DOE/RW-0184-R1
Volume 2

CHARACTERISTICS OF POTENTIAL REPOSITORY WASTES

JULY 1992

Prepared for the
U.S. DEPARTMENT OF ENERGY
OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
Washington, D.C. 20585

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
managed by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under
Contract No. DE-AC05-84OR21400

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Chapter 4. Non-LWR Spent Fuels

Chapter 5. Miscellaneous Wastes

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LIST OF ACRONYMS

AC	Allis Chalmers
ANF	Advanced Nuclear Fuels Corporation
ANL	Argonne National Laboratory
AP	Activation products
APSR	Axial power shaping rod
ASTM	American Society for Testing and Materials
B-C	Battelle-Columbus
B&W	Babcock and Wilcox
BPRA	Burnable poison rod assembly
BWR	Boiling-water reactor
CC	Complexant concentrate
CDB	Characteristics Data Base
CE	Combustion Engineering
CEA	Control element assembly
CEU	Consolidated Edison uranium
CFR	Code of Federal Regulations
CH	Contact handled
CWMS	Civilian Waste Management System
DHLW	Defense high-level waste
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
ECF	Expended Core Facility
EFPD	Equivalent full-power days
EIA	Energy Information Administration
EIS	Environmental impact statement
EPRI	Electric Power Research Institute
FFTF	Fast Flux Test Facility
FIS	Federal Interim Storage
FP	Fission products
FSV	Fort St. Vrain
GAPSR	Gray axial power shaping rod
GE	General Electric
GTCC	Greater than Class C
GWd	Gigawatt-days
GW(e)	Gigawatts (electric)
HANF	Hanford Site
HEDL	Hanford Engineering Development Laboratory
HEPA	High-efficiency particulate air
HLW	High-level waste
HTGR	High-temperature gas-cooled reactor
HWVP	Hanford Waste Vitrification Plant
ICPP	Idaho Chemical Processing Plant
IDB	Integrated Data Base
INEL	Idaho National Engineering Laboratory
LANL	Los Alamos National Laboratory
LER	Licensee Event Report
LLW	Low-level waste
LTA	Lead test assembly
LWBR	Light-water breeder reactor
LWR	Light-water reactor
MOX	Mixed oxide
MRS	Monitored retrievable storage

LIST OF ACRONYMS (continued)

MSRE	Molten Salt Reactor Experiment
MT	Metric tons
MTIHM	Metric tons of initial heavy metal
MTR	Materials Test Reactor
MW(t)	Megawatts (thermal)
MW(e)	Megawatts (electric)
NCAW	Neutralized current acid waste
NCRW	Neutralized cladding removal waste
NFA	Non-fuel assembly
NFB	Non-fuel bearing
NFS	Nuclear Fuel Services, Inc.
NMMSS	Nuclear Materials Management and Safeguards System
NPR	New Production Reactor
NRC	Nuclear Regulatory Commission
NRF	Naval Reactors Facility
NWTSP	National Waste Terminal Storage Program
O/U	Oxygen/uranium atom ratio
OCRWM	Office of Civilian Radioactive Waste Management
OFA	Optimized fuel assembly
ORA	Orifice rod assembly
ORNL	Oak Ridge National Laboratory
PB1	Peach Bottom Unit 1
PC	Personal computer
PCI	Pellet-clad interaction
PFP	Plutonium finishing plant
PIE	Postirradiation examination
PNL	Pacific Northwest Laboratory
PNS	Primary neutron source
PWR	Pressurized-water reactor
QA	Quality assurance
QC	Quality control
RH	Remotely handled
RNS	Regenerative neutron source
SAS	Statistical Analysis System
SFD	Spent fuel disassembly
SNF	Spent nuclear fuel
SRL	Savannah River Laboratory
SRP	Savannah River Plant
SRS	Savannah River Site
SS	Stainless steel
SST	Single-shell tanks
TMI-2	Three Mile Island 2
TRIGA	Training Research Isotopes - General Atomics
TRU	Transuranic (waste)
TRUW	Transuranic waste
UN	United Nuclear
VEPCO	Virginia Electric Power Co.
WAC	Waste acceptance criteria
WAPS	Waste Acceptance Preliminary Specification
WE or W	Westinghouse
WIPP	Waste Isolation Pilot Plant
WVDP	West Valley Demonstration Project

4. NON-LWR SPENT FUELS

4.1 INTRODUCTION

The LWR spent fuels discussed in Section 2 of this report comprise about 99% of all domestic non-reprocessed spent fuel. In this section we discuss other types of spent fuels which, although small in relative quantity, consist of a number of diverse types, sizes, and compositions. Many of these fuels are candidates for repository disposal. Some non-LWR spent fuels are currently reprocessed or are scheduled for reprocessing in DOE facilities at the Savannah River Site, Hanford Site, and the Idaho National Engineering Laboratory. It appears likely that the reprocessing of fuels that have been reprocessed in the past will continue and that the resulting high-level wastes will become part of defense HLW. However, it is not entirely clear in some cases whether a given fuel will be reprocessed, especially in cases where pretreatment may be needed before reprocessing, or where the enrichment is not high enough to make reprocessing attractive. Some fuels may be canistered, while others may require special means of disposal.

4.1.1 Scope

The major categories covered in this chapter include HTGR spent fuel from the Fort St. Vrain and Peach Bottom-1 reactors, research and test reactor fuels, and miscellaneous fuels. This information is arranged as follows:

Sect. 4.2. Fort St. Vrain fuel

Sect. 4.3. Peach Bottom-1 fuel

Sect. 4.4. Research and test reactor fuels. This includes fuels from privately owned commercial reactors, government-owned reactors, and reactors used for educational purposes.

Sect. 4.5. Miscellaneous fuels. These are arranged by site.

Additional information on the reactors and fuels discussed in this Section is given in the following Appendices:

Appendix 4A. Nuclear Reactors at Educational Institutions in the United States

Appendix 4B. Data Sheets for Nuclear Reactors at Educational Institutions

Appendix 4C. Supplemental Data for Fort St. Vrain Spent Fuel

Appendix 4D. Supplemental Data for Peach Bottom-1 Spent Fuel

Appendix 4E. Supplemental Data for Fast Flux Test Reactor

4.2 FORT ST. VRAIN REACTOR FUEL

The Fort St. Vrain (FSV) reactor is a high temperature, gas-cooled reactor located in Platteville, Colorado. The coolant gas is helium. Operation of the reactor started in 1974 with a rated power of 842 MW thermal. The total initial core loading was 774 kg of 93.5% 235U and 15,905 kg Th. In December 1988, Public Service of Colorado announced that final shutdown of the reactor would take place no later than June 1990. Actual shutdown took place on August 18, 1989, and no further operation of the reactor is planned.

4.2.1 Core Design and Discharge Schedule

The FSV reactor core is divided into 37 separate refueling regions. Figure 4.2.1 shows a core plan view with the 37 regions identified. The core contains a total of 247 fuel columns. Thirty-one of the 37 regions have 7 columns each, a center control fuel column and six surrounding columns made up of fuel elements of conventional design. The other six regions, which are located near the edge of the core, each contain one control fuel column and four fuel element columns. The full core consists of this pattern stacked six-high, bringing the total to 1,482 fuel elements (Morissette et al., 1986).

The FSV reactor was designed to operate on a graded fuel cycle with about one-sixth of the reactor core being replaced at each refueling. It was planned that a full refueling cycle would consist of five reloads of 240 elements each and one reload of 282 elements. Table 4.2.1 identifies the regions of the core that were planned to be replaced in each of the six reloads of the first full reloading cycle (Nirschl 1973).

Table 4.2.2 shows the actual schedule of spent fuel discharged from the FSV reactor through the end of 1989 and the projected schedule through 1991. The total in-core inventory at the time of shutdown was 1,482 elements. All this fuel is scheduled to be removed by the end of 1991, giving a total cumulative discharge of 2,208 elements (Brey 1990, DOE 1990). All spent fuel discharged prior to December 31, 1988, is located at the Idaho Chemical Processing Plant (ICPP). Fuel removed from the core in 1989 and 1990 remains on-site in temporary storage wells until shipment to the ICPP can be accomplished or an independent spent fuel storage installation is built for permanent storage.

4.2.2 FSV Fuel Element Characteristics

The core design used three types of fuel elements: standard, control, and bottom control fuel elements. The fuel elements (Fig. 4.2.2) are hexagonal graphite blocks drilled with a multiplicity of fuel holes and coolant channels. Internal coolant channels within each element are aligned with coolant channels in elements above and below. The active fuel is contained in an array of small-diameter holes, which are parallel with the coolant channels, and occupy alternating positions in a triangular array within the graphite structure.

Lateral alignment of the six-layered fuel element column is maintained by a system of three graphite dowels located on the top face of each element. A normal coolant channel passes through the center of each dowel. The dowels are threaded into the graphite structure and affixed with a carbonaceous cement.

4.2.2.1 Standard Fuel Element

All standard fuel elements have 210 fuel holes 0.500-in. in diameter and 108 coolant passages (Bingham and Evans 1976). When fully loaded they contain 3,132 fuel rods, which are right cylinders made of coated particles bonded together with a low density graphite matrix. A standard fuel rod has a diameter of 0.5 in. and a length of 1.94 in.

4.2.2.2 Control Fuel Elements

The center control rod fuel element in each region is similar to a standard fuel element, but contains enlarged channels for the two control rods and the reserve shutdown absorber material (Fig. 4.2.3). Each control rod fuel element contains 120 fuel holes loaded with a total of 1782 fuel rods, and has 57 coolant channels. The control rod channels have a diameter of 4.00-in. and a centerline separation distance of 9.72 in. The reserve shutdown channel has a diameter of 3.75 in.

4.2.2.3 Bottom Control Fuel Elements

The bottom element in the control rod column extends below the core about 7.5 in. The fuel holes and the absorber channel hole are arranged so that all elements at the bottom of the core are at the same elevation. Each bottom control fuel element contains 120 fuel holes loaded with a total of 1,302 fuel rods (Fig. 4.2.4).

4.2.2.4 Burnable Poison

All of the standard elements have 0.500 in.-diameter holes in each of their six corners for possible insertion of burnable poison rods. All of the control and bottom control elements have similar holes on four corners for burnable poison rods. In the initial loading, some burnable poison rods were placed in selected standard fuel elements, but none were placed in the control or bottom control elements.

The burnable poison rods are 2.00 in. long and 0.45 in. diameter. They were added as required and did not always fill the complete hole. Further information is given in Appendix 4C.

4.2.2.5 Coated Fuel Particles

The fissile and fertile fuel particles are TRISO-coated microspheres of uranium and thorium carbide. As shown in Fig. 4.2.5, each fuel particle consists of a spherical kernel covered with four main layers of coating material plus a thin intermediate seal coating. The kernel dimensions, coating designations, and coating thicknesses for the fissile and fertile particles are listed in Table 4.2.3. Reading from the kernel toward the outer surface, the coatings are as follows: (1) a buffer coating of low-density pyrolytic carbon, (2) a thin seal coating (material unspecified), (3) an inner isotropic coating of high-density pyrolytic carbon, (4) a silicon carbide coating, and (5) an outer isotropic coating of high-density pyrolytic carbon. The overall outside diameter of the fissile and fertile particles are approximately 460 and 730 microns, respectively. This size difference would have allowed separation of fissile and fertile particles if reprocessing of the fuel had been carried out as originally planned.

4.2.2.6 Fuel Element Weights

The weights of the various types of fuel elements vary from 128 kg to 109 kg and are listed for each type in Table 4.2.4 (Morissette 1986).

4.2.3 FSV Fuel Chemical Characteristics

The fuel block is made of nuclear grade graphite, type H-327 or type H-451, manufactured by Great Lakes Carbon Company. This graphite has very low levels of impurities. The specified maximum concentrations for impurities in the H-451 graphite (Engle 1979) were as follows:

Boron (755 barns/atom)	5 ppm
Iron (2.4 barns/atom)	100 ppm
Titanium (5.6 barns/atom) and Vanadium (5.1 barns/atom)	100 ppm total
Total ash	1,000 ppm

The maximum impurities for type H-327 graphite are the same as above except that titanium and vanadium each have an individual limit of 50 ppm rather than a combined limit of 100 ppm (Disselhorst 1972).

Dowels and plugs used in the fuel element are of the same type of graphite as the element and are bonded to the block with a carbonaceous cement.

4.2.3.1 Fuel Rods

The fuel rods consist of close-packed coated fuel particles bonded together with a low-density graphite matrix (GA 1975). Limits for impurity concentrations in the fuel rods are shown in Table 4.2.5. The values shown are excerpted from GA specification GA-10600, issue BP, dated October 1982. This specification has undergone several changes since its original issue, so the values given in Table 4.2.5 do not necessarily apply to all fuel loadings.

4.2.3.2 Coated Fuel Particles

The coated fuel particles in spent fuel contain mainly uranium, thorium, and mixed fission products. Small amounts of transuranic actinides are also present. The uranium and thorium are in the form of carbides.

Approximately 0.3 to 0.5 percent of the coatings are expected to be failed in the first three refueling segments (Kowal 1984, Moore 1978, and Graul 1982).

4.2.3.3 Poison Rods

Selected elements contain burnable poison rods. The rods are made of boron carbide particles in a carbon matrix. Limits for impurities in these rods (Beavan 1973) are shown in Table 4.2.6 (Beavan 1973, Rickard 1991).

4.2.4 Postirradiation Condition of Fuel

A nondestructive examination of various fuel elements was performed after each set of elements was removed from the core. Nearly all of the elements had shrunk slightly in both axial and radial dimensions. However, the inspected elements were generally in good condition. Minor cracks, chips, and scratches were observed on some elements. A more detailed analysis of postirradiation effects is contained in Appendix 4C.

All of the fuel discharged to date has experienced much lower burnup than that expected for the equilibrium core, which is about 100,000 MWd/MTIHM. The maximum burnup for the discharged fuel occurred in an element from segment 3 and was slightly under 47,000 MWd/MTIHM.

The fuel burnup is calculated for each fuel element removed. These calculations employ a three-dimensional model of the FSV HTGR; a computer code named "BUGATT" is used. The results for the fuel elements in the three discharged segments are stored on floppy discs in the format shown in Table 4.2.7 (Morissette 1986).

In order to establish the accuracy of the calculated values for burnup, measured and calculated element average values for surveillance element 1-0743 were compared (GA 1975) and are summarized in Table 4.2.8. The differences between calculated vs measured composite burnups are -3.5% $\pm$ 2.0% (1 σ) for the GAUGE analysis, -9.9% $\pm$ 1.9% (1 σ) for the GATT analysis, and -17.6% $\pm$ 1.7% (1 σ) for the FEVER analysis. The GATT analysis was used in calculating burnups for the first three segments which have been discharged (Morissette 1986).

A comparison of measured and calculated uranium isotopic concentrations for the same surveillance element

1-0743 is given in Table 4.2.9. The U-234 and U-235 concentrations are slightly lower than predicted, while the U-236 and U-238 concentrations are higher than predicted. U-233 concentration is not reported (Morissette 1986).

The radiological characteristics of average FSV fuel irradiated to 100,000 MWd/MT have been calculated for various decay times ranging from 120 days to 1 million years. Calculations were made by ORNL using the ORIGEN2 code; the starting composition for the decay calculations was taken from Morissette 1986. The calculated radioactivity for selected nuclides as a function of time is shown in Table 4.2.10; the calculated thermal power is given in Fig. 4.2.6. The "bump" at 10^4 to 10^5 years is due to the decay daughters of U-233 in the fertile particles. For fuels with reduced irradiation, acceptable first approximations of radioactivity and thermal power should be obtained by using linear interpolation from the values at 100,000 MWd/MTIHM. For transuranic content, linear interpolation will give a conservative (i.e., too high) result.

4.2.5 Container Description

The FSV spent fuel elements are currently stored at ICPP in 18-in.- diam, 1/4-in.-thick carbon steel canisters with a length of 11 ft. They have ungasketed lids which are held in place by remotely operable DE-STA-CO clamps; see Fig. 4.3.4 and Bingham 1976 for additional details. Each canister contains four FSV elements. The current inventory of 724 elements thus requires 181 canisters. Information on the serial numbers of the elements and the canister numbers in which they are contained is available at INEL.

4.2.6 Repository Canister Requirements

As indicated previously (Table 4.2.2), it is expected that the reactor will be completely defueled by the end of year 1991. When defueling is completed, the total cumulative spent fuel in storage will be 2,208 elements with a total mass of 24.0 MTIHM. No determination has been made as to whether the spent fuel elements will be disassembled for reprocessing or sent to a repository or MRS facility as intact spent fuel elements. If the elements are stored without disassembly, about 552 canisters will be required for storage (Salmon and Notz 1990).

4.2.7 Fuel Types and Identifying Markings

The initial core loading consisted of 84 different types of fuel elements. The variations in design result from differences in the block, different fuel loadings, the positioning of the burnable poison rods, and the neutron sources. A unique identification system consisting of three digits engraved on the side of the hexagonal block and a serial number also engraved on the block insure that the history of each element can be appropriately traced as needed. A detailed exposition of the numbering logic is given in Appendix 4C.

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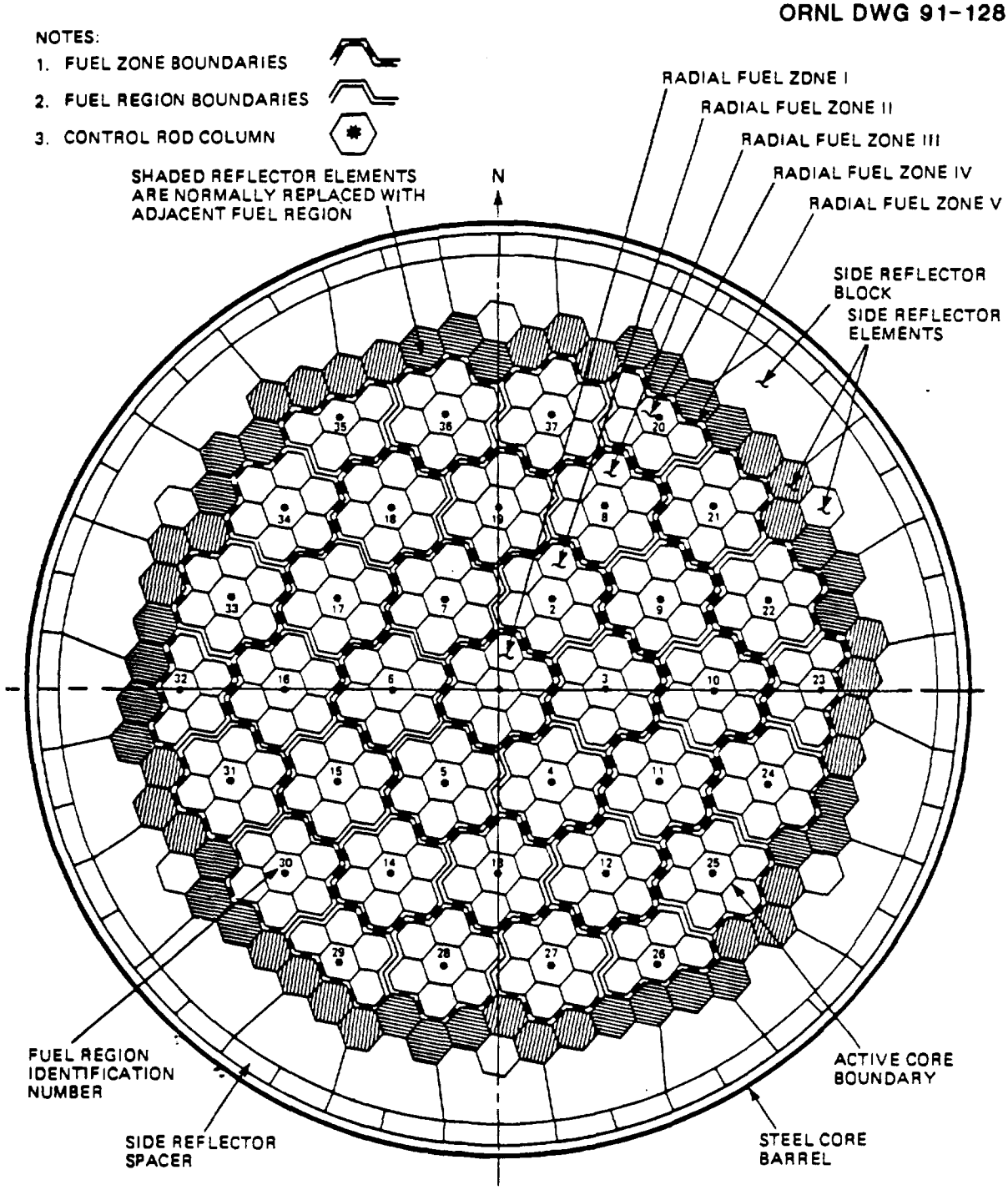

Fig. 4.2.1. FSV core plan view.

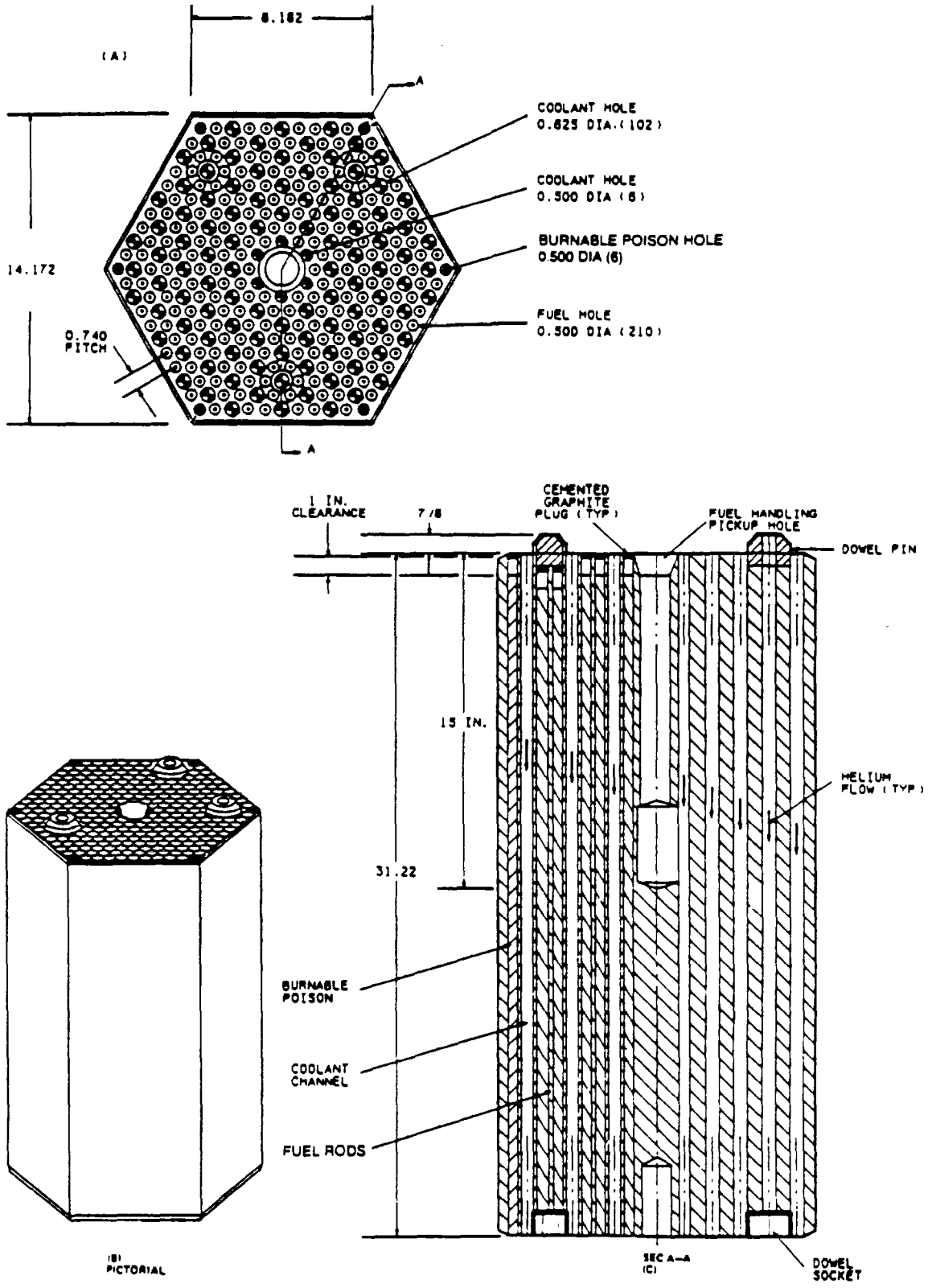

Fig. 4.2.2. FSV standard fuel element.

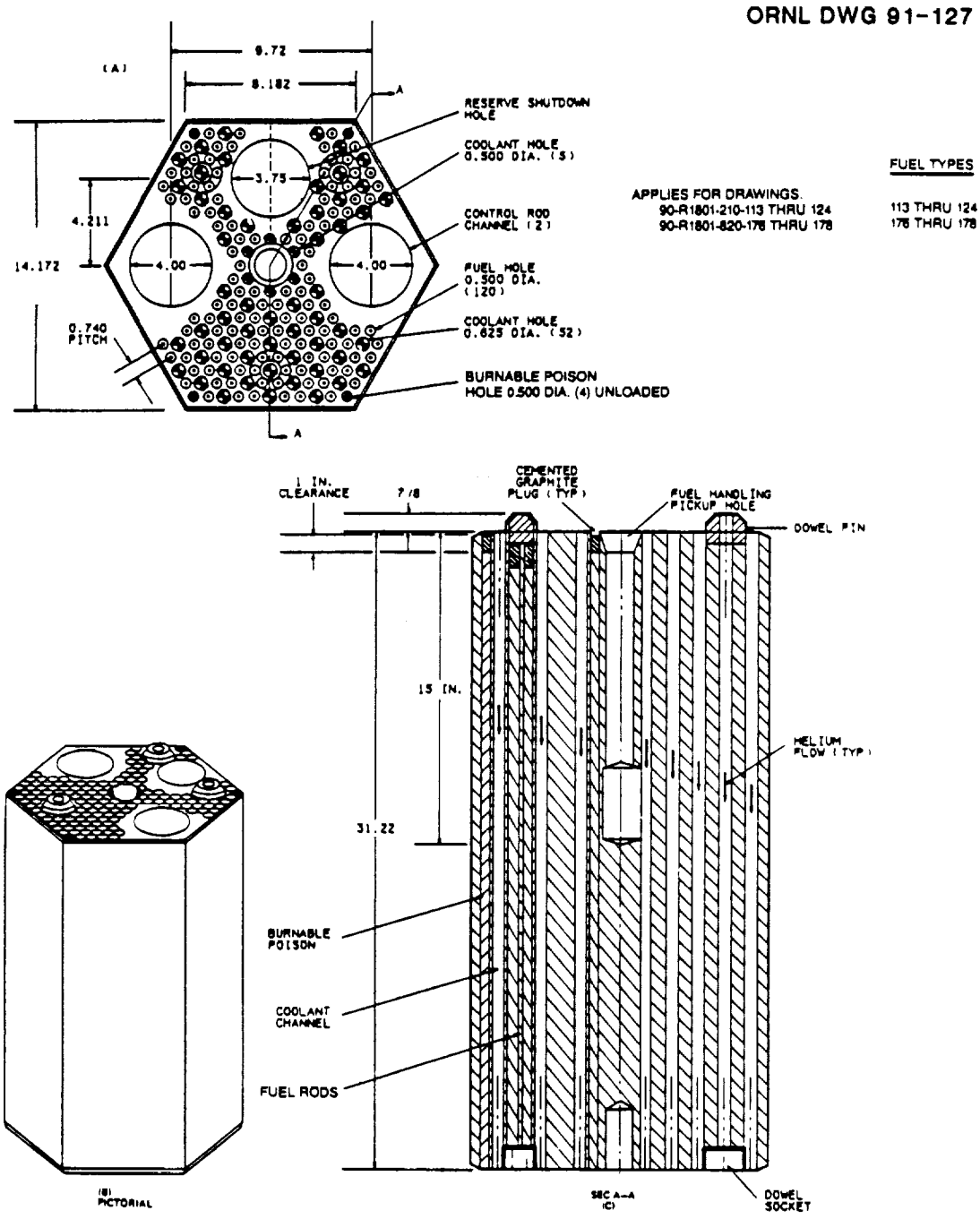

Fig. 4.2.3. Control fuel elements and surveillance control element.

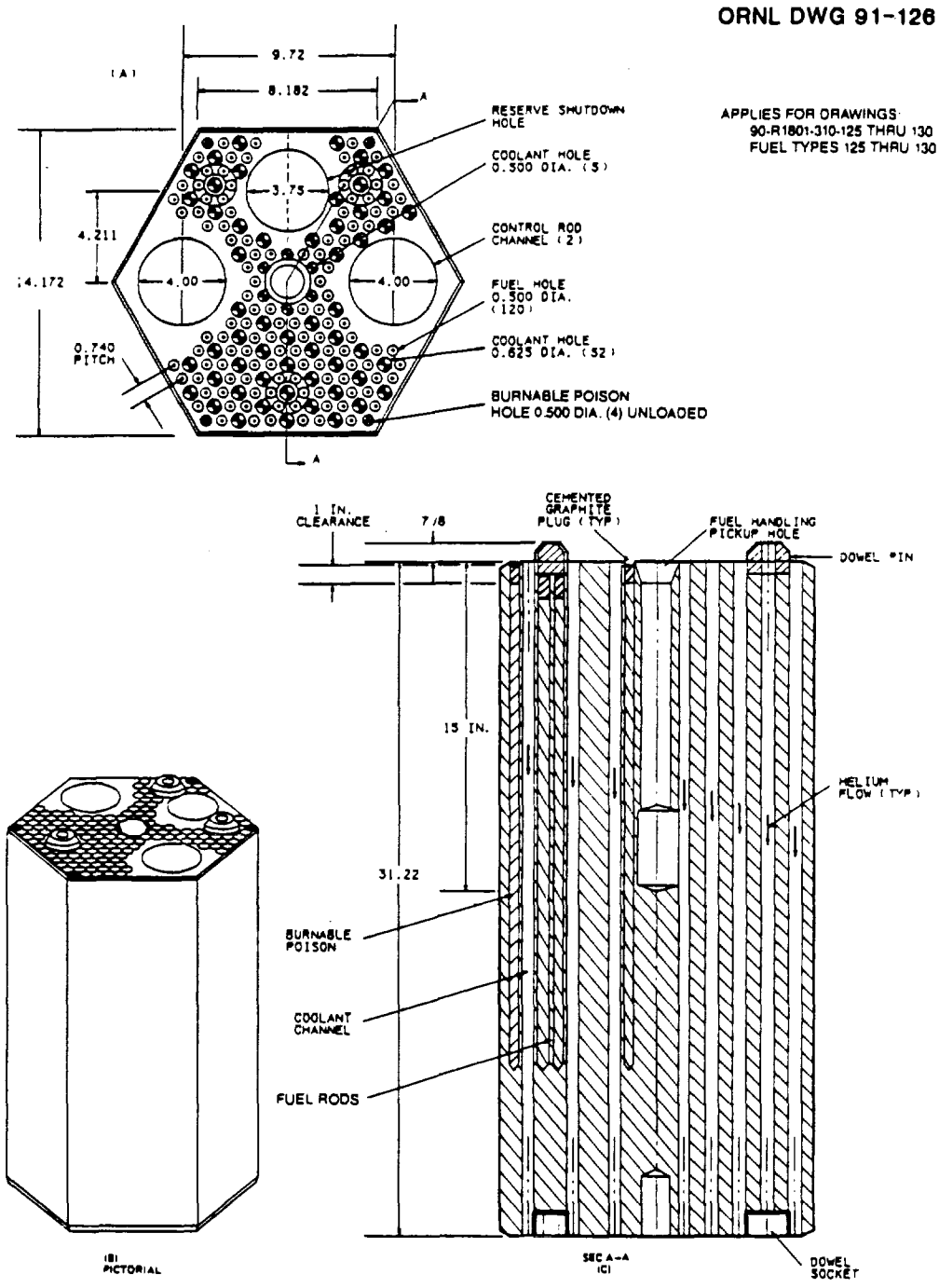

Fig. 4.2.4. Bottom control fuel element.

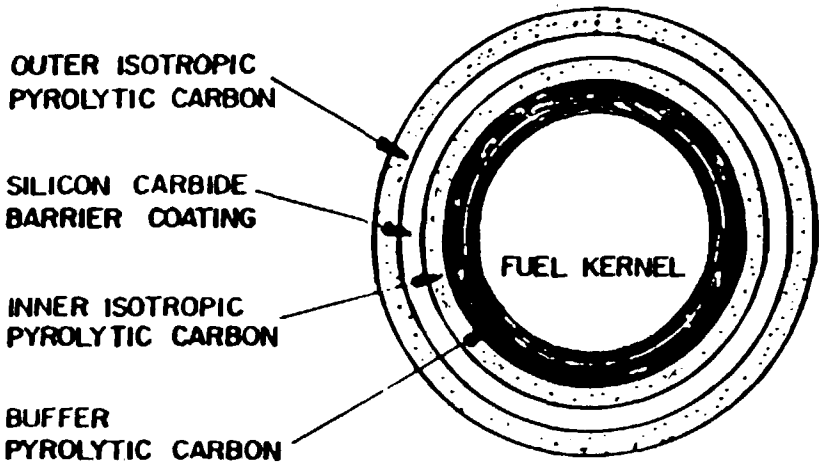

Fig. 4.2.5. Fertile and fissile fuel particles used in Fort St. Vrain reactor (approximately 100X).

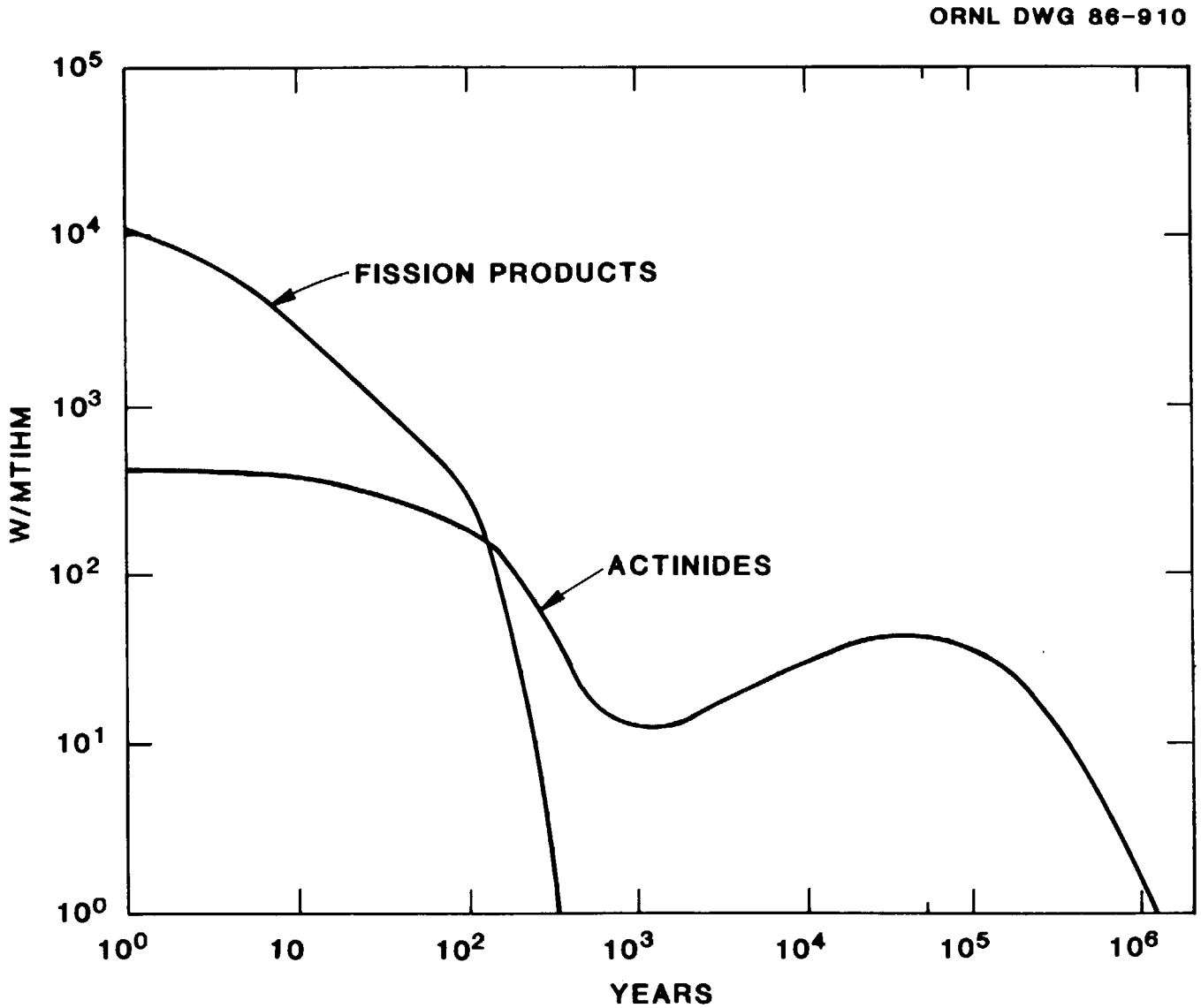

Fig. 4.2.6. Thermal power for Fort St. Vrain spent fuel. Basis: 1 metric ton initial heavy metal irradiated to 100,000 MWd/MTIHM.

Table 4.2.1. Refueling sequence and reload segment description for the FSV refueling cycle (Nirschl 1973)

Reload No.	Core regions, refueled	Segment No.	No. of spent fuel elements
1	5, 10, 17, 21, 28, 35	1	245[a]
2	4, 8, 15, 25, 32, 36	2	240
3	3, 13, 18, 22, 29, 33	3	240
4	2, 11, 16, 26, 30, 37	4	240
5	1, 7, 9, 14, 23, 27, 34	5	282
6	6, 12, 19, 20, 24, 31	6	240

[a]Includes replacing one fuel element with a test fuel element in five other regions. These five other regions are: 25, 22, 30, 27, and 24.

Table 4.2.2. Historical and projected spent fuel discharged from the Fort St. Vrain HTGR[a]

End of calendar year	Number of fuel assemblies discharged		Mass of fuel discharged (MTIHM)	
	Annual	Cumulative	Annual	Cumulative
1979	246[b]	246	2.80	2.80
1980	0	246	0.00	2.80
1981	240	486	2.77	5.57
1982	0	486	0.00	5.57
1983	0	486	0.00	5.57
1984	240	726	2.85	8.42
1985	0	726	0.00	8.42
1986	0	726	0.00	8.42
1987	0	726	0.00	8.42
1988	0	726[c]	0.00	8.42
1989	126[d]	852	1.32	9.74
1990	615[d,e]	1,467	6.47	16.21
1991	741[f]	2,208	7.49	24.00

[a]Based on Brey 1990 and DOE 1990.

[b]This refueling replaced 246 spent fuel elements made up of 240 standard fuel elements and 6 fuel test elements.

[c]All spent fuel discharged prior to December 31, 1988 is located at the ICPP.

[d]Fuel removed from the core in 1989 and 1990 remains on-site in temporary storage wells until shipment to the ICPP can be accomplished or an independent spent fuel storage installation is built for permanent storage.

[e]1990: 330 fuel blocks have been removed from the core prior to February 28, 1990.

[f]It is expected that the entire core will be defueled by the end of 1991.

Table 4.2.3. Fort St. Vrain fuel particle diameters and coating thicknesses[a]

Measurement	Fissile particle (10^{-6} m)	Fertile particle (10^{-6} m)
Kernel diameter[b]	100–275	300–500
Coating thicknesses		
Buffer[c]	45–110	45–65
Seal[d]	<5	<5
Inner isotropic[c]	20–30	20–40
Silicon carbide[c]	20–30	20–30
Outer isotropic	≥25	≥30
Outside diameter[e]	460	730

[a]Source: GA specification GA-10600, issue BS.
[b]Reference value; not a requirement.
[c]Sample mean.
[d]Ten percent may have regions $>5 \times 10^{-6}$ m thick.
[e]Approximate.

Table 4.2.4. Weights of FSV fuel elements and components[a]

Item	Weight
Fuel elements	
Standard	128 kg
Surveillance	128 kg
Neutron source	128 kg
Californium neutron source	128 kg
Test	126 kg
Bottom control	111 kg
Control	109 kg
Surveillance control	109 kg
Fuel element components	
Graphite body	
Regular fuel element	86 kg
Control rod fuel element	85 kg
Bottom control rod fuel element	94 kg
Fuel rod (per rod)[b]	
Thorium	2.7 g
Uranium	0.1 g
Silicon	0.8 g
Coatings	4.1 g
Matrix	0.8 g

[a]All weights are approximate. Source: Morissette 1986.

[b]This is for an individual fuel rod, which is about 1.27 cm (0.5 in.) diam by 4.93 cm (1.94 in.) long. Data given in Morissette are for an earlier version 3-in. rod and have been scaled down to a 1.94-in. rod. The minimum rod diameter is 0.487 in. The maximum is "loose fit in fuel hole."
Ref. GA90R1801-920/G.

Table 4.2.5. Fort St. Vrain reactor: fuel rod impurity limits[a]

Property	Maximum acceptable value, ppm
Total boron equivalents of impurities in fired fuel rods	
For any fuel rod lot:	
Total burnable[b]	10
Total nonburnable[c]	5
For any fuel element:	
Total burnable[b]	7.5
Total nonburnable[c]	2
Other impurities[d]	
Sulfur	1,200
Hydrogen (residual after firing)	200
Residual chlorine	150
Iron (segment weighted average)	20
Transition metals (each metal)[e]	55

[a]Source: GA 1982. The values shown are excerpted; for a more complete discussion, see the reference specification (GA-10600, issue BP, October 1982). The specification has undergone changes since its original issue and, therefore, does not necessarily apply to all the fuel loadings in the reactor.

[b]The total burnable value is the total boron equivalent content of B, Cd, Eu, Gd, and Sm.

[c]The total nonburnable value is the total boron equivalent content of Al, Ag, Ba, Bi, Ca, Cr, Co, Cu, Fe, Mg, Mn, Mo, Ni, P, Na, Sr, S, Sn, Ti, V, and Zn.

[d]These are maximum values for any fuel rod lot and are given in terms of mass of impurity, not boron equivalents.

[e]The transition elements referred to are Co, Cr, Mn, and Ni. The limit of 55 ppm (mass) applies to each element.

Table 4.2.6. Fort St. Vrain reactor: poison rod impurity limits[a]

Impurity	Limits of impurities, ppm	
	GA-18-R-09 Issue G (1973)	GA-18-R-09 Issue S (1982)
Iron	250	500
Cadmium	250	250
Hafnium	250	250
Sulfur	250	500
Total specified metals[b]	5,000	5,000
Boron oxide	c	c

[a]Source: GA document 18-R-09, issue G (Beavan 1973) and issue S (Rickard 1991).

[b]Specified metals in Issue G are: Al, Ba, Ca, Co, Cr, Cu, Fe, Mg, Mn, Mo, Na, Ni, Pb, Si, Sn, Sr, Ti, V, and Zn. After Issue G, V was dropped from this list.

[c]At least 99 wt % of the boron present must be in the carbide form. Boron in the oxide form must not exceed 1 wt % of total boron.

