Table 2.7.6. Sample Material Report for Inconel X-750 from LWR Assemblies Database

Radiological Report (Curies)
for
Inconel X-750
Boiling Water Reactor - In Core Zone
Standard Burnup (30000 MWd) - 15 Years After Discharge

Isotope	Curies
(Values are per kg of irradiated material)	
C 14	2.226E-02
FE 55	9.179E-01
CO 60	1.314E+02
NI 59	2.726E-01
NI 63	4.055E+01
NB 94	1.706E-02
TC 99	3.058E-14
TOTAL	1.733E+02

Table 2.7.7. Sample SFD Hardware Radiological Report from LWR Assemblies Database

Radiological Report Page 1
for GE BWR/4-6 8 X 8 Prepres.
Boiling Water Reactor
Standard Burnup (30000 MWd) - 15 Years After Discharge

Isotope	Curies
T. TIE PLATE (Top Zone)	
Weight: 2.10 Kg; Material(s): St.Steel 304(100.0%)	
C 14	4.670E-03
FE 55	1.105E+00
CO 60	3.415E+00
NI 59	7.066E-03
NI 63	1.051E+00
NB 94	3.987E-05
Totals	5.583E+00
SPACER-INCORE (In Core Zone)	
Weight: 2.31 Kg; Material(s): Zircaloy-4(85.7%), Inconel X-750(14.3%)	
C 14	1.004E-02
FE 55	3.358E-01
CO 60	4.372E+01
NI 59	8.985E-02
NI 63	1.337E+01
SR 90	1.262E-05
ZR 93	2.362E-03
NB 93M	1.293E-03
NB 94	6.075E-03
TC 99	5.638E-08
SN121M	3.946E-03
SB125	3.053E-01
TE125M	7.446E-02
I129	1.856E-16
Totals	5.792E+01
WATER ROD (In Core Zone)	
Weight: 1.88 Kg; Material(s): Zircaloy-2(100.0%)	
C 14	2.576E-03
FE 55	2.163E-02
CO 60	3.796E-01
NI 59	3.550E-04
NI 63	5.288E-02

*For St.Steel, radiological characteristics are based on St.Steel 304.

Table 2.7.7 (continued)

Radiological Report (continued) Page 2
for GE BWR/4-6 8 X 8 Prepres.
Boiling Water Reactor
Standard Burnup (30000 MWd) - 15 Years After Discharge

Isotope	Curies
WATER ROD (In Core Zone)	
Weight: 1.88 Kg; Material(s): Zircaloy-2(100.0%)	
SR 90	1.201E-05
ZR 93	2.248E-03
NB 93M	1.231E-03
NB 94	4.288E-04
TC 99	5.369E-08
SN121M	3.755E-03
SB125	2.905E-01
TE125M	7.087E-02
I129	1.767E-16
Totals	8.261E-01
B. TIE PLATE (Bottom Zone)	
Weight: 4.67 Kg; Material(s): St.Steel 304(100.0%)	
C 14	1.559E-02
FE 55	3.690E+00
CO 60	1.140E+01
NI 59	2.359E-02
NI 63	3.510E+00
NB 94	1.331E-04
Totals	1.864E+01
FINGER SPRINGS (Bottom Zone)	
Weight: 0.05 Kg; Material(s): Inconel X-750(100.0%)	
C 14	1.603E-04
FE 55	6.609E-03
CO 60	9.464E-01
NI 59	1.962E-03
NI 63	2.920E-01
NB 94	1.229E-04
TC 99	2.202E-16
Totals	1.247E+00

*For St.Steel, radiological characteristics are based on St.Steel 304.

Table 2.7.7 (continued)

Spent Fuel Disassembly Hardware — Page 3
for
GE BWR/4-6 8 X 8 Prepres.

Top Zone — Units are Curies
Weight: 2.10 kg. — Standard Burnup (30000 MWd)
Reduced Volume: 0.0002616 cu m. — 15 Years After Discharge

Totals by isotope for Top Zone:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	4.670E-03	2.226E-03	1.785E+01
FE 55	1.105E+00	5.267E-01	4.224E+03
CO 60	3.415E+00	1.628E+00	1.305E+04
NI 59	7.066E-03	3.368E-03	2.701E+01
NI 63	1.051E+00	5.010E-01	4.018E+03
NB 94	3.987E-05	1.900E-05	1.524E-01

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
C 14	1.785E+01	80.00	0.22
NI 59	2.701E+01	200.00	0.14
NB 94	1.524E-01	0.20	0.76
NI 63	4.018E+03	7000.00	0.57
Totals			1.69

Curies in Top Zone by Material:

St.Steel 304
5.583E+00

Top Zone Totals:

in Curies	Curies/kg	Curies/cu m
5.583E+00	2.661E+00	2.134E+04

Table 2.7.7 (continued)

Spent Fuel Disassembly Hardware Page 4
for
GE BWR/4-6 8 X 8 Prepres.

In Core Zone — Units are Curies
Weight: 4.19 kg. — Standard Burnup (30000 MWd)
Reduced Volume: 0.0006274 cu m. — 15 Years After Discharge

Totals by isotope for In Core Zone:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	1.262E-02	3.016E-03	2.011E+01
FE 55	3.574E-01	8.540E-02	5.696E+02
CO 60	4.410E+01	1.054E+01	7.029E+04
NI 59	9.020E-02	2.155E-02	1.438E+02
NI 63	1.342E+01	3.207E+00	2.139E+04
SR 90	2.463E-05	5.885E-06	3.926E-02
ZR 93	4.610E-03	1.102E-03	7.348E+00
NB 93M	2.524E-03	6.031E-04	4.023E+00
NB 94	6.504E-03	1.554E-03	1.037E+01
TC 99	1.101E-07	2.631E-08	1.755E-04
SN121M	7.701E-03	1.840E-03	1.227E+01
SB125	5.958E-01	1.424E-01	9.496E+02
TE125M	1.453E-01	3.472E-02	2.316E+02
I129	3.623E-16	8.657E-17	5.774E-13

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
C 14	2.011E+01	80.00	0.25
NI 59	1.438E+02	200.00	0.72
NB 94	1.037E+01	0.20	51.83
TC 99	1.755E-04	3.00	< 0.01
NI 63	2.139E+04	7000.00	3.06
SR 90	3.926E-02	7000.00	< 0.01
I129	5.774E-13	0.08	< 0.01
Totals			55.86

Curies in In Core Zone by Material:

Zircaloy-2	Zircaloy-4	Inconel X-750
8.261E-01	8.237E-01	5.710E+01

In Core Zone Totals:

in Curies	Curies/kg	Curies/cu m
5.875E+01	1.404E+01	9.364E+04

Table 2.7.7 (continued)

Spent Fuel Disassembly Hardware — Page 5
for
GE BWR/4-6 8 X 8 Prepres.

Bottom Zone — Units are Curies
Weight: 4.72 kg — Standard Burnup (30000 MWd)
Reduced Volume: 0.0005881 cu m. — 15 Years After Discharge

Totals by isotope for Bottom Zone:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	1.575E-02	3.338E-03	2.678E+01
FE 55	3.697E+00	7.836E-01	6.287E+03
CO 60	1.235E+01	2.618E+00	2.100E+04
NI 59	2.555E-02	5.415E-03	4.345E+01
NI 63	3.802E+00	8.058E-01	6.465E+03
NB 94	2.560E-04	5.426E-05	4.353E-01
TC 99	2.202E-16	4.667E-17	3.744E-13

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
C 14	2.678E+01	80.00	0.33
NI 59	4.345E+01	200.00	0.22
NB 94	4.353E-01	0.20	2.18
TC 99	3.744E-13	3.00	< 0.01
NI 63	6.465E+03	7000.00	0.92
Totals			3.65

Curies in Bottom Zone by Material:

Inconel X-750	St.Steel 304
1.247E+00	1.864E+01

Bottom Zone Totals:

in Curies	Curies/kg	Curies/cu m
1.989E+01	4.215E+00	3.382E+04

Table 2.7.7 (continued)

Spent Fuel Disassembly Hardware Page 6
for
GE BWR/4-6 8 X 8 Prepres.

All Zones Units are Curies
Weight: 11.00 kg. Standard Burnup (30000 MWd)
Reduced Volume: 0.0014771 cu m. 15 Years After Discharge

Totals by isotope for All Zones:

Isotope	Curies	Curies/kg	Curies/cu m
C 14	3.304E-02	3.003E-03	2.237E+01
FE 55	5.160E+00	4.690E-01	3.493E+03
CO 60	5.987E+01	5.442E+00	4.053E+04
NI 59	1.229E-01	1.117E-02	8.320E+01
NI 63	1.827E+01	1.661E+00	1.237E+04
SR 90	2.463E-05	2.239E-06	1.667E-02
ZR 93	4.610E-03	4.191E-04	3.121E+00
NB 94	6.800E-03	6.181E-04	4.604E+00
NB 93M	2.524E-03	2.294E-04	1.709E+00
TC 99	1.101E-07	1.001E-08	7.454E-05
SN121M	7.701E-03	7.000E-04	5.214E+00
SB125	5.958E-01	5.416E-02	4.034E+02
TE125M	1.453E-01	1.321E-02	9.837E+01
I129	3.623E-16	3.293E-17	2.453E-13

Class C Limit Calculations:

Isotope	Curies/cu m	Class C Limit	Ratio
C 14	2.237E+01	80.00	0.28
NI 59	8.320E+01	200.00	0.42
NB 94	4.604E+00	0.20	23.02
TC 99	7.454E-05	3.00	< 0.01
NI 63	1.237E+04	7000.00	1.77
SR 90	1.667E-02	7000.00	< 0.01
I129	2.453E-13	0.08	< 0.01
Totals			25.48

Curies in All Zones by Material:

Zircaloy-2	Zircaloy-4	Inconel X-750	St.Steel 304
8.261E-01	8.237E-01	5.835E+01	2.422E+01

All Zones Totals:

in Curies	Curies/kg	Curies/cu m
8.422E+01	7.656E+00	5.702E+04

2.8 NON-FUEL ASSEMBLY HARDWARE

2.8.1 Overview

According to the standard contract between the Department of Energy and the nuclear utilities, "Nonfuel components including, but not limited to control spiders, burnable poison rod assemblies, control rod elements, thimble plugs, fission chambers, and primary and secondary neutron sources, that are contained within the fuel assembly, or BWR channels that are an integral part of the fuel assembly, which do not require special handling, may be included as part of the spent nuclear fuel delivered for disposal pursuant to this contract." (10 CFR 961). Nonfuel components by themselves (apart from fuel assemblies) are classified as nonstandard fuel class NS-2: Nonfuel Components. Fuel assemblies containing nonfuel components are classified as standard fuel Class S-1 or S-2 if the dimensions are less than the maximum nominal physical dimensions. If these dimensions are exceeded, these assemblies are classified as nonstandard fuel Class NS-1: Physical Dimensions. Nonfuel components other than fuel channels are generally used within or between assemblies but are not permanently attached to an assembly. In addition, these nonfuel components are usually retired from service on a schedule that is different from that of the fuel assemblies. For this reason, they are characterized separately from assemblies (and spent fuel disassembly hardware) and are called non-fuel assembly (NFA) hardware.

This section describes the functions of NFA hardware, the physical characteristics of NFA hardware, estimates the quantities to be disposed of, and discusses the radiological characteristics of these wastes. The calculation of their radiological properties requires procedures tailored to suit the various cases.

Broad descriptions of the physical characteristics of NFA hardware, segregated according to a classification scheme developed for the Energy Information Administration (Moore 1991b), are given in this section. This information is primarily taken from vendor-supplied data for PWRs and supplemented by information from the Federal Docket for BWRs. Detailed physical characterization of these components is given in the NFA Hardware PC Data Base. Table 2.8.1 gives an example of a physical description report from this data base.

The amount of NFA hardware the Civilian Radioactive Waste Management System (CRWMS) will accept is not well-defined at this time. Only limited information on the usage and replacement rates for NFA hardware is available and many unresolved questions exist. At least some NFA hardware has been, and is being, disposed of as low-level waste, relieving the CRWMS of the responsibility for these wastes. In this report, estimates of NFA hardware usage are developed based on vendor descriptions and information contained in the Federal Docket. These estimates are compared with information collected from the reactor operators by the Edison Electric Institute (Farrell 1990).

