi to 1 mCi [p7-23]	$1x10^6$ d/m/100cm^2
Hot Cells						
Quantity	–	–	–	2	–	–
Description	–	–	–	Constructed with interlocking lead bricks	–	–
Contamination Level (inside)[a]	–	–	–	$2x10^{10}$ to $2x10^{12}$ d/m/100cm^2, 10mCi to 1 Ci [p7-20]	–	–

Table 2-90. Characterization of Reference Laboratories (continued)

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Laboratory Workbenches						
Quantity	$15m^2$	$11m^2$	$6m^2$	$3m^2$	$1.5m^2$	$18m^2$
Description	Steel with plastic laminate tops	Wood with plastic laminate tops	Steel with stainless steel tops	Wood with plastic laminate tops	Steel with stainless steel tops	Wood with plastic laminate tops
Contamination Level[a]	Generally less than $1x10^3$ $d/m/100cm^2$, with some spots as high as $5x10^4$ $d/m/100cm^2$ [p7-10]	Generally less than $1x10^3$ $d/m/100cm^2$, with some spots as high as $5x10^4$ $d/m/100cm^2$ [p7-13]	Generally less than 100 $d/m/100cm^2$, with some spots as high as $1x10^3$ $d/m/100cm^2$ [p7-16]	Generally less than $1x10^3$ $d/m/100cm^2$, with some spots as high as $5x10^4$ $d/m/100cm^2$ [p7-20]	Generally less than 20 $d/m/100cm^2$, with some spots as high as 200 $d/m/100cm^2$ [p7-23]	Generally less than 500 $d/m/100cm^2$, with some spots as high as $1x10^4$ $d/m/100cm^2$ [p7-33]
Sink and Drain						
Quantity	–	1	1	1	–	3
Description	–	Stainless steel sink. 2m of 0.1m diameter pipe	Stainless steel sink. 5m of 0.1m diameter pipe	Stainless steel sink. 4m of 0.1m diameter pipe	–	Stainless steel sinks. 15m of 0.1m diameter pipe
Contamination Level[a]	–	$1x10^4$ to $5x10^4$	Less than 100 $d/m/100cm^2$ [p7-16]	Generally less than $1x10^3$ $d/m/100cm^2$, with some spots as high as $5x10^4$ $d/m/100cm^2$ [p7-20]	–	$1x10^4$ to $5x10^4$ $d/m/100cm^2$

Table 2-90. Characterization of Reference Laboratories (continued)

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Ventilation Ductwork						
Quantity	40m	30m	18m	23m	38m	32m
Description	Sheet metal	Sheet metal	Sheet metal	Sheet metal	Polyvinyl-chloride	Sheet metal
Contamination Level[a]	$5x10^4$ to $1x10^5$ d/m/100cm^2	Generally less than $1x10^5$ d/m/100cm^2, with some spots in excess of $1x10^6$ d/m/100cm^2 [p7-13]	200 to 2000 d/m/100cm^2	$1x10^3$ to $5x10^4$ d/m/100cm^2	Generally less than 50 d/m/100cm^2, with spots as high as 1000 d/m/100cm^2 [p7-23]	Generally less than $1x10^4$ d/m/100cm^2, with spots as high as $1x10^5$ d/m/100cm^2 [p7-33]
Lead Vault (inside) [p7-33]						
Quantity	–	–	–	–	–	1
Description	–	–	–	–	–	Interlocking lead bricks [p9-18]
Contamination Level[a]	–	–	–	–	–	Generally less than $2x10^4$ d/m/100cm^2, with spots as high as $2x10^5$ d/m/100cm^2 [p7-33]

Table 2-90. Characterization of Reference Laboratories (continued)

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Refrigerator (inside) [p9-8]						
Quantity	2	2	1	–	–	1
Description	–	–		–	–	–
Contamination Level[a]	$1x10^3$ d/m/100cm^2 [p9-8]	$1x10^3$ d/m/100cm^2 [p9-9]	$1x10^2$ d/m/100cm^2 [p9-11]	–	–	Generally less than $1x10^3$ d/m/100cm^2, with spots as high as $5x10^4$ d/m/100cm^2 [p7-33]
Freezer (inside) [p9-8]						
Quantity	1	1	–	–	–	–
Description	–	–	–	–	–	–
Contamination Level[a]	$1x10^3$ d/m/100cm^2 [p9-8]	$1x10^3$ d/m/100cm^2 [p9-9]	–	–	–	–

Table 2-90. Characterization of Reference Laboratories (continued)

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Storage Cabinet [p9-8]						
Quantity	2	2	1	–	–	–
Description	Wood painted with latex enamel	Wood painted with latex enamel	Steel with glass panels	–	–	–
Contamination Level[a]	1x10^3 d/m/100cm^2 [p9-8]	1x10^3 d/m/100cm^2 [p9-9]	1x10^2 d/m/100cm^2 [p9-11]	–	–	–
Shelves [p9-11]						
Quantity	–	–	4.5m^2	–	–	–
Description	–	–	Steel	–	–	–
Contamination Level[a]	–	–	1x10^2 d/m/100cm^2 [p9-11]	–	–	–
Transfer Tunnels [p9-15]						
Quantity	–	–	–	–	4m	–
Description	–	–	–	–	0.45m x 0.45m stainless steel	–
Contamination Level[a]	–	–	–	–	1x10^6 to 1x10^8 d/m/100cm^2 [p9-15]	–

Table 2-90. Characterization of Reference Laboratories (continued)

Facility Component	H-3 Laboratory[b]	C-14 Laboratory[b]	I-125 Laboratory[b]	Cs-137 Laboratory[c]	Am-241 Laboratory[c]	Institutional Laboratory[d,e]
Animal Cages (inside) [p7-33]						
Quantity	–	–	–	–	–	1
Description	–	–	–	–	–	Steel [p9-18]
Contamination Level[a]	–	–	–	–	–	Generally less than $1x10^3$ d/m/100cm^2, with spots as high as $5x10^4$ d/m/100cm^2 [p7-33]
Total Cost to Decommission[f]	$67,000 [p9-8]	$58,600 [p9-10]	$52,700 [p9-12]	$53,300 ($15,000 credit for lead salvage) [p9-14]	$73,900 [p9-16]	$62,900 [9-19]

Source: NRC 1981, Table 2.5-1, unless otherwise noted.

a Removable contamination.

b Contamination levels are based on experience at a large commercial laboratory for the manufacture of labeled compounds (New England Nuclear Corporation). [p7-10, 7-13, 7-16]

c Contamination levels are based on experience at a large commercial laboratory for the manufacture of sealed sources (New England Nuclear Corporation). [p7-20]

d Contamination levels are author's estimate. [p7-33]

e The facility provides work enclosures and equipment for the synthesis and use of organic compounds containing H-3, C-14, P-32, S-35, I-125, I-131, and other isotopes. [p7-28]

f Costs are in 1978 dollars. [p9-8]

Table 2-91. Decommissioning Details That Determine the Cost Estimates for the Reference Laboratories

Facility Component	H-3 Laboratory	C-14 Laboratory	I-125 Laboratory	Cs-137 Laboratory	Am-241 Laboratory	Institutional Laboratory
Floor	Decontaminated to unrestricted release levels (Tiles that cannot be easily decontaminated are removed and replaced with new tiles)[pF-2]	Decontaminated to unrestricted release levels (Tiles that cannot be easily decontaminated are removed and replaced with new tiles)[pF-5]	Decontaminated to unrestricted release levels (Tiles that cannot be easily decontaminated are removed and replaced with new tiles)[pF-9]	Decontaminated to unrestricted release levels (Tiles that cannot be easily decontaminated are removed and replaced with new tiles)[pF-13]	Decontaminated to unrestricted release levels [pF-16]	Decontaminated to unrestricted release levels (Tiles that cannot be easily decontaminated are removed and replaced with new tiles)[pF-20]
Walls	Decontaminated to unrestricted release levels [pF-2]	Decontaminated to unrestricted release levels [pF-5]	Decontaminated to unrestricted release levels [pF-9]	Decontaminated to unrestricted release levels [pF-12]	Decontaminated to unrestricted release levels [pF-16]	Decontaminated to unrestricted release levels [pF-19]
Ceiling	Panels packaged for disposal [pF-2]	Panels packaged for disposal [pF-5]	Decontaminated to unrestricted release levels [pF-9]	Decontaminated to unrestricted release levels [pF-12]	Decontaminated to unrestricted release levels [pF-16]	Panels packaged for disposal [pF-19]
Fume Hoods	3 decontaminated to unrestricted release levels; 2 cleaned to remove loose contamination and then packaged for disposal at a shallow land burial ground [pF-2]	3 decontaminated to unrestricted release levels; 1 cleaned to remove loose contamination and then packaged for disposal at shallow land burial ground [pF-5]	Decontaminated to unrestricted release levels [pF-9]	Decontaminated to unrestricted release levels [pF-12]	Those located in the low-level alpha lab are decontaminated to unrestricted release levels. [pF-15]	4 decontaminated to unrestricted release levels; 1 cleaned to remove loose contamination and then packaged for disposal at a shallow land burial ground [pF-18]