Table 4.2.7. Sample of fuel accountability data (Morissette 1986)

Serial number	1-1773		Accountability date: 3/31/86
Core location			
Region	18		
Column	7		
Layer	7		
		Heavy metal weights, g	
Particle	Nuclide	Initial	Current
Fertile	Th-232	8,331.77	8,056.46
Fertile	Pa-231	.00	.03
Fertile	U-232	.00	.03
Fertile	U-233[a]	.00	152.78
Fertile	U-234	.00	14.19
Fertile	U-235	.00	1.58
Fertile	U-236	.00	.10
Fissile	Th-232	1,832.23	1,771.69
Fissile	Pa-231	.00	.01
Fissile	U-232	.00	.01
Fissile	U-233[a]	.00	33.60
Fissile	U-234	3.24	5.30
Fissile	U-235	407.07	123.40
Fissile	U-236	1.24	49.72
Fissile	U-238	25.46	22.58
Fissile	Np-237	.00	3.44
Fissile	Pu-238	.00	.72
Fissile	Pu-239[b]	.00	.54
Fissile	Pu-240	.00	.24
Fissile	Pu-241	.00	.20
Fissile	Pu-242	.00	.13
Total		10,601.00	10,236.76
Total fissile uranium		407.07	311.36
Total uranium		437.00	403.29
Total fissile plutonium		.00	.75
Total plutonium		.00	1.85
Effective U-233 enrichment, %		.00	46.21
Effective U-235 enrichment, %		93.15	30.99
U-232, ppm		c	c
Fertile particle fima, %		.00	1.29
Fissile particle fima, %		.00	11.27
Burnup, MWd/metric ton		32,601.50	
Cumulative EFPD			657.30

[a]Includes full decay of Pa-233.
[b]Includes full decay of Np-239.
[c]The initial and final U-232 concentrations are 0.00 and 90.79 ppm, respectively.

Table 4.2.8. Comparison of calculated vs measured fuel burnup for FSV fuel element 1-0743 (Morissette 1986)

	Burnup, %										
	Measured[a]		Case I[b]			Case II[c]			Case IV[d]		
Particle type	FIMA	$\pm 1\sigma$	FIMA	Z[e]	$\pm 1\sigma$[f]	FIMA	Z[e]	$\pm\sigma$[f]	FIMA	Z[e]	$\pm\sigma$[f]
$(Th,U)C_2$	6.38	0.15	6.2	-2.8	2.3	5.90	-7.5	2.2	5.30	-16.9	2.0
ThC_2	0.32	0.01	0.3	-6.2	2.9	0.25	-21.9	2.4	0.25	-21.9	2.4
Composite	1.42	0.03	1.37	-3.5	2.0	1.28	-9.9	1.9	1.17	-17.6	1.7

[a]Determined by averaging $(Th,U)C_2$ burnups at location of monitors 21 and 81 and ThC_2 burnups for fuel rods 12-4 and 279-3. These averages should be approximately equivalent to element average burnups.

[b]SURVEY-detailed GAUGE analysis.

[c]GATT analysis.

[d]Calculations based on FEVER-calculated fluxes.

[e]In all cases, Z (%) = 100 (calculated/measured) - 100.

[f]Progressed uncertainty due to measurement uncertainty only.

Table 4.2.9. Comparison of calculated and measured uranium isotopic concentrations for UC_2 burnup monitors irradiated in FSV fuel element 1-0743 (Morissette 1986)

	Isotopic concentration				
	Measured[a]			Relative difference	
Isotope	Atom percent	σ	Calculated[b] atom percent	Z (%)[c]	$+\sigma s$[d]
U-234	0.797	0.002	0.8	0.38	0.25
U-235	79.62	0.02	82.6	3.74	0.03
U-236	10.98	0.02	8.9	-18.94	0.15
U-238	8.60	0.01	7.7	-10.46	0.10

[a]Average values for monitors 21 and 81. The average neutron flux for these two monitors is approximately equivalent to the element average flux.
[b]Calculations based on fluxes obtained from the FEVER code.
[c]Z (%) = 100 (Calculated/Measured) - 100.
[d]Progressed uncertainty due to measurement uncertainty only.

Table 4.2.10. Radioactivity of Fort St. Vrain reactor spent fuel[a]
(Based on one MTIHM; 100,000 MWd/MTIHM)

	Curies							
Nuclide	120.0D	1.0 YR	10.0 YR	100.0 YR	1000.0 YR	10.0 KY	100.0 KY	1.0 MY
				Actinides and daughters				
TL207	0.0	3.795E-03	4.760E-02	1.718E-01	1.765E-01	1.518E-01	5.180E-02	3.432E-02
TL208	0.0	1.404E+01	5.897E+01	2.557E+01	3.789E-02	3.348E-02	3.348E-02	3.347E-02
TL209	0.0	2.747E-04	3.957E-03	4.060E-02	3.896E-01	2.583E+00	2.941E+00	7.437E-02
PB209	0.0	1.272E-02	1.832E-01	1.880E+00	1.804E+01	1.196E+02	1.362E+02	3.443E+00
PB210	0.0	1.835E-07	3.588E-05	2.118E-03	7.214E-02	2.081E+00	1.567E+01	2.624E+00
PB211	0.0	3.806E-03	4.773E-02	1.723E-01	1.770E-01	1.522E-01	5.195E-02	3.441E-02
PB212	0.0	3.909E+01	1.641E+02	7.118E+01	1.054E-01	9.317E-02	9.317E-02	9.317E-02
PB214	0.0	1.813E-05	2.651E-04	3.157E-03	7.216E-02	2.081E+00	1.567E+01	2.625E+00
BI210	0.0	1.835E-07	3.589E-05	2.118E-03	7.214E-02	2.081E+00	1.567E+01	2.624E+00
BI211	0.0	3.806E-03	4.773E-02	1.723E-01	1.770E-01	1.522E-01	5.195E-02	3.441E-02
BI212	0.0	3.909E+01	1.641E+02	7.118E+01	1.054E-01	9.317E-02	9.317E-02	9.317E-02
BI213	0.0	1.272E-02	1.832E-01	1.880E+00	1.804E+01	1.196E+02	1.362E+02	3.443E+00
BI214	0.0	1.813E-05	2.651E-04	3.157E-03	7.216E-02	2.081E+00	1.567E+01	2.625E+00
PO210	0.0	5.514E-08	3.589E-05	2.118E-03	7.214E-02	2.081E+00	1.567E+01	2.624E+00
PO212	0.0	2.504E+01	1.052E+02	4.560E+01	6.756E-02	5.969E-02	5.969E-02	5.969E-02
PO213	0.0	1.244E-02	1.792E-01	1.839E+00	1.765E+01	1.170E+02	1.332E+02	3.369E+00
PO214	0.0	1.812E-05	2.651E-04	3.157E-03	7.214E-02	2.081E+00	1.567E+01	2.624E+00
PO215	0.0	3.806E-03	4.773E-02	1.723E-01	1.770E-01	1.522E-01	5.195E-02	3.441E-02
PO216	0.0	3.909E+01	1.641E+02	7.118E+01	1.054E-01	9.317E-02	9.317E-02	9.317E-02
PO218	0.0	1.813E-05	2.652E-04	3.158E-03	7.217E-02	2.081E+00	1.567E+01	2.625E+00
AT217	0.0	1.272E-02	1.832E-01	1.880E+00	1.804E+01	1.196E+02	1.362E+02	3.443E+00
RN219	0.0	3.806E-03	4.773E-02	1.723E-01	1.770E-01	1.522E-01	5.195E-02	3.441E-02
RN220	0.0	3.909E+01	1.641E+02	7.118E+01	1.054E-01	9.317E-02	9.317E-02	9.317E-02
RN222	0.0	1.813E-05	2.652E-04	3.158E-03	7.217E-02	2.081E+00	1.567E+01	2.625E+00
FR221	0.0	1.272E-02	1.832E-01	1.880E+00	1.804E+01	1.196E+02	1.362E+02	3.443E+00
RA223	0.0	3.806E-03	4.773E-02	1.723E-01	1.770E-01	1.522E-01	5.195E-02	3.441E-02
RA224	6.295E+04	3.909E+01	1.641E+02	7.118E+01	1.054E-01	9.317E-02	9.317E-02	9.317E-02
RA225	0.0	1.272E-02	1.832E-01	1.880E+00	1.804E+01	1.196E+02	1.362E+02	3.443E+00
RA226	0.0	1.813E-05	2.652E-04	3.158E-03	7.217E-02	2.081E+00	1.567E+01	2.625E+00
RA228	9.849E-01	9.251E-01	4.210E-01	9.320E-02	9.317E-02	9.317E-02	9.317E-02	9.317E-02
AC225	0.0	1.272E-02	1.832E-01	1.880E+00	1.804E+01	1.196E+02	1.362E+02	3.443E+00
AC227	0.0	3.806E-03	4.768E-02	1.722E-01	1.770E-01	1.522E-01	5.195E-02	3.441E-02
AC228	9.746E-01	9.252E-01	4.211E-01	9.320E-02	9.317E-02	9.317E-02	9.317E-02	9.317E-02
TH227	0.0	3.753E-03	4.707E-02	1.699E-01	1.746E-01	1.501E-01	5.123E-02	3.394E-02
TH228	0.0	3.909E+01	1.640E+02	7.118E+01	1.054E-01	9.317E-02	9.317E-02	9.317E-02
TH229	0.0	1.272E-02	1.832E-01	1.880E+00	1.804E+01	1.196E+02	1.362E+02	3.443E+00
TH230	6.225E-02	6.241E-02	6.459E-02	8.732E-02	3.316E-01	2.654E+00	1.553E+01	2.623E+00
TH231	0.0	3.412E-02	3.412E-02	3.412E-02	3.413E-02	3.420E-02	3.442E-02	3.441E-02
TH232	9.317E-02	9.317E-02	9.317E-02	9.317E-02	9.317E-02	9.317E-02	9.317E-02	9.317E-02
TH234	5.760E+01	5.065E-02	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.100E-04
PA231	1.799E-01	1.799E-01	1.799E-01	1.796E-01	1.769E-01	1.521E-01	5.193E-02	3.441E-02
PA233	1.187E+06	2.242E+03	9.919E-01	9.954E-01	1.016E+00	1.019E+00	9.899E-01	7.396E-01
PA234M	0.0	5.065E-02	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.100E-04
PA234	5.760E+01	6.660E-05	1.183E-06	1.183E-06	1.183E-06	1.183E-06	1.183E-06	1.183E-06
U232	1.806E+02	1.795E+02	1,646E+02	6.920E+01	1.195E-02	2.793E-40	0.0	0.0
U233	2.001E+02	2.007E+02	2.007E+02	2.006E+02	1.998E+02	1.921E+02	1.299E+02	3.316E+00
U234	2.686E+01	2.688E+01	2.714E+01	2.892E+01	3.056E+01	2.979E+01	2.308E+01	1.801E+00
U235	3.412E-02	3.412E-02	3.412E-02	3.412E-02	3.413E-02	3.420E-02	3.442E-02	3.441E-02
U236	0.0	1.507E-07	2.235E-06	2.551E-05	2.538E-04	1.656E-03	2.528E-03	2.461E-03
U238	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.101E-04	9.100E-04
NP237	9.918E-01	9.918E-01	9.919E-01	9.954E-01	1.016E+00	1.019E+00	9.899E-01	7.396E-01
NP239	0.0	3.285E+00	3.282E+00	3.254E+00	2.991E+00	1.284E+00	2.740E-04	5.339E-41
PU238	1.054E+04	1.049E+04	9.773E+03	4.800E+03	3.933E+00	1.652E-20	0.0	0.0
PU239	8.526E+00	8.526E+00	8.524E+00	8.512E+00	8.374E+00	6.914E+00	5.599E-01	3.087E-12
PU240	7.562E+00	7.600E+00	8.023E+00	8.953E+00	8.169E+00	3.146E+00	2.252E-04	0.0
PU241	4.339E+03	4.201E+03	2.724E+03	3.578E+01	5.475E-18	0.0	0.0	0.0

Table 4.2.10 (continued)

Nuclide	Curies							
	120.0D	1.0 YR	10.0 YR	100.0 YR	1000.0 YR	10.0 KY	100.0 KY	1.0 MY
Actinides and daughters (continued)								
AM241	7.700E+00	1.229E+01	6.094E+01	1.328E+02	3.164E+01	1.706E-05	0.0	0.0
AM242M	5.339E-01	5.323E-01	5.109E-01	3.389E-01	5.594E-03	8.411E-21	0.0	0.0
AM242	0.0	5.296E-01	5.083E-01	3.372E-01	5.566E-03	8.369E-21	0.0	0.0
AM243	3.285E+00	3.285E+00	3.282E+00	3.254E+00	2.991E+00	1.284E+00	2.740E-04	5.339E-41
CM242	2.385E+03	8.420E+02	4.213E-01	2.789E-01	4.603E-03	6.942E-21	0.0	0.0
CM243	8.766E-01	8.624E-01	6.929E-01	7.763E-02	2.421E-11	0.0	0.0	0.0
CM244	5.472E+02	5.333E+02	3.779E+02	1.206E+01	1.323E-14	0.0	0.0	0.0
SUBTOT	1.269E+06	1.903E+04	1.451E+04	5.822E+03	4.385E+02	1.217E+03	1.403E+03	6.170E+01
Fission products								
KR 85	5.074E+04	4.858E+04	2.715E+04	8.062E+01	4.476E-24	0.0	0.0	0.0
SR 89	3.484E+05	1.202E+04	3.046E-16	0.0	0.0	0.0	0.0	0.0
SR 90	2.795E+05	2.751E+05	2.220E+05	2.607E+04	1.296E-05	0.0	0.0	0.0
Y 90	2.809E+05	2.752E+05	2.221E+05	2.607E+04	1.297E-05	0.0	0.0	0.0
Y 91	4.003E+05	2.190E+04	2.672E-13	0.0	0.0	0.0	0.0	0.0
ZR 95	5.460E+05	3.831E+04	1.309E-11	0.0	0.0	0.0	0.0	0.0
NB 95	1.050E+05	7.623E+04	2.906E-11	0.0	0.0	0.0	0.0	0.0
NB 95M	0.0	2.842E+02	9.709E-14	0.0	0.0	0.0	0.0	0.0
RU103	7.367E+04	9.726E+02	6.281E-23	0.0	0.0	0.0	0.0	0.0
RH103M	7.421E+04	8.768E+02	5.662E-23	0.0	0.0	0.0	0.0	0.0
RU106	7.200E+04	4.538E+04	9.312E+01	1.244E-25	0.0	0.0	0.0	0.0
RH106	7.279E+04	4.538E+04	9.312E+01	1.244E-25	0.0	0.0	0.0	0.0
SN123	2.566E+03	6.882E+02	1.507E-05	0.0	0.0	0.0	0.0	0.0
SB125	6.545E+03	5.533E+03	5.819E+02	9.629E-08	0.0	0.0	0.0	0.0
TE125M	0.0	1.265E+03	1.420E+02	2.349E-08	0.0	0.0	0.0	0.0
TE127	6.681E+04	1.368E+04	1.143E-05	0.0	0.0	0.0	0.0	0.0
TE127M	6.645E+04	1.397E+04	1.167E-05	0.0	0.0	0.0	0.0	0.0
TE129	2.139E+04	8.742E+01	3.090E-28	0.0	0.0	0.0	0.0	0.0
TE129M	2.114E+04	1.343E+02	4.748E-28	0.0	0.0	0.0	0.0	0.0
I129	0.0	1.233E-04	1.241E-04	1.241E-04	1.241E-04	1.240E-04	1.235E-04	1.187E-04
CS134	5.667E+03	4.522E+03	2.195E+02	1.622E-11	0.0	0.0	0.0	0.0
CS137	2.859E+05	2.815E+05	2.286E+05	2.858E+04	2.660E-05	0.0	0.0	0.0
BA137M	2.845E+05	2.663E+05	2.163E+05	2.703E+04	2.516E-05	0.0	0.0	0.0
BA140	2.747E+03	4.637E-03	0.0	0.0	0.0	0.0	0.0	0.0
LA140	3.102E+03	5.337E-03	0.0	0.0	0.0	0.0	0.0	0.0
CE141	1.683E+05	9.028E+02	3.299E-28	0.0	0.0	0.0	0.0	0.0
PR143	4.790E+03	1.729E-02	0.0	0.0	0.0	0.0	0.0	0.0
CE144	1.312E+06	7.215E+05	2.383E+02	3.679E-33	0.0	0.0	0.0	0.0
PR144	1.311E+06	7.215E+05	2.383E+02	3.679E-33	0.0	0.0	0.0	0.0
PR144M	0.0	8.658E+03	2.860E+00	4.415E-35	0.0	0.0	0.0	0.0
ND147	3.520E+02	7.438E-05	0.0	0.0	0.0	0.0	0.0	0.0
PM147	5.401E+05	4.523E+05	4.195E+04	1.974E-06	0.0	0.0	0.0	0.0
SM147	0.0	2.153E-06	1.221E-05	1.324E-05	1.324E-05	1.324E-05	1.324E-05	1.324E-05
SM151	6.771E+03	6.736E+03	6.285E+03	3.142E+03	3.067E+00	2.409E-30	0.0	0.0
EU154	4.026E+02	3.814E+02	1.847E+02	1.307E-01	4.108E-33	0.0	0.0	0.0
EU155	3.850E+03	3.505E+03	9.962E+02	3.429E-03	0.0	0.0	0.0	0.0
SUBTOT	6.418E+06	3.343E+06	9.672E+05	1.110E+05	3.067E+00	1.372E-04	1.368E-04	1.319E-04

[a]Nuclides contributing <0.0010% are omitted.

4.3 PEACH BOTTOM UNIT 1

Peach Bottom Unit 1 was a high-temperature gas-cooled reactor with a rated capacity of 115 MW(t). It was located at Peach Bottom, Pennsylvania, and operated from 1966 to 1974. The reactor contained 804 fuel elements per fuel load. The total number of fuel elements irradiated in the core was 1,639, which exceeds two times 804 because of replacement fuel and test fuel. The graphite-based fuel elements were 3.5 in. in diameter and 12 ft in length, containing varying amounts of uranium and thorium. These heavy metals were present as carbon-coated uranium carbide and thorium carbide particles that had been formed into compacts by sintering with carbonaceous materials.

Two cores were used during the lifetime of the reactor. The design burnup of the fuel was ~73,000 MWd/MTIHM; however, excessive fuel failures during operation of Core 1 resulted in removal of that core at about half the design burnup. The fuel failure was attributed to the fuel particle coating system. This system was modified for the second core, which performed satisfactorily and reached design burnup. The reactor was shut down at this point. The total initial heavy metal loadings of the two cores were 1.686 and 1.419 MT of U + Th, respectively.

4.3.1 Physical Description of Fuel

The basic fuel element, shown in Fig. 4.3.1, is a solid semihomogeneous type in which graphite served as the moderator, reflector, cladding, fuel matrix, and structure. Each fuel element consists of an upper reflector assembly, a fuel bearing middle section, a lower reflector, and an internal fission product trap. The fuel materials, part of the lower reflector, and the fission product trap are contained in a sleeve of low-permeability graphite that joins the upper reflector on one end and a bottom connector fitting on the other. A stainless steel screen installed at the bottom of each fission product trap retains any charcoal granules that might be released from the graphite body of the internal trap. Within the sleeve, the mixture of fissile and fertile materials making up the fuel is contained in annular compacts stacked on cylindrical graphite spines; the Core 1 fuel compact assembly is shown in Fig. 4.3.2. The Core 2 fuel compact assembly is shown in Appendix 4D.

The reactor core consisted of a number of fuel elements that were instrumented with thermocouples and (in Core 1 only) acoustic thermometers. Thirty-six such instrumented elements were included in the 804 fuel elements required for each core loading. In addition, 33 fuel test elements were irradiated in Core 2 to various exposures; the purpose of this was to measure the thermal, physics, fission product, and materials behavior of commercial HTGR fuel concepts utilizing test assemblies in a representative commercial HTGR neutron spectrum and a helium coolant environment.

Three basic fuel element configurations were irradiated in both Peach Bottom reactor cores: standard fuel elements, instrumented fuel elements, and test elements. Standard fuel elements are described below. Instrumented and test elements are described in Appendix 4D. The external appearance of all configurations is the same.

4.3.1.1 Core 1 Fuel Element

The Core 1 standard fuel element (Fig. 4.3.1) has as its primary components a bottom connector, a sleeve, a screen, an internal fission product trap assembly, a lower reflector piece, fuel compacts, spines, burnable poison compacts (in selected elements), a fuel cap, and an upper reflector assembly. The bottom connector and the sleeve are joined by a silicon braze, and together they form the main barrier against fission-product leakage from the fuel element. The fuel cap is a graphite disk that slips loosely into the upper end of the sleeve. All three of these components (bottom connector, sleeve, and fuel cap) are made of graphite, which has a helium permeability of 3×10^{-3} cm^2/s or less and an effective permeability to gaseous fission products of approximately 10^{-5} cm^2/s at reactor conditions.

The screen, internal trap assembly, lower reflector piece, fuel compacts with spines, and fuel cap are stacked, in that order, within the sleeve. The weight of these components is supported by the bottom connector. The lower reflector piece is a 3-in.-long graphite cylinder made of reactor-grade graphite. The annular fuel compacts are stacked on the cylindrical graphite spine sections. These spine sections are approximately 30 in. long and about 1.75 in. in diameter. There are two types of spines: one made of solid graphite and one with a 0.89-in.-diam hole designed to contain burnable poison compacts. The screen, which is used to retain any charcoal granules that might be released from the graphite body of the internal trap, is made of 18-8 stainless steel.

The upper reflector assembly is a machined graphite component that is threaded and cemented into the sleeve of the fuel element. The cement consists of furnace-cured carbonaceous material. The upper end of the reflector piece is machined to engage with the fuel handling machines. A 0.25 in.-diam hole down the centerline of the reflector serves as an inlet channel for purge gas. A porous plug cemented and retained within the upper reflector provides a controlled pressure drop for inflowing purge gas.

The uranium and thorium within the fuel compacts are in the form of carbides uniformly dispersed as coated particles in the graphite matrix. The particle coating is monolithic, laminar, pyrolytic carbon obtained by sintering at 1800°C.

The Core 1 fuel compacts consisted of carbides of uranium [enriched to 93.15% ^{235}U at the beginning of life (BOL)] and thorium, uniformly dispersed as coated

particles in a graphite matrix. The total carbon within the carbide substrates was between 11 and 16%, by weight, at BOL. The pyrolytic carbon-coated particles were between 210 and 595 mm in diameter, with coating thicknesses of 55 ± 10 μm. The size distribution of the particles was designed to ensure that the volume fraction of the coated particles did not exceed 30% of the total compact volume.

Burnable poison compacts, cylindrical in shape, were placed in hollow spines of some of the fuel elements. Each compact contains 0.436 ± 0.030 g of natural boron in the form of zirconium diboride pressed into a graphite matrix. The maximum particle size of the zirconium diboride is 100 μm.

4.3.1.2 Core 2 Fuel Element

The Core 2 standard fuel elements were essentially the same as the Core 1 elements (see Appendix 4D). The only design difference was in the coated particles and the external appearance of the fuel compacts. The coating of the Core 2 fuel and fertile particles consisted of an inner coating of low-density pyrolytic carbon surrounded by an outer isotropic layer of pyrolytic carbon ("BISO" particle). The total coating thickness was between 90 and 130 μm. The coated particles were ~340 and 630 μm in diameter, respectively, for the fuel and fertile particles. The Core 2 compacts were smooth and had slots on the ends.

4.3.2 Materials and Masses

4.3.2.1 Compositions

The compositions of the various fuel element components are indicated in Table 4.3.1.

4.3.2.2 Weights

The weights of the several styles of fuel elements, fuel element components, and certain filled storage apparatus are indicated in Table 4.3.2. The metal loadings in the four fuel element types found in each of the two cores are provided in Table 4.3.3.

4.3.3 Postirradiation Condition of Spent Fuel

Because the condition of Core 1 varied significantly from that of Core 2, the cores are discussed separately in the following paragraphs.

4.3.3.1 Core 1

Core 1 contained fuel particles coated with a single layer of pyrolytic graphite. Dimensional changes caused by fast neutrons and damage due to fission product recoils resulted in cracking and distortion of the coatings on the fuel particles. The broken coatings, in the process of curling and changing dimensions, caused the compacts to distort and swell. The radial expansion produced in the compacts caused them to bind against the graphite sleeve, leading to fracture in some cases. A total of 90 elements in Core 1 developed cracked sleeves (Scheffel et al. 1976). Two elements were broken during core removal.

The fuel in the balance of the core remained intact and was removed and then packaged for disposal. It can be assumed that some of the fuel particles had failed and some of the compacts had experienced swelling in this fuel. Several Core 1 elements were examined, and the results were reported in a series of documents (Scheffel et al. 1976). Based on these examinations, it is expected that the compacts can be removed from the graphite sleeves if this becomes a viable treatment option.

4.3.3.2 Core 2

Core 2 operated close to its full design lifetime of 900 equivalent full-power days (EFPD). The design of a new coated fuel particle resolved the problem experienced in Core 1, and all elements were in good condition after removal from the reactor.

Postirradiation examinations were performed on several Core 2 regular fuel elements. Data on the condition of this fuel is reported by Scheffel, Dyer, Wichner, and co-workers (Scheffel et al. 1976, Dyer 1976, 1978; Wichner 1977a, b, 1978, 1979).

4.3.4 Radiological Characteristics

Core 1 was irradiated to 451 EFPD, and Core 2 to 897 EFPD, as compared with the design core lifetime of the fuel of 900 EFPD. Burnup data for the two cores are summarized in Table 4.3.4.

4.3.4.1 Heavy Metals Content of Discharged Cores

The heavy-metal content of each fuel element has been calculated. The results are available in hard copy at INEL. Table 4.3.5 provides the sums of all the calculated amounts for the 813 elements irradiated as Core 1 and the 804 elements discharged as Core 2. Data on the loadings of specific elements were provided to INEL by Philadelphia Electric (Conti 1971) or with shipping records.

4.3.4.2 Spent Fuel Element Radioactivity and Thermal Power

Table 4.3.6 shows the radionuclide content of a Peach Bottom Core 2 spent fuel element with a fuel burnup of 73,000 MWD/MTIHM and a cooling time of 120 days. The radionuclide contents shown are given in a safety analysis report on the INEL irradiated fuels storage facility and are based on 2.5 years of reactor operation at 114 MW(t) (INEL 1976). Using the radionuclide contents

at a cooling time of 120 days as input, a series of calculations was made by means of the ORIGEN2 code to determine the radioactivity (curies) and thermal power (watts) per fuel element at total decay times of 120 days to one million years. The initial cooling period of 120 days was included in the total decay time. These results are shown in Tables 4.3.7 and 4.3.8.

4.3.5 Spent Fuel Inventory

The Peach Bottom 1 reactor was shut down on October 31, 1974, and all of the spent fuel was shipped to storage. The total inventory of spent fuel from the reactor consists of two cores (Core 1 and Core 2), some replacement elements, and a number of test elements. There were 819 fuel elements (818 regular elements and one test element) in Core 1. There were 820 fuel elements (787 regular elements and 33 test elements) in Core 2. The total number of elements in both cores was 1639.

Most of the spent fuel (813 Core 1 and 785 Core 2 elements) is stored at Idaho National Engineering Laboratory (INEL). All the spent fuel shipped to General Atomic Corporation has subsequently been shipped to INEL for storage. Twelve elements were shipped to ORNL. Two of these were destroyed in the course of examination; the remaining ten are in retrievable underground storage. This leaves 29 Peach Bottom 1 fuel elements unaccounted for. The number of fuel elements per container and the quantities of total uranium and ^{235}U per container are shown in Appendix 4D.

The data received from INEL (Denney 1986) on the Peach Bottom spent fuel does not allow a detailed inventory of each element by serial number or type; however, such information does exist (Morissette 1986). Core 1 elements are stored in groups of 18 or less, while the Core 2 elements are stored in groups of 12 or less.

4.3.6 Packaging

4.3.6.1 Core 1

Core 1 is currently stored in open-field drywells at the ICPP Fermi I Blanket Storage Facility at INEL. The fuel was placed in sealed aluminum canisters with stainless steel liners at Peach Bottom after removal from the reactor. The failed fuel was removed from the core with a stainless steel failed fuel element tool, and both the tool and the element were placed in a sealed canister. Figure 4.3.3 describes the canister without a removal tool. The loaded canisters weigh about 150 lb. Appendix 4D describes both a canister with a removal tool and a salvage canister surrounding a leaking canister. The part numbers given on these figures are identification numbers defined by Philadelphia Electric (USAEC/PEC 1971).

The canisters of fuel were shipped to INEL in the Peach Bottom fuel shipping cask. The elements were positioned in the cask by means of a basket assembly with a diameter of 25.5 in. At INEL, an entire basket loaded with canisters was lowered into a drywell. A loaded basket assembly weighs 3,400 lb. Forty-six baskets are situated in dry wells.

Removal and canning of the failed Core 1 fuel resulted in a number of package types. These are described in Appendix 4D (USAEC/PEC 1971).

4.3.6.2 Core 2

The Core 2 spent fuel was packaged for shipment using canisters of the same type as those used for Core 1. The Core 2 fuel was placed in the Irradiated Fuel Storage Facility at INEL. This required removing the fuel from the canister and cutting the top reflector so that the element could be placed in the 11-ft-long storage canister. Therefore, the resulting element length is approximately 10 ft 6 in. Each canister, shown in Fig. 4.3.4, contains 12 Peach Bottom elements (INEL 1976).

4.3.7 Quantities To Be Disposed Of

Table 4.3.9 summarizes the spent fuel quantities for the Peach Bottom 1 reactor.

4.3.8 References and Bibliography for Section 4.3 and Appendix 4D

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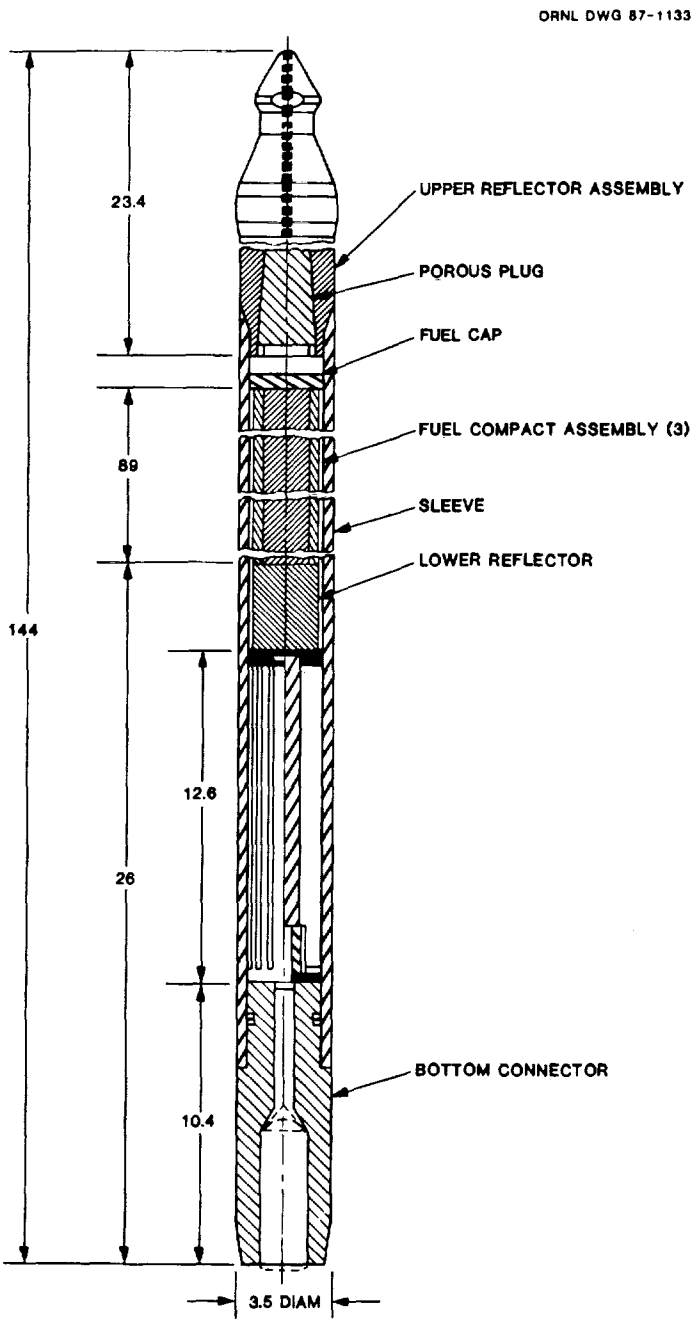


Note: All dimensions are in inches.

Fig. 4.3.1. Peach Bottom Unit 1, Core 1 fuel element.

ORNL DWG 91-129

FUEL TYPE	NUMBER OF ELEMENTS REQUIRED
1	54
2	564
3	84
4	102
	804

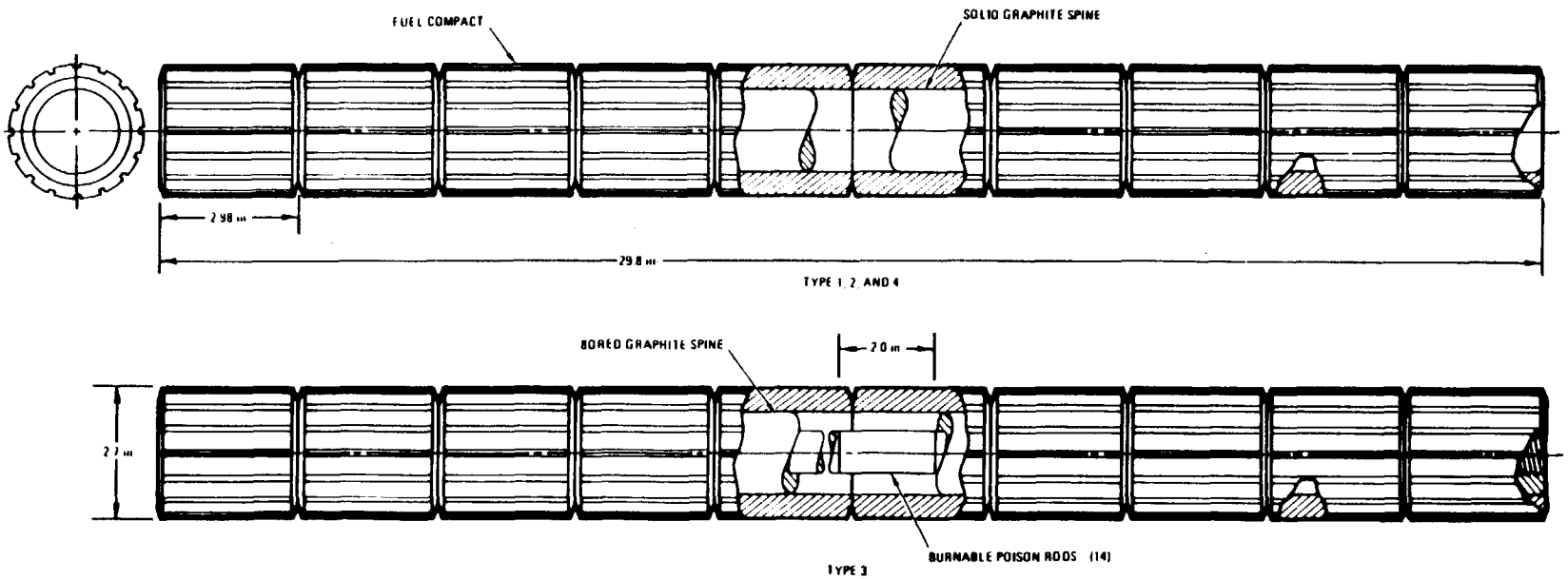
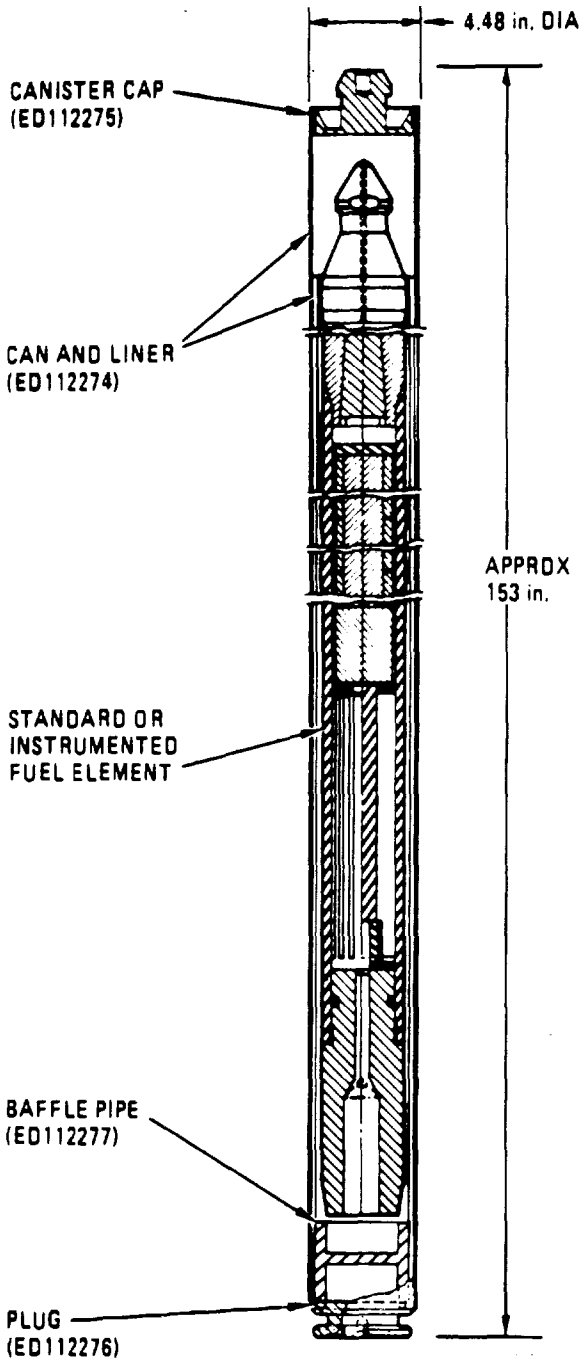


Fig. 4.3.2. Peach Bottom Unit 1, Core 1 fuel compacts.

ORNL DWG 91-132



FUEL PACKAGE TYPE	NUMBER OF PACKAGES
1	528
9	71
14	98
17	1
18	18
20	3
21	4
	723

Fig. 4.3.3. Peach Bottom Unit 1, Core 1 non-failed fuel element in storage canister.

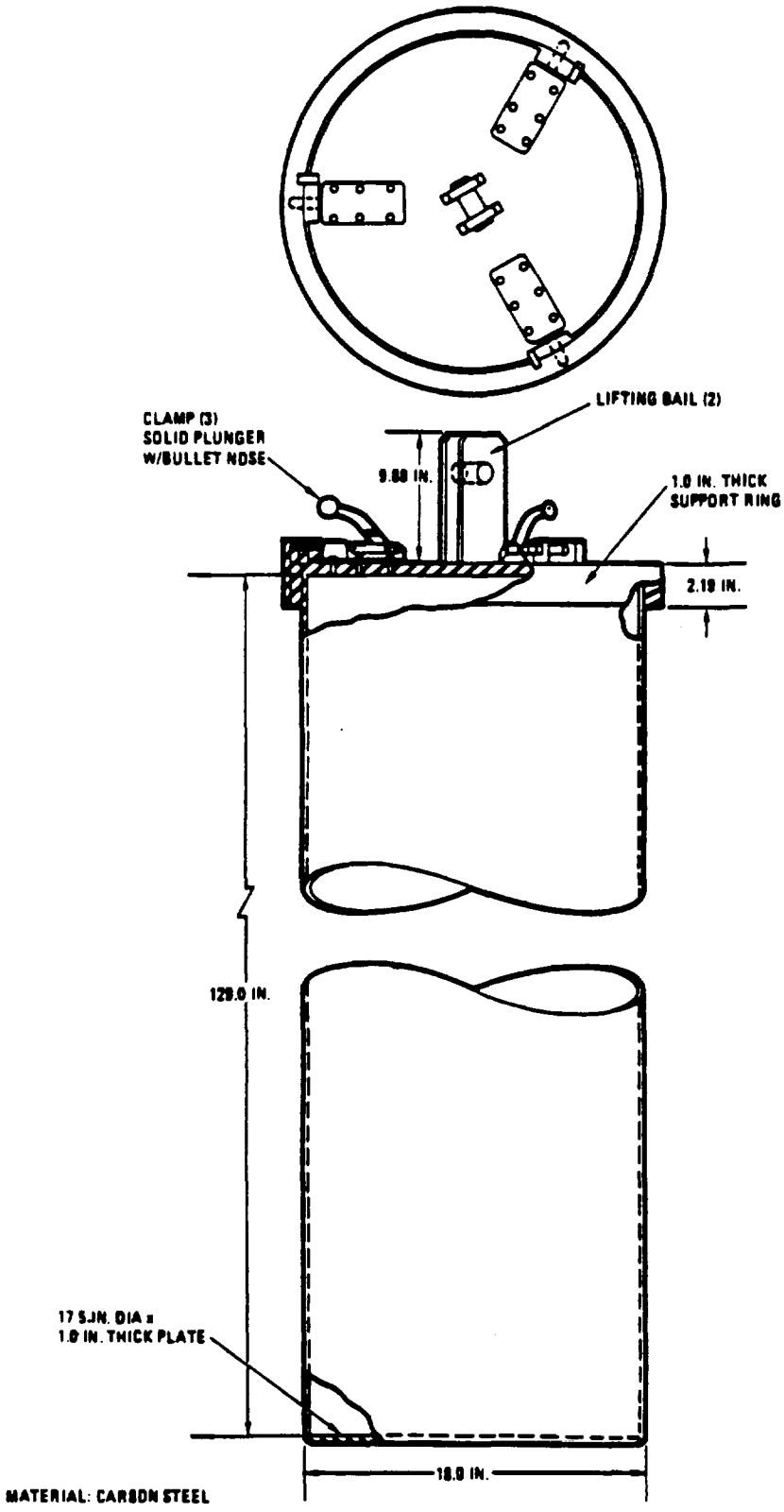


Fig. 4.3.4. Peach Bottom Unit 1, Core 2 storage canister.

Table 4.3.1. Materials used in Peach Bottom 1 fuel element components[a]

Component	Material
Fuel compact assemblies	
Fuel compacts	Pyrolitic carbon coated UC_2/ThC_2 particles in graphite matrix
Solid or bored spines	Graphite
Burnable poison compacts	ZrB_2 in graphite matrix
Non-fuel components	
Upper reflector	Graphite
Porous plug	Graphite
Fuel cap	Graphite
Sleeve	Graphite
Lower reflector	Graphite
Internal trap	Graphite
Screen	Stainless steel
Brazing ring	Silicon
Bottom connector	Graphite
Instrumented bottom connector (instrumented elements only)	Graphite, stainless steel, Inconel
Thermocouples (instrumented elements only)	Inconel sheath, tungsten-rhenium, chromel-alumel Nb-1% Zr sheath
Test samples	Niobium canned, fission product release samples

[a]Source: Morissette 1986.