Because NFA hardware may remain in the reactor for many cycles and because more neutron flux zones are involved, radiological characterization of the NFA hardware requires a more flexible methodology than that used for SFD hardware. A method for estimating the radiological properties of NFA hardware, based on the application of previously calculated zone factors (Luksic 1989) to ORIGEN2 data, is presented. Experimental measurements of the radiological characteristics of NFA hardware are currently ongoing at Pacific Northwest Laboratories and will be used for a future revision.

Other acceptable methods of characterizing the radiological properties of NFA Hardware exist. A common method is using the ORIGEN2 code to determine a fractional isotope mix for a particular set of NFA Hardware. A shielding code is used to calculate a dose rate based on the isotope mix. This dose rate is compared to the actual dose rate and the radiological properties of the isotope mix are adjusted to match the actual dose rate. This method has been widely used and accepted to qualify wastes for disposal as LLW.

2.8.2 Functions of NFA Hardware Components

This subsection describes the functions of various types of NFA hardware. In commercial LWRs, NFA hardware components are used for the initiation, monitoring, limitation, and control of the nuclear chain reaction within the reactor core, or to direct coolant flow through the reactor.

In BWRs, **fuel channels** are used to control the flow of water through the core. The fuel channel "(1) forms the fuel bundle flow path outer periphery for bundle coolant flow, (2) provides surfaces for control rod guidance into the reactor core, (3) provides structural stiffness to the fuel bundle . . ., (4) minimizes . . . coolant bypass flow at the channel/lower tieplate interface, (5) transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structures, (6) provides a heat sink during loss-of-coolant accident (LOCA), and (7) provides a stagnation envelope for in-core fuel sipping." (GE 1986)

In PWRs, **thimble plug assemblies** (TPAs), or **orifice rod assemblies** (ORAs), are inserted into the guide tubes of fuel assemblies, effectively blocking the flow of coolant through empty guide tubes, where it is not needed.

Primary and secondary neutron sources provide a base source of neutrons to initiate the nuclear chain reaction. Primary neutron sources are typically polonium- or plutonium-beryllium alloys, which produce neutrons via an (α,n) reaction on the Be, or californium, which produces neutrons via spontaneous fission. Secondary neutron sources are typically antimony-beryllium alloys, which produce neutrons from a (γ,n) reaction on the Be. Neutron activation of the natural antimony produces antimony isotopes which are strong gamma emitters. These wastes should be a minor contributor to NFA Hardware quantities. In the core, PWR neutron sources assemblies (NSAs) are placed within the guide tubes of

assemblies; BWR neutron sources are placed between fuel assemblies. Because they are used incore, neutron sources are likely to be highly activated.

In-core instrumentation (INST) is used to monitor the neutron flux within the core. In BWRs, instrumentation is typically inserted between fuel channels, away from the locations of the cruciforms. In PWRs, instrumentation is inserted into an instrument tube in the fuel assembly. This instrument tube is typically in the center of the fuel assembly. It resembles a guide tube, although it is typically slightly larger. Instrumentation consists of some sort of neutron detector (typically U^{235} or rhodium) and the associated electronics required to process and transmit a signal to monitoring devices outside the reactor vessel. Instrumentation will probably be a minor contributor to NFA hardware quantities. The detectors/emitters on these pieces of hardware may be highly activated. The long lead wires of these components are outside the core zone and are far less activated; because they are a single unit, the averaging rule can be applied and the detectors are buried as LLW.

BWR **control rod blades** (CRBs) or cruciforms and PWR **control rod assemblies** (CRAs) provide control over the nuclear chain reaction by limiting the number of neutrons (both fast and thermal) in the core. Made of strongly neutron absorbing materials, such as boron carbide, hafnium, or a silver-indium-cadmium alloy, and attached to control rod/blade drives, cruciforms and control rods are inserted into the reactor core at planned shutdowns and during scrams. Because they are used to control the power level during cycles and are also attached to control rod drives, **axial power shaping assemblies** (APSAs) (used only in B&W 15 X 15 Class reactors) are included with control elements. Control elements may also be partially inserted into the core region to provide additional reactivity control during power operation. All commercial BWRs in this country use CRBs inserted from below as their primary means of reactivity control. Most PWRs use CRAs designed to fit into the guide tubes of fuel assemblies; these are inserted from above. Three PWRs - Indian Point 1, Palisades, and Yankee Rowe - use(d) cruciforms for reactivity control, inserted from below. Nearly all PWR CRAs are designed to fit into the guide tubes of a single assembly. However, twelve-element Control Element Assemblies (CEAs) for CE's SYSTEM 80 reactors fit into the guide tubes of five assemblies. Control elements may contribute significantly to the volume of NFA hardware waste. Because they are used incore, they are highly activated. Some CRB's are Greater Than Class C wastes, although a significant quantity of CRB's have been buried as LLW.

Burnable absorber assemblies (BAAs) are inserted and locked into the guide tubes of PWR fuel assemblies to provide additional reactivity control. Both borosilicate glass and a boron carbide/aluminum oxide mixture have been utilized as the absorber for these assemblies. Because these absorbing materials are rapidly depleted, BAAs are useful in limiting power output in the higher-enriched assemblies that are now being placed into service. Currently, new absorber assemblies are typically used each cycle. As a result, these assemblies will be a large contributor to the volume of NFA hardware waste. Some newer fuel assemblies include integral burnable absorbers (gadolinia or zirconium diboride) in the fuel rods in place of NFA burnable absorber. Because of this, the use of burnable absorber assemblies varies greatly from utility to utility, depending on the fuel vendor and the utility fuel management strategy. Because they are used in the core of the reactor, their levels of neutron activation are high.

2.8.3 Physical Descriptions of NFA Hardware

In 1986, information was obtained from Babcock & Wilcox (Cooper 1986), Combustion Engineering (Hayduk 1987), and Westinghouse (Westinghouse 1986) concerning the descriptions of the fuel assemblies and NFA hardware they had manufactured. These responses were used in the preparation of the original Characteristics Data Base (DOE 1987). This information remains the best general description of NFA hardware, although it has now been supplemented by information from the Federal Docket and the open literature, as well as an increased understanding of fuels and hardware usage. Information on the physical descriptions of NFA hardware for GE reactors is taken exclusively from the Docket and reports available in the open literature. The descriptions that follow are classified on the basis on assembly class, as defined in Section 2.2.

2.8.3.1 Babcock & Wilcox Reactors

Babcock & Wilcox (B&W) is the nuclear steam supply system (NSSS) vendor for three assembly classes: B&W 15 X 15, B&W 17 X 17, and Indian Point 1. Descriptions of NFA hardware from these assembly classes follow.

B&W 15 X 15 Class Reactors

Control rod assemblies for B&W 15 X 15 Class reactors consist of 16 individual rods with their upper ends fastened to a spider assembly. The control rod drive mechanism engages the spider assembly to withdraw and position the CRA. The spider assembly consists of 7 pounds of CF3M, 304, and 316 stainless steels. The control rods are about 13 feet long and 0.440 inch in diameter and the cladding is 304 stainless steel with 304 or 308 stainless steel end plugs. The thickness is not given, but the weight of the 16 tubes with lower end plugs is 18.4 pounds. The nuts, upper end plugs, and spring spacers weight 7.5 pounds. The Ag-In-Cd alloy weighs 95 pounds, and the total weight of the assembly is 130 pounds. B&W reactors use 61 CRAs in the core.

Axial power shaping assemblies have a similar spider assembly that weighs 7.8 pounds. The rods for these assemblies are the same length and diameter as control rods and use 304 stainless steel for cladding, whereas end

plugs, intermediate plugs, and nuts may be 304 or 308 stainless steel. The stainless steel parts, excluding the spider, weigh 24.7 pounds. The Ag-In-Cd absorber is shorter and weighs 23.4 pounds. The overall assembly weight is 57 pounds. B&W reactors use 8 APSAs in the core; these appear to be changed out about every 5 cycles.

Gray axial power shaping rod assemblies also have a similar spider assembly that weighs 7.5 pounds. The gray axial power shaping rods are 156 inches long and 0.444 inch in diameter and use 304 stainless steel for cladding while end plugs, intermediate plugs, and nuts may be of 304 or 308 stainless steel. The stainless steel parts weigh 29.8 pounds. The absorber is Inconel 600. It is 63.3 inches long and weighs 33.8 pounds. The total weight of the assembly is 71 pounds. The overall length of the assembly is 159.75 inches. No information is currently available on the usage of gray axial power shaping assemblies.

The description of **primary neutron sources** is incomplete. Weights are given for three of the ten components, but no overall weight is given. The shroud tube, intermediate plug, and lower end plug are made of 304 stainless steel and weigh one pound. No description of the source is given except that it is Am-Be-Cu and is B&W proprietary.

Regenerative (secondary) neutron sources consist of a coupling spider assembly and eight rods. There is no mention of orifice plugs for the other eight locations. The spider assembly is made of CF3M, 304, and 316 stainless steel and weighs 7.8 pounds. The clad and end plugs for the rods are made of 304 stainless steel, but the weights are not given. The length of the source rods is about 11 feet 8 inches and the diameter is 0.440 inch. The total weight of the cluster is 46.3 pounds. By comparison to other types of rods, it may be assumed that the weight of stainless steel is about 12 pounds; thus, the antimony-beryllium source is estimated to weigh 26.4 pounds. The source composition is B&W proprietary.

The spider for **burnable absorber assemblies** reactors is made of CF3M, 304, and 316 stainless steel and weighs 7.8 pounds. The 16 absorber rods use Zircaloy-4 for cladding, end fittings, and the nuts to fasten the rods to the spider. Each rod is about 12 feet 6 inches long and 0.430 inch in diameter. The hold-down spring is 302 or 304 stainless steel. The absorber material is a B&W proprietary mixture of Al_2O_3 and B_4C. The absorber material weighs 20.8 pounds, the Zircaloy 25.2 pounds, the springs 2.1 pounds, and the spider assembly 7.8 pounds. The overall weight of the assembly is 57 pounds.

The spider assembly for **orifice rod assemblies** is made of CF3M and 304 stainless steel and weighs 7.8 pounds. The orifice rods are about 12 inches long and 0.480 inch in diameter. The 16 rods and associated nuts weigh 7.7 pounds. The rods are made of 304 stainless steel whereas the nuts are made of 304 or 308 stainless steel. B&W's submittal indicates that this assembly should have "orifice plugs" made of 304 stainless steel but does not assign them a weight. The total assembly weight is given as 15.8 pounds; this leaves 0.3 pound for the weight of the plugs themselves.

B&W provides **in-core instrumentation** but no descriptive information is available at this time.

B&W 17 X 17 Class Reactors

At this time, no B&W 17 X 17 Class reactors have been used in commercial service. Three reactors of this class (WNP-1, Bellefonte 1, Bellefonte 2) were ordered, but none have been completed to date. For this reason, no information on NFA hardware from this class has been included in this report.

Indian Point 1 Class Reactor

No data has been collected to this point on NFA hardware from the Indian Point 1 reactor.

2.8.3.2 Combustion Engineering Reactors

Combustion Engineering (CE) is the NSSS vendor for six assembly classes: CE 14 X 14, CE 16 X 16, CE SYSTEM 80, Fort Calhoun, St. Lucie 2, and Palisades. Descriptions of NFA Hardware from these assembly classes follow. Combustion Engineering uses fueled and nonfueled burnable absorbers which are integral to fuel assemblies; CE does not use ORAs or separate BAAs in their reactors.

CE 14 X 14 Class Reactors

Control element assemblies for CE 14 X 14 class reactors consist of five rods fastened to a spider at their upper ends. All CEAs for CE 14 X 14 reactors use Inconel 625-clad control rods which are 0.948 inch in diameter (O.D.) with a 0.040-inch wall thickness. The pellet diameter is 0.86 inch. All CE 14 X 14 CEAs are 161 inches long. The spider is made of 304 stainless steel and weighs 7.5 pounds.

