DOE/RW-0184
VOLUME 7 of 8
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Office of Civilian Radioactive Waste Management

CHARACTERISTICS OF SPENT FUEL, HIGH-LEVEL WASTE, AND OTHER RADIOACTIVE WASTES WHICH MAY REQUIRE LONG-TERM ISOLATION

JUNE 1988

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Washington, D.C. 20585

DISCLAIMER

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UC-70,-71, and -85

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PREFACE

Volumes 7 and 8 complete the first edition of this report. Volumes 1-6, dated December 1987, covered chapters 1 through 3 (Vol. 1) and the supporting appendices (Vols. 2-6). Volume 7 provides chapters 4 and 5, and Vol. 8 provides the appendices to Chapter 7.

The purpose of this report, and the information contained in the associated computerized data bases, is to establish the DOE/OCRWM reference characteristics of the radioactive waste materials that may be accepted by DOE for emplacement in the mined geologic disposal system as developed under the Nuclear Waste Policy Act of 1982. This report provides relevant technical data for use by DOE and its supporting contractors and is not intended to be a policy document.

This document is backed up by five PC-compatible data bases, written in a user-oriented, menu-driven format, which were developed for this purpose. These are:

LWR Assemblies Data Base:	Physical properties of intact assemblies and radiological properties of spent fuel disassembly hardware.
LWR Radiological Data Base:	Radiological properties of intact spent fuel as a function of burnup and age.
LWR Quantities Data Base:	Inventories and projected quantities of LWR spent fuel.
LWR NFA Hardware Data Base:	Physical and radiological properties of Non-Fuel Assembly hardware.
High-Level Waste Data Base:	Quantities and radiological properties of HLW as a function of age, for both interim and immobilized forms.

The above data bases may be ordered using the form printed on the following page.

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Systems Integration Program
c/o Dr. Karl J. Notz
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6233

For further information, call (615) 574-6632 or FTS 624-6632.

ACKNOWLEDGEMENTS

This document and the supporting data bases were prepared under the direction and sponsorship of the Systems Integration team of the Office of Civilian Radioactive Waste Management and with support from the Office of Geologic Repositories, the Office of Resource Management, and the Energy Information Administration. The work was carried out by the Oak Ridge National Laboratory with input from other laboratories and contractors. The following made substantial contributions:

Advanced Nuclear Fuels (formerly Exxon)
Analysas Corporation
Babcock & Wilcox
Combustion Engineering
GA Technologies
Idaho National Engineering Laboratory
E. R. Johnson Associates, Inc.
Maxima Corporation
PAI Corporation
Battelle-Pacific Northwest Laboratory
Savannah River Plant
Westinghouse Advanced Power Systems
Westinghouse Hanford Co.
West Valley Nuclear Services Co.

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 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
DHLW	defense high-level waste
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
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
FWMS	Federal Waste Management System
GAPSR	gray axial power shaping rod
GE	General Electric
GTCC	Greater than Class C
HANF	Hanford
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
LWBR	Light-water Breeder Reactor
LWR	light-water reactor
MOX	mixed oxide
MRS	monitored retrievable storage
MSRE	Molten Salt Reactor Experiment
MTIHM	metric tons of initial heavy metal
MTR	Materials Test Reactor
NCAW	neutralized current acid waste

NFA	nonfuel assembly
NFB	nonfuel bearing
NMMSS	Nuclear Materials Management and Safeguards System
NRC	Nuclear Regulatory Commission
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
PBI	Peach Bottom Unit I
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
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
WAC	waste acceptance criteria
WAPS	Waste Acceptance Preliminary Specification
WE	Westinghouse
WIPP	Waste Isolation Pilot Plant
WVDP	West Valley Demonstration Project

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4. NON-LWR SPENT FUELS

4.1 INTRODUCTION

Although LWR spent fuel constitutes, by far, the major portion of all domestic spent fuels, the fuels from other reactor types introduce a large number of specialized fuel forms and compositions. While the quantities of these other spent fuels are generally relatively minor, the variety of materials to be addressed may impose a significant additional burden. Problems not encountered with LWR spent fuel will have to be dealt with in a manner that does not compromise the primary purpose of a geologic repository, namely, to receive and safely isolate immobilized HLW and LWR spent fuel. These other spent fuels usually involve different chemical compositions, either in the fuel itself or in the cladding (or other matrix material), or in both. Many are of low burnup, which minimizes the thermal load and radioactivity but, when associated with highly enriched uranium, requires careful attention to prevention of possible criticality, and to appropriate safeguard procedures.

At this time, we have characterized other spent fuels and the quantities thereof in inventory on the basis of presently available information and estimated their projected future quantities. Quantities are stated in terms of number of fuel assemblies or mass of initial heavy metal. This report does not include estimates of repository volume requirements for these materials.

Relatively detailed chemical and physical descriptions of the Fort St. Vrain and Peach Bottom I fuels are either provided or referenced in this report. In general, the miscellaneous fuels located at eight different sites are described only superficially. In some cases, reference report numbers are provided. There is no immediate plan to develop detailed information on these miscellaneous materials.

In cases where miscellaneous spent fuels are presently being reprocessed or scheduled for reprocessing (in DOE facilities at the Savannah River Plant and the Idaho National Engineering Laboratories),

we have assumed that the existing situation would continue. (In some cases, there is substantial uncertainty in this regard.) Where specialized treatment may be required, we have made minimal assumptions.

The major categories covered in this chapter include HTGR spent fuel from the Fort St. Vrain and Peach Bottom I reactors, research and test reactor fuels for which the TRIGA reactors are a prominent contributor, and miscellaneous other fuels that are described in terms of the facility where they are presently stored.

4.2 FORT ST. VRAIN REACTOR FUEL

The Fort St. Vrain (FSV) reactor is a high-temperature, gas-cooled (helium) reactor located in Platteville, Colorado. Its operation started in January 1979 with a rated power of 842 MW thermal. The total initial core loading was 774 kg of 93.5% ^{235}U and 15905 kg Th.

Although at the present time (June 1987) the reactor is operating at about 35% of its design capacity, much of the time recently has been spent in upgrading the system to meet NRC requirements imposed as a result of the TMI incident. If scheduled operation proceeds favorably, the next (fourth) reload cycle is expected to occur in late 1988.

4.2.1 FSV Reactor Core

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 total fuel columns number 247. Thirty-one out of 37 of the regions consist of seven columns, a center control fuel column and six surrounding columns made up of fuel elements of conventional design. The other six regions located near the edge of the core contain one control fuel column and four fuel element columns. The full core consists of this pattern stacked six-high, bringing the total to 1482 fuel elements (Morrissette 1986).

The FSV reactor is designed to operate on a graded fuel cycle with about one-sixth of the reactor core being replaced at each refueling. A full refueling cycle consists of five reloads of 240 elements each and one reload of 282 elements.

4.2.2 FSV Fuel Element Characteristics

The reactor core is made up of three types of fuel elements: standard, control, and bottom control fuel elements. The fuel elements (Fig. 4.2.2) are hexagonal, graphite blocks which have been 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.

Standard Fuel Element

All standard fuel elements have 210 fuel holes 0.500-in. in diameter and 108 coolant passages (Bingham 1976). When fully loaded they contain 3132 fuel rods, which are right cylinders made of coated particles that are bonded together with a low density graphite matrix. Nominal fuel rod dimensions are 0.5-in. by 1.94-in. long.

Control Fuel Elements

The center control rod fuel element in each region is similar to the standard fuel elements, 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 57 coolant channels. The control rod channels have a diameter of 4.00 in. and a centerline of separation 9.72 in. The reserve shutdown channel has a diameter of 3.75 in.

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 1302 fuel rods (Fig. 4.2.4).

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 in the control or bottom control elements.

Coated Fuel Particles

The fuel particles are TRISO-coated microspheres of uranium and thorium carbide. The particle coating is comprised of four layers: a low-density pyrolytic carbon coating, a high-density pyrolytic carbon coating, a silicon carbide coating, and an outer layer of high-density pyrolytic carbon. Two particle sizes are used, about 460 and 730 microns in diameter.

Fuel Element Weights

The weights of the various types of fuel elements vary from 128 kg to 109 kg and are listed by type in Table 4.2.1 (Kowal 1984 and Kapernick 1973).

4.2.3 FSV Fuel Chemical Characteristics

The fuel block is nuclear grade graphite, type H-327 or type H-451, manufactured by Great Lakes Carbon Company with a very low level of impurities.

The specified maximum concentrations for impurities in the graphite (Disselhorst 1972) were as follows:

Boron (755 barns/atom)	-	5 ppm
Iron (2.4 barns/atom)	-	100 ppm
Titanium (5.6 barns/atom) + Vanadium (5.1 barns/atom)	}	100 ppm total
Nitrogen (not specified)		25 ppm
Total ash	-	1000 ppm

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

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:

Iron	≤500 ppm
Sulphur	≤1200 ppm
Titanium	≤50 ppm
Vanadium	≤50 ppm
Hydrogen (residual core average)	≤200 ppm
Residual ash	≤300 ppm at 900°C
H_2O	≤400 ppm
Nitrogen (not specified)	~25 ppm
Total concentration (boron equivalent)	<5 ppm

Coated Fuel Particles

The coated fuel particles in spent fuel contain mainly uranium, thorium, and mixed fission products. A small amount of transuranic actinides is also present. The uranium and thorium are in the form of carbide.

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

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 specified as:

Boron	250 ppm
Cadmium	250 ppm
Hafnium	250 ppm
Sulfur	250 ppm
Total, all other metals	5000 ppm

4.2.4 FSV Radiological Characteristics

Three reactor segments have been discharged from FSV as of June 1987.

4.2.4.1 Discharge Schedule

Table 4.2.2 shows the schedule of spent fuel discharged from the FSV reactor and the currently-estimated projection of discharges through the year 2002; no projections are available beyond this point. The data shown are from the Integrated Data Base for 1987 (DOE 1987).

4.2.4.2 Post Irradiation 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 shrank 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 of about 100,000 MWd/MTIHM. The maximum burnup for the discharged fuel occurred in an element from segment 3 at 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 4.2.3.

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 4.2.4. The differences between calculated and measured composite burnups (~ total power generation) 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.

A comparison of measured and calculated uranium isotopic concentrations for the same surveillance element 1-0743 is given in Table 4.2.5. 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 is not reported.

The radiological characteristics at extended decay times of average FSV fuel irradiated to 100,000 MWD/T have been calculated for 120-d decayed fuel (Morissette 1986). The calculated radioactivity for selected nuclides as a function of time is shown in Table 4.2.6; the calculated total instantaneous heat is given in Fig. 4.2.5. For fuels with reduced irradiation, an acceptable first approximation should be using linear interpolation from the values for 100,000 MWD/MTIHM.

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 which are 11 ft long. They have ungasketed lids which are held in place by remotely operable DE-STA-CO clamps (see Fig. 4.2.5 and also Bingham 1976 for additional details). Each canister contains four FSV elements. The current inventory of 724 elements is contained in 181 canisters. The ICPP has requisite information on element serial numbers and canister numbers currently housing the elements.

4.2.6 Quantities to be Disposed

As indicated previously (Table 4.2.2), the total projected spent fuel discharges through the year 2002 are 2724 elements with a total mass of 29.5 MTIHM. If spent fuel discharges continue at the rate shown in Table 4.2.2 through the year 2020, the total at that time would be about 4,900 fuel assemblies, or about 53 MTIHM. Fuel has been fabricated for ten reloads, i.e. 7 more discharge cycles, through year 2000 based on the schedule in Table 4.2.2.

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 ensure that the history of each element can be appropriately traced as needed. A detailed exposition of the numbering logic is contained in the Appendix 4C.

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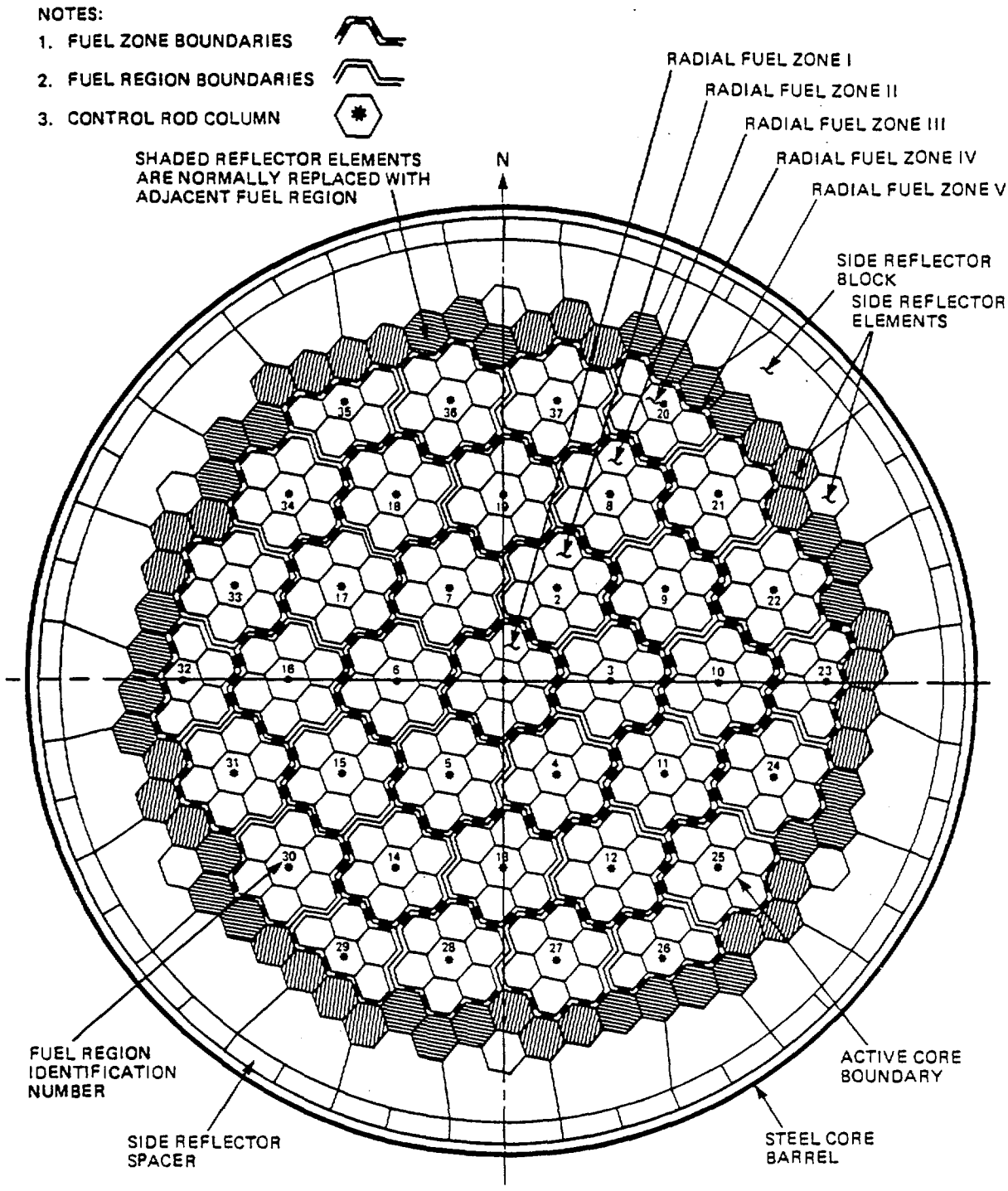

Fig. 4.2.1. FSV core plan view.

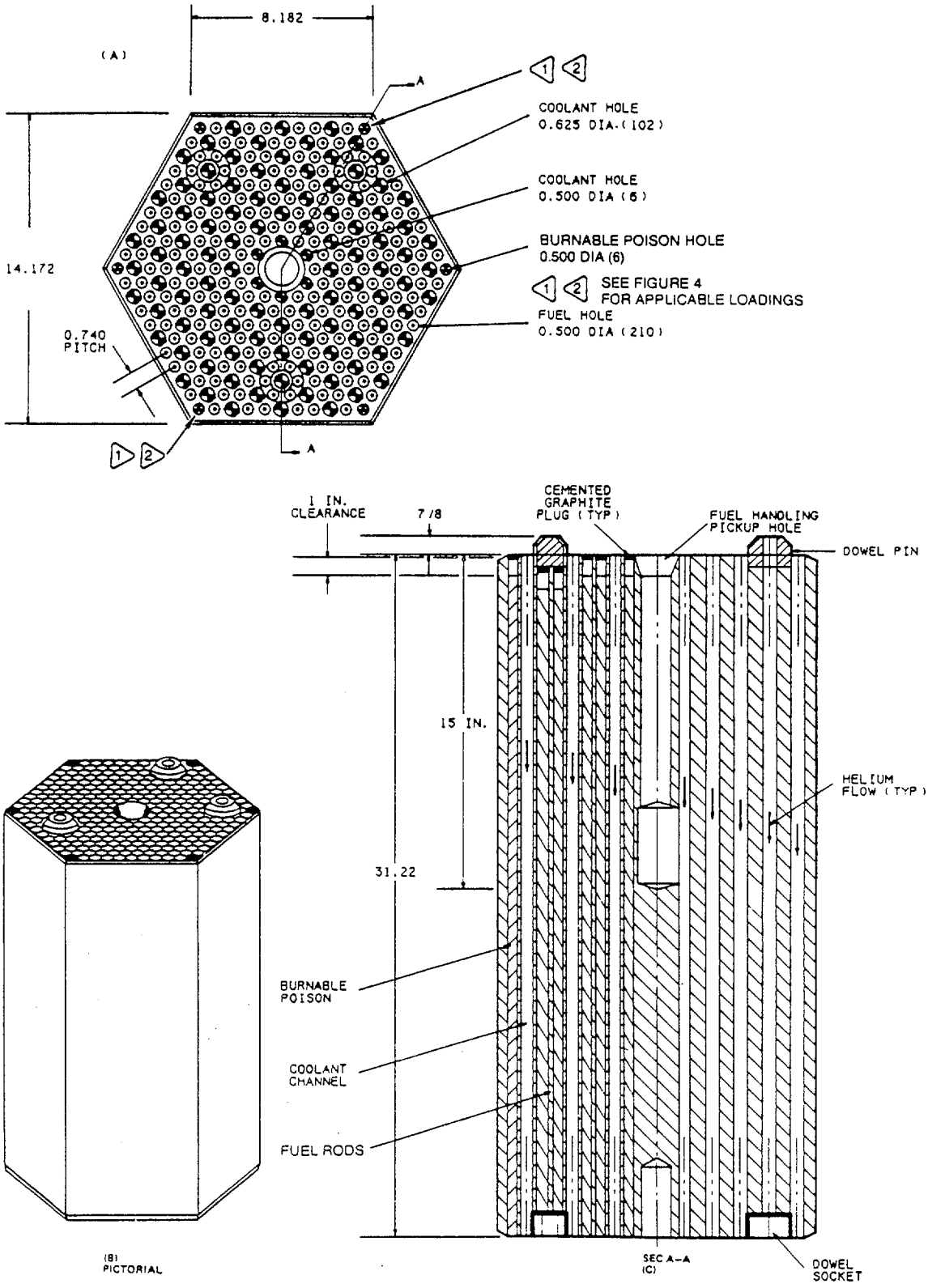

Fig. 4.2.2. FSV standard fuel element.

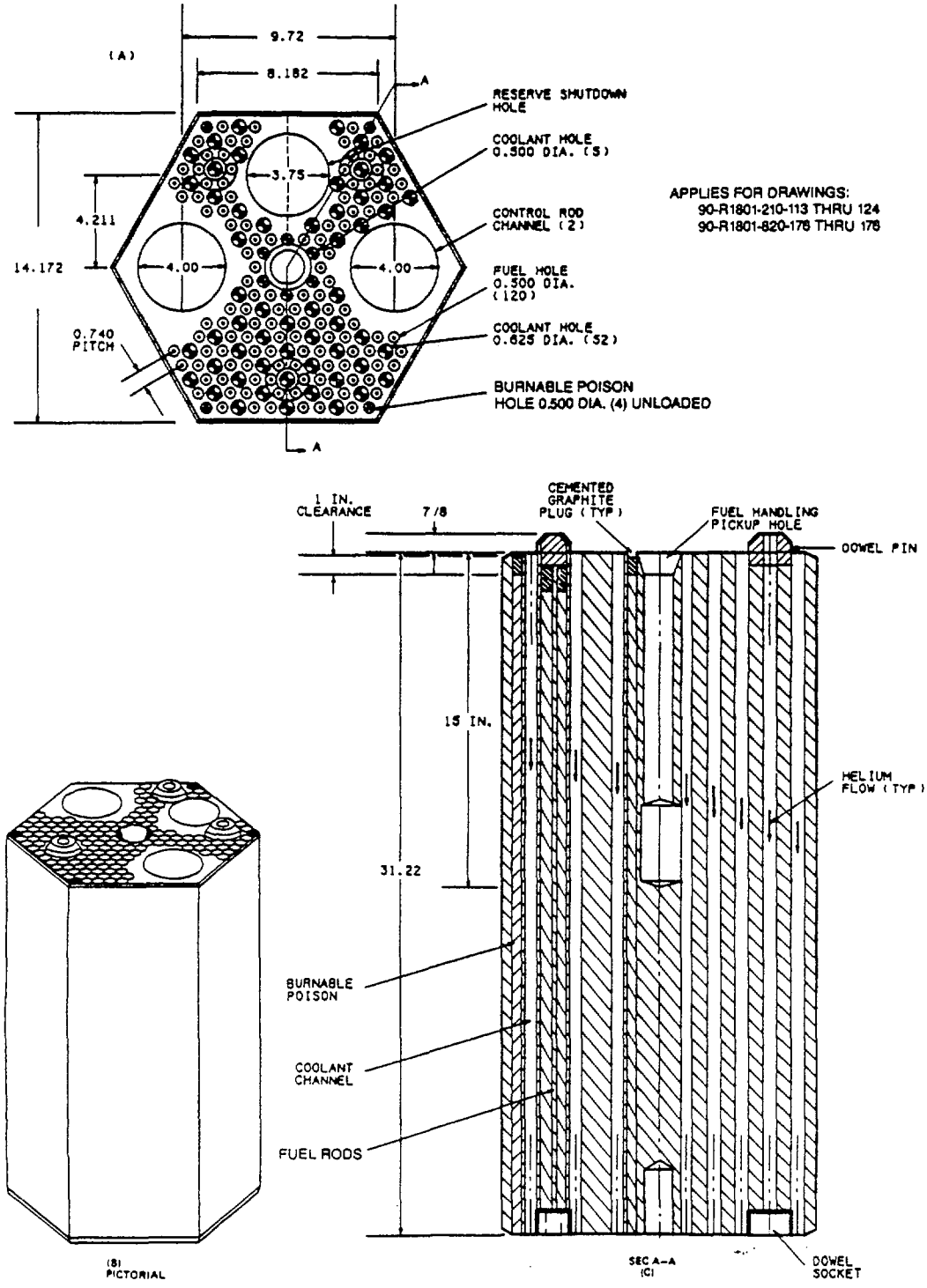

Fig. 4.2.3. Control fuel elements and surveillance control element.

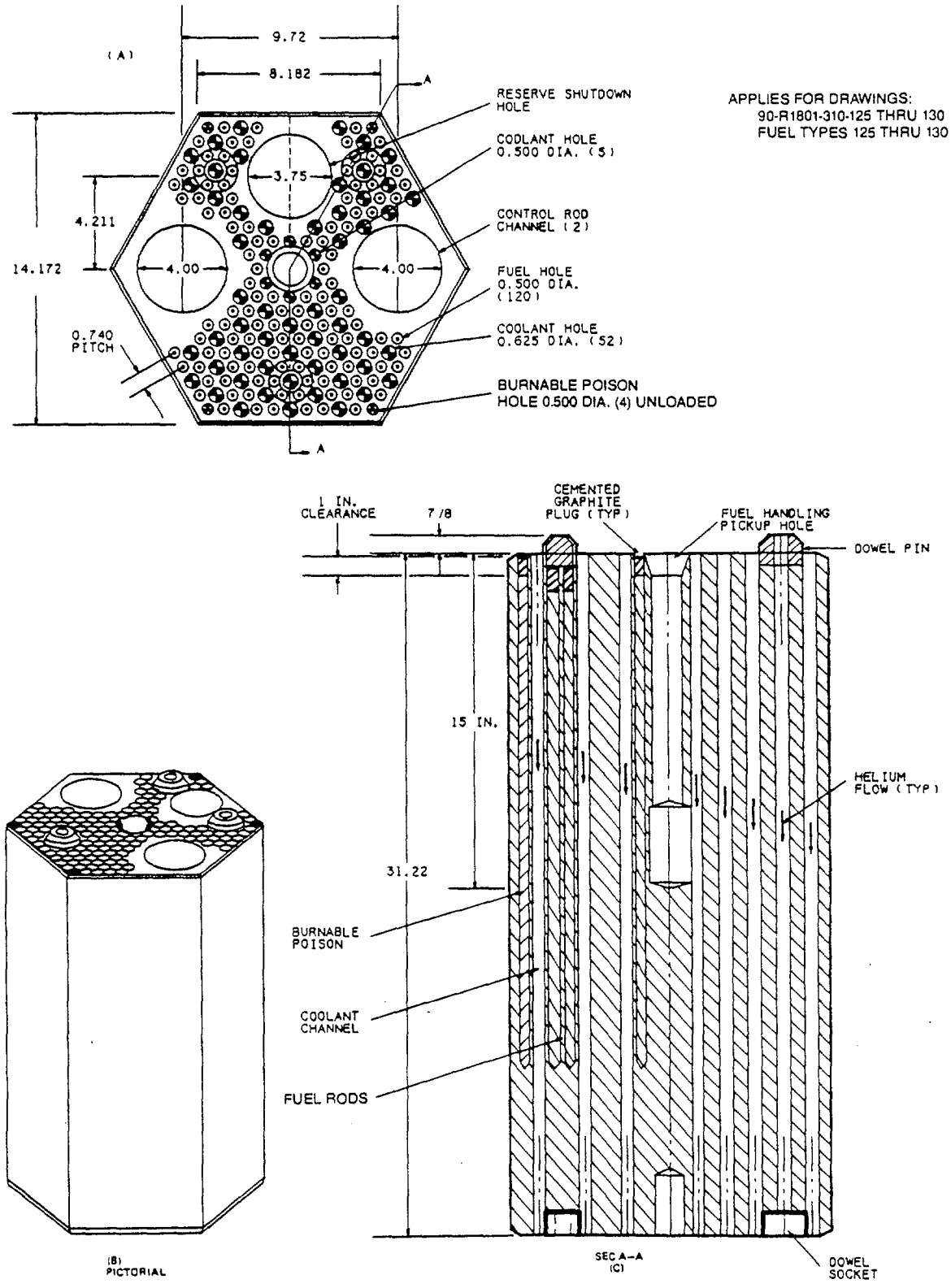

Fig. 4.2.4. Bottom control fuel element.

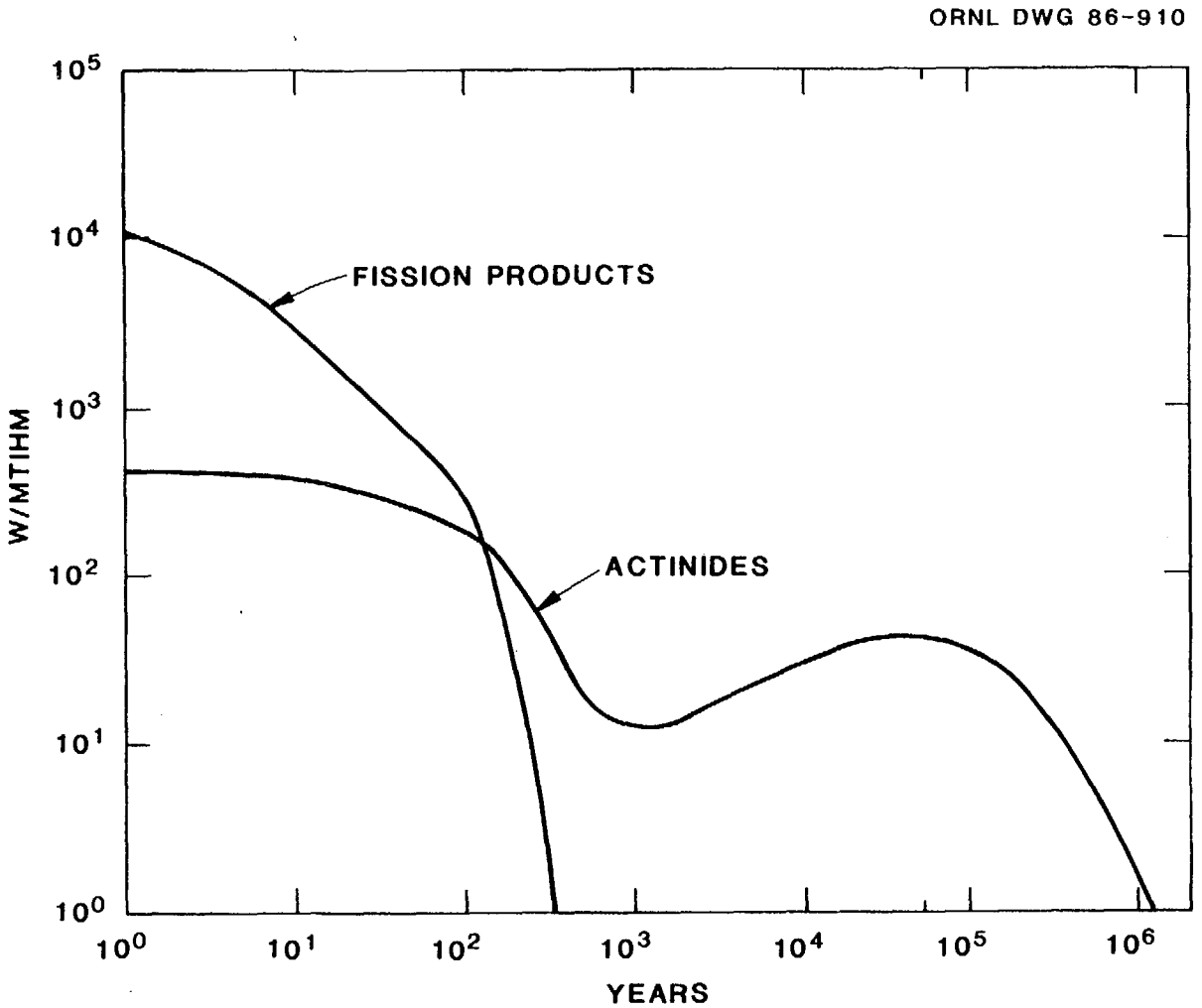

Fig. 4.2.5. Decay heat for Fort St. Vrain spent fuel. Basis: 1 metric ton initial heavy metal irradiated to 100,000 MWd/MTIHM.

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, 3, 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. FSV fuel element total and component weights

Element Total Weights

Element type	Weight
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

Component Weights

Component identity		
Graphite body:		
Regular fuel element		86 kg
Control rod fuel element		85 kg
Bottom control rod fuel element		94 kg
Fuel rod		13 g
Thorium	4.2 g	
Uranium	0.2 g	
Silicon	1.3 g	
Coatings	6.4 g	
Matrix	1.3 g	
Poison rod		100 g

NOTE: All weights are approximate.

Table 4.2.3. Sample of fuel accountability data

Serial number		1-1773	Accountability date:	3/31/86
Core location				
Region 18				
Column 7				
Layer 7				
			Heavy metal weights (GM)	
	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		
	Fissile	Pa-231	1,832.23	1,771.69
	Fissile	U-232	.00	.01
	Fissile	U-233	.00	.01
	Fissile	U-234	.00	33.60
	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
PPM U-232			.00	90.79
Fertile particle fima, %			.00	1.29
Fissile particle fima, %			.00	11.27
Burnup (MWd/tonne)				32,601.50
Cumulative EFPD				657.30

[a]Includes full decay of Pa-233.

[b]Includes full decay of Np-239.

Table 4.2.4. Comparison of calculated and measured fuel burnup for FSV fuel element 1-0743

	Burnup										
			Case I[b]			Case II[c]			Case IV[d]		
	Measured[a]			$Z = \frac{Calc}{Meas} - 1$			$Z = \frac{Calc}{Meas} - 1$			$Z = \frac{Calc}{Meas} - 1$	
Particle type	FIMA (%)	$\pm 1\sigma$ (%)	FIMA (%)	Z (%)	$\pm 1\sigma^e$ (%)	FIMA (%)	Z (%)	$\pm\sigma^e$ (%)	FIMA (%)	Z (%)	$\pm\sigma^e$ (%)
$(Th,U)C_2$	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]Progressed uncertainty due to measurement uncertaintv only.

Table 4.2.5. Comparison of calculated and measured uranium isotopic concentrations for UC_2 burnup monitors irradiated in FSV fuel element 1-0743

	Isotopic concentration				
	Measured[a]			Relative difference $Z = \frac{Calc}{Meas} - 1$ (%)	
Isotope	Atom percent	σ	Calculated[b] atom percent	Z	$\pm\sigma$[c]
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]Progressed uncertainty due to measurement uncertainty only.