Table 4.3.2. Weights of Peach Bottom 1 fuel elements, fuel element components, and fuel element storage apparatus[a]

Item	Approximate wt, kg
Fuel elements	
Standard fuel element	41
Instrumented fuel element	41
Fuel test element (PTE designs)	45
Fuel test element (others)	41
Core 2 cut-off fuel element	38
Core 2 cut-off instrumented fuel element	38
Storage apparatus	
Core 1 fuel element with storage canister	68
Storage basket with core 1 fuel	1642
Fuel element components	
Upper reflector	6
Sleeve	13
Lower reflector	0.6
Internal trap	2
Bottom connector	3
Fuel compact assembly (each)	5[b]
Fuel compact	0.4
Fuel element materials	
Carbon (standard element)	33
Stainless steel	5
Uranium	c
Thorium	c
Rhodium	c
Boron	c
Silicon	15 g

[a]Source: Morissette 1986.

[b]Weight shown is for each assembly. There are three assemblies per element.

[c]These weights are shown in Table 4.3.3.

Table 4.3.3. Peach Bottom 1 fuel element initial metal loadings, g[a]

	Type 1		Type 2		Type 3		Type 4	
	Core		Core		Core		Core	
	1	2	1	2	1	2	1	2
Uranium, 93%	313	250	313	250	313	250	166	141
Thorium	1563	1374	1563	1374	1563	1374	3461	2598
Rhodium 103	18.5	18.5	6.16	6.16	6.16	6.16	0	0
Boron	0		0		18.3	18.3	0	0

[a]Source: Morissette 1986.

Table 4.3.4. Peach Bottom 1 burnup data for cores 1 and 2[a]

	Core 1	Core 2
EFPD[b]	451.5	897.4
MW(t) hours[c]	1,246,089	2,476,454
Shutdown date	October 3, 1969	October 31, 1974
Heavy metal loading	1,686.14 kg	1,418.6 kg
Burnup	30,795 MWd/MTHM	72,717 MWd/MTHM

[a]Source: Morissette 1986.
[b]Equivalent full-power days.
[c]Reactor core output 115 MW(t).

Table 4.3.5. Postirradiation heavy metal loadings: Peach Bottom 1 cores[a]

Nuclide	Mass per core, kg	
	Core 1	Core 2
Th-232	1,439.31	1,172.54
U-232	0.0015	0.0075
U-233	20.52	25.95
U-234	2.96	4.55
U-235	156.52	66.96
U-236	14.27	21.12
U-238	12.32	9.25
Pu-239	0.411	0.200
Pu-240	0.083	0.069
Pu-241	0.063	0.112
Pu-242	0.008	0.054
Np-237		1.625
Total U	206.59	127.83
Uranium assays		
U-235, wt %	0.7576	0.5238
U-233, wt %	0.0994	0.2030
U-232, ppm	7.08	58.55

[a]Based on Conti 1971.

Table 4.3.6. Radionuclide content of Peach Bottom spent fuel element[a]

Radionuclide	Activity, Ci
Kr-85	5.0000E+01
Sr-89	1.1700E+03
Sr-90	3.9300E+02
Y-90	3.9300E+02
Y-91	1.7300E+03
Zr-95	2.1000E+03
Nb-95	3.9800E+03
Ru-103	4.8000E+02
Rh-103m	4.8000E+02
Ru-106	4.0000E+02
Rh-106	4.0000E+02
Te-127m	2.2000E+01
Te-127	2.2000E+01
Te-129m	3.1000E+01
Te-129	3.1000E+01
Cs-137	6.0000E+02
Ba-137	6.0000E+02
Ba-140	1.8000E+00
La-140	2.0000E+00
Ce-141	6.3500E+02
Pr-143	1.9000E+01
Ce-144	5.0600E+03
Pr-144	5.0600E+03
Pm-147	1.5000E+03
Sm-151	1.3000E+01
Pa-233	2.2000E+04
U-233	4.2000E-01
U-234	4.6500E-02
Pu-238	9.5000E+00
Pu-239	2.6900E-02
Pu-240	2.3000E-02
Pu-241	2.0000E+01

[a]Basis: one Peach Bottom fuel element, 900 equivalent full-power days, 120-day cooled.
Source: INEL 1976. Exposure is equivalent to 73,000 MWd/MTIHM.

Table 4.3.7. Radioactivity of Peach Bottom-1 reactor spent fuel based on one fuel element; burnup is 73,000 MWd/MTIHM (curies per element)

	120.00	1.0YR	10.0YR	100.0YR	1000.0YR	10.0KY	100.0KY	1000.0KY
				Actinides and daughters				
TL209	0.000E+00	5.872E-07	8.484E-06	8.707E-05	8.356E-04	5.539E-03	6.294E-03	1.252E-04
PB209	0.000E+00	2.719E-05	3.928E-04	4.031E-03	3.868E-02	2.564E-01	2.914E-01	5.795E-03
PB210	0.000E+00	2.764E-13	7.880E-10	4.989E-07	8.303E-05	3.298E-03	2.547E-02	4.272E-03
PB214	0.000E+00	4.088E-11	8.484E-09	9.018E-07	8.305E-05	3.299E-03	2.548E-02	4.273E-03
BI210	0.000E+00	2.764E-13	7.881E-10	4.989E-07	8.303E-05	3.298E-03	2.547E-02	4.272E-03
BI213	0.000E+00	2.719E-05	3.928E-04	4.031E-03	3.868E-02	2.564E-01	2.914E-01	5.795E-03
BI214	0.000E+00	4.088E-11	8.484E-09	9.018E-07	8.305E-05	3.299E-03	2.548E-02	4.273E-03
PO210	0.000E+00	6.573E-14	7.881E-10	4.989E-07	8.303E-05	3.298E-03	2.547E-02	4.272E-03
PO213	0.000E+00	2.660E-05	3.843E-04	3.944E-03	3.785E-02	2.509E-01	2.851E-01	5.670E-03
PO214	0.000E+00	4.087E-11	8.482E-09	9.016E-07	8.303E-05	3.298E-03	2.547E-02	4.272E-03
PO218	0.000E+00	4.089E-11	8.486E-09	9.019E-07	8.307E-05	3.300E-03	2.548E-02	4.274E-03
AT217	0.000E+00	2.719E-05	3.928E-04	4.031E-03	3.868E-02	2.564E-01	2.914E-01	5.795E-03
RN222	0.000E+00	4.089E-11	8.486E-09	9.019E-07	8.307E-05	3.300E-03	2.548E-02	4.274E-03
FR221	0.000E+00	2.719E-05	3.928E-04	4.031E-03	3.868E-02	2.564E-01	2.914E-01	5.795E-03
RA225	0.000E+00	2.719E-05	3.928E-04	4.031E-03	3.868E-02	2.564E-01	2.914E-01	5.795E-03
RA226	0.000E+00	4.089E-11	8.486E-09	9.019E-07	8.307E-05	3.300E-03	2.548E-02	4.274E-03
AC225	0.000E+00	2.719E-05	3.928E-04	4.031E-03	3.868E-02	2.564E-01	2.914E-01	5.795E-03
TH229	0.000E+00	2.719E-05	3.928E-04	4.031E-03	3.868E-02	2.564E-01	2.914E-01	5.795E-03
TH230	0.000E+00	2.811E-07	4.060E-06	4.264E-05	4.427E-04	4.232E-03	2.526E-02	4.270E-03
PA233	2.200E+04	4.045E+01	4.175E-07	1.597E-05	1.066E-04	1.342E-04	1.303E-04	9.737E-05
U233	4.201E-01	4.303E-01	4.303E-01	4.301E-01	4.284E-01	4.119E-01	2.779E-01	5.531E-03
U234	4.651E-02	4.652E-02	4.676E-02	4.835E-02	4.977E-02	4.852E-02	3.760E-02	2.931E-03
U236	0.000E+00	4.572E-10	6.583E-09	6.752E-08	6.459E-07	4.197E-06	6.405E-06	6.237E-06
U237	0.000E+00	4.751E-04	3.080E-04	4.046E-06	6.187E-25	0.000E+00	0.000E+00	0.000E+00
NP237	0.000E+00	2.417E-09	4.175E-07	1.597E-05	1.066E-04	1.342E-04	1.303E-04	9.737E-05
PU238	9.501E+00	9.451E+00	8.802E+00	4.323E+00	3.533E-03	0.000E+00	0.000E+00	0.000E+00
PU239	2.690E-02	2.690E-02	2.690E-02	2.683E-02	2.614E-02	2.017E-02	1.510E-03	8.322E-15
PU240	2.300E-02	2.300E-02	2.298E-02	2.276E-02	2.069E-02	7.967E-03	5.716E-07	0.000E+00
PU241	2.000E+01	1.937E+01	1.256E+01	1.649E-01	2.522E-20	0.000E+00	0.000E+00	0.000E+00
AM241	0.000E+00	2.118E-02	2.460E-01	5.818E-01	1.387E-01	7.482E-08	0.000E+00	0.000E+00
Subtotal	2.203E+04	6.982E+01	2.214E+01	5.631E+00	9.782E-01	2.574E+00	2.903E+00	9.775E-02
				Fission products				
KR 85	5.000E+01	4.788E+01	2.676E+01	7.946E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 89	1.170E+03	4.039E+01	1.023E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	3.931E+02	3.868E+02	3.122E+02	3.666E+01	1.823E-08	0.000E+00	0.000E+00	0.000E+00
Y 90	3.931E+02	3.869E+02	3.123E+02	3.666E+01	1.823E-08	0.000E+00	0.000E+00	0.000E+00
Y 91	1.730E+03	9.468E+01	1.155E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 95	2.100E+03	1.474E+02	5.035E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95	3.980E+03	3.216E+02	1.118E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	0.000E+00	1.093E+00	3.735E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU103	4.801E+02	6.337E+00	4.083E-25	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH103M	4.801E+02	5.713E+00	3.681E-25	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	4.001E+02	2.521E+02	5.174E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	4.001E+02	2.521E+02	5.174E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE127	2.200E+01	4.531E+00	3.785E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE127M	2.200E+01	4.626E+00	3.864E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129	3.101E+01	1.281E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	3.100E+01	1.968E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	0.000E+00	1.808E-07	1.819E-07	1.819E-07	1.819E-07	1.819E-07	1.811E-07	1.741E-07
CS137	6.001E+02	5.908E+02	4.799E+02	5.999E+01	5.583E-08	0.000E+00	0.000E+00	0.000E+00
BA137M	6.001E+02	5.589E+02	4.540E+02	5.675E+01	5.282E-08	0.000E+00	0.000E+00	0.000E+00
BA140	1.800E+00	3.039E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	2.000E+00	3.497E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	6.351E+02	3.404E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR143	1.900E+01	6.858E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE144	5.061E+03	2.783E+03	9.190E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	5.061E+03	2.783E+03	9.190E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144M	0.000E+00	3.340E+01	1.103E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM147	1.500E+03	1.256E+03	1.165E+02	5.486E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM147	0.000E+00	5.979E-09	3.392E-08	3.678E-08	3.678E-08	3.678E-08	3.678E-08	3.678E-08
SM151	1.300E+01	1.293E+01	1.207E+01	6.034E+00	5.890E-03	0.000E+00	0.000E+00	0.000E+00
Subtotal	2.518E+04	9.974E+03	1.717E+03	1.962E+02	5.890E-03	2.186E-07	2.179E-07	2.109E-07

Table 4.3.8. Decay heat of Peach Bottom-1 reactor spent fuel based on one fuel element; burnup is 73,000 MWd/MTIHM (watts per element)

	120.0D	1.0YR	10.0YR	100.0YR	1000.0YR	10.0KY	100.0KY	1000.0KY
				Actinides and daughters				
TL209	0.000E+00	9.757E-09	1.410E-07	1.447E-06	1.388E-05	9.203E-05	1.046E-04	2.080E-06
PB209	0.000E+00	3.126E-08	4.517E-07	4.635E-06	4.449E-05	2.949E-04	3.351E-04	6.664E-06
PB210	0.000E+00	6.404E-17	1.825E-13	1.156E-10	1.923E-08	7.641E-07	5.901E-06	9.897E-07
PB214	0.000E+00	1.304E-13	2.706E-11	2.876E-09	2.649E-07	1.052E-05	8.125E-05	1.363E-05
BI210	0.000E+00	6.374E-16	1.817E-12	1.150E-09	1.915E-07	7.605E-06	5.874E-05	9.851E-06
BI211	0.000E+00	3.572E-20	9.967E-17	6.212E-14	1.099E-11	9.483E-10	2.573E-08	3.666E-08
BI213	0.000E+00	1.143E-07	1.651E-06	1.695E-05	1.626E-04	1.078E-03	1.225E-03	2.436E-05
BI214	0.000E+00	5.239E-13	1.087E-10	1.156E-08	1.064E-06	4.228E-05	3.265E-04	5.476E-05
PO210	0.000E+00	2.107E-15	2.526E-11	1.599E-08	2.662E-06	1.057E-04	8.166E-04	1.370E-04
PO213	0.000E+00	1.346E-06	1.945E-05	1.996E-04	1.915E-03	1.270E-02	1.443E-02	2.869E-04
PO214	0.000E+00	1.898E-12	3.938E-10	4.186E-08	3.855E-06	1.531E-04	1.183E-03	1.984E-04
PO218	0.000E+00	1.482E-12	3.075E-10	3.268E-08	3.010E-06	1.196E-04	9.234E-04	1.549E-04
AT217	0.000E+00	1.160E-06	1.676E-05	1.720E-04	1.651E-03	1.094E-02	1.243E-02	2.473E-04
RN222	0.000E+00	1.355E-12	2.812E-10	2.989E-08	2.752E-06	1.093E-04	8.444E-04	1.416E-04
FR221	0.000E+00	1.049E-06	1.516E-05	1.556E-04	1.493E-03	9.897E-03	1.125E-02	2.237E-04
RA225	0.000E+00	1.906E-08	2.754E-07	2.827E-06	2.713E-05	1.798E-04	2.043E-04	4.064E-06
RA226	0.000E+00	1.181E-12	2.450E-10	2.604E-08	2.398E-06	9.527E-05	7.358E-04	1.234E-04
AC225	0.000E+00	9.496E-07	1.372E-05	1.408E-04	1.351E-03	8.958E-03	1.018E-02	2.024E-04
TH229	0.000E+00	8.317E-07	1.202E-05	1.233E-04	1.183E-03	7.845E-03	8.914E-03	1.773E-04
TH230	0.000E+00	7.956E-09	1.149E-07	1.207E-06	1.253E-05	1.198E-04	7.147E-04	1.208E-04
PA233	4.994E+01	9.182E-02	9.477E-10	3.625E-08	2.418E-07	3.046E-07	2.958E-07	2.210E-07
U233	1.221E-02	1.251E-02	1.251E-02	1.250E-02	1.245E-02	1.197E-02	8.079E-03	1.608E-04
U234	1.340E-03	1.340E-03	1.347E-03	1.393E-03	1.434E-03	1.398E-03	1.083E-03	8.443E-05
U235	0.000E+00	4.659E-13	6.710E-12	6.906E-11	6.838E-10	6.029E-09	2.274E-08	2.407E-08
U236	0.000E+00	1.239E-11	1.783E-10	1.829E-09	1.750E-08	1.137E-07	1.735E-07	1.690E-07
NP237	0.000E+00	7.388E-11	1.276E-08	4.881E-07	3.257E-06	4.101E-06	3.983E-06	2.976E-06
PU238	3.149E-01	3.132E-01	2.917E-01	1.433E-01	1.171E-04	1.555E-35	0.000E+00	0.000E+00
PU239	8.291E-04	8.291E-04	8.289E-04	8.267E-04	8.056E-04	6.216E-04	4.652E-05	2.565E-16
PU240	7.163E-04	7.162E-04	7.155E-04	7.087E-04	6.442E-04	2.481E-04	1.780E-08	0.000E+00
PU241	6.201E-04	6.004E-04	3.893E-04	5.113E-06	7.819E-25	0.000E+00	0.000E+00	0.000E+00
AM241	0.000E+00	7.036E-04	8.171E-03	1.933E-02	4.608E-03	2.485E-09	0.000E+00	0.000E+00
Subtotal	5.027E+01	4.217E-01	3.158E-01	1.789E-01	2.794E-02	6.699E-02	7.397E-02	2.379E-03
				Fission products				
KR 85	7.490E-02	7.172E-02	4.008E-02	1.190E-04	6.370E-30	0.000E+00	0.000E+00	0.000E+00
SR 89	4.045E+00	1.396E-01	3.537E-21	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	4.562E-01	4.490E-01	3.624E-01	4.254E-02	2.116E-11	0.000E+00	0.000E+00	0.000E+00
Y 90	2.178E+00	2.144E+00	1.731E+00	2.032E-01	1.011E-10	0.000E+00	0.000E+00	0.000E+00
Y 91	6.214E+00	3.400E-01	4.149E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 95	1.064E+01	7.464E-01	2.550E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95	1.909E+01	1.543E+00	5.362E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	0.000E+00	1.519E-03	5.190E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU103	1.606E+00	2.120E-02	1.366E-27	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH103M	1.105E-01	1.315E-03	8.472E-29	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	2.379E-02	1.499E-02	3.076E-05	4.070E-32	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	3.837E+00	2.418E+00	4.962E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE127	2.971E-02	6.118E-03	5.111E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE127M	1.184E-02	2.488E-03	2.078E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129	1.108E-01	4.578E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	5.436E-02	3.451E-04	1.220E-33	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	0.000E+00	8.363E-11	8.416E-11	8.416E-11	8.416E-11	8.413E-11	8.379E-11	8.053E-11
CS137	6.638E-01	6.535E-01	5.308E-01	6.635E-02	6.175E-11	0.000E+00	0.000E+00	0.000E+00
BA137M	2.356E+00	2.195E+00	1.783E+00	2.228E-01	2.074E-10	0.000E+00	0.000E+00	0.000E+00
BA140	5.023E-03	8.479E-09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	3.353E-02	5.863E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	9.298E-01	4.984E-03	1.821E-33	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR143	3.540E-02	1.278E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE144	3.357E+00	1.846E+00	6.096E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	3.720E+01	2.046E+01	6.755E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144M	0.000E+00	1.143E-02	3.773E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM147	5.381E-01	4.506E-01	4.179E-02	1.968E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM147	0.000E+00	8.187E-11	4.645E-10	5.036E-10	5.036E-10	5.036E-10	5.036E-10	5.036E-10
SM151	1.524E-03	1.517E-03	1.415E-03	7.075E-04	6.906E-07	5.423E-37	0.000E+00	0.000E+00
Subtotal	9.360E+01	3.352E+01	4.502E+00	5.357E-01	6.916E-07	5.877E-10	5.874E-10	5.841E-10

Table 4.3.9. Peach Bottom 1 reactor: summary of spent fuel quantities currently in inventory[a]

Source	No. of fuel elements	MTIHM[b]
Core 1	813	1.686
Core 2	785	1.419
Total	1598	3.105

[a]At INEL. There are also 10 elements in retrievable storage at ORNL, giving a total of 1608 elements currently in storage. Two elements were destroyed during examination and 29 are missing. This accounts for the total number of elements (1639) in both cores.

[b]Source: Morissette 1986.

4.4 RESEARCH AND TEST REACTOR FUELS

4.4.1 Introduction

All fuels that have been irradiated in a reactor are covered under the Nuclear Waste Policy Act, which requires that all non-defense spent nuclear fuels be disposed of in a geologic repository. The research and test reactors discussed in this section use a wide variety of different types of fuel, which require different handling, treatment, and disposal procedures. It is expected that some of the spent fuels from research reactors will be reprocessed at government-owned plants such as SRP or INEL; if so, the resulting high-level wastes from these operations will go to a repository as part of canistered defense high-level waste. For some of the fuels, a final decision as to whether to reprocess or not reprocess has not yet been made. Those fuels that are not reprocessed will require some type of packaging for repository disposal or MRS storage.

The research and test reactors listed here are divided into the following four categories: (1) reactors owned by commercial and industrial firms and used for private research and test purposes, (2) reactors owned and operated by U.S. government agencies other than DOE, (3) reactors owned and operated by DOE, and (4) reactors used for educational purposes and research at universities and other educational institutions. The objective of this section is to characterize the various fuels used in these reactors with regard to quantity, dimensions, activity level, and other characteristics (such as chemical composition or cladding material) that might affect their disposal.

A further classification of reactors in accordance with fuel type has been used in this report because of the importance of fuel characteristics in planning for fuel disposal. Fuels have been categorized into the following eight types:

1. plate-type U-Al alloy fuels, including flat plates, curved plates, and concentric tubes; enrichments are typically either 93% or 19.8%;
2. TRIGA (uranium-zirconium hydride) fuels, stainless steel, aluminum, or Incoloy-800 clad, predominantly 19.8% enriched, although some are 70% enriched;
3. homogeneous UO_2-polyethylene disks or blocks, 19.8% enriched;
4. low-enriched (4 to 6%) UO_2 pin-array types, such as PULSTAR;
5. aqueous solution liquid fuels;
6. uranium-molybdenum alloy highly enriched (~93%) fuels;
7. UO_2-PuO_2 mixed oxide fuels; and
8. other types.

4.4.2 Organization of Data

Table 4.4.1 lists the number of operational research and test reactors in each of the four ownership categories. There are currently 84 operational reactors: 5 privately-owned, 7 non-DOE government-owned, 37 DOE-owned, and 35 at educational institutions. Two of the DOE-owned reactors are on standby; that is, they are shut down temporarily but could be restarted if desired. Reactors that have been permanently shut down are not included in this table. In most cases, the fuel from shut-down non-DOE reactors has been returned to DOE.

Table 4.4.2 shows the number of research and test reactors of each fuel type, separated according to ownership category. The two types of fuel most widely used are the enriched U-Al plate or concentric tube type and the TRIGA uranium-zirconium hydride type. Again, reactors that have been permanently shut down are not included.

Table 4.4.3 lists the 5 private commercial research and test reactors that are currently operational and gives the types of fuel and other details. Table 4.4.4 lists 6 private commercial research and test reactors that are no longer in operation.

Table 4.4.5 lists the 7 research and test reactors owned and operated by non-DOE government agencies. Table 4.4.6 gives fuel data for these reactors.

Table 4.4.7 summarizes the 37 operational research and test reactors owned and operated by DOE. This table is arranged according to site location for 10 individual sites and gives the power level and fuel category of each reactor. Two of these reactors are in a standby condition; these are the Power Burst Facility at Idaho National Engineering Laboratory and the Bulk Shielding Reactor at Oak Ridge National Laboratory. Five reactors that have been permanently shut down are listed in the footnotes to the table. In addition, there are 5 critical assemblies at Los Alamos National Laboratory that are frames only (not fueled until an experiment is assembled) and are not counted here as reactors. These critical assembly frames are COMET, HONEYCOMB, MARS, PLANET, and VENUS.

Tables 4.4.8 through 4.4.17 give additional information on the DOE reactors and their fuels. These tables are arranged by site location. Some reactors that have been shut down or are on standby are included. Because of security considerations, no information was obtained on the four naval research reactors at Knolls Atomic Power Laboratory, so no table is included for that site.

Table 4.4.18 summarizes the 35 university/educational reactors that are currently operational. These reactors are separated according to fuel type in Tables 4.4.19–4.4.22, which give more detailed information on the individual reactors. Table 4.4.23 lists 26 educational reactors that have been shut down within the last 20 years.

Additional information on research and test reactors can be found in IAEA 1960, Burn and Bilof 1983, Burn 1988, IAEA 1989, and OSTI 1990.

Table 4.4.24 summarizes the estimated total quantities of fuel that will ultimately require disposal, calculated as of the end of the year 2020. This includes discharged fuel and fuel in reactors at that time. These quantities were estimated by extrapolating operating plans as of the year 1991, and therefore must be considered subject to future changes.

Radioactivity (curies) and thermal power (watts) per fuel element were estimated using ORIGEN2 (Croff 1980) for fuel types for which sufficient information was available. Calculations were made for decay times ranging from 0 to 10^6 years after discharge. These estimates are shown in Tables 4.4.25 through 4.4.27.

Table 4.4.28 gives physical data on TRIGA fuel. A summary of projected Fast Flux Test Facility (FFTF) fuel discharges is given in Table 4.4.29. Additional data on the FFTF are given in Appendix 4E.

4.4.3 Fuel Quantities and Characteristics

Fuel quantities and characteristics are discussed in the following subsections, 4.4.3.1–4.4.3.8. These subsections are arranged by fuel type and are in the same order as the list of fuel type categories given in Table 4.4.2.

4.4.3.1 U-Al Plate-Type or Concentric-Tube Fuels

4.4.3.1.1 Physical characteristics and dimensions

This fuel category consists of enriched plate-type fuels (MTR, Argonaut, etc.), fabricated of uranium-aluminum alloy fuel-bearing plates with aluminum cladding, and arranged in various geometrical configurations.

The Materials Testing Reactor (MTR), located at INEL, was the original source of much of this fuel. The MTR reached initial criticality in 1952, was operated as a test reactor until its final shutdown in 1970, and was decommissioned in 1974. It used a highly-enriched (93%) U-Al alloy fuel in the form of assemblies of curved plates. After the shutdown of the MTR, its fuel was widely distributed to a number of research and test reactors throughout the U.S. A typical MTR plate-type element (see Fig. 4.4.1) consists of a number of individual plates. An element typically contains about 0.15 to 0.18 kg of ^{235}U. Typical fuel element dimensions (cm) are about 7.6 × 7.6 × 90; however, both cross-sections and lengths may vary. Some of the reactors in this category use flat plates rather than curved plates. The High Flux Isotope Reactor (Table 4.4.14) uses two concentric annular fuel elements in which the annuli contain parallel curved fuel plates (Fig. 4.4.2).

In accordance with a policy statement issued by the Nuclear Regulatory Commission (NRC 1982), research reactor fuels containing highly-enriched uranium (HEU) are being replaced with low-enriched uranium (LEU) fuels where possible. LEU fuels have less than 20% enrichment; typically they have enrichments of about 19.8%. A few exceptions have been granted where physical limitations make such a change impracticable. For example, the Massachusetts Institute of Technology 5 MW reactor (Table 4.4.19) has been granted an exception because technological limits would prevent the reactor from reaching criticality with <20%-enriched fuel. The University of Missouri 10 MW reactor at Columbia has also been granted an exception for reasons related to neutronic performance. A number of reactors have already made the changeover to <20%-enriched fuel; among these are Ohio State University, University of Michigan, Manhattan College, and Worcester Polytechnic Institute. In this report, the LEU plate-type fuel is considered part of fuel category 1, since it is used in the same reactors as the HEU plate-type fuel and has similar disposal options.

The Manhattan College reactor uses a cylindrical fuel element configuration. Each element contains 6 concentric cylindrical shells; each shell consists of three 120-degree sections that run the length of the element.

The Westinghouse training reactor at Zion, Illinois, which has been shut down, also had concentric cylinder fuel elements, using 93% enriched uranium-aluminum alloy fuel with aluminum cladding. The reactor contained 24 elements, each of which consisted of three concentric annular cylinders about 4 mm thick, with an overall outside diameter of about 6.5 cm and an overall length of about 1.0 m.

The MIT reactor uses 24 vertical fuel elements about 66.7 cm in height. The cross-section of each element is a regular 60–120 degree parallelogram, each side of which is about 6.98 cm in length. Each element contains 15 parallel flat plates, about 6.42 cm wide by 58.4 cm long by 0.2 cm thick, placed parallel to two opposite faces of the parallelogram.

4.4.3.1.2 Quantity of fuel requiring disposal

Fuel of this general category is used by 1 commercial research reactor (Table 4.4.3), 2 government-owned non-DOE reactors (Tables 4.4.5-6), 12 DOE-owned reactors (Table 4.4.7), and 13 university/educational reactors (Table 4.4.19). The in-core fuel in the 13 university reactors amounts to about 62 kg of ^{235}U; the General Electric NTR reactor contains about 4 kg, and the 14 operating and 3 shut-down government-owned DOE and non-DOE reactors contain about 127 kg of ^{235}U. The total quantity of ^{235}U in the cores of all these reactors is about 193 kg. The additional quantity required for the refueling of operational reactors through the year 2020 is estimated to be about 15,000 to 30,000 kg of ^{235}U.

4.4.3.1.3 Treatment and disposal options

Fuels of this type have been reprocessed on a regular basis at INEL and SRS, and it is expected that this arrangement will continue. It appears likely that the reprocessing wastes from these fuels will continue to be part of the defense HLW, and their disposal will be handled via immobilization and shipment to a repository, as discussed in Sect. 3 of this report.

The number of canisters required for immobilized HLW from research and test reactor fuels is very small. Over a 30-year period, 15,000–30,000 kg (total uranium) of this type of fuel represents about 0.6–1.2 m^3 of HLW, based on 17 kg of waste per metric ton of uranium processed (White 1986) and assuming 1,650 kg of glass containing 25% waste solids per m^3. To provide a more conservative estimate, the amount of waste per metric ton of uranium processed was increased by a factor of 3 to account for the higher enrichment of the fuel. This increases the quantity of glass to about 2–4 m^3, which is less than 7 canisters. This indicates that the quantity of high-level waste generated by the reprocessing of research and test reactor U-Al plate-type fuels is so small compared with defense HLW that the quantity projections for the latter will require little or no adjustment for research reactor fuels.

4.4.3.2 TRIGA Fuels

4.4.3.2.1 Physical characteristics and dimensions

TRIGA reactors utilize uranium-zirconium hydride fuels clad with aluminum, stainless steel, or Incoloy-800. The atomic ratio of hydrogen to zirconium in the fuel is about 1.6. Stainless steel is now the standard cladding material. Both the aluminum-clad and the stainless-steel-clad elements are 3.8-cm-diam by 76-cm-long rods, including the end fittings. The elements clad with Incoloy 800 are of the same length but have a smaller diameter (1.37 cm). Fuel follower control rod elements are 3.8-cm-diam rods and range from 114.3 to 168.9 cm long. Physical configuration and dimensions of aluminum-clad and stainless-steel-clad elements are shown in Figs. 4.4.3 and 4.4.4 and Table 4.4.28 (Tomsio 1986).

The most common ^{235}U enrichment in TRIGA fuels is about 19.8%, although some 70% enriched fuels are in use. The latter are being replaced in accordance with the NRC policy statement mentioned earlier (NRC 1982).

4.4.3.2.2 Quantity of fuel requiring disposal

The United States has 31 TRIGA reactors, of which about 24 are still in operation. The number of fuel elements in a reactor varies from 60 to 100, depending on reactor size. A good average is probably about 80 elements per reactor. At present, there are about 3,000 fuel elements in reactors, or stored as spares at reactors, plus approximately 900 spent fuel elements. About 650 of the latter are at INEL, and about 240 are in storage at the various reactor sites. A few TRIGA reactors still contain 70%-enriched fuel, and it is planned to replace this with 19.8%-enriched fuel in all but a very few reactors. The exceptions will be those that require high-enriched fuel for special reasons. Refueling for 30 years will require an additional number of elements, estimated to be about 600. The total number of fuel elements requiring disposal is estimated to be between 2,500 and 4,500 by the year 2020.

4.4.3.2.3 Treatment and disposal options

The feasibility of reprocessing TRIGA fuels at SRS or INEL remains to be evaluated. The presence of hydride in the fuel may be found to cause difficulties in processing. Hanford has no plans for reprocessing such fuels (White 1986). At this time, there is a strong possibility that this type of fuel will be considered for disposal in packaged form. However, the presence of graphite parts in the fuel elements may conflict with repository waste acceptance criteria. The presence of hydrides of uranium and zirconium in the fuel may also be of concern.

4.4.3.2.4 Radioactivity and thermal power

Spent fuel burnups for aluminum-clad and stainless-steel-clad elements range typically from 10 to 20% (Tomsio 1986). Using a ^{235}U enrichment of 20% for standard TRIGA fuel, these figures correspond to burnups of 20,000 to 40,000 MWd per metric ton of uranium. In the absence of specific TRIGA information, radioactivity (curies) and thermal power (watts) were estimated by using data for PWR fuel at a burnup of 33,000 MWd per metric ton of uranium (Roddy et al. 1986). The results, based on a standard TRIGA spent fuel element containing 195 g (total) of uranium, are shown in Table 4.4.25 for decay times ranging from 10 days to 10^6 years. For example, at a decay time of 1 year, a package containing 120 fuel elements would have a radioactivity of about 54,000 Ci and a thermal power of about 230 W.

4.4.3.3 Homogeneous UO_2-Polyethylene Fuels

4.4.3.3.1 Physical characteristics and dimensions

Fuel in the third category consists of homogeneous UO_2-polyethylene material shaped into disks (four reactors) or blocks (one reactor). Each of the four reactors using disk-type fuel has nine fuel disks with diameters of 25.75 cm and thicknesses of 3.9 cm (4 disks), 2.3 cm (3 disks), and 1.0 cm (2 disks). The reactor using block-type fuel has 12 fuel elements, each containing 2 fuel blocks with dimensions of 7.3 × 7.78 × 12.7 cm. Enrichment in each

case is 19.9% ^{235}U, and the density of ^{235}U dispersed in the polyethylene matrix is 0.057 g/cm^3.

4.4.3.3.2 Quantity of fuel requiring disposal

The five reactors in this category (this includes shutdown reactors) contain 55 fuel disks (4.23 kg of ^{235}U, total) and 24 fuel blocks (0.81 kg of ^{235}U, total), giving a total ^{235}U inventory of 5.04 kg. Burnup and refueling requirements are negligible. Hence, the total amount of ^{235}U to be disposed of is about 5.04 kg. The total volume of polyethylene matrix in which this ^{235}U is dispersed is about 0.1 m^3.

4.4.3.3.3 Treatment and disposal options

The desirability and cost-effectiveness of reprocessing UO_2-polyethylene fuels remain to be evaluated. Presumably, chemical reprocessing would first involve removal of the polyethylene, which could require further process studies. Disposal of the fuel in a repository without first removing the polyethylene may or may not be allowable under future criteria. To date, other disposal options have not been developed. As far as we know, no assumption can be made at this time regarding the probable mode of treatment or disposal of this type of fuel.

4.4.3.3.4 Radioactivity and thermal power

Radioactivity and thermal power per kilogram of discharged UO_2-polyethylene fuel were estimated using ORIGEN2 (Croff 1980), based on an estimated maximum burnup of 300 MWd per metric ton of uranium. Most of the educational reactors using this type of fuel will accumulate much less exposure than this during their lifetime. Using the 300-MWd/MTU figure, radioactivity and thermal power were ratioed from PWR data at a burnup of 33,000 MWd/MTU. The results, shown in Table 4.4.26, are on a per-kilogram-of-uranium basis rather than on a fuel-element basis because the size of a fuel element varies. A UO_2-polyethylene fuel disk with a diameter of 25.75 cm and a thickness of 4.0 cm contains about 0.6 kg of uranium. The total fuel to be disposed of was estimated to be about 28 kg of uranium. Thus, the total radioactivity of all the fuel at discharge would be about 28 × 1.5E+3, or 42,000 Ci, and the total thermal power would be about 28 × 6.5E+0, or 183 W.

4.4.3.4 Low-Enriched Pin-Type Fuels

4.4.3.4.1 Physical characteristics and dimensions

As shown in Table 4.4.1, four educational reactors are using fuel in this category. These reactors are listed in Table 4.4.18. The two at North Carolina State and the State University of New York are of the PULSTAR type using low-enriched (4% and 6%) UO_2 fuels with Zircaloy-2 cladding. The reactors contain 25 and 32 fuel elements, respectively; each element consists of 25 fuel pins in a 5 × 5 square array. The pins have a diameter of about 1.2 cm and a length of about 66 cm. Each 5 × 5 array is enclosed in a Zircaloy-2 box with outside dimensions of 7.0 × 8.0 × 82 cm. The overall length of the fuel element, including end fittings, is 96.5 cm (Orlosky 1986; Miller 1986).

The Cornell University zero-power reactor contains 815 fuel elements, each consisting of an aluminum-clad cylindrical pin with an outside diameter of 1.69 cm and an overall length of from 150 to 158 cm, including end fittings. The fuel consists of UO_2 pellets with an enrichment is 2.1% (Aderhold 1986).

Table 4.4.4 shows a number of private research and test reactors that have been shut down. The first reactor in this table was a Babcock and Wilcox critical assembly that used low-enriched UO_2 pin-type fuel elements. Two types of fuel elements were used; these were a 2.5%-enriched aluminum-clad type and a 4.0%-enriched stainless-steel-clad type. The fuel from this reactor has been shipped to Savannah River Site.

The DOE-owned Loss of Fluid Test (LOFT) Reactor at INEL used 4% enriched UO_2 pellets in Zircaloy-4 pins. This reactor has been shut down.

4.4.3.4.2 Quantity of fuel requiring disposal

The North Carolina State and State University of New York reactors contain 57 fuel elements, which represent a total of 30.9 kg of ^{235}U, or about 620 kg of uranium. The Cornell zero-power reactor contains 815 fuel elements, which represent a total of 35 kg of ^{235}U, or about 1,670 kg of uranium. The estimated refueling requirements through the year 2020 are 25 elements for the North Carolina State reactor, 64 elements for the State University of New York reactor, and 10 elements for the Cornell zero-power reactor. Therefore, the total number of elements to be disposed of, including those in reactor cores in the year 2020, is about 146 PULSTAR 5 × 5 elements (80 kg ^{235}U, or about 1,600 kg of uranium) and 825 Cornell ZPR fuel pins (35.4 kg of ^{235}U, or about 1,700 kg of uranium).

4.4.3.4.3 Treatment and disposal options

The PULSTAR fuel assemblies are very similar in composition and cladding to commercial LWR power reactor fuels. In view of this, it may be reasonable to assume that the approach to their disposal will be similar to that used for LWR spent fuels. The Cornell ZPR fuel pins could also be disposed of in a similar manner. Fuel from the shutdown Babcock & Wilcox reactors has been sent to SRP for reprocessing; the DOE-owned LOFT reactor fuel is reprocessed at INEL. Thus, the only fuel elements in

this category requiring repository disposal are the PULSTAR and Cornell ZPR fuels.

4.4.3.4.5 Radioactivity and thermal power

The North Carolina State PULSTAR reactor contains 12.7 kg of ^{235}U, or about 318 kg of uranium, and accumulates an exposure of about 32 MWd per year of operation (Burn and Bilof 1983). This corresponds to an average burnup of 100 MWd per metric ton of uranium per year; and, assuming 40 years of operation on the same core, the average burnup would be about 4,000 MWd per metric ton of uranium. The State University of New York PULSTAR spent fuel has a somewhat higher estimated burnup, ranging from an average of 7,000 MWd/MTU to a peak of 15,000 MWd/MTU. The burnup of the Cornell zero-power reactor fuel is negligible; based on its full power of 100 W and a total of 35 kg of uranium in the core, an exposure of 40 years would amount to only 42 MWd/MTU. Using the highest burnup of these three reactors (15,000 MWd/MTU), the radioactivity and thermal power of spent PULSTAR fuel per kilogram of uranium were estimated by ratioing to PWR data at 33,000-MWd/MTU burnup. The results are shown in Table 4.4.27 for decay times ranging from 0 to 10^6 years. At a decay time of 1 year, the radioactivity is 2.3E+3 Ci/kg U and the thermal power is 1.0E+1 W/kg U. The uranium content of a canister containing 24 PULSTAR 5 × 5 fuel elements, for example, would be about 260 kg; thus, its radioactivity and thermal power at a decay time of 1 year would be about 598,000 Ci and 2600 W, respectively.

4.4.3.5 Aqueous Liquid Fuels

4.4.3.5.1 Physical characteristics and dimensions

The fifth category consists of aqueous liquid fuels. There are no operating reactors in this category. The two shut-down educational reactors in this category used solutions of uranyl sulfate in water; the enrichments are 20% in one case and 89% in the other. The volume of solution is about 60 L combined. The shut-down Solution High Energy Burst Assembly (SHEBA) reactor at Los Alamos listed in Table 4.4.13 used about 80 L of a 4.95% enriched uranyl fluoride solution. The fuel solution of this reactor has been saved for possible future use.

4.4.3.5.2 Quantity of fuel requiring disposal

The combined volume of aqueous solutions for the three reactors is about 140 L. The total quantity of ^{235}U is about 2.7 kg.

4.4.3.5.3 Treatment and disposal options

In view of the very small volume involved, it would seem feasible to blend these liquids into defense HLW reprocessing tanks. If necessary, the fuel solution could be treated and then vitrified along with defense HLW. If this is done, the increase in vitrified defense HLW would be less than 1 canister. Because the need for and feasibility of this require further evaluation, no specific disposal assumption is made here for aqueous liquid fuels.

4.4.3.6 Uranium-Molybdenum Alloy Fuels

Six government-owned reactors use fuels consisting of highly enriched (93.2%) uranium-molybdenum alloy clad with nickel or aluminum. These reactors include the DOE-owned Sandia Pulsed Reactor II, Sandia Pulsed Reactor III, the Health Physics Test Reactor at Oak Ridge National Laboratory, the Los Alamos Fast Burst Research Reactor (Table 4.4.13), and the two U.S. Army-owned 10 kW fast-burst test reactors at Aberdeen Proving Grounds and White Sands Missile Range (Table 4.4.5). The two latter reactors are identical to the Oak Ridge Health Physics Reactor.

4.4.3.6.1 Physical characteristics and dimensions

The Oak Ridge National Laboratory Health Physics Research Reactor is typical of these reactors. It has disc-type fuel elements (about 25 cm diameter) into which a plunger is inserted when a pulse is desired. The Sandia Pulsed Reactor II uses 10 elements per core, each 3.47 cm × 20.5 cm, with a total core loading of 87 kg of U-235. Sandia Pulsed Reactor III uses 18 elements per core, each 2.26 cm × 29.7 cm, with a total core loading of 215.9 kg of U-235. Fuel burnup is small, and no refueling of either reactor is anticipated.

4.4.3.6.2 Quantity of fuel requiring disposal

The quantity of fuel burned up is negligible, and generally little or no fuel is removed from these reactors during their lifetime.

4.4.3.6.3 Treatment and disposal options

Because this fuel is highly enriched, it is assumed that it will eventually be reprocessed in DOE facilities and that the resulting waste will be included with defense high-level waste.

4.4.3.7 UO_2-PuO_2 Mixed Oxide Fuel

4.4.3.7.1 Physical characteristics and dimensions

The only reactor in this category is the DOE-owned Fast Flux Test Facility (FFTF) at Hanford. This is a sodium-cooled 400 MW(t) fast reactor fueled with mixed oxide UO_2-PuO_2 pellets contained in 0.58-cm diameter stainless-steel-clad fuel pins. The pins are assembled into hexagonal assemblies, 366 cm in length, each containing 217 pins. Average plutonium content of the core fuel is about 25-27%. A more detailed discussion of this reactor and its fuel is given in Appendix 4E.

4.4.3.7.2 Quantity of FFTF fuel requiring disposal

The quantity of fuel discharged, and that projected for future years, is summarized in Table 4.4.29. Some of this fuel has been packaged into 104-cm long cylinders (see Table 4.5.6). At present (January 1991), it is assumed that irradiated FFTF fuel assemblies will accumulate and be stored on-site. No reprocessing of irradiated driver or test mixed-oxide fuel is projected through the period 1991-2007 because reprocessing capability is not planned to exist during this period (Honeyman 1991).

4.4.3.8 Other Fuel Types (Fuel Category 8)

4.4.3.8.1 Physical characteristics and dimensions

There are 4 reactors at ANL West in fuel category 8 (EBR-II, Fast Source, Transient Test, and Zero Power Physics reactors). The fuel characteristics of these reactors are described in Table 4.4.9. The INEL Power Burst Facility (now on standby) is described in Table 4.4.12. The Los Alamos Big Ten critical facility has a base fuel load of about one metric ton of 10% enriched uranium metal, to which other experimental fuel components may be added as desired (Table 4.4.13). Rocky Flats has four critical assemblies in fuel category 8. Two of these (Solution System and Tank Reservoir) are capable of using aqueous liquid fuel. These critical assemblies generate no spent fuel; various experiments can be mocked up. No specific dimensions or characteristics can be ascribed to these fuels.