Four variations of the CEA's (one full-length and three types of part-length) have been identified for CE 14 X 14 reactors. The full-length CEAs use 134 inches of B_4C in the center finger. Each of the outside fingers has 2.6 inches of Inconel on the tip, followed by 8.0 inches of Ag-In-Cd alloy and 124 inches of B_4C. The full-length assemblies contain 7.5 pounds of stainless steel, 39 pounds of Inconel, 24.2 pounds of B_4C, and 6.1 pounds of Ag-In-Cd. Each reactor uses 65 full-length CEAs in the core.

The part-length CEAs have several variations in the number and kinds of control material that they employ. Arrangements include:

- one B_4C rod and four stainless steel rods; this CEA weighs 104.5 pounds; twelve CEAs are used per core.
- one rod of Al_2O_3 and four rods of Ag-In-Cd and B_4C; this CEA weighs 82.4 pounds; eight CEAs are used per core.

- one rod of Al_2O_3, two rods of Ag-In-Cd and B_4C, and two rods having 10 inches of stainless steel and 124 inches of Al_2O_3; this CEA weighs 63.0 pounds; twelve CEAs are used per core.

Two **neutron source assemblies,** which combine the primary and secondary sources, are placed in the guide tubes of perimeter assemblies on opposite sides of the core. Their life expectancy is 4000 EFPD. They are stationary fixtures with an upper shoulder resting on a post of the fuel assembly and held down by a plunger and spring. The NSA consists of two subassemblies. The lower subassembly is 42.5 inches long and contains the sources, a tubular spacer at the bottom, and a hold-down spring at the top. There are 15.65 inches of 0.735-inch-diameter antimony-beryllium pellets in the center with a plutonium-beryllium capsule 10.75 inches long and 0.675 inch in diameter above and below them. The upper subassembly consists of the upper fitting, a coupler to connect to the lower subassembly, and a tube of the proper length to center the lower assembly in the active zone of the core. The plunger, upper subassembly, cladding, and spacer for the lower subassembly are 316 stainless steel. The cladding and tubing diameter is 0.875 inch. The assemblies contain 9.4 pounds of stainless steel, 0.5 pound of beryllium, 0.4 pound of nickel-based alloy springs, and 0.3 pound each of plutonium and antimony.

The **in-core instrument assemblies** are located strategically about the reactor core in positions not designated for control element assemblies. The emitters are rhodium attached to Inconel 600 lead wires surrounded by an aluminum oxide insulator and sheathed in Inconel 600, which is 0.064 inches in diameter. Four emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a stainless steel outer sheath which is 0.35 inch maximum diameter. These assemblies are 38 - 41 feet long and weigh 11.7 - 12.8 pounds each. There are 45 strings per reactor.

CE 16 X 16 Class Reactors

Control element assemblies for the CE 16 X 16 Class reactors with 16 X 16 fuel vary slightly in length from 180.8 - 181.3 inches. Three variations of full-length CEAs and two variations of part-length CEAs have been identified; all are clad in Inconel 625. The cladding is 0.816 inch in diameter with a 0.035 inch wall thickness. The pellets of neutron absorbing material are 0.737 inch in diameter. The plenum springs are 302 stainless steel. The spider is made of 304 stainless steel and weighs 8 pounds.

Full-length CEAs used at Arkansas Nuclear One (ANO), Unit 2 weigh 71.3 pounds, have a center finger with 140.0 inches of B_4C and 9.2 inches of Inconel 625, and have outside elements with 135.5 inches of B_4C and 12.5 inches of Inconel 625. Full-length CEAs used at other CE 16 X 16 Class reactors weigh 72 pounds and have 136 inches of B_4C and 12.5 inches of Ag-In-Cd in both the center and outside fingers. A few CEAs for other CE 16 X 16 Class reactors have been fabricated without the center finger.

The variations in the part-length CEAs are very slight. Both use five fingers using 75.0 inches of Inconel 625 near the tip and 16 inches of B_4C above it. The differences are the slight variations in length (180.8 for ANO, Unit 2 and 181.3 for other CE 16 X 16 reactors) and weight (91.1 and 92.0 pounds, respectively).

The **neutron source assemblies** for CE 16 X 16 Class reactors appear to be identical to the NSAs for other CE reactors with 16 X 16 fuel (St. Lucie 2 and CE SYSTEM 80 Classes). The assembly, which is 113 inches long, consists of two subassemblies. The lower subassembly is 42.5 inches long and contains the sources, a tubular spacer at the bottom, and a hold-down spring at the top. There are 15.65 inches of 0.654-inch-diameter antimony-beryllium pellets in the center with a plutonium-beryllium capsule 6.0 inches long and 0.654 inch in diameter above and below them. The upper subassembly consists of the upper fitting, an coupler to connect to the lower subassembly, and a tube of the proper length to center the lower assembly in the active zone of the core. The plunger, upper subassembly, cladding, and spacer for the lower subassembly are 304 stainless steel. The cladding and tubing diameter is 0.812 inch. The assemblies contain 9.1 pounds of stainless steel, 0.5 pound of beryllium, 0.4 pound of nickel-based alloy springs, and 0.3 pound each of plutonium and antimony, giving a calculated total weight of 10.6 pounds.

The **in-core instrument assemblies** have emitters which are made of rhodium attached to Inconel 600 lead wires, surrounded by an aluminum oxide insulator, and sheathed in Inconel 600, which is 0.064 inches in diameter. Four emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a Inconel 600 outer sheath, which is 0.35 inch maximum diameter. These assemblies are 37 - 38 feet long, weigh 6.4 - 7.3 pounds, and have a life expectancy of 1200 EFPD. There are 44 instrumentation strings at ANO2, and 56 at the other CE 16 X 16 Class reactors.

CE SYSTEM 80 Class Reactors

Control element assemblies for CE SYSTEM 80 reactors are distinctly different from the CEAs of any other U.S. commercial LWR. In addition to being significantly longer (253 inches), some of the CEAs for SYSTEM 80 reactors have twelve rods and are designed to be inserted in five different assemblies at the same time. These features make the transportation and storage of fuel assemblies with CEAs integral to the assemblies practically impossible. The intact CEAs are significantly longer than standard spent fuel cases, and, if accepted by the CRWMS, will require special handling.

Each SYSTEM 80 reactor uses 48 full-length 12-element CEAs, 28 full-length 4-element CEAs, and 13 part-length 4-element CEAs. Both of the full-length CEAs use 148 inches of B_4C in each rod. The part-length CEAs use 16 inches of B_4C, followed by 75 inches of

Inconel at the tip of each rod. The 12-element full-length CEAs weigh 192.2 pounds, the 4-element full-length CEAs weigh an estimated 68 pounds, and the part-length CEAs weigh 95.0 pounds. All are clad in Inconel 625. The cladding is 0.816 inch in diameter with a 0.035 inch wall thickness. The pellets of neutron absorbing materials are 0.737 inch in diameter. The plenum springs are 302 stainless steel. The spiders for these CEAs are made of 304 stainless steel. The spider for the twelve element array weighs 19.5 pounds.

Eight-element SYSTEM 80 CEAs (fitting into 3 assemblies) are mentioned in the Palo Verde Preliminary Safety Analysis Reports. However, in CEA patterns shown in reload submittals for both Palo Verde 1 and 3, no 8-element CEAs are shown. Additionally, CE's data submittal did not describe 8-element CEAs. From this, we conclude that 8-element CEAs were planned, but not used in CE SYSTEM 80 reactors.

The **neutron source assemblies** for CE SYSTEM 80 Class reactors appear to be identical to the NSAs for other CE reactors with 16 X 16 fuel (St. Lucie 2 and CE 16 X 16 Classes) and are described in the section on CE 16 X 16 Class fuel.

The **in-core instrument assemblies** are located strategically about the reactor core in positions not designated for control element assemblies. The emitters are rhodium, attached to Inconel 600 lead wires surrounded by an aluminum oxide insulator, and sheathed in Inconel 600, which is 0.064 inches in diameter. Four emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a Inconel 600 outer sheath, which is 0.45 inch maximum diameter. These assemblies are 116 feet long, weigh 34.3 pounds, and have a life expectancy of 1200 EFPD. There are 61 instrumentation strings per reactor.

Fort Calhoun Class Reactor

Fort Calhoun uses **control element assemblies** which are 152 inches long; they use 128 inches of B_4C in all five fingers of the full-length CEAs and the center finger of the part-length CEAs. The four outside fingers of the part-length CEAs have 32 inches of B_4C. The full-length assembly contains 7.5 pounds of stainless steel, 34 pounds of Inconel, and 25 pounds of B_4C, for a total weight of 66.5 pounds. The part-length assembly weight is 63.0 pounds. Both are clad in Inconel 625. The cladding is 0.948 inch in diameter with a 0.040 inch wall thickness. The pellets of neutron absorbing materials are 0.860 inch in diameter. The plenum springs are 302 stainless steel. The spider is 304 stainless steel and weighs 7.5 pounds.

The **neutron source assemblies** consists of two subassemblies and are approximately 100 inches long. The lower subassembly is 42.5 inches long and contains the sources, a tubular spacer at the bottom, and a hold-down spring at the top. There are 15.65 inches of 0.734-inch-diameter antimony-beryllium pellets in the center with a plutonium-beryllium capsule 10.75 inches long and 0.675 inch in diameter above and below them. The upper subassembly consists of the upper fitting, an coupler to connect to the lower subassembly, and a tube of the proper length to center the lower assembly in the active zone of the core. The plunger, upper subassembly, cladding, and spacer for the lower subassembly are 316 stainless steel. The cladding and tubing diameter is 0.875 inch. The assemblies contain 7.2 pounds of stainless steel, 0.4 pound of beryllium, 0.2 pound each of nickel-based alloy springs, plutonium, and antimony.

The **in-core instrument assemblies** are located strategically about the reactor core in positions not designated for control element assemblies. The emitters are rhodium attached to Inconel 600 lead wires surrounded by an aluminum oxide insulator and sheathed in Inconel 600, which is 0.064 inches in diameter. Four emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a stainless steel 304 outer sheath which is 0.35 inch maximum diameter. These assemblies are 39 feet long, weigh 11.6 pounds, and have a life expectancy of 1200 EFPD. There are 28 instrumentation strings in the Fort Calhoun reactor.

St. Lucie 2 Class Reactor

The **control element assemblies** for the St. Lucie 2 reactors are 162.8 inches long and have five rods. Each rod is clad in Inconel 625. The cladding is 0.816 inch in diameter with a 0.035 inch wall thickness. The poison materials are 0.737 inch in diameter. The full-length control rods have 123.0 inches of B_4C and 12.5 inches of Ag-In-Cd as the control material on each of the five fingers. The part-length CEAs have 14.0 inches of B_4C and 68.5 inches of Inconel 625 at the tip of each of the five fingers. The plenum springs are 302 stainless steel. The part-length CEAs for St Lucie 2 weigh 83.0 pounds.

The **neutron source assemblies** for St. Lucie 2 appear to be identical to the NSAs for other CE reactors with 16 X 16 fuel (CE 16 X 16 and CE SYSTEM 80 Classes) and are described in the section of CE 16 X 16 Class components.

The **in-core instrument assemblies** are located strategically about the reactor core in positions not designated for control element assemblies. The emitters are rhodium attached to Inconel 600 lead wires surrounded by an aluminum oxide insulator and sheathed in Inconel 600, which is 0.064 inches in diameter. Four emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a Inconel 600 outer sheath which is 0.35 inch maximum diameter. These assemblies are 35 feet long, weigh 7.0 pounds, and have a life expectancy of 1200 EFPD. There are 56 instrumentation strings at St. Lucie 2.

Palisades Class Reactor

Palisades is a special case; the **control blade assemblies** are in the form of cruciform blades and are expected to last for the lifetime of the unit. There are 45 assemblies in the reactor with an overall length of 151

inches and a weight of 214 pounds. Each of the four blades extends 6.125 inches from the center line and ranges from 0.32 inch thick at the root to 0.18 inch thick at the edge. The absorber is Ag-In-Cd clad in 304 stainless steel. The control rod drive mechanism engages a hanger to withdraw and position the CEA. The hanger is made of 304 and 308 stainless steel. The stainless steel hanger weighs 62.2 pounds and the absorber 151.8 pounds.