Table 2-91. Decommissioning Details That Determine the Cost Estimates
for the Reference Laboratories (continued)

Facility Component	H-3 Laboratory	C-14 Laboratory	I-125 Laboratory	Cs-137 Laboratory	Am-241 Laboratory	Institutional Laboratory
Glove Boxes	3 cleaned to unrestricted release levels; 3 packaged for disposal [pF-2]	3 cleaned to unrestricted release levels; 1 packaged for disposal [pF-5]	Packaged and shipped to a shallow land burial ground for disposal [pF-9]	–	Those located in the low-level alpha lab are decontaminated to unrestricted release levels. Those located in the high-level alpha lab are cleaned to remove loose contamination and then packaged and shipped to a shallow land burial site for disposal [pF-16]	Decontaminated to unrestricted release levels [pF-19]
Hot Cells	–	–	–	Disassembled and lead-glass windows and cell liners are packaged for disposal by shallow land burial. The lead bricks are monitored and 65% of the bricks are decontaminated and sold for salvage. The remaining bricks are packaged for disposal. [pF-12]	–	–

Table 2-91. Decommissioning Details That Determine the Cost Estimates for the Reference Laboratories (continued)

Facility Component	H-3 Laboratory	C-14 Laboratory	I-125 Laboratory	Cs-137 Laboratory	Am-241 Laboratory	Institutional Laboratory
Laboratory Workbenches	Cleaned to unrestricted release levels [pF-2]	Cleaned to unrestricted release levels [pF-2]	Cleaned to unrestricted release levels [pF-9]	Cleaned to unrestricted release levels [pF-12]	Cleaned to unrestricted release levels [pF-16]	Cleaned to unrestricted release levels [pF-19]
Sink and Drain	–	Sink cleaned to unrestricted release level; Drain line sectioned and packaged for disposal [pF-5]	Sink cleaned to unrestricted release level; Drain line sectioned and packaged for disposal [pF-9]	Sink cleaned to unrestricted release level; Drain line sectioned and packaged for disposal [pF-12]	–	Sink cleaned to unrestricted release level; Drain line packaged for disposal [pF-19]
Ventilation Ductwork (including filters)	Ductwork sectioned and packaged for disposal; Filters packaged for disposal [pF-2]	Filters packaged for disposal [pF-5] (Ductwork not indicated in report)	Ductwork sectioned and packaged for disposal; Filters packaged for disposal [pF-9]	Ductwork sectioned and packaged for disposal; Filters packaged for disposal [pF-12]	Ductwork sectioned and packaged for disposal; Filters packaged for disposal [pF-16]	Packaged for disposal. [pF-19]
Lead Vault	–	–	–	–	–	Cleaned to unrestricted release levels [pF-19]
Refrigerator	Cleaned to unrestricted release levels	Cleaned to unrestricted release levels [pF-2]	Cleaned to unrestricted release levels [pF-9]	–	–	Cleaned to unrestricted release levels [pF-19]
Freezer	Cleaned to unrestricted release levels	Cleaned to unrestricted release levels [pF-2]	–	–	–	–
Storage Cabinet	Cleaned to unrestricted release levels	Cleaned to unrestricted release levels [pF-2]	Cleaned to unrestricted release levels [pF-9]	–	–	–
Shelves	–	–	Cleaned to unrestricted release levels [pF-9]	–	–	–

Table 2-91. Decommissioning Details That Determine the Cost Estimates for the Reference Laboratories (continued)

Facility Component	H-3 Laboratory	C-14 Laboratory	I-125 Laboratory	Cs-137 Laboratory	Am-241 Laboratory	Institutional Laboratory
Transfer Tunnels	–	–	–	–	Those located in the high-level alpha lab are cleaned to remove loose contamination and then packaged and shipped to a shallow land burial site for disposal [pF-16]	–
Animal Cages	–	–	–	–	–	Packaged for disposal at a shallow land burial ground [pF-19]
Total Cost to Decommission[a]	$67,000 [p9-8]	$58,600 [p9-10]	$52,700 [p9-12]	$53,300 ($15,000 credit for lead salvage) [p9-14]	$73,900 [p9-16]	$62,900 [9-19]

Source: NRC 1981, Appendix F
a Costs are in 1978 dollars. [p9-8]

Table 2-92. Summary of Estimated Costs for Decommissioning Facility Components

Facility Component and DECON Option	Estimated Costs ($thousands)[a] to Decommission Components with Indicated Contaminant				
	H-3	C-14	I-125	Cs-137	Am-241
Fume Hood					
Decontamination	6.0	5.9	6.2	6.2	7.7
Packaging and Disposal w/o Volume Reduction	9.5	9.5	9.5	9.5	10.2
Packaging and Disposal w/ Supercompaction	6.5	6.5	6.5	6.5	7.1
Packaging and Disposal w/ Incineration	7.0	7.0	7.0	7.0	7.7
Glove Box					
Decontamination	4.4	4.1	4.5	–	5.7
Packaging and Disposal w/o Volume Reduction	4.0	4.0	4.0	–	4.5
Packaging and Disposal w/ Supercompaction	3.8	3.8	3.8	–	4.6
Packaging and Disposal w/ Incineration	4.0	4.0	4.0	–	4.7
Small Hot Cell					
Decontamination	–	–	–	8.6	–
Packaging and Disposal w/o Volume Reduction w/o Lead Salvage	–	–	–	10.1	–
Packaging and Disposal w/o Volume Reduction w/ Lead Salvage	–	–	–	12.0	–
Packaging and Disposal w/ Supercompaction w/ Lead Salvage	–	–	–	11.9	–
Packaging and Disposal w/ Incineration w/ Lead Salvage	–	–	–	12.3	–
Laboratory Workbench					
Decontamination	2.0	2.1	2.1	2.1	2.1
Packaging and Disposal w/o Volume Reduction	9.0	9.0	9.0	9.0	9.0
Packaging and Disposal w/ Supercompaction	4.7	4.7	4.7	4.7	4.7
Sink and Drain					
Decontamination	–	1.3	1.3	1.3	–
Packaging and Disposal w/o Volume Reduction	–	2.3	2.3	2.3	–
Packaging and Disposal w/ Supercompaction	–	1.9	1.9	1.9	–

Table 2-92. Summary of Estimated Costs for Decommissioning Facility Components (continued)

Facility Component and DECON Option	Estimated Costs ($thousands)[a] to Decommission Components with Indicated Contaminant				
	H-3	C-14	I-125	Cs-137	Am-241
Ventilation Ductwork					
Packaging and Disposal w/o Volume Reduction	11.8	11.8	11.8	11.8	12.3
Packaging and Disposal w/ Supercompaction	6.1	6.1	6.1	6.1	7.1
Packaging and Disposal w/ Incineration	6.9	6.9	6.9	6.9	7.9
Walls					
Decontamination	19.5	19.5	21.4	21.9	21.4
Floor					
Decontamination	8.8	8.8	8.8	8.8	8.5

Source: NRC 1989, p. 2.4

a Costs are in January 1988 dollars and include a 25 percent contingency.

2.5.3.2 NUREG-1496, *Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities,* Draft August 1994 (NRC 1994); Final 1997 (NRC 1997)

This GEIS was prepared in support of the NRC decommissioning rulemaking. (The rule has now been finalized and is set forth in Subpart E of 10 CFR 20 .) To enable the evaluation of regulatory alternatives, a cost/benefit analysis was performed that included the development of reference facilities representing the full range of NRC licensees. As such, the reference facilities cited in the GEIS represent an excellent starting point for developing reference facilities related to clearance of solid materials for recycling and reuse.

The reference facilities characterized in the final version of NUREG-1496 include a nuclear power plant, a uranium fuel fabrication plant, a sealed source manufacturer, and a rare metal extraction plant. In addition to these reference facilities, the draft version of NUREG-1496 included a reference broad research and development facility, a reference uranium mill, and a reference independent spent fuel storage installation. Of these, the sealed source manufacturer and the broad research and development facility are of interest to this section of the report. The data and information pertinent to these facilities are presented in Sections 2.5.5 and 2.5.6.