The Sandia Annular Core Research Reactor is a pulsed reactor with an average power of about 15 kW and pulses of much higher power. Fuel is UO_2 pellets with an enrichment of 35%. Fuel burnup is negligible. Eventually, removal of a full core would discharge about 70 kg of heavy metal (236 fuel elements). Elements are approximately 3.8 cm diameter × 51 cm long.

4.4.3.8.2 Quantity of fuel requiring disposal

It is assumed that the EBR-II fuel will continue to be reprocessed. The other three reactors at ANL West in fuel category 8 (Fast Source, Transient Test, and Zero Power Physics reactors) use no refueling, and it is assumed that the final discharges from these reactors will also be reprocessed. The INEL Power Burst Facility has pin-type oxide fuel of 18.5% enrichment; this reactor also uses no refueling. At final discharge, its core load of 68 elements (104 kg U-235) could either be reprocessed or packaged for repository disposal.

The five critical facilities in fuel category 8 (the Los Alamos Big Ten and the four at Rocky Flats), which have no refueling requirement, probably will produce little or no spent fuel requiring repository disposal. If reprocessing of any of this fuel is required, the amount will be small. The 1 metric ton of 10% enriched uranium in the Big Ten facility (Table 4.4.13) is the largest item that might require reprocessing.

The Sandia Annular Core Research Reactor, as noted above, will eventually discharge a full core containing about 70 kg U in the form of pin-type elements containing UO_2 pellets. No conclusion has been reached here as to the probably method of disposal of this fuel, although the relatively high enrichment suggests some economic value in the recovery of the U-235. If the fuel elements are not reprocessed, they could be packaged for repository disposal.

4.4.3.8.3 Other reactors

Table 4.4.7 lists 3 reactors at Idaho National Engineering Laboratory and 4 reactors at Knolls Atomic Power Laboratory for which no data were obtained. These reactors are all related to the naval reactors program. For reasons of security, no attempt was made to obtain data on these reactors or their fuels.

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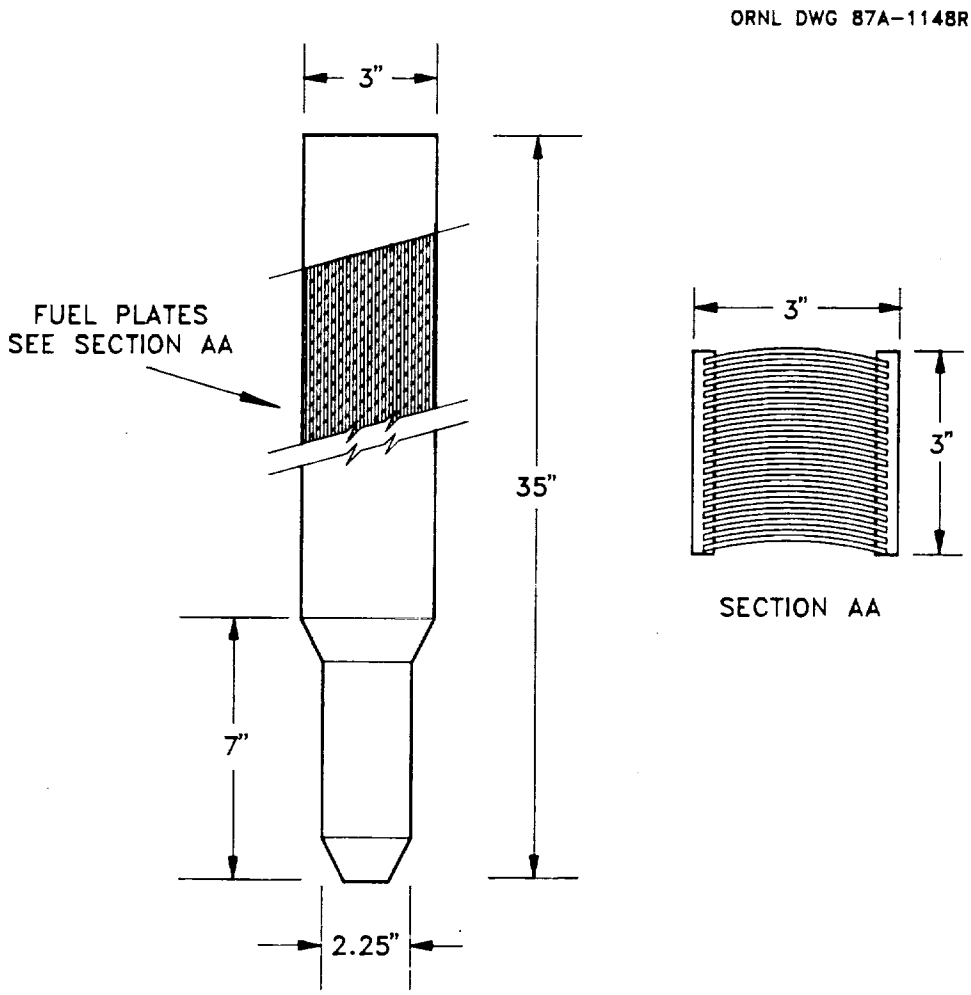


Fig. 4.4.1. MTR plate-type fuel assembly.

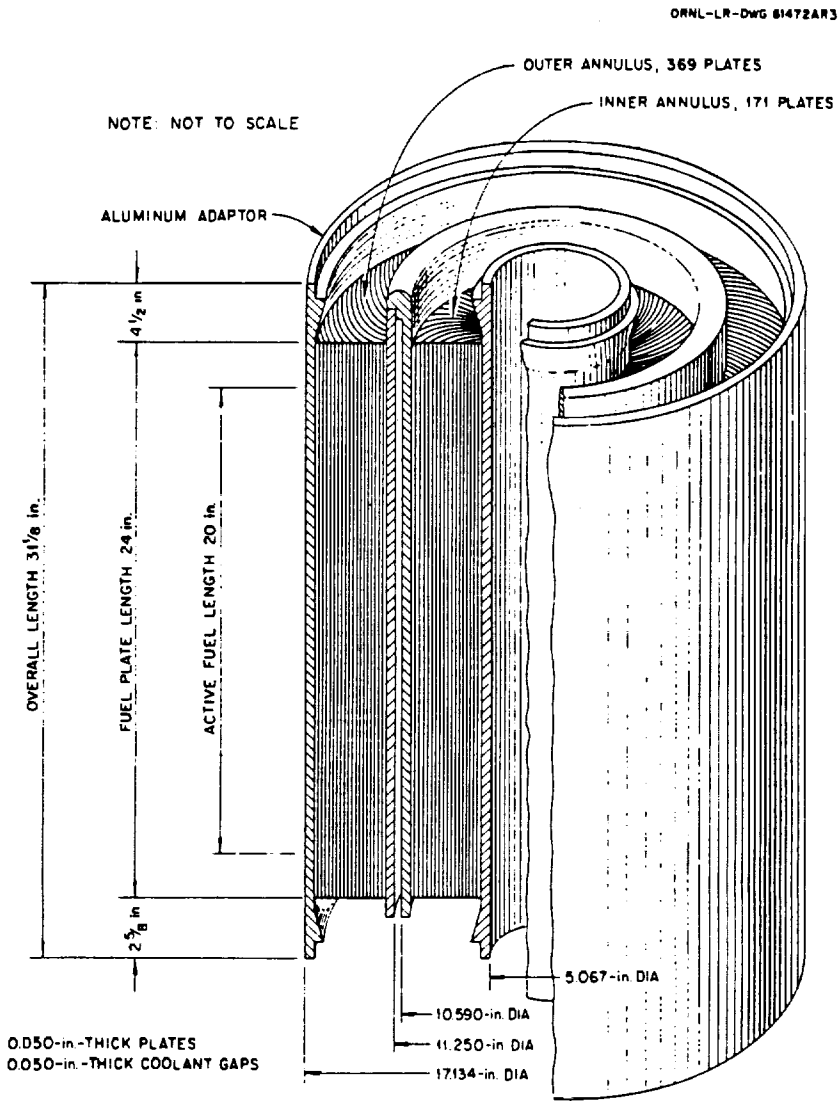
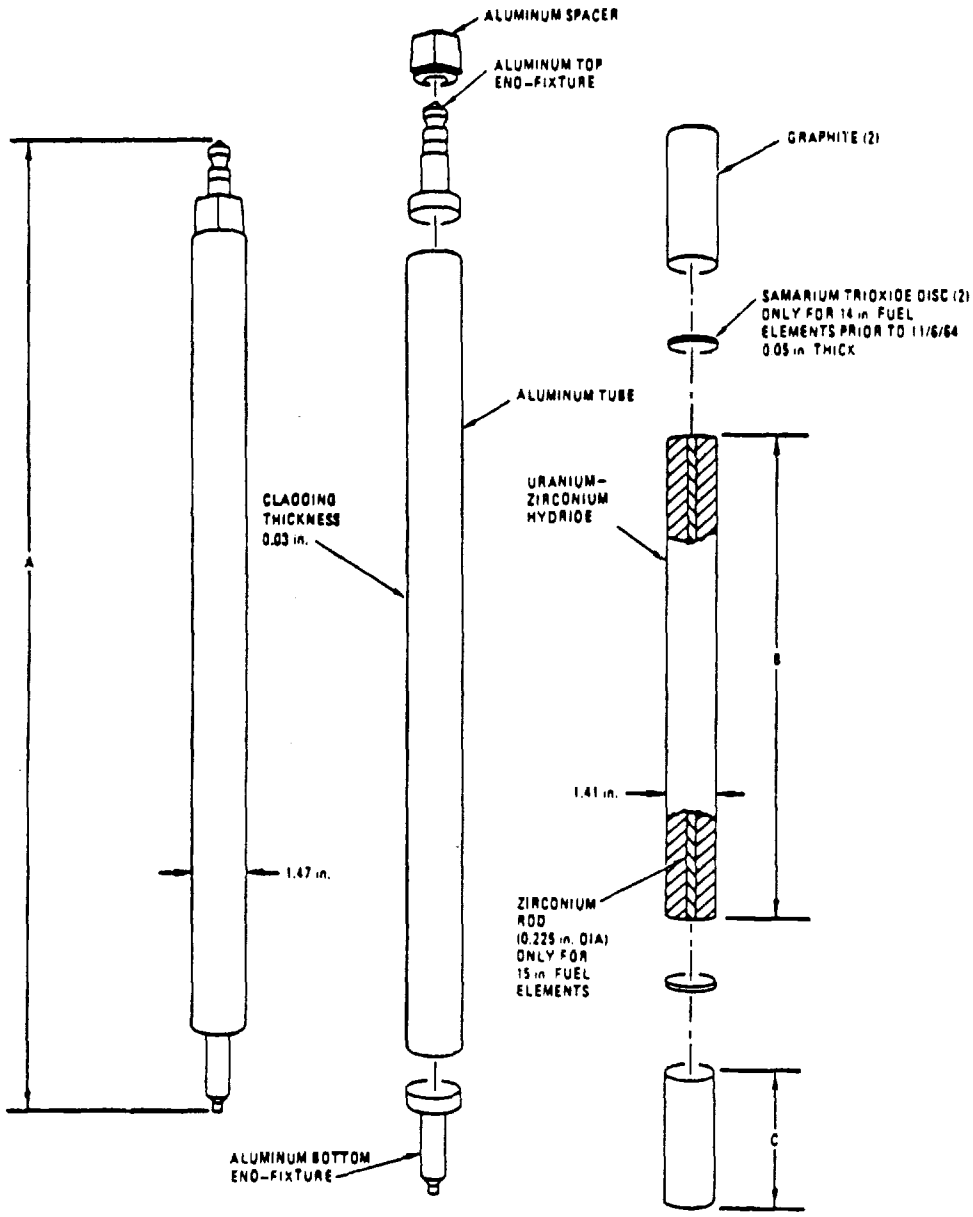


Fig. 4.4.2. High Flux Isotope Reactor fuel element. Source: HFIR 1982.



Drawing No.	TRIGA Fuel Type	A (in.)	B (in.)	C (in.)
TOS210D160	Original - 14 in.	28.37	14.0	3.95
TOS210D130	Original - 15 in.	28.3	15.0	3.53

Fig. 4.4.3. TRIGA aluminum-clad fuel element. Source: Tomsio 1986.

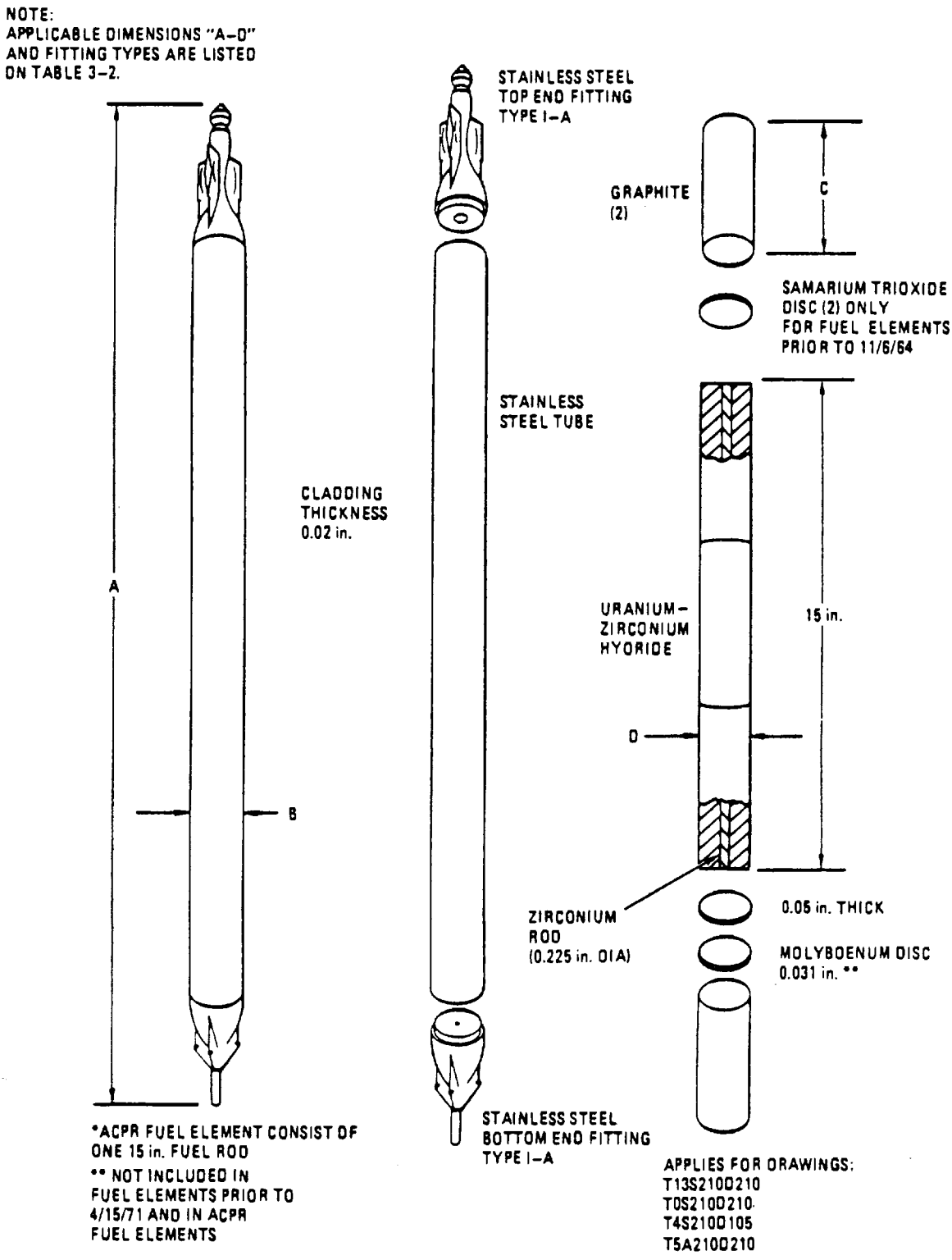


Fig. 4.4.4. TRIGA stainless-steel-clad fuel element. Source: Tomsio 1986.

Table 4.4.1. Research and test reactors by ownership category[a]

Reactor category	Number of reactors[b]
Commercial research	5
Government-owned non-DOE	7
Government-owned DOE	37
Educational	35
Total	84

[a]Reactors that have been permanently shut down are not included. Two DOE-owned reactors that are on standby (could be restarted if desired) are included.

[b]Numerous sources were contacted in compiling this table. See Tables 4.4.3, 4.4.4, 4.4.8 through 4.4.16, and 4.4.23.

Table 4.4.2. Number of operational research and test reactors in each fuel type category[a]

Fuel type	Private research and test	Government owned (Non-DOE)	Government owned (DOE)	University/ educational	Total
1. Plate or tube type, U-Al alloy, high enrichment (MTR, Argonaut, etc.)	1	2	12	13	28
2. TRIGA (U-Zr-H)	4	3	2	15	24
3. UO_2-polyethylene discs or blocks	0	0	0	3	3
4. PULSTAR and other low-enriched pin type	0	0	0	4	4
5. Liquid fuels (aqueous solutions)	0	0	0	0	0
6. U-Mo alloy, high-enriched (93.2%)	0	2	4	0	6
7. UO_2-PuO_2 mixed oxide fuel	0	0	1	0	1
8. Other	0	0	18[b]	0	18
Totals	5	7	37	35	84

[a]Reactors that have been permanently shut down are not included. Two DOE-owned reactors that are on standby (could be restarted if desired) are included. For sources of information, see Tables 4.4.3, 4.4.4, 4.4.8 through 4.4.16, and 4.4.23.

[b]This category includes classified reactors for which no information on fuel was obtained.

Table 4.4.3. Private commercial research and test reactors (operational)

Owner	Normal power	Type	Enrichment (%)	Normal core loading (kg ^{235}U)	Cladding material	Average full-power hours/year	Average ^{235}U burnup (g/year)	Disposition of spent fuel
Aerotest Operations, Inc.[a] San Ramon, CA	250 kW	TRIGA	<20	2.74	Al	1,200		
Dow Chemical Co.[b] Midland, MI	100 kW	TRIGA	19.9	2.8	SS			
General Atomic[c] San Diego, CA	250 kW	TRIGA	<20	2.6	Al or SS	150–300	2.0	Shipped to INEL
General Atomic[c] San Diego, CA	1,500 kW	TRIGA F	70	13.7	SS	300–400	20.0	Shipped to INEL
General Electric NTR[d] Pleasanton, CA	100 kW	Disk plate fuel	93	3.7	Al	600		[e]

[a]J. J. Haskins, telephone (415) 837-4248.
[b]C. W. Kocher, telephone (517) 636-0304.
[c]W. L. Whittemore, telephone (714) 455-3277.
[d]Denny Smith, telephone (415) 862-2211.
[e]Fuel (full core) probably will be sent to INEL for reprocessing between 1991 and 1995 (Smith 1986).

Table 4.4.4. Private commercial research and test reactors (shut down)

Owner	Normal power	Type	Enrichment (%)	Normal core loading, (kg ^{235}U)	Cladding material	Average full-power hours/year	Average ^{235}U burnup (g/year)	Disposition of spent fuel
Babcock and Wilcox[a] Lynchburg, VA	1 kW	Pin-type fuel	2.5-4.0		Al or SS			Shipped to SRP
Babcock and Wilcox[a] Lynchburg, VA	200 kW	U-Al plate-type fuel	93					Shipped to SRP
Babcock and Wilcox[b] Lynchburg, VA		U-Al plate-type fuel	93	6.44	Al			Shipped to SRP
General Electric GETR[c] Pleasanton, CA	50,000 kW	Plate-type fuel	93		Al			Shipped to INEL
Northrop Research Center[d] Palos Verdes, CA	1,000 kW	TRIGA F	20		SS			
Cintichem, Inc.[d] Tuxedo, NY	5,000 kW	Plate-type fuel	93	5.0	Al			
Westinghouse Electric[f] Zion, IL	100 W	Concentric tube fuel element	93.5	4.4	Al			[g]

[a]T. C. Engelder, telephone (804) 522-5145.
[b]Burn 1988.
[c]Denny Smith, telephone (415) 862-2211.
[d]G. B. Cozens, telephone (213) 970-2297.
[e]J. J. McGovern, telephone (914) 351-2131.
[f]Karen Rueter and R. J. Banchak, telephone (312) 872-4585. Also Burn 1988.
[g]Approximately 3.2 kg of highly enriched fuel was sent to SRP in 1981 (Banchak 1986).

Table 4.4.5. Government-owned non-DOE research and test reactors[a]

Location	Power level (kW)	Core loading, number of elements	Core loading, kg U-235	Enrichment (%)
Highly-enriched plate-type fuels[b]				
National Institute of Standards and Technology, Gaithersburg, MD	20,000	30	5.7	93.2
Rhode Island Nuclear Science Center[c]	2,000	35	3.47	93.2
TRIGA type (U-Zr hydride)				
Armed Forces Radiobiology Research Institute Bethesda, MD	1,000	87	3.31	19.9
U.S. Geological Survey Denver, CO	1,000	116	3.08	19.9
Veterans Administration Omaha, NE	18	56	2.03	19.9
U-10% Mo alloy fuel[d]				
U.S. Army Aberdeen Proving Grounds, MD	10	11	97	93.2
U.S. Army, White Sands Missile Range, NM	10	11	97	93.2

[a]Sources of general information: Burn 1988 and OSTI 1990. Specific sources: Raby 1989, White 1991, Dubyoski 1989, Simeneon 1990, Blotcky 1990, and DeBey 1990.

[b]The two reactors listed in this category have U-Al alloy plate fuel clad with Al.

[c]The reactor at the Rhode Island Nuclear Science Center is owned by the State of Rhode Island.

[d]The two reactors listed in this category are essentially identical to the Oak Ridge National Laboratory Health Physics Reactor, which has been shut down.

Table 4.4.6. Government-owned non-DOE reactors — fuel data[a]

Location	Fuel element dimensions (cm)	Full-power hours/year	Spent fuel burnup	Refueling schedule
Highly-enriched plate-type fuels				
National Institute of Standards and Technology, Gaithersburg, MD	7.6 × 8.55 × 174.8	6,800	60-80%	
Rhode Island Nuclear Science Center	7.62 × 7.62 × 100.3	1,750	~12%	
TRIGA type (U-Zr hydride)				
Armed Forces Radiobiology Research Institute Bethesda, MD	3.73 diam × 72	30	~11g U-235/yr	
U.S. Geological Survey Denver, CO	3.73 diam × 72	900		
Veterans Administration Omaha, NE	3.76 diam × 72	1,850		
U-10% Mo alloy fuel				
U.S. Army Aberdeen Proving Grounds, MD	Discs ~25 cm diam		Negligible	None expected
U.S. Army, White Sands Missile Range, NM	Discs ~25 cm diam		Negligible	None expected

[a]Sources of information: Burn 1988, OSTI 1990, and specific sources listed in Table 4.4.6.

Table 4.4.7. Summary of DOE-owned research and test reactors[a]

Location and name	Power level		Fuel type
Argonne National Laboratory			
Bioresearch Reactor (JANUS)	200	kW	1
Argonne West, Idaho Falls			
Experimental Breeder Reactor II	62	MW	8
Fast source reactor	1	kW	8
Transient test reactor	80	kW	8
Neutron radiography reactor	250	kW	2
Zero power physics reactor	c		8
Brookhaven National Laboratory			
High flux beam reactor	60	MW	1
Medical research reactor	3	MW	1
Hanford			
Neutron radiography	250	kW	2
Fast Flux Test Reactor	400	MW	7
Idaho National Engineering Laboratory			
Advanced Test Reactor	250	MW	1
ATR Criticality Facility	c		1
Advanced Reactivity Measurement Facility	c		1
Coupled Fast Reactor Measurement	c		1
Power Burst Facility	28	MW	8
Large Ship Reactor	b		b
Natural Circulation Reactor	b		b
Submarine Reactor Facility	b		b
Knolls Atomic Power Laboratory			
Destroyer Dual Reactor	b		b
Modifications and Additions Facility	b		b
Submarine Advanced Reactor	b		b
Trident Prototype Reactor	b		b
Los Alamos National Laboratory			
OMEGA West Reactor	5	kW	1
Big Ten	c		8
Fast Burst SKUA	c		6
Fast Burst GODIVA	c		6
Oak Ridge National Laboratory			
High Flux Isotope Reactor	100	MW	1
Bulk Shielding Reactor	2	MW	1
Tower Shielding Reactor II	1	MW	1
Pool Critical Assembly	c		1
Rocky Flats			
Horizontal Split Table	c		8
Solution System	c		8
Tank Reservoir	c		8
Vertical Split Table	c		8
Sandia National Laboratory			
Pulsed Reactor II	5	kW	6
Pulsed Reactor III	5	kW	6
Annular Core Research Reactor	2	MW	8

[a]Total number of reactors in this list is 37. Reactors that have been permanently shut down are not listed. Fuel type numbers are referred to in Table 4.4.2. Sources: Burn 1988 and other sources listed in Tables 4.4.8 through 4.4.17.

[b]No data obtained. These reactors are counted in fuel category 8.

[c]Critical facility.

Table 4.4.8. DOE-owned reactors at Argonne National Laboratory, Argonne, IL[a]

Name	Power level	Reactor and fuel data	Refueling schedule
Biological Research Reactor (JANUS)	200 kW	JANUS is a tank-type reactor using 93%-enriched U-Al alloy fuel. A fuel element consists of 3 concentric cylinders and has overall dimensions of 7.62 cm diam × 128.65 cm. Each element contains 0.14 kg U-235. There are 19 elements in the core, giving a total core loading of about 2.66 kg U-235. Fuel burnup is negligible	JANUS is operational and has only been discharged once since reaching initial criticality in 1964. No further fuel discharge is expected until final shut down
Reactors permanently shut down:			
Thermal Source Reactor	10 kW	93%-enriched U-Al alloy plate-type fuel. Element has 10 plates. Element dimensions (cm) are 8.2 × 7.62 × 67.9. Full core contains about 3 kg of U-235. Cladding is aluminum. Fuel burnup is negligible	The reactor has been shut down for about 3 years. The core is being removed and will be shipped to INEL; about half the core had been removed as of February 1991

[a]Sources: Burn 1988; OSTI 1990; John Hartig, ANL, 1991; and Roland Armani, ANL, 1991.

Table 4.4.9. DOE-owned reactors at ANL West, Idaho Falls, ID[a]

Name	Power level	Reactor and fuel data	Refueling schedule
Experimental Breeder Reactor II	62 MW	Pool-type sodium-cooled metallic-fueled fast breeder reactor. Fuel element dimensions are 5.82 cm equivalent diam × 167.1 cm long; or 233.3 cm long if top and bottom fittings are included. Element has 91 rods containing a total of 3.01 kg U-235. Rods are U-10% Zr alloy, bonded with sodium to stainless steel cladding. Enrichment is 67% U-235. Core normally contains 113 standard elements and up to 12 control elements. Total core loading is about 240 kg U-235 plus about 10 kg Pu-239	Normal operating cycle is 2,700–5,400 MW days followed by a 5-day refueling shutdown. Runs about 6,100 full-power hours per year. Fuel is shipped to INEL for reprocessing. Typical burnup is 6–8% U-235. Typical refueling replaces 6 standard elements and 1–2 control elements
Fast Source Reactor	1 kW	Fuel element is a circular cylinder with dimensions (cm) of 11.5 diam × 5.1 cm long. Fuel is 93% enriched uranium metal with stainless steel cladding	This reactor is used infrequently. There is no refueling. Burnup is negligible
Transient Test Reactor	80 kW	Fuel elements are highly enriched (93%) UO_2 dispersed in graphite with Zircaloy cladding. Element dimensions are 10.16 × 10.16 × 121.9 cm. There are between 250 and 350 elements (10–15 kg U-235) in the reactor	Burnup is negligible; fuel is expected to last indefinitely and at final discharge is expected to be shipped to INEL
Neutron Radiography Reactor	250 kW	This is a TRIGA reactor using 70% enriched U-Zr-H fuel. Element dimensions (cm) are 7.62 diam × 83.8 long. There are about 90 elements in the core. There is currently no plan to change to 20% enriched fuel	There is no refueling planned; fuel is expected to last indefinitely. On final shut down, fuel is expected to be shipped to INEL
Zero Power Physics Reactor (ZPR)		This is a critical assembly using 93% enriched uranium metal with 304 SS cladding and U-Pu-Mo ternary alloy fuels. Element dimensions (cm) are 0.55 × 5.0 × 2.26 to 20.03 long. A variety of experiments can be mocked up	No refueling is planned. Burnup is negligible

[a]Sources: Burn 1988; Leon Walters, Daniel Pruett, and Lawrence J. Harrison, ANL West, 1991.

Table 4.4.10. DOE-owned reactors at Brookhaven National Laboratory, Upton, NY[a]

Name	Power level	Reactor and fuel data	Refueling schedule
High Flux Beam Reactor	60 MW	Pressurized heavy-water moderated and cooled. High-enriched (93%) plate-type fuel using U_3O_8 cermet with 6061 aluminum cladding. Fuel element has 18 curved plates. Element dimensions are 8.18 × 7.16 × 145.4 cm. Core loading is 8.23 kg U-235 (28 elements)	Refueling frequency about 25 days. About 24 elements replaced at each refueling. Fuel is reprocessed at INEL. Burnup is about 37%
Medical Research Reactor	3 MW	Tank type, light-water-moderated and cooled. Typical plate-type fuel element (93.2% enriched) has 0.14 kg U-235 in 18 curved plates. Fuel element dimensions are 7.62 × 7.62 × 87.0 cm. Core contains about 30 fuel elements with a total core loading of about 4.2 kg U-235	Refueling is about 1 element every 2–5 years. Burnup is about 40%

[a]Sources of data: Burn 1988; OSTI 1990; David Rorer, BNL, 1989.

Table 4.4.11. DOE-owned research and test reactors at Hanford Site, Richland, WA[a]

Name	Power level	Reactor and fuel data	Refueling schedule
Hanford Neutron Radiography Reactor	250 kW	TRIGA-type reactor, U-Zr-H fuel. Enrichment 19.9%. Reactor contains 65 elements, 3.81 cm diam × 72.4 cm long	Annual refueling. Uses about 1 element per year or less
Fast Flux Test Facility[b]	400 MW	Sodium-cooled fast reactor research facility. Fuel contains PuO_2 - natural UO_2 pellets in pin-type assemblies, stainless steel clad. Core contains 83 elements, 74 of which are driver fuel elements, with a total of 550 kg of Pu. This reactor and its fuel are discussed in detail in Appendix 4E	Average rate of fuel discharge is about 42 driver elements/year

[a]Sources of data: OSTI 1990; A. T. Luksic, PNL, 1988; G. H. Beeman, PNL, 1988.
[b]For further information see Appendix 4E.

Table 4.4.12. DOE-owned reactors at Idaho National Engineering Laboratory, Idaho Falls, ID[a]

Name	Power level	Reactor and fuel data	Refueling schedule
Advanced Test Reactor	250 MW	High-flux test reactor, pressurized light-water-moderated and cooled. High-enriched (93%) U-Al plate fuel, clad with 6061 aluminum. Fuel element dimensions (cm) 6.5 × 10.5 × 168.0. Element consists of 19 curved plates. There are 40 elements in core. Normal core loading 35–43 kg U-235	Fuel is reprocessed at INEL (ICPP)
Advanced Test Reactor Criticality Facility	10 kW	This is a pool-type, low-power criticality facility. It uses the same fuel as the Advanced Test Reactor	Fuel use is negligible; no refueling is needed or anticipated
Advanced Reactivity Measurement Facility	10 kW	Critical facility, open-pool, light water. Plate fuel, high enrichment (93%), consisting of U-Al alloy with Al cladding, 15 plates per fuel element. Element dimensions (cm) 8.28 × 8.28 × 98.8. Core contains 5.08 kg U-235 (28 elements)	No refueling anticipated
Coupled Fast Reactor Measurement Facility	10 kW	This is identical to the Advanced Reactivity Measurement Facility and is located in the same pool	No refueling anticipated
Large Ship Reactor		No information obtained on this reactor	
Natural Circulation Reactor		No information obtained on this reactor	
Submarine Reactor Facility		No information obtained on this reactor	
Reactors in standby:			
Power Burst Facility	28 MW	Open tank, light-water-moderated, rod-type fuel. Rods are 1.91 cm OD × 12.06 cm long. Rods are stainless-steel clad. An element may contain from 28 to 62 rods. Core contains 60 standard and 8 control elements. Normal core loading is 104 kg U-235. Fuel is 18.5% enriched and is composed of UO_2-ZrO_2-CaO pellets	No refueling is used or anticipated. Reactor is shut down but could be restarted if desired

[a]Sources of data: Burn 1988 and D. M. Sherick, INEL, 1991.

Table 4.4.13. DOE-owned reactors at Los Alamos National Laboratory, Los Alamos, NM[a]

Name	Power level	Reactor and fuel data	Refueling schedule
OMEGA West Reactor	8 MW	Closed tank reactor using MTR plate-type fuel. Element dimensions are 7.61 × 8.56 × 108.3 cm. Element has 18 or 19 plates, 93% enriched U-Al alloy, clad with aluminum. Total of 5.5–6.0 kg U-235 in core	Reactor operates regularly every day. Fuel is expected to go to INEL for reprocessing. No fuel has been shipped for about 5 years
Fast Burst SKUA	b	Contains about 175 kg of 93% uranium as a metallic uranium alloy. Element is a right circular cylindrical annulus about 30.5 cm diam × 30.5 cm high	No refueling expected
Fast Burst GODIVA	b	Contains about 65.5 kg of 93% enriched uranium as a metallic uranium alloy. Element dimensions are 17.8 cm diam × 17.8 cm high	No refueling expected
Big Ten Reactor	b	Critical assembly; contains about 1 metric ton of uranium metal fuel, about 10% enriched. Various configurations can be mocked up as needed	No refueling expected
Reactors shut down:			
Solution High Burst Reactor (SHEBA)	b	The Solution High Burst Reactor was fueled by an aqueous solution of uranium fluoride. The reactor has been disassembled and removed	Fuel solution has been removed and has been saved for possible future use

[a]Sources of data: Burn 1988; Gerald Ramsey and Richard Anderson, LANL, 1991.
[b]Critical facility, no normal power.

Table 4.4.14. DOE-owned reactors at Oak Ridge National Laboratory, Oak Ridge, TN[a]

Name	Power level	Reactor and fuel data	Refueling schedule
High Flux Isotope Reactor	100 MW	Uses highly-enriched (93%) fuel. Fuel is a sandwich of fuel-bearing cermet sealed between covers of 6061 aluminum. Fuel is in two concentric elements containing curved plates. Outside diameter of element is about 43 cm and length is about 79 cm. Core loading is about 9.4 kg of U-235, 6.8 kg in the outer element and 2.6 kg in the inner element	The reactor is expected to remain in operation. Fuel is shipped to Savannah River Plant for reprocessing, about 1 fuel element per month
Tower Shielding Reactor II	1 MW	Light-water-cooled and moderated spherical-core reactor with high-enriched uranium alloy aluminum-clad plate fuel	This reactor does not replace fuel. Fuel burnup will probably not be more than 10% by year 2020
Pool Critical Assembly		Plate-type fuel, <20% enriched. Element dimensions (cm) 7.6 × 8.0 × 87.6. Normal core loading 3.4 kg U-235 (26 standard and 4 control elements)	There is no fuel replacement. Fuel burnup is negligible
Reactors on standby:			
Bulk Shielding Reactor[b]	2 MW	93%-enriched plate-type fuel (MTR type). Fuel element has 19 plates and contains about 190 g U-235. Element dimensions (cm) 7.6 × 8.0 × 87.6. Normal core loading 4–5 kg U-235 (18–28 standard elements plus 6 control elements)	Reactor was shut down about 1987 but could be restarted again if desired. When operating, fuel was reprocessed at Savannah River Plant
Reactors permanently shut down:			
Health Physics Reactor	10 kW	93.1% enriched U metal alloyed with 10% molybdenum. Fuel is in disks about 25 cm diam. Core is about 25 cm diam × 23 cm high and contained 11 fuel disks. About 100 mg of U-235 was burned during the 20-year operating period	Permanently shut down and all fuel removed. There was no refueling when reactor was operational
Oak Ridge Research Reactor	30 MW	U-Al plate 93% enriched MTR-type fuel, similar to that used in the Bulk Shielding Reactor	Permanently shut down and removed

[a]Sources of information: Burn 1988; OSTI 1990; Stover 1989; Corbett 1989; and Holland 1989.

[b]The Low-Temperature Neutron Irradiation Facility, which is listed in OSTI 1990, is not a reactor; it is a cryostatic cell for irradiating samples in the Bulk Shielding Reactor (Coleman 1992).

Table 4.4.15. DOE-owned reactors at Rocky Flats Plant, Golden, CO[a]

Name	Power level	Reactor and fuel data	Refueling schedule
Horizontal Split Table		See note b	None
Solution System		See note b	None
Tank Reservoir		See note b	None
Vertical Split Table		See note b	None

[a]Sources of information: Burn 1988.
[b]These are all critical assemblies and have no normal power level. Various experiments can be mocked up as required. No spent fuel is generated.

Table 4.4.16. DOE-owned reactors at Sandia National Laboratory, Albuquerque, NM[a]

Name	Power level	Reactor and fuel data	Refueling schedule
Pulsed Reactor II	5 kW	Identical to ORNL Health Physics Reactor. Fuel is highly enriched (93% U-235) uranium +10% molybdenum alloy. Fuel is in the form of disks about 20.5 cm diam × 3.45 cm thick, clad with aluminum. Core loading is 87 kg U-235	Fuel usage and burnup are negligible. No refueling is expected. Eventually, 87 kg of U-235 will be discharged
Pulsed Reactor III	5 kW	Identical to Pulsed Reactor II	Same as Pulsed Reactor II
Annular Core Research Reactor	2 MW	Pulsed reactor; average power is about 15 kW, short pulses up to 2 MW. Open pool type, light-water-cooled and moderated. Core contains 236 standard elements and 11 control elements. Elements are stainless steel clad pins, 3.73 cm diam × 73.15 cm long, containing pellets of UO_2, 35% enrichment, and BeO. Total core loading is 23.6 kg U-235	No refueling expected

[a]Sources of data: Burn 1988; Theodore F. Luera, Sandia, 1989.

Table 4.4.17. DOE-owned research and test reactors at Savannah River Site[a]

Name	Power level	Reactor and fuel data	Refueling schedule
Reactors permanently shut down:			
Savannah River Test Pile	50 W	Critical assembly. No regular operating schedule. Various fuel configurations can be mocked up. No appreciable burnup of fuel	None

[a]Sources: Burn 1988; R. Benton, SRS, 1991.

Table 4.4.18. Operational university/educational reactors — summary[a]

Fuel type and location	Power, kW	Core loading no. of elements	Core loading kg U-235
Highly-enriched plate-type fuels			
Georgia Institute of Technology	5,000	17	3.01
Iowa State University	10	12	3.19
Manhattan College	Negligible	16	3.02
Massachusetts Inst. of Technology	4,900	24	12.2
Ohio State University	10	20	3.18
Purdue University	10	16	2.65
University of Florida	100	24	3.35
University of Lowell	1,000	26	3.50
University of Michigan	2,000	35	9.00
University of Missouri (Columbia)	10,000	8	6.20
University of Missouri (Rolla)	200	28	2.85
University of Virginia	2,000	20	3.30
Worcester Polytechnic Institute	10	24	3.26
TRIGA type (U-Zr-H fuel)			
Cornell University	500		
Kansas State University	250	80	2.7
Oregon State University	1,000		11.2
Pennsylvania State University	1,000	95-101	3.4
Reed College	250		2.3
Texas A&M University	1,000		10.5
University of Arizona	100		3.3
University of California, Irvine	250	81	2.9
University of Illinois LOPRA	10	55	2.1
University of Illinois	1,500	100	3.8
University of Maryland	250	96	3.4
University of Texas	1,100		2.8
University of Utah	250	87	2.9
University of Wisconsin	1,000	91	8.0
Washington State University	1,000	110	6.7
UO_2-polyethylene fuel			
Idaho State University	Negligible	9	0.67
Texas A&M University	Negligible	9	0.69
University of New Mexico	Negligible	9	0.67
Low-enriched UO_2 pin-type fuel			
Cornell University zero-power reactor	Negligible	815	35.
North Carolina State University	1,000	25	12.7
Rensselaer Polytechnic Institute	Negligible		
State University of New York (Buffalo)	2,000	32	18.2

[a]Total number of reactors in this table is 35. Reactors that have been permanently shut down are not included. For a list of reactors recently shut down, see Table 4.4.23. Sources: Burn 1988, OSTI 1990, and other sources listed in Table 4.4.23 and Sect. 4.4.4.

Table 4.4.19. Educational reactors — highly enriched plate-type fuels[a]

Location	Fuel element dimensions (cm)	Full-power hours/year	Spent fuel burnup (%)	Refueling schedule
Georgia Institute of Technology	7.52 × 7.04 × 219.4		30	None
Iowa State University	7.62 × 14.06 × 66.0			None
Manhattan College	8.89 diam × 93.98[b]	150		None
Massachusetts Institute of Technology	6.98 × 6.98 × 66.7[c]		40	6 elements/year
Ohio State University	7.62 × 7.62 × 88.9	150–200		
Purdue University	7.52 × 7.52 × 81.9			None
University of Florida	7.23 × 5.44 × 65.1			Element lifetime >10 years
University of Lowell	7.62 × 7.62 × 101.6	500	15	
University of Michigan	7.47 × 8.26 × 87.4		20–30[d]	1 element per 17 days
University of Missouri (Columbia)	7.04 × 14.63 × 82.6[e]		20	8 elements per 17 weeks
University of Missouri (Rolla)	7.57 × 8.74 × 87.0			None
University of Virginia	7.61 × 8.26 × 93.7	2,000	17	f
Worcester Polytechnic Institute	7.75 × 7.75 × 101.6	100		None

[a]All these reactors originally used U-Al alloy fuel of about 93% enrichment, clad with Al. A change to LEU (low-enriched fuel) of less than 20% enrichment (typically 19.8% enriched) is in progress, in accordance with a 1982 policy statement by the Nuclear Regulatory Commission. Some of these reactors have already made this change. Sources for this table are Burn 1988 and telephone conversations listed in Sect. 4.4.4.

[b]Each fuel element consists of 6 concentric cylindrical shells spaced about 0.4 cm apart. Each shell consists of three 120° curved plates. The innermost cylinder has a diameter of 3.81 cm; the outermost has a diameter of 8.89 cm.

[c]The cross section of each fuel element is a 60–120° regular parallelogram with faces 6.98 cm wide (Kwok 1991).

[d]The enrichment is 19.5%. Source: Burn 1991.

[e]The fuel elements of this reactor have a pie-shaped configuration in which each element is a 45° slice of a circle. The enrichment is 93%. Source: Hultsch 1991.

[f]Reactor typically runs 3 day/week, 10 hours/day. A shipment of spent fuel is made to Savannah River Site about once every 3 years.