Palisades uses two **startup (primary) neutron sources** and two **sustaining (secondary) neutron sources.** They are about 115 inches long, 0.34 inch in diameter, and clad with 304 stainless steel. The sustaining source is made of antimony-beryllium; it is 0.286 inch in diameter and 72 inches long. The startup source is the same size but consists of 12 inches of polonium-beryllium in the center with 30 inches of antimony-beryllium above and below. Each source assembly consists of 4.1 pounds of stainless steel and 0.2 pound of beryllium. The sustaining source contains 0.2 pound each of antimony and beryllium. The total weight of each is 4.5 pounds.

The **in-core instrument assemblies** are located strategically about the reactor core in positions not designated for control element assemblies. The emitters are rhodium, attached to Inconel 600 lead wires surrounded by an aluminum oxide insulator, and sheathed in Inconel 600, which is 0.064 inches in diameter. Four emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a stainless steel 304 outer sheath, which is 0.35 inch maximum diameter. These assemblies are 37 feet long, weigh 9.3 pounds, and have a life expectancy of 1200 EFPD. The Palisades reactor uses 61 instrumentation strings.

2.8.3.3 Westinghouse Reactors

Westinghouse (WE) is the NSSS vendor for seven assembly classes: WE 14 X 14, WE 15 X 15, WE 17 X 17, South Texas, San Onofre, Haddam Neck, and Yankee Rowe. Descriptions of NFA Hardware from these assembly classes follow.

Data on nonfuel components are primarily taken from Nuclear Fuel Data (Westinghouse 1986). This document was prepared by Westinghouse Electric Corporation for Martin Marietta Energy Systems and describes both the fuel assemblies and the nonfuel components manufactured by Westinghouse. Updated data, which resolve apparently conflicting information and address recent developments, are needed.

Westinghouse 14 X 14 Class Reactors

For WE 14 X 14 Class reactors, CRAs, NSAs, BAAs, and TPAs consist of 16 individual rods fastened by the upper ends to a spider or hold-down device.

Control rod assemblies for WE 14 X 14 Class reactors use a spider consisting of 6.25 pounds of 304 and 308 stainless steel and 1.61 pounds of Inconel X-750. The overall length of the CRAs is 156.6 - 158.5 inches. Two types of control rods have been identified. Standard CRAs use 142 inches of Ag-In-Cd, clad in 304 stainless steel, as the neutron absorber. These absorber rods are 152.7 inches long and weigh 7.59 pounds apiece. The overall length of the assembly is 158.5 inches and the weight is 129.4 pounds. A second type of CRA for WE 14 X 14 reactors is a part-length CRA. This CRA uses 118 inches of Ag-In-Cd, clad in 304 stainless steel, as the neutron absorber. These rods are 128 inches long and weigh 6.33 pounds apiece. The overall length and weight of these assemblies is given in WSTD-TME-148 as 156.6 inches and 128 pounds; these values seem larger than expected from component descriptions. A length less than 140 inches and a weight of about 110 pounds seems to be indicated.

Neutron source assemblies for WE 14 X 14 Class reactors consist of primary and/or secondary neutron source rods attached to the spider along with burnable absorber rods or thimble plugs. The source rods are typically 0.431 - 0.437 inches in diameter and clad in 304 stainless steel. These NSAs are fitted into fuel assemblies that are not located under CRAs. The spider consists of 7.8 pounds of 304 stainless steel and 0.5 - 1.1 pounds of Inconel springs.

Five configurations of **primary neutron source assemblies** for WE 14 X 14 class reactors have been identified. The overall length of these assemblies range from 143 - 158 inches. Each configuration has only one primary source rod. In addition to the primary source rod, each assembly contains 3 or 4 secondary source rods. The remaining 11 or 12 locations contain burnable absorber rods or thimble plugs. Two of the configurations use only thimble plugs with californium and plutonium-beryllium sources. These two assemblies weigh 26 and 28 pounds, respectively. The other three configurations use pyrex burnable absorber rods with californium, plutonium-beryllium, and polonium-beryllium sources. All five configurations use antimony-beryllium secondary source rods. These assemblies weigh 48 and 50 pounds.

Only one basic configuration of **secondary neutron source assemblies** has been identified for WE 14 X 14 Class reactors, although several variations of this configuration exist. These assemblies consist of 4 antimony-beryllium secondary sources and 12 thimble plugs attached to a spider assembly. Variations in the length of the secondary source rods give rise to overall length variations of 91 - 157 inches and corresponding weight variations of 20.7 - 27.7 pounds.

Two types of **burnable absorber assemblies** (rods) have been used in WE 14 X 14 Class reactors. The earlier BAAs utilized a borosilicate glass in the form of 142.6 - 142.7 inches of pyrex tubing as a neutron absorber. These rods have two diameters, 0.437 and 0.445 inches. They are clad in 304 stainless steel, have a overall length of 150.4 - 152.2 inches, and weigh 2.2 - 2.3 pounds. Eight configurations of pyrex BAAs have been identified, based on varying numbers of burnable absorber rods per assembly. Assemblies containing 1, 2, 4, 6, 8, 12,

14, and 16 burnable absorber rods have weights of 11, 13, 17, 21, 25, 32, 36, and 40 pounds, respectively. All appear to be 156.1 inches in length.

More recently **Wet Annular Burnable Absorber (WABA)** assemblies have been used. WABA rods are smaller in diameter (0.381 inches), shorter (150.1 inches), and lighter (1.8 pounds per rod). WABA rods use 134.0 inches of B_4C as the neutron absorber. Five configurations of WABA assemblies have been identified, based on varying numbers of WABA rods per assembly. Assemblies containing 4, 8, 9, 12, and 16 burnable absorber rods have weights of 15.6, 22.1, 23.4, 27.7, and 34.2 pounds, respectively. All appear to be 153.6 inches in length.

Thimble plugs used in NFA components for WE 14 X 14 Class reactors are made from 304 or 308 stainless steel and range from 0.434 - 0.502 inches in diameter, 6.56 - 8.02 inches in length, and 0.29 - 0.40 pounds. **Thimble plug assemblies** vary from 9.5 - 13.2 pounds in weight and from 10.8 - 13.6 inches in length.

Imaging and Sensing Technology Corporation (formerly a division of Westinghouse) provides in-core instrumentation, but these were not described in WSTD-TME-148.

Westinghouse 15 X 15 Class Reactors

For WE 15 X 15 Class reactors, CRAs, NSAs, BAAs, and TPAs consist of 20 individual rods fastened by the upper ends to a spider assembly or holddown device.

Control rod assemblies for WE 15 X 15 Class reactors use a spider consisting of 6.25 pounds of 304 and 308 stainless steel and 1.61 pounds of Inconel X-750. The overall length of the assemblies is 156.6 - 158.5 inches. Only one type of CRA for WE 15 X 15 Class reactors has been identified. Standard CRAs use 142 inches of Ag-In-Cd, clad in 304 stainless steel, as the neutron absorber. These absorber rods are 150.9 - 152.7 inches long and weigh 7.80 pounds apiece. The overall weight of the assembly is 165.0 pounds. Based on their use in WE 14 X 14 Class and WE 17 X 17 Class reactors, part-length CRAs have probably been used in WE 15 X 15 Class reactors.

Neutron source assemblies for WE 15 X 15 Class reactors consist of primary and/or secondary neutron source rods attached to a spider pack or holddown assembly along with burnable absorber rods or thimble plugs. The source rods are typically 0.431 - 0.437 inches in diameter and clad in 304 stainless steel. These NSAs are fitted into fuel assemblies that are not located under CRAs. The spider packs consist of 7.8 pounds of 304 stainless steel; the holddown assembly consists of 3.6 - 3.83 pounds of 304 or 308 stainless steel and 0.80 pound of 302 stainless steel, Inconel X-750, or Inconel 718 springs.

Five configurations of **primary neutron source assemblies** for WE 15 X 15 class reactors have been identified. The overall length of these assemblies range from 143 - 158 inches. Each configuration has only one primary source rod. In addition to the primary source rod, each assembly may contain secondary source rods and does contain burnable absorber rods and thimble plugs. The configurations are as follows:

- a californium source with 4 burnable absorber rods and 15 thimble plugs, weighing 20.2 pounds;
- a californium source with 12 burnable absorber rods and 7 thimble plugs, weighing 34 - 40.4 pounds;
- a plutonium-beryllium source with 1 antimony-beryllium secondary source, 16 burnable absorber rods, and 2 thimble plugs, weighing 43.5 pounds;
- a plutonium-beryllium source with 3 antimony-beryllium secondary source, 12 absorber rods, and 4 thimble plugs, weighing 52 pounds; and
- a polonium-beryllium source with 3 antimony-beryllium secondary source, 12 burnable absorber rods, and 4 thimble plugs, weighing 52 pounds.

Three basic configurations of **secondary neutron source assemblies** have been identified for WE 15 X 15 Class reactors. Each configuration uses 4 antimony-beryllium secondary source rods. The differences are in the number of burnable absorber rods and thimble plugs attached to the spider/holddown assembly as follows:

- 16 thimble plugs and a weight ranging from 19 - 33.6 pounds;
- 12 burnable absorber rods and 4 thimble plugs, weighing 52 pounds; and
- 7 burnable absorber rods and 9 thimble plugs, weighing 35.2 pounds.

Two types of **burnable absorber assemblies** have been used in WE 15 X 15 Class reactors. The earlier BAAs utilized rods containing borosilicate glass in the form of 142.6 - 142.7 inches of pyrex tubing as a neutron absorber. These rods have two diameters, 0.437 and 0.445 inches. They are clad in 304 stainless steel, have a overall length of 150.4 - 152.2 inches, and weigh 2.2 - 2.3 pounds. Ten configuration of pyrex BAAs have been identified, with 4 (17.0 - 20.4 pounds), 5 (20.0 pounds), 6 (no weight available), 8 (26.0 - 29.2 pounds), 9 (27.4 pounds), 10 (33.7 pounds), 12 (33.0 - 36.8 pounds), 13 (35.5 pounds), 16 (40.5 - 44.8 pounds), and 20 (50.0 - 53.7 pounds) rods per assembly. All are 156.1 - 156.9 inches in length.

More recently **Wet Annular Burnable Absorber** rods have been used. WABA rods are smaller in diameter (0.381 inches), shorter (150.1 inches), and lighter (1.8 pounds per rod). WABA rods use 134.0 inches of B_4C as the neutron absorber. Six configurations of WABA assemblies have been identified for WE 15 X 15 Class rectors, based on varying numbers of WABA rods per assembly. Assemblies containing 4, 6, 8, 12, 16, and 20 burnable absorber rods have weights of 16 - 19, 21, 22 - 24.3, 29 - 30, 34 - 39.4, and 46.8 pounds, respectively. Two overall lengths have been used, 146.6 and 153.7 inches.

Thimble plugs used in NFA components for WE 15 X 15 Class reactors are made from 304 or 308 stainless steel and range from 0.434 - 0.502 inches in diameter,

6.56 - 8.02 inches in length, and 0.29 - 0.40 pounds. **Thimble plug assemblies** vary from 9.5 - 13.2 pounds in weight and from 10.8 - 13.6 inches in length.

Imaging and Sensing Technology Corporation (formerly a division of Westinghouse) provides in-core instrumentation, but these were not described in WSTD-TME-148.

Westinghouse 17 X 17 Class Reactors

For WE 17 X 17 Class reactors, CRAs, NSAs, BAAs, and TPAs consist of 24 individual rods fastened by the upper ends to a spider assembly or holddown device.