One of the conclusions presented in NUREG-1496 (NRC 1997, p. C.1-3) regarding reference non-fuel-cycle facilities is as follows:

> *Non-fuel-cycle material licensees include universities, medical institutions, radioactive source manufacturers, and companies that use radioisotopes for industrial purposes. Over 75 % of the NRC's materials licensees use either sealed radioactive sources or small amounts of short-lived radioactive materials. Decommissioning of these facilities should be relatively inexpensive and of short duration because there is usually little or no residual radioactive contamination to be cleaned up and disposed. Of the remaining 25 % of licensees, a small number (e.g., radioactive source manufacturers, radiopharmaceutical producers, and radioactive ore processors) conduct operations which could produce substantial radioactive contamination during the life of the facilities. The reference non-fuel-cycle material licensees analyzed in the EIS include a sealed source manufacturer and a rare metal extraction plant.*

NUREG-1496 provides information on the nature and extent of the contamination in structures undergoing license termination. However, since the current report is concerned with the clearance of materials and equipment for possible reuse, recycle, or disposal, it is concerned not only with structures, but also with the equipment and other materials within the structures. In addition, this report considers not only the disposition of materials at the time of license termination, but also clearance during routine operations. As such, this inquiry probes more deeply into additional subcategories of materials licensees in order to better understand the possible impacts of the alternatives on the different categories of licensees.

2.5.4 Large Medical Facilities (1,224 NRC licensees)

The investigation into NRC-licensed medical licensees was divided into three phases. In the first phase, the available and pertinent literature was reviewed. In the second phase, the information obtained from the review was validated and expanded upon by meetings and correspondence with Radiation Safety Officers (RSOs) at selected hospitals. In the third phase, information gathered in phases 1 and 2 was put into a form more useful for assessing the costs and benefits of the clearance alternatives.

2.5.4.1 Literature Review and Interviews

"Lessons Learned in Decommissioning Medical Facilities"

This paper, prepared by Victor Evdokimoff of Boston University Medical Center (BUMC) and published in *The Radiation Protection Journal*, Vol. 77, No. 5 Supplement, November 1999, describes the decontamination of two medical facilities.

The first decontamination operation consisted of renovating seven floors of a 10-story hospital in 1994, where all existing structures and equipment were removed with only the exterior of the building remaining. Since this building had been used for 50 years, the cleanup activities involved contamination accumulated over a long period of time. The second decommissioning project involved the demolition of three buildings in 1997.

The 1994 decommissioning was performed in accordance with NRC guidance provided in NUREG/CR-5849, *Manual for Conducting Radiological Surveys in Support of License Termination* (NRC 1994). Since many of the MARSSIM concepts were first introduced in this guidance, the manual can be considered a precursor to MARSSIM. Unlike MARSSIM, the guidelines also incorporate specific cleanup criteria, as presented in Table 2-93. These levels are identical to those set forth in Regulatory Guide 1.86.

Table 2-93. Acceptable Surface Contamination Levels Based on NUREG/CR-5849

Nuclides[a]	Average[b,c,f] (dpm/100 cm²)	Maximum[b,c,d,f] (dpm/100 cm²)	Removable[b,c,e,f] (dpm/100 cm²)
U-nat, U-235, U-238, and associated decay products	5,000 α	15,000 α	1,000 α
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-133, I-131	1,000	3,000	200
Beta/gamma emitters (nuclide with decay modes other than alpha emissions or spontaneous fission) except Sr-90 and other noted above	5,000 βγ	15,000 βγ	1,000 βγ

a Where surface contamination by both alpha- and beta/gamma-emitting nuclides exists, the limits established for alpha- and beta/gamma-emitting nuclides should apply independently.

b As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrument.

c Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

d The maximum contamination level applies to an area not more than 100 cm^2.

e The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, then pertinent levels should be reduced proportionately and the entire surface should be wiped.

f The average and maximum radiation levels associated with surface contamination resulting from beta/gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

In accordance with these guidelines, the seven floors of the building (about 60,000 square feet) were divided into affected and unaffected areas. An area was defined as either a room in a lab, a room with many labs, or even a piece of equipment within a room, such as a hood. A total of 291 affected areas were identified. Within these areas, 70 contaminated spots were found in 32 hoods, 70 were found on 28 bench tops, 66 were found on the floors of 23 rooms, and 31 were found in cabinets in nine rooms. The 290 unaffected areas identified were found to contain two contaminated spots. Out of 4,114 dry wipe tests taken, only 12 swipes indicated removable contamination above the release limits, and 5 of these were contaminated with tritium.

It was determined that very little of the contamination was removable. The average level of contamination was 30,000 dpm/100 cm^2 primarily from tritium and C-14. The authors concluded that about 14 percent of the total surface area of the floors was contaminated, and only 0.3 percent of the contaminated areas contained removable contamination.

The 1997 decommissioning was performed in accordance with MARSSIM guidance. A scoping survey was performed which surveyed 10 percent of the affected areas and 1 percent of the unaffected areas. The scoping survey found no contamination above background in Building 1, which was a four-story patient care facility abandoned 15 years prior to the survey. Building 2 was a four-story research building also abandoned 15 years prior to the survey. Out of 30 areas designated as affected, 4 contaminated spots were found. These consisted of H-3 and C-14 on some animal cages, inside freezers, and on some counter tops. Since the scoping survey found some contamination, a 100-percent survey of Building 2 was performed, but no additional contamination was found. There was no contamination detected in the areas of Building 2 designated as unaffected.

Building 3 was a six-story research building in use for 40 years and vacated several months prior to the survey. In the 21 areas designated as affected, seven contaminated spots were found on counter tops. As a result, a 100-percent survey was performed which uncovered no additional contamination.

In the areas designated as unaffected, the scoping survey found Ra-226 contamination on the floor and under the floor tiles in a corridor. The contamination resulted from a 70 mCi spill and contaminated a 40 square foot area of concrete under the floor tiles. The contamination was removed with a jackhammer. As a result of this finding, a 100-percent survey was performed, but no additional contamination was found.

The overall cost of the 1994 program was several hundred thousand dollars, while the overall cost of the 1997 program was tens of thousands of dollars. The primary difference in the costs was the result of MARSSIM protocols that reduced the number of samples required for analysis and the extent of the surveys.

Interview with Victor Evdokimoff

In addition to his paper, Mr. Evdokimoiff supplied additional information in telephone interviews characterize more fully the potential impacts of the alternatives on medical facilities. Mr. Evdokimoff also answered several questions in writing.

In the 1994 decontamination, the total floor area of the 7 floors was 60,000 square feet, which included an average of about 30 labs per floor, of which about half used radioactive materials. During the decontamination operation of these labs, 15 cubic feet of incinerator ash (containing 10 μCi of H-3/C-14), and 13.6 cubic feet of miscellaneous waste (containing 300 μCi of H-3/C-14) were disposed of as low-level waste. In addition, 30 linear feet of plumbing and drains contaminated with S-35 were placed in storage for radioactive decay.

In the 1997 decommissioning, 22.5 cubic feet of scabbled concrete (containing 100 μCi of Ra-226) and 30 cubic feet of miscellaneous metal and other wastes (containing 5.6 mCi of H-3/C-14) were disposed of as low-level radioactive waste.

Mr. Evdokimoff provided the following written comments:

1. *Given sufficient time to allow for decay of I-125, P-32 and S-35, most contaminants are weak beta emitters H-3, C-14.*

2. *Most contamination is fixed on floors, benchtops, hoods, cabinets and hot sinks. The contamination was easily removed from the surfaces saving considerably on radwaste costs. Wipe testing for removable contamination is a waste of time and money.*

3. *Since H-3 is a major contaminant, institutions that only do wipe tests and count them in a LSC will miss fixed H-3 contamination. Most medical research facilities do not have a windowless Proportional counter which is able to detect H-3 on surfaces.*

4. *I-125 release limits are exceedingly strict. Most institutions will not be able to meet the detection limit for free release of 20 dpm/100 cm sq removable.*

5. *Older medical facilities have labs that are separated from one another by doors. In my opinion this minimizes the spread of radioactive contamination. Today's medical facilities have open bench designs in which a floor could contain 30-50 benchtops one after another. This could result in declaring the whole floor an impacted area since many of these benchtops are radioactive use areas.*

6. *Most medical research institutions have competent and vigilant health physics staff that help keep contamination from researchers to a minimum. In my experience in reviewing thousands of lab surveys over 25 years at*

Boston University Medical Center (BUMC) and as a consultant, radioactive contamination in labs is very infrequent. In addition, if there is contamination, the micro to millicurie use levels contribute to low levels of contamination