Table 4.4.20. Educational reactors — TRIGA fuel (U-Zr hydride)[a]

Location	Fuel element dimensions (cm)	Full-power hours/year	Spent fuel burnup	Refueling schedule
Cornell University				None
Kansas State University		300		None
Oregon State University		~900	4–8%	b
Pennsylvania State University		~560	40–50,000 MWd/ton	~6 elements/2 year
Reed College		~150		None
Texas A&M University		~2,500		b
University of Arizona		~0.3 MWd/year		None
University of California, Irvine		~650		1–2 year
University of Illinois (10 kW)		~1		None
University of Illinois (1500 kW)		~270		~2 elements/year
University of Maryland		~48	~0.3 g ^{235}U/year	None
University of Texas				c
University of Utah				d
University of Wisconsin		~600–800	37–63%	b
Washington State University		~900	36 g ^{235}U/year	b

[a]TRIGA fuel elements are cylindrical and typically have a diameter of about 3.73 cm and a length of about 72 cm. See Figs. 4.4.2 and 4.4.3. Enrichment is usually 19.9% but may be as high as 70%. Cladding is stainless steel or aluminum. Sources for this table are Burn 1988 and telephone conversations listed in Sect. 4.4.4.

[b]Planning to change from 70% enriched to <20% enriched around 1994. Do not anticipate any appreciable refueling after that.

[c]No refueling expected until 1996; perhaps a few elements per year after 1996.

[d]About 4–6 new elements would satisfy needs for foreseeable future.

Table 4.4.21. Educational reactors — UO_2-polyethylene fuel[a]

Location	Fuel element dimensions (cm)	Full-power hours/year	Spent fuel burnup	Refueling schedule
Idaho State University	25 diam × 1.0, 2.5, or 5.0 thick		Negligible	None
Texas A&M University	25.75 diam × 1.0, 2.3, or 3.9 thick[b]		Negligible	None
University of New Mexico	25.6 diam × 1.0, 2.0, or 4.0 thick		Negligible	None

[a]These fuels consist of disks or blocks of polyethylene containing homogeneously dispersed UO_2 with an enrichment of 19.9%. They operate at a very low power level, typically 5 W, and are not refueled. Burnup of the fuel is negligible. The fuel elements have no cladding. Typical fuel configuration is a stack of flat disks.

[b]This reactor was removed from Catholic University and sent to Texas A&M. Dimensions are from Burn 1988.

Table 4.4.22. Educational reactors — low-enriched UO_2 pin-type fuels

Location	Fuel element dimensions (cm)	Full-power hours/year	Spent fuel burnup	Refueling schedule
Cornell University ZPR	1.69 diam × 158[a]		Negligible	None
North Carolina State University	8.0 × 6.96 × 96.5 (5 × 5 array)	~1200	55 g ^{235}U/year[b]	
Rensselaer Polytechnic Institute[c]	SPERT-type pins about 1.18 cm diam × 105.7 cm long		Negligible	None
State University of New York (Buffalo)[d]	8.0 × 6.96 × 96.5 (5 × 5 array)	~6200	7000–5000 MWd/ton	>10 year

[a]Length shown is for a decappable fuel element. A normal fuel element has a length of 150.5 cm. Both types have an outside diameter of 0.666 in. (1.69 cm). Aderhold 1986.

[b]Source: Burn 1988.

[c]This is a critical facility and was still in use as of February 1992 (Connor 1992). Dimensions are from Burn 1988.

[d]Sources: Orlosky 1986 and Burn 1988.

Table 4.4.23. Educational reactors permanently shut down[a]

Type and location	Power level	Shutdown date	Fuel still at reactor?
High-enriched U-Al plate fuel			
University of California (LA)	100 kW	1984	Shipped to INEL[b]
University of Kansas	10 kW	1987	Shipped to SRP[c]
University of Virginia	100 W	1987	Still at U. of Virginia
Virginia Polytechnic Institute	100 kW	1984	Shipped to SRP[d]
University of Washington	100 kW	1988	Shipped to INEL[e]
TRIGA type			
Michigan State University	4900 kW	1989	Shipped to USGS and INEL[f]
Columbia University	250 kW	1985	None[g]
University of Illinois		1968	All shipped[b]
University of California (Berkeley)	1000 kW	1987	Shipped to INEL[h]
University of Texas (Austin)	250 kW	1990	Fuel is still at U. of Texas[i]
UO_2-polyethylene fuel			
Catholic University	Negligible	1986	Will be shipped to DOE[j]
Memphis State University	Negligible	1985	
University of Utah	Negligible	1990	In cask for shipping to DOE[k]
University of Oklahoma	Negligible	1988	Shipped to Oak Ridge[l]
Calif. Polytech. Inst.	Negligible	1980	
University of Delaware	Negligible	1978	Shipped to Oak Ridge[b]
Georgia Inst. of Tech.	Negligible	1986	Shipped to Oak Ridge[m]
Colorado State University	Negligible	1974	Shipped to South Korea[n]
Oregon State University	Negligible	1981	Still at Oregon State[o]
Polytech. Inst. of N.Y.	Negligible	1974	Shipped to Oak Ridge[b]
Tuskeegee Institute	Negligible	1984	Returned to DOE[b]
W. Virginia University	Negligible	1972	Shipped to Univ. of Okla.[b]
Rice University	Negligible	1965	Shipped to Texas A&M[b]
Aqueous liquid fuel			
University of California (Santa Barbara)	Negligible	1986	Shipped to INEL[p]
Brigham Young University	Negligible	1987	Shipped to INEL[q]
University of Wyoming	Negligible	1974	Shipped to Atomics Int'l.[b]

[a]List of shut down reactors was compiled from Burn 1988 and OSTI 1990. For a more complete list of university/educational reactors shut down, see OSTI 1990.

[b]Burn 1988.

[c]Telephone conversation, Harold Rosson, Univ. of Kansas, and R. Salmon, ORNL, June 7, 1991.

[d]Telephone conversation, Thomas F. Parkinson, Virginia Polytech, and R. Salmon, ORNL, June 10, 1991.

[e]Telephone conversation, D. L. Fry, Univ. of Washington, and R. Salmon, ORNL, June 7, 1991.

[f]Telephone conversation, Bruce Wilkinson, Mich. State Univ., and R. Salmon, ORNL, June 7, 1991. About one-third was sent to USGS (Denver) and two-thirds to INEL.

[g]No fuel was ever procured for this reactor (Burn 1988).

[h]Telephone conversation, Keith Brown, EG&G Idaho, and R. Salmon, ORNL, March 6, 1989

[i]Telephone conversation, Tom Bauer, Univ. of Texas, and R. Salmon, ORNL, December 11, 1990. Fuel is being used in the 1100 kW TRIGA at Univ. of Texas.

[j]Telephone conversation, Harold Keene, Catholic Univ., and R. Salmon, ORNL, June 7, 1991.

[k]Telephone conversation, Gary Sandquist, Univ. of Utah, and R. Salmon, ORNL, December 11, 1990.

[l]Telephone conversation, Paul Skierkowski, Univ. of Okla., and R. Salmon, ORNL, June 7, 1991.

[m]Telephone conversation, R. Karam, Georgia Inst. of Tech., and R. Salmon, ORNL, June 10, 1991.

[n]Telephone conversation, James Johnson, Colorado State, and R. Salmon, ORNL, June 7, 1991.

[o]Telephone conversation, Brian Dodd, Oregon State, and R. Salmon, ORNL, June 10, 1991.

[p]Telephone conversation, A. E. Profio, Univ. of Calif., and R. Salmon, ORNL, June 10, 1991.

[q]Telephone conversation, Gary Jensen, Brigham Young, and R. Salmon, ORNL, June 7, 1991.

Table 4.4.24. Estimated disposal requirements for research and test reactor fuels[a]

Fuel category	Fuel type	Number of reactors	Estimated number of fuel elements	Total U-235 (kg)	Estimated disposal volume (m^3)[b]
1	Plate-type U-Al alloy fuels, some 93% enriched, some 19.8% enriched	29	20,000–40,000	10,000–20,000	c
2	TRIGA U-ZrH fuels, SS or Al clad, some 19.8% enriched, some 70% enriched	24	2,500–4,500	100–200	40
3	Homogeneous UO_2-polyethylene discs or blocks, 20% enriched	3	87	6	1
4	PULSTAR and other low-enriched UO_2 pin-array fuels	4	1,000	115	10
5	Liquid fuels (aqueous)	0	0	3	c
6	U-Mo alloy highly-enriched fuels	6	c	c	c
7	UO_2-PuO_2 mixed oxide fuel	1	d	d	d
8	Other types	18	e	e	e

[a]Estimated total through year 2020, including discharged fuel and fuel in reactors at that time. These are ORNL estimates based on extrapolation of current operations.

[b]Assuming criticality safety can be achieved by addition of neutron poisons and that canistered TRIGA fuels will be acceptable for repository disposal.

[c]It is assumed that these fuels will be reprocessed and disposed of as vitrified defense HLW (less than 10 canisters total).

[d]Not determined, but it appears likely that the FFTF fuel will be reprocessed, either on-site or at INEL.

[e]Not determined.

Table 4.4.25. Radioactivity and thermal power of TRIGA spent fuel element[a]

Decay time (years)	Total radioactivity per fuel element (Ci)	Total thermal power per fuel element (W)
0.03	4.60E+3[b]	7.50E+0[b]
1	4.52E+2	1.90E+0
2	2.67E+2	9.50E-1
3	1.58E+2	7.00E-1
5	1.01E+2	5.20E-1
10	7.63E+1	4.20E-1
15	6.61E+1	4.02E-1
20	5.73E+1	3.85E-1
30	4.30E+1	3.50E-1
50	2.15E+1	3.07E-1
100	7.96E+0	2.00E-1
300	1.65E+0	5.00E-2
1,000	3.41E-1	1.07E-2
10,000	9.14E-2	2.63E-3
100,000	1.09E-2	2.05E-4
1,000,000	8.00E-4	1.00E-4

[a]Basis: one standard stainless-steel-clad spent fuel element containing 195 g total uranium, initial enrichment of 19.9%, and a burnup of 30,000 MWd/MTU. Radioactivity is estimated based on PWR fuel with a burnup of 33,000 MWd/MTU. Thermal power is based on data from Tomsio (1986) for decay times up to 100 years and on PWR data for subsequent decay times. This table is based on the results of ORNL calculations using ORIGEN2.

[b]Quantities shown are for a decay time of 10 days after discharge from reactor.

Table 4.4.26. Radioactivity and thermal power of UO_2-polyethylene spent fuel[a]

Decay time (years)	Total radioactivity (Ci/kg U)	Total thermal power (W/kg U)
0	1.5E+3	6.5E+0
1	2.3E+1	1.0E-1
10	3.9E+0	1.1E-2
100	4.0E-1	2.8E-3
1,000	1.8E-2	5.5E-4
10,000	4.7E-3	1.4E-4
100,000	5.6E-4	1.0E-5
1,000,000	4.8E-5	5.0E-6

[a]Burnup was assumed to be 300 MWd/MTU, which was estimated to be the maximum that will be encountered with this type of reactor. Burnup for most educational reactors of this type will be considerably less. Radioactivity and thermal power were estimated, by ratio, from PWR data at 33,000 MWd/MT. A UO_2-polyethylene fuel disk 25.75 cm diam × 4.0 cm thick contains about 0.6 kg of uranium. This table is based on the results of ORNL calculations using ORIGEN2.

Table 4.4.27. Radioactivity and thermal power of PULSTAR spent fuel[a]

Decay time (years)	Total radioactivity (Ci/kg U)	Total thermal power (W/kg U)
0	1.5E+5	6.5E+2
1	2.3E+3	1.0E+1
10	3.9E+2	1.1E+0
100	4.0E+1	2.8E-1
1,000	1.8E+0	5.5E-2
10,000	4.7E-1	1.4E-2
100,000	5.6E-2	1.0E-3
1,000,000	4.8E-3	5.0E-4

[a]Burnup was assumed to be 33,000 MWd/MTU; radioactivity and thermal power values were assumed to be the same as those for PWR fuel with a burnup of 33,000 MWd/MTU. This table is based on the results of ORNL calculations using ORIGEN2.

Table 4.4.28. Dimensions of TRIGA stainless-steel-clad fuel element[a]

Drawing No.	TRIGA fuel type	Fitting type	A (cm)	B (cm)	C (cm)	D (cm)
T13S210D210	Standard-streamline	I-A	75.39	3.754	6.50[b]	3.645
T0S210D210	Standard-plain	II-A	73.40	3.754	8.69	3.645
T4S210D105	Four rod cluster	III-A	75.90	3.592	8.69	3.480
T5A210D210	ACPR[c]	IV-A	73.38	3.754	8.76	3.556

[a]Source: Tomsio 1986. Dimensions refer to Fig. 4.4.3.
[b]Lower graphite is longer than upper graphite. Lower graphite = 9.45 cm.
[c]Annular core pulsed reactor.

Table 4.4.29. Historical and projected spent fuel discharged from the FFTF[a]

End of calendar year	Number of fuel assemblies discharged		Mass of fuel discharged (MTIHM)	
	Annual	Cumulative	Annual	Cumulative
1984	52	52	2.02	2.02
1985	27	79	1.05	3.07
1986	56	135	2.17	5.24
1987[b]	35	170	1.18	6.42
1988	36	206	1.22	7.64
1989	36	242	1.22	8.86
1990	45	287	1.52	10.38
1991	30	317	1.02	11.40
1992	30	347	1.03	12.43
1993	30	377	1.04	13.47
1994	30	407	1.05	14.52
1995	30	437	1.04	15.56
1996[c]	30	467	0.99	16.55
1997	30	497	0.99	17.54
1998	30	527	0.99	18.53
1999	30	557	0.99	19.52
2000	30	587	0.99	20.51
2001	30	617	0.99	21.50
2002	30	647	0.99	22.49
2003[d]	30	677	0.99	23.48

[a]Based on DOE 1987.

[b]Initial fuel generally consists of a mixture of UO_2 and PuO_2. A small number of assemblies discharged from 1984 through 1995 contain enriched uranium (generally 30% ^{235}U).

[c]All spent fuel assemblies discharged after 1995 are projected to have 39% enriched uranium.

[d]Data for years following 2003 are not available.

4.5 MISCELLANEOUS FUELS

4.5.1 Introduction

As a result of various research and test programs, quantities of miscellaneous spent fuels are stored at a number of sites throughout the United States. The amounts of individual fuels stored range from less than 1 kg to several metric tons of heavy metal. In this section, the following types of data are provided on these miscellaneous fuels, to the extent such data are available:

1. masses of heavy metals,
2. chemical form,
3. general description,
4. estimated burnup (MWd/MTIHM), and
5. sources for additional information.

The information provided in this section is organized along two main lines. The first part, Sects. 4.5.2 through 4.5.9, discusses the quantities of miscellaneous fuels stored at various sites; this discussion is arranged by sites. A general summary of this information is provided in Table 4.5.1; the quantity of total candidate materials shown amounts to about 254 MT of heavy metals. These data are mainly from the Integrated Data Base report for 1990 (DOE 1990), and the quantities cited are as of December 31, 1989. Most of this information was provided by the sites as part of their IDB submittals. The materials listed in Tables 4.5.1 through 4.5.9 as "miscellaneous radioactive materials" are essentially all miscellaneous spent fuels. Some of these fuels were removed from various research and test reactors already discussed in Sect. 4.4. In estimating ultimate disposal requirements, care must be taken not to count such fuels twice. The second part, Sects. 4.5.10 through 4.5.12, gives more specific information on the fuels from certain reactors; this discussion is arranged by reactors. More detailed information on disposal requirements is available in Salmon and Notz 1988.

4.5.2 Argonne National Laboratory West (ANL-W)

The spent fuel materials stored at ANL-W (Idaho Falls) are briefly described in Table 4.5.2. All of these materials are classed as scrap. No determination has been made as to which of these fuels may be amenable to reprocessing. Some of the fuels have a reactive metal (sodium or a sodium-potassium eutectic) bond between the fuel material and the cladding. It is probable that removal of the reactive metals from such fuels would be required in order to meet repository acceptance criteria; however, chemical reprocessing is a possible alternative.

4.5.3 Babcock & Wilcox (B&W)

The radioactive materials stored by Babcock & Wilcox are principally derived from LWR fuels. The fuel assemblies or fuel rods have been cut into pieces small enough to fit inside 4.25-in.-diam by 33-in.-long containers. A sketch of the container is presented in Fig. 4.5.1. The basic information concerning these materials is provided in Table 4.5.3.

4.5.4 Battelle Pacific Northwest Laboratory (BPNL)

The spent fuels and other highly radioactive materials held in storage at BPNL are briefly described in Table 4.5.4. With the exception of two small batches of glass mix, all of these materials are conventional LWR spent fuels, a small portion of which has been opened to obtain specimens.

4.5.5 Hanford Engineering Development Laboratory (HEDL)

Table 4.5.5 provides information concerning spent fuels being held at Hanford. The listed materials consist of fuel sections and whole fuel pins, plus a number of TRIGA fuel assemblies. The TRIGA fuel assemblies are packaged into 13 concrete-filled, 55-gal steel drums, 6 to 7 assemblies per drum.

4.5.6 Idaho National Engineering Laboratory (INEL)

The highly radioactive materials stored at INEL/ICPP are briefly described in Table 4.5.6. The total mass of heavy metal in storage is about 230 MT; this includes an estimated 82 MT for TMI-2 fuel. About 40 MT of the remaining 148 MT is in the general form of conventional LWR fuel, while another 34 MT consists of blanket material from the Fermi I reactor. This latter material consists of U-Mo metal in stainless steel tubes, with a sodium metal bond between the blanket material and its cladding. In addition to such uranium-based wastes, there is about 67 MT of predominantly thorium-containing waste. The latter is from the Shippingport light-water breeder reactor (LWBR) and from two high-temperature, gas-cooled reactors (HTGRs), Fort St. Vrain and Peach Bottom. The LWBR fuel is an oxide, whereas the HTGR fuel is a carbide, as described in a previous section of this report. The remaining 7 MT consists of miscellaneous items.

The Fermi I blanket fuel contains a quantity of sodium metal. It is likely that removal of these reactive constituents will be necessary in order to meet repository waste acceptance criteria.

4.5.7 Los Alamos National Laboratory (LANL)

The material stored at LANL is listed in Table 4.5.7. The fuel types are oxides, carbides, and nitrides clad in stainless steel. It is planned that these fuels will be packaged and shipped to another site (probably either SRS

or INEL) within the next few years. A sketch of the container planned for this purpose is provided in Fig. 4.5.2 (Serma 1986).

4.5.8 Oak Ridge National Laboratory (ORNL)

The materials held at ORNL are briefly described in Table 4.5.8. The largest portion (by weight) is the Consolidated Edison material, which is in the form of solidified U_3O_8 cake contained in 3.5-in.-OD × 24-in.-long stainless steel cans. The principal uranium isotopes present, as percentages of total U, are ^{233}U (9.7%), ^{234}U (1.4%), ^{235}U (76.5%), ^{236}U (5.6%), and ^{238}U (6.8%). This material has substantial radioactivity because its ^{232}U content is approximately 140 parts per million parts of total uranium. The storage containers are described in Fig. 4.5.3. An outer can encapsulates the stainless steel container. The primary container mates with a magnetic lifting tool, while the outer shell mates with a mechanical grasping device (McGinnis 1986). A conceptual waste disposal package for this material is shown in Fig. 4.5.4.

Substantial amounts of other materials are contained in ~3.5-in.-diam stainless steel cans of various lengths.

Fuel from the Molten Salt Reactor Experiment is described in greater detail in Sect. 4.5.10.

Also stored at ORNL are ten Peach Bottom Unit 1 Core 2 fuel assemblies. A description of these assemblies is given in Sect. 4.3.

4.5.9 Savannah River Site (SRS)

The miscellaneous fuels stored at the Savannah River Site (SRS) are listed in Table 4.5.9. The total quantity is about 19 MT of heavy metal (U + Pu + Th), of which the largest portions are fuels from Elk River (~5 MT), Dresden (~2 MT), Experimental Boiling Water Reactor (~7.5 MT), and Sodium Reactor Experiment (~2 MT). Since the early 1970s, SRS has required that incoming materials be accompanied by extensive descriptive matter, including drawings, compositions, etc. Such information is available from SRS on items of significant quantity (O'Rear 1987).

4.5.10 Molten Salt Reactor

Fuel from the Molten Salt Reactor Experiment (MSRE) is stored at Oak Ridge National Laboratory. The MSRE was a graphite-moderated, homogeneous-fueled reactor built to investigate the practicality of the molten-salt reactor concept for application to central power stations. It was operated from June 1965 to December 1969 at a nominal full-power level of 8.0 MW. The circulating fuel solution was a eutectic mixture of lithium and beryllium fluorides containing uranium fluoride as the fuel and zirconium fluoride as a chemical stabilizer. The initial fuel charge was highly enriched ^{235}U, which was later replaced with a charge of ^{233}U. Processing capabilities were included as part of the facility for on-line fuel additions, removal of impurities, and uranium recovery. A total of 105,737 MWh was accumulated in the two phases of operation (Notz 1985, 1987).

Following reactor shutdown, the fuel salt was drained into two critically safe storage tanks and isolated in a sealed hot cell, along with a third tank containing the flush salt. Figure 4.5.5 shows one of these drain tanks. The fission product activities (mainly beta-gamma) of these salts, decayed to 1985, total about 32,000 Ci. The alpha activity from transuranic isotopes and their daughters amounts to about 2,000 Ci. These isotopes are divided roughly 98–99% in the fuel salt and 1–2% in the flush salt. The total alpha activity of the fuel salt is about 400,000 nCi/g, while that of the flush salt is about 6,000 nCi/g. The total decay heat at present is about 150 W, with three-fourths coming from the beta-gamma component and the remainder from the alpha emissions. Figures 4.5.6 through 4.5.8 show the fission product activity (including Zr-93, an activation product), the actinide plus daughters activity, and the thermal output in watts, for decay times ranging from zero to one million years (Notz 1985).

4.5.11 Elk River (Rural Co-op) Reactor

The Elk River Rural Cooperative Power Association reactor at Elk River, Minnesota was a boiling-water reactor with a capacity of 22 MW(e). It was operated by United Power Association for the U.S. Atomic Energy Commission from 1962 to 1968 and was shut down in February 1968. Dismantling and removal of the facility was completed in 1974.

Table 4.5.10 summarizes the general characteristics of the reactor. The fuel assemblies were 5 × 5 pin arrays containing UO_2-ThO_2 pellets. Each assembly was 6.8 ft long and contained 26.84 kg of U + Th. The uranium enrichment was greater than 90%. Normal operating power was 58 MW(th), and normal burnup was about 9,500 MWd/metric ton of U + Th. Cross-sectional dimensions of the fuel assembly were about 3.5 in. × 3.5 in. Fuel pin diameter was 0.452 in. (AEC 1974).

The specific power of the Elk River reactor was 14.7 MW(th)/metric ton at a normal power level of 58.2 MW(th), indicating that the mass of fuel in the reactor was about 4.0 metric tons of U + Th. About 95% of the heavy metal was Th. The fuel loading was 26.84 kg U + Th per assembly, and the number of fuel assemblies in the reactor was 148. At a burnup of 9,500 MWd/metric ton U + Th, the residence time of the fuel in the reactor would be 646 days, or about two years.

Rural Cooperative Power Association is no longer in business. Telephoned information from the successor company, United Power Association, indicates that some spent fuel was removed from the reactor and shipped to Italy for reprocessing. Monthly operating reports by United

Power from the Elk River Reactor from the Elk River Reactor site during 1968 and 1969 (COO-651-55 through COO-651-79) make no mention of shipments to Italy, but state that at least three shipments of spent fuel were made to SRP for reprocessing. These shipments were described as being by rail cask containing 28 fuel assemblies per shipment. The cask was returned empty from SRP to the reactor site for reloading after each shipment, so there seems to be no doubt that the spent fuel was unloaded at SRP. This is confirmed by SRP's report of miscellaneous fuels on hand, as given in the IDB report, which shows Elk River Reactor spent fuel in the amount of 5,042.94 kg of U + Th (DOE 1988). These are described as UO_2-ThO_2 SS-clad assemblies, 3.5 × 3.5 × 81.62 in. The number of assemblies is not stated, but based on the 26.84 kg U + Th per assembly from other data sources, this would amount to 188 assemblies.

The total quantity of fuel that was at the Elk River reactor, as given in Elk River operating report COO-651-59 for the month of April 1968, was 7,737.7 kg U + Th. This included 4,025.6 kg U + Th in the reactor, 2,483.2 kg U + Th in the spent fuel storage pool, and 1,228.9 kg U + Th in the fresh fuel storage rack. Thus, the amount now in storage at SRS does not account for the total fuel at the reactor as of April 1968; the net difference unaccounted for is 2,594.8 kg U + Th. It appears, therefore, that some fuel may have been reprocessed. A paper by G. Orsenigo and S. Cambi presented at Gatlinburg, Tennessee, May 3–6, 1966, referred to plans for a reprocessing campaign on Elk River fuel (Orsenigo and Cambi 1966). However, no data have been located showing that such a campaign actually occurred.

4.5.12 Shippingport Reactor

The Shippingport Atomic Power Station, located at Shippingport, Pennsylvania, was the first large-scale central-station nuclear power plant in the United States. It started up in 1957, was permanently shut down in 1982, and was decommissioned over the ensuing few years. It had a net capacity of 60 MW(e). Initially it was a uranium-fueled pressurized light-water reactor (LWR), but it was converted around 1976–1977 to a light-water breeder reactor (LWBR) based on the U-233-thorium fuel cycle. The LWR started operation in 1957 and was shut down in 1974 to prepare for installation of the LWBR core. During its lifetime (1957 to 1974), the LWR used two cores of different design, referred to as PWR Core 1 and PWR Core 2. (Connors 1979). The PWR operations and fuel disposition have already been discussed in Sect. 2.6. Figures 4.5.9 and 4.5.10 show the LWBR reactor cross-sectional configuration and details of a blanket module. Table 4.5.11 summarizes the LWBR design characteristics and gives data on the fuel. The core had a total U + Th loading of 42,557 kg.

The Shippingport LWBR fuel was shipped to the Naval Reactors Expended Core Facility at Idaho Falls. These shipments are described in some detail in Selsley 1987. Three M-130 shipping containers were modified to accept LWBR seed, blanket, and reflector fuel modules for rail shipment. A total of 39 LWBR fuel modules were transferred in 10 shipments; however, the mass was not stated. Table 4.5.6 shows a total of about 57.1 MTHM Shippingport LWBR fuel now at ICPP. These materials are not currently scheduled to be reprocessed.

4.5.13 References for Section 4.5

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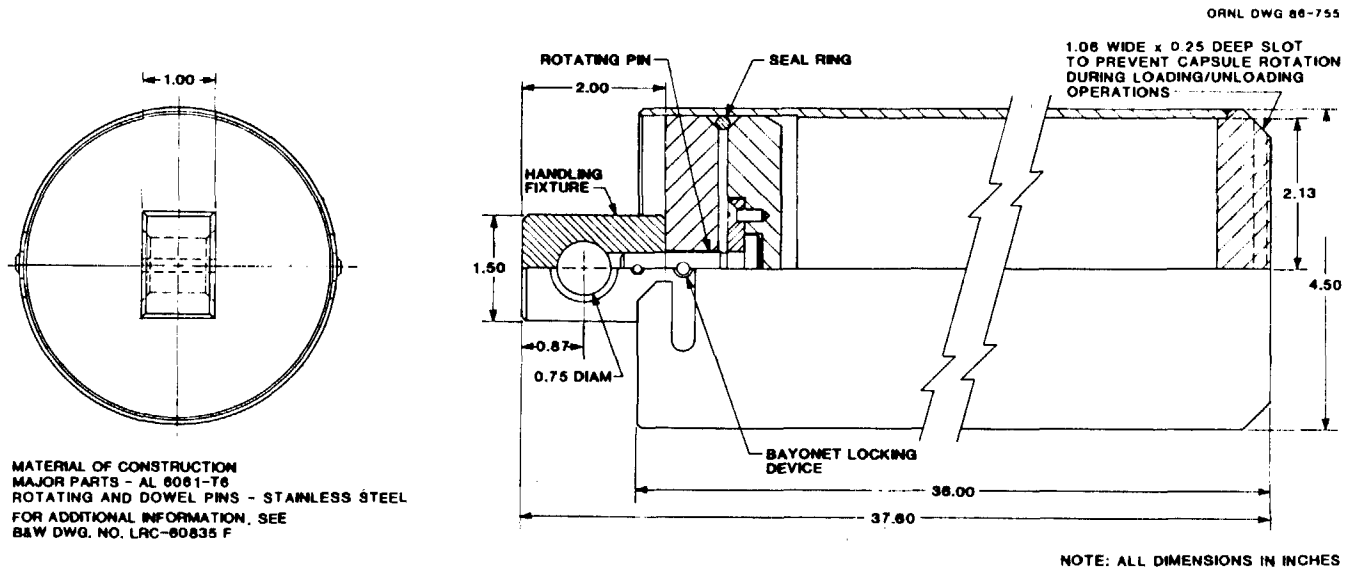

Fig. 4.5.1. Babcock and Wilcox spent fuel container.

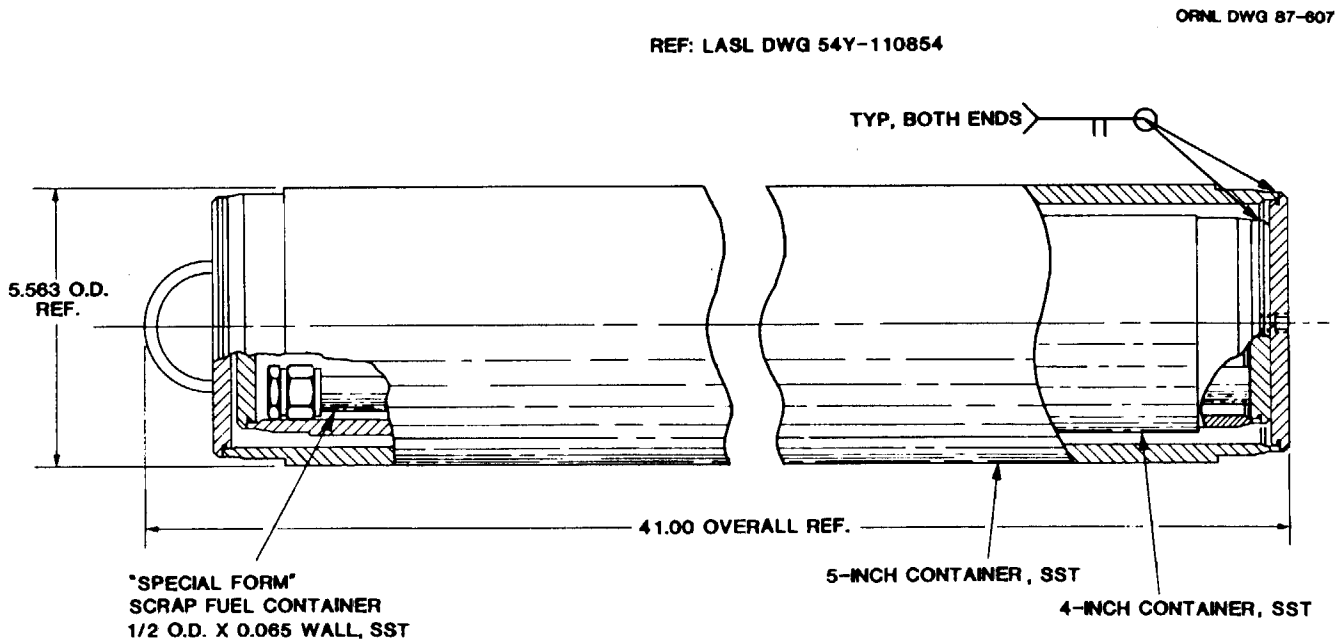

Fig. 4.5.2. The Los Alamos National Laboratory multicontainer system for scrap fuel.

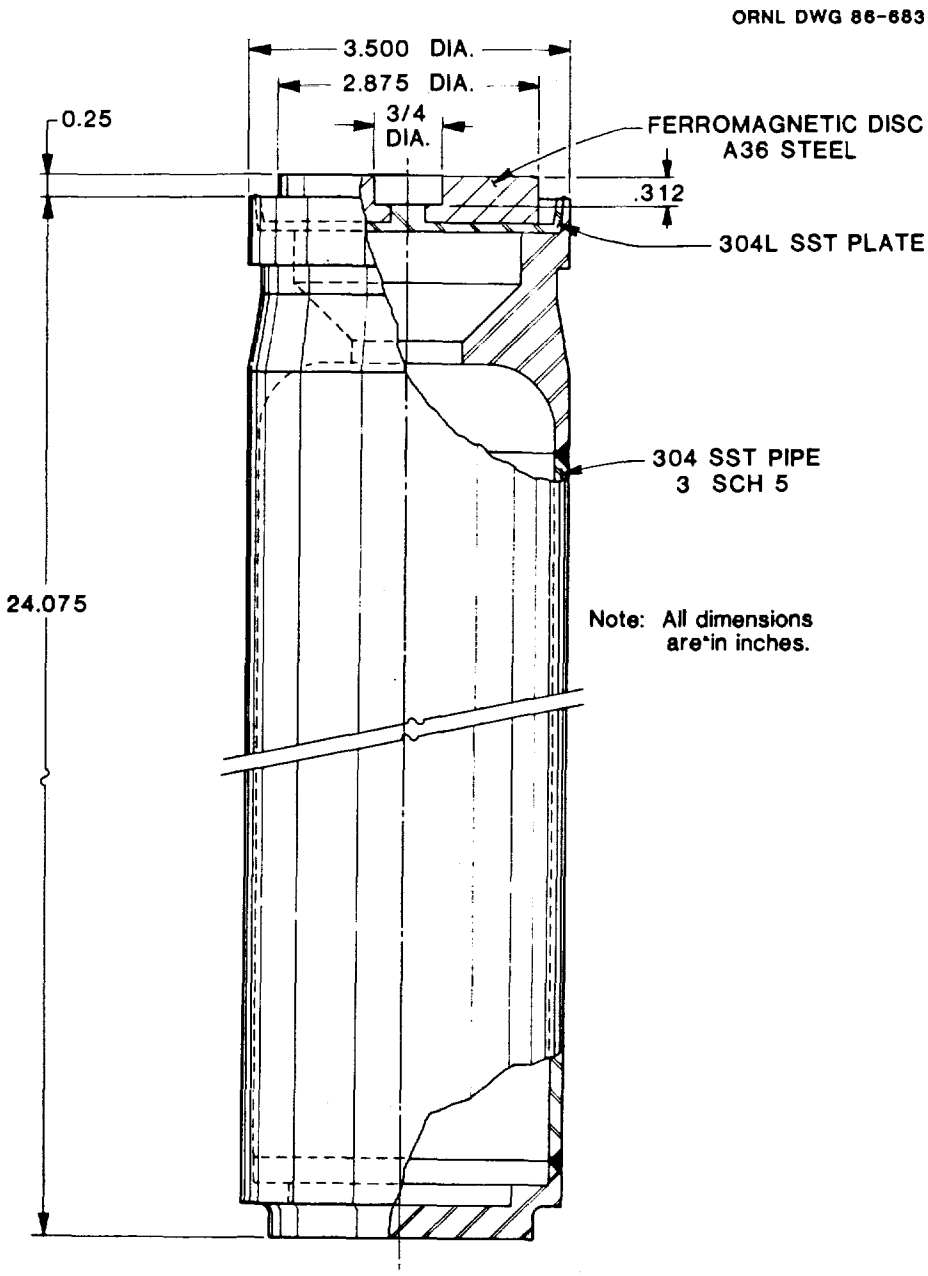

Fig. 4.5.3. Storage can assembly for CEU solidified waste.

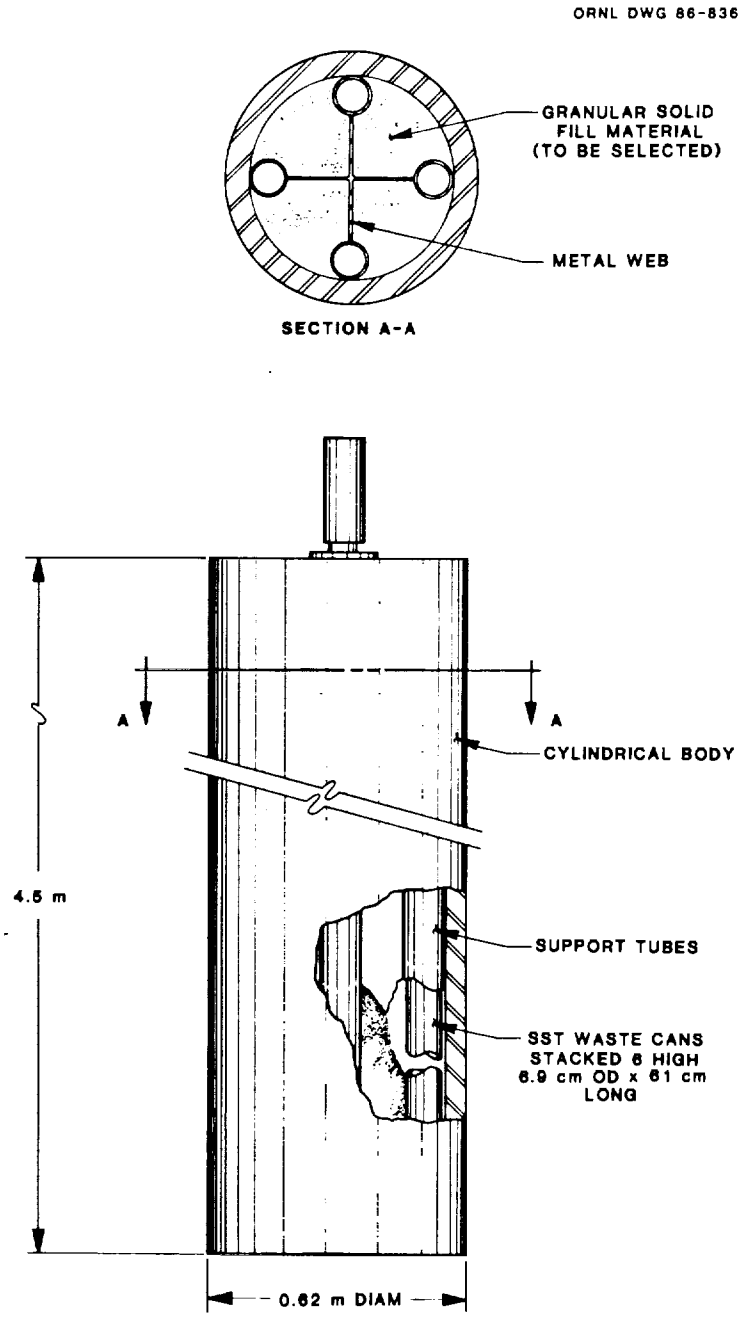

Fig. 4.5.4. CEU waste disposal package concept.

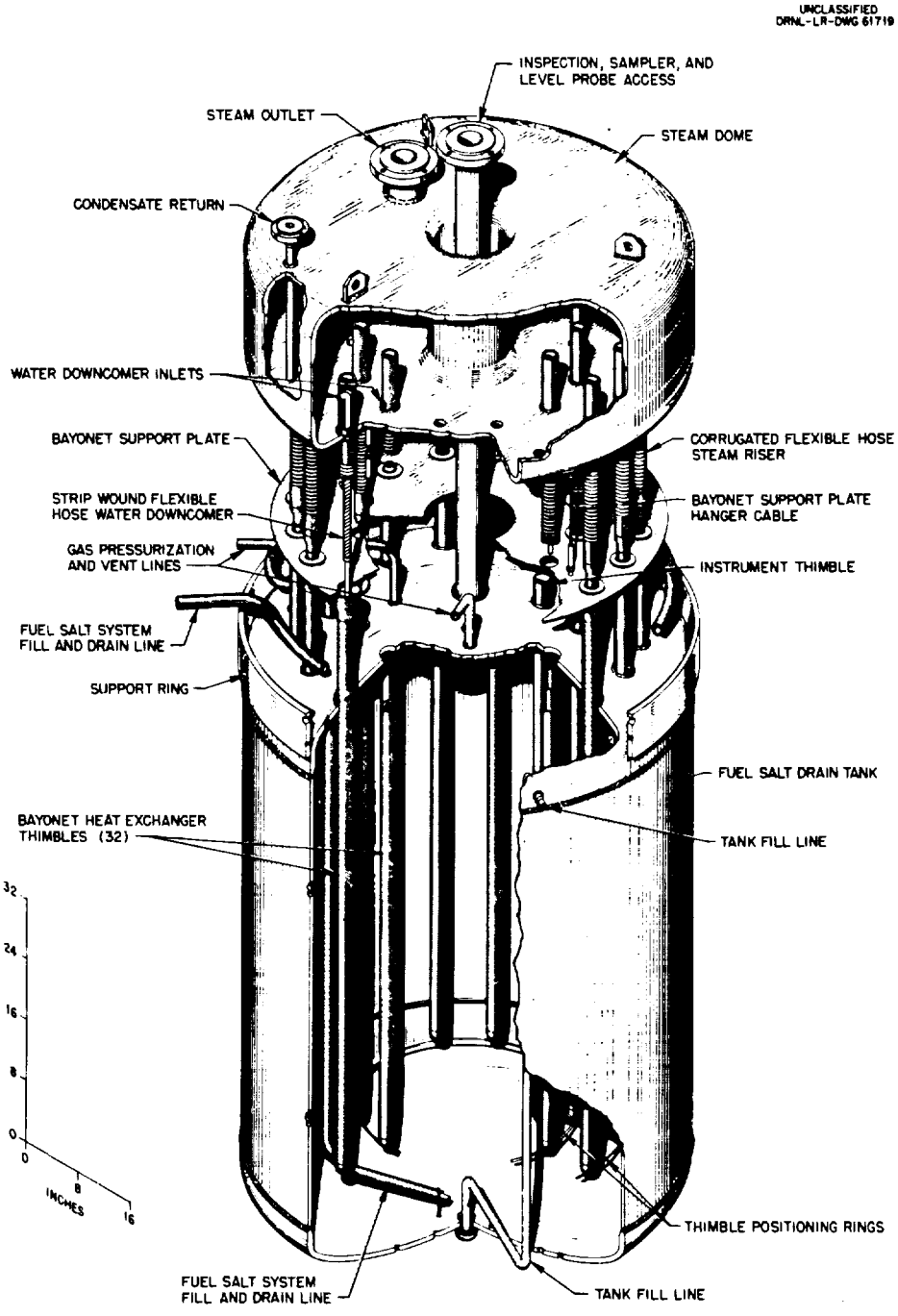

Fig. 4.5.5. MSRE fuel-salt drain tank.