Control rod assemblies for WE 17 X 17 Class reactors use a spider consisting of 4.80 - 7.6 pounds of 304 and 308 stainless steel and 1.65 pounds of Inconel X-750. The overall length of the assemblies is 160.9 inches. Five types of CRAs for WE 17 X 17 Class reactors have been identified. Four of these are full-length control assemblies and one is a part-length assembly. They are described below:

- A full-length CRA, using Ag-In-Cd as the absorber material and a lighter (8.6 pounds) spider assembly. These assemblies have a calculated weight of 149.5 pounds;
- A full length CRA, using Ag-In-Cd as the absorber material and a heavier (9.3 pounds) spider assembly. These assemblies have a calculated weight of 150.2 pounds;
- A full length CRA, using hafnium as the absorber material. These assemblies have a calculated weight of 181.2 pounds. These CRAs were typically used at standardized Westinghouse plants (Westinghouse 1972). As a result of several instances of irradiation induced swelling of these control elements, many reactors are replacing these assemblies with Ag-In-Cd CRAs;
- A full length, "hybrid" CRA, using 102 inches of boron carbide above 40 inches of Ag-In-Cd. These assemblies have a listed weight of 92.7 pounds; and
- A part length CRA, using 36 inches of Ag-In-Cd in the bottom of the absorber rod. A 106 inch aluminum oxide spacer is above the Ag-In-Cd. These assemblies have a listed weight of 100 pounds.

Neutron source assemblies for WE 17 X 17 Class reactors consist of primary and/or secondary neutron source rods attached to the spider along with burnable absorber rods and/or thimble plugs. The source rods are 0.385 inches in diameter and clad in 304 stainless steel. These NSAs are fitted into fuel assemblies that are not located under CRAs. The spiders consist of about 5 pounds of 304 stainless steel and 0.6 pound - 1.4 pounds of Inconel 718 springs.

Six configurations of **primary neutron source assemblies** for WE 17 X 17 class reactors have been identified. The overall length of these assemblies is 156 inches. Each configuration has one primary source rod. For WE 17 X 17 Class reactors, only californium has been used as the source material. One configuration has, in addition the primary source rods, 3 secondary (antimony-beryllium) source rods, 12 burnable absorber rods, and 8 thimble plugs. This configuration weighs between 37 and 39 pounds. Four other configurations use only burnable absorber rods and thimble plugs to fill the remaining 23 locations on the holddown device. These configurations have 12 (34 - 35 pounds), 16 (40 pounds), 20 (44 - 46 pounds), and 23 (51 pounds) burnable absorber rods. The other configuration uses only 20 thimble plugs (and no burnable absorber rods) and weighs 15 - 16 pounds.

Five configurations of **secondary neutron source assemblies** have been identified for WE 17 X 17 Class reactors. These assemblies consist of 4 antimony-beryllium secondary sources and an combination of burnable absorber rods and thimble plugs attached to holddown device. Configurations have been identified with 0 (17 - 24 pounds), 8 (29 pounds), 12 (35 - 39 pounds), 16 (41 - 45 pounds), and 20 (47 -51 pounds) burnable absorber rods. These assemblies also have an overall length of 156 inches.

Two types of **burnable absorber assemblies** have been used in WE 17 X 17 Class reactors. The earlier BAAs utilized a borosilicate glass in the form of 142 inches of pyrex tubing as a neutron absorber. These rods have a diameter of 0.385 inches. They are clad in 304 stainless steel, have a overall length of 152.4 inches, and weigh 1.8 pounds. Eleven configurations of pyrex BAAs have been identified with 3 (15.1 - 17.7 pounds), 4 (16.8 - 19.4 pounds), 5 (18.4 - 21.5 pounds), 6 (20.0 - 23.0 pounds), 8 (23.3 - 25.4 pounds), 9 (25.0 - 27.5 pounds), 10 (26.6 - 29.0 pounds), 12 (29.5 - 35.0 pounds), 16 (36.5 - 38.2 pounds), 20 (43.0 - 44.5 pounds), and 24 (49.0 - 50.3 pounds) burnable absorber rods.

More recently, **Wet Annular Burnable Absorber** assemblies have been used. WABA rods are smaller in diameter (0.381 inches), shorter (150.0 inches), and slightly heavier (1.9 pounds per rod). WABA rods use 134.0 inches of B_4C as the neutron absorber. Twelve configurations of WABA assemblies have been identified with 3 (15.2 - 17.8 pounds), 4 (16.9 - 19.5 pounds), 5 (17.8 - 20.9 pounds), 6 (19.5 - 22.4 pounds), 8 (23.1 - 26.0 pounds), 9 (24.6 - 27.2 pounds), 10 (26.2 - 28.7 pounds), 12 (29.6 - 32.4 pounds), 15 (36.4 pounds), 16 (36.3 - 38.2 pounds), 20 (43.5 - 49.8 pounds), and 24 (50.5 - 51.7 pounds) rods per assembly. All appear to be 153.6 inches in length.

Thimble plugs used in NFA components for WE 17 X 17 Class reactors are made from 304 or 308 stainless steel and range from 0.434 - 0.502 inches in diameter, 6.56 - 8.02 inches in length, and 0.29 - 0.40 pounds. **Thimble plug assemblies** attach 24 thimble plugs to a spider and vary from 9.5 - 13.2 pounds in weight and from 10.8 - 13.6 inches in length.

Imaging and Sensing Technology Corporation (formerly a division of Westinghouse) provides in-core

instrumentation, but these were not described in WSTD-TME-148.

South Texas Class Reactors

Fuel for the South Texas Class reactors is similar to fuel for a WE 17 X 17 Class reactor in that it uses a 17 X 17 array and is 8.4 inches wide. However, fuel for South Texas is longer than fuel for WE 17 X 17 Class reactors. Fuel for South Texas is approximately 199 inches long (compared to 159.7 inches for WE 17 X 17 Class fuel). No specific information of NFA hardware for South Texas has been obtained but information on NFA Hardware for WE 17 X 17 Class reactors should be descriptive.

San Onofre 1 Class Reactor

Fuel for the San Onofre 1 reactor is similar to fuel for a WE 14 X 14 Class reactor in that it uses a 14 X 14 array and is 7.76 inches wide. However, fuel for San Onofre 1 is shorter than fuel for WE 14 X 14 Class reactors. Fuel for San Onofre 1 is approximately 137.1 inches long (compared to 159.7 inches for WE 14 X 14 Class fuel). No specific information of NFA hardware for San Onofre 1 has been obtained but information on NFA Hardware for WE 14 X 14 Class reactors should be descriptive.

Haddam Neck Class Reactor

Fuel for the Haddam Neck reactor is similar to fuel for a WE 15 X 15 Class reactor in that it uses a 15 X 15 array and is 8.4 inches wide. However, fuel for Haddam Neck is shorter than fuel for WE 15 X 15 Class reactors. Fuel for Haddam Neck is approximately 137.1 inches long (compared to 159.7 inches for WE 15 X 15 Class fuel). No specific information of NFA hardware for Haddam Neck has been obtained but information on NFA Hardware for WE 15 X 15 Class reactors should be descriptive.

2.8.3.4 General Electric Reactors

General Electric (GE) is the NSSS vendor for five assembly classes: GE BWR/2,3, GE BWR/4-6, Big Rock Point, Dresden 1, and Humboldt Bay. Descriptions of the NFA hardware from these assembly classes follow.

GE BWR/2,3 and GE BWR/4-6 Class Reactors

Both General Electric and ASEA-Atom have supplied **control rod blades** for use in GE BWR/2,3 and GE BWR/4-6 Class reactors. Because of their shape, CRBs have often been referred to as cruciforms. While the CRBs for American BWRs have many similar physical characteristics, there are several important variations in neutron absorbing materials, methods of construction, and size differences based on lattice design.

The primary neutron absorbing material which has been used in CRBs is boron carbide. As operating experience has grown, both GE and ASEA have begun using hafnium as the absorbing material in high-exposure locations. A typical CRB is 9.75 inches wide, has a control length of 143 inches, and is positioned below the core in the reactor. The weight of the CRBs varies from 218 - 225 pounds. The end roller bearings were initially made of alloys which were high in cobalt (Stellite-3 or Haynes-25). The natural cobalt activates to ^{60}Co to levels high enough that these bearings were sometimes removed and packaged for use as ^{60}Co sources. The heat output from these bearings can present localized hot spots. These high-cobalt alloys are no longer being used, having been replaced with lower-cobalt alloys. ASEA has replaced the roller bearings with guide pads fabricated from Inconel X-750.

Original GE CRBs utilized vertical stainless steel tubes which were filled with boron carbide. These tubes were sheathed in a stainless steel jacket. In subsequent versions, GE replaced the boron carbide in three outside vertical tubes on each wing with hafnium rods and designated these as Hybrid I Control Rods (HICRs). Later, GE replaced the top several inches of the wings with a hafnium plate. These are designated as Advanced Long Life Control Rods (ALLCRs).

ASEA-Atom manufactures CRBs from solid stainless steel plates into which horizontal holes are drilled. These holes are approximately 4 inches deep and are filled with the neutron absorber (either boron carbide or hafnium). When closed, they are connected by a narrow slit, which allows the pressure between holes to be equalized without allowing movement of the absorbing material. This features eliminates concerns with settling (and reduction of reactivity control). ASEA initially replaced boron carbide in the top 19 horizontal tubes on each wing with hafnium metal rods to form its hafnium-tip (Model CR-82) CRB. Subsequently, the boron carbide on the outside half-inch of the wings was replaced with hafnium on the top 75% of the "ultimate" (Model CR-85) CRB. Both the CR-82 and CR-85 are available in "matching" and "highworth" designs. "Highworth" versions are designed for 5 to 15% (relative) higher rod worth than original equipment rods. "Matching" versions have up to 5% (relative) higher rod worth.

Both GE and ASEA CRBs also have also variations according to the lattice design of the reactor core. All GE BWR/2 and GE BWR/3 reactors, and many GE BWR/4 reactors use a D lattice core. In D lattice cores, the spacing between assemblies on the sides adjacent to the CRB is greater than the spacing between assemblies on the side away from it. GE BWR/4 reactors that do not use D lattice core and all GE BWR/5 reactors use C lattice cores. C lattice cores have equal spacing between assemblies on all sides. GE BWR/6 reactors use S lattice cores, which have equal spacing between assemblies, but which use the slightly smaller fuel assembly width specific to GE BWR/6 reactors. Each design of CRB described here could have been manufactured for D, C, or S lattice

cores. The primary distinction between the design of the CRB for different lattices is the width of the wing.

GE uses five to seven **neutron source rods** per reactor. A mixture of antimony-beryllium has been used as the neutron source for these rods, including those neutron source rods used for the initial startup of the reactor. Each source rod consists of two antimony rods within a single beryllium cylinder. Both the antimony and beryllium are encased in stainless steel tubes. No weights or dimensions are available at this time.

Temporary poison curtains made of borated (3800 - 5400 ppm boron) stainless steel sheets were used to control reactivity in the initial core of early reactors. These curtains, which are no longer used, were 141.25 inches long, 9.20 inches wide, and 0.0625 inch thick, were placed between fuel assemblies in water gaps without control rods. The weight is not given, but density considerations indicate a mass of about 25 pounds. Currently, GE incorporates integral gadolinia absorbers into selected fuel rods to provide reactivity control.

GE reactors use four types of **in-core instrumentation**: source range monitors, intermediate range monitors, and local power range monitors (LPRMs), and transverse incore probes (TIPs). Most reactors seem to have four source range monitors and four intermediate range monitors, which are used during periods of low neutron fluxes (i.e. during reactor shutdowns and startup). Recently, wider range neutron monitors have been introduced which replace both source and intermediate range monitors. The number of LPRMs is dependent of the number of control-rod groups. One four-element LPRM is at the center of each four-control-rod group. At least three vendors (GE, Imaging and Sensing Technology Corporation, and ABB Atom) supply in-core instrumentation for BWRs. LPRMs are approximately 40 ft. long, 13.5 ft. of which is in the core. The total weight is approximately 48 lbs., 8 lbs. of which is in the core. The diameter of the end in the core is approximately 0.75 in; the out of core end is approximately 1.05 in. in diameter.