7. *I agree with many of the observations provided by my colleague Ken Miller at HMC (see Appendix C). Our institution is two to three times larger than HMC. I believe my decommissioning findings can be generalized to other medical research facilities. My findings are that most areas and equipment are not contaminated. What little contamination exists is fixed from weak beta emitters.*

8. *Medical research facilities do not have the financial resources to pay for decommissionings that turn up very little contamination prior to release. We paid $300,000 for a contractor with 3 people working 7 months in the 1994 decommissioning alone.*

9. *Hospitals that utilize nuclear medicine should not be confused with medical research facilities at a university, hospital or biotech firm. Medical research utilizes "CHIPS". The most commonly used radionuclides today are P-32 and S-35. There is a declining use of H-3, C-14 and 1-125. Iodinations have declined and 1-125 RIA kits are almost never used.*

10. *In our experience, 30% of the total radioactive use labs are turned over in a year. The researcher moves to another building. Our researchers seem to be moving around the institution constantly. Some leave and new Principle Investigators come. Almost all researchers use some radionuclides. Their grants depend on the use of radioactive materials. If they could not use radionuclides, they could not do research. Money is very tight for researchers who have to provide their own funding to come AND REMAIN at a medical research facility. If they lose a grant, they usually have to leave.*

Mr. Evdokimoff also commented on the clearance alternatives, as follows:

1. *1 mrem/yr, 0.1 mrem/yr, and zero above background. These dose limits are not justified. The LLD to detect these levels cannot be obtained. This effectively prevents release of materials that are probably not contaminated. If we cannot survey, then these materials would have to go out as radioactive waste. No one could afford this. Actions of this sort would cause researchers not to use radioactive tracers even those that are decayable because labs*

cannot be used for certain lengths of time. [S-35, a common radionuclide, has a half-life of 90 days.] Grants would be affected and ultimately medical research.

2. *Prohibition of release is not justified because of low risk, low contamination potential.*

The following are Mr. Evdokimoff's comments on NUREG /CR-1754 (the study of reference non-fuel-cycle facilities discussed in Section 2.5.3.1):

1. *This guide is too old to be used at all for generic medical research facilities.*

2. *H-3 and C-14 are usually used on the same benchtop. No mention of P-32 and S35 use. No research facility uses unsealed Am-241 or Cs-137.*

3. *The contamination levels seem high. Most surveys yield negative results. Walls and ceilings were not found to be contaminated in our decommissionings.*

4. *Medical facilities do not use glove boxes or hot cells.*

5. *Hot sinks, benchtops, floors cabinets, hoods and animal cages do get contaminated. Iodination hoods and its ductwork can be contaminated with I-125. Plumbing and hot sink traps also get contaminated.*

Site Visit

In order to validate and supplement the information summarized in the literature review, Hershey Medical Center in Harrisburg, Pennsylvania, was visited. Appendix D summarizes the results of that visit. This summary was reviewed and approved by Ken Miller, RSO of Hershey Medical Center.

Subsequent to the visit, additional questions arose that helped to clarify some of the material provided in Appendix D. The first question dealt with lower limits of detection. Mr. Miller explained that the medical center employs 100 dpm/100 cm^2 as the definition of "clean," and this is determined by counting wipe samples that can achieve the lower limits of detection necessary to demonstrate that a given surface meets tthe "clean" criteria. Surface scans using handheld survey instruments can observe contamination at the limits set forth in Regulatory Guide 1.86.

The second question dealt with the volume and characteristics of low-level radioactive waste generated during the process of clearing a room. Mr. Miller explained that room clearance generates hardly any waste because the rooms are kept clean, i.e., <100 dpm/100 cm^2 as normal policy. In general, clearing a room usually generates less than one cubic foot of low-level waste. The amount of contamination in the waste is small as is evidenced by the fact that the total quantity of H-3 shipped in LLW each year from the entire facility is about 100 mCi. However, if a room cannot be declared as clean because of widespread contamination, it is possible that the entire room would have to be gutted, including tables, cabinetry, etc., producing at least 100 cubic feet of LLW.

2.5.4.2 Integration

The information compiled from the literature review and interaction with RSOs establishes a basis for assessing the potential impacts of the seven clearance alternatives on large medical facilities. The first step in the assessment was to estimate the total amount of material, equipment, furniture, etc., that is "at play" at a typical large research and university hospital, in this case the Hershey Medical Center. The term "at play" refers to the totality of material that is under regulatory control at any given time and that may be cleared per year. The quantity of material at play was determined by estimating the number of rooms under regulatory control at any given time and the number cleared per year, and the inventory of material in the rooms, as characterized in Table 2-90. In the next step, an estimate was made of the inventory of radionuclides in the contaminated material, equipment, furniture, etc., in the rooms based on a combination of the information provided in Table 2-90 and Appendix D.

This inventory thus represents the types, quantities, and characteristics of material, equipment, furniture, etc., including radionuclide composition, that would require license disposal at a low-level radioactive waste disposal facility if the clearance criteria prohibited clearance, or were set so stringently that the RSO could not use feasible and practical methods to determine compliance with the clearance criteria.

The next step in the process was to determine the quantity and characteristics of the material, equipment, furniture, etc., that would be sent to licensed disposal each year or upon license termination if the criteria were set at levels that could be detected.

This approach provides the technical basis for assessing the costs, benefits, and feasibility of the alternative clearance criteria.

Material Inventories

The preceding review revealed that there are about 250 rooms under regulatory control at Hershey Medical Center at any particular time, and that 6 to 12 of these are cleared each year. In contrast to this, BUMC experiences an approximate 30 percent per year turnover rate of its labs. Based on Appendix D and Table 2-90, the contents of a typical room may be described as follows:

- Floor area 50 to 100 m^2, 1/4 inch asphalt tile over plywood or concrete (~0.5 m^3; 1,000 lb)
- Wall area 50 to 80 m^2, 1/2 inch plaster board (~1 m^3; 2,000 lb)
- Ceiling area 50 to 100 m^2, 1/2 inch plaster board (~0.5 m^3; 1,000 lb)
- 1 fume hood 38 (1/2" L, 27 1/2" D, 55 1/4" T, ~425 lb)
- 1 glove box (27"L, 13 1/2"W, 24 1/2" T, ~40 lb)
- 10 m^2 wooden, steel, or plastic laminated workbench (assumed 100 lb)
- 1 sink and drain (assumed 50 lb)
- 20 m of sheet metal ductwork (assumed 100 lb)
- 1 refrigerator (ISCO ~175 lb)
- 1 storage cabinet (Polystyrene 14"w, 14.5"h, 14.5"d, ~35 lb)

Based on information provided at www.labx.com, a used lab equipment web site, the following are the physical characteristics of typical equipment and furniture of this type:

- Fume Hoods:
  - Steriguard ~1000 lb.
  - Bio-guard 38 1/2" L, 27 1/2" D, 55 1/4" T, ~425 lb.
  - NuAire LabGuard 78"W, 60"H, 30"D, ~700lb.
  - Benchtop 27"L, 13 1/2"W, 24 1/2" T, ~40 lb.

- Glove Boxes:
  - Forma Scientific 42"W, 30"D, 30"H, ~1,000 lb.
  - Forma Anaerobic 43", 30", 31" D, ~400lb.

- Refrigerators:
  - ISCO ~175 lb.
  - Stainless Steel 40ft^3, 7' tall, 6' wide, 31" D, ~900lb.

- Freezers: Ultra Low 18"D, 23.5"W, ~250 lb.

- Freezer/Refrig: Sanyo 1.4 ft^3 freezer, 6.2 ft^3 refrig, ~415 lb.

- Storage Cabinet: Polystyrene 14"w, 14.5"h, 14.5"d, ~35 lb.

The inventory of materials in the reference room is about 5,000 pounds. To obtain the total inventory at a reference large medical center, which contains 250 rooms under regulatory control, simply multiply the room inventory by 250, yielding about 600 tons of potentially clearable material in the reference medical facility at any given point in time. To obtain the total inventory in the United States., this value is multiplied by the number of reference hospitals licensed in the country. Based on the NRC tracking system database in combination with the Dun and Bradstreet SIC code information, the number of personnel employed by Hershey Medical Center is 5100, and the total number of employees in the U.S. hospitals licensed by the NRC is about one million (see Appendix C). Thus, the total amount of material "at play" in U.S. hospitals licensed by the NRC can be crudely estimated by multiplying the quantity of material in the reference large hospital by 200[5]. Finally, since the total number of NRC licensees is about 1/4th the total number of NRC plus Agreement State licensees, this number is multiplied by three to obtain an overall estimate of the amount of material at play in the United States; i.e., about 480,000 tons.