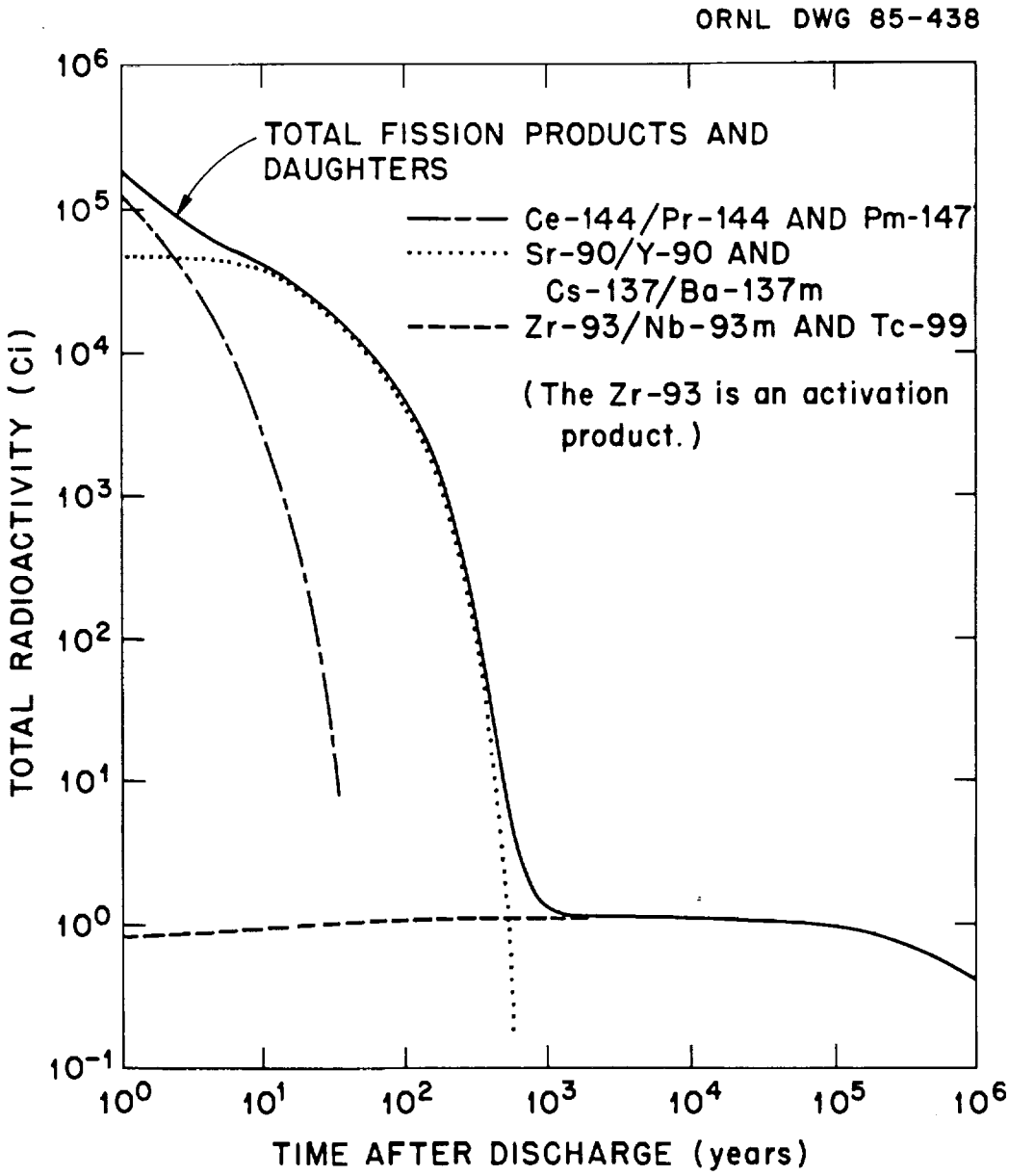

Fig. 4.5.6. Fission product activities of MSRE fuel and flush salts.

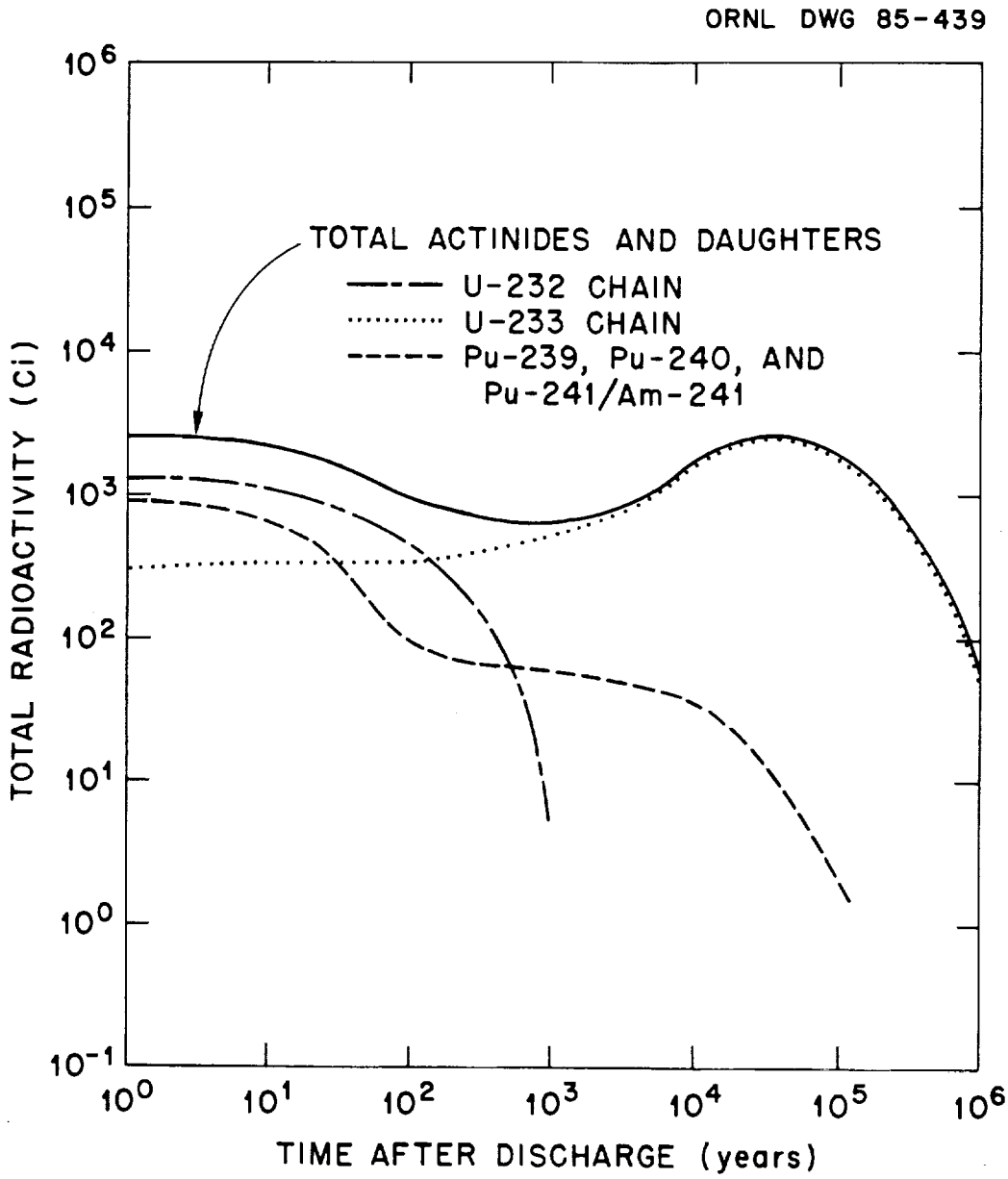

Fig. 4.5.7. Actinide and daughter activities of MSRE fuel and flush salts.

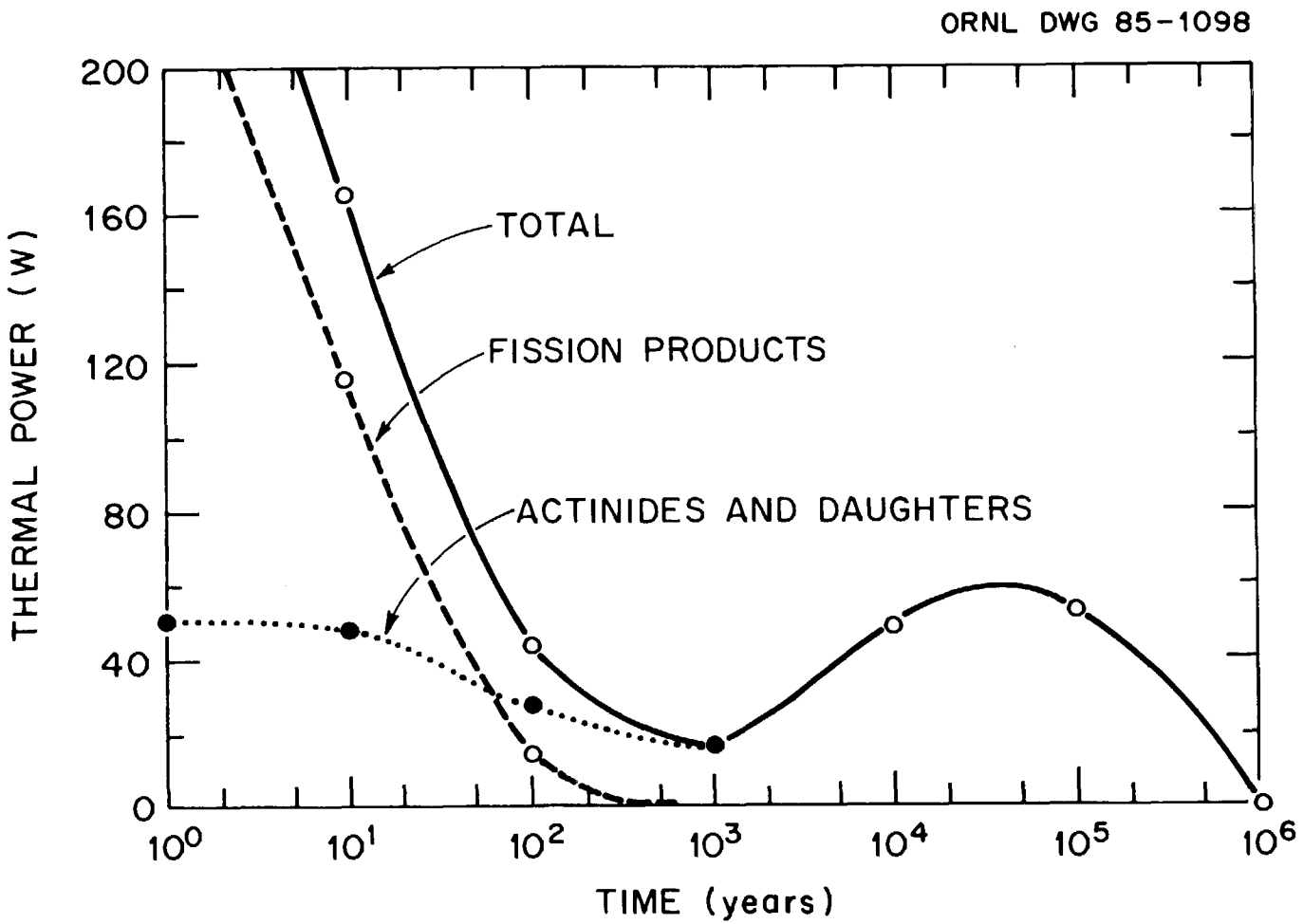

Fig. 4.5.8. Thermal output of MSRE fuel and flush salts.

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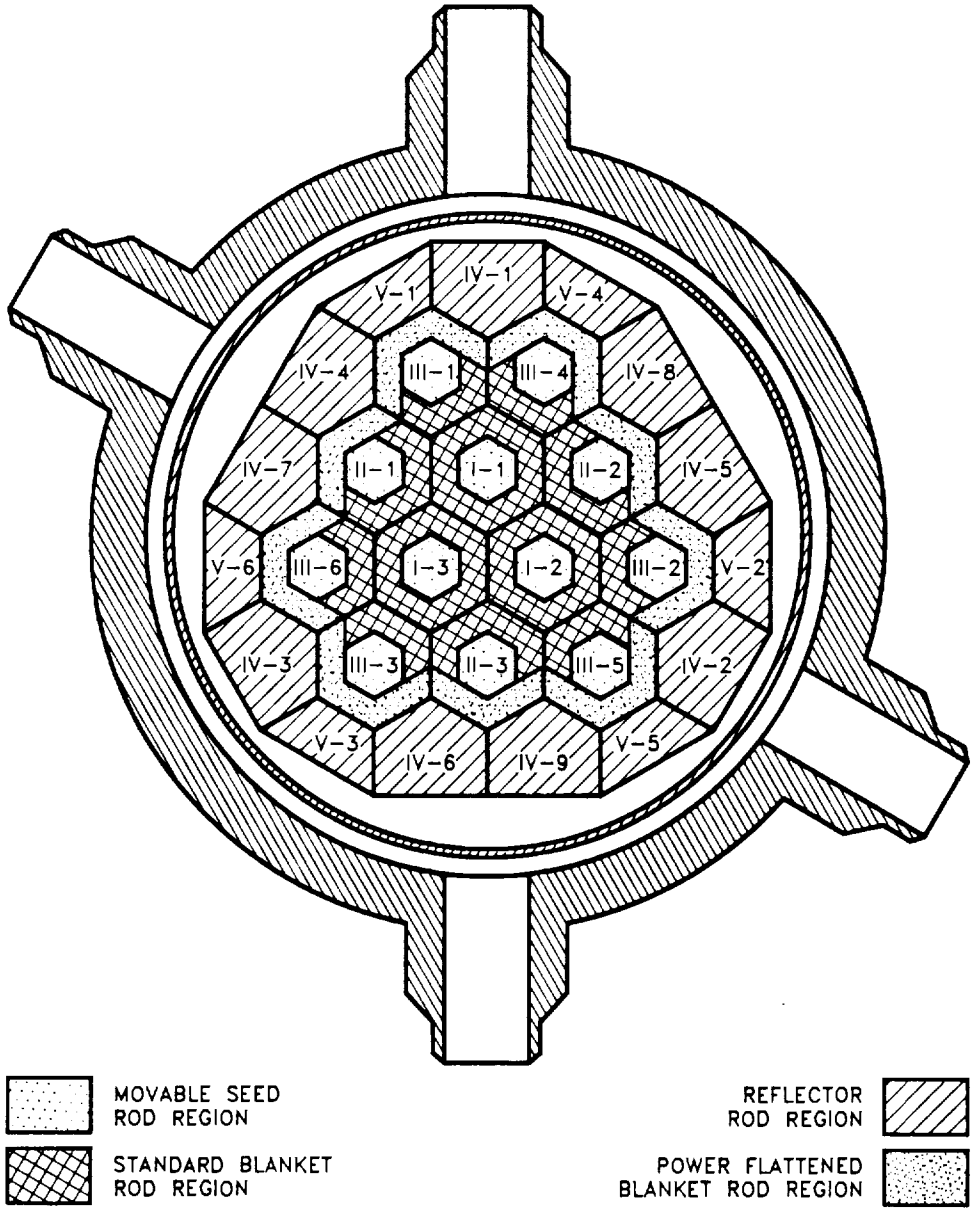

Fig. 4.5.9. Shippingport LWBR core cross section.

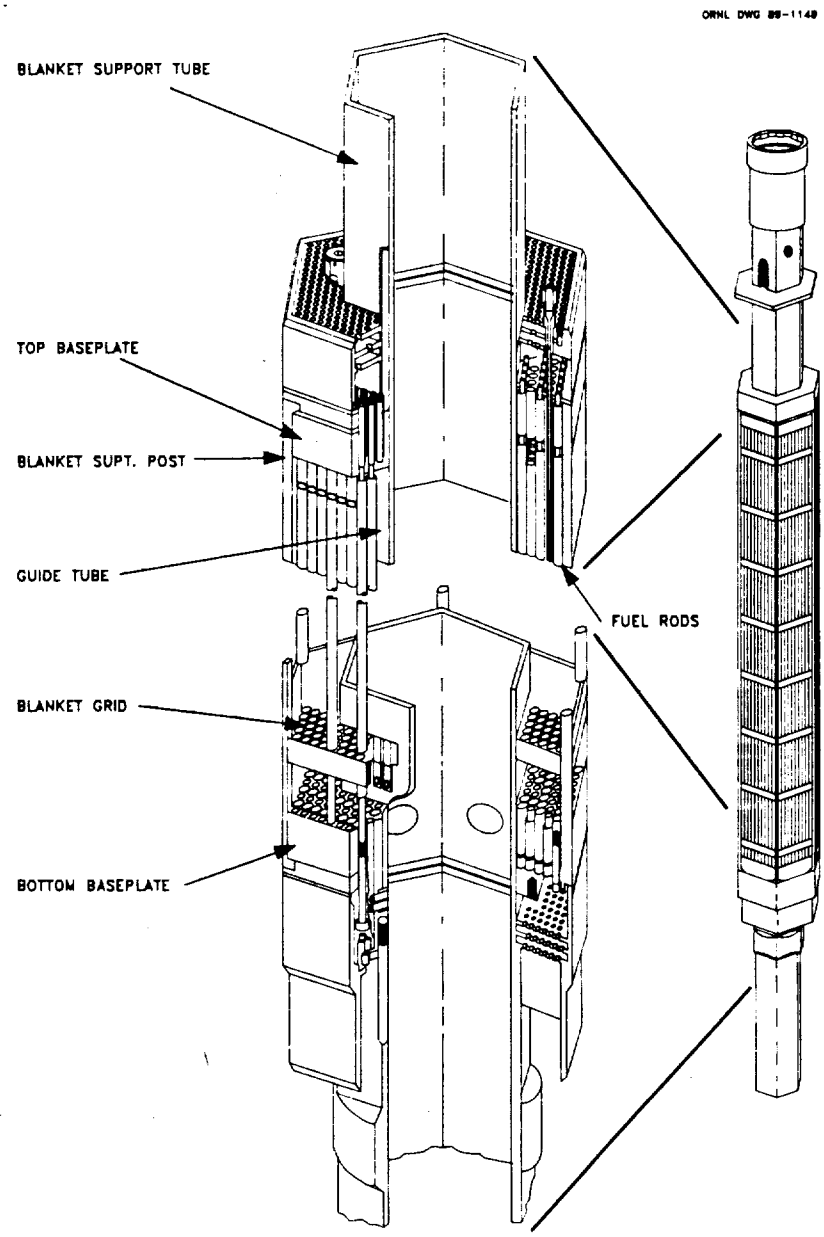

Fig. 4.5.10. Shippingport LWBR blanket module.

Table 4.5.1. Total quantities of miscellaneous radioactive materials stored at various sites as of December 31, 1989

Storage site and location	Total candidate materials (kg)	Uranium content, kg			Total plutonium content (kg)	Total thorium content (kg)
		Total	^{235}U	^{233}U[a]		
Reported potential miscellaneous materials inventory						
Argonne National Laboratory-West; Idaho Falls, ID	311.60	302.65	20.050		8.950	
Babcock & Wilcox, Naval Nuclear Fuel Division (NNFD) Research Laboratory; Lynchburg, VA	88.45	87.66	1.379		0.790	
Battelle Pacific Northwest Laboratory; Richland, WA	2,347.9	2,311.9	21.6		29.3	6.7
Hanford 200-Area burial grounds; Richland, WA	263.33	230.35	42.21		32.98	
Idaho National Engineering Laboratory; Idaho Falls, ID[b]	148,560.16	81,339.36	1,936.47	959.46	273.80	66,947.0
Los Alamos National Laboratory; Los Alamos, NM	38.03	31.68	22.45	0.134	6.35	
Oak Ridge National Laboratory; Oak Ridge, TN	1,253.72	1,252.92	798.7	280.29	0.801	
Savannah River Site; Aiken, SC	19,110.39	10,419.52	761.04	31.16	42.67	8,648.2
Total reported	171,973.58	95,976.04	3,603.90	1,271.04	395.64	75,601.9
Estimated potential miscellaneous materials inventory						
Three Mile Island-Unit 2[c]	82,023	82,023	2,064.4			

[a]Some of the ^{233}U waste may be certifiable as TRU waste.

[b]Many of the fuels at ICPP have a lower uranium enrichment than that of fuels normally processed. These fuels could be reprocessed in a special campaign, if required.

[c]The quantity shown (82,023 kg) is the initial fuel loading, kg HM. The total spent fuel and core debris shipped to INEL as of the end of 1989 was 145.3 MT (Conaway 1990).

Table 4.5.2. Miscellaneous radioactive materials stored at Argonne National Laboratory-West, as of December 31, 1988[a]

Source of material	Composition	Description[b]	U content, kg		Total Pu content (kg)
			Total	^{235}U	
Radioactive Waste and Scrap Facility[c]					
Basic research – ANL	Scrap	Stored in canister[d]	182.00	12.980	5.052
EBR-2 blanket subassembly	Scrap	Stored in canister[d]	104.80	0.230	0.180
LMFBR test fuel	Scrap	Stored in canister[d]	13.33	5.253	3.026
Postirradiation test on NUMEC LMFBR	Scrap	Stored in canister[d]	0.72	0.345	0.123
Sodium Loop Safety Facility	Scrap	Stored in canister[d]	1.80	1.242	0.569
Total			302.65	20.050	8.950

[a]Ref. Wallace 1990.

[b]No information regarding the burnup of this scrap is available.

[c]Radioactive Scrap and Waste Facility is located approximately 0.5 miles north of ANL-W site.

[d]Canisters are retrievable and constructed of stainless steel with minimum dimensions of 8-in. OD and 5-ft length. The canister lid is gasketed and tightly screwed on, welded closed, or screwed into a canister fitted with pipe threads.

Table 4.5.3. Miscellaneous radioactive materials stored at Babcock & Wilcox, NNFD Research Laboratory, as of December 31, 1989[a]

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg		Total Pu content (kg)
				Total	^{235}U	
Arkansas I	UO_2, Zr-clad	Stored in four 4.25-in.-diam x 33-in. Al canisters	47,000	11.761	0.046	0.133
B&W Test Reactor	UO_2, Zr-clad	Stored in fourteen 4.25-in.-diam x 33-in. Al canisters	Unknown[c]	0.015	0.005	<0.0005
Consolidated Edison	UO_2, Zr-clad	Stored in a 4.25-in.-diam x 33-in. Al canister	29,523	10.849	0.060	0.088
Oconee I	UO_2, Zr-clad	Stored in twenty-six 4.25-in.-diam x 33-in. Al canisters	18,686	0.531	0.004	0.003
			24,080	2.159	0.028	0.017
			26,480	6.482	0.033	0.056
			31,160	4.275	0.041	0.037
			39,180	11.000	0.057	0.101
			50,000	10.579	0.037	0.117
Oconee I	UO_2-Gd_2O_3, Zr-clad	Stored in four 4.25-in.-diam x 33-in. Al canisters	15,000	7.911	0.103	0.048
Oconee II	UO_2, Zr-clad	Stored in seven 4.25-in.-diam x 33-in. Al canisters	17,000	10.711	0.105	0.095
			31,000	6.329	0.057	0.056
			36,000	2.105	0.015	0.019
TMI-Unit 2	UO_2 debris	Stored in a 4.25-in.-diam x 33-in. Al canister	Unknown[c]	0.047	0.0307	<0.0005
Various fuel scrap samples	UO_2, Zr-clad	Stored in a 4.25-in.-diam x 33-in. Al canister	Unknown[c]	2.908	0.757	<0.0005
Hot cell solid waste	Miscellaneous[d]	Stored in forty-one 80-gal drums, thirty-three 55-gal drums, and fifty-two 30-gal drums	-	e	e	<0.082[f]
Total				87.662	1.379	0.790

[a]Ref. Long 1990.
[b]Zr-clad = Zircaloy-clad.
[c]Currently in underground storage tubes.
[d]Miscellaneous materials from periodic hot cell cleanup.
[e]Negligible.
[f]Calculated assuming a contaminated level of <0.5 g of plutonium per drum.

Table 4.5.4. Miscellaneous radioactive materials stored at Battelle Pacific Northwest Laboratory, as of December 31, 1989[a]

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg		Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U		
Calvert Cliffs	UO_2, Zr-clad	0.440-in. diam x 147 in.					
		(stored as 175 intact rods, 1 cut rod[c])	30,000	370.5	2.6	5.3	
		(stored as 154 intact rods, 1 cut rod[c])	45,000	293.2	1.7	7.7	
Cooper	UO_2, Zr-clad	98 rods[c]	26,000	365.3	2.5	3.1	
Point Beach-1	UO_2, Zr-clad	Stored as three intact fuel assemblies, miscellaneous cut samples	32,000	1,163.6	10.3	10.6	6.7
H. B. Robinson	UO_2, Zr-clad	Stored as 19 cut fuel rod sections[c]	30,000	30.2	2.2	0.2	
Shippingport				3.9	0.1	0.1	
VBWR[d]	UO_2, Zr-clad	Twelve 3-ft fuel rod segments	20,000-30,000	11.1	0.1	0.7	
PNL Lot Numbers:							
ATM-5		Glass mix		0.1	e	<0.1	
ATM-6		Glass mix		0.1	e	<0.1	
Miscellaneous scrap and fuel	Cut pieces, scrap	Stored in hot cells		68.5	2.0	1.5	
Miscellaneous fuel	Cut pieces	Stored in hot cell	Unknown	5.4	0.1	0.1	
Total				2,311.9	21.6	29.3	6.7

[a]Ref. Dickman 1990.
[b]Zr-clad = Zircaloy-clad.
[c]Stored in a hot cell.
[d]Vallecitos boiling-water reactor.
[e]Negligible.

Table 4.5.5. Miscellaneous radioactive materials stored at the Hanford 200 Area burial grounds, as of December 31, 1989[a]

Source of material	Composition	Description[b]	U content, kg		Total Pu content (kg)
			Total	^{235}U	
EBR II (Experimental Breeder Reactor) From INEL	UO_2/PuO_2, SS-clad	Stored in four 30-in.-diam × 59.5-in. shielded carbon steel casks	45.53	7.64	3.60
From LANL	UO_2/PuO_2, SS-clad	Stored in eight 30-in.-diam × 59.5-in. shielded carbon steel casks	29.18	17.57	14.19
From INEL and FFTF (Fast Flux Test Facility) at Hanford	UO_2/PuO_2, SS-clad	Stored in five 30-in.-diam × 59.5-in. shielded carbon steel casks	34.65	7.55	9.81
Fast Critical Facility and SEFOR (Southwest Experimental Fast Oxide Reactor) from GE, Vallecitos, CA	UO_2/PuO_2	Stored in twenty-two 75.5-in. × 65.5-in. × 65.5-in. concrete casks	40.49	4.88	4.70
K reactor	Unknown	12 americium target elements stored in one 30-in.-diam × 69-in. Zircaloy container	0.024[c]	0.024	.074
LWR from GETR,[d] Monticello Reactor, Quad Cities 1 Reactor, and Millstone Reactor	UO_2 pellets	Stored in six 30-in.-diam × 59.5-in. shielded carbon steel casks	63.28	1.29	0.59
TRIGA (Training Reactor, Isotopes, General Atomic) from Oregon State University	Zr-U hydride (8 wt % U), Al-clad	3.6-cm diam × 72 cm fuel assemblies stored/buried in thirteen 55-gal concrete-filled drums, six to seven assemblies per drum	17.2	3.26	0.013
Total			230.35	42.21	32.98

[a]Ref. Turner 1990.
[b]No information regarding the burnup of this fuel is available.
[c]Enrichment of uranium not provided.
[d]General Electric (GE) Testing Reactor.

Table 4.5.6. Miscellaneous radioactive materials stored at the Idaho National Engineering Laboratory, as of December 31, 1989[a]

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Defense plus other government agency material stored at ICPP								
GCRE (Gas-Cooled Reactor Experiment)	UO_2-BeO, Hastelloy X clad	One SS tube, 5 in. × 25.5 in.		0.984	0.918			
LWBR (Shippingport Light-Water Breeder Reactor)	Ceramic pellets, Zr-clad, Th blanket	65 units		982.173	10.349	826.016	0.177	56,167.0
Misc. fuels and scrap	Scrap	Stored in 92 SS and Al cans		168.195	137.330	0.119	0.079	36.0
PWR Core 2 (Shippingport Pressurized-Water Reactor)	UO_2 pellets, Zr-clad	28 units		392.026	305.802			
SM-1A (Stationary Media)	UO_2, SS-clad	Stored in 93 SS cans		65.759	56.648			
TORY-11A	UO_2-BeO crushed to 0.25 in. × 0.06 in.	Stored in 147 Al cans 3.25 in. × 1.5 in.		48.645	45.325			
TORY-11C	UO_2-Y_2O_3-ZrO_2-BeO ceramic	Stored in three Al cans 2.68 in. × 52.5 in.		59.065	55.022			
Subtotal				1,716.847	611.394	826.135	0.256	56,203.0
DOE/Civilian Development Programs material stored at ICPP								
EBR Scrap (Experimental Breeder Reactor)	Scrap			1.618	0.839			
Fermi 1 Blanket	U-Mo (97% U), sodium-bonded, SS-clad	Stored in 510 SS cans, 0.4-in. diam × 41 in. or 61 in.		34,165.000	120.000		6.522	
FSVR (Fort St. Vrain Reactor)	U-Th carbide and Th carbide, pyrolytic carbon-coated particles in graphite matrix	732 hexagonal graphite blocks 14.2 in. across flats × 31.2 in.		299.758	164.431	87.013	0.752	8,124.0

Table 4.5.6 (continued)

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Civilian Development Programs material stored at ICPP (continued)								
Pathfinder	UO_2-B_4C pellets, SS-clad	417 rods in 17 cans; each can is 9-in. diam x 80 in.		53.406	48.242			
Peach Bottom	U-Th carbide, pyrolytic carbon-coated particles in graphite matrix	1,603 graphite blocks 3.5-in. diam x 12 ft	>1[c]	332.420	223.540	46.310	0.870	2,620.0
Pulstar, State University of New York at Buffalo	UO_2 pellets in Zr-clad pins	Stored in 24 SS cans, 3 in. x 3 in. x 35.5 in.		251.431	12.083		0.793	
TRIGA (Training Reactor, Isotopes, General Atomic)	Al- or SS-clad elements	852 units stored in 121 cans		160.974	33.839			
VBWR (Geneva) (Vallecitos Boiling-Water Reactor)	UO_2 and UO_2-TiO_2, SS-clad	142 rods stored in four 6-in.-diam x 36-in. Al cans	8[c]	12.383	2.606			
Subtotal				35,276.890	606.580	133.323	9.037	10,744.0
DOE material stored at NRF[d]								
Shippingport PWR Core 1	UO_2 pellets, Zr-clad	Seed and blanket fuel assemblies	11,100	570.02	1.63		3.4	
Shippingport PWR Core 2	UO_2 wafers, Zr-clad	Seed and blanket fuel assemblies	14,273	1,260.92	164.45		8.9	
Subtotal				1,830.94	166.08		12.3	
DOE/Civilian Development Programs material stored at INEL (other than ICPP and NRF)								
CANDU (Canadian Deuterium Reactor)	UO_2 pellets, Zr-clad	8 pins	5,000	2.660	0.261			
Connecticut Yankee	UO_2, Zr-clad	1 assembly		378.485	5.204		3.774	

Table 4.5.6 (continued)

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Civilian Development Programs material stored at INEL (other than ICPP and NRF) (continued)								
Dresden	UO_2, Zr-clad	54 pins (depleted U)		165.0	Unknown		1.064	
EMAD[e] (Engine Maintenance Assembly & Disassembly)	UO_2 pellets, Zr-clad	18 assemblies	25,000-30,000	7,831.273	58.103		65.255	
GAP CON (Gap Conductance)	UO_2 pellets, Zr-clad	20 pins	42-115	12.838	1.285			
GE (General Electric)	UO_2 pellets, Zr-clad	Pins		18.644	0.394		0.071	
Halden Assy	UO_2 pellets, Zr-clad	5 pins	4,000	2.313	0.233		0.005	
Halden 226 and 239 Assy	UO_2-PuO_2 pellets, Zr-clad	12 pins					0.324	
IE (Irradiation Effects)	UO_2 pellets, Zr-clad	Pins	27-17,600	7.833	0.867		0.012	
LLR (LOFT Lead Rod)	UO_2 pellets, Zr-clad	7 pins	36-150	3.510	0.327			
LOC (Loss of Coolant)	UO_2 pellets, Zr-clad	60 pins	16-150	7.777	0.816		0.010	
LOFT (Loss of Fluid Test)	UO_2 pellets, Zr-clad	15+ assemblies	0-1,050	2,201.696	89.371		2.029	
MAPI (Mitsubishi Atomic Power Industries)	UO_2 pellets, Zr-clad	43 pins	2,990-8,770	22.499	1.267		0.032	
Miscellaneous fuel pins	UO_2 pellets, Zr-clad	Pins	Varies	173.354	1.758		2.626	
Miscellaneous rods and scrap	Scrap	Stored in 8 cans	Varies	13.553	1.197			
OPTRAN (Operational Transient)	UO_2 pellets, Zr-clad	Pins	0-15,000	19.669	0.472		0.087	

Table 4.5.6 (continued)

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Civilian Development Programs material stored at INEL (other than ICPP and NRF) (continued)								
PBF (Power-Burst Facility)	UO_2-ZrO_2-CaO; Zr sleeves, SS-clad	Pins		725.690	132.890			
PCM (Power Coolant Mismatch)	UO_2 pellets, Zr-clad	30 pins	<70	18.828	6.557			
Peach Bottom	UO_2 pellets, Zr-clad	1 assembly and pieces		364.1	2.512		1.878	
RIA (Reactivity Initiated Accident)	UO_2 pellets, Zr-clad	23 pins	0-6,090	8.989	0.504		0.013	
H. B. Robinson	UO_2 pellets, Zr-clad	Pins	28,000	263.916	1.890		2.153	
Saxton	UO_2 pellets, Zr-clad	21 pins	10,400-18,530	7.607	0.660		0.025	
SFD (Severe Fuel Damage)	UO_2 pellets, Zr-clad	143 pins		50.867	2.711		0.150	
TC (Thermocouple)	UO_2 pellets, Zr-clad	Pins	0-<20	6.186	0.683			
TMI-Unit 2	Rubble			(Quantities unknown until entire core received)				
VEPCO (Virginia Electric Power Company)		69 assemblies		30,207.295	242.457		172.695	
Subtotal				42,514.582	552.419		252.203	
Total at INEL				81,339.359	1,936.473	959.458	273.796	66,947.0

[a]Refs. Berreth 1990 and Connors 1990. Many of the fuels at INEL have lower uranium enrichment than is found in those fuels that are normally processed. These fuels could be reprocessed in a special campaign, if required.
[b]Zr-clad = Zircaloy-clad.
[c]Data expressed in percentage.
[d]Based on Connors 1990.
[e]Turkey Point Fuel.

Table 4.5.7. Miscellaneous radioactive materials stored at the Los Alamos National Laboratory, as of December 31, 1989[a]

Source of material	Composition	Description	U content, kg			Total Pu content (kg)
			Total	^{235}U	^{233}U	
EBR-2	U-Pu oxide, carbide or nitride SS-clad fuel rod segments	0.3-in. diam x 13.5 in.[b]	26.08	17.71	0.134	6.35
B&W (Lynchburg, VA)	UO_2 spent fuel elements	Stored in racks	5.60[c]	4.74		
Total			31.68[c]	22.45	0.134	6.35

[a]Ref. Erkkila 1990.
[b]No information regarding the burnup of this fuel is available.
[c]Includes 0.348 kg of ^{236}U.

Table 4.5.8. Miscellaneous radioactive materials stored at the Oak Ridge National Laboratory, as of December 31, 1989

Source of material	Composition[a]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)
				Total	^{235}U	^{233}U	
CEU (Consolidated Edison Uranium)	U_3O_8-CdO solid cake	Stored in 401 3.5-in.-OD x 24-in. SS cans	b	1,044.38	797.70	101.32	
Dresden-1	UO_2, Zr-clad	Sheared fuel pins stored in two 1-qt paint cans	~24,000	5.00	0.024		0.020
		9/16-in.-diam x 8-in. fuel rod sections plus short lengths	20,000	0.930	0.005		0.006
GETR (General Electric Test Reactor)	UO_2, Zr-clad	9/16-in.-diam x 8-in. fuel test capsules	1,000-2,000	0.399	0.022		
Monticello	UO_2, Zr-clad	1/2-in.-diam x 6-in. fuel rod sections plus short lengths	40,000	1.00	0.004		0.008
MSRE[c] (Molten Salt Reactor Experiment)	LiF_2-BeF_2-ZrF_2-UF_4	See ref. 13	~5 x 10^4 Ci total (see ref. 13)	36.95	0.940	31.01	0.743
Oconee-1	UO_2, Zr-clad	1/2-in.-diam x 6-in. fuel rod sections plus short lengths	38,000	1.00	0.005		0.005
Peach Bottom-2	UO_2, Zr-clad	9/16-in.-diam x 8-in. fuel rod sections plus short lengths	10,000	0.324	0.001		0.001
Quad City-1	UO_2, Zr-clad	1/2-in.-diam x 6-in. fuel rod sections plus short lengths	40,000	1.00	0.004		0.008
H. B. Robinson	UO_2, Zr-clad	1/2-in.-diam x 12-in. fuel rod sections plus short lengths	30,000	1.00	0.005		0.004
BR-3 (Belgium)	UO_2, Zr-clad	3/8-in.-diam x 6-in. fuel rod lengths	42,000	0.837	0.020		0.006
ORNL Inventory Item Nos. AUA-67/AUA-70 from LANL	U metal chunks	Stored in two 3.75-in.-OD x 18-in. SS cans	b	6.02		5.89	
CZA-91 from ANL	UO_x powder	Stored in two 3.5-in.-OD x 13-in. SS cans	b	0.881		0.856	

Table 4.5.8 (continued)

Source of material	Composition[a]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)
				Total	^{235}U	^{233}U	
HUA-2A from HEDL	UO_x powder	Stored in five 3.75-in.-OD x 7-in. SS cans	b	0.317		0.307	
LAE-03	Metal	Stored in one 3-in.-OD x 10-in. SS can	b	0.01		0.01	
RCP-02 from SRO	UO_2 powder	Stored in thirty-two 3.5-in.-OD x 24-in. SS cans	b	11.14		10.72	
RCP-03 from SRO	UO_2 powder	Stored in 140 3.88-in.-OD x 10-in. SS cans	b	67.41		61.61	
RCP-04 from SRO	UF_4-LiF_2 powder converted from UO_2	Stored in six 3.5-in.-OD x 24-in. SS cans	b	3.19		2.92	
RCP-06	U_3O_8-CdO solid cake	Stored in twenty-seven 3.5-in.-OD x 24-in. SS cans	b	65.55		60.60	
RCP-20/JZBL from LANL	U metal chunks	Stored in five 3.5-in.-OD x 24-in. SS cans	b	5.15		5.05	
Total				1,252.92	798.7	280.29	0.801

[a]Zr-clad = Zircaloy-clad.

[b]No information regarding the burnup of this fuel is available.

[c]The Molten Salt Reactor Experiment was concluded in 1969, and the fuel has never been removed from the facility. A surveillance and monitoring program has been in force since shutdown. See Notz 1988.

Table 4.5.9. Miscellaneous radioactive materials stored at the Savannah River Site, as of December 31, 1989[a]

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Civilian Development Programs material stored at SRS								
CANDU (Canadian Deuterium Reactor)	UO_2, Zr-clad	Rods stored in three 5.0-in.-diam x 14-ft cans; pieces stored in three 3.5-in.-diam x 1-ft cans	6,500	50.22	0.231			
Carolinas-Virginia Tube Reactor	UO_2-Zr or SS-clad	One bundle of 34 rods in a 5.0-in.-diam x 14-ft can	Unknown	67.37	0.640		0.200	
Dresden	UO_2-ThO_2, SS-clad	Intact assemblies stored in 4.4-in. x 4.4-in. x 135-in. cans	4,000-10,000	683.88	37.545	15.391	1.879	1,857.2
ERR (Elk River Reactor)	UO_2-ThO_2, SS-clad	Assemblies 3.5 in. x 3.5 in. x 81.62 in.	Max. 50,000	224.34	186.159	14.722		4,818.6
LWR samples (Light-Water Reactors)	UO_2-PuO_2, SS- and Zr-clad	Fuel rod pieces stored in five 3.75-in.-diam x 32.5-in.-long cans	Unknown	12.631	0.192		0.109	
Nereide (a French Experiment using DOE fuel)	UAl-Si_x, Al-clad	Materials Test Reactor plate-type fuel assembly 34.37 in. x 2.98 in. x 3.14 in.	600	35.45	7.015			
H. B. Robinson	UO_2-PuO_2, Zr-clad, SS casing	Four 6- to 8-in.-long fragments in 4.5-in.-diam x 32-in.-long can	6,800-30,000	0.52	0.004		0.003	
Saxton	UO_2-PuO_2, Zr- or SS-clad	567 rods stored in eight 5.0-in.-diam x 14-ft cans and 64 rods stored in one 3.75-in.-diam x 50-in. can	1,000	276.67	1.411		15.408	
	UO_2, Zr-clad	Multiple pins stored in four 5.0-in.-diam x 14-ft cans and one bundle stored in one 12-in.-diam x 14-ft can	1,600	66.79	6.866		0.233	

Table 4.5.9 (continued)

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Civilian Development Programs material stored at SRS (continued)								
VBWR (Vallecitos Boiling-Water Reactor)	UO_2, Zr-clad	Stored in four 3.5-in.-diam x 12-in. cans	1,500	4.04	0.998		0.003	
Subtotal				1,421.911	241.061	30.113	17.835	6,675.8
DOE/Defense plus other government agencies material stored at SRS								
B&W scrap	UO_2-PuO_2, SS-clad	Stored in 3.5-in.-diam x 32-in. cans	6-54	0.025	0.013		0.048	
EBR-2 (Experimental Breeder Reactor)	UO_2-PuO_2, SS-clad (from ANL)	Eight rods stored in a 3.5-in.-diam x 30-in. can	120 kW total in 1975	0.44	0.376		0.114	
	UO_2-PuO_2, SS-clad (from HEDL)	Rod segments stored in 0.5-in.-diam x 42-in. cans	10,000-34,000	2.04	1.624		0.680	
EBWR (Experimental Boiling-Water Reactor)	UO_2, SS-clad	Assemblies 3.75 in. x 3.75 in. x 62.5 in.	1,600	1.73	1.612			
	UO_2, Zr-clad	Assemblies 3.75 in. x 3.75 in. x 62.5 in.	1,600	1,604.30	95.456			
	UO_2-Zr, Zr-clad	Assemblies 3.75 in. x 3.75 in. x 62.5 in.	1,600	5,031.77	73.967		9.092	
	UO_2-ZrO_2-CaO, Zr-clad	Assemblies 3.75 in. x 3.75 in. x 62.5 in.	1,600	28.93	26.651			
	UO_2-PuO_2, Zr-clad	Assemblies 3.75 in. x 3.75 in. x 62.5 in.	1,600	907.39	2.087		13.952	
EPR-1	PuO_2, SS-clad	Pieces stored in 4.5-in.-diam x 32-in. can	Unknown				0.022	

Table 4.5.9 (continued)

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Defense plus other government agencies material stored at SRS (continued)								
GCRE (Gas-Cooled Reactor Experiment)	UO_2 or UO_2-BeO, Hastelloy-clad	Four 2-in.-diam x 32-in. Al cans of scrap pieces; two 1.5-in.-diam Al cans of plates; 66 pin-type assemblies		61.290	56.559			
HWCTR (Heavy-Water Components Test Reactor)	U and UO_2, Zr-clad	Intact assemblies 3 in. diam x 132 in. Pieces stored in 3.5-in.-diam x 12-in. cans	6,200	863.958	8.294		0.007	
	U-Zr, Zr-clad			37.165	31.590			
HTRE (High-Temperature Reactor Experiment)	UO_2-BeO, Nichrome-clad	Segments and pieces of fuel assemblies and test pieces in thirteen 4-in.-diam x 36-in. Al cans		3.698	3.423			
ML-1 (Mobile Low Power Plant No. 1)	UO_2 and PuO_2-BeO, SS-clad	Sixty-eight 19-pin assemblies		58.575	54.478			
ORNL (Oak Ridge National Laboratory)	U, Zr-clad	Stored in three 4.5-in.-diam x 9.25-in. Al cans		0.184	0.171			
ORNL mixed oxide	UO_2-PuO_2, Zr- or SS-clad	Stored in one 3.5-in.-diam x 15.12-in. can	Unknown but low	0.376	0.030		0.094	
Shippingport	UO_2, Zr-clad	Stored in a 10.5-in.-diam x 15-in. container	18,000	16.429	0.023		0.108	
SPERT-3 (Special Power Excursion Reactor Test)	UO_2, Zr-clad	Stored in three 4.0-in.-diam x 12-ft cans	Unknown	12.64	0.603			
SRE (Sodium Reactor Experiment)	U, Th rods, SS-clad	Stored in 3.5-in.-diam x 110.25-in. cans	10,000	155.24	143.410	1.045		1,972. 4
	UC, SS-clad			47.42	4.344		0.016	

Table 4.5.8 (continued)

Source of material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
DOE/Defense plus other government agencies material stored at SRS (continued)								
SRS (Savannah River Site)	UO_2-PuO_2, Zr-clad	Stored in a 12.0-in.-diam x 14-ft container	Unknown	69.00	0.304		0.161	
ORR-LEU (Oak Ridge Reactor Low Enriched Uranium)	U_3Si_2, Al-clad	Stored in fourteen 3.5-in.-diam x 168-in. Al cans	15,600	95.006	14.960	-	0.537	-
Subtotal				8,997.606	519.875	1.045	24.831	1,972.4
Total				10,419.517	761.036	31.158	42.666	8,648.2

[a]Ref. Brock 1990. The spent fuels listed in this table are not reprocessible in existing facilities.
[b]Zr-clad = Zircaloy-clad.