BWR **fuel channels** have been supplied to domestic reactors by at least three suppliers: General Electric, Carpenter Technology of San Diego, and ABB Atom. As the primary supplier of BWR fuel in the American market, GE has supplied the preponderance of these channels. Since only information on GE channels has been obtained, this discussion focuses only on GE channels. Early versions of BWR fuel for Dresden 1, Humboldt Bay, and Big Rock Point used various materials (stainless steel, Zircaloy-2, and Zircaloy-4) for channels. By the advent of the BWR/2 reactors (Nine Mile Point 1 and Oyster Creek), Zircaloy-4 had become the standard channel material. Since that time, Zircaloy-4 has continued to be used for the manufacture of fuel channel. New methods of treatment (e.g., annealing, β-quenching, and autoclaving) have been developed to increase the strength and corrosion resistance of the channels.

Available data on fuel channel dimensions comes primarily from engineering drawings furnished by GE on six channel designs and from discussions held with GE during a visit to the GE fuel fabrication plant in Wilmington, NC (Moore 1991a). Fuel channels for GE BWR/2,3 Class reactors have an overall length of 162.2 inches. Fuel channel length for BWR/4 and BWR/5 reactors is 166.9 inches and 167.4 inches for BWR/6 reactors. Fuel channel thickness had historically been 0.080 inches for fuel for BWR/2, BWR/3, BWR/4, and BWR/5 reactors when GE introduced the BWR/6 reactor design in 1972. The BWR/6 introduced the 8 X 8 fuel rod array and a thicker (0.120 inch) fuel channel. In order to keep the exterior dimension of the fuel assemblies (5.438 inches with channels attached) the same as BWR/2-5 fuel assemblies, GE reduced the pitch of the fuel rods in the BWR/6 design, reducing the exterior dimension of the fuel bundle (without the channel) from 5.278 inches to 5.198 inches. Since the introduction of the BWR/6 reactor design, channel thickness for most BWR/4 and BWR/5 reactor fuel (and presumably some BWR/2 and BWR/3 reactors fuel) has increased from 0.080 to 0.100 inches, increasing the outside dimension to 5.478 inches. Recently, GE introduced an "interactive" fuel channel. This channel reduces neutron losses by reducing the thickness of the side flats to 0.065 inches, while keeping the corners 0.100 inch thick for strength. Thus, eight basic types of channels have been identified. Many subtypes exist, and some variations could be measurably different from those described below:

- 80 mil BWR/2 and BWR/3 channel - 162.2 inches long, about 64 pounds
- 80 mil BWR/4 and BWR/5 channel - 166.9 inches long, about 66 pounds
- 100 mil BWR/2 and BWR/3 channel - 162.2 inches long, about 80 pounds
- 100 mil BWR/4 and BWR/5 channel - 166.9 inches long, about 82 pounds
- 120 mil BWR/6 channel - 167.4 inches long, about 98 pounds
- Interactive BWR/2 and BWR/3 channel - length and weight undetermined
- Interactive BWR/4 and BWR/5 channel - length and weight undetermined
- Interactive BWR/6 channel - length and weight undetermined

A separate issue with respect to fuel channels is the width of the channel, including the spacer buttons and the channel fastener clips. Informal discussions at GE indicated that the maximum width of an unirradiated fuel channel, including the spacer button, channel fastener, and adapter plate appeared to be 5.798 inches. This may not be an absolute maximum, but should be close. The maximum width of the newest fuel channel was 5.795 inches, and the maximum width of a typical 80-mil channel was 5.768 inches. In drawings furnished by GE, the maximum width shown was 5.771 inches.

Big Rock Point, Dresden 1, and Humboldt Bay Class Reactors

No information has been obtained on NFA hardware for the Big Rock Point, Dresden 1, and Humboldt Bay Classes. However, the NFA hardware should be similar to the NFA hardware used in other BWRs.

2.8.3.5 Allis-Chalmers Reactors - LaCrosse Class Reactor

Allis-Chalmers is the NSSS vendor for one assembly class: LaCrosse. No information has been obtained on NFA hardware for the LaCrosse Class reactor. However, the NFA hardware should be similar to the NFA hardware used in other BWRs.

2.8.4 Estimated Quantities of NFA Hardware

No single reliable source of data on the quantities of NFA Hardware generated by commercial reactors is available. This is probably because NFA hardware has for many years been viewed as a minor contributor to the wastes to be disposed of by the CRWMS. While discharged and in-core fuel data are collected from the utilities by the EIA each year, similar information has not been collected on NFA hardware in the past. The issue of quantities of NFA hardware is further complicated by the fact that the amounts and types of NFA hardware used by commercial reactors varies from reactor to reactor, and also changes with time. In addition, several utilities have disposed of some NFA hardware components as Low-level Waste. In this section, the historical and present usage of NFA hardware components is described, future trends are speculated on, and crude estimates of the quantities of NFA hardware are made and compared to the data which are available.

An overview of the estimates of quantities of NFA hardware that will be discharged from commercial reactors in the U.S. is given in Table 2.8.2. These data have been normalized to a "number per 100 assemblies" basis. The rationale for this is that the number of discharged assemblies is a realistic measure of reactor operating time and EFPD. It is also a number that is readily obtainable.

Several methods for estimating NFA quantities are used. Data from the vendors on quantities of NFA hardware shipped to the reactors are used, along with quantities of fuel shipped to the reactors to obtain one estimate. Data derived from knowledge of NFA hardware usage are also used to estimate quantities. Vendor-supplied component lifetime data are used in conjunction with reactor operations data for another estimate. Finally, the Edison Electric Institute requested data from the utilities on nonfuel components usage, to which 52 reactors responded, with varying degrees of completeness (Farrell 1990). The results of this survey, along with historical and projected fuel discharge data, were also used to estimate NFA components usage. All fuel discharge information used in making these estimates are from the EIA RW-859 Data Base (EIA 1989), for CY-1989 data.

2.8.4.1 Babcock & Wilcox Reactors

Data supplied by B&W listed shipments of fuel assemblies and NFA hardware components to the reactors it built (Cooper 1986). A summary of this information is presented in Table 2.8.3. All of the operating reactors in the US which were built by B&W are B&W 15 X 15 Class reactors. Several trends are apparent upon examination of these data:

- Most reactors received an initial complement of NFA hardware consisting of 61 CRAs, 68 BAAs, 8 APSAs, 2 primary NSAs, and 108 ORAs. For the second cycle, 2 regenerative (secondary) NSAs were shipped to each reactor. After 7 or 8 cycles, the regenerative NSAs appear to be replaced.
- After 4 or 5 cycles, the eight APSAs are replaced. Given the trends at the other reactors, the 24 APSAs shipped to Oconee 1 in November of 1979 may have been used at Oconee 1, Oconee 2, and Oconee 3.
- BAAs, originally designed to be used in startup operations, appear to be have been used regularly since 1979 for additional reactivity control. Approximately 93 BAAs have been shipped to reactors (since 1979) for every 100 fuel assemblies. While Three Mile Island (to 1986) had not begun using BAAs on a regular basis, this is likely the result of the extended shutdown of this unit after the TMI-2 incident in 1979. Reload information for TMI-1 cycle 6 indicates the use of 68 BAAs (B&W 1986).
- Shipments of replacement CRAs are not extensive, although some appear to have been replaced intermittently. Larger numbers of CRAs were shipped to Oconee 3 and Arkansas Nuclear One, Unit 1 after about seven cycles of operation, potentially indicating the gradual replacement of the CRAs there. This is generally consistent with the supposition that CRAs for PWRs will be replaced one or twice during a standard 40-year reactor lifetime.
- Only very limited numbers of ORAs have been replaced and later reactors received fewer than the earlier reactors. Combined with the trend in PWRs toward removing these assemblies from use in the core, this indicates limited numbers requiring disposal.

These observations are used to estimate the quantities of NFA hardware based on the following assumptions:

- A total of 10032 fuel assemblies are projected to be discharged from B&W 15 X 15 Class reactors.
- A maximum number of CRAs can be estimated by comparing the number of CRAs shipped to the number of fuel assemblies shipped. This works out

to approximately 12 CRAs and 3 APSAs for every 100 fuel assemblies.

- An estimated 93 BAAs are used per 100 fuel assemblies for fuel inserted after the use of fresh BAAs in each cycle begins. Prior to this, the actual number of BAAs shipped (554) is divided by the total number of fuel assemblies shipped to that point (2817). This gives an estimate of 554 from early use and 6710 form regular use [0.93 * (10032 - 2817)], for a total of 7264 BAAs. Overall, this works out to 72 BAAs per 100 fuel assemblies.
- The number of ORAs shipped through 1986 (784) is taken as the total number which the CRWMS must dispose. (8 ORAs per 100 fuel assemblies).
- An estimated 2 primary and 8 secondary NSAs are assumed per reactor, for a total of 80. (0.8 NSAs per 100 fuel assemblies.)

Two B&W 15 X 15 Class reactors responded to the EEI survey. Based on EIA estimates, these two reactors (Davis-Besse and Crystal River 3) will discharge an estimated 2460 fuel assemblies during their lifetimes. Estimates given by the reactors were 165 CRAs, 40 APSAs, and 2268 BAAs. This is 7 CRAs, 2 APSAs, and 92 BAAs per 100 fuel assemblies. One reactor, however, listed future discharges of only 8 CRAs, which is almost certainly low. These reactors also reported a total of 50 ORAs and 8 NSAs as discharged and no projected discharges.

Based on these estimates, factors for B&W Class reactors are as follows:

Estimated NFA Hardware from B&W Reactors
(items per 100 discharged fuel assemblies)

NFA Type	Shipping Record	EEI Survey	Value Used	Rationale
BAA	72	92	82	Average
APSA	3	2	3	Larger
CRA	12	7	12	Better Value
ORA	8	-	8	Only Estimate
NSA	0.8	-	0.8	Only Estimate

2.8.4.2 Combustion Engineering Reactors

Combustion Engineering gives an estimate of lifetimes of 4000 EFPD, 4000 or 3500 EFPD, and 1200 EFPD for its CEAs, NSAs, and instrument strings, respectively (Hayduk 1986). Using these estimates, fuel discharge data from the EIA, and the number of NFA components per core, "lifetime" estimates of the usage of NFA hardware in Combustion Engineering reactors are obtained. These estimates are given in Table 2.8.5 and are based on intermediate calculations given in Table 2.8.4. The CE reactors on which these estimates are made will discharge 27,183 assemblies. On this basis, quantities of NFA hardware are estimated as 13 CRAs, 0.3 NSAs, and 23 instrument strings per 100 discharged fuel assemblies.

Seven CE-built reactors responded to the EEI survey regarding NFA hardware usage. Five reactors gave estimates on CEA usage. Based on the future discharges from these reactors, the EEI survey reports about 14 CEAs per 100 fuel assemblies. Of these, three reactors estimated the use of 4 sets of CEAs over the reactor lifetime; the "lifetime" estimate assumed 3 sets. Another reactor reported the only current quantities in storage; the value reported agrees well with the "lifetime" estimate if one assumes that the second of three sets of CEAs is currently in-core.

Four reactors gave estimates on quantities of NSAs and instrument strings. For these four reactors, the "lifetime" estimate gives 26 NSAs and 1970 instrument strings. While the number of NSAs is substantially below the reactor estimate of 87 (1.0 neutron sources per 100 assemblies), the number of instrument strings agrees reasonably well with the reactor estimate of 1615 strings (18 per 100 assemblies).

Based on these estimates, factors for CE Class reactors are as follows:

Estimated NFA Hardware from CE Reactors
(items per 100 discharged fuel assemblies)

NFA Type	Lifetime Estimate	EEI Survey	Value Used	Rationale
CEA	13	14	13	Better Value
NSA	0.3	1.0	1.0	Larger Value
INST	23	18	23	Larger Value

2.8.4.3 Westinghouse Reactors

Information supplied by Westinghouse (Westinghouse 1986) with respect to CRAs indicates that, through June 30, 1986, Westinghouse had shipped 2841 assemblies for WE 14 X 14 Class reactors and San Onofre Class reactors, 5967 assemblies for WE 15 X 15 Class reactors and Haddam Neck, and 8881 assemblies for WE 17 X 17 Class reactors. Quantities of CRAs shipped are 234, 514, and 1467 for these same groupings, representing 8.2%, 8.6%, and 16.5%, respectively (or 13 CRAs per 100 fuel assemblies). It should be noted that the percentage for WE 17 x 17 Class reactors is artificially high. This is because WE 17 x 17 Class reactors are still being brought on-line (at least 10 of these reactors had not discharged any fuel by the end of 1986) and the amount of fuel discharged is very low, relative to the number of CRAs shipped to reactor sites.