In any given year, it can be assumed that about 2 to 5 percent (i.e., 6 to 12 rooms out of 250 rooms) of this material may be subject to clearance, or about 9,600 to 24,000 tons per year. Alternatively, this amount could be about 10-fold higher if the clearance rate is 30 percent per year, as is the experience at BUMC. This is an estimate of the total quantity of material that may require licensed disposal from hospitals <u>if the prohibition option were implemented</u>. Of course,

[5] Other metrics may be more suitable for scaling the reference facility to the universe of medical facilities. For example, the amount of R&D money spent at each hospital might be used. This issue requires further investigation.

only a very small fraction of this material would actually require licensed disposal because the vast majority of this material is clean. For example, Hershey Medical Center ships about ten 55-gallon drums (74 cubic feet) per year for licensed disposal. This includes the waste from routine operations plus the waste from clearing rooms.

Fixed Radionuclide Contamination Inventories in Rooms

The next step in the analysis consisted of determining the radionuclide inventory at play. Based on information obtained from Hershey Medical Center (see Appendix D), this material is expected to be clean except for some very localized areas contaminated with difficult to remove C-14 or H-3. According to the fixed contamination report provided by Hershey Medical Center, the current inventory of fixed surface contamination that would be disposed of as low-level radioactive waste is several square feet of floor area contaminated with S-35 (it is assumed that material containing S-35 would be held for decay and not contribute to LLW disposal volumes) at 3,000 dpm per 100 cm^2, several square feet of equipment contaminated with C-14 at 25,000 dpm/100 cm^2, several square feet in a freezer contaminated with H-3 at 2,000 dpm/100 cm^2, and several square feet of table tops contaminated with C-14 at about 1,000 dpm/100 cm^2. This converts to the following approximate inventory of fixed radionuclide contamination in the hospital at any given time that will contribute to low-level waste. The amount of LLW is independent of the regulatory alternatives under consideration.

Table 2-94. Estimated Fixed Contamination Profile for Hospital Room

Item	Nominal Area (ft^2)	Tritium Contamination (dpm/100 cm^2)	C-14 Contamination (dpm/100 cm^2)	Total (mCi H-3)	Total mCi C-14)
Equipment	10		25,000	0	0.01045
Freezer	10	2,000		0.000836	0
Table tops	10		1,000	0	0.000418
Floor	10			0	0
Total	40			0.000836	0.010868

Based on this review, it can be concluded that only those alternatives that would either prohibit clearance of material from regulatory control, or establish clearance levels not readily implementable using current monitoring protocols and equipment would have an impact on this segment of the NRC-licensed community.

2.5.5 Research and Development Laboratories (566 NRC-licensed facilities)

In many respects, research and development (R&D) labs are similar to the research labs at university hospitals, such as Hershey Medical Center and Boston University Medical Center described above. In addition, Table 2-90 was designed primarily to capture the nature and extent of contamination at R&D facilities. This section summarizes the information on R&D facilities contained in the draft of NUREG-1496. This information, along with Table 2-90, is used to construct a reference R&D facility. The characterization of the reference R&D lab is then scaled up to the United States as a whole using the list of facilities presented in Appendix C.

Draft NUREG-1496 characterized this segment of non-fuel-cycle licensees as follows:

> *Research and development facilities using radioactive materials cover a broad range of activities including the use of laboratories or health treatment facilities that use radioisotopes. Both short-lived (^3H) and long-lived isotopes (^{14}C) may be used. The reference facility includes rooms for synthesis of labeled compounds and for preparing radioactive samples, a laboratory, a counting room, and a storage room. Only long-lived nuclides are included in this analysis. Contaminated facilities associated with the reference broad research and development facility include:*
>
> - *glove boxes, fume hoods, sinks, workbenches;*
> - *laboratory floor and wall areas;*
> - *a storage area.*
>
> *A generic single building facility is used in the analyses for a reference broad R&D facility, because such facilities vary widely in size. However, for an R&D facility comprised of several buildings, the waste volumes, costs, etc., can be reasonably approximated by multiplying the results for a single building by the number of buildings in the facility.*
>
> *The floor area of the facility is estimated to be approximately 6,000 ft^2. Approximately 10% of this area is estimated to be contaminated with ^{137}Cs and ^{60}Co. The crack length is estimated to be approximately 320 ft. The wall area is estimated to be approximately 4,600 ft^2. Approximately 5% of this area is estimated to be contaminated.*

Table 7.1.1 of Appendix C of Draft NUREG-1496 states that, for the reference broad R&D facility, the total floor area at the facility is 6000 ft^2 of which 10 percent is contaminated, and the total wall area of the facility is 4,600 ft^2, of which 5 percent is contaminated. In addition, the level of contamination is stated to be 102,000 dpm/100 cm^2 for Co-60 and 33,300 dpm/100 cm^2

for Cs-137. An analysis of the depth profile of the contamination in concrete, based on data compiled from Indian Point Unit 1 (see Figure 4.8.3 of Appendix C of Draft NUREG-1496), shows that the profile declines exponentially with depth, and about 90 percent of the activity is in the top 5 mm (about 1/4 inch), but some contamination extends down to about 30 mm (a little over one inch).

The implication is that a total of about 80 m^2 of floors and walls is contaminated. In addition, assuming the contamination extends down to about 1/2 inch, the volume of contaminated material (assumed to be about 2 g/cm^3 in situ) is about 1 m^3. When scabbled and compressed, this material may double in volume to an overall density of 1 g/cm^3, and weigh $2x10^6$ g or about 5,000 lb.

The total inventory of Co-60 and Cs-137 in the scabbled concrete is about $1.3x10^7$ Bq (0.35 mCi) of Co-60 and $4.2x10^6$ Bq (0.11 mCi) of Cs-137. The average Co-60 and Cs-137 concentrations in the scabbled material are 6.5 Bq/g and 2.1 Bq/g, respectively (176 and 57 pCi/g, respectively).

In addition to the contaminated structural material, the reference R&D facility also contains the types of furniture and equipment listed in Table 2-90. It is assumed that the inventory of such equipment and material is proportional to the floor area of the facility. The reference Cs-137 lab in Table 2-90 has a floor area of 48 m^2, while this reference R&D lab has a floor area of 6000 ft^2 (557 m^2). The implication is that the reference R&D lab has about 10 rooms. Hence, the inventory of equipment in the facility is assumed to be as follows[6]:

- Hot Cells - 20 contaminated at $2x10^{10}$ to $2x10^{12}$ dpm/100cm^2
- Fume Hoods - 20 made of steel contaminated at $2x10^6$ to $2x10^7$ dpm/100cm^2. Assumed to weigh 1,000 lb each.
- Work Benches - 30 m^2 contaminated at 1,000 dpm/100cm^2, with hot spots up to 50,000 dpm/100cm^2. Assumed to weigh 300 lb. (About $3x10^5$ Bq, assuming 10 percent is contaminated at the high end.)
- Sinks and Drain - 10 stainless steel sinks, and 10 sets of 4 m by 0.1 m diameter pipe contaminated at 1,000 dpm/100cm^2, with hot spots up to 50,000

[6] The mass estimates presented here are based on the information presented previously under **Materials Inventories** or on engineering judgment.

dpm/100cm^2. Assumed to weigh 50 lb each. (About 4,000 Bq, assuming 10 percent is contaminated at the high end.)

- Ventilation Ductwork - 230 m of sheet metal contaminated at 1,000 dpm/100cm^2, with hot spots up to 50,000 dpm/100cm^2. Assumed to be 1,000 lb. Assuming each meter of ductwork flattens out to 1 meter wide, there is about 230 m^2 of ductwork containing about 2.3x10^6 Bq of contamination (assuming 10 percent is contaminated at the high end).

Not including the hot cells and fume hoods (because they are contaminated at levels too high to be considered for clearance), the total weight of equipment, etc., in the facility that may be subject to clearance is about 2,000 lb.

Extrapolation

Extrapolation of the reference R&D lab to all 566 NRC-licensed R&D labs in the United States is difficult because the reference lab described above applies to a 6,000 ft^2 facility that handles Cs-137 and Co-60. Many R&D labs also use C-14, H-3, and other radionuclides, as indicated in Table 2-90. The overall size (number of rooms) of these facilities has not yet been determined. In addition, the distribution of R&D labs (rooms) that use the various isotopes identified in Table 2-90 has not yet been determined. Inspection of Table 2-90 reveals differences among the different labs with respect to the size of the rooms (the sizes range from 48 m^2 to 120 m^2), the types of equipment (e.g., hot cells, hoods, freezers, refrigerators, etc.), and the levels of contamination (100 dpm/100 cm^2 for I-125 labs to 1,000 dpm/100 cm^2 for H-3, C-14, and Cs-137 labs).