Table 4.5.10. Characteristics of the Elk River Reactor[a]

General characteristics	
Location	Elk River, Minnesota
Reactor-system designer	Allis-Chalmers Manufacturing Company
Architect-engineer	Sargent and Lundy, Engineers
Owner	USAEC
Operator-lessee	United Power Assoc., Elk River, Minnesota
Reactor thermal output	58.2 MW
Superheater thermal output	14.8 MW
Total thermal output	73.0 MW
Gross electric output	23.8 MW
Net electric output	22.5 MW
Net efficiency	30.8%
Operating pressure	922 psig
Operating temperature	536°F
Reactor vessel	
Inside height	25 ft
Inside diameter	7 ft
Wall thickness	3 in.
Base material	Carbon steel, Type A 302B
Cladding material	Stainless Steel, Type 304
Minimum cladding thickness	0.109 in.
Design pressure	1250 psig
Test pressure	1875 psig
Core	
Configuration	Right cylinder
Height	60 in.
Diameter	60 in.
Volume fractions of core materials	
Steel	4.25%
Zirconium	8.25%
Water	68.25%
Fuel	19.25%
Number of fuel assembly positions available	164
Number of fuel assemblies in a complete loading	148

[a]Source: Final Elk River Reactor Program Report, COO-651-93, November 1974.

Table 4.5.11. Shippingport LWBR design characteristics[a]

Rod center - center spacing, in.	0.3686	0.6304	0.6304	0.9005
Rod outer diameter, in.	0.3063	0.5717	0.5274	0.8323
Rod surface-surface spacing, in.	0.0623	0.0587	0.1030	0.0682
Clad thickness, in.	0.02217	0.02808	0.02642	0.0419
Clad thickness/diameter ratio	0.072	0.049	0.050	0.050
Number of grid levels	9	8	8	6
Number of grids in fuel height	7.5	6.5	6.5	6
Grid fraction/level, in fuel lattice	0.846	0.79	0.79	0.80
Grid volume/fuel rod,[b] $in.^3$	0.130	0.211	0.211	0.422
Metal/water volume ratio[c]	1.740	2.981	1.764	3.486
Total number of fuel rods	7428	3434	3581	3047
Number of flux-well rods	None	3	4	1
Total fissile loading, kg	198.6	116.3	186.1	None
Total Th-232 loading, kg	5206.5	9487.1	8788.3	18574.2

[a]Source: H. C. Hecker, *Summary of the Nuclear Design and Performance of the Light Water Breeder Reactor*, WAPD-TM-1326, June 1979 (Hecker 1979).

[b]Volume in fuel rod lattice based on number of grids in fuel height and the grid fraction per level in the fuel lattice.

[c]Under nominal hot conditions and with grid volume per fuel rod homogenized throughout the fuel regions.

5. MISCELLANEOUS WASTES

5.1 INTRODUCTION

Chapters 2, 3, and 4 have discussed the principal sources of repository wastes: light-water-reactor spent fuels, high-level wastes, and non-LWR spent fuels. These sources account for over 99% of all the wastes (on a curie basis) expected to go to the repository. Most of the wastes discussed in this chapter are in the greater-than-Class C low-level waste (GTCC LLW) category. GTCC LLW is defined as low-level waste generated by a licensee of the Nuclear Regulatory Commission (NRC) or Agreement States that exceeds the radionuclide concentration limits established by NRC for Class C LLW, and is therefore not acceptable for near-surface disposal. The NRC has recently ruled that GTCC LLW must be disposed of in a geologic repository unless disposal elsewhere has been approved by the NRC (NRC 1989). Typical examples of wastes that might require disposal as GTCC LLW are activated metal hardware (such as control rods), spent fuel disassembly hardware, ion exchange resins, filters, evaporator residues, sealed sources used in medical and industrial applications (such as oil-well logging), moisture and density gauges, scrap, and contaminated trash. Such wastes are generated by routine operations at nuclear power plants, by reactor fuel research facilities, and by manufacturers of radiopharmaceuticals and sealed sources, and will be generated in the future by the decommissioning of nuclear reactors, and possibly by the decommissioning of other facilities.

The wastes discussed in this chapter as possible GTCC LLW are divided into the following general categories:

1. wastes from routine LWR operations,
2. sealed radioisotope sources,
3. wastes from decommissioning LWRs,
4. other sources of waste, and
5. wastes from decommissioning other facilities.

In addition to those listed above, a few other miscellaneous wastes are also discussed. These include wastes generated or owned by the U.S. Department of Energy (DOE) that are not suitable for near-surface disposal, and other wastes that could potentially require repository disposal but whose classification or ownership is uncertain at present. This includes wastes held by DOE under agreements with non-DOE generators. Some of these wastes could eventually be classified as GTCC LLW.

5.1.1 Definitions Used in Waste Classification

The NRC, in 10 CFR Part 61, established radionuclide limits that determine whether or not a LLW is acceptable for near-surface disposal. Near-surface disposal is defined as disposal within the upper 30 m of the earth's surface. Low-level wastes that are acceptable for near-surface disposal are divided into classes A, B, and C. The limits for each class are given in Table 5.1.1. The term "greater-than-Class C low-level waste" (GTCC LLW) is applied to any LLW generated by a licensee of NRC or an Agreement State that exceeds the radionuclide limits established by the NRC for Class C.

The classification system established by NRC does not apply to wastes generated by the U.S. Department of Energy (DOE), which has a "TRU waste" category. The DOE has defined transuranic (TRU) waste as waste that is contaminated with alpha-emitting radionuclides having atomic numbers greater than 92 and half-lives greater than 20 years, in concentrations greater than 100 nCi/g of waste at the time of assay. The term "TRU waste" applies only to DOE-generated wastes and should not be applied to wastes generated by licensees of NRC or Agreement States. Likewise, the term GTCC LLW should not be applied to wastes generated by DOE; that term is limited to wastes generated by licensees of NRC or Agreement States. TRU waste is not required to be disposed of in an NRC-licensed facility, but is not acceptable for shallow-land disposal. The Waste Isolation Pilot Plant (WIPP) facility is designated for TRU waste.

The Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA), which became Public Law 99-240 on January 15, 1986, defines LLW as (A) radioactive material that is not high-level radioactive waste, spent nuclear fuel, or by-product material [as defined in section 11e(2) of the Atomic Energy Act of 1954], and (B) which the Nuclear Regulatory Commission (NRC), consistent with existing law and paragraph (A), classifies as low-level radioactive waste. The LLRWPAA states that the federal government (specifically, the U.S. Department of Energy) is responsible for the disposal of GTCC LLW, and that such disposal must be in a facility licensed by the NRC. The LLRWPAA also states that DOE is responsible for the disposal of (a) low-level radioactive waste (LLW) owned or generated by DOE, (b) LLW owned or generated by the U.S. Navy as a result of the decommissioning of vessels of the U.S. Navy, and (c) LLW owned or generated by the federal government as a result of any research, development, testing, or production of any atomic weapon. However, the disposal of these types of waste is not required to be in a facility licensed by NRC. Finally, Section 3(b)(3) of the LLRWPAA required the U.S. Department of Energy (DOE) to submit a report to the Congress setting forth DOE's recommendations for the management of GTCC LLW.

5.1.2 The National LLW Management Program

Pursuant to the requirements of the LLRWPAA, DOE issued a Report to Congress in 1987 containing estimates of the types and quantities of GTCC LLW, along with a discussion of technical, managerial, legislative, and institutional issues related to DOE's acceptance of responsibility for the disposal of GTCC LLW (DOE 1987). It was evident that estimates of future volumes of this waste varied significantly. Projections of the volume that will be on hand by the year 2020 ranged from 2,000 m^3 in DOE 1987 to 17,000 m^3 in the update of the Part 61 Impacts Analysis Methodology (Oztunali 1986).

Because of the wide differences in these estimates and in other data, the DOE National LLW Management Program (NLLWMP) initiated activities to develop best estimates of volumes and radioactivities of GTCC LLW for use in planning for the disposal of this waste. The NLLWMP issued a report on this work in August 1991 (NLLWMP 1991). Volume projections from NLLWMP 1991 and other sources are included in this chapter.

5.1.3 Waste Volume Projections

Prior to the publication of NLLWMP 1991, a 1990 paper by the NLLWMP (Hutchison 1990) described the sources of, and uncertainties in, the data used for earlier GTCC LLW volume projections. These earlier projections were principally based on a survey conducted by DOE in July and August of 1986 through the Energy Information Administration, with the cooperation of the NRC. In this survey, the EIA surveyed 1,275 possible generators of GTCC LLW in an effort to obtain information on current and future waste generation. Response to the survey was good; 1,085 survey forms were returned. The results identified 115 current or potential generators of GTCC LLW. However, analysis of the results showed that the information obtained was incomplete and contained a number of uncertainties. For example, additional experience in decommissioning processes was needed before the volumes and activities of GTCC LLW from these processes could be predicted. Also, some wastes were held on hand because they might be GTCC LLW but were not yet definitely known to be GTCC LLW.

After supplementing the results of the EIA survey with additional data obtained from the literature, eight specific areas were identified by the NLLWMP in which significant uncertainties required resolution. The following description of these areas of uncertainty is highly abridged, and the reader is referred to Hutchison's paper for a complete discussion:

1. Concentration averaging. For example, if GTCC LLW and Class C wastes are combined in a single package, the resulting package may meet Class C standards. Regulating agencies in different states may have differing criteria as to how concentration averaging may or may not be used.
2. Sealed sources. The initial survey was found to be incomplete. The NRC has now performed a more detailed sealed source survey, the results of which are used in the NLLWMP report. Preliminary indications are that the number of sealed sources identified in the NRC survey will be several orders of magnitude larger than the number identified in the EIA survey.
3. Disposition of commercial GTCC LLW currently held by DOE. The disposal of such waste raises legal issues that require resolution.
4. Projections of operations waste. The survey revealed that a number of operators generated GTCC LLW in past operations but did not project generation of GTCC LLW in the future. Clarification of the reasons for this is needed.
5. Timing of decommissioning wastes. Stretchouts of the timing of reactor decommissioning could have a significant effect on the schedule for the receipt of GTCC LLW.
6. Non-fuel-bearing components of reactors. The question of whether some of these components may be disposed of with spent fuel requires resolution. This could have an appreciable effect on the volume of activated metal hardware disposed of as GTCC LLW.
7. Ion-exchange resins. A draft report evaluating the volume of GTCC LLW from spent ion-exchange resins has been prepared (Vance 1989).
8. Waste packaging assumptions. In some cases, packaging assumptions may have been based on near-surface burial rather than on GTCC LLW disposal. In general, the assumptions underlying the packaging factors used in the current volume projections require validation.

Many of the technical problems and issues described above could not be resolved with the available information, and some involved difficult legal questions. The DOE NLLWMP carried out a multi-phased technical review process to develop the assumptions needed to resolve these issues. The assumptions from the technical review process permitted resolution of a number of these issues and were used to finalize the August 1991 report on projected volumes and radioactivities. In some areas, however, studies are continuing.

Table 5.1.2 summarizes the waste volume projections discussed in this chapter. Most of these are based on data from NLLWMP 1991.

5.1.4 References for Section 5.1

DOE 1987. *Recommendations for Management of Greater-than-Class C Low-Level Radioactive Waste*, DOE/NE-0077, February 1987.

Hutchison 1990. David Hutchison and Mary Magleby, "Greater-than-Class-C Low-Level Waste Characterization Technical Review Process," presented at INMM Conference, Los Angeles, July 15–18, 1990.

NLLWMP 1991. R. A. Hulse, *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics*, DOE/LLW-114, August 1991.

NRC 1989. U.S. Nuclear Regulatory Commission, amendments to 10 CFR Part 61, "Disposal of Radioactive Wastes," final rule, *Fed. Regist.* 54(100), 22578-83 (May 25, 1989).

Oztunali 1986. O. I. Oztunali, W. D. Pon, R. Eng, G. W. Roles, *Update of Part 61 Impacts Analysis Methodology*, NUREG/CR-4370, Vol. 2, January 1986.

Vance 1989. J. N. Vance and D. W. James, *Estimated GTCC Waste Volumes of Ion-Exchange Resins and Filters Generated by the Commercial Nuclear Power Industry*, Vance and Associates (draft), Ruidoso, New Mexico, October 1989.

Table 5.1.1. Rules for classifying LLW[a,b]

Long half-life radionuclides	$t_{1/2}$, years	Maximum concentration Column (L) (Ci/m^3)	Maximum concentration Column (L) (nCi/g)
C-14	5,730	8	
C-14 in activated metal	5,730	80	
Ni-59 in activated metal	76,000	220	
Nb-94 in activated metal	20,300	0.2	
Tc-99	213,000	3	
I-129	1.57E07	0.08	
Alpha emitters with $t_{1/2}$ > 5 years			100
Pu-241	14.4		3,500
Cm-242	0.45		20,000

Short half-life radionuclides	$t_{1/2}$, years	Maximum concentration, Ci/m^3 Column (A)	Column (B)	Column (C)
All radionuclides with $t_{1/2}$ < 5 years		700	NL	NL
H-3	12.3	40	NL	NL
Co-60	5.27	700	NL	NL
Ni-63	100	3.5	70	700
Ni-63 in activated metal	100	35	700	7,000
Sr-90	28.5	.04	150	7,000
Cs-137	30.0	1	44	4,600

[a]Apply these tests in the following order, choosing the lowest class that meets the test:

1. The waste is Class A if concentration exceeds neither Column (A) nor 10% of Column (L);
2. The waste is Class B if concentration exceeds neither Column (B) nor 10% of Column (L);
3. The waste is Class C if concentration exceeds neither Column (C) nor Column ((L);
4. The waste is greater than Class C if concentration exceeds either Column (C) or Column (L).

Note: For mixtures of radionuclides, limits are obtained by a sum-of-fractions rule. The entry NL means there is no limit for this nuclide in this class.

[b]Source: NRC regulation 10 CFR 61.55, Federal Register 27(248) pp. 57473-4, Dec. 27, 1982. The procedure shown above is the same as that described in 10 CFR 61.55 but is more condensed. For a more complete discussion, the reader is referred to 10 CFR 61.55.

Table 5.1.2. Summary of waste volume projections discussed in this chapter[a]

Type	Source	Base 2035 (m^3)	High 2035 (m^3)
GTCC LLW	LWR operations[b]	1,325	2,580
GTCC LLW	Sealed radioisotope sources[c]	6	18
GTCC LLW	Decommissioning LWRs[d]	523	1,794
GTCC LLW	Other sources of waste[e]	269	320
Potential GTCC LLW	Decommissioning of non-LWRs and fuel cycle facilities[f]	1,507	1,507
Total		3,630	6,219

[a]These are packaged waste volumes. Most of the projections are based on data in NLLWMP 1991. The base case and high case for year 2035 are as defined in NLLWMP 1991 (see Sect. 5.2.3 for definitions).

[b]See Table 5.2.1.

[c]See Sect. 5.3.

[d]See Table 5.4.3.

[e]See Table 5.5.1.

[f]See Table 5.6.3. Includes DOE-held wastes. It has not been completely determined how much of this waste is GTCC LLW.

5.2 GTCC LLW WASTES FROM ROUTINE LWR OPERATIONS

5.2.1 Introduction

Low-level wastes in classes A, B, and C, which meet the requirements for shallow-land burial, are routinely shipped from LWRs to commercial burial grounds. The NRC requires semiannual reporting of the volume and radioactivity of all nuclear power plant wastes shipped for commercial disposal. The categories defined by NRC are wet, dry compactible, irradiated components, and other. Reactor wastes account for more than half of the Class A, B, and C radioactive wastes shipped to commercial LLW burial grounds (A. Kibbey, in Forsberg 1985).

Wastes classified as GTCC LLW cannot be disposed of in commercial burial grounds and are kept in storage at the reactor sites; or, in some cases, are shipped to DOE sites for temporary storage by agreement with DOE. However, some wastes are not definitely known to exceed Class C limits and are kept in storage at reactor sites only because of the possibility that an assay at a future time may show that the waste is GTCC LLW. Also, packaging regulations may have an effect on waste classification. For these reasons, surveys of the quantities of GTCC LLW on hand at reactor sites may not give an accurate measure of the quantities of GTCC LLW that will be generated by routine LWR operations in the future.

The August 1991 report by the DOE National LLW Management Program (NLLWMP 1991) is the most up-to-date source of information on GTCC LLW generation rates and projections of the volumes of these wastes expected to be generated in the future. This section summarizes the NLLWMP's estimates of GTCC LLW volumes expected from routine LWR operations. Estimates from earlier studies are also briefly discussed.

In the NLLWMP report, routinely-discharged activated metals are included in operations waste, whereas this was not true in some of the earlier studies. In this report, the principal discussion of activated metal hardware is given in Chapter 2.

5.2.2 Earlier Estimates of Waste Volumes

Data were obtained by Cline et al. in 1985 on nearly 900 samples representing routine wastes from more than 50 BWRs and PWRs. Analyses showed that about 1.3% of the samples exceeded Class C limits. The samples in the Cline study consisted of typical radwaste streams (e.g., spent resins, sludges, evaporator bottoms, filter sludge, filter cartridges) and did not include non-fuel assembly hardware, i.e., activated metal components removed from inside the reactor pressure vessel (Cline 1985). The only BWR waste that exceeded Class C limits was filter sludge; 4.8% of the filter sludge samples exceeded Class C limits. For PWR waste, the categories filter sludge, filter cartridges, and evaporator bottoms all contained samples that exceeded Class C limits. The percentages of samples in these categories that were GTCC LLW were 2.2%, 9.4%, and 2.0%, respectively.

A breakdown of BWR and PWR operations wastes into categories, including those cited above from the Cline study, was done by Forsberg, Carter, and Kibbey (Forsberg 1985). It was reported that the waste from BWRs is 27% (by volume) filter sludge, and the waste from PWRs is 0.8% (by volume) filter cartridges, 0.1% filter sludge, and 42% evaporator bottoms. Combining the Cline 1985 and the Forsberg 1985 projections leads to a total volume of GTCC LLW from reactor operations of about 12,800 m^3 by the year 2020. Previously published estimates had been as high as 19,000 m^3. Both the Cline 1985 and the Forsberg 1985 studies excluded activated metal hardware from their GTCC LLW volume estimates.

5.2.3 Estimates of GTCC LLW Volumes by the NLLWMP

The NLLWMP report (NLLWMP 1991) shows estimated GTCC LLW volumes for four cases, referred to as low, base, and high for year 2035, and high for year 2055. All of these cases showed volumes considerably lower than previous estimates. For LWR operations waste, the packaged volumes shown for the year 2035 base and high cases were 1,325 m^3 and 2,580 m^3, respectively.

The cases presented in the NLLWMP report were defined as follows:

> Three cases were developed to give a range in the projections of GTCC LLW. The base case is intended to represent the most probable waste generation projections. That case is based on current generation rates and disposal practices using packaging and concentration averaging which are in use at the present time. The low case is the lower limits of the base case, assuming different practices in packaging and concentration averaging techniques. The high case is presented in two parts:
>
> 1. High Case (2035)
>
> The base case unpackaged data (projected to 2035) are modified for use in the high case by including more material (core barrel) from decommissioning of nuclear utilities and using higher packaging factors and more stringent concentration averaging practices. These more stringent concentration averaging practices assume no averaging between components. However, it is assumed that the radionuclide activity is homogeneously distributed throughout that component.

2. High Case (2055)

The second high case is similar to the high case above, except that life extension is assumed for 70% of all operating or licensed reactors. It is assumed that all life extension will be for 20 years. Only operational waste volumes will increase, and decommissioning waste volumes will not be affected.

A number of factors were discussed in the NLLWMP report that are expected to contribute to the reduction in estimated GTCC LLW volumes. Among these are (1) fewer components from reactor internals will be included as GTCC LLW, (2) efficient and reasonable packaging factors will be used, and (3) concentration averaging will be applied to reduce the volume of GTCC LLW in some cases.

The NLLWMP report used actual current production data on GTCC LLW quantities from operating LWR facilities. These data were provided to the NLLWMP project by a company that measures and calculates LLW activities, performs concentration averaging, and packages and ships wastes from nuclear utilities to commercial disposal sites (see Appendices F and I of NLLWMP 1991). One of the points noted was that most utilities are phasing out their evaporators because the evaporator bottoms are too variable and difficult to characterize. Another company provided an evaluation of the potential for classification of activated metal hardware as GTCC LLW (see Appendix G of NLLWMP 1991). A third study for the NLLWMP (Vance 1989) showed that the volume of GTCC LLW from filter cartridges and filter sludge is less than that indicated by the earlier estimates.

Table 5.2.1 summarizes estimates of GTCC LLW packaged volume projections from routine LWR operations given in the NLLWMP report. The NLLWMP report also shows possible reductions of 380 and 850 m^3 in operations waste due to concentration averaging in the low and base cases, respectively. One of the points mentioned by the NLLWMP was that the use of concentration averaging may cease, or be significantly reduced, when new LLW disposal facilities open for operation.

5.2.4 References for Section 5.2

Cline 1985. J. E. Cline, J. R. Noyce, L. J. Coe, and K. W. Wright, *Assay of Long-Lived Radionuclides in Low-Level Wastes from Power Reactors*, NUREG/CR-4101, 1985.

Forsberg 1985. C. W. Forsberg, W. L. Carter, and A. H. Kibbey, *Flowsheets and Source Terms for Radioactive Waste Projections*, ORNL/TM-8452, March 1985.

IDB 1990. *Integrated Data Base for 1990: Spent Fuel and Radioactive Waste Inventories, Projections and Characteristics*, DOE/RW-0006, Rev. 6, October 1990.

Kibbey 1990. A. H. Kibbey, H. W. Godbee, and S. M. DePaoli, *An Update of the Source Terms and Rationale Used for Low-Level Radioactive Waste projections in the 1988 Department of Energy Integrated Data Base*, ORNL/TM-11710, Oak Ridge National Laboratory, Oak Ridge, Tennessee (in preparation).

NLLWMP 1991. R. A. Hulse, *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics*, DOE/LLW-114, August 1991.

Vance 1989. J. N. Vance and D. W. James, *Estimated GTCC Waste Volumes of Ion-Exchange Resins and Filters Generated by the Commercial Nuclear Power Industry*, Vance and Associates (draft), Ruidoso, New Mexico, October 1989.

Table 5.2.1. NLLWMP estimates of GTCC LLW volumes from LWR operations[a]

Waste type	Estimated packaged volume, m^3			
	Low 2035	Base 2035	High 2035	High 2055
Control rod blades	353	441	883	1,200
Local power range monitors	58	97	193	263
Dry tubes	13	21	44	58
Decontamination resins	58	274	545	545
Pool filters	2	34	67	91
Control rod drive strainers	1	23	47	63
Cartridge filters	12	244	481	655
Crud tank filters	0	5	9	13
In-core detectors	25	43	84	112
Source rods	1	1	1	1
Instrument strings	28	46	92	122
Thimble plug assemblies	39	79	117	133
Control rod drives	17	17	17	17
Totals	607	1,325	2,580	3,273

[a]Source: NLLWMP 1991, Figs. 11-1 through 11-38.

5.3 SEALED RADIOISOTOPE SOURCES

5.3.1 Introduction

Sealed radioisotope sources, also known as radioactive sealed sources or simply sealed sources, are manufactured for use in a wide variety of industrial, medical, and other applications. The activities of sealed sources can range from a few curies to several thousand curies. The radioisotopes used in the manufacture of sealed sources in the United States can be purchased from a number of suppliers, the largest of which is DOE. Some of the major radioisotopes, particularly the transuranics, can be obtained for United States use only from DOE. The major radioisotopes that have been distributed thus far in industrial quantities are Co-60, Sr-90, Cs-137, Pu-238, Pu-239, Am-241, Cm-244, and Cf-252. Other radioisotopes that may be significant from the standpoint of quantities produced or potential disposal needs are H-3 (tritium), C-14, Kr-85, Tc-99, Eu-151, Eu-152, and Gd-153. Some of these sealed radioisotope sources will require future disposal as GTCC LLW; however, others may not. Some of the radioisotopes discussed here do not have a Class C limit. Thus, discussion of particular radioisotopes in this section is not meant to imply that they will necessarily require repository disposal; those that do not qualify as GTCC LLW are so identified.

Table 5.3.1 gives the half-lives and other physical data on the radionuclides discussed in this section. These were obtained from the ORIGEN2 radionuclide data base. Table 5.3.1 also shows whether there is a Class C limit on the radionuclide; this information was obtained by referring to Table 5.1.1.

Estimates of quantities of sealed sources that may require disposal as GTCC LLW are given in Sects. 5.3.3 and 5.3.4. These are based on the August 1991 report of the NLLWMP (NLLWMP 1991).

5.3.2 Types and Uses of Sealed Sources

5.3.2.1 Tritium (H-3)

Tritium (half-life 12.35 years) has been widely distributed for commercial uses such as leak testing and manufacture of luminous dials. This radionuclide is not normally used as a sealed source and does not have a Class C limit.

5.3.2.2 Carbon-14

Carbon-14 (half-life 5,730 years) has a Class C limit of 8 Ci/m^3 unless it is a component of activated metal, in which case the limit is raised to 80 Ci/m^3. Carbon-14 is used in research for "tagging" organic compounds as an aid in following their reactions. It is not normally used as a sealed source.

5.3.2.3 Cobalt-60

Cobalt-60 (half-life 5.27 years; high-energy gamma) has a variety of commercial and medical uses, the largest of which is the sterilization of medical supplies and instruments. The normal form of use is cobalt metal. The major supplier is Atomic Energy of Canada (about 50 million curies/year); DOE supplies about 2 million curies/year. Approximately 100 million curies were in use in 1987, and the quantity was increasing. At that time, a supply of about 10 to 12 million curies/year was required to replace the amount decaying each year. As Table 5.3.1 indicates, there is no Class C limit on Co-60, so this radionuclide does not qualify as GTCC LLW; however, it is discussed in this section for completeness.

5.3.2.4 Krypton-85

Krypton-85 (half-life 10.72 years) has been widely distributed for commercial uses such as leak testing and manufacture of luminous dials. In 1986 ORNL distributed about 5,000–6,000 Ci/year of Kr-85. This quantity, although appreciable, is small compared to the amount produced annually by commercial power reactor operation. Krypton-85 is also produced during the reprocessing of reactor fuels at defense sites. This radionuclide is not normally used as a sealed source and does not have a Class C limit.

5.3.2.5 Strontium-90

Strontium-90 (half-life 28.5 years) is not widely used as a sealed source. Its major use is as a heat source. Its short-lived daughter, Y-90, emits a hard gamma that requires considerable shielding. As discussed in Chapter 3 and Appendix 3B of this report, until 1985 Sr-90 and Cs-137 were separated from the HLW produced at Hanford, converted to SrF_2 and CsCl, and stored in capsules. Some of the capsules were distributed for commercial and medical uses; however, those that were unopened have been returned to Hanford for disposal as HLW and are therefore covered in Chapter 3 and Appendix 3B. Strontium-90 has a Class C limit of 7,000 Ci/m^3.

5.3.2.6 Technicium-99

Technicium-99 (half-life 213,000 years) is mainly used for research purposes and in 1984 was being sold at the rate of about 130 g/year. Several kg of this isotope have been sold over the years, and an inventory of about 1 kg was on hand at ORNL in 1986. Its Class C limit is 3 Ci/m^3.

5.3.2.7 Cesium-137

Cesium-137 (half-life 30.0 years) is most commonly used in applications that require a strong gamma source.

The hard gamma radiation for such applications comes from the short-lived daughter radionuclide Ba-137m. Gamma density logging is used in two important applications in the oil industry; these are measuring the density of rock and soil formations and measuring the density of fluids at specific locations within the borehole. Cesium-137 is the most commonly used radioisotope for these applications. The sealed sources used in density logging devices may contain from about 0.5 Ci to about 5.0 Ci of Cs-137 depending on the design. The sources are doubly encapsulated and are usually made of stainless steel, although other materials such as monel have been used. As mentioned in Sect. 5.3.2.5, a number of capsules of CsCl were distributed from Hanford for commercial and medical uses. Those that were unopened have been returned to Hanford for disposal as HLW, and are discussed in Chapter 3 and Appendix 3B.

Cesium-137 has a Class C limit of 4,600 Ci/m^3.

5.3.2.8 Gadolinium-153, Europium-151, and Europium-152

Gadolinium-153 (half-life 242 days) has recently become important as an isotope used in the diagnosis of osteoporosis, and its demand for this purpose has grown by a factor of almost 20 in the past four years. In 1986 ORNL was planning to expand its production of Gd-153 to a rate of several thousand curies per year, but this plan was interrupted by the shutdown of the High Flux Isotope Reactor. The isotope is produced by the irradiation of natural europium, followed by separation and die-pressing into 1-Ci pellets. The by-products Eu-151 and Eu-152, which are produced as waste, are chemically removed from the Gd-153 and stored together in double-containment stainless steel capsules. Europium-151 is stable, but Eu-152 (half-life about 13.6 years) emits a hard gamma. Neither Gd-153 nor Eu-152 has a Class C limit.

5.3.2.9 Plutonium-238

The alpha-emitting radioisotope Pu-238 (half-life 87.7 years) is used with beryllium to provide a source of neutrons by the (α,n) reaction. Such neutron sources are used in oil-well logging devices and as start-up sources for nuclear reactors. Plutonium-238 is also used as a heat source in special applications where more conventional sources of heat are not usable. An example of this is space power applications for satellites; here it is encased in iridium to avoid burnout during reentry. In this application, Pu-238 is by far the most widely used radioisotope. Because Pu-238 is classed as a special nuclear material, its use is subject to certain restrictions.

5.3.2.10 Plutonium-239

Plutonium-239 (half-life 24,100 years) was used in the past in certain special applications but is no longer used commercially. The principal reason for including it here is that the NRC survey identified a number of sealed sources containing Pu-239 that are on hand and will have to be disposed of. Plutonium-239 has a Class C limit of 100 nCi/g.

5.3.2.11 Americium-241

Americium-241 (half-life 432 years; neutron source when mixed with beryllium) is used commercially for oil-well logging and for the production of smoke alarms. The first of these uses account for most of the Am-241 produced, about 1–2 kg/year; the total annual use for smoke alarms is only about 10 g. The amount used in a smoke detector is so small that no administrative controls on the user are necessary. Because of this, smoke detectors are not included in the count of sealed sources in this report. Oil-well logging sources require larger amounts of Am-241, and these sources are licensed by NRC; some also may be licensed by Agreement States, although this has not been confirmed. Thus far, about 8–10 kg have been distributed.

A typical oil-well logging device consists of a high-energy neutron source and a capture gamma detector located at a fixed distance from the neutron source. Most sources used for neutron-capture gamma well logging make use of the (α,n) reaction with either Am-241 or Pu-238 as the alpha source and beryllium as the neutron-producing target material. Americium-241 has the advantage of being less difficult to license or export than Pu-238. A typical neutron source capsule has a diameter of about 1.0 in. and a length of about 5.4 in. The Am-241 is used in the oxide form and is mixed and pelletized with beryllium metal powder, which is then doubly encapsulated in stainless steel. The quantity of Am-241 varies with the application but typically is about 10–20 Ci.

Although Am-241 has a half-life of 432 years, the useful life of an oil-well logging device is typically only a few years because of mechanical or electronic obsolescence. No data were found as to what percentage of the Am-241 source capsules, if any, are returned to the manufacturer for reuse.

Americium-241 has a Class C limit of 100 nCi/g. An encapsulated sealed source containing 16 Ci of Am-241 and having a volume of 4.2 in^3 (69.5 cm^3) would weigh about 500 g. Its radioactivity per unit mass would be about 0.032 Ci/g. This exceeds the Class C limit by a factor of about 3×10^6.

5.3.2.12 Curium-244

Curium-244 (half-life 18.1 years) is used in sealed sources to a very minor extent. Its principal uses are in X-ray fluorescence analyzers and fixed gauges. It is also used as a heat source in some special applications, although the use of Pu-238 in this application is much more

common. The only other significant use of Cm-244 is as a target material for the production of higher radionuclides.

5.3.2.13 Californium-252

Californium-252 (half-life 2.64 years) is widely used as a neutron source, making use of the fact that about 3% of its decay is by spontaneous fission; the other 97% is by alpha decay. Californium-252 has been produced at a rate of ~500 mg/year at DOE's Transplutonium Element Production Program facilities at Oak Ridge National Laboratory, which consist of the High Flux Isotope Reactor and the Transuranium Processing Plant.

As a neutron source, Cf-252 is unique in providing a highly concentrated and reliable neutron spectrum from a very small assembly. Over the past 30 years, Cf-252 has been applied to cancer therapy, neutron radiography, startup sources for nuclear reactors, fission activation for QA of commercial nuclear fuel, and other uses. Recently, Cf-252 has been introduced as a neutron source in oil-well logging devices. It has the advantage of producing a physically smaller device than those that use Am-241, and thus can be used in more confined spaces. However, the short half-life of Cf-252 is a disadvantage. At present, only a small fraction of oil-well logging devices use Cf-252. Californium-252 decays to a long-lived daughter actinide, Cm-248, which is useful for research purposes.

Data on commercial sales of Cf-252 are tabulated in Table 5.3.2. A breakdown of Cf-252 sales by DOE in terms of final application is given in Table 5.3.3. About half the sources (and half of the contained Cf-252) are for reactor startup. Fuel rod scanners and activation analysis are the next biggest segments of the sales market.

As shown in Table 5.3.1, there is no Class C limit on Cf-252. However, its daughter radionuclide Cm-248 has a Class C limit of 100 nCi/g. This quantity of Cm-248 would be produced in 2.64 years by the decay of an initial quantity of 0.0257 Ci of Cf-252. This initial quantity corresponds to about 48 μg of Cf-252.

5.3.3 Quantities of Sealed Sources That Are GTCC LLW

Estimates of the number of sealed sources in existence have varied greatly. In the 1987 DOE Report to Congress, it was estimated that there were less than 2,000 sealed sources in storage that were GTCC LLW (DOE 1987). The 1988 OTA background paper for Congress reported that the NRC estimated that there may be 25,000–30,000 GTCC sealed sources in use in the United States (OTA 1988).

The number of sealed sources now in existence that may eventually be disposed of as GTCC LLW was recently evaluated in a 1989 survey by NRC. The NRC survey was limited to specific licensees. Specific licensees are those that hold broad Type A NRC or similar Agreement State licenses. The results of the NRC survey were used by the NLLWMP in its 1991 report on GTCC LLW (NLLWMP 1991). Based on an analysis of the NRC survey, the NLLWMP report estimated that there were about 27,000 sealed sources currently in the possession of specific licensees that would qualify as GTCC LLW. In analyzing its survey results, the NRC estimated that there are also about 18,500 general licensees possessing about 65,500 sealed sources that could qualify as GTCC LLW. These were not taken into account in the estimation of disposal volumes.

The NLLWMP has indicated that additional investigations into quantities and volumes of sealed sources are planned by both NRC and NLLWMP during FY 1992, so these estimates should be considered as subject to change.

5.3.4 Volumes of Sealed Sources Requiring Disposal as GTCC LLW

The August 1991 report of the NLLWMP estimated that the total base case and high case packaged volumes of sealed sources to be disposed of as GTCC LLW by year 2035 would be 6 and 18 m^3, respectively. The NLLWMP has stated that these estimates should be considered as subject to further development. They appear to be on the low side, based on 27,000 sealed sources and a packaging factor of 380, as used by the NLLWMP.

However, the estimation of repository disposal volumes for sealed sources is uncertain, because designs of disposal packages and decisions as to modes of disposal have not reached a point where accurate estimates of total disposal volumes can be made. It has not yet been determined whether shipping packages will be used as repository disposal packages, and if so, what the designs of the shipping packages will be, and how many sources each package will carry. In view of these uncertainties, and of the fact that further study is being carried out by the NLLWMP, no estimates of disposal volumes are made in this report beyond those given in the preceding paragraph.

5.3.5 GTCC LLW from the Manufacture of Sealed Sources

In DOE's 1987 Report to Congress, it was pointed out that GTCC LLW is generated during the manufacture of sealed sources. The waste generated consists of various materials contaminated with Sr-90, Cs-137, Pu-238, Am-241, and other radionuclides.

The NLLWMP report (NLLWMP 1991) discusses GTCC LLW from the manufacture of sealed sources under the category "Other Generators Waste." This category of waste is discussed in Sect. 5.5.

5.3.6 Chemical Forms of Radionuclides in Sealed Sources

Table 5.3.4 shows the chemical forms in which some of the most widely used radionuclides are ordinarily used in sealed sources (Monsanto 1981).

5.3.7 Reporting of Radioisotope Shipments

Until 1985, a report was prepared annually by PNL summarizing the radioisotope shipments for the year, giving the names of the customers and the amounts shipped. This document was prepared for the Office of Health and Environmental Research (ER-73), Office of Energy Research, DOE, and listed DOE's radioisotope production and distribution activities of its facilities at Argonne National Laboratory, Brookhaven National Laboratory, Hanford Engineering Development Laboratory, Pacific Northwest Laboratory, Idaho Operations Office, Los Alamos National Laboratory, Oak Ridge National Laboratory, Savannah River Plant, and UNC Nuclear Industries, Inc. The information included was generally as follows:

- A list of the suppliers of isotopes and the name of the contact person for each DOE facility;
- A list of customers and quantities of isotopes purchased, along with the identification of the DOE facility supplying each isotope; and
- A summary of radioisotope shipments for the fiscal year, with appropriate dollar value.

Both foreign and domestic customers are included. Shipments may be either purchased or leased. If leased, the title to the radioisotope remains with DOE. If purchased, it is transferred to the purchaser.

Table 5.3.5 shows typical information on radioisotope quantities excerpted from PNL's report for FY 1984 (Baker 1985). This table is not a complete list, but shows some of the major radioisotopes of commercial interest.

5.3.8 References for Section 5.3

Baker 1985. D. A. Baker, *List of DOE Radioisotope Customers with Summary of Radioisotope Shipments, FY 1984*, DOE Report PNL-5492, August 1985.

DOE 1987. *Recommendations for Management of Greater-than-Class C Low-Level Radioactive Waste*, DOE/NE-0077, February 1987.

Hutchison 1990. David Hutchison and Mary Magleby, "Greater-than-Class-C Low-Level Waste Characterization Technical Review Process," presented at INMM Conference, Los Angeles, July 15-18, 1990.

Monsanto 1981. *Radioactive Sources, Their Manufacture and Characteristics*, Monsanto Research Corporation, Dayton, Ohio, October 1981.

NLLWMP 1991. R. A. Hulse, *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities and Other Characteristics*, DOE/LLW-114, August 1991.

OTA 1988. *An Evaluation of Options for Managing Greater-than-Class-C Low-Level Radioactive Waste*, Office of Technology Assessment, October 1988.

Wolfe 1986. Sylvia Wolfe, telephone conversation with Royes Salmon, dated April 8, 1986.

Table 5.3.1. Reference data on radioisotopes discussed in Section 5.3[a]

Radioisotope	Half-life, years (or as noted)	Curies per gram	Thermal power (W/Ci)	Class C limit[b]
H-3	12.35	9.65E+03	3.36E-05	NL
C-14	5,730	4.46E+00	2.93E-04	c
Co-60	5.27	1.13E+03	1.54E-02	NL
Kr-85	10.72	3.92E+02	1.50E-03	NL
Sr-90	28.5	1.36E+02	1.16E-03	7,000 Ci/m^3
Y-90	2.67 days	5.44E+05	5.54E-03	d
Tc-99	213,000	1.70E-02	5.01E-04	3 Ci/m^3
Cs-137	30.0	8.70E+01	1.11E-03	4,600 Ci/m^3
Ba-137m	153 sec	5.38E+08	3.93E-03	e
Eu-152	13.6	1.73E+02	7.65E-03	NL
Gd-153	242 days	3.53E+03	9.03E-03	NL
Pu-238	87.74	1.71E+01	3.31E-02	100 nCi/g
Pu-239	24,064	6.22E-02	3.08E-02	100 nCi/g
Am-241	432	3.43E+00	3.32E-02	100 nCi/g
Cm-248	339,000	4.25E-03	1.24E-01	100 nCi/g
Cf-252	2.64	5.38E+02	7.13E-02	f

[a]Source: ORIGEN2 radionuclide library.

[b]NL = no Class C limit.

[c]Class C limit is 8 Ci/m^3 except when contained as a component in activated metal, in which case the limit is raised to 80 Ci/m^3.

[d]There is no Class C limit on Y-90, which is the gamma-active daughter of Sr-90.

[e]There is no Class C limit on Ba-137m, which is the gamma-active daughter of Cs-137.

[f]There is no Class C limit on Cf-252 itself, but there is a Class C limit on its daughter Cm-248.

Table 5.3.2. Commercial sales of ^{252}Cf by Savannah River Office[a]

Primary customer	1981	1982	1983	1984	1985
GE-Vallecitos	4.8	6.6	6.4	2.6	6.0
Monsanto	11.8	5.4	11.2	20.9	21.5
Amersham	12.0	-	8.1	16.4	10.0
Karlsruhe	1.0	2.1	14.9	-	3.0
CEN France	7.0	7.4	10.5	7.6	15.0
Total	36.6	21.5	51.1	47.5	55.5

[a]Estimated by DOE Savannah River Office, based on customer information. Quantities are in mg.

Table 5.3.3. Distribution of secondary sales of ^{252}Cf by market applications[a]

Market	Number of sources, %	mg of ^{252}Cf, %
Reactor startup	55.4	48.3
Fuel rod scanner	8.2	25.3
Activation analysis	18.9	19.4
Education/research	5.0	2.4
Medical research	3.1	0.7
Calibration/dosimetry	3.1	0.1
Gauging/miscellaneous	6.3	3.8

[a]Data based on about 260 sources sold by Monsanto during 1971–1982. Other suppliers reportedly have similar distributions.