Data on WE CRAs from the EEI report is available for 19 reactors. These reactors project 2296 CRAs. These 21 plants will discharge a total of 34,322 fuel assemblies over their lifetimes. Adjusting this number by a factor of 0.90 (to represent the number of reactors reporting), approximately 8% of the assemblies will contain CRAs.

Data with regards to BAAs indicates that the use of burnable absorbers in Westinghouse-built reactors has

been similar to their use in B&W-built reactors (about 20 per 100 assemblies). Early Westinghouse reactors used borosilicate glass BAAs, presumably primarily in the first few cycles. Around 1979, WE introduced a new type of BAA called a WABA rod. WABA rods use a B_4C-Al_2O_3 absorber material in an annular rod. The rod is inserted into the fuel assembly guide tube and reactor coolant flows through the annular portion. WABA rods and assemblies have received significant usage since that time. At the Farley reactor site during the period 1986 to 1989, 387 fresh fuel assemblies were inserted (McDonald 1986a,b, 1988, Hairstan 1987, 1989a,b). During this same period of time, 4528 burnable absorber rods were also inserted. This represents a minimum of 190 BAAs, and is, more probably, representative of 380 BAAs (based on 24 and 12 rods per assembly, respectively). This is about 98 BAAs per 100 fuel assemblies. We use 95 BAAs per 100 fuel assemblies, taking the 92% figure from B&W into account. Currently, integral burnable absorbers (in the form of a zirconium diboride coating on selected fuel pellets) are being used in nearly one-third of Westinghouse-built plants (Kramer 1989). If this trend continues, we expect the number of BAAs to decrease. Estimates of the total number of BAAs from WE reactors are based upon the number of discharged assemblies and the estimated percentage of assemblies with BAA from several time periods, as shown in Table 2.8.6. Based on this data, approximately 28% of the assemblies from WE-built reactors might have BAAs.

A second method for estimating the usage of BAAs is from WE shipping records. Quantities of BAAs shipped to reactors using array sizes of 14 X 14, 15 X 15, and 17 X 17 (through June 1986) were 558, 1499, and 2946, representing 20%, 25%, and 33%, respectively, of the number of assemblies shipped (28% overall).

Comparison to the number of BAAs projected by the WE-built reactors responding to the EEI survey also serves as a check. Fourteen of the twenty-one WE reactors that responded to the survey gave both historical and projected discharges that appeared to be real. These reactors reported a total of 13,177 BAA discharges [One reported number, 2298 for Plant No. 17, was an obvious typographical error. 298 BAAs were used for this reactor]. These 21 plants will discharge a total of 34,322 fuel assemblies over their lifetimes. Adjusting this number by a factor of 0.67 (to represent the number of reactors reporting), indicates that reactors will discharge approximately 57 BAAs per 100 fuel assemblies.

Thus, estimates from the reactors seem to be approximately double the estimates that are obtained from actual shipping records and estimates of usage. This is not surprising, however, in that the actual shipping record covers mostly time before the full-scale implementation of recurring BAA usage. Additionally, the estimate based on usage data assumes a significant reduction in the number of BAAs in 1995. This implies that assemblies being inserted into reactors at the present time have largely ended the use of BAAs. On the other hand, the estimates by the reactor may not fully account for the increasing use of integral burnable absorbers. Therefore, the average of these results (43 BAAs per 100 fuel assemblies) will be used.

An estimate of the quantity of TPAs comes from the Westinghouse report. As with the B&W ORAs we assume that the number shipped as of June 1986 is that total number that the CRWMS will have to deal with. For array sizes 14 X 14, 15 X 15, and 17 X 17, 1680, 3166, and 3362 TPAs have been shipped. This represents 19%, 20%, and 6%, respectively (10% overall).

In the EEI survey, 17 of the 21 reactors which responded furnished both historical and projected quantities of TPAs discharges. These reactors reported a total of 1775 TPAs discharges. Adjusting the number of fuel assemblies from these reactors by a factor of 0.81 (to represent the number of reactors reporting), approximately 6% of the assemblies will contain TPAs.

Since the Westinghouse reports identifies actual TPAs that have actually been shipped, an estimate of 10 TPAs per 100 fuel assemblies will be used.

Based on these estimates, factors for WE Class reactors are as follows:

Estimated NFA Hardware from WE Reactors
(items per 100 discharged fuel assemblies)

NFA Type	Shipping Record	EEI Survey	Value Used	Rationale
BAA	28	57	43	Average
CRA	13	8	8	Better Value
ORA	10	-	10	Only Estimate

2.8.4.4 General Electric Reactors

Estimating the quantities of NFA hardware from BWR reactors is even more uncertain than for PWRs. Part of the reason for this difficulty is that undetermined portions of the BWR NFA hardware has been disposed of as Low-level waste. Of the 19 GE-built BWRs that responded to the EEI survey,[3] 10 reported that they had disposed of one or more of the following types of NFA hardware: CRBs, LPRMs, neutron sources, fuel channels, and poison curtains. For this reason, in estimating quantities of NFA hardware from the EEI report, only projected quantities have been used.

Data on CRBs from the EEI report are available for 16 reactors. These reactors project the discharge of 5659 CRBs. The 19 plants will have future discharges totaling 63,271 fuel assemblies over their lifetimes. Adjusting this number by a factor of 0.84 (to represent the percentage of reactors reporting), the number of CRBs is approximately 11 per 100 fuel assemblies.

Data on LPRMs from the EEI report is available for 15 reactors. These reactors project the discharge of 3792 LPRMs. Adjusting the number of projected future fuel assembly discharges by a factor of 0.79 (to represent the percentage number of reactors reporting), the number of LPRMs is approximately 8 per 100 fuel assemblies.

In the EEI survey, 12 of the 19 GE BWR reactors which responded furnished both historical and projected quantities of discharged channels.. These reactors reported a total of 59,638 channels. Adjusting the number of fuel assemblies from these reactors by a factor of 0.63 (to represent the number of reactors reporting), the number of channels is approximately 112% of the number of assemblies to be discharged. This implies that more than one channel will be used on some fuel assemblies. However, if only historical discharges are considered, the number of fuel channels is approximately 93% of the number of assemblies. At one time, the reuse of fuel channels was considered to be viable, but problems (NRC 1990) with bowing in reused channels will severely limit this option. It is likely that each assembly will use a single fuel channel and that the use of more than one channel on an assembly or reuse of channels will be rare. In this light, an estimate of 100 fuel channels per 100 fuel assemblies will be used.

Supporting information regarding the use of NFA hardware is available. At the Susquehanna Power Station, 100 CRBs and 72 LPRMs will be characterized, processed, packaged, transported, and disposed of as LLW by Chem-Nuclear (Nuclear News 1991). To date, the two Susquehanna reactors have discharged 1720 fuel assemblies. Thus, reasonable lower limits for the number of CRBs (6 per 100 fuel assemblies) and LPRMs (4 per 100 fuel assemblies) can be estimated.

Since the EEI data comes from multiple plants, those estimates will be used as follows:

Estimated NFA Hardware from GE Reactors
(items per 100 discharged fuel assemblies)

NFA Type	EEI Survey	N. News Value	Value Used	Rationale
CRB	11	6	11	Better Value
LPRM	8	4	8	Better Value
CHAN	112	-	100	Better Value

2.8.5 Radiological Properties of NFA Hardware

Because NFA hardware is not physically attached to the fuel assemblies, it is not necessarily discharged from the core on the same schedule as the assemblies. Some NFA hardware is in the core for only a single cycle; other NFA hardware is designed to last for the life of the reactor. Additionally, some hardware resides within the active core while some is normally positioned outside the fueled region. Other hardware is moved between the two regions routinely. For these reasons, radiological characterization of NFA hardware requires procedures tailored to suit the various cases. A method for estimating the radiological properties of NFA hardware, based on the application of previously calculated zone factors[4] to ORIGEN2 data, has been developed. In the method, radiological properties of irradiated NFA hardware are based upon the materials of construction, a weighted neutron flux scaling factor, and the duration of the component's exposure to the neutron flux. Experimental radiological characterization of NFA hardware that is currently ongoing at Pacific Northwest Laboratories will be available in the future and will provide a basis for checking these calculated parameters and, if appropriate, adjusting the values.

Because of the severe conditions which materials are exposed to in the core of a nuclear power reactor, relatively few materials have been used in the fabrication of NFA hardware. These materials are alloys of zirconium (Zircaloy-2 and Zircaloy-4), alloys of nickel (Inconel-625, Inconel-718, and Inconel X-750), stainless steels (stainless steel 304), and strongly neutron-absorbing materials (hafnium, a silver-indium-cadmium alloy, and boron carbide - both in powder form and contained in an aluminum oxide matrix). The elemental compositions of many of the structural materials used in NFA hardware are given in Table 2.7.3. Because they are the precursors to the isotopes that determine low-level waste classifications, the initial amounts of nitrogen, nickel, cobalt, and niobium in these materials are of particular interest. Nitrogen in the materials is activated to ^{14}C; natural nonradioactive isotopes of nickel are activated to ^{59}Ni and ^{63}Ni; natural niobium activates to ^{94}Nb; each of these isotopes must be considered towards the Class C limit. For niobium in particular, trace quantities may be sufficient for the material to exceed the Class C limits after irradiation. The amount of cobalt in the material is a concern only for dose rate considerations; cobalt is not a factor in Class C limit calculations. Since these trace quantities do not significantly affect the physical and chemical characteristics of the material, the actual amount may not be measured by the ingot manufacturer or hardware fabricator. As with SFD hardware and intact fuel assemblies, the radiological characterization of materials irradiated in the core of the reactor is based on ORIGEN2 calculations. ORIGEN2 radionuclide inventories for many of the materials (SS304, Zircaloy-2 and -4, Inconel-718 and X-750) used in NFA hardware have been developed and are described in Section 2.7.3. ORIGEN2-calculated inventories for other materials (Ag-In-Cd, B_4C/Al_2O_3 mixtures, etc) are scheduled for development and will be published separately. The results of these calculations are directly applicable to materials irradiated in the core's fueled region.

Outside the fueled region, the ORIGEN2 results must be scaled down because of a decrease in the neutron flux and shifts in the neutron spectrum. For NFA hardware, as with SFD hardware, much of the material of interest is located outside the fueled region, at or beyond the end fittings. To estimate the radionuclide inventory in these components, scaling factors are used to compensate for the decreased neutron flux outside the fueled region. The scaling factors used in these calculations for the top end fitting, gas plenum, and bottom end fitting zones are based upon the data of Luksic (Luksic 1989). These factors, which have an uncertainty of $\pm 50\%$, are presented in Table 2.8.7.

Because much of the NFA hardware is located beyond the top and bottom end fittings of the fuel assemblies, two additional neutron exposure zones have been used. In both zones, the neutron scaling factor is assumed to be 10% of the applicable neutron scaling factor in the zone adjacent to it. These flux scaling factors are also presented in Table 2.8.7. These factors are intended for average radionuclide inventory estimates, and should not be applied to small sections due to significant variations in the neutron flux with respect to even a small change in position.

In order to simplify the application of these factors, weighted scaling factors have been developed for each type of NFA hardware used by each of the major reactors vendors. These weighted scaling factors represent the neutron flux, relative to the core flux, that each type of hardware receives. It is obtained by summing the products of the scaling factor by the percentage of the NFA hardware component in that exposure zone. These percentages were obtained by analysis of information provided by the fuel vendors on NFA hardware (Cooper 1986, Hayduk 1987, Westinghouse 1986). These percentages, as well as the weighted scaling factors are presented in Table 2.8.8.