Notwithstanding the incomplete characterization of this sector, and until more complete information is obtained, an equal number of H-3, C-14, Cs-137, and Co-60 labs among the 566 NRC-licensed R&D facilities is assumed.

2.5.6 Nuclear Pharmacies (including both manufacturers and regional and local distributors) (52 facilities)

The NRC issues commercial nuclear pharmacy licenses pursuant to 10 CFR Part 30 and 10 CFR 32.72, for the possession and use of radioactive materials for the manufacture, preparation, or transfer for commercial distribution of radiopharmaceuticals (radioactive drugs) containing byproduct material for medical use under Part 35. Radiopharmaceuticals produced from NORM or accelerator-produced radionuclides are not within the regulatory authority of the NRC, although they may be subject to state licensing requirements. Preparation includes the making of radiopharmaceuticals from reagent kits and from raw materials. Typically, nuclear pharmacies are also authorized to transfer for commercial distribution (per 10 CFR 31.11) *in vitro* test kits, radiopharmaceuticals to licensees authorized to possess them for other than human medical use (i.e., veterinary medicine and research licensees), and radiochemicals to those licensees authorized to possess them, pursuant to 10 CFR Part 30. Additionally, nuclear pharmacies are authorized to redistribute (transfer) sealed sources for calibration and medical use initially distributed by a manufacturer licensed pursuant to 10 CFR 32.74.

The NRC database identifies 52 nuclear pharmacies. Examination of the authorized materials and possession limits of these licensees shows that a typical licensee will be authorized about 750 mCi of I-131, 2 Ci of Xe-133, 100-200 Ci of Mo-99/Tc-99m, 75 mCi of radiopharmaceuticals for use, pursuant to 10 CFR 35.100, in uptake, dilution, and excretion studies; 500 mCi of radiopharmaceuticals used, pursuant to 10 CFR 35.200, in imaging and localization studies; and about 2 Ci of radiopharmaceuticals used, pursuant to 10 CFR 35.300, in therapeutic administrations. Additionally, the licensee will be authorized to possess and distribute sealed sources and will have approximately 200-600 kg of depleted uranium for use in shielding its Mo-99/Tc-99m generators. Finally, about 40 percent of nuclear pharmacies are authorized to possess about 1 Ci of any isotope except iodines and Tc-99m with an atomic number not greater than 83.

Although almost half of the nuclear pharmacy licensees are authorized to possess about 1 Ci of any isotope with an atomic number of 83 or less, all the radiopharmaceuticals these licensees prepare, regardless of the specific radioisotope or compound, have one common characteristic - a short half-life. This is important from the perspective of the clearance alternatives. In NUREG-SR 1556, Vol. 13, *Consolidated Guidance about Materials Licenses: Program Specific Guidance about Commercial Radiopharmacy Licenses* (NRC 1999a), the NRC staff observes the following in its discussion of compliance with the regulatory requirements in 10 CFR 20.1406 regarding minimization of contamination and generation of radioactive wastes:

> *All applicants for new licenses need to consider the importance of designing and operating their facilities to minimize the amount of radioactive contamination generated at the site during its operating lifetime and to minimize the generation of radioactive waste during decontamination. In the case of commercial*

radiopharmacy applicants, these issues usually do not need to be addressed as a separate item, as they are included in responses to other items of the application. The bulk of unsealed radioactive material utilized by radiopharmacies have short half-lives (under 120 days). These radionuclides do not pose a source of long-term contamination. Additionally, nearly all radioactive waste generated by radiopharmacies is stored for decay rather than transferred to a radioactive waste disposal facility.

The following table, from NCRP Report No. 124 (NCRP 1996), indicates that the radionuclides used in nuclear medicine are very short-lived. The implication is that hold up of material for a relatively short time will eliminate the source of radioactivity whether during routine operations or during license termination.

Radionuclide	*Physical Half Life*
Tc-99m	*6 h*
I-123	*13 h*
I-131	*8 d*
Tl-201	*73 h*
Ga-67	*78 h*
Xe-133	*5.3 d*
In-111	*68 h*
Rb-82	*1.25 min*
O-15	*2.04 min*
C-11	*20.48 min*
F-18	*1.9.74 min*
N-13	*9.97 min*

The licensee may possess and redistribute sealed sources that contain radionuclides with long half lives. These sealed sources have been approved by NRC or an Agreement State and, if used according to the respective SS&D Registration Certificate, usually pose little risk of contamination. Leak tests performed at the frequency specified in the SS&D Registration Certificate should identify defective sources. Leaking sources must be immediately withdrawn from use and decontaminated, repaired, or disposed of according to NRC requirements. These steps minimize the spread of contamination and reduce radioactive waste associated with decontamination efforts.

For nuclear pharmacies, decommissioning for license termination will typically involve the removal of all sealed sources and depleted uranium and maintenance of active radiological control of the facility until 10 half-lives of the longest half-life material used at the facility have elapsed. A confirmatory survey after the appropriate elapsed time would then complete decommissioning efforts.

The origin of the radioisotopes used in radiopharmaceuticals is either nuclear reactors or accelerators. Nuclear reactors are addressed in other parts of this report. Accelerators are not licensed by the NRC but are of concern to this investigation because the State authorities that license and regulate accelerators may be influenced by the clearance alternatives. However, information characterizing the physical and radiological characteristics of accelerators was not readily available for incorporation into this report.

2.5.7 Manufacturers of Source and Radio-Labeled Compounds (63 facilities)

This section is divided into three parts. The first part summarizes the literature pertinent to defining a reference source and radio-labeled compound manufacturer as provided in NUREG-1496. The second part presents pertinent material provided in the decommissioning plan for the facility issued on February 18, 2000. The third part uses the material in NUREG-1496 and the decommissioning plan to construct an updated reference sealed source manufacturer.

2.5.7.1 Literature Review (NUREG-1496)

The following material was extracted directly from Section C.4.3 of NUREG-1496:

> *The sealed source manufacturing process is a hand operation that is carried out in buildings which contain a number of small laboratories, each of which is devoted to a specific process and/or isotope. The reference sealed source manufacturer is a laboratory which processes ^{137}Cs and ^{60}Co. Contaminated facilities associated with the reference sealed source manufacturer include:*
>
> - *hot cells, fume hoods, workbenches, sinks*
> - *laboratory floor and wall areas*
> - *building areas used for storage of waste drums.*
>
> *Advanced Medical Systems, Inc. (AMS) is used as the reference sealed source manufacturer. It is a licensed non-fuel-cycle plant in Cleveland, Ohio, that manufactures ^{137}Cs and ^{60}Co capsule sources for use in medical teletherapy devices and radiography machines (NRC, 1993).**

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* *NRC 1993. Site Decommissioning Plan, NUREG-1444, U.S. Nuclear Regulatory Commission, Washington, DC.*

The AMS operations occupy about one quarter of an 8,000-ft^2 (ground floor) warehouse building. The remainder of the building is unused. The facility occupies portions of three floors in the warehouse. The first floor consists of an office area, an isotope shop area, a hot cell, a shielded work room, and a storage area. The second floor area houses a mechanical equipment room and an exhaust ventilation equipment room. A liquid waste handling room and the former liquid waste holdup tank room and dry waste storage area are located in the basement. Waste is stored in a locked room with roped areas on the south side of the warehouse area.. The floor surface areas are estimated to be 6,000 ft^2 (assuming three floors). The indoor surface area of the walls (estimated at 10 ft high) is estimated to be 4,600 ft^2.

A 1985 survey by Oak Ridge Associated Universities (ORAU) found surface contamination in a hot cell, the ventilation system, the dry waste storage area, the liquid waste area, and the holding tank and its piping. No offsite contamination was found. However, some detectable activity (attributed to stack effluent releases) was found in sediments, soil, and vegetation in the southern portion of the AMS property. The ORAU survey showed contamination up to 1.51×10^6 dpm/100 cm^2 in the hot cell access port in the isotope shop area, an area normally expected to be highly contaminated. A water sample from the liquid waste room floor contained 1.75×10^5 pCi ^{60}Co/L. Sediment from the loading dock drain showed low but detectable levels of activity.