Table 4.2.6. Radioactivity of Fort Saint Vrain reactor spent fuel.

BASED ON ONE MTIHM; 100,000 MWD/MTIHM

CURIES

NUCLIDE	120.0D	1.0YR	10.0YR	100.0YR	1000.0YR	10.0KY	100.0KY	1.0MY
				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

Table 4.2.6 (continued)

CURIES

NUCLIDE	120.0D	1.0YR	10.0YR	100.0YR	1000.0YR	10.0KY	100.0KY	1.0MY
				ACTINIDES AND DAUGHTERS				
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
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

Table 4.2.6 (continued)

CURIES

NUCLIDE	120.0D	1.0YR	10.0YR	100.0YR	1000.0YR	10.0KY	100.0KY	1.0MY
				FISSION PRODUCTS				
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

NUCLIDES CONTRIBUTING < 0.0010 % ARE OMITTED

4.3 PEACH BOTTOM UNIT 1

The Peach Bottom Unit 1 (PB1) rated at 115 MW(t), was a high-temperature gas-cooled reactor that operated during the period 1966 to 1974. It utilized a 3.5 in.-diam by 12-ft-long cylindrical fuel element (assembly) made up largely of graphite but containing about 1.8 kg of uranium and thorium. These heavy metals were present as carbon-coated particles that were formed into compacts by addition and sinter carbonaceous materials. The heavy-metal loading in this reactor, ~1.4 Mg, was contained in 804 elements. The design burnup for the PB1 fuel was ~73,000 MWd/MTIHM; however, excessive fuel failures which occurred 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.

Most of the fuel from both cores is now located at INEL in forty-six 24-in.-diam baskets (Core 1) and forty-four 18-in.-diam baskets (Core 2). A small quantity (10 elements) is located at ORNL. Some fuel (28 elements) is unaccounted for (Morissette 1986).

4.3.1 Physical Description

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 PB 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 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 PB 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. Externally, the appearance of all configurations is the same.

Figure 4.3.1 shows the Core 1 standard fuel element. Its primary components are 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 the 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 was machined to engage with the fuel handling machines. A 0.25 in.-diam hole down the centerline of the reflector served as an inlet channel for purge gas. A porous plug cemented and retained within the upper reflector provided 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 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 for Core 1. The pyrolytic carbon-coated particles are between 210 and 595 μm in diameter, with coating thicknesses of 55 $\pm$ 10 μm for Core 1. The size distribution of the particles was selected such that the volume fraction of the coated particles does 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.

The Core 2 standard fuel elements are essentially the same as the Core 1 elements (see Appendix 4D). The only design difference is 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, low-density, pyrolytic carbon coating surrounded by an outer isotropic layer of pyrolytic carbon. 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 are smooth and have 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

The condition of Core 1 varies significantly from that of Core 2; therefore, each core is 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. Fast-neutron-induced dimensional changes 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 designed core lifetime of the fuel of 900 EFPD. The 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 specific elements loadings were provided to INEL by Philadelphia Electric (Conti 1971) or with shipping records.

4.3.4.2 Fuel Element Radioactivity

Table 4.3.6 summarizes the calculated major fission product and heavy-metal nuclide inventory as a function of decay time for what is considered maximum burnup fuel, assuming full exposure (ICP 1976). Data on actual fission product inventories for each fuel element, based on its exposure in the core, are not available for PBl. The basis for the calculations is uncertain since the reference is an undocumented personal communication. However, the ^{233}U content used as the basis of these calculations was compared with the amount calculated by dividing the Core 2 discharge quantity by the number of fuel elements (804); the fuel element amount was approximately 33% larger than its calculated fractional share (1/804 times the Core 2 discharge quantity). Additionally, the ^{85}Kr, ^{90}Sr, and ^{137}Cs contents of this hypothetical element were compared with those for LWR fuel irradiated to the same level. LWR fuel nuclide contents were obtained by using the values for a burnup of 60,000 MWd/MTIHM and ratioing them upward to account for a burnup of 72,959 MWd/MTIHM. The curie quantities of the above three nuclides was about a factor of 1.6 greater for the PB element values than were obtained using LWR values and assuming average power generation. This leads us to believe that the PB element values are probably for a maximum power generation element rather than for an average one.

4.3.4.3 Decay Heat

Figure 4.3.3 shows the calculated decay heat from 1-MTIHM PBI fuel having the isotopic content used in Table 4.3.6 for calculating radioactivity. As indicated previously, these rates are probably applicable to fuel in the most highly irradiated element rather than for an average element. We note that the calculated heat generation rate per metric ton of initial heavy metal for PBI spent fuel is greater than the value calculated for FSV spent fuel, even though, on the average, FSV fuel is more highly irradiated (100,000 versus 73,000 MWd/ton). This discrepancy must be due to the use of different bases for the two cases. We expect to rectify this anomaly in the future.

4.3.5 Spent Fuel Inventory

The PBI 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.

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 28 PBl 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 PB 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 4.3.4 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 PB fuel shipping cask. The elements were positioned in the cask with a 25.5-in.-diam basket assembly. At INEL, an entire basket loaded with canisters was lowered into a drywell. A loaded basket assembly weighs 3400 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 given in Appendix 4D (USAEC/PEC 1971).

4.3.6.2 Core 2

The Core 2 spent fuel was packaged for shipment using the same type of canister that was used for Core 1. However, the Core 2 fuel was placed in the Irradiated Fuel Storage Facility at INEL. This required removal of the fuel from the existing canister and cutting of 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. 4.3.5, contains 12 PB elements (ICP 1976).

4.3.7 Quantities to be Disposed

Table 4.3.7 summarizes the spent fuel quantities for the Peach Bottom I reactor.

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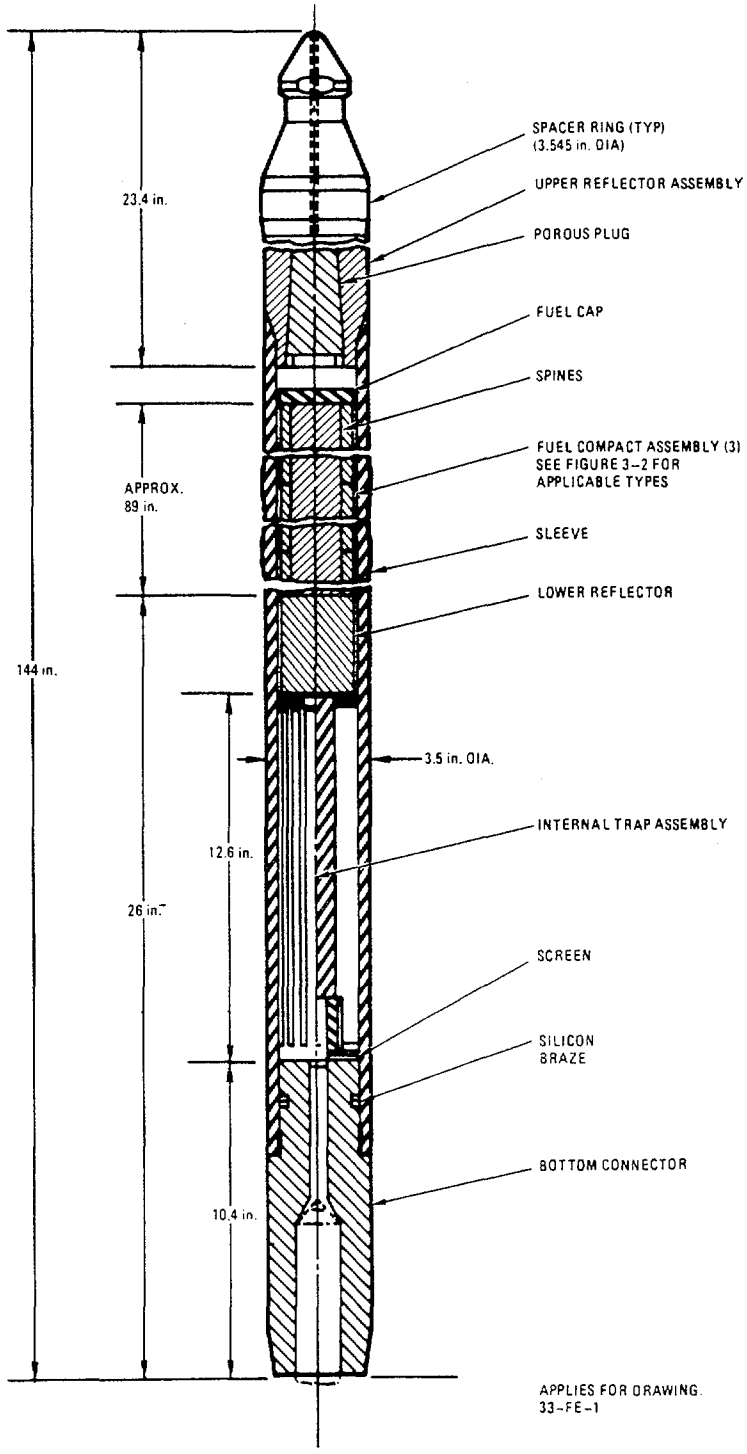


Fig. 4.3.1. Peach Bottom Unit 1, Core 1 fuel element

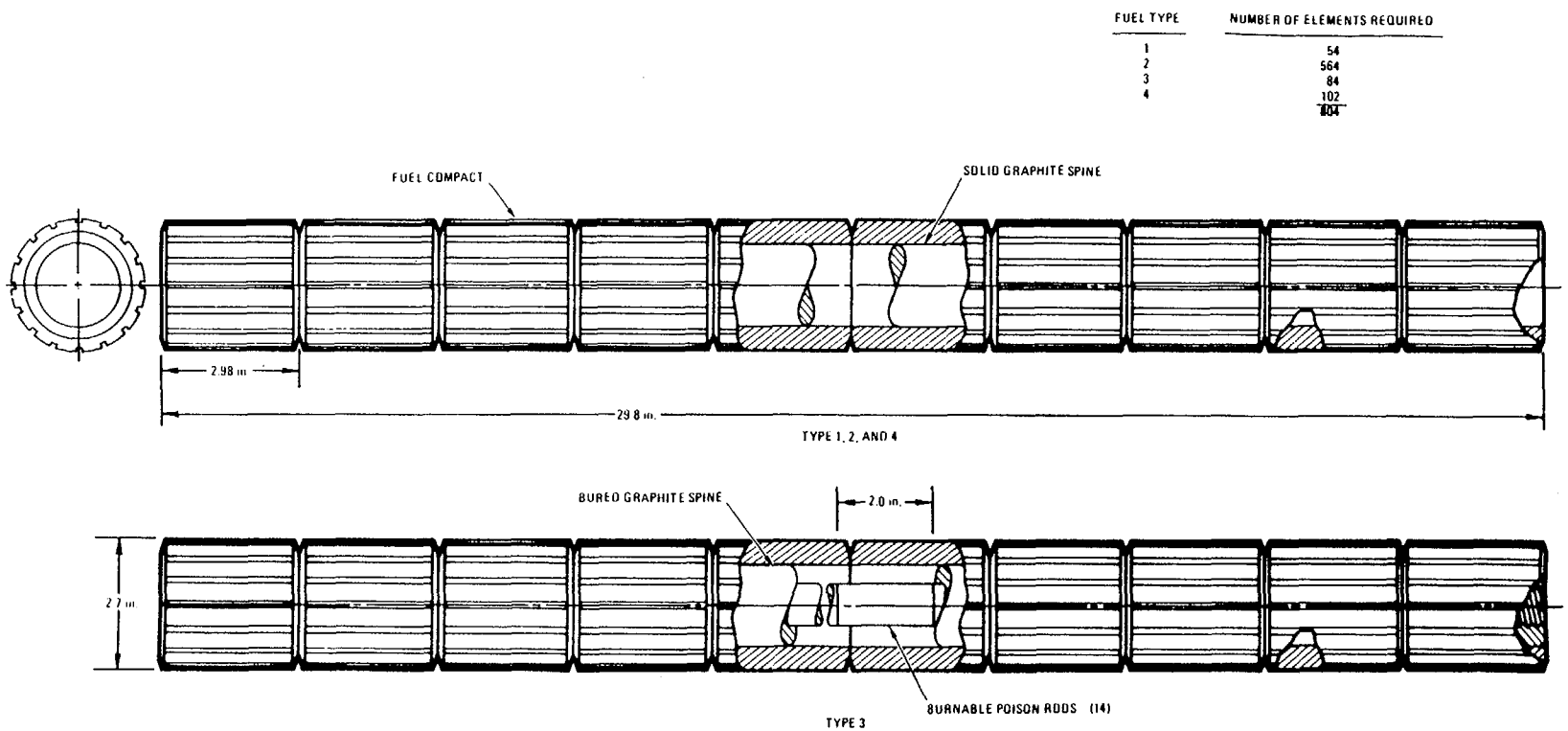


Fig. 4.3.2. Peach Bottom Unit 1, Core 1 fuel compacts

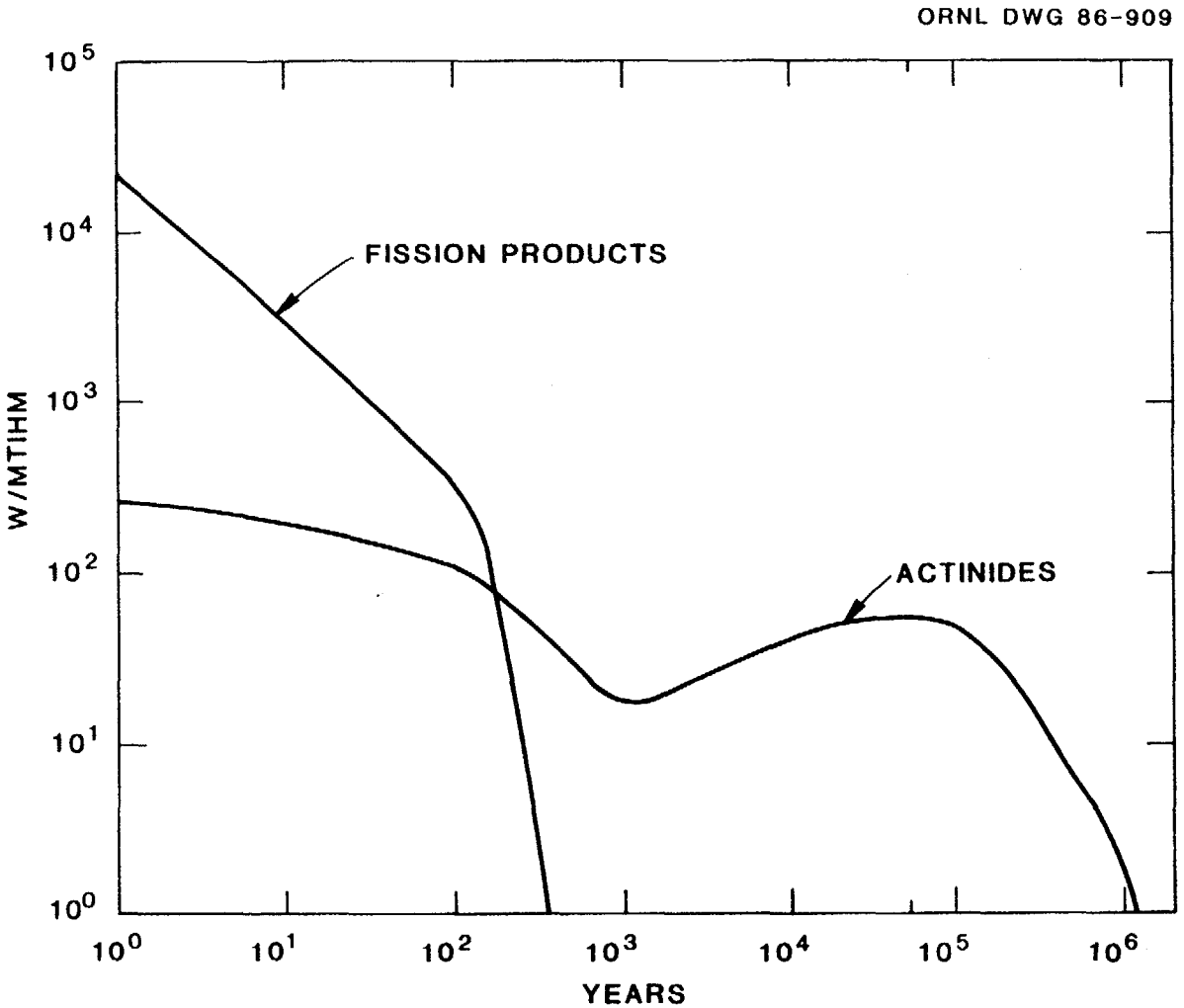
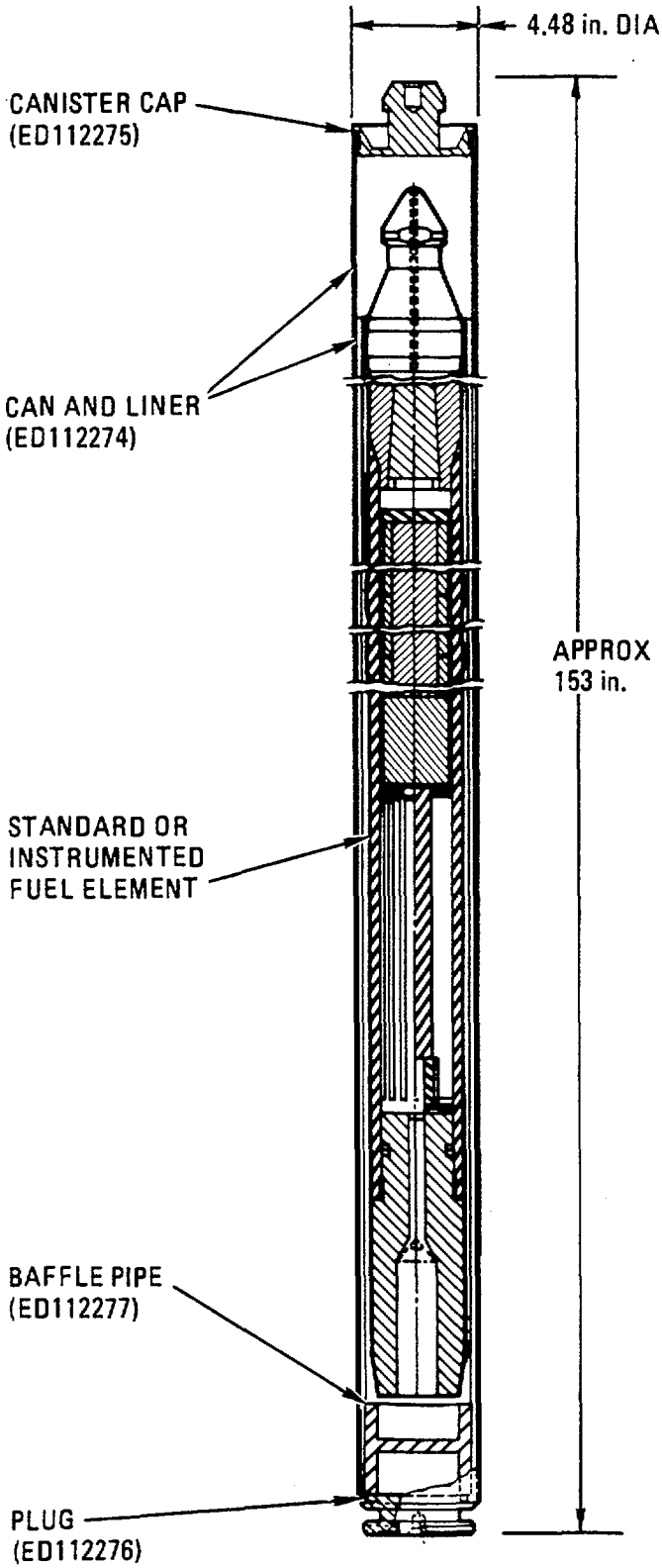


Fig. 4.3.3. Calculated decay heat of fully irradiated Peach Bottom Unit 1 fuel. Basis: 1 Mg initial heavy metal.



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

Fig. 4.3.4. PB1/1 non-failed fuel element in storage canister

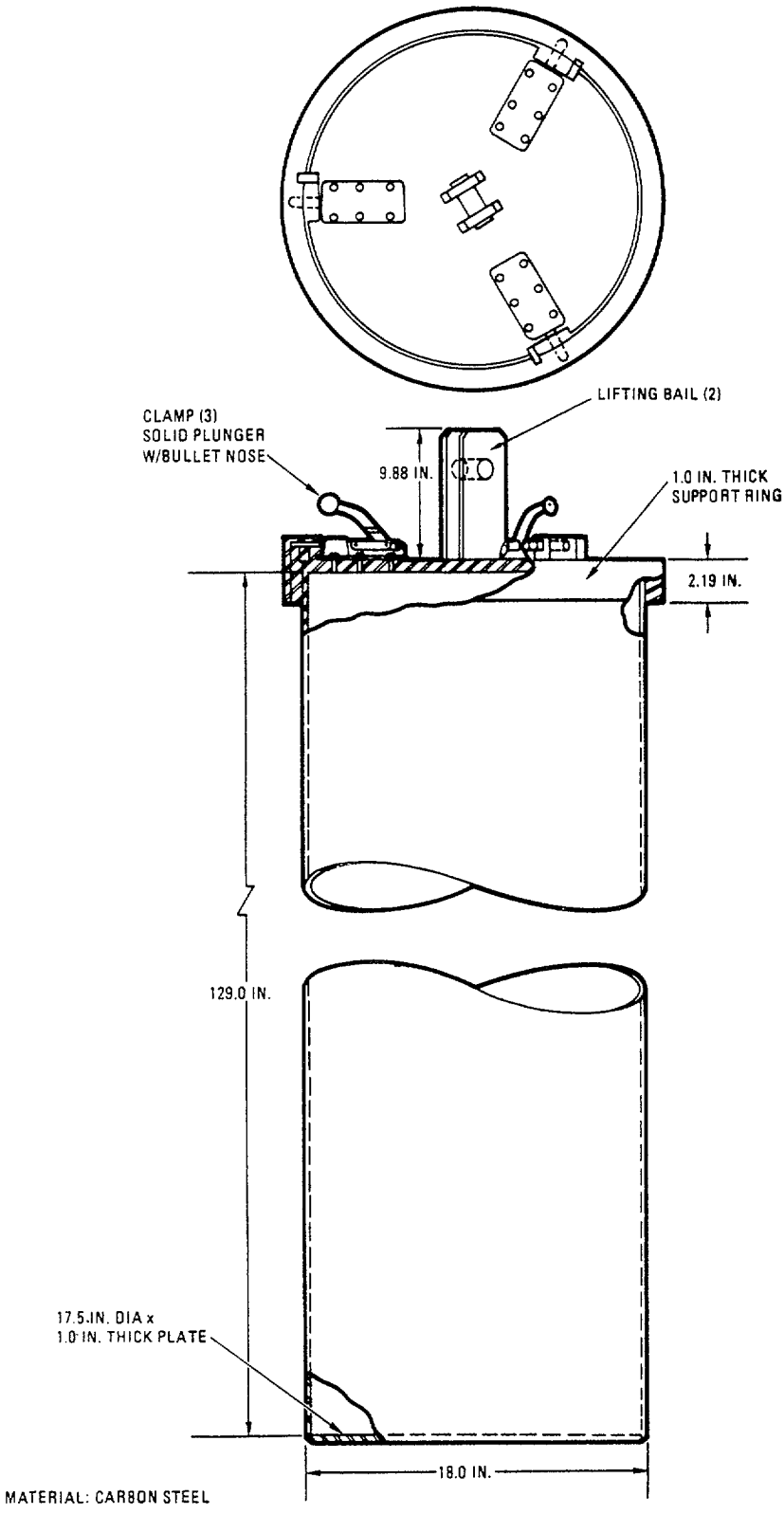


Fig. 4.3.5. PB1/2 storage canister

Table 4.3.1. Materials used in fuel element components

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-alumal Nb-1% Zr sheath
Test samples	Niobium canned, fission product release samples

Table 4.3.2. Weights of fuel elements, fuel element components, and fuel element storage apparatus

Item description	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	70
Storage basket with core 1 fuel	1550
Fuel Element Components	
Upper reflector	6
Sleeve	13
Lower reflector	0.6
Internal trap	2
Bottom connector	3
Fuel compact assembly (3 per element)	5/15
Fuel compact	0.4
Fuel Element Materials	
Carbon	39
Stainless steel	5
Uranium	(see Table 4.3.3)
Thorium	(see Table 4.3.3)
Rhodium	0-103
Boron	0-18
Silicon	15

Table 4.3.3. PB1 fuel element initial metal loadings (Refs. 24 and 25), grams

	Fuel element type 1		2		3		4	
	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	3468	2598
Rhodium 103	0	18.5	30.8	6.16	103	6.16	0	0
Boron	0		0		0	18.3	0	0

Table 4.3.4. Burnup data for cores 1 and 2

	Core 1	Core 2
EFPD	451.5	897.4
MW(t)-h[a]	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 MW-d/MTHM	72,717 MW-d/MTHM

[a]Reactor core output 115 MW(t).

Table 4.3.5. Postirradiation total core heavy metal loadings

Nuclide	Data	Totals	
		Core 2	Core 1
Th-232	kg	1,172.54	1,439.31
U-232	Milligrams	7,484.56	1,460
U-233	Grams	26,251.46	20,523.82
U-234	Grams	4,546.84	2,956.24
U-235	Grams	66,962.86	156,518.24
U-236	Grams	21,116.46	14,266.21
U-238	Grams	9,252.53	12,324.92
Pu-239	Milligrams	199,505.53	417,170
Pu-240	Milligrams	69,211.53	82,850
Pu-241	Milligrams	112,470.13	63,340
Pu-242	Milligrams	53,696.54	8,310
Np-237	Grams	1,624.52	
U	Grams	127,832.20	206,592.89
U-235	Weight fraction	0.5238	0.7576
U-233	Weight fraction	0.2030	.0994
U-232	ppm	58.55	7.08

TABLE 4.3.6. RADIOACTIVITY OF PEACH BOTTOM-1 REACTOR SPENT FUEL

BASED ON ONE MTIHM; BURNUP IS UNCERTAIN

CURIES

NUCLIDE	120.0D	1.0YR	10.0YR	100.0YR	1000.0YR	10.0KY	100.0KY	1.0MY
				ACTINIDES AND DAUGHTERS				
TL207	0.0	1.840E-08	3.487E-06	1.796E-04	2.555E-03	2.329E-02	1.078E-01	1.225E-01
TL208	0.0	2.480E-04	1.621E-02	3.258E-02	3.258E-02	3.258E-02	3.258E-02	3.258E-02
TL209	0.0	3.669E-04	5.301E-03	5.440E-02	5.221E-01	3.461E+00	3.932E+00	7.823E-02
PB209	0.0	1.699E-02	2.454E-01	2.518E+00	2.417E+01	1.602E+02	1.821E+02	3.622E+00
PB210	0.0	1.716E-10	4.891E-07	3.097E-04	5.157E-02	2.049E+00	1.582E+01	2.654E+00
PB211	0.0	1.845E-08	3.497E-06	1.801E-04	2.563E-03	2.336E-02	1.081E-01	1.229E-01
PB212	0.0	6.903E-04	4.511E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02
PB214	0.0	2.537E-08	5.266E-06	5.598E-04	5.158E-02	2.049E+00	1.583E+01	2.655E+00
BI210	0.0	1.716E-10	4.891E-07	3.097E-04	5.157E-02	2.049E+00	1.582E+01	2.654E+00
BI211	0.0	1.845E-08	3.497E-06	1.801E-04	2.563E-03	2.336E-02	1.081E-01	1.229E-01
BI212	0.0	6.903E-04	4.511E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02
BI213	0.0	1.699E-02	2.454E-01	2.519E+00	2.417E+01	1.602E+02	1.821E+02	3.622E+00
BI214	0.0	2.537E-08	5.266E-06	5.598E-04	5.158E-02	2.049E+00	1.583E+01	2.655E+00
PO210	0.0	4.080E-11	4.891E-07	3.097E-04	5.157E-02	2.049E+00	1.582E+01	2.654E+00
PO212	0.0	4.422E-04	2.890E-02	5.809E-02	5.810E-02	5.810E-02	5.810E-02	5.809E-02
PO213	0.0	1.662E-02	2.401E-01	2.464E+00	2.365E+01	1.568E+02	1.781E+02	3.543E+00
PO214	0.0	2.537E-08	5.265E-06	5.597E-04	5.157E-02	2.049E+00	1.582E+01	2.654E+00
PO215	0.0	1.845E-08	3.497E-06	1.801E-04	2.563E-03	2.336E-02	1.081E-01	1.229E-01
PO216	0.0	6.903E-04	4.511E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02
PO218	0.0	2.538E-08	5.267E-06	5.599E-04	5.159E-02	2.050E+00	1.583E+01	2.655E+00
AT217	0.0	1.699E-02	2.454E-01	2.519E+00	2.417E+01	1.602E+02	1.821E+02	3.622E+00
RN219	0.0	1.845E-08	3.497E-06	1.801E-04	2.563E-03	2.336E-02	1.081E-01	1.229E-01
RN220	0.0	6.903E-04	4.511E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02	9.067E-02
RN222	0.0	2.538E-08	5.267E-06	5.599E-04	5.159E-02	2.050E+00	1.583E+01	2.655E+00
FR221	0.0	1.699E-02	2.454E-01	2.519E+00	2.417E+01	1.602E+02	1.821E+02	3.622E+00
FR223	0.0	2.546E-10	4.823E-08	2.485E-06	3.536E-05	3.224E-04	1.491E-03	1.695E-03
RA223	0.0	1.845E-08	3.497E-06	1.801E-04	2.563E-03	2.336E-02	1.081E-01	1.229E-01
RA224	0.0	6.903E-04	4.511E-02	9.067E-02	9.068E-02	9.067E-02	9.067E-02	9.067E-02
RA225	0.0	1.699E-02	2.454E-01	2.519E+00	2.417E+01	1.602E+02	1.821E+02	3.622E+00
RA226	0.0	2.538E-08	5.267E-06	5.599E-04	5.159E-02	2.050E+00	1.583E+01	2.655E+00
RA228	0.0	6.085E-03	5.734E-02	9.067E-02	9.068E-02	9.068E-02	9.067E-02	9.067E-02
AC225	0.0	1.699E-02	2.454E-01	2.519E+00	2.417E+01	1.602E+02	1.821E+02	3.622E+00
AC227	0.0	1.845E-08	3.495E-06	1.801E-04	2.563E-03	2.336E-02	1.081E-01	1.229E-01
AC228	0.0	6.085E-03	5.734E-02	9.067E-02	9.068E-02	9.068E-02	9.067E-02	9.067E-02
TH227	0.0	1.820E-08	3.448E-06	1.776E-04	2.527E-03	2.304E-02	1.066E-01	1.212E-01
TH228	0.0	6.903E-04	4.510E-02	9.067E-02	9.068E-02	9.068E-02	9.067E-02	9.067E-02
TH229	0.0	1.699E-02	2.454E-01	2.518E+00	2.417E+01	1.602E+02	1.821E+02	3.622E+00
TH230	0.0	1.745E-04	2.520E-03	2.648E-02	2.750E-01	2.629E+00	1.569E+01	2.653E+00
TH231	0.0	1.224E-01	1.224E-01	1.224E-01	1.224E-01	1.225E-01	1.229E-01	1.229E-01
TH232	9.068E-02	9.068E-02	9.068E-02	9.068E-02	9.068E-02	9.068E-02	9.067E-02	9.067E-02
PA231	0.0	1.739E-06	2.507E-05	2.579E-04	2.562E-03	2.335E-02	1.080E-01	1.229E-01
PA233	1.383E+07	2.570E+04	2.651E-04	1.014E-02	6.766E-02	8.521E-02	8.276E-02	6.183E-02

TABLE 4.3.6 (CONTINUED)

CURIES

NUCLIDE	120.0D	1.0YR	10.0YR	100.0YR	1000.0YR	10.0KY	100.0KY	1.0MY
				ACTINIDES AND DAUGHTERS				
U233	2.624E+02	2.688E+02	2.689E+02	2.687E+02	2.677E+02	2.574E+02	1.737E+02	3.457E+00
U234	2.887E+01	2.888E+01	2.902E+01	3.003E+01	3.092E+01	3.014E+01	2.336E+01	1.821E+00
U235	1.224E-01	1.224E-01	1.224E-01	1.224E-01	1.224E-01	1.225E-01	1.229E-01	1.229E-01
U236	0.0	2.844E-07	4.094E-06	4.199E-05	4.017E-04	2.610E-03	3.984E-03	3.879E-03
U237	0.0	3.017E-01	1.956E-01	2.569E-03	3.923E-22	0.0	0.0	0.0
NP237	0.0	1.535E-06	2.651E-04	1.014E-02	6.766E-02	8.521E-02	8.276E-02	6.183E-02
PU238	5.974E+03	5.943E+03	5.535E+03	2.719E+03	2.221E+00	2.950E-31	0.0	0.0
PU239	1.677E+01	1.677E+01	1.676E+01	1.672E+01	1.629E+01	1.257E+01	9.407E-01	5.186E-12
PU240	1.431E+01	1.431E+01	1.429E+01	1.416E+01	1.287E+01	4.955E+00	3.563E-04	0.0
PU241	1.270E+04	1.230E+04	7.973E+03	1.047E+02	1.599E-17	0.0	0.0	0.0
AM241	0.0	1.345E+01	1.562E+02	3.694E+02	8.809E+01	4.751E-05	0.0	0.0
SUBTOT	1.385E+07	4.428E+04	1.400E+04	3.544E+03	6.135E+02	1.609E+03	1.815E+03	6.318E+01
				FISSION PRODUCTS				
KR 85	3.110E+04	2.978E+04	1.664E+04	4.942E+01	2.696E-24	0.0	0.0	0.0
SR 89	7.286E+05	2.515E+04	6.373E-16	0.0	0.0	0.0	0.0	0.0
SR 90	2.434E+05	2.396E+05	1.934E+05	2.270E+04	1.129E-05	0.0	0.0	0.0
Y 90	2.446E+05	2.396E+05	1.934E+05	2.271E+04	1.129E-05	0.0	0.0	0.0
Y 91	1.079E+06	5.902E+04	7.202E-13	0.0	0.0	0.0	0.0	0.0
ZR 95	1.308E+06	9.174E+04	3.135E-11	0.0	0.0	0.0	0.0	0.0
NB 95	2.503E+06	2.004E+05	6.960E-11	0.0	0.0	0.0	0.0	0.0
NB 95M	0.0	6.806E+02	2.326E-13	0.0	0.0	0.0	0.0	0.0
RU103	2.997E+05	3.954E+03	2.554E-22	0.0	0.0	0.0	0.0	0.0
RH103M	2.982E+05	3.566E+03	2.302E-22	0.0	0.0	0.0	0.0	0.0
RU106	2.489E+05	1.568E+05	3.219E+02	4.381E-25	0.0	0.0	0.0	0.0
RH106	2.490E+05	1.568E+05	3.219E+02	4.381E-25	0.0	0.0	0.0	0.0
TE127	1.377E+04	2.820E+03	2.356E-06	0.0	0.0	0.0	0.0	0.0
TE127M	1.369E+04	2.879E+03	2.405E-06	0.0	0.0	0.0	0.0	0.0
TE129	1.941E+04	7.929E+01	2.803E-28	0.0	0.0	0.0	0.0	0.0
TE129M	1.918E+04	1.218E+02	4.306E-28	0.0	0.0	0.0	0.0	0.0
I129	0.0	1.119E-04	1.126E-04	1.126E-04	1.126E-04	1.125E-04	1.121E-04	1.077E-04
CS137	3.754E+05	3.696E+05	3.002E+05	3.752E+04	3.492E-05	0.0	0.0	0.0
BA137M	0.0	3.496E+05	2.840E+05	3.550E+04	3.304E-05	0.0	0.0	0.0
BA140	1.132E+03	1.911E-03	0.0	0.0	0.0	0.0	0.0	0.0
LA140	1.245E+03	2.199E-03	0.0	0.0	0.0	0.0	0.0	0.0
CE141	3.951E+05	2.120E+03	7.749E-28	0.0	0.0	0.0	0.0	0.0
PR143	1.184E+04	4.273E-02	0.0	0.0	0.0	0.0	0.0	0.0
CE144	3.151E+06	1.733E+06	5.724E+02	8.828E-33	0.0	0.0	0.0	0.0
PR144	3.149E+06	1.733E+06	5.724E+02	8.829E-33	0.0	0.0	0.0	0.0
PR144M	0.0	2.079E+04	6.869E+00	1.059E-34	0.0	0.0	0.0	0.0
PM147	9.318E+05	7.803E+05	7.237E+04	3.418E-06	0.0	0.0	0.0	0.0
SM147	0.0	3.714E-06	2.107E-05	2.284E-05	2.284E-05	2.284E-05	2.284E-05	2.284E-05
SM151	8.357E+03	8.314E+03	7.757E+03	3.878E+03	3.785E+00	2.973E-30	0.0	0.0
SUBTOT	1.532E+07	6.210E+06	1.070E+06	1.224E+05	3.786E+00	1.354E-04	1.349E-04	1.306E-04

NUCLIDES CONTRIBUTING < 0.001 % ARE OMITTED

Table 4.3.7. Peach Bottom I Reactor. Summary of spent fuel quantities currently in inventory.