Table 5.3.4. Chemical forms in which radioisotopes are commonly used[a]

Radionuclide	Chemical form
Cobalt-60	Co metal
Strontium-90	$SrTiO_3$
Cesium-137	CsCl
Promethium-147	Pm_2O_3
Polonium-210	Po metal
Plutonium-238	PuO_2
Americium-241	AmO_2[b]
Curium-242	Cm_2O_3
Curium-244	Cm_2O_3
Californium-252	Cf_2O_3

[a]Source: Monsanto 1981.

[b]Sometimes combined with powdered beryllium when used as a neutron source by the (α,n) reaction.

Table 5.3.5. Quantities of various radioisotopes shipped from DOE laboratories during FY 1984[a]

Radioisotope	Quantity shipped	
	g	Ci
Tritium-3	52.8	510,000
Cobalt-60	577	652,000
Strontium-90	0.368	50
Technetium-99	25	0.425
Cesium-137	993	86,350
Cerium-144	0.0011	3.56
Gadolinium-153	0.022	78.5
Iridium-192	637.6	5,860,000
Americium-241	453	1,554
Curium-244	0.012	0.97
Californium-252	0.0235	12.6

[a]Source: Baker 1985. Only shipments to non-DOE domestic customers are shown. Additional radioisotopes are listed in the reference.

5.4 POTENTIAL GTCC LLW FROM DECOMMISSIONING LWRs

5.4.1 Introduction

Commercial nuclear reactors and other nuclear facilities must be decommissioned at the end of their useful life. Decommissioning means the steps taken at the end of a facility's life to safely retire it from service. This may be accomplished in one of three ways, as defined by the U.S. Nuclear Regulatory Commission (NRC 1988):

> SAFSTOR. The nuclear facility is placed and maintained in such condition that it can be safely stored and subsequently decontaminated to levels that permit release of the property for unrestricted use. Also known as mothballing.
>
> ENTOMB. The radioactive contaminants are encased in a structurally long-lived material such as concrete. The entombment structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting unrestricted release of the property. Also known as in-place entombment.
>
> DECON. The equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations. Also known as prompt removal or removal and dismantling.

These three decommissioning methods are described by the International Atomic Energy Agency (IAEA) as Stage 1, Stage 2, and Stage 3, respectively. The IAEA term "stage" does not imply a step-by-step process, but is intended to indicate the nature of the decommissioning measures taken. Thus, a Stage 3, or DECON, decommissioning operation could be performed without ever performing operations in the other 2 stages.

The delayed dismantling alternative, SAFSTOR, allows significant radioactive decay (though little reduction in GTCC volumes) to occur and thus reduces the occupational exposure during future decommissioning operations. This benefit must be compared with the increased costs of storage during delayed decommissioning. Most studies suggest that immediate dismantlement (DECON) is more cost effective (Tsoulfanidis 1991, EPRI 1988).

NRC regulations do not allow GTCC LLW to be placed in LLW shallow-land burial facilities. This means that GTCC wastes from the decommissioning of LWRs must be placed in a federal geologic repository or other NRC-approved and licensed site (NRC 1989).

5.4.2 Activated Metals and Other Sources of GTCC LLW from Decommissioning

GTCC LLW can be classified as either remote-handled or contact-handled. The bulk of GTCC LLW from LWR decommissioning will consist of remote-handled, activated metal components removed from the reactor core region (Peters 1990). As Table 5.1.1 shows, the primary radionuclides (with half-lives) in these activated metals that have Class C limits are ^{59}Ni (76,000 years), ^{63}Ni (100 years), and ^{94}Nb (20,300 years). Short-term radioactivity is dominated by ^{60}Co but, as shown in Table 5.1.1, this radionuclide has no Class C limit. However, ^{60}Co could be present in a waste that is GTCC LLW because of other radionuclides and could contribute significantly to the radioactivity and shielding requirements of such a waste.

Decontamination of surfaces and treatment of liquid wastes during decommissioning operations will result in the production of some GTCC LLW, consisting principally of combustible trash, filter cartridges, and ion exchange resins contaminated primarily with transuranics (DOE 1987).

5.4.3 Historical LWR Decommissioning Data

Currently, 13 commercial LWR reactors have been shut down, as shown in Table 5.4.1. Two of these reactors, Elk River and Shippingport, have been completely dismantled (DECON). Since the wastes from Elk River and Shippingport have already been buried, no repository waste will be generated from these activities. Elk River, a 22-MW(e) plant, was the first government-owned power reactor producing commercial electric power to be dismantled. The plant, which operated from 1964 to 1968, was completely dismantled by 1974 after producing 510,620 MW(e)-h of electricity. The Elk River pressure vessel weighed 36 tons and had an estimated activity of 1,110 Ci. The core shroud and shields had a combined activity of 7,790 Ci. All of the Elk River wastes were disposed of at the Sheffield shallow-land burial site in Illinois. The Elk River reactor was not representative of later reactor designs.

Shippingport was the first domestic commercial power reactor. Two PWR cores and one light-water breeder fuel core were used over its 25-year operating history from 1957 to 1982. During its 80,324 hours of operation, the plant produced 7.4 million MW(e)-h of electricity. Decommissioning of Shippingport started in 1985 and continued through 1990. The activated metals from the DECON process were packaged in the reactor pressure vessel, which was then shipped intact to Hanford in 1989 for shallow-land burial at the DOE Hanford site. The reactor vessel, as packaged for shipment, had a volume of 283 m^3 and an estimated activity of 16,000 Ci (IDB 1991). According to DOE, decommissioning of the Shippingport reactor did not generate any GTCC LLW (GAO 1990); the

basis for this was not defined, although it was noted in GAO 1990 that Shippingport was not licensed by the NRC.

There are a number of significant differences between the Shippingport reactor and new commercial reactors. Shippingport was relatively small [60 MW(e)] compared to modern power reactors [typically 1,000 MW(e)]. The pressure vessel, 25 feet in height and weighing 153 tons, was disposed of in one piece. This would be very difficult to accomplish for today's power reactors, which can have vessels 70 feet in height and weighing as much as 1,000 tons. It is expected that the DECON of large power reactors will generate GTCC LLW, even after a SAFSTOR period of 50 years.

5.4.4 Projected LWR Decommissioning Data

As shown in Table 5.4.2, there were 111 operable, commercial LWR power reactors in the United States as of December 1991. Three additional reactors shown are expected to be operational by 1995. This table also shows the reactor thermal and electrical power and the year of initial reactor startup (year criticality was first achieved). The projected shutdown year is based on the assumption that the reactor will operate for 40 years after initial criticality.

Based on the 40-year life assumption, over one-half of the currently operating LWR reactors will be shut down by 2017, and all existing reactors will be ready for decommissioning by 2035. After a reactor license expires, the utility has up to 60 years under NRC rules to complete reactor decommissioning. Alternatively, the utility can request a NRC license extension. This is expected to extend many existing reactor licenses by about 20 years (GAO 1990). For these reasons, extensive decommissioning of U.S. power reactors is not expected for a number of years.

The waste quantities generated from the decommissioning of a particular reactor will depend upon a number of factors such as reactor size, rated power level, neutron flux level, reactor capacity factor, decommissioning techniques used, composition of metal components, and other factors.

In order to account for these and other differences, all commercial LWR power reactors were grouped into 6 generic reactor types (2 BWR and 4 PWR) by the National Low-Level Waste Management Program (NLLWMP 1991). These generic or reference reactors were characterized in detail to determine the quantity of activated metal hardware that would be generated during the 40-year plant life. The projected volumes and activities of GTCC LLW for all 6 reference reactor types are given in Table 5.4.3. These assumptions were used to estimate the GTCC LLW quantities in the two following summary tables. The second column in Table 5.4.2 identifies the type.

The National Low-Level Waste Management Program (NLLWMP) has projected volumes of GTCC LLW from all LWR decommissioning activities as shown in Table 5.4.4. These GTCC LLW projections assume that reactor decommissioning begins five years after shutdown following 40 years of operation. An average capacity factor of 65% for BWRs and 70% for PWRs was used to estimate the radionuclide activities of GTCC LLW at the time of decommissioning.

The NLLWMP report (NLLWMP 1991) considers three scenarios for data projection of GTCC LLW quantities: (1) unpackaged volumes which consider only the actual (or displacement) volume of activated metal, (2) packaged volumes based on the application of realistic packaging factors to the unpackaged volumes, and (3) concentration averaging applied to the packaged volumes. These scenarios were used to represent the effects of packaging and concentration averaging on each specific waste stream. For reactor decommissioning waste, concentration averaging had no effect.

These scenarios are each represented for four projection cases: low 2035, base 2035, high 2035, and high 2055. The base case, projected to the year 2035, is intended to represent the most probable waste generation rate and is based on current disposal practices for packaging and concentration averaging. The low and high cases consider a range of assumptions for the packaging factors. The alternate high case, projected to 2055, assumes that 70% of the commercial LWRs will be granted a 20-year life extension by the NRC. Plant life extension was shown to have no effect on the volumes of GTCC LLW generated from LWR decommissioning. Additional information on the NLLWMP assumptions can be found in Sect. 5.2.3.

In all of these cases, the NLLWMP study projects GTCC LLW volumes that are considerably lower than previous estimates (Forsberg 1985, Oztunali 1986). These previous studies were not as detailed as the NLLWMP study. Earlier studies usually considered four PWR core components (core shroud, core barrel, thermal shields, and lower grid plate) to be GTCC LLW. Newer information from in-core measurements of neutron fluences, updated material compositions, and more detailed calculations of activation levels have determined that only the PWR core shroud is GTCC LLW, with the core barrel being GTCC LLW in a few cases. In the NLLWMP study and previous studies, only the core shroud was calculated to be GTCC LLW for BWRs. Studies by the NLLWMP are continuing and should help to reduce uncertainties.

The projected radioactivity of the GTCC LLW in 2035 is shown in Table 5.4.5. For each radionuclide, the activity is given for the BWR core shroud, PWR core shroud, and the PWR core barrel. The activity projections, made using the same assumptions as for Table 5.4.4, show that the total radioactivity of the GTCC LLW generated by the decommissioning of LWRs will be about 4.17×10^7 Ci in 2035.

5.4.5 References for Section 5.4

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Table 5.4.1. Commercial LWR reactors shut down or dismantled as of December 31, 1992[a]

Reactor	Location	Power		Year of initial criticality	Year of permanent shutdown	Decom. period[b]	Decom. method
		MW(e)	MW(t)				
		Boiling-water reactors					
Boiling Nuclear Superheater Power Station (BONUS)	Punta Higuera, PR	17	50	1964	1968	1970	ENTOMB
Dresden 1	Morris, IL	200	700	1960	1978	1978	SAFSTOR
Elk River	Elk River, MN	22	58	1962	1968	1971–74	DECON
Humboldt Bay 3	Eureka, CA	63	242	1963	1976	1985	SAFSTOR
La Crosse	Genoa, WI	50	165	1969	1987	1988	SAFSTOR
Pathfinder	Sioux Falls, SD	59	190	1966	1967	1972	SAFSTOR
Shoreham	Brookhaven, NY	820	2436	1985	1989	1989	SAFSTOR
Vallecitos	Pleasanton, CA	5	33	1957	1963	1963	SAFSTOR
		Pressurized-water reactors					
Indian Point 1	Buchanan, NY	257	615	1963	1974	1974	SAFSTOR
Rancho Seco	Clay Station, CA	913	2772	1975	1989	1989	SAFSTOR
Saxton Nuclear Experimental Reactor Project	Saxton, PA	3	24	1962	1972	1986–present[c]	DECON
Shippingport	Shippingport, PA	60	236	1957	1982	1985–90	DECON
Three Mile Island 2	Middletown, PA	906	2772	1978	1979	1986–present	Partial DECON[d]

[a]ANS 1992, DOE 1991, IDB 1991.

[b]Decommissioning period is either the year SAFSTOR or ENTOMB was initiated or the time period over which DECON occurred.

[c]Saxton reactor began SAFSTOR in 1973. DECON was initiated in 1986.

[d]TMI-2 is currently undergoing a partial decontamination after the 1979 accident. The plant will then be placed in monitored storage indefinitely.

Table 5.4.2. Projected shutdown dates for commercial LWR power reactors[a]

Reactor name	Reactor class	Reactor type	Reactor power MW(e)	Reactor power MW(t)	Initial startup	Projected shutdown
Yankee-Rowe	WE-1	PWR	167	600	1960	2000
Big Rock Point	GE-4	BWR	67	240	1962	2002
Haddam Neck	WE-1	PWR	565	1825	1967	2007
San Onofre 1	WE-1	PWR	436	1347	1967	2007
Ginna	WE-1	PWR	470	1520	1969	2009
Nine Mile Point 1	GE-4	BWR	610	1850	1969	2009
Oyster Creek	GE-4	BWR	620	1930	1969	2009
Dresden 2	GE-4	BWR	772	2527	1970	2010
Millstone 1	GE-4	BWR	654	2011	1970	2010
Monticello	GE-4	BWR	536	1670	1970	2010
Point Beach 1	WE-1	PWR	485	1518	1970	2010
Robinson 2	WE-1	PWR	700	2300	1970	2010
Dresden 3	GE-4	BWR	773	2527	1971	2011
Palisades	CE	PWR	768	2530	1971	2011
Quad Cities 1	GE-4	BWR	769	2511	1971	2011
Maine Yankee	CE	PWR	840	2630	1972	2012
Pilgrim	GE-4	BWR	670	1998	1972	2012
Point Beach 2	WE-1	PWR	485	1518	1972	2012
Quad Cities 2	GE-4	BWR	769	2511	1972	2012
Surry 1	WE-1	PWR	781	2441	1972	2012
Turkey Point 3	WE-1	PWR	666	2200	1972	2012
Vermont Yankee	GE-4	BWR	504	1593	1972	2012
Browns Ferry 1	GE-4	BWR	1065	3293	1973	2013
Fort Calhoun	CE	PWR	478	1500	1973	2013
Indian Point 2	WE-1	PWR	970	2758	1973	2013
Oconee 1	B&W	PWR	846	2568	1973	2013
Oconee 2	B&W	PWR	846	2568	1973	2013
Peach Bottom 2	GE-4	BWR	1100	3293	1973	2013
Prairie Island 1	WE-1	PWR	503	1650	1973	2013
Surry 2	WE-1	PWR	781	2441	1973	2013
Turkey Point 4	WE-1	PWR	666	2200	1973	2013
Zion 1	WE-1	PWR	1040	3250	1973	2013
Zion 2	WE-1	PWR	1040	3250	1973	2013
Arkansas 1	B&W	PWR	836	2568	1974	2014
Browns Ferry 2	GE-4	BWR	1065	3293	1974	2014
Calvert Cliffs 1	CE	PWR	825	2700	1974	2014
Cooper Station	GE-4	BWR	764	2381	1974	2014
Duane Arnold	GE-4	BWR	538	1658	1974	2014
Fitzpatrick	GE-4	BWR	757	2436	1974	2014
Hatch 1	GE-4	BWR	741	2436	1974	2014
Kewaunee	WE-1	PWR	503	1650	1974	2014

Table 5.4.2 (continued)

Reactor name	Reactor class	Reactor type	Reactor power MW(e)	Reactor power MW(t)	Initial startup	Projected shutdown
Oconee 3	B&W	PWR	846	2568	1974	2014
Peach Bottom 3	GE-4	BWR	1100	3293	1974	2014
Prairie Island 2	WE-1	PWR	500	1650	1974	2014
Three Mile Island 1	B&W	PWR	808	2568	1974	2014
Brunswick 2	GE-4	BWR	790	2436	1975	2015
Cook 1	WE-1	PWR	1020	3250	1975	2015
Millstone 2	CE	PWR	863	2700	1975	2015
Trojan	WE-2	PWR	1095	3411	1975	2015
Beaver Valley 1	WE-1	PWR	810	2660	1976	2016
Browns Ferry 3	GE-6	BWR	1065	3293	1976	2016
Brunswick 1	GE-4	BWR	790	2436	1976	2016
Calvert Cliffs 2	CE	PWR	825	2700	1976	2016
Indian Point 3	WE-1	PWR	965	3025	1976	2016
Salem 1	WE-2	PWR	1106	3411	1976	2016
St. Lucie 1	CE	PWR	839	2700	1976	2016
Crystal River 3	B&W	PWR	821	2544	1977	2017
Davis-Besse	B&W	PWR	874	2772	1977	2017
Farley 1	WE-1	PWR	814	2652	1977	2017
Arkansas 2	CE	PWR	858	2815	1978	2018
Cook 2	WE-1	PWR	1060	3411	1978	2018
Hatch 2	GE-4	BWR	761	2436	1978	2018
North Anna 1	WE-1	PWR	911	2893	1978	2018
North Anna 2	WE-1	PWR	909	2893	1980	2020
Salem 2	WE-2	PWR	1106	3411	1980	2020
Sequoyah 1	WE-2	PWR	1148	3411	1980	2020
Farley 2	WE-1	PWR	824	2652	1981	2021
McGuire 1	WE-2	PWR	1129	3411	1981	2021
Sequoyah 2	WE-2	PWR	1148	3411	1981	2021
Grand Gulf 1	GE-6	BWR	1142	3833	1982	2022
LaSalle 1	GE-6	BWR	1036	3323	1982	2022
San Onofre 2	CE	PWR	1070	3390	1982	2022
Summer	WE-1	PWR	885	2775	1982	2022
Susquehanna 1	GE-6	BWR	1032	3293	1982	2022
McGuire 2	WE-2	PWR	1129	3411	1983	2023
San Onofre 3	CE	PWR	1080	3390	1983	2023
St. Lucie 2	CE	PWR	839	2700	1983	2023
Callaway	WE-2	PWR	1125	3565	1984	2024
Diablo Canyon 1	WE-2	PWR	1073	3338	1984	2024
LaSalle 2	GE-6	BWR	1036	3323	1984	2024
Limerick 1	GE-6	BWR	1055	3293	1984	2024
Susquehanna 2	GE-6	BWR	1038	3293	1984	2024

Table 5.4.2 (continued)

Reactor name	Reactor class	Reactor type	Reactor power MW(e)	Reactor power MW(t)	Initial startup	Projected shutdown
Washington Nuclear 2	GE-6	BWR	1100	3323	1984	2024
Byron 1	WE-2	PWR	1105	3411	1985	2025
Catawba 1	WE-2	PWR	1129	3411	1985	2025
Diablo Canyon 2	WE-2	PWR	1087	3411	1985	2025
Enrico Fermi 2	GE-4	BWR	1075	3292	1985	2025
Palo Verde 1	CE	PWR	1221	3800	1985	2025
River Bend 1	GE-6	BWR	936	2894	1985	2025
Waterford 3	CE	PWR	1075	3390	1985	2025
Wolf Creek	WE-2	PWR	1135	3411	1985	2025
Catawba 2	WE-2	PWR	1129	3411	1986	2026
Hope Creek	GE-6	BWR	1031	3293	1986	2026
Millstone 3	WE-2	PWR	1142	3411	1986	2026
Palo Verde 2	CE	PWR	1221	3800	1986	2026
Perry 1	GE-6	BWR	1205	3579	1986	2026
Beaver Valley 2	WE-1	PWR	833	2660	1987	2027
Braidwood 1	WE-2	PWR	1120	3411	1987	2027
Byron 2	WE-2	PWR	1105	3411	1987	2027
Clinton	GE-6	BWR	930	2894	1987	2027
Harris	WE-1	PWR	860	2775	1987	2027
Nine Mile Point 2	GE-6	BWR	1080	3323	1987	2027
Palo Verde 3	CE	PWR	1304	3817	1987	2027
Vogtle 1	WE-2	PWR	1100	3411	1987	2027
Braidwood 2	WE-2	PWR	1120	3411	1988	2028
South Texas 1	WE-2	PWR	1250	3800	1988	2028
Limerick 2	GE-6	BWR	1055	3293	1989	2029
Seabrook	WE-2	PWR	1150	3411	1989	2029
South Texas 2	WE-2	PWR	1250	3800	1989	2029
Vogtle 2	WE-2	PWR	1097	3411	1989	2029
Comanche Peak 1	WE-2	PWR	1150	3411	1990	2030
Watts Bar 1	WE-2	PWR	1165	3411	1993	2033
Comanche Peak 2	WE-2	PWR	1150	3411	1993	2033
Watts Bar 2	WE-2	PWR	1165	3411	1995	2035

[a]ANS 1992, DOE 1991. Theoretical shutdown year was calculated by assuming that the reactor will operate 40 years after initial criticality.

Table 5.4.3. Projected GTCC LLW quantities for the reference reactor types used in the NLLWMP study[a]

Reactor type	No. of reactors	Reactor power [MW(t)]	Volume, ft³		Weight, lb		Activity, Ci	
			Core shroud	Core barrel	Core shroud	Core barrel	Core shroud	Core barrel
			Boiling-water reactors					
GE-4	22	2,440	102	NA	50,400	NA	9.87E+5	NA
GE-6	14	3,580	176	NA	87,000	NA	9.23E+5	NA
			Pressurized-water reactors					
B&W	7	2,770	52	90	44,400	25,600	2.89E+6	6.82E+5
CE	15	3,390	88	290	43,900	143,900	6.63E+6	3.30E+6
WE-1	28	3,020	45	125	22,300	61,700	5.19E+6	3.23E+5
WE-2	27	3,410	71	125	35,200	46,000	3.50E+6	4.37E+5

[a]NLLWMP 1991, Tables F-2, F-3, F-4, F-14, F-19, and F-20. The volumes shown are actual metal volumes with no packaging factors applied per individual reactor. Activities were calculated at time of shutdown.

Table 5.4.4. NLLWMP projected GTCC LLW volumes from LWR decommissioning[a]

Waste component	Estimated volumes of GTCC LLW, m^3					
	Unpackaged volume			Packaged volume		
	Low 2035	Base 2035	High 2035	Low 2035	Base 2035	High 2035
BWR						
Core shroud	129	129	129	180	257	386
PWR						
Core shroud	133	133	133	186	266	398
Core barrel	0	0	336	0	0	1010
Total	262	262	598	336	523	1794

[a]NLLWMP 1991, Figures 10-1, 10-2, and 10-8 through 10-15.

Table 5.4.5. NLLWMP projected GTCC LLW radioactivity from LWR decommissioning[a]

Radionuclide	Estimated radioactivity of GTCC LLW in 2035, Ci			
	BWR core shroud	PWR core shroud	PWR core barrel	Total radioactivity
C-14	3.52E+3	3.66E+4	2.90E+3	4.30E+4
Co-60	1.89E+6	1.59E+7	2.04E+6	1.98E+7
Fe-55	7.26E+5	8.46E+6	1.02E+6	1.02E+7
Mn-54	2.17E+2	1.55E+4	4.30E+3	2.00E+4
Ni-59	1.86E+4	1.08E+5	1.68E+4	1.43E+5
Ni-63	2.29E+6	7.14E+6	1.93E+6	1.14E+7
Nb-94	3.60E+1	5.74E+2	4.94E+1	6.59E+2
Total	4.93E+6	3.17E+7	5.01E+6	4.17E+7

[a]NLLWMP 1991, Figure 10-20.

5.5 OTHER SOURCES OF GTCC LLW

5.5.1 Introduction

Based on available information, this category of GTCC LLW comes (or will come) primarily as part of the waste stream resulting from the manufacture of sealed sources or from the future decommissioning of those facilities. This waste is thus contaminated with various radionuclides, including Sr-90, Cs-137, Am-241, several plutonium isotopes, and others. The majority of this waste is in the physical form of compactible trash such as gloves, plastic, paper, etc. Two major sources are available for data of this kind: the 1987 DOE Report to Congress on recommendations for management of GTCC LLW (DOE 1987) and the 1991 report by the NLLWMP on characterization of GTCC LLW (NLLWMP 1991). The DOE report includes some NRC estimates of decommissioning waste, while the NLLWMP report includes data obtained by the EIA in a survey conducted in 1986. These various studies do not use equivalent definitions or boundary conditions; thus there is some degree of uncertainty, not only in the figures themselves, but in how they should be interpreted and combined. It is anticipated that future work in this area by the NLLWPA will give improved definition of the waste quantities in this category.

5.5.2 Data from DOE 1987

At the time of this study, the various manufacturers involved had 30 m^3 of GTCC LLW on hand. The study quoted 1986 NRC data that projected at least 95 m^3 from future operations through 2020 involving Am-241 sealed source manufacturing by the largest manufacturers. Operations involving other radionuclides could be expected to add to this quantity. They also quoted NRC projections of 270 m^3 resulting from future decommissioning of these facilities. This gives a total of at least 395 m^3.

5.5.3 Data from NLLWMP 1991

This study gives a detailed breakdown by physical form (Table 5.5.1) and indicates that the largest category, compactible trash, comes mostly from sealed source manufacturing. However, other possible sources are apparently included, though not defined. The total volume for the base case in 2035 is 269 m^3 and for the high case in 2035 is 320 m^3, which are the values used in the summary for this chapter.

5.5.3 References for Section 5.5

DOE 1987. *Recommendations for Management of Greater-than-Class C Low-Level Radioactive Waste*, DOE/NE-0077, February 1987.

NLLWMP 1991. R. A. Hulse, *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities and Other Characteristics*, DOE/LLW-114, August 1991.

Table 5.5.1. Estimated volumes of GTCC LLW from other sources[a]

Type of waste	Packaged volumes, m^3		
	Low case 2035	Base case 2035	High case 2035
Metal shavings	5	5	5
Cartridge filters	2	46	92
Compactible trash	142	142	142
Contaminated equipment	23	23	23
Ion exchange resins	1	5	10
Filter media	9	9	9
Mixed oxide fuel pellets	4	4	4
Noncompactible trash	10	10	10
Organic liquids	6	6	6
Solidified resins	16	16	16
Metals from operations	3	3	3
Total	221	269	320

[a]Source: NLLWMP 1991, Fig. 11-41; referred to as "other generators." Data shown here are rounded.

5.6 POTENTIAL GTCC LLW FROM DECOMMISSIONING NON-LWRS AND FUEL CYCLE FACILITIES

5.6.1 Introduction

Potential GTCC LLW from miscellaneous sources, including the decommissioning of non-LWRs and other facilities, are discussed in this section. Potential GTCC LLW are wastes that could potentially be classified as GTCC LLW but whose classification at present is uncertain for various reasons. This includes wastes from the decommissioning of facilities owned or operated by licensees of NRC or Agreement States in connection with DOE defense-related or research-related activities. The classification of some of these wastes may require the resolution of legal questions.

As noted by the NLLWMP, another large volume of potential GTCC LLW comes from DOE-held potential GTCC LLW. No determination has yet been made concerning ultimate disposal requirements for DOE-held potential GTCC LLW (NLLWMP 1991).

For completeness, some wastes that have been reported as TRU wastes in earlier studies, but whose classification potentially might change, are included in this section.

5.6.2 Fort St. Vrain Reactor

As discussed in Sect. 4.2, the 330-MW(e) FSV reactor was shut down in August 1989. Projected wastes from DECON (dismantling) of this reactor have been estimated by Public Service of Colorado (PSC). The most recent estimate by PSC indicates that no GTCC LLW will be produced. An earlier estimate had indicated that 4 m^3 of GTCC LLW might be produced, out of a total of 4,000 m^3 of LLW. These were the control rod drive metal-clad reflectors. Subsequent calculations showed that these would fall into the Class C category (PSC 1991).

5.6.3 Peach Bottom Unit 1 Reactor

The 40-MW(e) Peach Bottom Unit 1 Reactor, discussed in Sect. 4.3, was shut down in 1974 and placed in SAFSTOR. The total volume of waste resulting from decommissioning was 400 m^3, including waste from the processing of 1 m^3 of contaminated liquid (IDB 1990). It was not stated by Philadelphia Electric whether any of this waste was GTCC LLW. It is possible that some of this waste may have already been disposed of; an effort is being made to resolve this possibility.

5.6.4 Cimarron Fuel Fabrication Facility

Decommissioning activities are in progress at the Cimarron fuel fabrication plant at Crescent, Oklahoma. These facilities were licensed by NRC. Table 5.6.1 shows the radioactive wastes removed through December 1989 and projected to be removed by the completion of the project. Over 95% of the waste is low specific activity, and the total potential GTCC LLW (listed as TRU waste in the table) was estimated to be 256 m^3, with a radioactivity of 10.87 Ci (IDB 1990, based on data from Cimarron).

5.6.5 NFS Fuel Fabrication Plant

A proposal was submitted by Nuclear Fuel Services (NFS) to DOE in 1983 whereby NFS would conduct decontamination and decommissioning of the NFS fuel fabrication plant at Erwin, Tennessee, as a demonstration project for DOE and supply DOE with data from the project, if DOE in turn would accept the waste and scrap generated by the project. This proposal was accepted and decontamination and decommissioning has begun. It was estimated by NFS that decontamination and decommissioning of the plant would generate about 1,130 m^3 of untreated solid and liquid low-level waste, which, after treatment, would yield about 156 m^3 of solid TRU waste (Daling 1986).

In June 1991 it was decided that any transuranic-contaminated waste generated will be shipped to Oak Ridge National Laboratory for interim storage. No new estimates are available of the quantities of waste that will be generated.

5.6.6 Exxon Nuclear Company Fuel Fabrication Plant

Decommissioning of an Exxon Nuclear Company mixed-oxide fuel fabrication plant was estimated to generate about 7 m^3 of TRU waste (34 55-gal drums). These are miscellaneous metallic equipment items and tools (Daling 1986).

5.6.7 Babcock and Wilcox Plutonium Fuels Laboratory

In a proposal by Babcock and Wilcox (B&W), it was estimated that decommissioning of the B&W plutonium fuels development laboratory at Lynchburg, VA, would generate about 44 m^3 of TRU waste (210 55-gal drums). The wastes were expected to consist of concrete, building rubble, soil, metallic equipment such as glove boxes and associated hardware, and solidified decontamination solutions (primarily from the removal of epoxy paint and tile cement from masonry surfaces) (Daling 1986). This may be the same waste that is listed in Table 10-23 of NLLWMP 1991. An attempt is being made to resolve this question.

5.6.8 Sequoyah Fuels Corporation Mixed Oxide Plant

Decontamination and decommissioning of the Sequoyah Fuels Corporation mixed oxide fuel facility near

Oklahoma City, Oklahoma, was expected to generate about 1,840 55-gal drums (380 m^3) of TRU waste. The waste materials include sectioned metal items (glove boxes, storage tanks, pulse columns), filters, wipers, plastics, glass, and cellulosic materials (Daling 1986, Hazelton 1983). This may be the same waste that is listed in Table 10-23 of NLLWMP 1991. An attempt is being made to resolve this question.

5.6.9 DOE-Held Potential GTCC LLW

Table 5.6.2 lists potential GTCC LLW that has been accepted by DOE from NRC and Agreement State licensees for storage until a disposal facility is in operation. The source of this table is the NLLWMP report (NLLWMP 1991). The data reflect information compiled in late 1990. It is not determined whether all these wastes will require disposal in an NRC-licensed facility; some may be decided to be owned by DOE and acceptable for disposal in a DOE facility (NLLWMP 1991). The total in the NLLWMP report is 1,076 m^3 for both the base and high cases in year 2035. This total includes portions of the decommissioning wastes discussed in Sections 5.6.4–5.6.8.

5.6.10 Summary of Potential GTCC LLW

Table 5.6.3 summarizes the total GTCC LLW volumes estimated in Sect 5.6. This includes the 1,076 m^3 of DOE-held potential GTCC LLW. The quantities of GTCC LLW from decommissioning non-LWRs and fuel cycle facilities have been adjusted by subtracting the quantities already shipped to DOE and listed as DOE-held waste.

5.6.11 Potential Wastes from Fuel Rod Consolidation

Fuel rod consolidation is a process whereby spent fuel rods are separated from fuel disassembly hardware in order to reduce the space requirements for rod storage or transport. The process has been demonstrated but thus far has not been used commercially. As part of the NLLWMP studies, follow-up contacts were made with utilities that had indicated possible use of fuel rod consolidation in the 1986 EIA survey. These contacts indicated that fuel rod consolidation is unlikely to be used. In the technical review process conducted by the NLLWMP, it was decided that any waste generated by fuel rod consolidation would not be considered GTCC LLW, and that such waste, if any, would be likely to be considered high-level waste (NLLWMP 1991). The possible use of fuel rod consolidation at an MRS facility is discussed in Sect. 5.7.

5.6.12 References for Section 5.6

Daling 1986. P. M. Daling, J. D. Ludwick, G. B. Mellinger, and R. W. McKee, *Repository Disposal Requirements for Commercial Transuranic Wastes (Generated Without Reprocessing)*, report PNL-5597, Pacific Northwest Laboratory, June 1986.

Hazelton 1983. R. F. Hazelton, "Commercial Transuranic Waste Inventory Survey," letter report, Pacific Northwest laboratory, Richland, Washington, 1983.

IDB 1990. *Integrated Data Base for 1990: Spent Fuel and Radioactive Waste Inventories, Projections and Characteristics*, DOE/RW-0006, Rev. 6, October 1990.

IDB 1991. *Integrated Data Base for 1991: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Rev. 7, October 1991.

NLLWMP 1991. R. A. Hulse, *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics*, DOE/LLW-114, August 1991.

PSC 1991. Letter from F. J. Novachek, PSC, to S. N. Storch, ORNL, July 30, 1991.

Table 5.6.1. Characteristics of wastes from decommissioning activities at the Cimarron Fuel Fabrication Facility[a]

Project area	Total waste removed through December 1989	
	Volume (m^3)	Radioactivity (Ci)
LLW of low specific activity		
Burial ground	1,833	5.37
Mixed oxide fuel plant	464	3.25
Uranium fuel plant	1,680	3.22
Liquid process waste evaporation ponds		
Mixed oxide plant pond	104	b
Uranium plant pond	184	0.23
Sanitary lagoons	1,559	2.93
Total LLW (low specific activity)	5,824	15.00
TRU waste	256	10.87
Total waste[c]	6,080	25.87

[a]Adapted from IDB 1990, Table 7.18. This was based on data from Cimarron.

[b]The radioactivity of this waste is 9.03E-06 Ci.

[c]It is projected by Cimarron that an additional 1,000 m^3 remain to be removed by the end of the project. This is all LLW of low specific activity from the uranium fuel plant area.

Table 5.6.2. Potential GTCC LLW from NRC- or Agreement State-licensed facilities currently held at U.S. Department of Energy sites[a]

Field office	Generator/user	Approximate packaged volume (m^3)	Estimated radioactivity (Ci)	Description	Status
ID	J.C. Haynes	6.1	25.4	5 drums in 2 boxes	Stored at ILTSF[b]
ID	Monsanto	5.7	453.4	27 55-gal drums	Stored at ILTSF
ID	Monsanto	13.6	6.6	4 bins	Stored at ILTSF
ID	Babcock & Wilcox (VA)	4.0	46.3	19 55-gal drums	Stored at ILTSF
ID	GPU Nuclear Corporation (TMI)	1.2	43.3	3 casks	Stored at TAN[c]
RL	GE Vallecitos	87.5	NA	8 boxes	Retrievable storage
RL	GE Vallecitos	4.2	60,000	6 casks	Retrievable storage
RL	Babcock & Wilcox (PA)	11.9	NA	2 boxes	Retrievable storage
RL	Babcock & Wilcox (PA)	63.8	NA	304 55-gal drums	Retrievable storage
RL	Westinghouse (PA)	78.5	NA	374 55-gal drums	Retrievable storage
RL	Westinghouse (PA)	173.6	NA	27 boxes	Retrievable storage
RL	Rockwell (CA)	0.2	NA	1 source in 55-gal drum	Retrievable storage
RL	Rockwell (CA)	41.0	NA	195 55-gal drums	Retrievable storage
RL	GPU Nuclear Corporation (TMI)	161.4	477,700	14 casks	Disposed in trench
RL	Battelle Columbus	0.2	NA	1 source in 55-gal drum	Retrievable storage
SR	Allied General Nuclear	10.9	NA	52 55-gal drums	Retrievable storage
	Total	664.	538,000		

[a]Source: NLLWMP 1991. Two entries (for NFS and Sequoyah) that are in the source table are deleted here since they are listed separately in Table 5.6.3.

[b]ILTSF = Intermediate Level Temporary Storage Facility (INEL).

[c]TAN = Test Area North (INEL).

Table 5.6.3. Summary of potential GTCC LLW from decommissioning non-LWRs and fuel cycle facilities, including DOE-held wastes[a]

Reactor or facility	Estimated total packaged volume of GTCC or TRU waste (m^3)
Fort St. Vrain Reactor	0
Peach Bottom Unit 1 Reactor	b
Cimarron Fuel Fabrication Facility	256
Nuclear Fuel Services Fabrication Plant	156
Exxon Nuclear Company Fuel Fabrication Plant	7
Babcock and Wilcox Plutonium Fuels Laboratory[c]	44
Sequoyah Fuels Corporation Mixed Oxide Plant	380
DOE-held potential GTCC LLW[d]	664
Total	1,507

[a]This table summarizes the estimates discussed in Section 5.6.
[b]Information not available.
[c]It is not clear from the available data if any or all of this is accounted for in Table 5.6.2.
[d]A detailed breakdown is given in Table 5.6.2.

5.7 OTHER DOE WASTES THAT MAY REQUIRE REPOSITORY DISPOSAL

5.7.1 Introduction

The Civilian Radioactive Waste Management System will consist of one or more repositories, a Monitored Retrievable Storage (MRS) facility (if authorized), and possibly smaller specialized facilities such as a transport cask maintenance and overhaul facility. Each of these facilities will in turn generate secondary wastes. Some of the secondary wastes will contain radioactivity from facility maintenance and decontamination operations, some may be hazardous as defined by the Resource Conservation and Recovery Act (RCRA) and regulated under 40 CFR 261, and some may be both, i.e., so-called "mixed" waste. For completeness, TRU waste generated at the West Valley Demonstration Project is included in this section, although it is expected that such TRU waste will go to the Waste Isolation Pilot Plant (WIPP) facility.

5.7.2 Secondary Wastes from an MRS Facility

A conceptual design study of an MRS facility was done by R. M. Parsons Co. in 1985 (Parsons 1985). The facility design included dry rod consolidation of spent fuel assemblies. It was estimated by Parsons that if dry rod consolidation of spent fuel assemblies is practiced at such a facility as much as 315 m^3/year of TRU waste might be generated. This was based on a fuel consolidation rate of 3,000 MT/year (Parsons 1985, 1987, Daling 1986). The estimate given in Daling 1986 showed an annual generation of 3,060 ft^3 (87 m^3) of contact-handled TRU waste and 8,050 ft^3 (228 m^3) of remote-handled TRU waste.

A breakdown of the Parsons estimate (Table 5.7.1) shows that 83% of the total estimated annual production of TRU waste was HEPA filters and frames. In the previous edition of this report, it was stated that this rate of consumption of HEPA filters seemed overly conservative. At the present time, a new design study of an MRS is under way. If dry rod consolidation is included in this study, updated information on HEPA filter usage and TRU waste production rates may be obtained. However, a Draft Mission Plan Amendment issued by the DOE Office of Civilian Radioactive Waste Management in September 1991 (DOE 1991) does not include dry rod consolidation as part of the mission of the MRS facility.

If dry rod consolidation is not performed at an MRS, but is performed elsewhere in the waste management system (for example, at a repository), it is expected that the rate of generation of waste would be about the same as that estimated for the MRS. If dry rod consolidation is not performed anywhere in the system, this would eliminate the major source of OCRWM-generated TRU waste, since other routine operations within the system, such as shipping, fuel handling, and underground operations, are not expected to generate any appreciable volumes of TRU wastes. However, the possibility must be considered that some such wastes might be generated.

5.7.3 Cask Maintenance Facility

A recent feasibility study done by Oak Ridge National Laboratory for OCRWM provides a basis for initiating the conceptual design of a cask maintenance facility (Rennich 1991). The report of the study discusses the functions of a cask maintenance facility (CMF) within the transportation system and gives preliminary costs, specifications, and schedules.

The primary functions of a CMF would include inspecting, servicing, testing, and repairing casks and their components. Servicing would include cleaning and decontamination. Other functions would be to reconfigure casks to accommodate different spent fuel baskets, to provide for the temporary storage of unloaded casks, and to prepare casks for decommissioning and disposal. A CMF could be located at the MRS, at the repository, or as a stand-alone facility (OCRWM 1991).

The feasibility study did not develop any estimates of the quantities of wastes that might be generated by the operation of a CMF. Casks received by the CMF are expected to be contaminated both internally and externally. It is likely, therefore, that some wastes would be generated by cleaning, decontamination, and decommissioning operations. It is not clear whether such wastes would be classified as DOE wastes or as wastes that would require disposal in an NRC - licensed facility. However, since these operations are all performed on unloaded (empty) casks, and since the levels of contamination of incoming casks are subject to controls imposed on the system, it seems unlikely that large quantities of waste requiring repository disposal would normally be produced.

5.7.4 West Valley Demonstration Project

TRU waste generated at the West Valley Demonstration Project through December 1989 is reported to be 42 m^3 with a total radioactivity of 66 Ci. This includes 14 m^3 (13.2 Ci) generated during 1989 from HEPA filters. The total quantity of TRU waste expected at the completion of the project is projected to be 300 m^3 with a radioactivity of 350 Ci (IDB 1990).

5.7.5 References for Section 5.7

Daling 1986. P. M. Daling, J. D. Ludwick, G. B. Mellinger, and R. W. McKee, *Repository Disposal Requirements for Commercial Transuranic Wastes (Generated Without Reprocessing)*, report PNL-5597, Pacific Northwest Laboratory, June 1986.

DOE 1991. Draft Mission Plan Amendment, U.S. Department of Energy Office of Civilian Radioactive Waste Management, September 1991.

IDB 1990. *Integrated Data Base for 1990: Spent Fuel and Radioactive Waste Inventories, Projections and Characteristics*, DOE/RW-0006, Rev. 6, October 1990.

OCRWM 1991. OCRWM Bulletin DOE/RW-0305p, April 1991.

Parsons 1985. *Integral Monitored Retrievable Storage (MRS) Facility Conceptual Design Report*. Vol. I, Book 2 "Design Description." MRS-11. Prepared for U.S. Department of Energy, Richland Operations Office, Richland, Washington, by Ralph M. Parsons Co., 1985.

Parsons 1987. *Evaluation of the Effluent Streams from the Conceptual Design of the Monitored Retrievable Storage Facility*, Ralph M. Parsons Co., June 1987.

Rennich 1991. M. J. Rennich, L. G. Medley, and C. R. Attaway, *Feasibility Study for A Transportation Operations System Cask Maintenance Facility*, ORNL/TM-11019, January 1991.

Table 5.7.1. Estimated annual generation of TRU waste from an MRS facility[a]

Type of waste	TRU waste generation rate	
	Volume, m^3/year	55-gal drums/year
Contact-handled TRU waste		
Cemented waste	44	221
HEPA filters (wooden-framed)	41	207
Noncombustibles (equipment, tools, etc.)	2	9
Remote-handled TRU waste		
Spent resin	2	10
HEPA filter frames (wooden)	8	39
Solidified evaporator bottoms	3	17
HEPA filter frames (metal)	104	523
HEPA filter media	111	562
Total	315	1,588

[a]Source: Parsons 1985, as cited in Daling 1986. Plant throughput is 3,000 MT/year.

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