2.5.7.2 Advanced Medical Systems, Inc. Decommissioning Plan

The Ohio Department of Health graciously provided a copy of "Decommissioning Funding Plan," submitted to Advanced Medical Systems, Inc., by Integrated Environmental Management Systems, Inc. The complete description of the AMS site characteristics is included as Appendix E. The following was excerpted from the decommissioning plan (AMS 2000):

AMS operations that involve licensed radioactive materials occupy approximately 25% of an 80,000 square foot warehouse and manufacturing building at the London Road address. The main floor of this three-story area includes an office area, the Isotope Shop area, a hot cell, a source storage area and irradiation facility, a shielded work room, and miscellaneous unoccupied areas. The second floor contains additional unoccupied office space, a mechanical equipment room, and the ventilation system equipment room. The basement contains a former waste storage area, additional unoccupied space, and a liquid waste holdup tank room (WHUT Room). The majority of the 6.3-acre property outside of the building is covered with asphalt or concrete.

The overall inventory of equipment and contaminated areas is as follows:

Table 2-95. Contaminated Components at the AMS Facility

Component	Number of Components	Dimension of Component (units)	Total Dimensions (units)
Glove Boxes	0		
Fume Hoods	2	1 m^3	2 m^3
Lab Benches	1	1 m^3	2 m^3
Sinks	3	0.5 m^3	1.5 m^3
Drains	14	10 linear feet	140 linear feet
Floors (Basement, Isotope Shop, Decon Room, HEPA Equipment Room)	450	m^2	450 m^2
Walls (Basement, Isotope Shop, Decon Room, HEPA Equipment Room)	350	m^2	350 m^2
Ceilings	0		
Ventilation Ductwork	5	m^3	5 m^3
Hot Cell	1	4 m^2	4 m^2
Equipment, Materials, Staged/Stored Waste	20	m^3	20 m^3
Soil Plots (Old lateral/manhole)	1	4 m^3	4 m^3
Storage Tanks (3000 gal. tank)	1	3,000 gal	3,000 gal
Storage Areas	0		
Radwaste Areas	0		
Scrap Recovery Areas	0		
Maintenance Shop	0		
Equipment Decontamination Areas	0		
Other (WHUT Room)	1	20 m^2	20 m^2

Source: AMS Decommissioning Plan, Table 3.5 (AMS 2000)

2.5.7.3 Interpretation and Extrapolation

It can be assumed that, at the time of decommissioning, good health physics practice would demand that all contaminated material and surfaces be decontaminated to the extent practicable regardless of which clearance alternative is in effect. Steps would then be taken to decontaminate the structure to comply with the license termination criteria set forth in 10 CFR 20. This may involve a broad range of decontamination strategies which would generate waste material, including scabbled rubble and equipment. At that point, the structure would comply with 10 CFR 20 license termination criteria, and the material generated during the decontamination and dismantlement process would be subject to the clearance alternatives considered in this report.

In order to characterize a reference facility, two alternatives were considered: using the reference facility as defined by NUREG-1496 and Table 2-90 or directly using the information provided in the decommissioning plan for AMS. Due to the very unusual levels of contamination at AMS, it was determined that the AMS facility (based on the decommissioning plan) was not representative of typical sealed sources manufacturers. Thus, the NUREG-1496/Table 2-90 approach was selected.

In NUREG-1496, the floor surface area of the structure was estimated to be 6,000 ft^2 of which 10 percent was contaminated with 102,000 dpm/100 cm^2 of Co-60 and 33,300 dpm/100 cm^2 of Cs-137. The wall area was estimated to be 4,600 ft^2 of which 5 percent was contaminated with the same concentrations of Co-60 and Cs-137 as the floors (from Table C.7.1.1 of NUREG-1496). The volume of concrete assumed to be contaminated is between 96 and 140 ft^3, depending on the scabbling depth ranging from 0.125 to 0.625 inches, which, in turn, depends on the allowable exposure limit for future occupants with modeled doses ranging from 485 down to 0.004 mrem/yr (see Table 7.5.1 of Appendix C of NUREG-1496).

The facility has several rooms; however, only one room houses the hot cell where the sealed sources are assembled. The reference Cs-137 lab described in Table 2-90 is used as the reference lab[7]. Details of the laboratory are as follows:

- Floor - 48 m^2 asphalt tile over concrete contaminated at 1,000 dpm/100cm^2, with hot spots up to 50,000 dpm/100cm^2. Assuming 1/4 inch tile, the quantity of contaminated flooring is estimated to be about 0.3 m^3. Assuming a density of 1, this corresponds to about 300 kg. Assuming 10 percent of the area is contaminated at the high levels, the radionuclide inventory is estimated to be about 4.0×10^5 Bq (48 m^2 x 0.1 x 50,000 dpm/100 cm^2 x 104 m^2/cm^2 ÷ 60 sec/min).

- Walls - 84 m^2 latex enamel paint over concrete contaminated at 1,000 dpm/100cm^2, with hot spots up to 10,000 dpm/100cm^2. Assuming 1/2 inch of contaminated concrete (based on the concrete contamination profiles presented in

[7] AMS assembles Co-60 teletherapy sources. However, the reference Cs-137 lab should serve the purposes of this anaylsis.

Appendix C of NUREG-1496), this corresponds to about 1 m^3. Assuming a density of 1, this corresponds to about 1000 kg. Assuming 10 percent of the area is contaminated at the high levels, the radionuclide inventory is estimated to be about $2.8x10^5$ Bq.

- Ceiling - 48 m^2 latex enamel paint over concrete contaminated at 1,000 dpm/100cm^2, with hot spots up to 10,000 dpm/100cm^2. This corresponds to about 0.3 m^3 and 300 kg. Assuming 10 percent of the area is contaminated at the high levels, the radionuclide inventory is estimated to be about $1.6x10^5$ Bq.

- Hot Cells - two contaminated at $2x10^{10}$ to $2x10^{12}$ dpm/100cm^2

- Fume Hoods - two made of steel contaminated at $2x10^6$ to $2x10^7$ dpm/100cm^2. Assumed to weigh 1,000 lb each.

- Work Benches - one 3 m^2 contaminated at 1,000 dpm/100cm^2, with hot spots up to 50,000 dpm/100cm^2. Assumed to weigh 30 lb.

- Sinks and Drain - one stainless steel sink and 4 m of 0.1 m diameter pipe contaminated at 1,000 dpm/100cm^2, with hot spots up to 50,000 dpm/100cm^2. Assumed to weigh 50 lb.

- Ventilation Ductwork - 23 m of sheet metal contaminated at 1,000 dpm/100cm^2, with hot spots up to 50,000 dpm/100cm^2. Assumed to weigh 100 lb.

Not including the hot cell and fume hoods, and other areas of the facility with prohibitively high contamination levels, the quantity of material in the reference hot cell lab that may be a candidate for clearance is estimated to be about 1,700 kg, consisting of a mixture of metal, concrete, and asphalt tile, with an inventory of about $1x10^6$ Bq of Co-60, or an estimated 1 Bq /g (27 pCi/g).

The Commission's license tracking system identified 63 sealed source and radio-labeled compound manufacturers licensed by the NRC. As is the case for R&D labs, there is a great deal of diversity among the individual facilities that make up this category. Not all facilities within this category manufacture Co-60 sealed sources. Some facilities manufacture radio-labeled compounds, and therefore may have more in common with the hospital and R&D labs. Nevertheless, given the overall quality of the data for the reference facility, simple scaling of the NUREG-1496/Table 2-90 reference facility by the number of sealed source and radio-labeled compound manufacturers (63) is believed to adequately capture the expected inventory from this group of non-fuel-cycle facilities.

2.6 Site Decommissioning Management Plan Inventories

2.6.1 Introduction

This section contains information of the inventories of potentially clearable materials from 28 radiologically contaminated sites that are part of NRC's Site Decommissioning Management Plan (SDMP). These sites usually have large amounts of soil contamination or unused settling ponds or burial grounds that may be difficult to remedy. The information contained in this section came from the year 2000 status document for decommissioning, SECY-00-0094 (NRC 2000a), and site characterization, decommissioning, and remediation planning documents contained in the docket for each site. There is little detailed information concerning the inventory of potentially clearable materials other than soils, which constitue the large majority of waste for these sites.

Potentially clearable materials of interest to this report are concrete, metal, and slag removed from buildings and properties during site cleanup. Soils are evaluated in another NRC study. Buildings and structures remaining onsite after decommissioning are not considered to be potentially clearable materials, because they are regulated by 10 CFR 20 Subpart E.

Table 2-96 presents a summary of all SDMP facilities. Facilities with only soil contamination are included for completeness but are not analyzed here. The table lists the type of facility and type of waste, together with the waste volumes and contamination levels. Almost half the sites (12 out of 28) are (or were) involved in metal alloy production with large volumes of soil and slag waste. Seven of the sites are former nuclear fuel production facilities. Four sites were land disposal facilities, three sites were involved in military ordnance testing or production, and two were used for miscellaneous purposes.