Source	Number of fuel elements	MTIHM[a]
Core 1	813	2.0
Core 2	804	1.6
Total	1617	3.6

[a]Based on quantities shown in Table 4.3.5.

4.4 RESEARCH AND TEST REACTOR FUELS

4.4.1 Introduction and Summary

The research and test reactors considered in this section are divided into the following four categories: (1) reactors used for educational purposes and research at universities and other educational institutions, (2) reactors owned by commercial and industrial firms and used for private research and test purposes, (3) reactors owned and operated by U.S. government agencies other than DOE, and (4) reactors owned and operated by DOE. The number of reactors in each of these categories is listed in Table 4.4.1; there are 50 educational, 11 commercial research, 6 government-owned non-DOE, and 51 DOE reactors, giving a total of 118. This section presents estimates of the contributions of these reactors to the total spent fuel load of the repository and of the forms and characteristics of these fuels when received at the repository.

The research and test reactors use a wide variety of different types of fuel, which require different handling, treatment, and disposal procedures. Some of the spent fuels will be reprocessed at goverment-owned plants such as SRP or INEL; if so, the resulting wastes become part of the defense high-level waste, and their immobilization and disposal have already been covered in Sect. 3 of this report. Some of the fuels, however, are not planned to be reprocessed, and for others the decision to reprocess or not reprocess has not been made. Those fuels which are not reprocessed may eventually require disposal in a repository as canistered spent fuel. Thus, the task of this section is to catalog the various fuel types used in research and test reactors; to determine which of these fuel types might require repository disposal; to characterize these fuels with regard to quantity, dimensions, activity level, and other characteristics that might affect their disposal.

4.4.1.1 Fuel Type Categories

Since the disposal requirements are expected to be sensitive to the characteristics of the fuel, it is logical to categorize the fuels into different types based on their physical and chemical properties and to discuss each type separately. Each of the following subsections describes a fuel type and gives data on physical/mechanical characteristics, the quantity of fuel requiring disposal, possible treatment and disposal options, and the approximate disposal volume required if repository disposal is used. The fuels are divided into the following eight categories:

1. MTR plate-type, highly enriched, U-Al alloy fuels;
2. TRIGA (uranium-zirconium hydride fuels), stainless steel or alumium clad, 19 to 20% enriched;
3. homogeneous UO_2-polyethylene disks or blocks, 19 to 20% enriched;
4. PULSTAR and other low-enriched (4 to 6%) UO_2-pin array types
5. liquid fuels;
6. uranium-molybdenum alloy highly enriched fuels;
7. FFTF UO_2-PuO_2 fuel; and
8. miscellaneous types.

4.4.1.2 Assumptions Regarding Disposal

For the purposes of this report, it was assumed that fuel in categories 1, 5, 6, 7, and 8 will be reprocessed at SRP, HANF, or INEL and will become part of defense HLW; and that fuel in categories 2, 3, and 4 will be packaged for emplacement in a repository. No attempt was made to design packages or to estimate repository volume requirements.

4.4.1.3 Summary of Data Presented

The following tables summarize data presented in this section: Table 4.4.2 gives the number of university/educational reactors, commercial reseach/test reactors, government-owned DOE reactors, and government-owned non-DOE reactors in each of the eight fuel type categories, for a total of 118. Tables 4.4.3 through 4.4.7 list the

university/educational reactors in each fuel category and provide information on fuel characteristics and dimensions, quantity of fuel in the reactor, and frequency of refueling. Table 4.4.8 gives similar information for private commercial research and test reactors, while Tables 4.4.9 and 4.4.10 give similar information for government-owned DOE and non-DOE research and test reactors. Table 4.4.11 summarizes the feasibility of reprocessing the various types of fuels at SRP or INEL, and Table 4.4.12 summarizes the total quantities of spent fuels requiring disposal or reprocessing.

Radioactivity (curies) and thermal power (watts) per fuel element were estimated for those fuel categories that were assumed to be packaged. Decay times ranging from 0 to 10^6 years after placement in packages were used. These estimates are shown in Tables 4.4.13 through 4.4.15.

Table 4.4.16 gives physical data on TRIGA fuel.

A summary of projected FFTF fuel discharges is given in Table 4.4.17.

It should be noted that some of the research and test reactor fuels discussed in this section have already been shipped to various DOE sites and are listed as miscellaneous spent fuels at these sites in Sect. 4.5 of this report. Thus, in estimating final disposal requirements, care must be taken not to count such fuels twice.

4.4.1.4 Appendix on Educational Reactors

Reactors at universities and other educational institutions in the United States require special mention because of their fuel supply and disposal arrangements. Under DOE's university assistance program, DOE supplies the fuel for university and educational reactors and retains ownership of the fuel; thus, DOE is responsible for disposal of the fuel when it is removed from a reactor. Appendix 4A lists the university and other educational reactors, provides data on the fuel at each reactor, and describes the fuel supply and refueling schedule for each.

4.4.2 MTR Plate-Type Fuels

4.4.2.1 Physical Characteristics and Dimensions

Fuels in the first category are highly enriched (90 to 93%) plate-type fuels (MTR, Argonaut, etc.), most of which consist of uranium-aluminum alloy fuel with aluminum cladding. Typical fuel element dimensions (cm) are about 7.6 x 7.6 x 90; the length may vary from 66 to 220 cm. An element (see Fig. 4.4.1) consists of a number of individual plates and typically contains about 0.15 to 0.18 kg of ^{235}U. The Westinghouse training reactor at Zion, Illinois, has a different type of fuel element (cylindrical) but is included in this category because it is 93% enriched uranium-aluminum alloy fuel with aluminum cladding. The reactor contains 24 elements, each of which consists 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.

4.4.2.2 Quantity of Fuel Requiring Disposal

This type of fuel (Table 4.4.3) is being used by 20 university/educational reactors, one commercial research reactor (the Westinghouse reactor listed in Table 4.4.8), and 15 DOE-owned reactors (Table 4.4.10). The in-core fuel in the 20 university reactors amounts to about 66 kg of ^{235}U; the Westinghouse reactor contains about 4 kg, and the 15 DOE reactors contain about 360 kg of ^{235}U. Thus, the total quantity of ^{235}U in the cores of all these reactors is about 430 kg. The additional quantity required for the annual refueling of these reactors for 30 years is about 2000 to 4000 kg of ^{235}U.

4.4.2.3 Treatment and Disposal Options

Fuels of this type have been reprocessed on a regular basis at INEL and SRP in the past. It is expected that future shipments will continue to be handled in the same way. Thus, 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.

Over a 30-year period, 5000 kg (total uranium) of this type of fuel represents about 0.2 m^3 of HLW, based on 17 kg of waste per metric ton of uranium processed (White 1986) and assuming 1650 kg of glass containing 25% waste solids per m^3. Even if the amount of waste per metric ton of uranium processed is increased by a factor of 5 to account for the high enrichment of the MTR fuel, the quantity of glass would still only require about 1 m^3. This indicates that the quantity of high-level waste generated by research and test reactor fuels that can be reprocessed is so small compared with defense HLW that the quantity projections for the latter would require no adjustment.

4.4.3 TRIGA Fuels

4.4.3.1 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 is about 1.6. The most prevalent ^{235}U enrichment is 20%, although some 70% enriched fuels are still in use. Stainless steel is now the standard cladding material. Both the aluminum-clad and the stainless-steel-clad elements are 1.5-in.-diam by 30-in.-long rods, including the end fittings. The elements clad with Incoloy 800 are of the same length but have a smaller diameter (0.54 in.). Fuel follower control rod elements are 1.5-in.-diam rods and range from 45.0 to 66.5 in. long. Physical configuration and dimensions of aluminum-clad and stainless-steel-clad elements are shown in Figs. 4.4.2 and 4.4.3 and Table 4.4.16 (Tomsio 1986).

4.4.3.2 Quantity of Fuel Requiring Disposal

The United States has 31 TRIGA reactors, of which 27 are still in operation. The number of fuel elements in a reactor varies from 60 to 100, depending on reactor size. At present, there are about 3000 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. Refueling for 30 years will require an additional number of elements, estimated to be about 600. Thus, the total number of fuel elements requiring disposal will be about 4500.

4.4.3.3 Treatment and Disposal Options

Defense reprocessing facilities at INEL and SRP are not well suited to hydride fuels since the presence of hydride is likely to cause difficulties in the various treatment steps. Thus, the feasibility of reprocessing these fuels at SRP or INEL remains to be evaluated. Hanford has no plans for reprocessing such fuels (White 1986).

4.4.3.4 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 4.4.13 for decay times ranging from 10 days to 10^6 years. At a decay time of (for example) 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.

4.4.4 Homogeneous UO_2-Polyethylene Fuels

4.4.4.1 Physical Characteristics and Dimensions

Fuel in the third category consists of homogeneous UO_2-polyethylene material shaped into disks (six reactors) or blocks (one reactor). Each of the six 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 x 7.78 x 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.

4.4.4.2 Quantity of Fuel Requiring Disposal

The seven reactors in this category 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.

4.4.4.3 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 would require development 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. If the fuel is reprocessed at one of the defense reprocessing sites, the amount of material added to defense HLW would be negligible; the additional volume of immobilized HLW would be only a few liters.

4.4.4.4 Radioactivity and Thermal Power

Radioactivity and thermal power per kilogram of discharged UO_2-polyethylene fuel were estimated, 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 4.4.14, 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. (see Sect. 4.4.4.2). Thus, the total radioactivity at discharge would be about 28 x 1.5E+3, or 42,000 Ci, and the total thermal power would be about 28 x 6.5E+0, or 183 W.

4.4.5 PULSTAR and Other Low-Enriched Pin-Type Fuels

4.4.5.1 Physical Characteristics and Dimensions

As shown in Table 4.4.2, six reactors are using fuel in this category: three educational, two private research, and one government-owned. The educational reactors are listed in Table 4.4.6. 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 x 5 square array. The pins have a diameter of about 1.2 cm and a length of about 66 cm. Each 5 x 5 array is enclosed in a Zircaloy-2 box with outside dimensions of 7.0 x 8.0 x 82 cm. The overall length of the fuel element, including end fittings, is 96.5 cm (Orlosky 1985, 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. Fuel enrichment is 2.1% (Aderhold 1986).

The two private research reactors using low-enriched pin-type fuel assemblies are the Babcock & Wilcox reactors shown in Table 4.4.8. Both of these reactors have been shut down, and their fuel has been shipped to SRP. Enrichment of these fuels was 2.5 and 4.0%, and the cladding material was aluminum and stainless steel, respectively.

The DOE-owned reactor is the Loss of Fluid Test (LOFT) Reactor at INEL, which uses 4% enriched UO_2 pellets in Zircaloy-4 pins. Fuel from the LOFT reactor has been reprocessed at INEL, and it is assumed that this practice will continue.

4.4.5.2 Quantity of Fuel Requiring Disposal

The two educational PULSTAR 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 1670 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 x 5 elements (80 kg ^{235}U, or about 1600 kg of uranium) and 825 Cornell ZPR fuel pins (35.4 kg of ^{235}U, or about 1700 kg of uranium).

4.4.5.3 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 seems reasonable for purposes of this report to assume that the disposal approach 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.

4.4.5.4 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 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 4000 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 4.4.15 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 x 5 fuel elements 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.

4.4.6 Aqueous Liquid Fuels

4.4.6.1 Physical Characteristics and Dimensions

The fifth category consists of aqueous liquid fuels. Both of the educational reactors in this category use a solution 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 third reactor in this category is the Solution High Energy Burst Assembly (SHEBA) reactor at Los Alamos, which uses about 80 L of a 4.95% enriched uranyl fluoride solution, as indicated in Table 4.4.9.

4.4.6.2 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.

4.4.6.3 Treatment and Disposal Options

Sulfate is not desirable in defense waste processing because of its adverse effect on the properties of borosilicate glass (Berreth 1986). However, in view of the very small volume involved, it would seem feasible to blend these liquids into defense HLW reprocessing tanks; or, if necessary, the sulfate could be replaced via ion exchange. This fuel would then be vitrified along with defense HLW. It is assumed here that this option is the one that will be used. The volumetric addition of these liquid fuels to defense HLW is insignificant; the additional HLW volume is approximately 10 liters.

4.4.7 Uranium-Molybdemum Alloy Fuels

Six government-owned reactors use fuels consisting of highly enriched (93.2%) uranium-molybdemum alloy clad with nickel or alumimum.

4.4.7.1 Physical Characteristics and Dimensions

Data on dimensions and physical characteristics of these fuels have not been obtained.

4.4.7.2 Quantity of Fuel Requiring Disposal

No data on quantity have been obtained.

4.4.7.3 Treatment and Disposal Options

The reprocessing of these fuels is currently handled by DOE. It is assumed that this practice will continue and that the resulting waste will be included with DOE high-level waste.

4.4.8 UO_2 - PuO_2 Fast Reactor Fuel

4.4.8.1 Physical Characteristics and Dimensions

The Fast Flux Test Facility (FFTF) at Hanford is a sodium-cooled 400 MW(t) sodium-cooled fast reactor fueled with mixed oxide UO_2 - PuO_2 pellets contained in 0.23-in diameter stainless-steel-clad fuel

pins. The pins are assembled into hexagonal assemblies, 12 ft in length, each containing 217 pins. Physical dimensions of the assemblies are shown in Fig. 4.4.4. Average Pu content of the core fuel is about 25-27%. Additional data on the fuel characteristics are given in Table 4.4.17.

4.4.8.2 Quantity of Fuel Requiring Disposal

The quantity of fuel discharged, and that projected for future years, is summarized in Table 4.4.18. Some of this fuel has been packaged into 41-in. long cylinders (see Table 4.5.6). Some has already been reprocessed and more probably will be.

4.4.9 Quantities to be Disposed

The estimated quantities of fuels discussed in the preceding subsections are summarized in Table 4.4.12. The MTR-type highly enriched uranium-aluminum alloy fuels and aqueous liquid fuels are assumed to be reprocessed, and their addition to defense HLW is negligible (less than 1 m^3).

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White 1985. J. D. White, DOE, Richland, letter to M. G. O'Rear, DOE, Savannah River, dated March 29, 1985.

Wilde 1986a. Letter from R. T. Wilde, Rockwell Hanford Operations, to M. W. Shupe, DOE/RLO, dated Apr. 18, 1986.

Wilde 1986b. Letter from R. T. Wilde, Rockwell Hanford Operations, to H. W. Godbee, ORNL, dated May 5, 1986.

ORNL DWG 87-1148

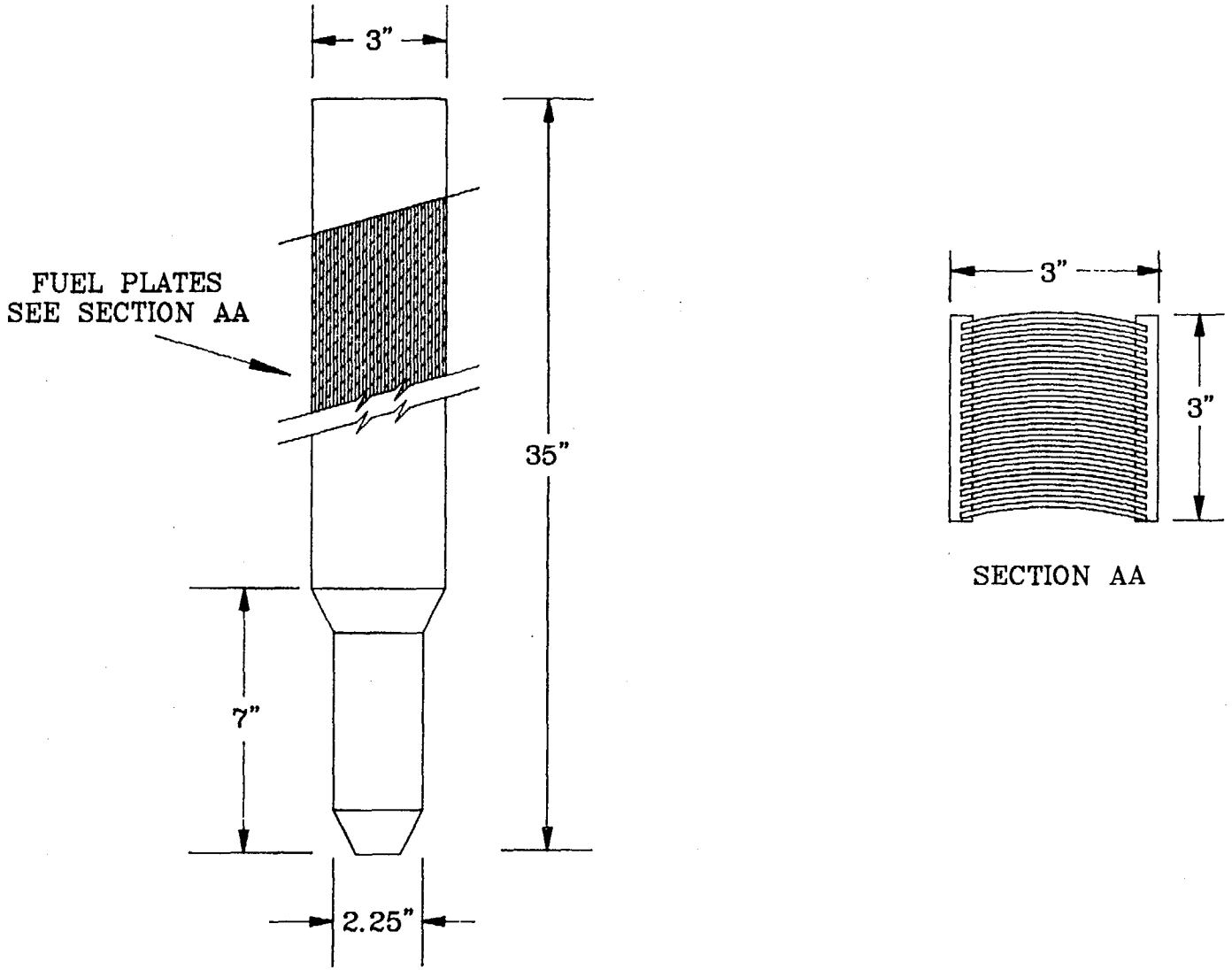
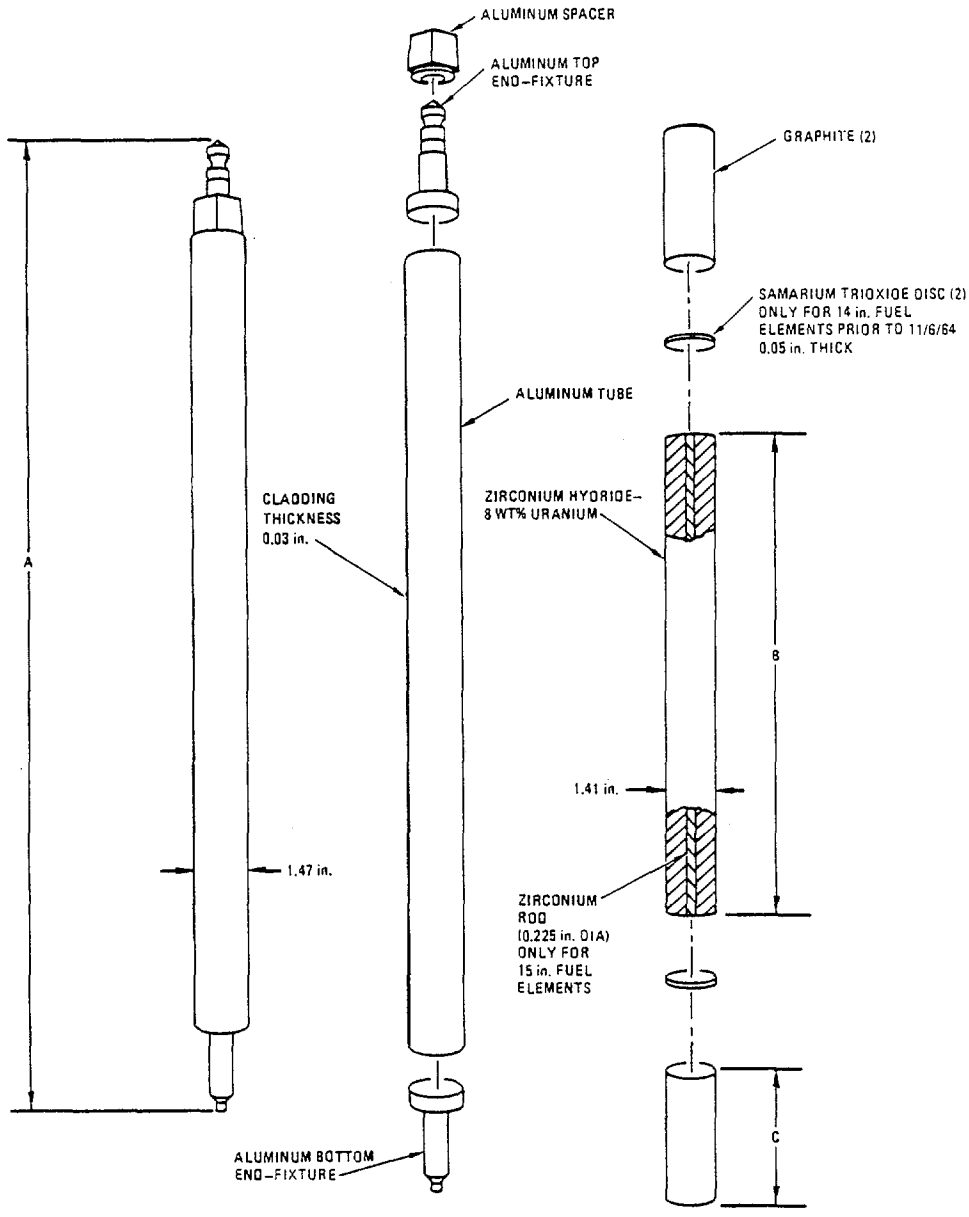


Fig. 4.4.1. MTR-plate type fuel assembly.



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. 4.4.2. TRIGA aluminum-clad fuel element.
Source: Tomsio 1986.

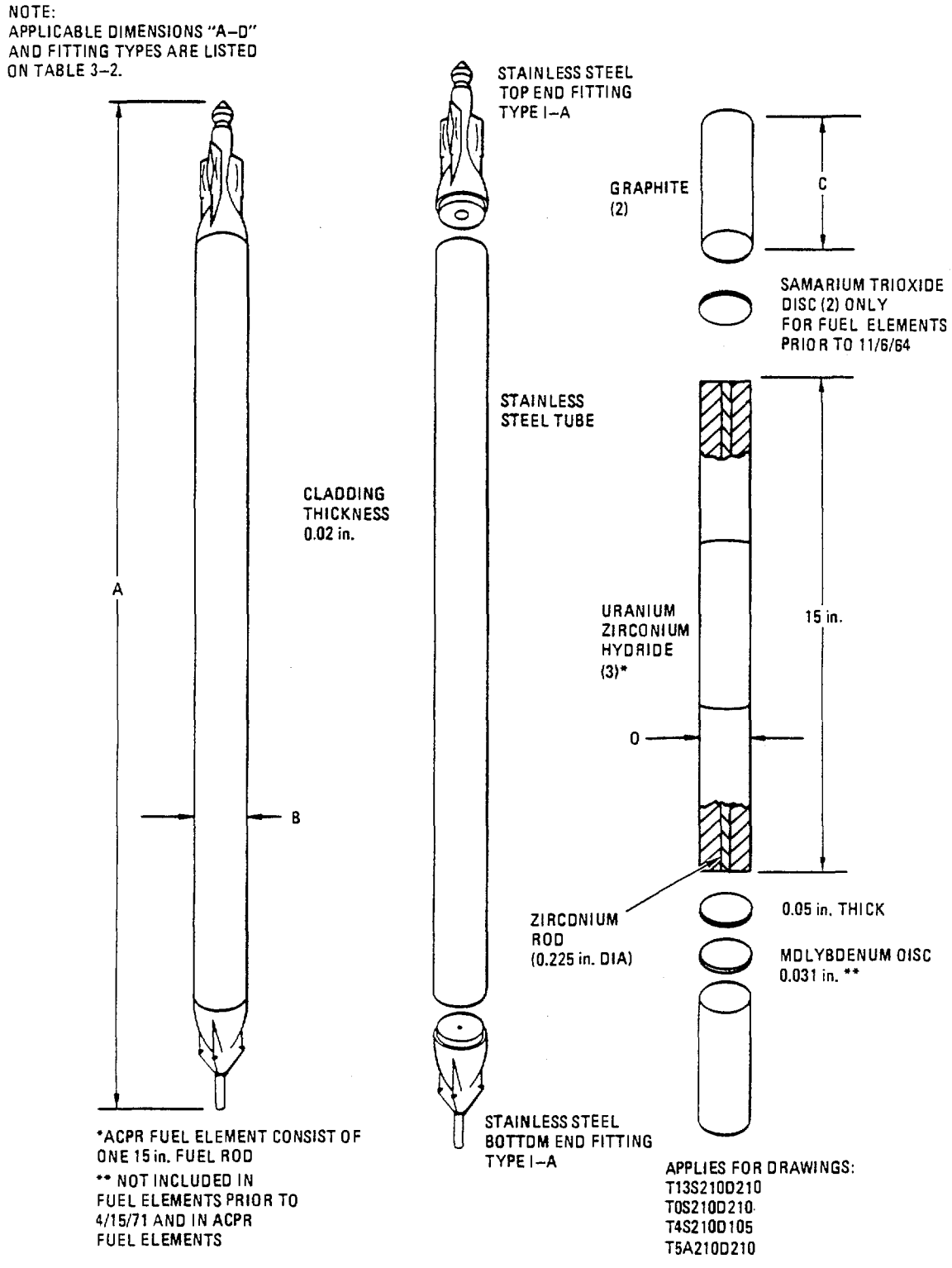


Fig. 4.4.3. TRIGA stainless-steel-clad fuel element. Source: Tomsio 1986.

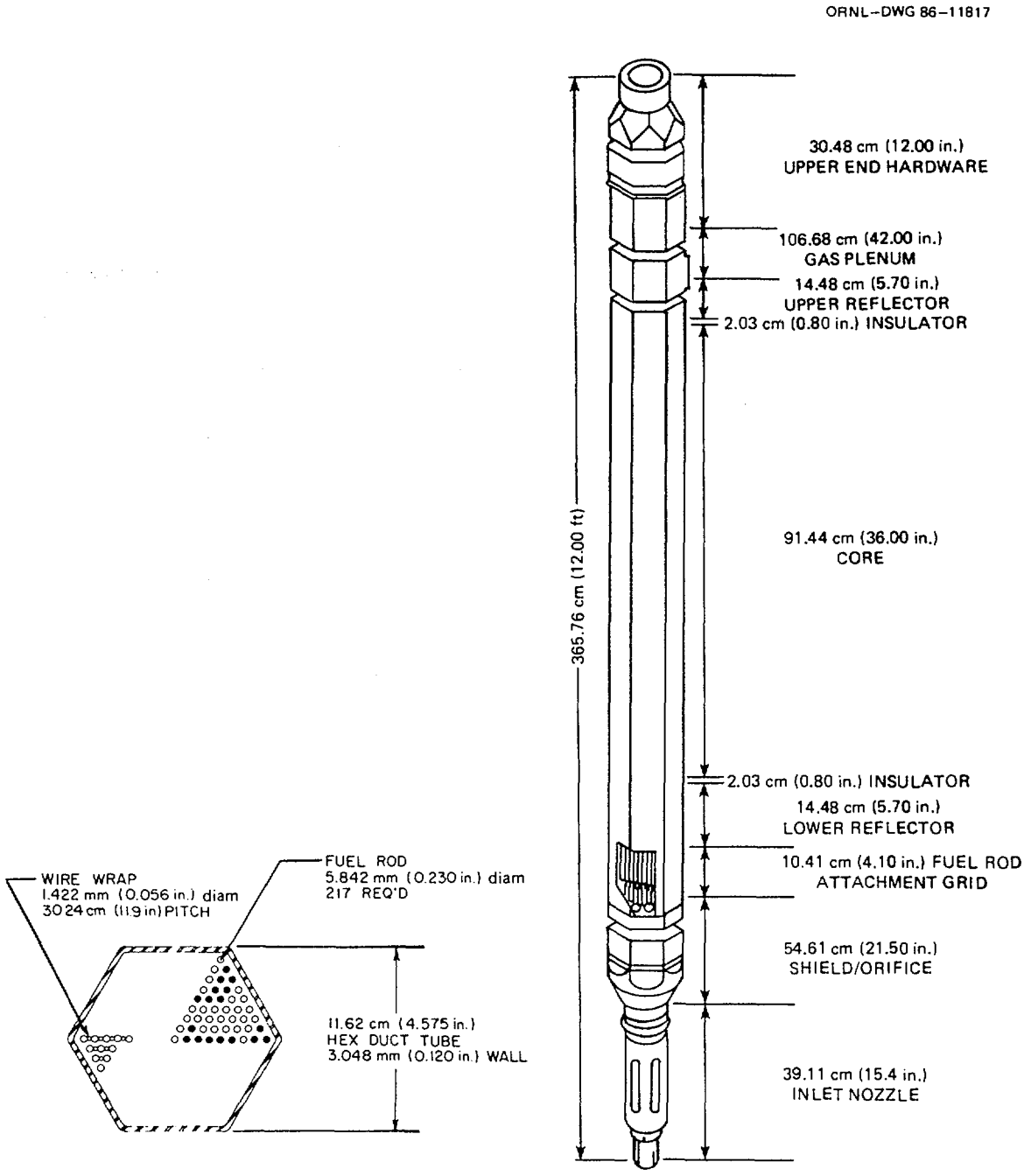


Fig. 4.4.4. Fuel assembly used in Fast Flux Test Facility. Source: Sears and Welch, 1986.

Table 4.4.1. Research and test reactors by user category

Reactor category	Number of reactors
Educational	50
Commercial research	11
Government-owned non-DOE	6
Government-owned DOE	52
Total	119

Table 4.4.2. Number of research and test reactors in each fuel type category

	Fuel type	University/ educational	Private research and test	Government owned (DOE)	Government owned (non DOE)
1.	MTR plate type, U-Al alloy, high enrichment	20	4	16	1
2.	TRIGA (U-ZrH_2 fuel)	18	5	2	3
3.	UO_2-polyethylene discs or blocks	7	0	0	0
4.	PULSTAR and other low-enriched pin type	3	2	1	0
5.	Liquid fuels (aqueous solutions)	2	0	1	0
6.	U-Mo alloy, high-enriched (93.2%)	0	0	4	2
7.	FFTF (UO_2-PuO_2 pellets)	0	0	1	0
8.	Miscellaneous	0	0	27	—
		50	11	52	6

Table 4.4.3. Educational reactors with highly enriched U-Al plate-type fuel assemblies

Ref. no.[a]	Location	Power level	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total ^{235}U in reactor (kg)
6	Georgia Institute of Technology	5 MW	U-Al	93	Al	17	3.01
8	Iowa State University	10 kW	U-Al	90	Al	12	3.19
10	Manhattan College	0.1 W	U-Al	92	Al	16	3.02
11	Massachusetts Institute of Technology	4.9 MW	U-Al	93	Al	24	12.2
15	Ohio State University	10 kW	U-Al	93	Al	20	3.18
18	Purdue University	1 kW	U-Al	93	Al		
20	Rensselaer Polytechnic Institute	Critical assembly	UO_2 in SS	93	SS		
27	University of California (LA)	100 kW	U-Al	93	Al	24	3.56
29	University of Florida	100 kW	U-Al	93	Al	24	3.35
32	University of Kansas	10 kW	U-Al	90	Al	16	2.50
33	University of Lowell	1 MW	U-Al	93	Al	26	3.50
35	University of Michigan	2 MW	U-Al	93	Al	35	6.35
36	University of Missouri	10 MW	U-Al	93.15	Al	8	6.2
37	University of Missouri, Rolla	200 kW	U-Al	89-93	Al	28	2.85
43	University of Virginia	100 W	U-Al	93	Al		
44	University of Virginia	2 MW	U-Al	93	Al	20	3.3
45	University of Washington	100 kW	U-Al	93	Al	24	3.43
47	Virginia Polytechnic Institute	100 kW	U-Al	90	Al	12	3.19
49	Worcester Polytechnic Institute	10 kW	U-Al	93	Al	24	3.26
50	Rhode Island Nuclear Science Center	2 MW	U-Al	93	Al	35	3.47

[a]Reference numbers refer to Appendix 4A, Table 4A.2.

Table 4.4.3 (continued)

Ref. no.[a]	Location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup (%)	Full power hours/year
6	Georgia Institute of Technology	7.52 x 7.04 x 219.4	Yes; to SRP	2 elements per 90 MWd	30	
8	Iowa State University	7.62 x 15.24 x 66.0	Yes; to ORNL			
10	Manhattan College	8.89 diam x 93.98	No			
11	Massachusetts Institute of Technology	6.11 x 66.7	Yes; to SRP, INEL	1 element per 17 operating days	40	
15	Ohio State University	7.62 x 7.62 x 88.9	No			150-200
18	Purdue University	7.52 x 7.52 x 8.19		0.02 g U235 in 24 yrs. of operation		
20	Rensselaer Polytechnic Institute	7.54 x 7.54 x 68.6				
27	University of California (LA)	6.03 x 7.23 x 68.6	Yes; to INEL			
29	University of Florida	7.23 x 5.44 x 65.1	Yes; to SRP	Element lifetime >10 years		
32	University of Kansas	7.62 x 7.62 x 86.3				
33	University of Lowell	7.62 x 7.62 x 101.6	No	To be refueled with 20% enriched fuel	15	500
35	University of Michigan	7.47 x 8.26 x 87.4	Yes; to SRP	1 element per 17 days	20	
36	University of Missouri	7.04 x 14.63 x 82.6	Yes; to SRP	8 elements per 2 weeks	25-50	
37	University of Missouri, Rolla	7.57 x 8.74 x 87.0	No	None anticipated		
43	University of Virginia	7.61 x 8.26 x 93.7				
44	University of Virginia	7.61 x 8.26 x 93.7	Yes; to SRP		17	2000
45	University of Washington	7.11 x 6.1 x 65.1	No			100
47	Virginia Polytechnic Institute	7.62 x 15.24 x 66.0		1-2 plates every 2 years		
49	Worcester Polytechnic Institute	7.75 x 7.75 x 101.6	No	None anticipated		100
50	Rhode Island Nuclear Science Center	7.62 x 7.62 x 100.3	Yes; to SRP	30 elements/1.5 years	12	1750

Table 4.4.4. Educational Reactors - TRIGA type (U-Zr hydride)

Ref. no.[a]	Location	Power level	Fuel	Enrichment %	Cladding	Number of elements in reactor	Total U-235 in reactor (kg)
3	Columbia University	250 kW	U-ZrH	20	SS		
4	Cornell University	100 kW	U-ZrH	19	SS		
9	Kansas State University	250 kW	U-ZrH	<20	SS	80	2.7
13	Michigan State University	250 kW	U-ZrH	20	SS		
16	Oregon State University	1 MW	U-ZrH	70	SS		11.17
17	Pennsylvania State University	1 MW	U-ZrH	<20	SS	95-101	3.42
19	Reed College	250 kW	U-ZrH	20	SS, Al		2.3
23	Texas A&M University	1 MW	U-ZrH	20-70	SS		
24	University of Arizona	100 kW	U-ZrH	20	SS		3.31
25	Univ. of Calif. Berkeley	1 MW	U-ZrH	20	SS	106	3.6
26	Univ. of Calif. Irvine	250 kW	U-ZrH	19.9	SS	81	2.9
30	University of Illinois	1 W	U-ZrH	20	Al	55	2.09
31	University of Illinois	1.5 MW	U-ZrH	20	SS	100	3.8
34	University of Maryland	250 kW	U-ZrH	20	SS	96	3.4
40	University of Texas	250 kW	U-ZrH	20	SS		
42	University of Utah	100 kW	U-ZrH	<20	Al	87	2.9
46	University of Wisconsin	1 MW	U-ZrH	70	SS	91	
48	Washington State University	1 MW	U-ZrH	20-70	SS	110	6.7

Table 4.4.4. (continued)

Ref. no.[a]	Location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/year
3	Columbia University	3.74 diam x 72.05	No	None anticipated		600
4	Cornell University					
9	Kansas State University	3.74 diam x 72.05	No	None		300
13	Michigan State University	3.76 diam x 72.4			~1.0 g/yr	
16	Oregon State University	3.73 diam x 72.06	Yes, HANF	~1 element/year	4-8%	~900
17	Pennsylvania State University	3.73 diam x 72.14	No	~6 elements/2 years	40-50,000 MWd/MTU	561
19	Reed College	3.73 diam x 72.14				~150
23	Texas A&M University	3.58 diam x 76.2				2500
24	University of Arizona	3.73 diam x 72.3				~0.3 MWd/yr
25	Univ. of Calif. Berkeley	3.63 diam			~10 g U-235/10 years	150
26	Univ. of Calif. Irvine	3.81 diam x 71.12		1-2 years		650
30	University of Illinois	3.73 diam x 71.12	No	None anticipated		1
31	University of Illinois	3.73 diam x 71.12		~2 elements/yr		270
34	University of Maryland	3.58 diam x 68.58	No	None anticipated	~0.3 g U-235/yr	48
40	University of Texas	3.76 diam x 72.06				
42	University of Utah		No	None anticipated		
46	University of Wisconsin	3.58 diam x 68.3	Not stated	~10 years	37-63%	600-800
48	Washington State University			~10 years	47-65 g U-235/yr	900

[a]Reference numbers refer to Appendix 4A.

Table 4.4.5. Educational reactors - AGN type (UO_2-polyethylene)

Ref. no.[a]	Location	Power level (W)	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	^{235}U in reactor (kg)
2	Catholic University	0.1	UO_2-polyethylene disks	19.9	None	9	0.69
7	Idaho State University	5	UO_2-polyethylene disks	19.9	None	9	0.67
12	Memphis State University	0.1	UO_2-polyethylene disks	19.9	None	9	0.66
22	Texas A&M University	5	UO_2-polyethylene disks		None	9	0.69
38	University of New Mexico	5	UO_2-polyethylene disks		None	9	
39	University of Oklahoma	15	UO_2-polyethylene blocks	19.84	Epoxy	12	0.81
41	University of Utah	5	UO_2-polyethylene disks	19.5	None	9	0.69

[a]Reference numbers refer to Appendix 4A, Table 4A.2.

Table 4.4.5 (continued)

Ref. no.[a]	Location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full-power hours/ year	Type
2	Catholic University	25.75 diam x 3.9, 2.3, or 1.0 thick	None	None anticipated	Negligible	50	AGN-201
8	Idaho State University	25 diam x 5.0, 2.5, or 1.0 cm	None	None anticipated	Negligible	100	AGN-201M
12	Memphis State University	25.4 diam	1 shipment to ORNL in DOT Type 6J drums	None anticipated	Negligible	~100	AGN-201-108
22	Texas A&M University	25.4 diam x 3.9, 2.3, or 1.0 thick			Negligible		AGN-201M
38	University of New Mexico	25.6 diam x 4.0, 2.0, or 1.0 thick			Negligible		AGN-201M-112
39	University of Oklahoma	7.32 x 7.77 x 70.17	None	None anticipated	Negligible	~100	AGN-211P
41	University of Utah	25.6 diam x 4.0, 2.0, or 1.0 thick			Negligible		AGN-201-107

[a]Reference numbers refer to Appendix 4A, Table 4A.2.

Table 4.4.6. Educational reactors - PULSTAR and other low-enriched pin-type

Ref. no.[a]	Location	Power level	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total ^{235}U in reactor (kg)
14	North Carolina State University	1 MW	UO_2	4	Zircaloy-2	25	12.7
21	State University of New York	2 MW	UO_2	6	Zircaloy-2	32	18.2
5	Cornell University	0 W	UO_2	2.1	Al	815	35

(Continuation)

Ref. no.[a]	Location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/year
14	North Carolina State University	8.0 x 6.96 x 96.5 (5 x 5 pin array)			0.032 kg ^{235}U/year	~ 750
21	State University of New York	8.0 x 6.96 x 96.5 (5 x 5 pin array)	Yes, to INEL (in 1979)	>10 year	7,000-15,000 MWd/ton	6,240
5	Cornell University	1.69 diam x 158	None	None anticipated	Negligible	Zero-power crit. facility

[a]Reference numbers refer to Appendix 4A, Table 4A.2.

Table 4.4.7. Educational reactors - liquid fuel

Ref. no.[a]	Location	Power level (W)	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total ^{235}U in reactor (kg)
1	Brigham Young University	10	Uranyl sulfate in water	20	None	Liquid volume 35.0 liters	1.45
28	University of California Santa Barbara[b]	10	Uranyl sulfate in water	89	None	Liquid volume 35.0 liters	1.23

(Continuation)

Ref. no.[a]	Location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/ year	Type
1	Brigham Young University	Liquid volume 35.0 l	None	None anticipated	Negligible	~100	AI Type L-77
28	University of California Santa Barbara[b]	Liquid volume 35.0 l	None	None anticipated	Negligible	~20	AI Type L-77

[a]Reference numbers refer to Appendix 4A.

[b]This reactor has been shut down.

Table 4.4.8. Private commercial research and test reactors

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	Average rate of discharge
Aerotest Operations, Inc.[a] San Ramon, CA	250 kW	TRIGA	<20	2.74	Al	1200			
Babcock and Wilcox[b] Lynchburg, VA	1 kW	Pin-type fuel	2.5-4.0		Al or SS	Shut down; all fuel shipped to SRP			
Babcock and Wilcox[c] Lynchburg, VA	200 kW					Shut down; all fuel shipped to SRP			
Dow Chemical Co.[d] Midland, MI	100 kW	TRIGA	<20	2.8	SS				
General Atomic[f] San Diego, CA	250 kW	TRIGA	20	2.6	Al or SS	150-300	2.0	Shipped to Exxon, Idaho	
General Atomic[f] San Diego, CA	1500 kW	TRIGA F	70	13.7	SS	300-400	20.0	Shipped to Exxon, Idaho	
General Electric NTR[g] Pleasanton, CA	100 kW	Disk plate fuel	93	3.7	Al	600		h	
General Electric GETR[i] Pleasanton, CA	50000 kW	Plate-type fuel	93		Al	Shut down		Shipped to INEL	
Northrop Research Center[j] Palos Verdes, CA	1000 kW	TRIGA F	20		SS				
Cintichem, Inc.[k] Tuxedo, NY	5000 kW	Plate-type fuel	93	5.0	Al	8000		Shipped to SRP for reprocessing	One element every 12 days (0.196 kg ^{235}U)
Westinghouse Electric[l] Zion, IL	100 W	Concentric tube fuel element	93	4.4	Al	1530		m	

[a]J. J. Haskins, (415) 837-4248.
[b]T. C. Engelder, (804) 522-5145.
[c]T. C. Engelder, (804) 522-5145.
[d]C. W. Kocher, (517) 636-0304.
[e]W. L. Whittemore, (714) 455-3277.
[f]W. L. Whittemore, (714) 455-3277.
[g]Denny Smith, (415) 862-2211.
[h]Fuel (full core) probably will be sent to INEL for reprocessing between 1991 and 1995 (Smith 1986).
[i]Denny Smith, (415) 862-2211.
[j]G. B. Cozens, (213) 970-2297.
[k]J. J. McGovern, (914) 351-2131.
[l]Karen Rueter and R. J. Banchak, (312) 872-4585.
[m]Approximately 3.2 kg of highly enriched fuel was sent to SRP in 1981 (Banchak, 1986).

Table 4.4.9. Research and test reactors owned by DOE

Name and location	Power level	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total U-235 in reactor (kg)
Argonne Thermal Source Reactor Argonne National Laboratory	10 kW	U-Al alloy	93.2	Al		
Biological Research Reactor Argonne National Laboratory	200 kW	U-Al alloy	93	Al		
Argonne Fast Source Reactor Idaho Falls, ID	1 kW	U metal	93	SS		
EBR-II Idaho Falls, ID	62.5 MW	U alloy	67	SS	87	240
Neutron Radiography Reactor Idaho Falls, ID	250 kW	U-Zr hydride	70	SS	64	7.5
Transient Test Reactor Idaho Falls, ID	80 kW	UO_2		Zircaloy		
Zero Power Plutonium Reactor Idaho Falls, ID	Critical facility	Pu-U-Mo alloy		SS		
Brookhaven Medical Research Reactor, Brookhaven National Laboratory	3 MW	U-Al alloy	90-93	U-Al alloy	28	3.85
High Flux Beam Reactor Brookhaven National Laboratory	60 MW	U_3O_8 cermet plates	93	Al	28	8.23

Table 4.4.9. (continued)

Name and location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/year
Argonne Thermal Source Reactor Argonne National Laboratory	8.2 x 7.62 x 67.9				
Biological Research Reactor Argonne National Laboratory	7.62 diam x 128.65				
Argonne Fast Source Reactor Idaho Falls, ID	11.5 diam x 5.1 cm				
EBR-II Idaho Falls, ID	5.82 diam x 167.1	Shipped to Exxon Chemical Reprocessing Plant, ID	6-8 elements every 6 weeks	28,000 MWd	6,130
Neutron Radiography Reactor Idaho Falls, ID	7.62 x 7.62 x 83.82	Shipped to Idaho Chemical Reprocessing Plant	20 years	20%	600
Transient Test Reactor Idaho Falls, ID	10.16 x 10.16 x 121.92				
Zero Power Plutonium Reactor Idaho Falls, ID					
Brookhaven Medical Research Reactor, Brookhaven National Laboratory	7.62 x 7.62 x 87.0	None	1 element per year	20%	300
High Flux Beam Reactor Brookhaven National Laboratory	8.18 x 7.16 x 145.42	None	7 elements every 22 days	37%	6,800

Table 4.4.9. (continued)

Name and location	Power level	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total U-235 in reactor (kg)
Destroyer Dual Reactor Knolls Atomic Power Lab.						
Modifications and Additions Facility Knolls Atomic Power Lab.						
Submarine Advanced Reactor Knolls Atomic Power Lab.						
Trident Prototype Reactor Knolls Atomic Power Lab.						
Advanced Reactivity Measurement Facility Idaho Falls, ID	10 kW	U-Al plate fuel	93	Al	28	5.08
Advanced Test Reactor Idaho Falls, ID	250 MW	U-Al plate fuel	93	Al	40	35-40
ATR Critical Facility Idaho Falls, ID	5 kW	U-Al plate fuel	93	Al		
Coupled Fast Reactor Measurement Facility Idaho Falls, ID	100 kW	U-Al plate fuel	93	Al	32	5.7
Loss of Fluid Test Reactor Idaho Falls, ID	50 MW	UO_2 pellets in pins	4	Zircaloy-4		520.0
Power Burst Facility Idaho Falls, ID	28 MW	UO_2 pellets in rods	18.5	SS	68	104.0

Table 4.4.9. (continued)

Name and location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/year
Destroyer Dual Reactor Knolls Atomic Power Lab.					
Modifications and Additions Facility Knolls Atomic Power Lab.					
Submarine Advanced Reactor Knolls Atomic Power Lab.					
Trident Prototype Reactor Knolls Atomic Power Lab.					
Advanced Reactivity Measurement Facility Idaho Falls, ID	8.28 x 8.28 x 98.81	None	No refueling anticipated	-	None
Advanced Test Reactor Idaho Falls, ID	6.5 x 10.5 x 168.0	Reprocessed at Idaho Chem. Reproc. Plant	15 to 40 days	35-56%	7,000
ATR Critical Facility Idaho Falls, ID	6.5 x 10.5 x 168.0				
Coupled Fast Reactor Measurement Facility Idaho Falls, ID	8.28 x 8.28 x 98.8		No refueling anticipated		
Loss of Fluid Test Reactor Idaho Falls, ID	21.4 x 21.4 x 200 and 18.8 x 26.6 x 200	Reprocessed at ICPP	6 mo - 2 yr	3,100-9000 MWD/MTU	~250
Power Burst Facility Idaho Falls, ID		To be shipped to INEL	No refueling anticipated		

Table 4.4.9. (continued)

Name and location	Power level	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total U-235 in reactor (kg)
Big Ten Critical Assembly Los Alamos National Lab.	Critical facility	U cylinder	10			
Jezebel Critical Facility Los Alamos National Lab.	Critical facility	Pu sphere		Nickel		
PARKA Critical Assembly Los Alamos National Lab.	Critical	U-graphite	93	None		
Fast Burst Research Reactor Los Alamos National Lab.		U alloy	93	Al		
Flattop Critical Assembly Los Alamos National Lab.	Critical facility	U-Pu	93-94.9	Ni		
COMET Critical Assembly Los Alamos National Lab.	Critical facility					
HONEYCOMB Critical Assembly Los Alamos National Lab.	Critical facility		Normally 93			
MARS Critical assembly Los Alamos National Lab.	Critical facility					
OMEGA West Reactor Los Alamos National Lab.	8 MW	U-Al alloy	93	Al		
Plasma Core Assembly Los Alamos National Lab.	100 W	U	93			

Table 4.4.9. (continued)

Name and location	Element dimensions (cm)	Frequency of Fuel shipments	Spent fuel refueling	Full power burnup	hours/year
Big Ten Critical Assembly Los Alamos National Lab.					
Jezebel Critical Facility Los Alamos National Lab.					
PARKA Critical Assembly Los Alamos National Lab.					
Fast Burst Research Reactor Los Alamos National Lab.	17.8 diam x 17.8				
Flattop Critical Assembly Los Alamos National Lab.					
COMET Critical Assembly Los Alamos National Lab.					
HONEYCOMB Critical Assembly Los Alamos National Lab.					
MARS Critical assembly Los Alamos National Lab.					
OMEGA West Reactor Los Alamos National Lab.	MTR plate fuel				
Plasma Core Assembly Los Alamos National Lab.					

Table 4.4.9. (continued)

Name and location	Power level	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total U-235 in reactor (kg)
Solution High Energy Burst Assembly Los Alamos National Laboratory	5 kW	UO_2F_2 solution	4.95			
Bulk Shielding Reactor Oak Ridge National Laboratory	2 MW	U-Al plates	93	Al	24-28	4.0-5.0
Health Physics Research Reactor Oak Ridge National Laboratory	10 kW	U-Mo discs	93.2	Ni		97.0
High Flux Isotope Reactor Oak Ridge National Laboratory	100 MW	U-Al plates	93	Al		9.4
Oak Ridge Research Reactor Oak Ridge National Laboratory	30 MW	U-Al plates	93	Al	30-33	5.0
Pool Critical Assembly Oak Ridge National Laboratory	10 kW	U-Al plates	93	Al	30	3.4
Tower Shielding Reactor II Oak Ridge National Laboratory	1 MW	U-Al plates	93	Al		8.37
Horizontal Split Table Rocky Flats Plant	Critical facility					
Solution System Rocky Flats Plant	Critical facility					
Tank Reservoir Rocky Flats Plant	Critical facility					

Table 4.4.9. (continued)

Name and location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/year
Solution High Energy Burst Assembly Los Alamos National Laboratory					
Bulk Shielding Reactor Oak Ridge National Laboratory	7.61 x 8.05 x 87.63	Shipped to SRP for reprocessing	4 elements per	50-70%	
Health Physics Research Reactor Oak Ridge National Laboratory		Ship to Y-12 facility, Oak Ridge		Negligible	
High Flux Isotope Reactor Oak Ridge National Laboratory	26.9 diam x 79.1	Shipped to SRP for reprocessing		30-35%	~8,000
Oak Ridge Research Reactor Oak Ridge National Laboratory	7.6 x 8.0 x 97.4	Shipped to SRP for reprocessing	~2 wks	50-70%	6,455
Pool Critical Assembly Oak Ridge National Laboratory	7.6 x 8.0 x 87.63				~300
Tower Shielding Reactor II Oak Ridge National Laboratory		Shipped to SRP for reprocessing			
Horizontal Split Table Rocky Flats Plant					
Solution System Rocky Flats Plant					
Tank Reservoir Rocky Flats Plant					

Table 4.4.9. (continued)

Name and location	Power level	Fuel	Enrichment (%)	Cladding	Number of elements in reactor	Total U-235 in reactor (kg)
Vertical Split Table Rocky Flats Plant	Critical facility					
Annular Core Research Reactor Sandia National Laboratory	2 MW	UO_2 pellets	35	SS	247	23.6
Sandia Pulsed Reactor II Sandia National Laboratory	5 kW	U-Mo plates	93	Al	10	87.0
Sandia Pulsed Reactor III Sandia National Laboratory	5 kW	U-Mo plates	93	Al	18	215.9
Nuclear Test Gage Savannah River Plant	Subcritical	U-Al plates	93			
Savannah River Test Pile Savannah River Plant	50 W					
Fast Flux Test Facility Hanford Eng. Development Lab.	400 MW	PuO_2-UO_2 pellets	Natural	SS	83	550 Pu
Neutron Radiograph Facility Hanford Eng. Development Lab.	250 kW	U-Zr hydride	<20	Al, SS	65	
Large Ship Reactor Idaho National Eng. Lab.						
Natural Circulation Reactor Idaho National Eng. Lab.						
Submarine Reactor Facility Idaho National Eng. Lab.						
Fast Burst Reactor White Sands Missile Range	10 kW	U-Mo alloy	93.2	Ni	97.0	

Table 4.4.9. (continued)

Name and location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/year	
Vertical Split Table Rocky Flats Plant						
Annular Core Research Reactor Sandia National Laboratory	3.73 diam x 73.15	None anticipated	No refueling	Negligible	~200	
Sandia Pulsed Reactor II Sandia National Laboratory	20.5 diam x 3.47	None anticipated	No refueling	Negligible	<100	
Sandia Pulsed Reactor III Sandia National Laboratory	29.7 diam x 2.26	None anticipated	No refueling	Negligible	<1,000	
Nuclear Test Gage Savannah River Plant						
Savannah River Test Pile Savannah River Plant						
Fast Flux Test Facility Hanford Eng. Development Lab.	11.0 diam x 365.8	100 days	25 elements per hvy metal	45-80 MWd/kg	~5,000	
Neutron Radiograph Facility Hanford Eng. Development Lab.	3.81 diam x 72.4		Annually		300	(TRIGA)
Large Ship Reactor Idaho National Eng. Lab.						
Natural Circulation Reactor Idaho National Eng. Lab.						
Submarine Reactor Facility Idaho National Eng. Lab.						
Fast Burst Reactor White Sands Missile Range						

Table 4.4.10. Research and test reactors owned by U.S. government agencies other than DOE

Location	Power level	Fuel	Enrichment %	Cladding	Number of elements in reactor	Total U-235 in reactor (kg)
U.S. Army Aberdeen Proving Grounds, MD	10 kW	U-10% Mo alloy	93.2	Ni	11	97
Armed Forces Radiobiology Research Institute Bethesda, MD	1 MW	TRIGA U-Zr Hydride	20	SS	87	3.31
National Bureau of Standards Washington, DC	10 MW	MTR plate fuel, U-Al	93	Al	30	5.7
U.S. Geological Survey Denver, CO	1 MW	TRIGA U-Zr Hydride	19	SS	116	3.08
Veterans Administration Omaha, Nebraska	18 kW	TRIGA U-Zr Hydride	20	Al	56	2.03
U.S. Army, White Sands Missile Range, NM	10 kW	U-10% Mo alloy	93.2	Ni	11	97

[a]Essentially identical to ORNL Health Physics Reactor.

Table 4.4.10. (continued)

Location	Element dimensions (cm)	Fuel shipments	Frequency of refueling	Spent fuel burnup	Full power hours/year
U.S. Army Aberdeen Proving Grounds, MD	200 diam x 21			Negligible	
Armed Forces Radiobiology Research Institute Bethesda, MD	3.73 diam x 71.9	Yes, to SRP	Entire core, at interval of 25-30 years	~11 g/year	30
National Bureau of Standards Washington, DC	7.6 x 8.55 x 174.8	Yes, to SRP	4 elements/6 weeks	60-80%	6800
U.S. Geological Survey Denver, CO	3.73 diam x 72.0	Yes, to SRP	>10 yr		900
Veterans Administration Omaha, Nebraska	3.76 diam x 72.1		None expected	<1 g/yr U-235	1854
U.S. Army, White Sands Missile Range, NM	200 diam x 21			Negligible	

Table 4.4.11. Reprocessing of spent fuels from research and test reactors

Fuel category	Fuel type	Fuel	Enrichment %	Cladding	Fuel shipments	Feasibility of reprocessing
1	MTR, plate type, ARGONAUT, etc.	U-Al alloy	90-93	Al	Some fuel has been shipped to INEL and SRP	Easily feasible[a]
2	TRIGA	U-Zr hydride	19-20	SS, Al	Some fuel has been shipped to INEL	Difficult
3	UO_2-polyethylene discs or blocks	UO_2-polyethylene	19.9	-	One shipment to ORNL	Difficult
4	PULSTAR, 5 x 5 pin arrays, low enriched pin-type fuels	UO_2	4-6	Zircaloy-II	Some fuel has been shipped to INEL	Feasible, but see note b
5	Liquid fuel	Uranyl sulfate in water	20-89	-	None	Feasible
6	U-Mo alloy	U-Mo alloy	93.2	Ni		c
7	Miscellaneous					c

[a]This fuel is routinely reprocessed at INEL and SRP.

[b]This fuel is similar to commercial power reactor (LWR) fuel and could be canistered for disposal without reprocessing.

[c]Not determined.

Table 4.4.12. Canister requirements for research and test reactor fuels

Fuel category	Fuel description	Number of reactors	Estimated number of fuel elements[a]	Total U-235 (kg)	Estimated disposal volume (m^3)[b]
1	MTR plate-type highly enriched U-Al alloy fuels	40	20,000	3,000	c
2	TRIGA U-ZrH fuels, SS or Al clad, mostly 20% enriched	28	4,500	160	40
3	Homogeneous UO_2-polyethylene discs or blocks, 20% enriched	8	87	6	1
4	PULSTAR and other low-enriched UO_2 pin-array fuels	6	971	115	10
5	Liquid fuels (aqueous)	3	-	3	c
6	U-Mo alloy highly-enriched fuels	6	d	d	d
7	FFTF mixed oxide fuel	1	d	d	d
8	Miscellaneous fuels	27	d	d	d

[a]Total through year 2020, including fuel in reactors at that time.
[b]Assuming criticality safety can be achieved by addition of neutron poisons.
[c]Reprocessed and disposed of as vitrified defense HLW (less than 1 canister).
[d]Not determined.

Table 4.4.13. 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 power 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.

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

Table 4.4.14. 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 diameter x 4.0 cm thick contains about 0.6 kg of uranium.

Table 4.4.15. Radioactivity and thermal power of PULSTAR spent fuel element[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.

Table 4.4.16. Dimensions of TRIGA stainless-steel-clad fuel[a]

Drawing No.	TRIGA fuel type	Fitting type	A (in.)	B (in.)	C (in.)	D (in.)
T13S210D210	Standard-streamline	I-A	29.68	1.478	2.56[b]	1.435
TOS210D210	Standard-plain	II-A	28.9	1.478	3.42	1.435
T4S210D105	Four rod cluster	III-A	29.88	1.414	3.42	1.37
T5A210D210	ACPR[c]	IV-A	28.89	1.478	3.45	1.40

[a]Source: Tomsio 1986.

[b]Lower graphite is longer than upper graphite. Lower graphite = 3.72 in.

[c]Annular core pulsed reactor.

Table 4.4.17. Historical and projected spent fuel discharged from the FFTF[a]

End of calendar year	Number of fuel assembies 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 highly enriched uranium (30-62% ^{235}U).

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

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

Table 4.4.18. 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

Quantities of miscellaneous spent fuels exist at a number of sites throughout the nation. The nature of the materials ranges from specimens usually containing no more than a few grams to multiple fuel assemblies containing several megagrams of irradiated heavy metal present in a different form from that considered previously. In this report section, we will make no attempt to describe individual waste units containing less than 100 kg of heavy metal. For larger quantities of spent fuel, primary emphasis will be on indicating where detailed descriptive material may be found. We will also attempt to provide the following data, realizing that in some cases the values presented constitute only best estimates based on minimal data and calculations:

- mass of heavy metal,
- chemical form,
- burnup (MWd/MTIHM),
- approximate disposal volume (estimated), and
- source for additional information.

A summary of the quantities of miscellaneous fuels stored at various sites is provided in Table 4.5.1. Some of these fuels were removed from various research and test reactors already discussed in Section 4.4. Thus, in estimating ultimate disposal requirements, care must be taken not to count these fuels twice.

4.5.1 Argonne National Laboratory West (ANL-W)

The materials stored by ANL-W are briefly described in Table 4.5.2. Only relatively small quantities of scrap are now being held at ANL-W some of which may be amenable to reprocessing. Also, others have a reactive metal (sodium or a sodium-potassium metal eutectic) bond between the fuel material and the cladding. It is conceivable that the reactive metal-containing materials could be emplaced without first

destroying the reactive metal, although it is probable that such removal will be required. If reactive metal elimination is required to permit acceptance by a repository, it is probable that the active metal removal would be followed by reprocessing.