Since data on materials other than soils were limited for the SDMP sites, a "reference facility" approach was utilized for estimating the quantities of other potentially clearable materials. This approach was needed because useful, site-specific information on the volumes and concentrations of most materials was very limited.

For estimating the inventory of reusable and recyclable material, two reference facilities (a metal processing facility and a uranium fabrication facility) were used. These two generic facilities cover about 70 percent of the SDMP sites. The facilities selected are the same as those described

Table 2-96. Summary of SDMP Facilities and Waste Inventory[1]

Site	Type of Facility	Waste Type and Volume	Radionuclide and Concentration
Air Manufacturing, Inc.	Thorium alloy manufacturing	Soil - 3,400 m^3	Thorium - > 13 pCi/g
B&W - Parks Operating Facility	Nuclear fuel fabrication	Soil - 1,400 m^3	Uranium - <1,000 pCi/g Am-241 - <13 pCi/g Co-60 - <199 pCi/g Cs-137 - <581 pCi/g
B&W - Parks Waste Trenches	Land disposal site	Low-level waste - 11,000 m^3	Various
Cabot Performance Materials, Inc.	Iron and tin ore processing	Slag - 5,100 m^3	Th-232 - 45 pCi/g U-238 - 30 pCi/g
Dow Chemical Company	Thorium alloy manufacturing	-	Thorium
Fansteel, Inc.	Metal recovery and processing	Soil and other residual -16,000 to 26,000 m^3	Uranium - <93 pCi/g Thorium - <51 pCi/g
Heritage Minerals Inc.	Rare metal mining	Tailings - 700 m^3	Uranium and Thorium
Jefferson Proving Ground	Military ordnance testing	Soil	Depleted uranium
Kaiser Aluminum Specialty Products	Magnesium recovery and processing	Soil and slag - 4,700 m^3	Thorium - <364 pCi/g
Kerr McGee - Cimarron	Nuclear fuel fabrication	Soil and building material - 12,000 m^3	Total uranium - <3,300 pCi/g Gross alpha - <6,000 dpm/100cm^2 Gross beta - <40,000 dpm/100 cm^2
Kerr McGee - Cushing	Nuclear fuel and oil refining	Soil, sludge, building material - 11,000 m^3	Th-232 - <20 pCi/g Uranium - <160 pCi/g
Kiski Valley Water Pollution Authority	Waste disposal lagoons	Sludge - 9,000 m^3	Uranium - 147 pCi/g
Lake City Army Ammunition Plant	Ammunition plant	Soil and building material - 22,000 m^3	Depleted uranium
Mallinckrodt Chemical, Inc.	Rare metal extraction and processing	Soil and building material	Uranium, thorium and radium
Michigan Department of Natural Resources	Landfill	Soil and slag	Thorium - >10 pCi/g Uranium

Table 2-96. Summary of SDMP Facilities and Waste Inventory (continued)

Site	Type of Facility	Waste Type and Volume	Radionuclide and Concentration
3M Company	Metal processing and waste disposal	Scrap - 50 m^3	U-234 - 2892 pCi/g U-235 - 96 pCi/g U-238 - 34 pCi/g Ra-228 - 528 pCi/g Th-228 - 305 pCi/g Th-232 - 1174 pCi/g
Molycorp - Washington, PA	Iron alloy production	Slag - 7,600 m^3, soil - 46,000 to 115,000 m^3	Thorium - <1,200 pCi/g Uranium
Molycorp - York, PA	Rare earth ore processing	Slag and soil - 5,000 m^3	Thorium
Permagrain Products, Inc.	Byproduct and irradiator facility	Soil - 120 to 360 m^3	Sr-90 - >5 pCi/g
Safety Light Corp.	Self-illuminating dial manufacturing	Soil and building material	Ra-226 - <670 pCi/g Cs-137 - <630 pCi/g
SCA Services	Landfill	Unknown	Thorium
Sequoyah Fuels Corp.	Nuclear fuel processing	Soil, sludge and building material - 155,000 to 340,000 m^3	Uranium - 5 to 500 pCi/g Thorium - <500 pCi/g Radium - 300 to 350 pCi/g
Shieldalloy Metal Corp.	Metal alloy manufacturing	Slag - 18,000 m^3, baghouse dust - 15,000 m^3	Uranium, thorium and radium
Union Carbide Corp.	Nuclear fuel development	Soil - 20 m^3, buildings	Uranium and Thorium on building surfaces - <430,000 dpm/100cm^2 Uranium in soil - <3,566 pCi/g
Watertown - GSA	Former Manhattan Project facility	Soil and building rubble	Uranium and depleted uranium - 2 to 93 pCi/g
Watertown Mall	Military ordnance facility	Piping - 360 m^3	Depleted uranium - 5,000 to 18,000 dpm/100 cm^2
Westinghouse Waltz Mill	Nuclear R&D	Building material, waste basin liners, and soil	Sr-90 - <680 pCi/g Cs-137 - <8,900 pCi/g
Whittaker Corp.	Iron alloy production	Slag and soil - 15,000 m^3	Thorium and uranium

[1] Taken from: ARR 1998; ARR 1999; BWTX 1995; BWTX 1996; Fansteel 1999; Kaiser 1999; Kerr-McGee 1998; Mallinckrodt 1997; MDNR 1999; Molycorp 1999; NRC 2000h&i; NRC 1999b.c,d,e,f; NRC 1997; NRC undated; PA 1998; SCA 1996; Sequoyah Fuels 1999; Shieldalloy 1999; US Army 1999; US COE, 2000; Westinghouse 1996; Whittaker 1998; Whittaker 1996.

in NRC's *Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning NRC-Licensed Nuclear Facilities*, NUREG-1496 (NRC 1997). Inventory information was taken from that document.

2.6.2 Radionuclides of Concern

Table 2-97 lists the frequency of occurrence of the principal radionuclides by site. The table shows that, by far, the most frequent radionuclides are those of uranium and thorium. This is because most SDMP facilities utilized ore or slag containing these naturally occurring radio-elements for either nuclear fuel or metal alloy production. In NUREG-1496, the NRC staff estimated that the uranium and thorium contamination on the surface structures before decommissioning activities was about 33 pCi/g for the two reference facilities. In addition, the slag was reported to contain an average value of 1,250 pCi/g of thorium.

Table 2-97. Frequency (Number of SDMP sites) of Sites Containing Individual Radionuclide Contamination

Element/Radionuclide					
U	***Th***	***DU***	***Pu***	***Am-241***	***Cs-137***
19	18	5	2	3	3
Co-60	***Sr-90***	***Tc-99***	***Pr***	***Ra***	***K***
2	5	1	1	5	1

2.6.3 Waste Volumes

Based on surface measurements and diffusion modeling, the NRC staff estimated that the volume of concrete removed for decommissioning was 288 m^3 for the uranium fabrication facility and 176 m^3 for the metal processing facility. They also assumed that the reference metal processing facility contained 7,000 m^3 of slag. This translates to about 4,100 m^3 of building material for all 19 facilities and 84,000 m^3 of slag for the 12 metal processing sites. Since the building material is assumed to be removed (e.g., scabbled) from structures as part of the cleanup process, this material would probably be disposed of as LLW under most regulatory options. The slag may be reprocessed for metal recovery or used for other commercial applications.

2.6.4 Regulatory Note

Besides using 10 CFR 20 Subpart E residual contamination criteria, about half the sites are using other standards and guidance for release of material. These include:

- *Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Materials,* U.S. Nuclear Regulatory Commission, August 1987.

- Option I of the 1981 Branch Technical Position (*Disposal or Onsite Storage of Thorium or Uranium Waste from Past Operations*, Federal Register, Vol. 46, No. 205, October 23, 1981, p. 52061). The numerical soil release criteria is 10 pCi/g for Th-232 and Th-228.

- Option II of the 1981 Branch Technical Position (*Disposal or Onsite Storage of Thorium or Uranium Wastes from Past Operations,* Federal Register, Vol. 46, No. 205, October 23, 1981, p. 52061). The criterion for soil is 30 pCi/g of total uranium, and the criterion for the disposal facility is <10 uR/hr above background.

- 1983 NRC directive for depleted uranium. This includes DU in soil at 35 pCi/g and a surface exposure rate of < 10 uR/hr above background.

- SDMP Action Plan (*Action Plan to Ensure Timely Remediation of Sites Listed in the Site Decommissioning Management Plan,* 57 FR 13389, April 1992). The proposed release criteria for building surfaces is < 5 uR/hr above background.