4.5.2 Babcock & Wilcox (B&W)

The radioactive materials stored by Babcock & Wilcox are derived, with one exception, from a few LWRs. The fuel assemblies or fuel rods have been cut into pieces small enough to fit inside a 4.25-in.-diam by 33-in.-long container, a sketch of which is presented in Fig. 4.5.1. The basic information concerning these materials is provided in Table 4.5.3. Additional information may be obtained by contacting K. D. Long (B&W Nuclear Materials Accountability Specialist), P.O. Box 11165, Lynchburg, Virginia, 24506-1165, telephone 804-5982, or FTS 671-1060, ext. 5982.

4.5.3 Battelle-Columbus Laboratories (B-C)

This institution currently has in storage minor quantities of spent fuels which were discharged from a number of LWRs. The bulk of the items comprise relatively few kilograms of heavy metals, but three essentially intact fuel assemblies contain almost 1.2 MT of the total (about 1.5 MT) heavy metal held by Battelle-Columbus.

Records are available at Battelle which reveal the origin of each quantity of waste material (Pasupathi 1986).

All the bare fuel that is stored at B-C has been placed in aluminum tubes which are closed with screwed pipe plugs. These tubes are, in turn, housed in dry storage casks.

Some cut rods and/or bare fuel pellets are encapsulated. Encapsulation consists of placing the material (fuel rod sections or pellets) inside a relatively close-fitting stainless steel tube, which is subsequently sealed by welding (Pasupathi 1986).

It is planned that these spent fuels be shipped to other sites, with the bulk of the material going to INEL/ICPP and the remainder going either to Morris, Illinois, or the Connecticut Yankee reactor. The transfer is expected to be completed by April 1987 (Pasupathi 1986).

A listing and brief description of the B-C held spent fuels is provided in Table 4.5.4.

4.5.4 Battelle Pacific Northwest Laboratory (BPNL)

The highly radioactive material being held in storage at BPNL is briefly described in Table 4.5.5. With the exception of two small batches of glass mix, all the described materials are conventional LWR spent fuels, a small portion of which has been opened to obtain specimens. This material should be accorded the same treatment as will be given to other LWR fuels. Much of the LWR fuel will be consumed in various experimental programs at BPNL, and will thus blend in with other waste streams, e.g. HLW.

4.5.5 Hanford Engineering Development Laboratory (HEDL)

Table 4.5.6 provides information concerning spent fuels being held at Hanford. The listed material consists of fuel sections and whole fuel pins, plus 90 TRIGA fuel assemblies. The Westinghouse Hanford Company is planning to send this material to retrievable burial. Prior to being sent to retrievable burial, the pins will be chopped to a size which will fit inside 5.1-in.-diam x 41-in.-long welded steel capsules. This includes a small quantity of FFTF fuel. It is expected that all of the listed fuel other than TRIGA will fit into 5 of these capsules. The TRIGA fuel assemblies will be packaged into 13 concrete-filled, 55-gal steel drums (Engel 1986). The ^{235}U content of each drum will then be ~257 g, which is well below the minimum critical mass of ~700 g. If the TRIGA fuel is left in the concrete-filled, 55-gal drums, the total volume of the materials held at HEDL will be about 5 m^3.

4.5.6 Idaho National Engineering Laboratory (INEL)

The highly radioactive materials being stored at INEL and potentially bound for a repository in the as-is condition are briefly described in Table 4.5.7. The total mass of heavy metal in storage is about 136 MT. About 40 MT of this is in the general form of conventional LWR fuel, while another 34 MT was used as blanket material in 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 the above uranium-based wastes, there is about 57 MT of predominantly thorium-containing waste. The latter is from a light water breeder reactor (LWBR) and from two high temperature gas-cooled reactors (HTGRs). The LWBR fuel is an oxide, whereas the HTGR fuel is a carbide, as described in a previous section of this report.

The Fermi I blanket fuel contains a quantity of reactive metal. There is some question as to whether the quantity present in a waste package would be sufficient to compromise the function of that package or to detract from the function of a geologic repository. However, the restrictions on reactive components will likely prohibit leaving the sodium with the fuel, so that some measure of predisposal treatment, to remove the sodium, may be necessary.

4.5.7 Los Alamos National Laboratory (LANL)

The material stored at LANL consists of irradiated test fuels from EBR-II (see Table 4.5.8). The fuel types are oxides, carbides, and nitrides clad in stainless steel. The total mass of heavy metal is reported to be 127.5 kg, with the ^{235}U and Pu contents being 54.5 kg and 31 kg, respectively. It is planned that these fuels be packaged and shipped to another site within the next few years. The packaging operation will consist of the following steps:

- cropping the major portion of unfueled tubing and hardware, leaving primarily fueled tubing that is unsealed;
- sealing the short, cut lengths of fueled tubing inside 0.5-in.-OD x 38-in.-long tubes that are closed by refrigeration-type fittings which employ a Neoprene rubber gasket;
- enclosing a number of the 0.5-in. by 38-in.-long tubes inside a 4-in. stainless steel tube that will be sealed by welding; and
- placing the filled 4-in. stainless steel tubes inside 5-in. stainless steel tubes that will be sealed by welding.

A sketch of the elements of this system 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 4.5.9. The bulk of the mass is in the form of solidified U_3O_8 cake contained within 3.5-in.-OD x 24-in.-long stainless steel cans. This material has substantial radioactivity because of its ^{232}U content of 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. Whereas the primary container mates with a magnetic lifting tool, 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 also contained in ~3.5-in.-diam stainless steel cans of various lengths. These are shown in Table 4.5.9.

Fuel from the Molten Salt Reactor Experiment (MSRE) is stored in Bldg. 7503 at ORNL. 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 2000 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 very high, about 400,000 nCi/g, while that of the flush salt is about 6000 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 ^{93}Zr, an activation product), the actinide plus daughters activity, and the thermal output in watts, all out to one million years (Notz 1985).

Also stored at ORNL are ten Peach Bottom Unit 1, core 2 fuel assemblies. (See Sect. 4.3 for a description of these assemblies.)

4.5.9 Savannah River Plant (SRP)

The miscellaneous fuels stored at the Savannah River Plant (SRP) are listed in Table 4.5.10. 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, SRP

has required that incoming materials be accompanied by extensive descriptive matter, including drawings, compositions, etc. Such information is available from SRP on items of significant quantity (O'Rear 1987).

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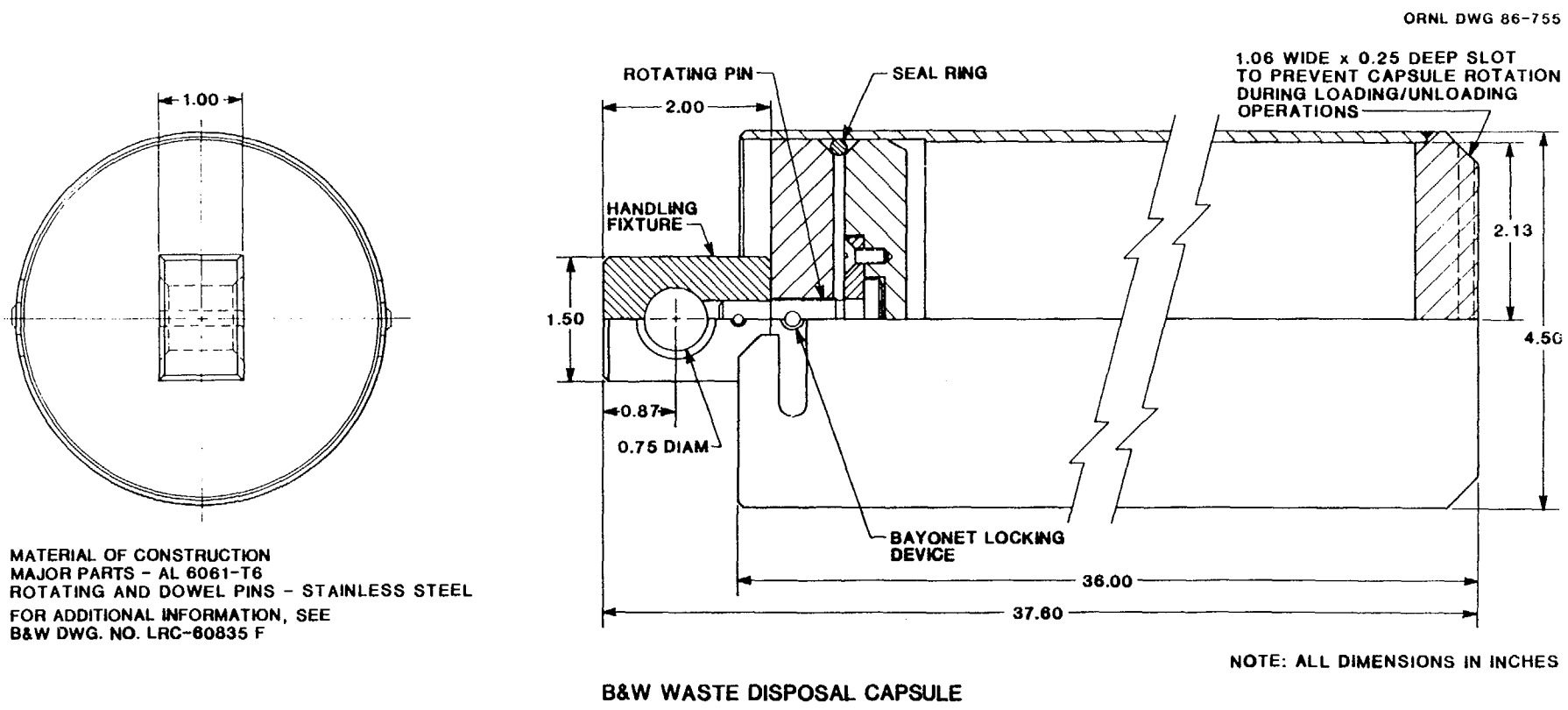

Fig. 4.5.1. B&W spent fuel container.

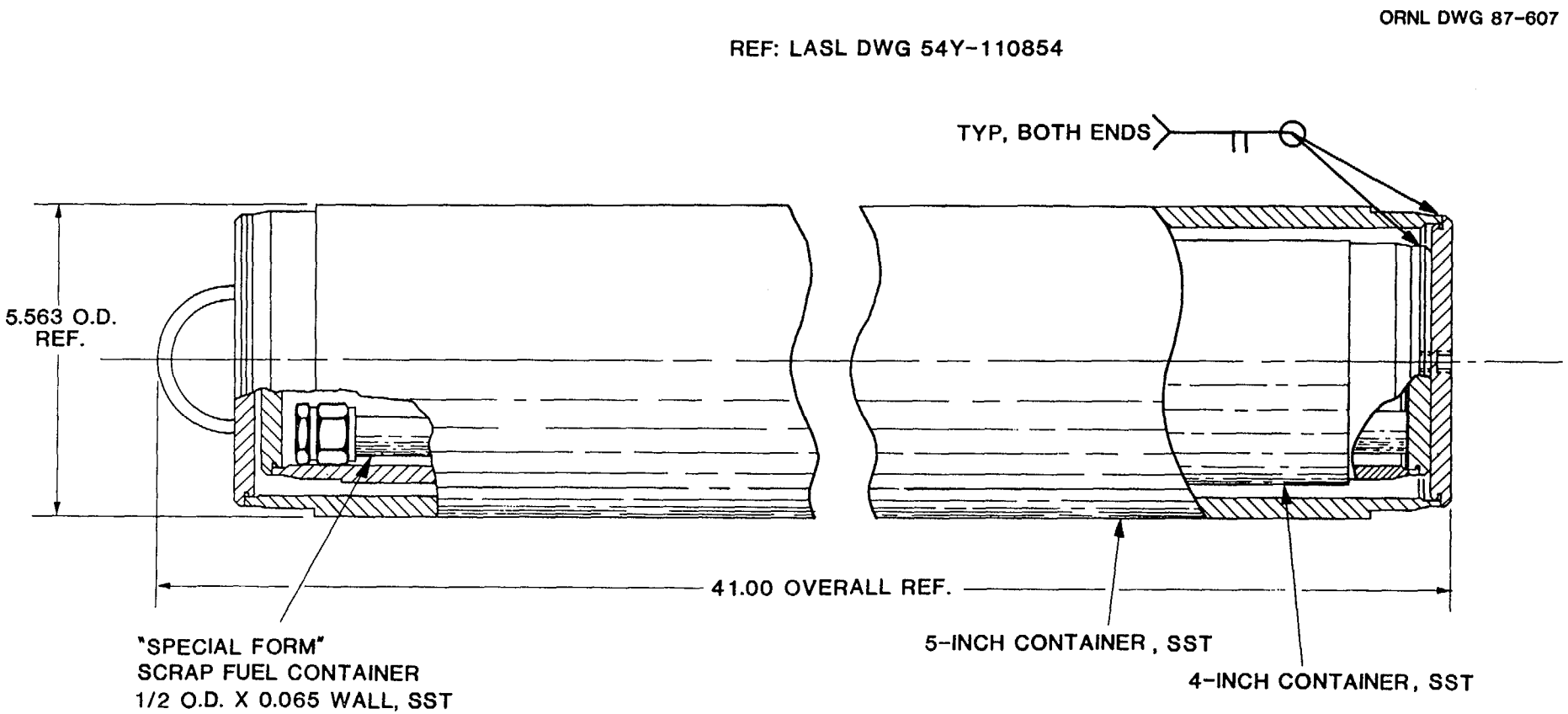

Fig. 4.5.2. The Los Alamos Scientific Laboratory multicontainer system for scrap fuel.

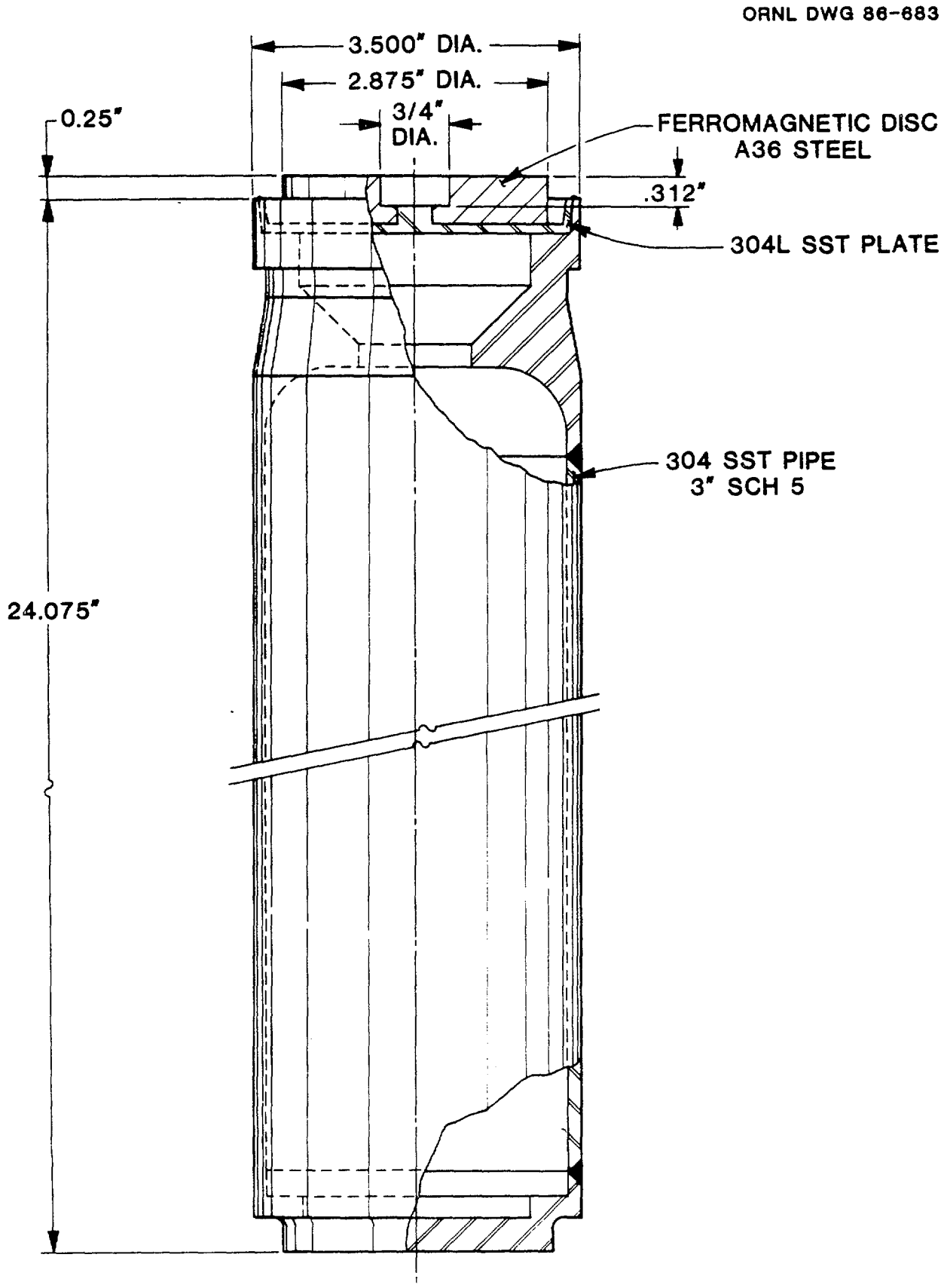

Fig. 4.5.3. Storage can assembly for CEU solidified waste.

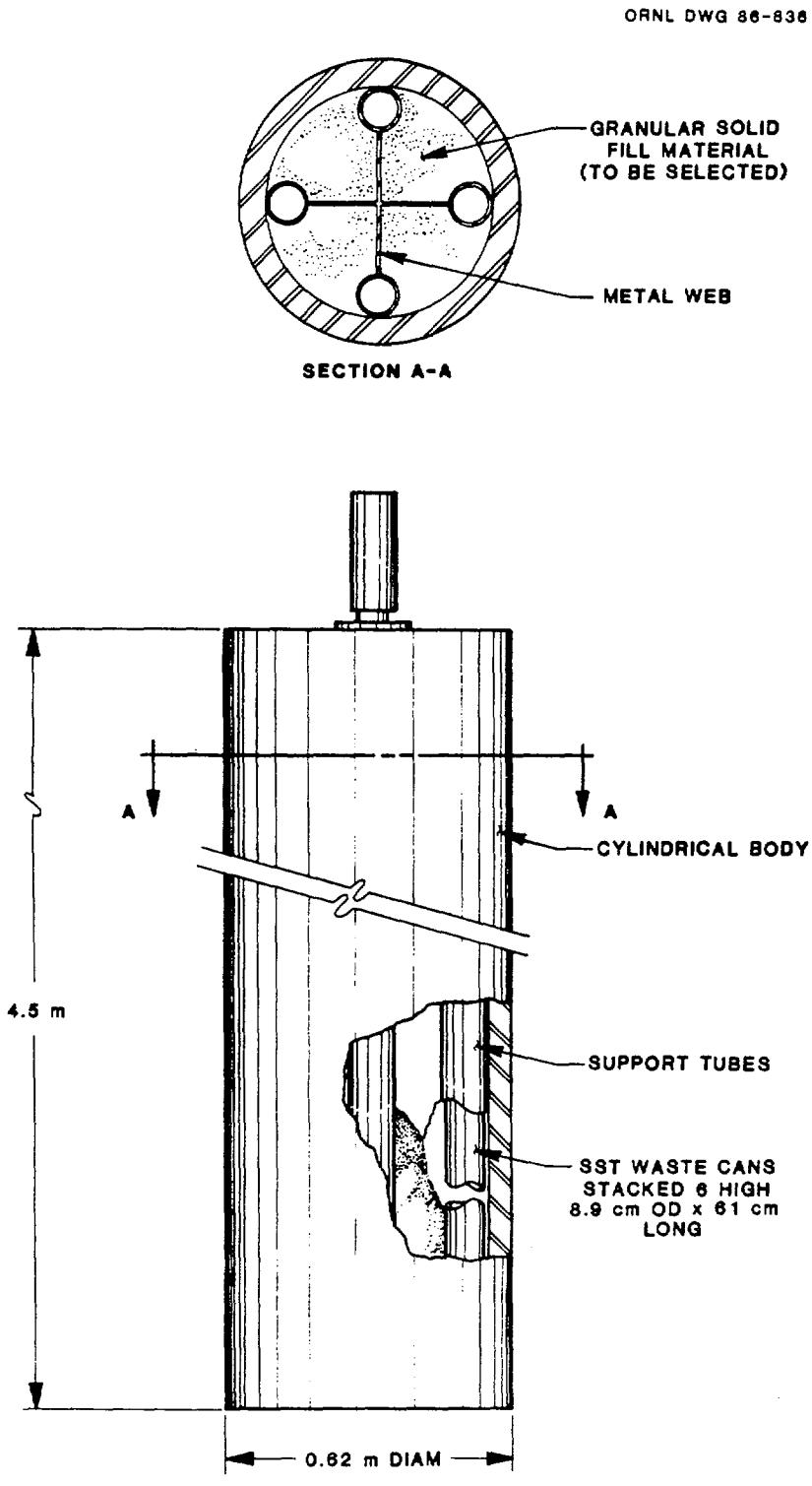

Fig. 4.5.4. CEU waste disposal package concept.

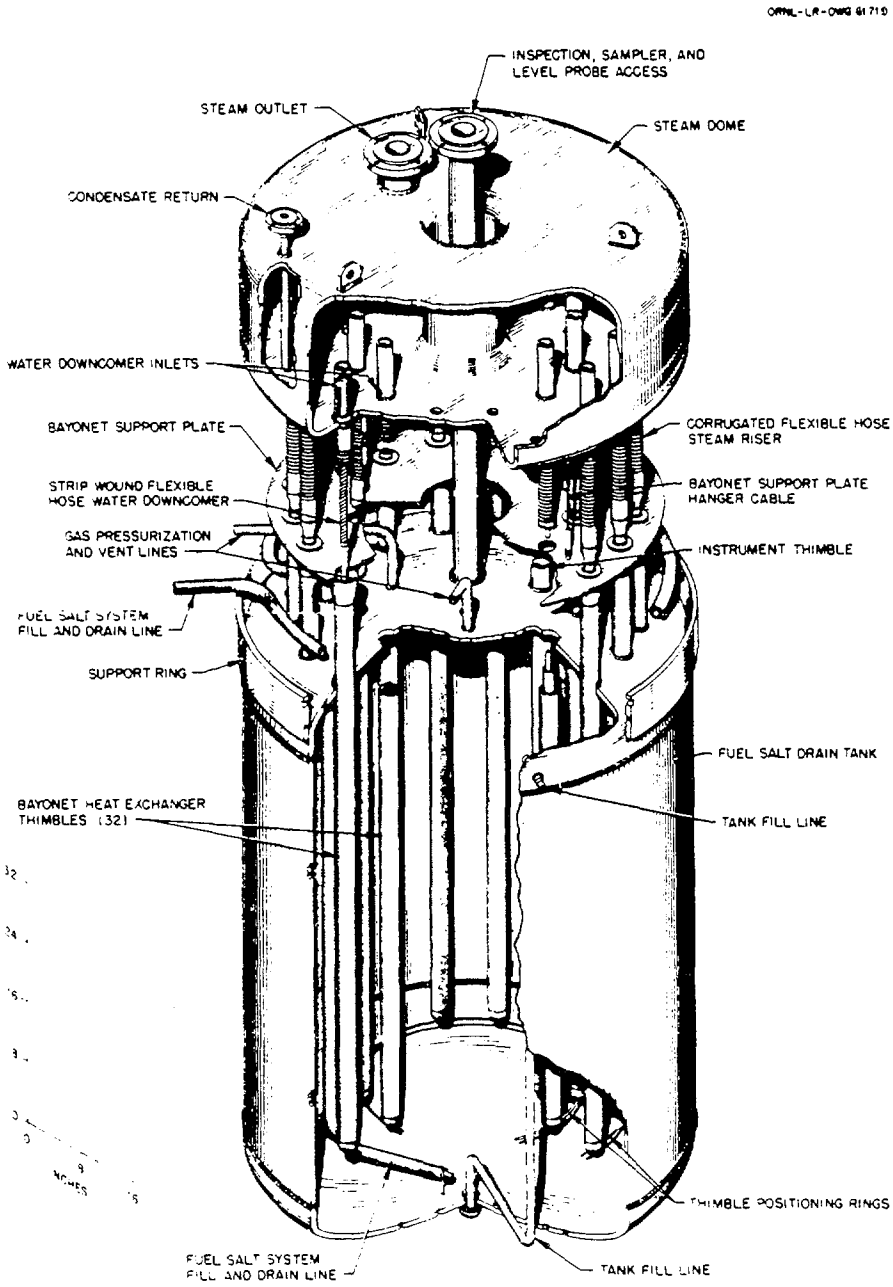

Fig. 4.5.5. Fuel-salt drain tank.

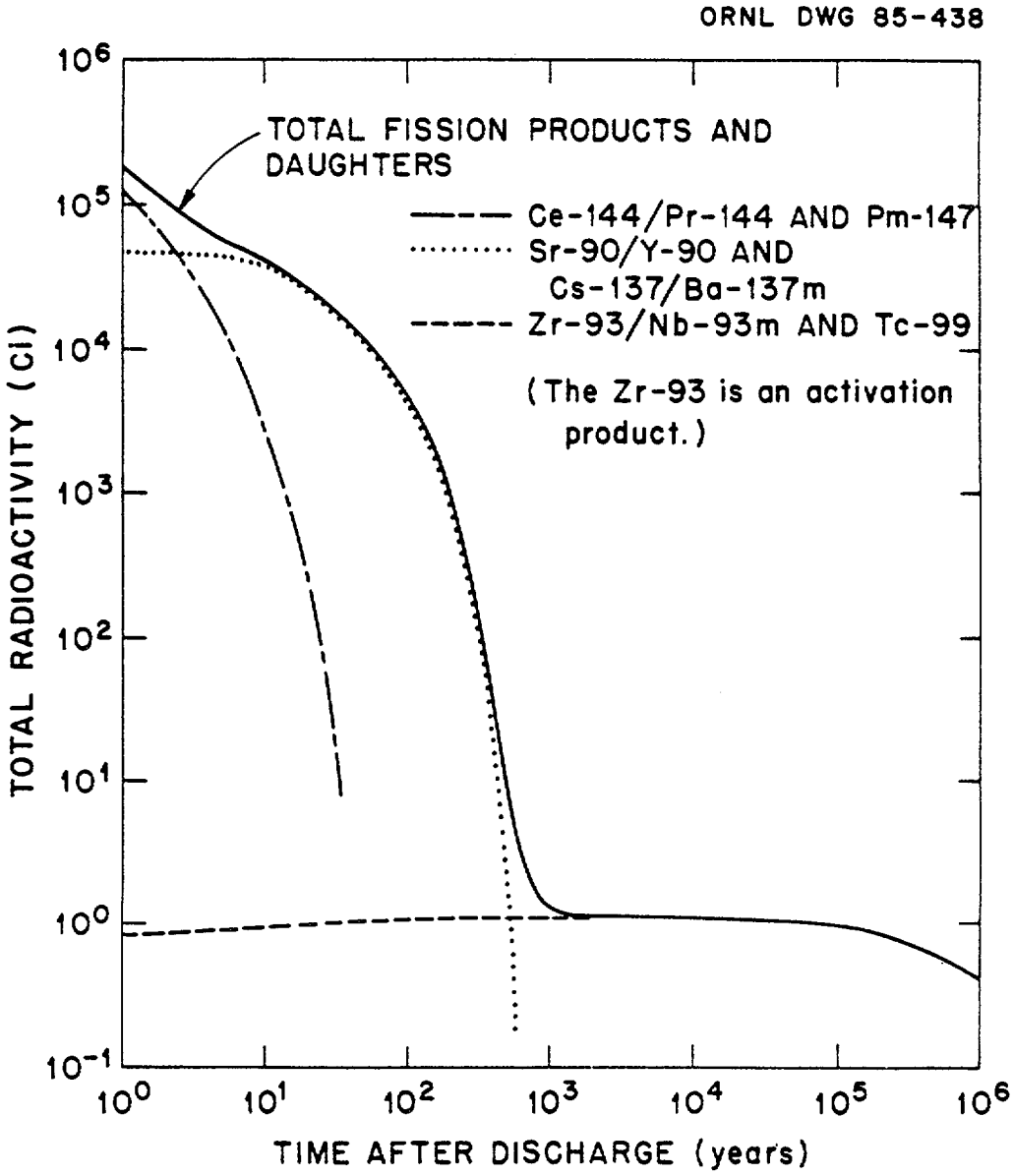

Fig. 4.5.6. Fission product activities of MSRE fuel and flush salts.

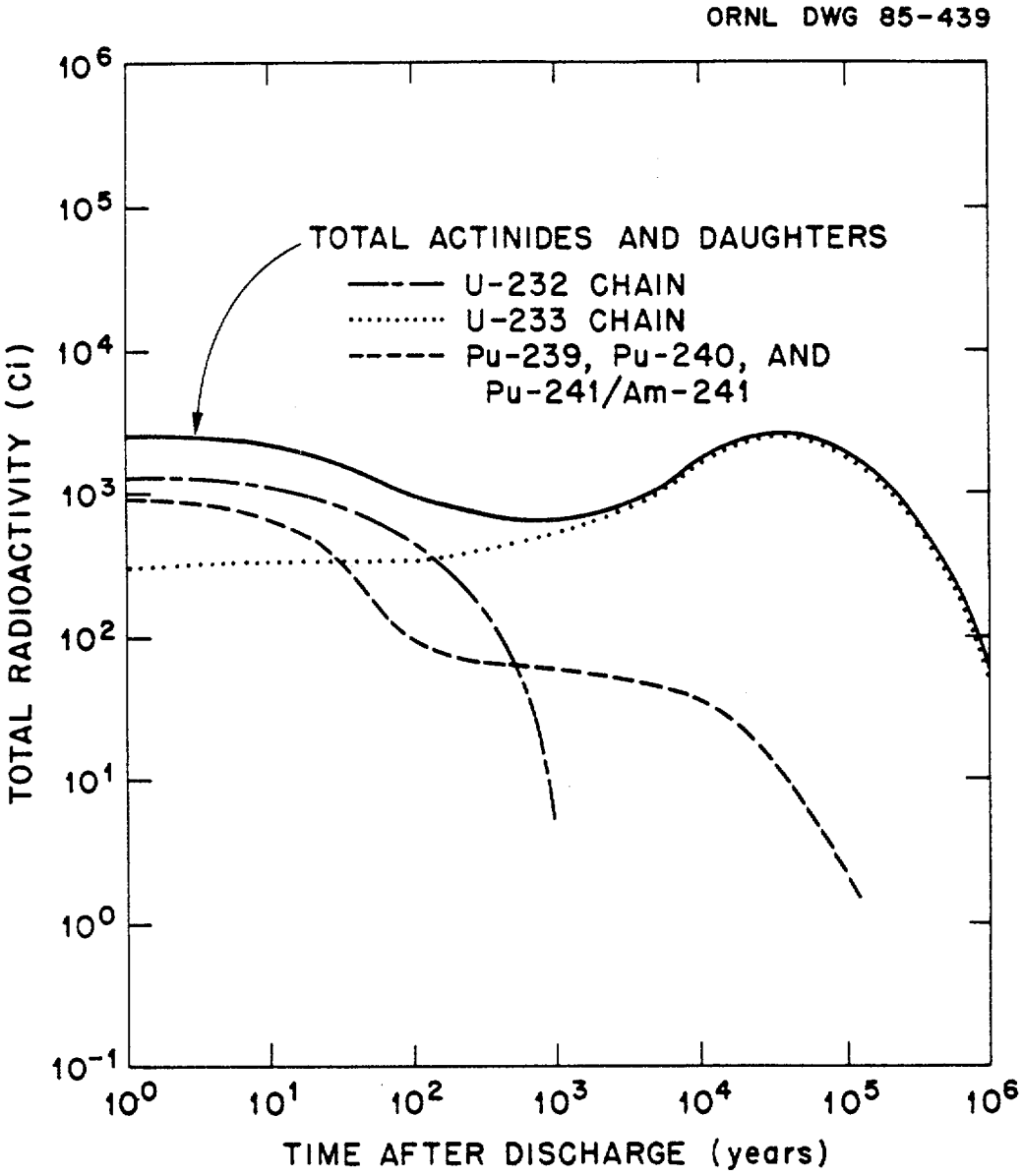

Fig. 4.5.7. Actinide and daughter activities of MSRE fuel and flush salts.

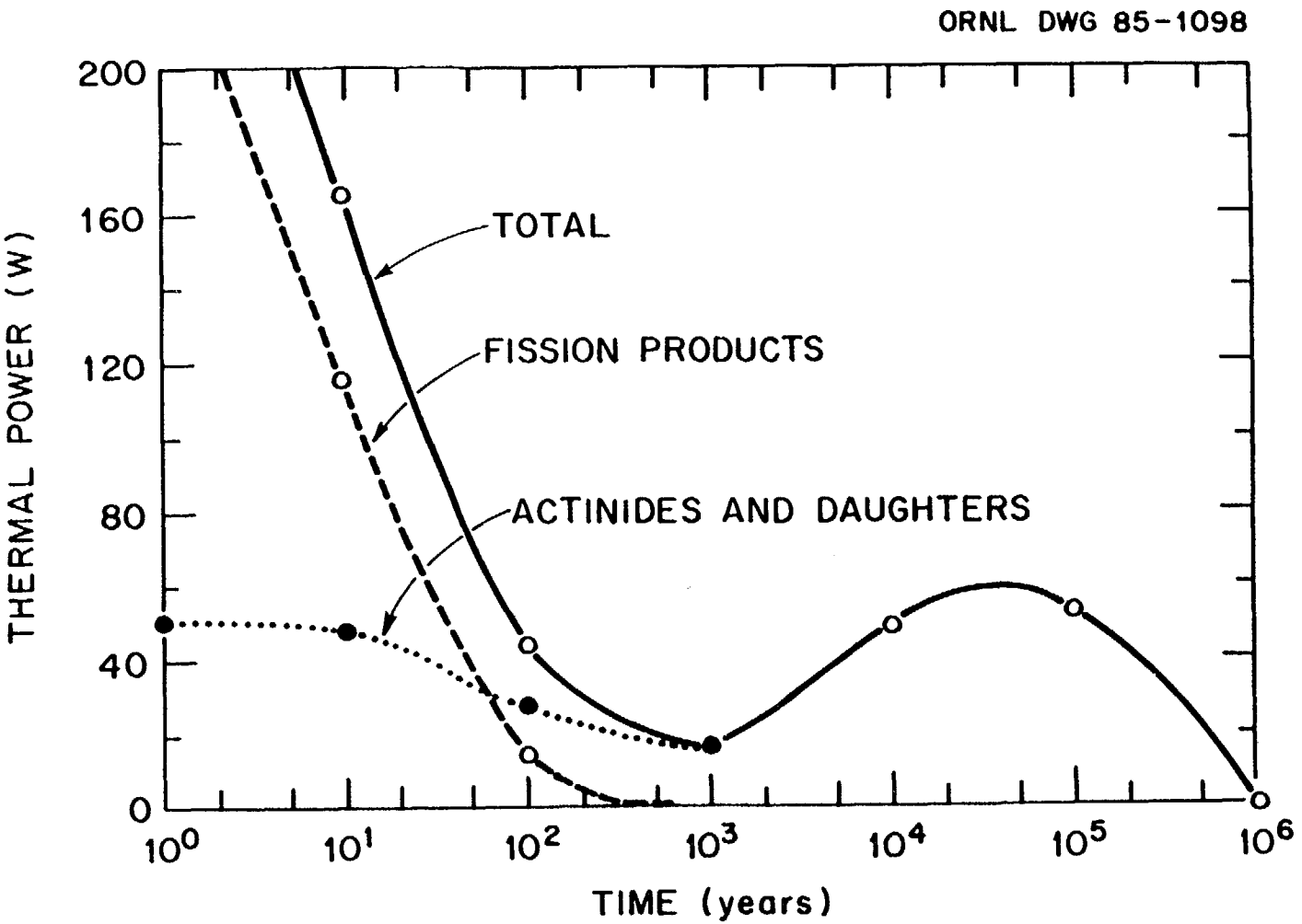

Fig. 4.5.8. Thermal output of MSRE fuel and flush salts.

Table 4.5.1. Total quantities of miscellaneous fuels stored at various sites, as of December 31, 1986

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]		
Argonne National Laboratory; Idaho Falls, ID	311.19	302.18	19.930		9.009	
Babcock & Wilcox; Lynchburg, VA	53.85	53.47	1.162		0.377	
Battelle Memorial Institute; Columbus, OH	1,505.22	1,492.10	11.764		13.123	
Battelle Pacific Northwest Laboratory; Richland, WA	2,251.47	2,218.0	17.86		26.77	6.7
Hanford Engineering Development Laboratory; Richland, WA	70.2	60.0	10.4		10.2	
Idaho Chemical Processing Plant[b] — Idaho National Engineering Laboratory; Idaho Falls, ID	136,015.83	77,790.15	1,329.55	862.26	251.68	57,974.0
Los Alamos National Laboratory; Los Alamos, NM	127.49	96.52	54.47	0.133	30.97	
Oak Ridge National Laboratory; Oak Ridge, TN	1,275.65	1,257.85	803.99	280.28	0.795	17.00
Savannah River Plant; Aiken, SC	19,020.57	10,330.06	746.22	31.15	42.31	8,648.2
Total	160,631.5	93,600.3	2,995.3	1,173.8	385.2	66,645.8

[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.

Table 4.5.2. Miscellaneous, highly radioactive materials stored at Argonne National Laboratory as of December 31, 1986[a]

Material	Composition	Description	Irradiation level	U content, kg		Total Pu content (kg)
				Total	^{235}U	
Radioactive Waste and Scrap Facility[b]						
Basic research — ANL	Scrap	Stored in canister[c]	1 to 500 R at 1 ft	181.53	12.860	5.111
EBR-2 blanket subassembly	Scrap	Stored in canister[c]	1 to 500 R at 1 ft	104.80	0.230	0.180
LMFBR test fuel	Scrap	Stored in canister[c]	1 to 500 R at 1 ft	13.33	5.253	3.026
Postirradiation test on NUMEC LMFBR	Scrap	Stored in canister[c]	1 to 500 R at 1 ft	0.72	0.345	0.123
Sodium Loop Safety Facility	Scrap	Stored in canister[c]	1 to 500 R at 1 ft	1.80	1.242	0.569
Total				302.18	19.930	9.009

[a] Ref. Teunis 1986.

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

[c] Canisters are retrievable and constructed of stainless steel with minimum dimensions of 8 in. OD and 5 ft long. 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, highly radioactive materials stored at Babcock & Wilcox, Lynchburg Research Center, as of December 31, 1986[a]

Material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg		Total Pu content (kg)
				Total	^{235}U	
B&W Test Reactor	UO_2, Zr-clad	Stored in fourteen 4.25-in.-diam × 33-in. Al canisters	Unknown[c]	2.126	0.659	<0.0005
Consolidated Edison	UO_2, Zr-clad	Stored in a 4.25-in.-diam × 33-in. Al canister	Unknown[c]	0.091	0.048	<0.0005
Oconee I	UO_2, Zr-clad	Stored in twenty-six 4.25-in.-diam × 33-in. Al canisters	20,000	2.66	0.032	0.044
			30,000	11.28	0.077	0.093
			40,000	11.00	0.057	0.101
			50,000	7.00	0.025	0.094
Oconee I	UO_2-Gd_2O_3, Zr-clad	Stored in four 4.25-in.-diam × 33-in. Al canisters	10,000	7.905	0.104	0.047
Oconee II	UO_2, Zr-clad	Stored in seven 4.25-in.-diam × 33-in. Al canisters	27,000	10.702	0.105	0.021
Unknown	UO_2, Zr-clad	Stored in a 4.25-in.-diam × 33-in. Al canister	Unknown[c]	0.706	0.055	<0.0005
Total				53.47	1.162	0.377

[a]See Long 1986.
[b]Zr-clad = Zircaloy-clad.
[c]Currently in underground storage tubes.

Table 4.5.4. Miscellaneous, highly radioactive materials stored at the Battelle Memorial Institute, Battelle Columbus Laboratories, as of December 31, 1986[a]

Material	Composition[b]	Description	Estimated burnup (MWd/MTIHM)	U content, kg		Total Pu content (kg)
				Total	^{235}U	
Connecticut Yankee	UO_2, SS-clad	0.442-in. diam × 144 in. (stored as intact fuel assembly)	35,000	388.00	5.335	3.871
Fort Calhoun	UO_2, Zr-clad	0.440-in. diam × 137 in. (stored as intact fuel assembly, intact fuel rods, and cut fuel rod sections)	40,000–45,000 ~55,000	39.00 348.00	0.203 0.766	0.035 3.745
Maine Yankee	UO_2, Zr-clad	0.440-in. diam × 137 in. (stored as cut fuel rod sections)	15,000	21.07	0.287	0.104
Monticello	UO_2, Zr-clad	0.563-in. diam × 144 in. (stored as cut fuel rod sections and punctured full-length fuel rods)	~31,000 50,000	16.00 17.00	0.056 0.042	0.162 0.147
Oconee-1	UO_2, Zr-clad	0.422-in. diam × 144 in. (stored as base fuel and cut fuel rod sections)	24,000–28,000	79.94	0.330	0.307
Oyster Creek	UO_2, Zr-clad	0.422-in. diam × 127 in. (stored as full-length and cut fuel rod sections)	~22,000	49.31	0.459	0.355
H. B. Robinson	UO_2, Zr-clad	0.422-in. diam × 144 in. (stored as cut fuel rod sections)	28,000	21.58	0.157	0.178
Shippingport	UO_2, Zr-clad	0.4-in. diam × 12 in. (stored as short pins)	18,000	2.00	0.012	0.005
Surry-1	UO_2, Zr-clad	0.422-in. diam × 144 in. (stored as cut fuel rod sections)	17,000–48,000	20.00	0.120	0.186
Turkey Point-3	UO_2, Zr-clad	0.422-in. diam × 144 in. (stored as intact fuel assembly, base fuel, long and short fuel rod sections)	30,000–32,000 22,000	26.00 431.10	0.175 3.652	0.222 3.442
Zion I-1	UO_2, Zr-clad	0.422-in. diam × 144 in. (stored as cut fuel rod sections)	17,000–56,800 ~50,000 ~20,000 ~27,000	21.00 4.00 2.00 6.10	0.063 0.022 0.014 0.071	0.224 0.042 0.020 0.050
Total				1,492.10	11.764	13.123

[a]Ref. Faust 1986.
[b]Zr-clad = Zircaloy-clad.

Table 4.5.5. Miscellaneous, highly radioactive materials stored at Battelle Pacific Northwest Laboratory, as of December 31, 1986[a]

Material	Composition[b]	Description	Estimated burnup (MWd/t)	U content, kg		Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U		
Calvert Cliffs	UO_2, Zr-clad	0.440-in. diam × 147 in.					
		(stored as 175 intact rods, 1 cut rod[c])	30,000	371.0	3.0	5.287	
		(stored as 154 intact rods, 1 cut rod[c])	45,000	293.7	1.7	7.723	
Cooper	UO_2, Zr-clad	98 rods[c]	26,000	366.5	2.6	3.065	
Point Beach-1	UO_2, Zr-clad	Stored as three intact fuel assemblies, miscellaneous cut samples	32,900	1,169.4	10.38	10.572	6.7
H. B. Robinson	UO_2, Zr-clad	Stored as 19 cut fuel rod sections[c]	30,000	5.1	0.04	0.04	
Shippingport				1.0	d	0.005	
VBWR[e]	UO_2, Zr-clad	Twelve 3-ft fuel rod segments	20,000-30,000	11.1	0.14	0.074	
PNL Lot Numbers:							
ATM-5		Glass mix		0.1	d	0.001	
ATM-6		Glass mix		0.1	d	<0.001	
Total				2,218.0	17.86	26.77	6.7

[a]Ref. Mendel 1986.
[b]Zr-clad = Zircaloy-clad.
[c]Stored in a hot cell.
[d]Negligible.
[e]Vallecitos boiling-water reactor.

Table 4.5.6. Miscellaneous, highly radioactive materials stored at the Hanford Engineering Development Laboratory, as of December 31, 1986[a]

Material	Composition	Description	Irradiation level	U content, kg		Total Pu content (kg)
				Total	^{235}U	
EBR-2 (Experimental Breeder Reactor)	UO_2/PuO_2, SS-clad	5.1-in. diam[b] × 41 in.	100 R at 1 m	10.1	6.5	3.1
FFTF (Fast Flux Test Facility)	UO_2/PuO_2, SS-clad	5.1-in. diam[b] × 41 in.	100 R at 1 m	22.7	0.5	7.0
TRIGA (Training Reactor, Isotopes, General Atomic)	Zr-U hydride (8 wt % U), Al-clad	3.6-cm diam[c] × 72 cm	3 R at 1 m	17.2	3.3	
Various LWR fuel sections	UO_2 pellets	5.1-in. diam[b] × 41 in.	100 R at 1 m	10.0	0.1	0.1
Total				60.0	10.4	10.2

[a]Ref. Engel 1986.

[b]Pins and pin sections are loaded into welded stainless steel canisters 5.1 in. in diameter by 41 in. long. Fissile loading is 4 kg Pu + ^{235}U.

[c]TRIGA assemblies are 3.6 cm in diameter by 72 cm long. Storage/burial is in 55-gal concrete-filled drums, six to seven assemblies per drum.

Table 4.5.7. Miscellaneous, highly radioactive materials stored at the Idaho National Engineering Laboratory, as of December 31, 1986[a]

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								
Bettis scrap	Scrap	Stored in 50 cans		30.338	7.015	0.119	0.079	36.0
GCRE (Gas-Cooled Reactor Experiment)	UO_2-BeO, Hastelloy X clad	1 SS tube, 5 in. × 25.5 in.		0.984	0.918			
LWBR (Light-Water Breeder Reactor)	Ceramic pellets, Zr-clad, Th blanket	65 units		770.554		729.037		47,208.0
PWR scrap (Pressurized-Water Reactor)	UO_2, Zr scrap			>79.743	79.743			
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	Stored in three Al cans 2.68 in. × 52.5 in.		59.065	55.022			
Subtotal				>989.329	188.023	729.156	0.079	47,244.0
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 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, carbon-coated particles in graphite matrix	582 hexagonal graphite blocks 14.2 in. across flats × 31.2 in.		299.096	164.169	86.779	0.748	8,110.0
Pathfinder	UO_2-B_4C pellets, SS-clad	417 assemblies 9-in. diam × 80 in.		53.406	49.242			
Peach Bottom	U-Th carbide, carbon-coated particles in graphite matrix	1,603 graphite blocks 3.5-in. diam × 12 ft	>1[c]	332.420	223.540	46.310	0.970	2,620.0
Pulstar, State University of New York at Buffalo	UO_2 pellets, Zr-clad	Stored in 24 SS cans, 3 in. × 3 in. × 35.5 in.		251.431	12.083		0.793	

Table 4.5.7. (continued)

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)						
TRIGA (Training Reactor, Isotopes General Atomic)	Al- or SS-clad elements	703 units stored in 83 cans		131.285	28.395			
VBWR (Geneva) (Vallecitos Boiling-Water Reactor)	UO_2 and UO_2-TiO_2, SS-clad	142 rods stored in four 6-in.-diam × 36-in. SS cans	8[c]	12.383	2.606			
Subtotal				35,246.639	600.874	133.089	9.033	10,730.0
		DOE/Civilian Development Programs material stored at INEL (other than ICPP)						
CANDU (Canadian Deuterium Reactor)	UO_2 pellets, Zr-clad	8 pins	5,000	2.660	0.261			
Dresden	UO_2, Zr-clad	54 pins (depleted U)		165.0	Unknown		1.064	
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	0.223		0.005	
Halden 226 and 239 Assy	UO_2-PuO_2 pellets, Zr-clad	12 pins		>0.007	0.007		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 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	16.669	0.472		0.087	

Table 4.5.7. (continued)

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 stored at INEL (other than ICPP) (continued)								
PBF (Power-Burst Facility)	UO_2-ZrO_2-CaO; Zr sleeves, SS-clad	Pins		725.690	130.890			
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-2 (Three Mile Island)	Rubble		(Quantities not known until entire core received)					
Turkey Point-3	UO_2 pellets, Zr-clad	17 assemblies	25,000-30,000	7,422.701	54.582		62.018	
VEPCO (Virginia Electric Power Company)		69 assemblies		30,207.295	242.457		172.695	
Subtotal				>41,554.178	>540.655		>242.566	
Total at INEL				>77,790.146	>1,329.552	862.245	>251.678	57,974.0

[a]See Berreth 1987. Many of the fuels at INEL have lower uranium enrichment than are normally processed. These fuels could be reprocessed in a special campaign, if required.

[b]Zr-clad = Zircaloy-clad.

[c]Data expressed in percentage.

Table 4.5.8. Miscellaneous, highly radioactive materials stored at the Los Alamos National Laboratory, as of December 31, 1986[a]

Material	Composition	Description	Irradiation level	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 × 13.5 in.	330 R/h at 1 m	96.52	54.47	0.133	30.97

[a]See Valencia 1986.

Table 4.5.9. Miscellaneous, highly radioactive materials stored at the Oak Ridge National Laboratory, as of December 31, 1986

Material	Composition[a]	Description	Estimated burnup (MWd/MTIHM) or irradiation level	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
CEU (Consolidated Edison Uranium)	U_3O_8-CdO solid cake	Stored in 401 3.5-in.-OD × 24-in. SS cans	~350 R/h at 1.5 in.	1,044.38	797.70	101.32		
Dresden-1	UO_2, Zr-clad	Sheared fuel pins stored in 2-qt paint cans	~24,000	5.00	0.024		0.020	
		9/16-in.-diam × 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 × 8-in. fuel test capsules	1,000-2,000	0.399	0.222			
KSTR [KEMA (Holland) Suspension Test Reactor]	UO_2-ThO_2 microspheres	Stored in 5.43-in.-OD × 40-in. SS vessel	~10 R/h at 1.5 in.	5.78	5.19			17.00
Monticello	UO_2, Zr-clad	1/2-in.-diam × 6-in. fuel rod sections plus short lengths	40,000	1.00	0.004		0.008	
MSRE[b] (Molten Salt Reactor Experiment)	LiF_2-BeF_2-ZrF_2-UF_4	Stored in three Inconel tanks, each 50-in. diam x 86-in. high	~5 x 10^4 Ci total (see ref. 10)	36.95	0.940	31.01	0.743	
Oconee-1	UO_2, Zr-clad	1/2-in.-diam × 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 × 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 × 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 × 12-in. fuel rod sections plus short lengths	30,000	1.00	0.005		0.004	
SRO (Savannah River Oxide)	UO_2 powder	Stored in thirty-five 3.88-in.-OD × 10-in. SS cans	~500 R/h at 1.5 in.	67.41		61.61		

Table 4.5.9. (continued)

Material	Composition[a]	Description	Estimated burnup (MWd/MTIHM) or irradiation level	U content, kg			Total Pu content (kg)	Total Th content (kg)
				Total	^{235}U	^{233}U		
TRIGA (Training Reactor, Isotopes, General Atomic)	U-Zr hydride, Incoloy-clad	1/2-in.-diam sheared fuel pins stored in 2-qt paint cans	1,000 R/h at 12 in.	0.441	0.089			
ORNL Inventory Item Nos.								
AUA-67/AUA-70	U metal chunks from LANL	Stored in two 3.75-in.-OD × 18-in. SS cans	~100 R/h at 1.5 in.	6.02		5.89		
CZA-91	UO_x powder from ANL	Stored in two 3.5-in.-OD × 13-in. SS cans	~100 R/h at 1.5 in.	0.881		0.856		
HUA-2A	UO_x powder from HEDL	Stored in five 3.75-in.-OD × 7-in. SS cans	~100 R/h at 1.5 in.	0.317		0.307		
RCP-02	UO_2 powder from SRO	Stored in thirty-two 3.5-in.-OD × 24-in. SS cans	~100 R/h at 1.5 in.	11.14		10.72		
RCP-04	UF_4-LiF_2 powder converted from SRO UO_2	Stored in six 3.5-in.-OD × 24-in. SS cans	~100 R/h at 1.5 in.	3.19		2.92		
RCP-06	U_3O_8-CdO solid cake	Stored in twenty-seven 3.5-in.-OD × 24-in. SS cans	~50 R/h at 1.5 in.	65.55		60.60		
RCP-20/JZBL	U metal chunks from LANL	Stored in five 3.5-in.-OD × 24-in. SS cans	~100 R/h at 1.5 in.	5.15		5.05		
Total				1,257.85	803.99	280.28	0.795	17.00

[a]Zr-clad = Zircaloy-clad.

[b]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. Several decontamination and decommissioning studies have been completed; however, at this time, extended storage of the solidified fuel salts is the most prudent and rational course to take. See Notz 1985.

Table 4.5.10. Miscellaneous, highly radioactive materials stored at the Savannah River Laboratory, as of December 31, 1986[a]

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 SRP								
CANDU (Canadian Deuterium Reactor)	UO_2, Zr-clad	Rods stored in three 5.0-in.-diam × 14-ft cans; pieces stored in three 3.5-in.-diam × 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 × 14-ft can	Unknown	67.37	0.640		0.200	
Dresden	UO_2-ThO_2, SS-clad	Intact assemblies stored in 4.4-in. × 4.4-in. × 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. × 3.5 in. × 81.62 in.	Max. 50,000	224.34	186.159	14.722		4,818.6
LWR samples (Light-Water Reactor)	UO_2-PuO_2, SS- and Zr-clad	Stored in four 4.5-in.-diam cans		17.476	0.334		0.291	
Nereide (a French Experiment using DOE fuel)	USi_x, Al-clad	Materials Test Reactor plate-type fuel assembly 34.37 in. × 2.98 in. × 3.14 in.	600	35.45	7.015			
H. B. Robinson	UO_2-PuO_2, Zr-clad		Unknown	0.56	0.004		0.003	
Saxton	UO_2-PuO_2, Zr- or SS-clad	567 rods stored in eight 5.0-in.-diam × 14-ft cans	1,000	276.67	1.411		15.408	
		64 rods stored in a 3.75-in.-diam × 50-in. can						
	UO_2, Zr-clad	Multiple pins stored in four 5.0-in.-diam × 14-ft cans	1,600	66.79	6.866		0.233	
		One bundle stored in a 12-in.-diam × 14-ft can						
VBWR (Vallecitos Boiling-Water Reactor)	UO_2, Zr-clad	Stored in four 3.5-in.-diam × 12-in. cans	1,500	4.04	0.998		0.003	
Subtotal				1,426.796	241.203	30.113	18.017	6,675.8

Table 4.5.10. (continued)

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 SRP						
B&W scrap	UO_2-PuO_2, SS-clad	Stored in 3.5-in.-diam × 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 × 30-in. container	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 × 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. × 3.75 in. × 62.5 in.	1,600	1.73	1.612			
	UO_2, Zr-clad	Assemblies 3.75 in. × 3.75 in. × 62.5 in.	1,600	1,604.30	95.456			
	UO_2-Zr, Zr-clad	Assemblies 3.75 in. × 3.75 in. × 62.5 in.	1,600	5,031.77	73.967		9.092	
	UO_2-ZrO_2-CaO, Zr-clad	Assemblies 3.75 in. × 3.75 in. × 62.5 in.	1,600	28.93	26.651			
	UO_2-PuO_2, Zr-clad	Assemblies 3.75 in. × 3.75 in. × 62.5 in.	1,600	907.39	2.087		13.952	
EPR-1	PuO_2, SS-clad	Pieces stored in 4.5-in.-diam × 32-in. can	Unknown				0.022	
GCRE (Gas-Cooled Reactor Experiment)	UO_2 or UO_2-BeO, Hastelloy-clad			61.284	56.559			
HWCTR (Heavy-Water Components Test Reactor)	U and UO_2, Zr-clad	Intact assemblies 3 in. diam × 132 in. Pieces stored in 3.5-in.-diam × 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			3.698	3.423			
ML-1 (Mobile Low Power Plant No. 1)				59.209	54.478			

Table 4.5.10. (continued)

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 SRP (continued)								
ORNL (Oak Ridge National Laboratory)				0.18	0.171			
ORNL mixed oxide	UO_2-PuO_2, Zr- or SS-clad	Stored in a 3.5-in-diam × 15.12-in. can	Unknown but low	0.42	0.030		0.094	
Shippingport	UO_2, Zr-clad	Stored in a 10.5-in.-diam × 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 × 12-ft cans	Unknown	12.64	0.603			
SRE (Sodium Reactor Experiment)	UTh metal, SS-clad	Stored in 3.5-in.-diam × 110.25-in. cans	10,000	155.24	143.410	1.045		1,972.4
	UC, SS-clad			47.42	4.344		0.016	
SRP (Savannah River Plant)	UO_2-PuO_2, Zr-clad	Stored in a 12.0-in.-diam × 14-ft container	Unknown	69.00	0.304		0.161	
Subtotal				8,903.268	505.015	1.045	24.294	1,972.4
Total				10,330.064	746.218	31.148	42.311	8,648.2

[a]See O'Rear 1987. The spent fuels listed in this table are not reprocessible in existing facilities.

[b]Zr-clad = Zircaloy-clad.

5. MISCELLANEOUS WASTES

5.1 INTRODUCTION

This chapter pertains to all other nonfueled wastes which may require long-term isolation, except for LWR hardware, which is included in Chapter 2. At least some of this waste may fall into the low-level waste "greater-than-Class C" category (GTCC) and therefore not be suitable for disposal in a LLW facility. Some wastes may be LLW of Class C or lower, but not eligible for commercial disposal. These wastes are categorized as follows:

- wastes generated by operations within the OCRWM system;
- commercial TRU waste arising from operations other than reprocessing;
- activated metal deriving from decommissioning of reactors;
- sealed radioisotope source capsules which have gone beyond their useful lifetime or are no longer needed; and
- wastes from routine LWR operations.

Section 5.2 deals with system-generated wastes, that is, wastes that arise from the operation of the Civilian Radioactive Waste Management System. These wastes arise from operations such as decontamination of shipping casks and fuel consolidation and/or repackaging.

Section 5.3 covers commercial TRU waste, since WIPP will accept only TRU waste from defense operations under present policy. Commercial reprocessing is no longer being done in this country; therefore, the only sources are from the decommissioning of the West Valley plant and fuel fabrication plants and from specialized industrial or institutional facilities.

Section 5.4 deals with the activated metals remaining in a reactor core after shutdown and which are removed during decommissioning. Section

5.5 covers sealed isotopic capsules, a rather specialized source of radwaste. Some of these capsules may require extensive shielding. Section 5.6 covers LWR operating wastes deriving from out-of-core, nonfuel operations.

Since much of the waste material covered in this chapter falls into the GTCC category, some discussion of GTCC is included in this introduction. Primary responsibility for this area resides with the DOE Office of Nuclear Energy (DOE 1987).

5.1.1 Low-Level Waste Greater than Class C

GTCC wastes are radioactive wastes that exceed Class C limits on radionuclide concentrations, as specified in 10 CFR Part 61, NRC's regulations for low-level waste disposal (U.S. NRC 1982). However, by definition, they do not include spent reactor fuel or high-level waste from reprocessing. GTCC wastes are classified as such either because they contain certain long-lived radionuclides above specified concentrations or certain shorter-lived radionuclides above specified concentrations (see Table 5.1.1):

Long Half-Life Radionuclides	Half-Life	
C-14	5,730	yrs
Ni-59	75,000	yrs
Nb-94	20,000	yrs
Tc-99	4.2×10^6	yrs
I-129	15.7×10^6	yrs
alpha emitters longer than 5 years	--	
Pu-241 (a precursor)	14.4	yrs
Cm-242 (a precursor)	163	days
Short Half-Life Radionuclides		
all shorter than 5 years	--	
H-3	12.28	yrs
Co-60	5.27	yrs
Ni-63	100	yrs
Sr-90	28.62	yrs
Cs-137	30.17	yrs

The Pu-241 is a precursor to Am-241 (432 y) and Np-237 (2.1×10^6 yrs), and Cm-242 is a precursor to Pu-238 (87.7 yrs) and U-234 (0.24×10^6 yrs). The maximum limits for each of the above species for Class C are given in Table 5.1-1; the table also gives the limits for these species for Classes A and B. For some species which are soft beta emitters, the limits are higher when they occur in activated metals because of the self-shielding effect.

The upper limit of GTCC (that is, the lower limit of HLW) has not yet been defined. NRC is currently working on this and has received input from DOE and others, but they have not yet announced a firm proposal. A number of suggestions have been offered; in one of these, the GTCC (or "intermediate" level) is limited by either thermal output or by an activity 30 times the Class C limit. The intermediate wastes, although not generally suitable for shallow-land burial, could qualify for "greater confinement" disposal in either intermediate-depth burial or in an engineered facility. Obviously, deep geologic disposal would also be acceptable in the absence of an intermediate facility. Thus, the final formal definition of the lower limit for HLW could have an impact on the quantities of waste falling in the GTCC classification, and these wastes might be sent to a high-level geologic repository. They cannot be disposed of at low-level waste sites, except in cases where the NRC grants an exemption on an individual basis (this has only been done in a few instances).

Thus, there is considerable interest in this waste category, but the interest is of relatively recent origin. For this reason, the available characterization data are still rather preliminary. The EIA is in the process of conducting a survey of all the nuclear utilities and all licensed users of radioisotopes (their questionnaire NE-869). EPRI is collecting data. Several workshops have been held. EGG/ID, by virtue of their lead role in low-level waste, provided a draft report on this topic to DOE/NE in October 1986, which was subsequently issued (DOE 1987).

The GTCC wastes may include a variety of materials, such as activated metal hardware (e.g., control rods) from the reactor core; some spent fuel assembly hardware; ion exchange resins, filters, and evaporator bottoms; sealed sources from medical and industrial applications such as irradiation devices, well-logging, radiation calibration and monitoring equipment; moisture and density gauges; scrap and other wastes from manufacture of radiopharmaceuticals; and solidified liquids, scrap, trash and equipment contaminated with transuranic elements. GTCC wastes are currently generated by nuclear utilities, reactor fuel research facilities, manufacturers and users of sealed sources, and radiopharmaceutical manufacturers. In the future, decommissioning of nuclear power plants will also create some GTCC waste. Decommissioning of reactor fuel fabrication plants, test and research reactors, the West Valley Reprocessing Center, and Three Mile Island Unit 2 are also expected to generate some GTCC wastes.

5.1.2 References for Section 5.1

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

Table 5.1.1. Rules for characterizing LLW

Long Half-Life Radionuclides Nuclide	(L) Maximum concentration	
C-14	8	Ci/m^3
C-14 in activated metal	80	Ci/m^3
Ni-59 in activated metal	220	Ci/m^3
Nb-94 in activated metal	0.2	Ci/m^3
Tc-99	3	Ci/m^3
I-129	0.08	Ci/m^3
Alpha emitters with $t_{1/2}$ longer than 5 yrs	100	nCi/gm
Pu-241	3,500	nCi/gm
Cm-242	20,000	nCi/gm

Short Half-Life Radionuclides	(A)	(B)	(C)
	Ci/m^3		
All radionuclides with $t_{1/2}$ less than 5 yrs	700	-	-
H-3	40	-	-
Co-60	700	-	-
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7000
Sr-90	.04	150	7000
Cs-137	1	44	4600

Class A waste:
Concentration less than column (A) and 10% of column (L).
Class B waste:
Concentration less than column (B) but more than column (A).
Class C waste:
Concentration less than column (C) but more than column (B), or concentration 10 to 100% of column (L).
Greater than C:
Concentration exceeds column (L) or column (C).

Note: For mixtures of radionuclides, limits are obtained by a sum-of-fractions rule.

5.2 OCRWM-GENERATED WASTES

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 Federal Interim Storage (FIS) or 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.

The quantities and characteristics of the secondary wastes are dependent on the detailed design characteristics of each facility. Neither detailed designs nor prototype equipment for OCRWM facilities currently exist. As a consequence, estimates of secondary wastes are uncertain. Of the various proposed facilities, design has proceeded the furthest on the MRS (Ralph M. Parsons Co. 1987). In this design the MRS conducts consolidation and packaging operations and hence generates most of the TRU wastes to be expected from OCRWM operations. If an MRS is not authorized, it is expected that the repository surface facility operations and secondary wastes will be similar to the MRS because the same functions would be conducted at the repository: receiving, consolidation, and packaging. If consolidation is not done, this would eliminate the major source of OCRWM-generated TRU waste. Underground repository operations are not expected to generate radioactive wastes. Different combinations of MRS and repository designs are possible but the total quantities of wastes from both facilities combined should be approximately constant given that the same functions are performed. On the other hand, if consolidation is done underwater (at utility sites), it would contribute little or no additional waste.

5.2.1 MRS-Generated Waste

Table 5.2.1 shows the secondary wastes expected from the MRS, the sources of the waste, and the description. Table 5.2.2 shows waste quantities and classifications.

The waste quantity estimates for the MRS arrived at by the A-E for the volume of HEPA filters were for an upper-bound case, and consequently may be high. Note that the total volume related to HEPA filters is ~9,000 ft^3 for an MRS feed rate of 3600 MTIHM/year, or ~2.5 ft^3 per ton. This volume is based on separating the filter media from the frames and accomplishing sufficient volume reduction such that the media from six filters are placed in one 55-gal drum while the chopped frames from six filters are placed in another drum. This constitutes a relatively small volume reduction of ~40%. The fraction of maximum density that is assumed for both the media and the framing is ~0.06. It seems unlikely that the frames will be sufficiently contaminated to have an alpha content of $\geq$100 nCi/g. This would require (assuming PWR fuel irradiated to 33,000 MWD/T and decayed for 10 years) that a single filter frame (weighing 17 lb) retain ~0.27 grams heavy metal, which is an improbably large amount of dust. Hence, if the media are separated from the frames, it seems quite probable that the frames will not be TRU waste, but will be low-level waste. The filter media, however, can be expected to be TRU waste.

There appears to be little reason to believe that HEPA filters which are in a stream that is alpha-particle-contaminated will require frequent replacement. All air entering zones housing fuel assemblies will be filtered to remove dust prior to entering the radioactive area; this will assure that the filters are not loaded with dust brought in from cold areas. Because the quantity of particulates that will be generated in the shielded space will be very small, filter lifetime can be expected to be quite long - experience with an existing facility at ORNL indicates a life of five or more years for even the primary filters (Chattin 1986). HEPA filters are commonly installed with at least two and sometimes three filters in series, and since the dust load to all except the first, or primary, filter is negligible, the life expectancy of secondary and tertiary filters should be several-fold longer, or perhaps 15 to 20 years.

Therefore, because the estimated number and compacted volume (~0.07 m^3/MT) of HEPA filters constitutes a dominant and significant quantity of TRUW produced by spent fuel consolidation, an examination of possible means of waste volume reduction appears to be appropriate. In summary, reasons for suspecting that reduction in estimated volumes may be reasonable include the following:

1. It appears likely that the filter frames will qualify as LLW, and that the filter media can be reduced in volume by a factor of 2 from the value used by the A-E. Thus, on the basis of routing separated filter frames to LLW and the filter media to TRUW, the annual TRUW volume for a 3500 MT/year consolidation facility might be reduced to

$$3500 \frac{MT}{y} \times \frac{1.0 \text{ filter media}}{MT} \times \frac{\text{drum}}{12 \text{ filter media}} = 292 \text{ drums/year}$$

2. It is conceivable that the TRUW volume might be reduced below this by a factor ≥ 4 via greater compaction of the media and/or through reducing the spread of contamination released when fuel rods are broken.
3. A filter usage rate of even 1 per MT spent fuel (initial heavy metal basis) still appears excessive, based on actual experience.

5.2.2 Other Waste Sources

Operation of a cask fleet and transport system will generate a variety of wastes from decontamination, replacement of failed equipment, malfunctions, etc. These wastes would normally be LLW. They would be handled using procedures appropriate for LLW. Any hazardous or mixed waste would have to be handled by procedures appropriate for those wastes.

5.2.3 References for Section 5.2

Ralph M. Parsons Company 1987. Evaluation of the Effluent Streams From the Conceptual Design of the Monitored Retrievable Storage Facility, (June 1987).

Chattin 1986. Personal communication to A. R. Irvine, dated 1986.

Table 5.2.1. Description of Treated MRS Secondary Waste Streams

Category	Original Sources	Description
Liquid		
Oily sewer	Diesel fuel tank locations Vehicle maintenance area	Primarily off-specification #2 fuel oil
Sanitary sewer	Toilets, showers, and sinks	Liquid effluent from sewer plant
Process sewer	Laboratory cooling tower blowdown	Liquid effluent from process sewer plant
Solids		
Process sludge	Sanitary sewage	Solidified sludge from process sewer plant
Sanitary sludge	Sanitary sewage	Sludge from sanitary sewer plant
LLW drums	Decontamination operations Truck washdown Contaminated failed equipment	Cemented drums
TRUW drums (transuranic waste)	In-cell HEPA filters Radwaste drains Contaminated failed equipment	55-gal drums

Table 5.2.2. Quantities of MRS secondary waste streams

General category	Quantity	Regular category LLW	HAW	Mixed	Effluent location
Liquids					
Oily sewer	38,850 lb/year	No	No	No	Reclaim
Sanitary sewer	16,250 gpd	No	No	No	River
Process sewer	22,530 gpd	No	No	No	River
Solids					
Process sludge	12,312 lb/yr	No	No	No	Landfill
Sanitary sludge	18,250 lb/yr	No	No	No	Landfill
LLW Drums		Yes	No	[a]	LLW disposal
Solid Combustable	265 drums/yr				
Noncombustable	11 drums/yr				
Filters	249 drums/yr				
Sludge	8 drums/yr				
	533 drums/yr				
TRUW drums		No	Yes	[a]	Long-term isolation disposal
HEPA filters[b]	675 drums/yr				
Filter media[c]	675 drums/yr				
Cemented evaporator bottoms	21 drums/yr				
Cemented Ion exchange Resin	12 drums/yr				
	1383 drums/yr				

[a]Possibly mixed waste depending upon treatment option chosen

[b]May be low-level waste.

[c]Analysis suggests can be reduced by factor of 2 easily and perhaps by up to a factor of 8 with further refinements.

5.3 COMMERCIAL TRANSURANIC WASTE (TRUW)

Transuranic wastes have been produced in the course of reactor operations and during the reprocessing of spent fuels at West Valley, New York. They are expected to be produced in the course of handling spent fuels at central fuel handling facilities such as may be used for fuel consolidation either at a monitored retrievable storage facility or, alternatively, at a geologic repository site. The following subsections deal with the quantities and characteristics of the TRU waste which have issued and will issue from these sources. Transuranic wastes are also generated in nonnuclear fuel cycle operations, which include research laboratories and certain industrial operations that utilize transuranic isotopes, and from decommissioning of fuel fabrication plants.

5.3.1 Reactor Operations

The TRU waste generation characteristics of nuclear power plants can best be considered by subdividing these operational wastes into two categories - normal and abnormal operations. TRU radioisotopes generated during normal nuclear power plant operations usually remain within the UO_2 fuel pellets. TRU radionuclides may escape from the fuel pellets if the fuel cladding fails or is otherwise breached. The amount of TRU isotopes released from a failed fuel rod is usually small enough that when the nuclides are removed from the reactor coolant or spent fuel storage pool water (depending upon where the fuel failure occurs), the resulting concentrations in filter media or ion exchange resins are extremely low. This observation is based on results of radionuclide analyses of low-level waste samples from operating nuclear power plants as reported by Cline et al. (1980). TRU and other nuclide concentrations were determined in that study for several categories of LLW generated at 25 nuclear power plants. Samples from evaporator bottoms, resins, filter sludge, smears and swipes, and reactor coolant insolubles were included in that study. Their results showed that mean and median

concentrations of the nuclide of most concern, Pu-239, for each category of solid waste are significantly below 10 nCi/g. Since Cline et al. (1980) included plants with a history of significant fuel failures, these results are believed to be applicable to most plants.

Cline et al. (1980) reported that less than 4% of the wastes analyzed had plutonium concentrations greater than 10 nCi/g. Even less than 4% of the wastes would be expected to contain over 100 nCi/g (the EPA definition of TRU wastes in 40 CFR 191 and NRC's in 10 CFR 61.55). The study analyzed dried samples of waste which had not been diluted with any immobilization agent such as concrete. If these samples had been mixed with other wastes and immobilized, the plutonium concentrations would be reduced further. In addition, most of the samples that contained concentrations above 10 nCi/g were atypical; for example, three primary filter sludge samples came from a plant that had experienced recent and substantial fuel failures, and three resin samples came from a plant experiencing difficulties with its radioactive waste evaporation and fuel pool cleanup systems.

The Hazelton survey supports the preceding conclusions with a few exceptions (Hazelton 1983). Hazelton surveyed 12 nuclear utilities representing 28 nuclear power plants, and only two plants reported that they were certain they had some TRU wastes. These two plants had abnormal operations and will be discussed below.

Abnormal operations at Three Mile Island Unit 2 (TMI-2) caused the core to be transformed to rubble. In the course of cleanup, no TRU waste was produced through August 1986, but some is expected to be produced later (Hahn 1986). The projected production of TRUW is ~8.5 m^3 (DOE 1985).

High fuel failure rates during the early operation of Oyster Creek Unit 1 resulted in the production of some wastes known to contain sufficient amounts of transuranic to exceed the limits of LLW, and some metal waste of unknown alpha activity level. The characteristics of these wastes are presented in Table 5.3.1.

The information on TRU waste produced by abnormally operating reactors is a very limited base from which to predict the amount of TRU waste which will be produced per TWeD of power production. This can be done in the absence of better data, but is of value mainly in placing an upper bound on the production rate. This is a reasonable interpretation because nuclear power generation is in the early stages, and deficiencies in design are still being identified. Hence, it seems reasonable that the recurrence rate of problems previously encountered will become lower in the future.

However, if we use the quantity generated per TWeD that can be calculated as of June 30, 1986, we obtain the following values:

Source	Volume, m^3/TWeD
TMI-2 type failure	0.06
Other abnormal operations	0.47-1.25

The total U.S. commercial nuclear power production as of June 30, 1986 was ~153 TWeD (from Nucleonics Week). The quantity of TRUW generated at Oyster Creek Unit 1 was multiplied by three (included in the above values) in an attempt to allow for other abnormal reactor operations which were not covered in Hazelton's survey. This is not to indicate a known incident which definitely produced TRU waste, but is simply to allow a margin for occurrences undiscovered by the survey. It appears that the above production rates are conservative (high), but this is yet to be confirmed or quantified.

5.3.2 Fuel Reprocessing Facilities

The West Valley Demonstration Project (WVDP) will demonstrate the retrieval, solidification, storage, transport, and disposal of liquid high-level wastes (HLW) currently being stored in tanks at the Western New York Nuclear Service Center (West Valley). Transuranic wastes will

be generated in all phases of the demonstration, including pre-solidification (initial decontamination), solidification (HLW processing), and postsolidification (final decontamination and decommissioning) phases. Although this project and the waste generated are being handled by DOE, it is a product that results from commercial activities. It is anticipated that vitrified West Valley HLW will be disposed of in a commercial repository.

The latest estimates of TRU waste from West Valley decommissioning are far lower - by a factor of 7 - than estimates prepared earlier (Bixby 1987). The latest data give 23 m^3 as of December 31, 1986, and project a total of 300 m^3 upon completion of the project. This is much less than earlier estimates, described in detail below, of 2000 m^3 (71,400 ft^3).

Earlier estimates of TRU waste generated by WVDP were extracted from a draft preconceptual design report and from Cwynar et al. (1984). These reports indicated that a total of approximately 70,000 ft^3 (packaged waste volume) of TRU wastes will be generated. About 60% of these wastes will be generated during final decontamination and decommissioning operations, 30% during initial decontamination, and the remainder during HLW processing operations.

The bulk of the TRU wastes will be treated in some manner prior to final packaging. Current plans are to compact LLW general rubbish and trash into 1.9- x 1.2- x 1.3-m rectangular steel boxes (Cwynar et al. 1984). TRU wastes are not planned to be compacted because of the contamination risk. It is planned that liquid wastes, such as decontamination solutions, will be filtered, dewatered (through evaporation), demineralized, and solidified in cement (Cwynar 1984). The WVDP plans to use a fluidized bed dryer (FBD) to process evaporator concentrates and processed neutralized decontamination solutions. The dried wastes

from the FBD, as well as filter sludges and spent ion exchange resins, will be solidified with cement and cast into 55-gal drums. Metallic wastes, such as contaminated chemical processing equipment and hardware, failed HLW processing equipment and hardware, and waste retrieval equipment, will be decontaminated and dismantled to the extent possible and cut where needed to sizes that will fit within the steel box mentioned above. Contaminated rubble from building demolition during the post-solidification phase will be packaged without treatment in two metal box sizes (1.9 x 1.2 x 1.3 m and 1.7 x 1.4 x 1.0, respectively). Waste generation estimates are summarized in Table 5.3.2.

The physical properties of the treated TRU wastes are uncertain at this time. However, the preconceptual design report mentioned previously assumed the following densities for the various waste forms:

- Wastes solidified in binder (cement) - 1600 kg/m^3
- Compacted wastes - 480 kg/m^3
- Contaminated rubble - 800 kg/m^3
- Contaminated equipment/hardware - 320 kg/m^3.

Radioisotope distributions are not available for these wastes. However, estimates of the surface dose rates of the various waste containers have been developed. The maximum surface dose rate is 50 R/hr for solidification resins and filters. It is estimated that over 50% of the 55-gal drums, 50% of the liners, and none of the boxes will be RH-TRU wastes (i.e., surface dose rate exceeds 200 mR/hr). The remainder will be CH-TRU wastes.

5.3.3 Other Sources

Two other sources of commercial TRU waste have been identified (Daling 1986):

- Industrial/institutional sources, including commercial research laboratories, and other industrial users of TRU isotopes; and
- Future decommissioning of commercial fuel fabrication plants.

As of December 1986, commercial research labs have generated 13 m^3, while other industrial users have generated 28 m^3. Projected future rates are less than 1 m^3/year and 10-40 m^3/year, respectively.

Projected plant decommissioning activites have been estimated to produce about 680 m^3 total TRU waste.

5.3.4 Volume for Eventual Disposal

The TRUW volumes from commerical sources can be divided into two categories — that which currently exists and that which is projected. That which currently exists has resulted mostly from fuel failures at nuclear reactors. Future generation of TRUW can be expected from this same source plus from WVDP, fuel rod consolidation operations, industrial/institutional operations, and from D&D of fuel fabrication plants. The estimated waste volumes from these various sources is shown in Table 5.3.3.

5.3.5 References for Section 5.3

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Table 5.3.1. TRU waste quantities generated at Oyster Creek Nuclear Power Plant

Waste description	Waste treatment process	Waste volume, ft^3	Waste container	Number of containers produced
Filters	Shredding and solidification	420	Drum	57
Sludges	Evaporation and solidification	430	Drum	58
Metals[a]	Cutting and sectioning	1400	Drum	390[b]

[a]Metal waste is irradiated equipment for which activity is unknown. May or may not be TRU waste.

[b]Wastes are assumed to be packaged in 55-gal drums. A 50% packing fraction was assumed to calculate the number of containers produced.

Table 5.3.2 Summary of TRU waste quantities projected for the West Valley Demonstration Project (WVDP)[a]

WVDP Phase	Waste description	Treatment process	Waste volume, ft^3
Pre-solidification	Rubbish, trash	Compaction	2,000
	Spent resins and filters	Dried and solidified	200
	Decontamination solutions	Dried and solidified	6,000
	Contaminated equipment/hardware	Dismantle and cut	14,500
Solidification	Rubbish, trash	Compacted	5,500
	Spent resins and filters	Dried and solidified	500
	Decontamination solutions	Dried and solidified	200
	Failed equipment	Dismantle and cut	500
Post-solidification	Rubbish, trash	Compaction	4,000
	Spent resins and filters	Dried and solidified	200
	Decontamination solutions	Dried and solidified	7,800
	Contaminated equipment/hardware	Dismantle and cut	16,000
	Waste retrieval equipment	Dismantle and cut	2,500
	Contaminated rubble	None	11,500
Totals			71,400

[a]Data from Cwynar (1984). Later data by Bixby (1987) project a much smaller total of 300 m^3 (10,500 ft^3).

Table 5.3.3. Estimated TRUW Volume

Source	Quantity (m^3)		
	Existing	Expected	Projected
Reactor operations	70-200	-	10-30/y[a]
West Valley Demonstration Project	23	300	-
Industrial/institutional	41	-	10-40/y
Fuel fabrication plant D&D	-	680	-

[a]Assuming nationwide nuclear power production of 22 TWeD/yr and waste volume = 0.5 to 1.25 m^3/TWed.

5.4 REACTOR DECOMMISSIONING

5.4.1. Introduction

Power reactors must be decommissioned at the end of their useful life. This may be accomplished in one of three ways defined by the U.S. Nuclear Regulatory Commission (NRC) as:

SAFSTOR - The nuclear facility is placed and maintained so that it can be safely stored and subsequently decontaminated to levels that permit unrestricted use. The International Atomic Energy Agency (IAEA) calls this approach Stage 1.

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 levels permitting unrestricted use of property (or until DECON is undertaken). The IAEA calls this approach Stage 2.

DECON - The portions of the facility containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after completion of these operations. The IAEA calls this approach Stage 3.

Public Law 99-240, "The Low-Level Radioactive Waste Policy Amendment Act of 1985," effective 1/15/86, made the U.S. Department of Energy (DOE) responsible for "any other low-level radioactive waste (LLW) with concentrations of radionuclides that exceed the limits established by the Nuclear Regulatory Commission for Class C", also called greater-than-class-C (GTCC). This law is acting as a stimulus to the resolution of such issues as:

- The volume and characteristics of all waste expected to be GTCC.
- The options for disposal of GTCC.
- The need for more regulatory specificity.

The required DOE response to Public Law 99-240 (DOE 1987) covers the status of and actions needed to ensure the safe disposal of GTCC LLW. If new facilities are required it will be some time before such disposal is possible. If existing facilities or facilities under development can be used, disposal services may be provided sooner.

Briggs et al. (1978) recommend delaying DECON at a dedicated site for 50 to 100 years to greatly reduce the volume of LLW and the associated risks

of handling and shipping the radioactive materials by allowing decay of Co-60 (half-life 5.3 years) and other radionuclides.

5.4.2 Shippingport Data

DOE has an active program for developing the technology needed for decommissioning nuclear facilities (Delaney 1986). The Surplus Facilities Management Program includes the ongoing DECON of the Shippingport Atomic Power Station. Table 5.4.1. presents current data and projections from that activity (Schreiber 1987). To date, indications are that essentially all activated metals from Shippingport are to be shipped with the "Reactor Vessel Package" which will be barged to Hanford for disposal as LLW by shallow land burial.

5.4.3 Projected Data

The characterization of the wastes generated during LWR reactor decommissioning is addressed in many reports (ORNL 1986, Forsberg 1985, Luksic 1986, Murphy 1984, Oztunali 1986). There are significant uncertainties in any projection for decommissioning wastes because of such variables as the size of reactor, exact metal composition, neutron flux, and time of exposure. Luksic (1986) calculated that after 40 years of full-power operation for a reference PWR only the core shroud would be GTCC, only the ^{94}Nb would exceed the Class C individual radioisotope limit, and the sum of fractions of all reported isotopes would be about twice the Class C (upper) limit. Forsberg (1985) presented data from calculations that showed:

- The PWR core shroud exceeded the individual Class C limits for ^{94}Nb and ^{63}Ni, and the sum-of-fraction of all reported isotopes was about five times the Class C limits.
- The BWR core shroud sum-of-fraction of all reported isotopes was 1.15 times greater than the Class C limit, although no individual radioisotope exceeded Class C limits.
- The PWR lower grid plate, the next most radioactive item, had a sum-of-fractions of all reported radioisotopes of about 70% of Class C upper limits. These data are shown in Table 5.4.2. It is clear from the data that it will be necessary to calculate each specific case to determine which, if any, component is GTCC.

Table 5.4.3. shows that the great preponderance of radioactivity is contained in the activated metal, which is a small portion of the total

volume of waste generated. Table 5.4.4. shows that most of the radioactivity in the activated metals is contained in the core shroud.

Table 5.4.5. is derived by using the following information from Oztunali (1986): (a) power reactors listed in Table 5.4.6. that are eligible for decommissioning after forty years of operation and (b) the core shroud waste volume shown in Table 5.4.4. Additional assumptions include:

- only the core shroud is considered as GTCC solid waste;
- all reactor core shrouds have the same volume as the same type reference reactor (even when they are considerably smaller in MWe rating);
- the waste is disposed of in equal and rounded increments over a 4-year period 3 years following reactor shutdown. For example, Peach Bottom 2 is scheduled to be decommissioned in 2013. Three years are allowed for licensing and engineering preparation. The 47 m^3 of core shroud are assumed to be available for disposal in 12 m^3 batches in 2016, 2017, 2018, and 2019 respectively.

The projection indicates there would be 1560 m^3 of GTCC wastes generated through 2020 and that the largest annual volume, in 2018, would be 228 m^3.

5.4.4 References For Section 5.4.

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Table 5.4.1. Characteristics of Wastes from the Shippingport Station Decommissioning Project[a]

Type of Waste	Total waste removed from Shippingport through December 1986[b] Waste volume (m^3)	Activity (Ci)	Projected waste vol. remaining (m^3)
Liquid Wastes	1,643	.6	341
Solid Wastes[c]			
Reactor Vessel Package	-	-	289
Resins	-	-	43
Asbestos	1,010	.5	0
Compacted Trash	24	.1	25
Irradiated Steel	9	1.2	0
Metallic Waste	521	6.4	1,497
Total	3,207	18.8	2,195

a. Source: Schreiber 1987.

b. During its history, the Shippingport reactor operated with three different cores. The last two of these were light-water cooled, seed-blanket LWBR-type. Physical dismantling began in September 1985 and is expected to be complete by July 1989.

c. Solid waste volume is total volume as packaged.

Table 5.4.2. Volume and Activity of Activated Metals from Power Reactor Decommissioning

Waste Type	Volume, m^3		Specific Activity, Ci/m^3				
	Burial	Minimum	C-14	Ni-59	Nb-94	Ni-63	Co-60
PWR[a]							
Vessel	114	17.4	8.52 E-3	1.59 E-2	1.28 E-4	1.70	-
Barrel	108	6.7	1.92	4.27	1.26 E-2	4.56 E+2	-
Shroud	11	1.6	23.80	51.50	0.247	5.50 E+3	-
Total	233						
PWR[b]							
Lower core barrel	91	5.4	-	2.0	1.4 E-2	3.2 E+2	2.6 E+3
Thermal shield	17	1.3	-	2.4	1.7 E-2	3.8 E+2	3.1 E+3
Core shroud	11	1.6	-	87.0	0.63	1.4 E+4	1.1 E+5
Lower grid plate	14	0.5	-	11.0	7.9 E-2	1.8 E+3	1.4 E+4
Total	133						
BWR[b]							
Core shroud	47	4.1	5.0	30.0	7.1 E-2	4.2 E+3	1.6 E+4
Lower Limits GTCC	-	-	80.0	220.0	0.20	7.0 E+3	-

a. Data from Luksic 1986

b. Data from Forsberg 1985

Table 5.4.3. Summary of Wastes from Decommissioning a Reference PWR and a Reference BWR[a]

Waste Stream	Volume (m^3)	Activity (Ci)
Reference 1155 MW(e) BWR:		
Activated metal[b]	138	6,552,310
Activated concrete[b]	90	170
Contaminated metal[b]	14,629	8,400
Contaminated concrete[b]	2,600	144
Dry solid waste (trash)[c]	3,390	1,806
Spent resins[d]	42	228
Filter cartridges[e]	--	--
Evaporator bottoms[f]	438	32,753
Reference 1175 MW(e) PWR:		
Activated metal[b]	485	4,840,820
Activated concrete[b]	707	2,000
Contaminated metal[b]	5,465	900
Contaminated concrete[b]	10,613	100
Dry solid waste (trash)[c]	1,415	757
Spent resins[d]	30	42,000
Filter cartridges[e]	9	5,000
Evaporator bottoms[f]	133	13,805

a Source: Oztunali 1986 (Table A26).

b As-packaged volumes.

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

d BWR spent resins actually include spent resins and filter sludge. Volumes shown are dewatered volumes.

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

f PWR and BWR evaporator bottom volumes are given as generated and prior to solidification.

Table 5.4.4. Volumes and Activities of Decommissioned LWR Activated Metals[a]

Component	Constr.[b] Material	Disposal Volume[c] (m^3)	Activity (Ci)	Disposal Concentration (Ci/m^3)
Reference BWR:				
Steam separator assembly	S	10	9,600	960
Fuel support pieces	S	5	700	140
Control rods and in-core instruments	S	15	189,000	12,600
Control rod guide tubes	S	4	100	25
Jet pump assemblies	S	14	20,000	1,429
Top fuel guide	S	24	30,000	1,254
Core support plate	S	11	650	59
Core shroud	S	47	6,300,000	134,043
Reactor vessel wall	C	8	2,160	270
Total		138	6,552,310	
Reference PWR:				
Pressure vessel cylindrical wall	C	108	19,170	178
Vessel head	C	57	<10	0.18
Vessel bottom	C	57	<10	0.18
Upper core support assembly	S	11	<10	0.91
Upper support columns	S	11	<100	9.1
Upper core barrel	S	6	<1,000	167
Upper core grid plate	S	14	24,310	1,736
Guide tubes	S	17	<100	6
Lower core barrel	S	91	651,000	7,154
Thermal shields	S	17	146,000	8,594
Core shroud	S	11	3,431,100	311,909
Lower grid plate	S	14	553,400	39,529
Lower support columns	S	3	10,000	333
Lower core forging	S	31	2,500	81
Miscellaneous internals	S	23	2,000	87
Reactor cavity liner	S	14	<10	0.7
Total		485	4,840,820	

a Source: Oztunali 1986 (Table A27).

b Construction material symbols: S = stainless steel, C = carbon steel.

c Disposal volumes include the disposal container after the activated metal components have been cut into manageable pieces.

Table 5.4.5. Projected Volume of Greater Than Class C Wastes From Decommissioning Reactors Power

Reactors Shutdown[a]			Volume (m^3)[b]	
Year	BWR	PWR	Yearly	Total
1997	0	1	0	0
8	0	0	0	0
9	1	0	0	0
2000	0	1	3	3
1	0	0	3	6
2	1	1	15	21
3	1	0	18	39
4	0	0	15	54
2005	0	0	30	84
6	0	0	30	114
7	1	2	27	141
8	0	0	27	168
9	2	1	12	180
2010	3	2	18	198
11	1	1	18	216
12	3	4	45	261
13	2	9	87	348
14	7	8	84	432
2015[c]	1	3	132	564
16	2	4	156	720
17	0	3	222	942
18	1	2	228	1170
19	0	1	216	1386
2020	0	2	174	1560

a For a list of reactors assumed to be shut down, see Table 5.4.6.

b Assumes the waste is disposed of in 4 equal yearly portions such that the PWR shutdown in 1997 has 11 m^3 (Table 5.4.3) disposed as 3 m^3 in 2001, 2002, 2003, and 2004, and the BWR shutdown in 1999 has 47 m^3 (Table 5.4.3) disposed of as 12 m^3 in 2002, 2003, 2004, and 2005.

c Projected volumes through 2020 include: all shutdowns by year 2014; 3/4 of the shutdowns in 2015; 1/2 of the shutdowns in 2016; 1/4 of the shutdowns in 2018; and none of the shutdowns in 2018, 2019, or 2020.

Table 5.4.6. Power Reactors Potentially Eligible for Decommissioning After Forty Years Operation[a]

Name	State Located	Type	Power MW(e)	Power MW(t)	Theoretical Start-up	Theoretical Shutdown
Shippingport	PA	PWR	60	236	1957	1997
Dresden 1	IL	BWR	200	700	1959	1999
Yankee Rowe	MA	PWR	175	600	1960	2000
Indian Point 1	NY	PWR	265	615	1962	2002
Big Rock Point	MI	BWR	72	240	1962	2002
Humboldt Bay	CA	BWR	65	242	1963	2003
Haddam Neck	CN	PWR	575	1825	1967	2007
LaCrosse	WI	BWR	50	165	1967	2007
San Onofre	CA	PWR	436	1347	1967	2007
Oyster Creek	NJ	BWR	650	1930	1969	2009
Nine Mile Point 1	NY	BWR	620	1850	1969	2009
R. E. Ginna 1	NY	PWR	470	1520	1969	2009
Millstone 1	CN	BWR	660	2011	1970	2010
H. B. Robinson	SC	PWR	700	2200	1970	2010
Dresden 2	IL	BWR	794	2527	1970	2010
Monticello	MN	BWR	545	1670	1970	2010
Point Beach 1	WI	PWR	497	1518	1970	2010
Dresden 3	IL	BWR	794	2527	1971	2011
Palisades	MI	PWR	805	2530	1971	2011
Maine Yankee	ME	PWR	825	2630	1972	2012
Vermont Yankee	VT	BWR	514	1593	1972	2012
Surry 1	VA	PWR	822	2441	1972	2012
Turkey Point 3	FL	PWR	693	2200	1972	2012
Point Beach 2	WI	PWR	497	1518	1972	2012
Quad-Cities 1	IL	BWR	789	2511	1972	2012
Quad-Cities 2	IL	BWR	789	2511	1972	2012
Indian Point 2	NY	PWR	873	2758	1973	2013
Peach Bottom 2	PA	BWR	1065	3293	1973	2013
Browns Ferry 1	AL	BWR	1065	3293	1973	2013
Oconee 1	SC	PWR	887	2568	1973	2013
Oconee 2	SC	PWR	887	2568	1973	2013
Surry 2	VA	PWR	822	2441	1973	2013
Turey Point 4	FL	PWR	693	2200	1973	2013
Prairie Island 1	MN	PWR	530	1650	1973	2013
Zion 1	IL	PWR	1040	3250	1973	2013
Zion 2	IL	PWR	1040	3250	1973	2013
Ft. Calhoun	NE	PWR	457	1500	1973	2013
Calvert Cliffs 1	MD	PWR	845	2700	1974	2014
Calvert Cliffs 2	MD	PWR	845	2700	1974	2014
J. A. Fitzpatrick	NY	BWR	821	2436	1974	2014

Table 5.4.6. (continued)

Name	State Located	Type	Power MW(e)	Power MW(t)	Theoretical Start-up	Theoretical Shutdown
Peach Bottom 3	PA	BWR	1065	3293	1974	2014
Pilgrim 1	MA	BWR	665	1998	1974	2014
Three Mile Island 1	PA	PWR	819	2535	1974	2014
Browns Ferry 2	AL	BWR	1065	3293	1974	2014
E. I. Hatch 1	GA	BWR	786	2436	1974	2014
Oconee 3	SC	PWR	887	2568	1974	2014
Duane Arnold 1	IA	BWR	538	1658	1974	2014
Kewaunee	WI	PWR	535	1650	1974	2014
Prairie Island 2	MN	PWR	530	1650	1974	2014
Arkansas 1	AR	PWR	850	2568	1974	2014
Cooper	NE	BWR	778	2381	1974	2014
Ft. St. Vrain	CO	HTGR	330	842	1974	2014
Rancho Seco 1	CA	PWR	918	2772	1974	2014
Millstone 2	CT	PWR	870	2700	1975	2015
Brunswick 2	NC	BWR	821	2436	1975	2015
D. C. Cook 1	MI	PWR	1054	3250	1975	2015
Trojan 1	OR	PWR	1130	3411	1975	2015
Beaver Valley 1	PA	PWR	852	2660	1976	2016
Indian Point 3	NY	PWR	965	3025	1976	2016
Salem 1	NJ	PWR	1090	3338	1976	2016
Browns Ferry 3	AL	BWR	1065	3293	1976	2016
Brunswick 1	NC	BWR	821	2436	1976	2016
St. Lucie 1	FL	PWR	802	2700	1976	2016
Crystal River 3	FL	PWR	825	2544	1977	2017
J. M. Farley 1	AL	PWR	829	2652	1977	2017
Davis-Besse 1	OH	PWR	906	2772	1977	2017
E. I. Hatch 2	GA	BWR	784	2436	1978	2018
North Anna 1	VA	PWR	907	2775	1978	2018
D. C. Cook 2	MI	PWR	1100	3391	1978	2018
Three Mile Island 2	PA	PWR	906	2772	1979	2019
North Anna 2	VA	PWR	907	2775	1980	2020
Arkansas 2	AR	PWR	912	2815	1980	2020

a Source: Oztunali 1986 (Table A36).

5.5 RADIOISOTOPE CAPSULES

5.5.1 Introduction

DOE distributes a number of radioisotopes to various organizations and firms for medical purposes, research, and industrial applications. The major radioisotopes that have been distributed thus far in industrial quantities are ^{90}Sr, ^{137}Cs, ^{60}Co, ^{241}Am, and ^{252}Cf. Other radioisotopes that may be significant from the standpoint of quantities produced or potential disposal needs are ^{99}Tc, ^{153}Gd, ^{151}Eu, ^{152}Eu, ^{85}Kr, and tritium (H-3). It is not clear which of these radioisotopes, if any, will require future repository disposal. In many cases, the half-lives are short enough so that interim storage will reduce the activity to a point where disposal as low-level waste becomes feasible. Thus, the discussion of particular radioisotopes in this section is not meant to imply that they will necessarily require repository disposal. The information given here is intended to facilitate decisions as to whether such disposal will ultimately become necessary or can be avoided by interim storage and disposal as low-level waste.

Table 5.5.1, which is included for reference purposes, gives the half-lives and other data on the radioisotopes discussed in this section.

5.5.2 Sr-90 and Cs-137 Capsules

The defense waste at Hanford has been processed to separate the bulk of the strontium and cesium in concentrated form from the remainder of the HLW. The ^{90}Sr (half-life 28.5 years) and ^{137}Cs (half-life 30.1 years) have been converted to solid forms as strontium fluoride and cesium chloride and placed in double-walled capsules for storage in water basins. Each capsule has an external diameter of about 6.7 cm and an overall length of about 53 cm. Detailed dimensional data are shown in Fig. 5.5.1. Separation and encapsulation were completed during 1984 and resulted in a total of 640 ^{90}Sr capsules and 1576 ^{137}Cs capsules (White 1986). The capsules include the short-lived daughter isotopes ^{90}Y and ^{137m}Ba that are in transient equilibrium with the parent radionuclides.

5.5.2.1 Total Radioactivity of Capsules

The data used here for total radioactivity are from the 1986 IDB (Integrated Data Base) submittals from Hanford (Wilde 1986a, b). At the end of 1985, the 640 strontium capsules had a total radioactivity of 6.6E+7 Ci, and the 1576 cesium capsules had a total radioactivity of 1.46E+8 Ci. Thus, at the end of 1985, each ^{90}Sr capsule had an average radioactivity of 1.03×10^5 Ci, and each ^{137}Cs capsule had an average radioactivity of 9.27×10^4 Ci. The radioactivities stated here include those of the daughter isotopes. Data given here on the number of capsules, as well as on the strontium and cesium curie values, represent revisions from earlier figures. The capsule count is slightly lower and is based on actual production data; no further production of capsules is currently planned. The ^{90}Sr curie figures are slightly lower and the ^{137}Cs curie value is about 12% higher than in the previous (1985) IDB data submittal. These changes are based on the use of actual production figures for the strontium and on a reassessment of calorimeter calibration data for the cesium capsules (Wilde 1986a).

5.5.2.2 Distribution of Strontium and Cesium Capsules

At the end of 1985, most of the SrF_2 and CsCl capsules produced at Hanford were still located at the capsule storage basins in the Hanford 200 East Area; some capsules had already been transferred to other locations for various useful purposes. As of October 1, 1985, 33 strontium capsules and 683 cesium capsules had been transferred. Most of these were shipped offsite, and are now located (or are to be located) at various places such as France; England; Federal Republic of Germany; Oak Ridge National Laboratory; Denver, Colorado (IOTECH, Inc.); New Mexico; and Westerville, Ohio (Radiation Sterilizers, Inc.). A number are also located at the Hanford 300 Area (Pacific Northwest Laboratory).

Arrangements have been made to lease approximately 400 of the cesium capsules to four lessees for a period of 10 years. The names of

the lessees and the quantities of cesium leased are given in Table 5.5.2. In addition, about 8 x 10^6 Ci of cesium capsules have been shipped to the Federal Republic of Germany. DOE has also set aside another 17 x 10^6 Ci for demonstrations of food irradiation use (Wolfe 1986).

Table 5.5.3 gives the current locations of the strontium and cesium capsules produced at Hanford (White 1986).

5.5.2.3 Disposal of Sr-Cs Capsules

Currently, DOE is planning to retain ownership of the capsules. At this time, the reference plan is for most of the shipped cesium capsules to be returned to Hanford, processed by overpacking into canisters, and shipped to a geologic repository. However, some of the cesium capsules have been or will be converted to other forms (that is, cut open and subdivided) for by-product utilization. In these cases, the capsules will not be returned to Hanford for processing to a geologic repository (quoted from White 1986). The strontium capsules also will be overpacked in canisters for repository disposal.

5.5.2.4 Canister Dimensions

Figure 5.5.2 shows two proposed designs for overpack canisters for strontium and cesium capsules (White, 1986). The design currently proposed in the Hanford reference plan is the thin-wall canister, which uses a carbon steel container with an outside diameter of 0.3 m (about 12 in.) and an overall length of 2.7 m (about 9 ft). An internal rack supports several (3 or 4) strontium or cesium capsules along the axis of the canister so that the decay heat is distributed over the entire area of the canister, thus avoiding excessive heat fluxes to the surrounding medium. This proposed canister design has not been finalized and is open to future changes. At present there are no plans to use the massive cast steel overpack (Watrous 1986).

5.5.2.5 Number of Canisters Produced

As stated previously, a total of 640 strontium and 1576 cesium capsules have been produced. The number of canisters that will be required depends on the heat content of the capsules at the time of filling and the thermal limit, which is based on geologic repository heat load limit. The assumed thermal limits are 1.17 kW/canister for strontium and 0.8 kW/canister for cesium; these figures include the daughter isotopes. Based on the 1986 IDB submittal (Wilde 1986a), the total thermal power of the capsules in January 1995 will be 180 kW for the 640 strontium capsules and 293 kW for the 1576 cesium capsules, including the daughter isotopes in each case. The daughter isotopes Y-90 and Ba-137m account for about 83% of the heat load for a strontium capsule and about 77% of the heat load for a cesium capsule.

Assuming that capsule overpacking commences in 1995, and assuming further that the strontium and cesium capsules will not be intermingled and that a given canister must contain an integral number of capsules, then, to stay within the heat load limits stated, a strontium canister will contain four strontium capsules and a cesium canister will contain four cesium capsules. This results in a total of 160 strontium canisters and 394 cesium canisters.

At the time the capsules are packaged into canisters, the number of canisters required will be somewhat less (up to 10%) than these totals due to conversion to other forms for by-product utilization. The date that the capsules will be placed in the canisters varies from 1995 to 2020 depending on when the capsules that have been shipped are returned to Hanford. A later date will result in a reduction in the total number of canisters due to decay of strontium and cesium (quoted from White, 1986).

5.5.2.6 Radioactivity and Thermal Power of Canisters

ORIGEN2 decay calculations were made to determine the radioactivity and decay heat per canister for decay times (from 1985) of up to 10^6

years. For calculation of curies and watts per canister, it was assumed that a strontium canister contains four capsules and a cesium canister contains four capsules, as already noted. The results, given in Table 5.5.4, show that activity and heat generation are negligible at decay times of 1000 years or more. The activities and heat loads given in the table include the daughter isotopes, and decay time is based on December 1985 as the starting time. The initial activities per capsule for this table were based on 1986 IDB data, that is, 1.03E+5 curies per strontium capsule and 9.27E+4 curies per cesium capsule as of December 1985. The detailed ORIGEN2 outputs are shown in Tables 5.5.5 and 5.5.6 for strontium and cesium capsules, respectively; these are the data on which Table 5.5.3 was based.

5.5.3 Cobalt-60

Cobalt-60 (half-life 5.27 years) 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/yr); DOE supplies about 2 million curies/yr. Approximately 100 million curies are currently in use, and this quantity is increasing. A supply of about 10 million to 12 million curies/yr is required to replace the amount decaying each year. In view of its commercial usefulness and short half-life, it is not clear whether disposal of Co-60 in a repository will ever be required; however, it is mentioned here for the sake of completeness (Ottinger 1986).

5.5.4 Americium-241

Americium-241 (half-life 432 years) is used commercially for oil-well logging and for the production of smoke alarms. The first of these uses accounts 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. Oil-well logging requires larger amounts, and these

sources are licensed by NRC. The only distributor of Am-241 in the U.S. is Oak Ridge National Laboratory. Thus far, about 8-10 kg have been distributed. Am-241 emits an alpha with a soft gamma and is ordinarily used in oxide form. The recent reduction in oil-well drilling has caused a decline in the annual quantity sold, and there is currently an inventory of about 3 kg at Oak Ridge National Laboratory (Ottinger 1986).

5.5.5 Californium-252

Californium-252 (half-life 2.62 years) is currently being produced at a rate of ~500 mg/yr 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, ^{252}Cf is unique in providing a highly concentrated and extremely reliable neutron spectrum from a very small assembly. Over the past 30 years, ^{252}Cf has been applied with great success to cancer therapy, neutron radiography of objects ranging from flowers to entire aircraft, startup sources for nuclear reactors, fission activation for QA of all commercial nuclear fuel, and many other beneficial uses.

Californium-252 decays to a long-lived daughter actinide, Cm-248, which is useful for research purposes.

5.5.5.1 Major Users of Cf-252

DOE provides ^{252}Cf to four major categories of users: sales to commercial customers, loans to DOE facilities and DOE contractors, loans to other government agencies, and loans to educational and medical institutions.

Commercial Sales: Bulk ^{252}Cf is sold to commercial encapsulators at a fixed price per microgram, plus handling and packaging costs, FOB Savannah River Plant. Since 1970, the fixed price has been \$10/μg. The handling and packaging costs provide full cost-recovery for these operations, which have averaged \$52,000/yr over the past four years. Recent

commercial sales are tabulated in Table 5.5.7. A breakdown of ^{252}Cf sales in terms of final application is given in Table 5.5.8. About half the sources (and half of the contained ^{252}Cf) are for reactor startup. Fuel rod scanners and activation analysis are the next biggest segments of the sales market.

Loans to DOE sites and DOE contractors: There is no charge for the ^{252}Cf nor for normal operating costs. The requestor is responsible for the extra costs incurred in fabricating special sources or packaging used sources and for transportation.

Loans to Other Government Agencies: Present policy is to make no charge for the ^{252}Cf, which will eventually be returned for recovery of ^{248}Cm. The requestor is responsible for the direct costs incurred in preparing a source for shipment and for transportation costs FOB Savannah River Plant or Oak Ridge National Laboratory.

Loans to Educational and Medical Institutions: No charge is made for the ^{252}Cf, which will eventually be returned, nor for handling and encapsulation. The requestor is responsible only for transportation costs.

5.5.6 Technicium-99

Technicium-99 (half-life 213,000 years) is mainly used for research purposes and is currently being sold at the rate of about 20 g/year. Several kg of this isotope have been sold over the years, and an inventory of about 1 kg is currently on hand at ORNL (Ottinger 1986).

5.5.7 Gd-153, Eu-151, and Eu-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. Currently ORNL is planning to expand its production of Gd-153 to a rate of several thousand curies per year. The isotope is produced by the irradiation of natural europium, followed by separation and die-pressing

into 1-curie 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. Eu-151 is stable, but Eu-152 (half-life about 13.6 years) emits a hard gamma, and the mixed europium isotopes will therefore have to be disposed of as high-level waste (Ottinger 1986).

5.5.8 Krypton-85 and Tritium

Krypton-85 and tritium (half-lives 10.72 and 12.35 years respectively) have been widely distributed for commercial uses such as leak testing and manufacture of luminous dials. ORNL distributes about 5000-6000 Ci/year of Kr-85. This quantity, although appreciable, is small compared to the amount produced annually by commercial power reactor operation. Kr-85 is also produced during the reprocessing of reactor fuels at defense sites (Ottinger 1986).

5.5.9 Reporting of Radioisotope Shipments

A report is prepared annually by PNL summarizing the radioisotope shipments for that year, giving the names of the customers and the amounts shipped (Baker 1985). This annual document is prepared for the Office of Health and Environmental Research (ER-73), Office of Energy Research, DOE, and lists 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 is 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;
- 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.

5.5.10 References for Section 5.5

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

Coony 1987. F. M. Coony, Rockwell Hanford, submission of Hanford HLW data to IDB, March 1987.

Ottinger 1986. C. Ottinger, Oak Ridge National Laboratory, private communication to R. Salmon, dated November 17, 1986.

Watrous 1986. Telephone conversation between R. Watrous (Rockwell Hanford) and R. Salmon (ORNL), dated July 23, 1986.

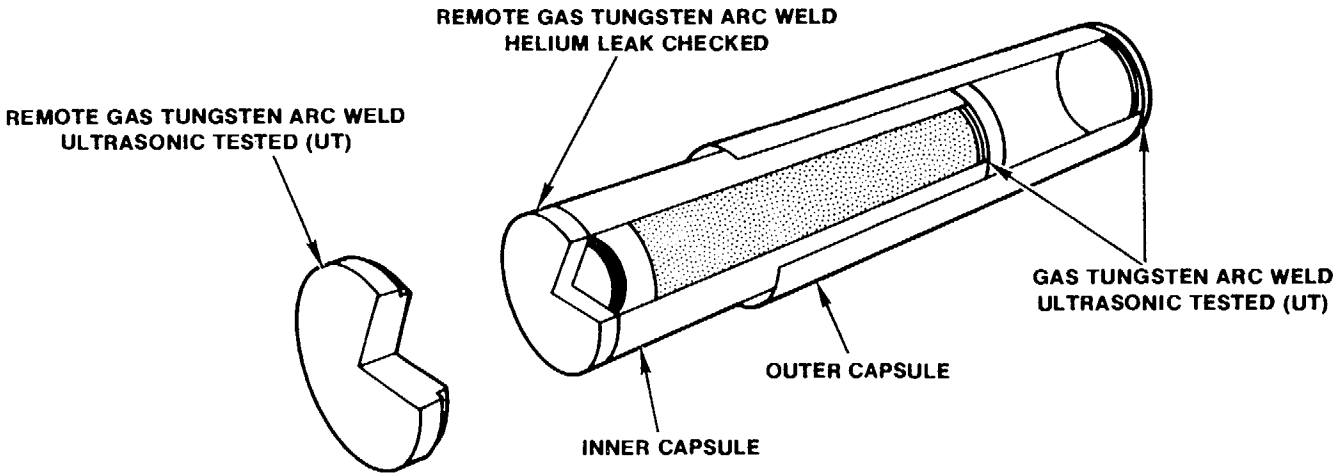
White 1986. Letter from J. D. White, Richland Operations Office, to W. R. Bibb, Oak Ridge Operations Office, dated July 3, 1986.

Wilde 1986a. Letter from R. T. Wilde, Rockwell Hanford, to M. W. Shupe, Richland Operations Office, dated April 11, 1986.

Wilde 1986b. Letter from R. T. Wilde, Rockwell Hanford, to Herschel W. Godbee, ORNL, dated May 5, 1986.

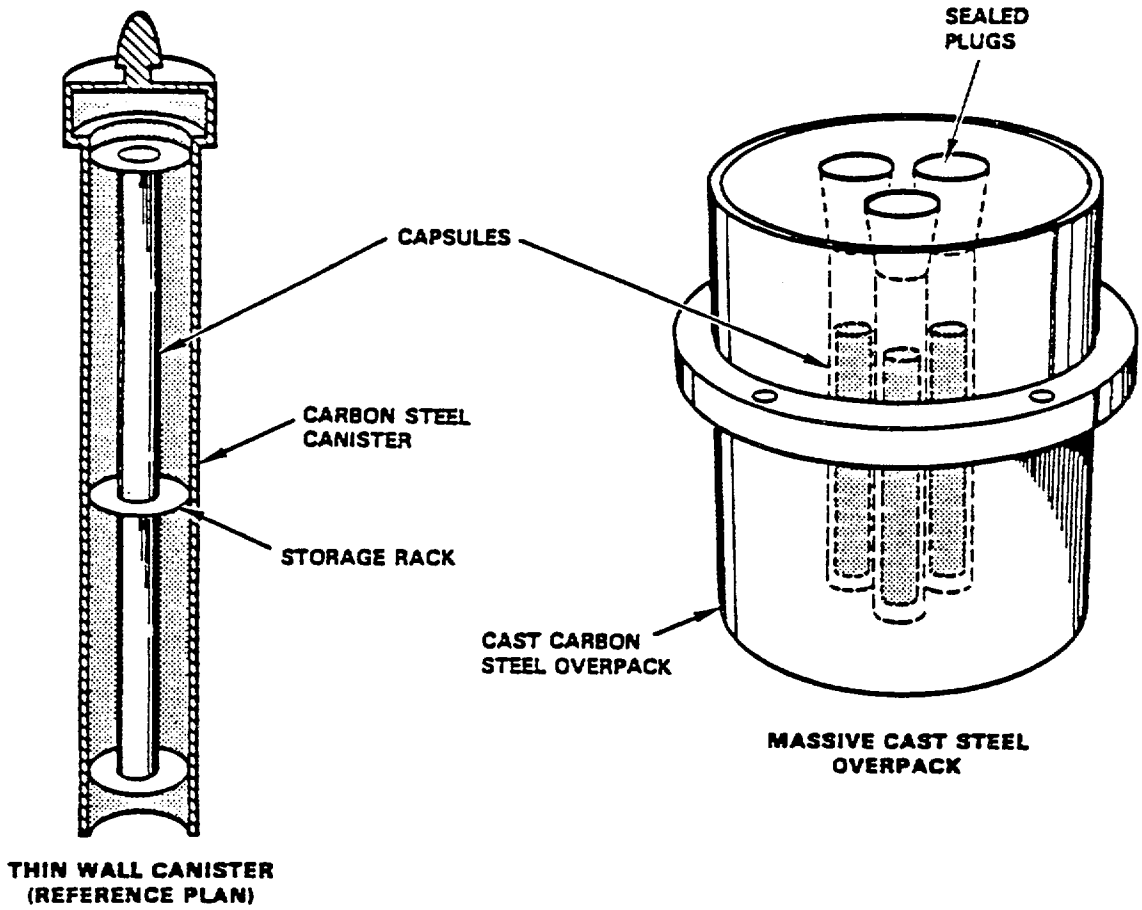
Wolfe 1986. Wolfe, Sylvia, telephone conversation with Royes Salmon, dated April 8, 1986.

	FORM	LOADING	PERCENT OF THEORETICAL DENSITY BASED ON TOTAL VOID SPACE OF CAPSULE	TEMPERATURE			
				AIR		WATER	
				CENTER LINE	SURFACE	CENTER LINE	SURFACE
STRONTIUM FLUORIDE	COMPACTED POWDER	150 kCi (MAX)	68	860°C	430°C	660°C	71°C
CESIUM CHLORIDE	MELT-CAST	70 kCi	65	450°C	200°C	327°C	58°C



	CAPSULE									
	INNER					OUTER				
	MATERIAL	WALL THICKNESS	OUTSIDE DIAMETER	TOTAL LENGTH	TOTAL CAP THICKNESS	MATERIAL	WALL THICKNESS	OUTSIDE DIAMETER	TOTAL LENGTH	TOTAL CAP THICKNESS
STRONTIUM FLUORIDE	HASTELLOY C-276 (UT)	0.305 (UT)	5.72	48.39	1.02	STAINLESS STEEL 316-L (UT)	0.277 (UT)	6.67	51.05	1.02
CESIUM CHLORIDE	STAINLESS STEEL 316-L (UT)	0.241 (UT)	5.72	50.10	1.02	STAINLESS STEEL 316-L (UT)	0.277 (UT)	6.67	52.77	1.02
NOTE: ALL DIMENSIONS ARE IN cm										

Fig. 5.5.1. Dimensional data for strontium and cesium capsules. Source: White 1986.



	OVERPACK DIMENSIONS		
	OVERALL LENGTH (m)	OUTSIDE DIAMETER (m)	APPROXIMATE REQUIRED
THIN WALL CANISTER	2.7	0.3	513
MASSIVE CAST STEEL OVERPACK	1.2	0.6	1300

PS8310-21

Fig. 5.5.2. Overpack concepts for strontium and cesium capsules. Source: White 1986.

Table 5.5.1. Reference data on radioisotopes discussed in Sect. 5.5

Radioisotope	Half-life, years (or as noted)	Curies per gram	Thermal power Watts per curie
H-3	12.35	9.65E+03	3.36E-05
Co-60	5.27	1.13E+03	1.54E-02
Kr-85	10.72	3.92E+02	1.50E-03
Sr-90	29.1	1.36E+02	1.16E-03
Y-90	2.67 days	5.44E+05	5.54E-03
Tc-99	213,000	1.70E-02	5.01E-04
Cs-137	30.0	8.70E+01	1.11E-03
Ba-137m	153 sec	5.38E+08	3.93E-03
Eu-152	13.6	1.73E+02	7.56E-03
Gd-153	242 days	3.53E+03	9.03E-03
Am-241	432	3.43E+00	3.32E-02
Cm-248	339,000	4.25E-03	1.24E-01
Cf-252	2.64	5.38E+02	7.13E-02

Table 5.5.2. Leasing of cesium capsules

Lessee	Location	Approximate quantity (10^6 Ci)
Iotech, Inc.	Colorado	15
Radiation Sterilizers, Inc.	Ohio and Georgia	21
ARECO	Virginia	1.25
Radiation Technology, Inc.	Arkansas and North Carolina	10
Total		~47

Source: Wolfe 1986.

Table 5.5.3. Location of Sr and Cs capsules produced at Hanford[a]

Strontium		
In storage basins at Hanford		605
Shipped		
Pacific Northwest Laboratories (Richland, Wa.)	5	
Oak Ridge, Tn.	24	
Nevada test site	4	
Cut[b]	2	
Total shipped		35
Total capsules produced		640
Cesium		
In storage basins at Hanford		343
Shipped		
Oak Ridge, Tn.	18	
England (Harwell and Stratford)	3	
New Mexico (Sandia)	19	
France (CEA)	4	
Pacific Northwest Laboratories	22	
Iotech 1 (Northglenn, Co.)	248	
Cut[b]	46	
Cut, Federal Republic of Germany[b]	143	
Radiation sterilizers, Inc. (Westerville, Ohio)	180	
Radiation sterilizers, Inc. (Atlanta, Ga.)	252	
Total shipped		935
Planned shipments from Hanford[c]		
TI[d]	4	
Iotech (Northglenn, Co.)	61	
ARECO (Lynchburg, Va.)[e,f]	25	
RTI (West Memphis, Ar.)[f,g]	208	
Total planned shipments		298
Total cesium capsules		1576

[a]Source: White 1986.

[b]Cut means that the capsules have been disassembled with no intent for recovery.

[c]Some of these capsules have already been shipped.

[d]TI is an acronym for Transportable (Cesium) Irradiator. The location of these capsules will vary depending on the location of the TI.

[e]ARECO is an acronym for Applied Radiant Energy Corporation.

[f]At the time of shipment, the destination may be other than that specified.

[g]RTI is an acronym for Radiation Technologies Incorporated.

Table 5.5.4. Radioactivity and thermal power of canisters containing strontium and cesium capsules[a]

Decay time (years)	Strontium canister (4 capsules)		Cesium canister (4 capsules)	
	Curies	Watts	Curies	Watts
0	412,400	1,380	371,000	918
5	366,300	1,230	330,000	817
10	325,200	1,090	294,000	727
20	256,300	860	233,000	577
50	125,500	420	117,000	290
100	38,200	128	36,800	91
200	3,530	12	3,650	9
300	327	1.1	360	0.9
1,000	1.9E-05	6.4E-08	3.4E-05	8.4E-08

[a]Based on ORIGEN2 calculations. Radioactivity and thermal power include the contributions of the daughter isotopes Y-90 and Ba-137M. Starting point for decay time is December 1985. The assumed thermal limits at a decay time of 10 years are 1,170 W/canister for Sr capsules and 800 W/canister for Cs capsules (Coony 1987).

Table 5.5.5. Radioactivity and thermal power of strontium capsule as a function of decay time[a]

RADIOACTIVITY, CURIES
HANFORD STRONTIUM CAPSULES. BASIS ONE CAPSULE, AVERAGE ACTIVITY

	DISCHARGE	5.0YR	10.0YR	20.0YR	50.0YR	100.0YR	200.0YR	300.0YR	1.0KY	10.0KY	100.0KY	1.0MY
SR 90	5.157E+04	4.578E+04	4.065E+04	3.204E+04	1.569E+04	4.772E+03	4.415E+02	4.086E+01	2.373E-06	0.0	0.0	0.0
Y 90	5.157E+04	4.579E+04	4.066E+04	3.204E+04	1.569E+04	4.773E+03	4.416E+02	4.087E+01	2.374E-06	0.0	0.0	0.0
SUMTOT	1.031E+05	9.158E+04	8.130E+04	6.408E+04	3.138E+04	9.545E+03	8.832E+02	8.172E+01	4.747E-06	0.0	0.0	0.0
TOTAL	1.031E+05	9.158E+04	8.130E+04	6.408E+04	3.138E+04	9.545E+03	8.832E+02	8.172E+01	4.747E-06	0.0	0.0	0.0

THERMAL POWER, WATTS
HANFORD STRONTIUM CAPSULES. BASIS ONE CAPSULE, AVERAGE ACTIVITY

	DISCHARGE	5.0YR	10.0YR	20.0YR	50.0YR	100.0YR	200.0YR	300.0YR	1.0KY	10.0KY	100.0KY	1.0MY
SR 90	5.985E+01	5.314E+01	4.717E+01	3.718E+01	1.821E+01	5.538E+00	5.125E-01	4.742E-02	2.754E-09	0.0	0.0	0.0
Y 90	2.858E+02	2.538E+02	2.253E+02	1.776E+02	8.696E+01	2.645E+01	2.448E+00	2.265E-01	1.316E-08	0.0	0.0	0.0
SUMTOT	3.457E+02	3.069E+02	2.725E+02	2.148E+02	1.052E+02	3.199E+01	2.960E+00	2.739E-01	1.591E-08	0.0	0.0	0.0
TOTAL	3.457E+02	3.069E+02	2.725E+02	2.148E+02	1.052E+02	3.199E+01	2.960E+00	2.739E-01	1.591E-08	0.0	0.0	0.0

[a]Initial time (shown as "discharge") is December 1985. Basis: Wilde 1986a and ORIGEN2 calculations.

Table 5.5.6. Radioactivity and thermal power of cesium capsule as a function of decay time[a]

RADIOACTIVITY, CURIES
HANFORD CESIUM CAPSULES. BASIS ONE CAPSULE, AVERAGE ACTIVITY

	DISCHARGE	5.0YR	10.0YR	20.0YR	50.0YR	100.0YR	200.0YR	300.0YR	1.0KY	10.0KY	100.0KY	1.0MY
CS137	4.759E+04	4.240E+04	3.777E+04	2.998E+04	1.499E+04	4.721E+03	4.684E+02	4.647E+01	4.395E-06	0.0	0.0	0.0
BA137M	4.505E+04	4.011E+04	3.573E+04	2.836E+04	1.418E+04	4.466E+03	4.431E+02	4.396E+01	4.157E-06	0.0	0.0	0.0
SUMTOT	9.264E+04	8.251E+04	7.351E+04	5.834E+04	2.917E+04	9.188E+03	9.115E+02	9.043E+01	8.552E-06	0.0	0.0	0.0
TOTAL	9.264E+04	8.251E+04	7.351E+04	5.834E+04	2.917E+04	9.188E+03	9.115E+02	9.043E+01	8.552E-06	0.0	0.0	0.0

THERMAL POWER, WATTS
HANFORD CESIUM CAPSULES. BASIS ONE CAPSULE, AVERAGE ACTIVITY

	DISCHARGE	5.0YR	10.0YR	20.0YR	50.0YR	100.0YR	200.0YR	300.0YR	1.0KY	10.0KY	100.0KY	1.0MY
CS137	5.264E+01	4.690E+01	4.178E+01	3.316E+01	1.658E+01	5.222E+00	5.181E-01	5.140E-02	4.861E-09	0.0	0.0	0.0
BA137M	1.769E+02	1.575E+02	1.403E+02	1.114E+02	5.568E+01	1.754E+01	1.740E+00	1.726E-01	1.632E-08	0.0	0.0	0.0
SUMTOT	2.295E+02	2.044E+02	1.821E+02	1.445E+02	7.226E+01	2.276E+01	2.258E+00	2.240E-01	2.118E-08	0.0	0.0	0.0
TOTAL	2.295E+02	2.044E+02	1.821E+02	1.445E+02	7.226E+01	2.276E+01	2.258E+00	2.240E-01	2.118E-08	0.0	0.0	0.0

[a]Initial time (shown as "discharge") is December 1985. Basis: Wilde 1986a and ORIGEN2 calculations.

Table 5.5.7. Recent commercial sales of ^{252}Cf by SRO (in mg)

Primary customer	1981	1982	1983	1984	1985[a]
GE-Vallecitos	4.8	6.6	6.4	2.6	6
Monsanto	11.8	5.4	11.2	20.9	21.5
Amersham	12.0	--	8.1	16.4	10
Karlsruhe	1.0	2.1	14.9	--	3
CEN France	7.0	7.4	10.5	7.6	15
Frontier	--	--	--	--	xx
Total	36.6	21.5	51.1	47.5	55.5

[a]Estimated by SRO, based on customer information.

Table 5.5.8. 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.

5.6 LWR OPERATIONAL WASTE

5.6.1 GTCC Wastes from Routine LWR Reactor Operations

The quantity of GTCC wastes that will be generated in future years by LWR nuclear reactors during normal operations can be estimated by multiplying the predicted total quantity of waste of all classes that will be produced by the estimated fraction of the waste that will exceed Class C limits. Data have been obtained (Cline 1985) on nearly 900 samples of waste from over 50 commercial pressurized and boiling water reactors. Only 12 (1.3%) of the samples contained TRU concentrations that exceeded the Class C limits. The percentages of the samples that contained TRU concentrations that exceeded Class C limits were 4.8% for BWR filter sludge, 9.4% for PWR filter cartridges, 2.2% for PWR filter sludge, 2.0% for PWR evaporator bottoms, and zero for all other types of samples. None of the samples contained concentrations of fission product and activation product radionuclides that approached Class C limits. These samples consisted of normal materials from typical radwaste streams (e.g., resins, sludges, evaporation bottoms, filters, and dry active wastes) and did not include neutron activated metal components removed from inside the reactor pressure vessel.

A breakdown of BWR and PWR operations waste into categories, including those cited above from the Cline study, has been done (Forsberg, Carter, and Kibbey 1985). They reported that the waste from BWRs is 27% (by volume) filter sludge, and the waste from PWRs is 0.8% filter cartridges, 0.1% filter sludge, and 42% evaporator bottoms. Multiplying these numbers by the percentages of the various waste types that exceed Class C limits yields the result that 1.3% of the BWR waste and 0.9% of the PWR waste would exceed Class C limits (Table 5.6.1). Virtually all of the waste samples that exceeded Class C limits were either BWR filter sludge or PWR evaporator bottoms.

The total volume of GTCC waste from this source can be calculated by multiplying these numbers by the total volume of radioactive wastes

of all classes generated by PWRs and BWRs. A recent IDB report (DOE/IDB 1985) gives, for the total volume of wastes generated by BWRs per year, 13 × 10^3 m^3 in 1985, increasing to 49 × 10^3 m^3 in 2020. The amount generated by PWRs per year was 7 × 10^3 m^3 increasing to 27 × 10^3 m^3 in 2020. Therefore, the predicted volume of GTCC waste that will be generated per year will increase from 220 m^3 in 1985 to 850 m^3 in 2020. According to the IDB (DOE/IDB 1985), the total nuclear electrical capacity will increase from 80 GW(e) in 1985 to 247.7 GW(e) in 2020. Therefore the predicted GTCC waste volume corresponds to 2.8 m^3/GW(E) in 1985 and 3.4 m^3/GW(e) in 2020 or, in round numbers, 3 m^3/GW(e)-yr.

Another study, using a different premise and different data, came up with a much smaller estimate. Daling et al. (1986) have predicted the volumes of TRU waste that will be generated by commercial power plants in the future from a survey of the volumes of TRU waste that have been produced and are stored at reactors now operating. From the volume of TRU wastes stored at each reactor site and the number of years each reactor has operated, they calculated the average volume of TRU waste produced per reactor per year. They then multiplied this number by the total number of reactors that are now operating, and the predicted number that will be operating in the future, to obtain the yearly production of TRU wastes through 2020. They estimated that the total volume of TRU waste that will be produced will increase from 14 m^3/yr at the present time to 25 m^3/yr in 2020. Or, in terms of reactor operation, 0.1 to 0.2 m^3/GW(e)-yr. Even this range of values may be high since it included atypical data from the Oyster Creek reactor (which had experienced abnormal operating conditions).

A number of factors could alter the estimates given above. No special treatment was involved, but changing circumstances could provide certain incentives. For example, in the Cline data, there was no dilution by blending with other, lower activity waste. None of the

(unpackaged) waste samples measured by Cline et al. (1985) had TRU concentrations that exceeded the Class C waste by more than a factor of 3.6. Therefore, if the wastes were diluted by more than this during packaging, then none of the waste packages would exceed Class C limits. Even dilution by a factor of only two would decrease the percentage of the waste that exceeded Class C limits by a factor of about two. Therefore, their calculated volume of GTCC waste should probably be viewed as an upper limit, when considered from this point of view.

On the other hand, if volume reduction techniques such as incineration or compaction are utilized, the actual volumes of GTCC wastes could actually increase, because the concentrating effect might escalate some Class C wastes into GTCC. However, these techniques are still in the developmental stage, so it is difficult to attempt to quantify the effects of volume reduction at the present time. Economics and the availability of disposal sites will control the eventual choice between a smaller volume of higher category waste and a larger volume of lower category waste. Individual utilities may elect different options, depending on individual circumstances and constraints.

5.6.2 References for Section 5.6

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Table 5.6.1. Estimates of the percentages of LWR reactor operating wastes that exceed Class-C limits[a]

Waste type	Boiling-water reactors			Pressurized-water reactors		
	% of measurements that are greater-than Class C limits[b]	% of total waste volume contributed by waste type[c]	% greater than-Class-C waste is of total waste volume	% of measurements that are greater-than Class C limits[b]	% of total waste volume contributed by waste type[c]	% greater than-Class-C waste is of total waste volume
Resin	0	3.7	0	0	2.8	0
Filter sludge	4.8	27.3	1.3	2.2	0.1	0.002
Filter cartridge	0	0	0	9.4	0.8	0.08
Evaporator bottoms	0	16.7	0	2.0	42.3	0.8
Compactable trash	0	45.7	0	0	47.9	0
Non-compactable trash	0	6.6	0	0	6.0	0
Total % of waste that exceeds Class-C limits			1.3			0.88

[a]Normal radwaste streams. Does not include neutron-activated metal components removed from the reactor pressure vessel.

[b]Cline, Noyce, Coe, and Wright, 1985.

[c]Forsberg, Carter, and Kibbey, 1985.