The weighted scaling factors represent only the relative flux. Since the total exposure to the neutron flux is also a function of the duration of the exposure to the neutron flux, allowance for the number of cycles of irradiation must be made. Table 2.8.9 presents the estimated number of cycles of irradiation for each type of hardware, the number of cycles per assembly lifetime, the weighted scaling factors, and an overall exposure factor. The overall exposure factor is obtained by multiplying the number of cycles of irradiation by the weighted scaling factor and dividing by the number of cycles per assembly lifetime. The overall exposure factor represents the averaged exposure to the neutron flux in the reactor core relative to an assembly lifetime. Thus, when multiplied by the ORIGEN2 radionuclide inventories for materials irradiated incore for one assembly lifetime, the overall exposure factor gives the estimated radionuclide inventories for the NFA hardware. Components which are largely in the incore zone for longer than an assembly lifetime will have overall exposure factors greater than unity.

2.8.6 References for Section 2.8

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Table 2.8.1 Sample Physical Description Report from LWR NFA Hardware PC Data Base

Physical Description Report Page 1

Assembly Class: CE SYSTEM 80
Component Name: 12 Rod Full-Length Control Element

Designed for:
Fuel Assembly with array size: 16 X 16
Pressurized Water Reactor

Dimensions:
Total Length: 253 inches
Total Weight: 192.2 pounds

Cladding:

Material:	Inconel 625
Outer Diameter:	0.816 inches
Wall Thickness:	0.035 inches
Diametral Gap:	0.009 inches

Absorber:

Primary Material:	B4C (CE)
Absorber Length:	148 inches
Pellet Diameter:	0.737 inches

Plenum Spring Material: St.Steel 302

Spider Material: St.Steel 304

Number of Control Rods: 12

Life Expectancy: 4000 Effective Full Power Days

Comments:
These CEA's fit simultaneously into 5 assemblies in the core.

Table 2.8.1 (cont.) Sample Physical Description Report from LWR NFA Hardware PC Data Base

Physical Description Report Page 2

Assembly Class: CE SYSTEM 80
Component Name: 12 Rod Full-Length Control Element

Composition:

Material	Weight (kg)	Neutron Zone
St. Steel 304	8.17	Above Top End Fitting
Inconel 625	53.62	Above Top End Fitting
Boron Carbide	20.90	Above Top End Fitting
St. Steel 304	8.17	Top End Fitting
Inconel 625	53.62	Top End Fitting
Boron Carbide	20.90	Top End Fitting
St. Steel 304	0.68	Gas Plenum
Inconel 625	2.20	Gas Plenum
Boron Carbide	1.60	Gas Plenum

Usable at the Following Reactors:

Palo Verde 1 Palo Verde 2 Palo Verde 3

Fuel Assembly Types Usable at these Reactors:

CE SYSTEM 80 16 X 16 CE

Table 2.8.2 Summary of estimated quantities of NFA hardware.

Reactor Vendor	Type of NFA Hardware	Number per 100 Assemblies	Estimated Number of Assemblies*	Total Pieces
B&W	Burnable Absorbers	82	13,145	10,800
	Axial Power Shaping Assemblies	3		394
	Control Rod Assemblies	12		1,580
	Orifice Rod Assemblies	8		1,050
	Neutron Source Assemblies	0.8		105
	Instrumentation	NA		NA
CE	Control Element Assemblies	13	27,183	3,530
	Neutron Sources	1.1		299
	Instrument Strings	23		6,250
WE	Control Assemblies	8	87,143	6,970
	Burnable Absorber Assemblies	43		37,500
	Thimble Plug Assemblies	10		8,714
	Instrumentation	NA		NA
GE	Cruciform Control Blades	11	155,181	17,100
	Local Power Range Monitors.	8		12,400
	Fuel Channels	100		155,181
	Other Instrumentation	NA		NA

* Based on 1989 EIA RW-859 Historical Discharges and No New Orders Case Projections through 2040 (Reference 12). Data for B&W include B&W 15 X 15 and B&W 17 X 17 Class reactors. Data for CE include CE 14 X 14, CE 16 X 16, CE SYSTEM 80, Ft. Calhoun, and St. Lucie 2 Class reactors. Data for WE include WE 14 X 14, WE 15 X 15, WE 17 X 17, SOUTH TEXAS, San Onofre 1, and Haddam Neck Class reactors. Data for GE include GE BWR/2,3 and GE BWR/4-6 Class reactors. NFA hardware estimates for Big Rock Point, Dresden 1, Humboldt Bay, Indian Point 1, LaCrosse, Palisades, and Yankee Rowe Class reactors are not included in this table.

Table 2.8.3 Shipment records of Fuel Assemblies and NFA hardware components from B&W to B&W-built reactors.[6]

Reactor	Cycle	Shipment Date	Number of FAs	Number of CRAs	Number of APSAs	Number of BAAs	Number of ORAs	Number of RNSAs
Ark. Nucl. One 1	1	3/74	177	61	8	68	108	-
	2	3/77	56	-	-	-	-	2
	3	11/77	56	-	-	-	-	-
	4	1/79	64	-	-	56	-	-
	5	11/80	68	-	8	64	-	-
	6	10/82	72	-	-	64	-	-
	7	11/84	68	12	-	64	-	-
	8	4/86	64	13	-	64	-	-
Crystal River	1	5/76	177	61	8	68	72	-
	2	11/78	56	-	-	-	3	2
	3	3/80	61	-	-	-	-	-
	4	6/81	51	-	-	64	-	-
	5	7/82,5/83	76	-	-	60	-	-
	6	2/85	60	-	8	40	-	-
Davis-Besse	1	11/76	177	53	8	68	57	-
	2	4/80	52	-	-	-	-	2
	3	8/81	40	-	-	-	-	-
	4	6/83	48	-	-	-	-	-
	5	6/84	64	-	-	64	-	-
	6	8/86	64	-	8	64	-	-
Oconee 1	1	2/73	177	61	8	68	108	-
	2	7/74	61	-	-	-	-	2
	3	10/75	60	-	-	-	-	-
	4	5/77	61	-	1	-	-	-
	5	8/78	56	-	-	-	-	-
	6	11/79	48	8	24 *	64	-	-
	7	7/81	68	-	-	60	-	-
	8	6/83	65	-	-	60	-	-
	9	10/84	64	-	10	60	-	-
	10	1/86	60	-	-	52	-	2

FA - Fuel Assemblies
CRA - Control Rod Assemblies
APSA - Axial Power Shaping Assemblies
BAA - Burnable Absorber Assemblies
ORA - Orifice Rod Assemblies
RNSA - Regenerative Neutron Source Assemblies

* Appears to be 3 sets of 8 APSAs; one set for Oconee 1, one set for Oconee 2, and one set for Oconee 3.

Table 2.8.3 Shipment records of Fuel Assemblies and NFA hardware components from B&W to B&W-built reactors.

Reactor	Cycle	Shipment Date	Number of FAs	Number of CRAs	Number of APSAs	Number of BAAs	Number of ORAs	Number of RNSAs
Oconee 2	1	10/73	177	61	8	68	108	-
	2	6/76	56	-	-	-	-	2
	3	5/77	56	-	-	-	-	-
	4	10/78	56	-	-	-	-	-
	5	1/80	68	-	-	56	-	-
	6	10/81	72	-	-	64	-	-
	7	9/83	72	-	-	64	-	-
	8	1/85	68	-	8	60	-	-
	9	6/86	60	4	-	60	-	2
Oconee 3	1	5/75	177	61	8	68	108	-
	2	5/77	56	-	-	-	-	2
	3	6/78	56	-	-	-	-	-
	4	8/78	56	-	-	-	-	1
	5	3/79	56	-	-	-	-	-
	6	9/80	68	-	-	60	-	-
	7	6/82	72	-	-	64	-	-
	8	2/84	68	13	-	60	-	-
	9	6/85	68	-	8	60	-	-
Rancho Seco	1	10/73	177	61	8	68	108	-
	2	6.77	56	-	-	-	-	2
	3	5/78	56	-	-	-	4	-
	4	11/79	52	-	-	52	-	-
	5	2/81	56	-	-	56	-	-
	6	6/82	40	-	8	48	-	-
	7	12/84	56	2	1	56	-	-
Three Mile Island 1	1	3/74	177	62	8	68	108	-
	2	12/75	56		-	-	-	2
	3	12/76	61	-	-	-	4	-
	4	1/78	60	-	-	-	-	-
	5	1/79	54	3	8	-	-	-
	6	11/85	59	-	-	-	-	-

FA - Fuel Assemblies
CRA - Control Rod Assemblies
APSA - Axial Power Shaping Assemblies
BAA - Burnable Absorber Assemblies
ORA - Orifice Rod Assemblies
RNSA - Regenerative Neutron Source Assemblies

Table 2.8.4 Calculation of Total Effective Full Power Days for CE-built reactors, based on Historical and Projected Fuel Discharges.

Reactor	Power Level	Hist./ Proj.	Discharges (MTU)*	Average Burnup*	Total Power (MW days)	EFPD
Ark.Nucl.One,Unit 2	2815	H	176.1	31,837	26,054,000	9,255
		P	458.4	44,607		
Calvert Cliffs 1	2700	H	274.9	31,089	29,206,000	10,817
		P	440.8	46,868		
Calvert Cliffs 2	2700	H	235.0	32,832	29,250,000	10,833
		P	438.8	49,076		
Fort Calhoun	1500	H	154.0	30,549	14,641,000	9,761
		P	235.0	42,284		
Maine Yankee	2630	H	347.0	26,005	26,202,000	9,963
		P	385.2	44,595		
Millstone 2	2700	H	246.4	30,627	27,503,000	10,186
		P	442.7	45,080		
Palo Verde 1	3800	H	77.0	23,489	39,303,000	10,343
		P	894.3	41,926		
Palo Verde 2	3800	H	45.0	18,034	39,993,000	10,525
		P	943.0	41,550		
Palo Verde 3	3817	H	43.4	15,816	39,283,000	10,292
		P	937.8	41,157		
San Onofre 2	3390	H	152.0	28,038	34,457,000	10,164
		P	676.3	44,648		
San Onofre 3	3390	H	105.7	25,144	34,552,000	10,192
		P	712.3	44,777		
St. Lucie 1	2700	H	231.2	30,556	31,187,000	11,551
		P	527.5	45,730		
St. Lucie 2	2700	H	120.6	27,712	33,605,000	12,446
		P	609.8	49,628		
Waterford	3390	H	107.3	24,722	37,302,000	11,004
		P	765.5	45,264		

* Based on 1989 EIA RW-859 Historical Discharges and No New Orders Case Projections through 2040 (Reference 12).

Table 2.8.5 Estimated "lifetime" quantities of NFA hardware from CE-built reactors.

	Control Elements			Neutron Sources			Instrumentation Strings		
	Number per Core	Number of Sets*	Total Discharges	Number per Core	Number of Sets*	Total Discharges	Number per Core	Number of Sets*	Total Discharges
Ark.Nucl.One,Unit 2	81	3	243	2	3	6	44	7.7	339
Calvert Cliffs 1	77	3	231	2	3	6	45	9.0	405
Calvert Cliffs 2	77	3	231	2	3	6	45	9.0	405
Fort Calhoun	49	3	147	2	3	6	28	9.1	255
Maine Yankee	77	3	231	2	3	6	45	8.3	374
Millstone 2	73	3	219	2	3	6	45	8.5	383
Palo Verde 1	89	3	267	2	3	6	61	8.6	525
Palo Verde 2	89	3	267	2	4	8	61	8.8	537
Palo Verde 3	89	3	267	2	3	6	61	8.6	525
San Onofre 2	91	3	267	2	3	6	61		
San Onofre 3	91	3	273	2	3	6	56	8.5	476
St. Lucie 1	73	3	219	2	3	6	45	9.6	432
St. Lucie 2	91	3	273	2	4	8	56	10.4	582
Waterford	91	3	273	2	4	8	56	9.2	515

* Basis for number of sets in estimated number of effective full power days for each reactor (from Table 2.8.5) divided by CE estimate of the lifetime of the NFA hardware item in effective full power days. This value is rounded up to the next integer for control element sets and neutron source sets and up to the next tenth for instrument strings.

Table 2.8.6 Estimation of Number of Burnable Absorber Assemblies from WE-built reactors

Years	Est. Percentage of Assemblies with BAA	WE 17 x 17 Class Number of FA	WE 17 x 17 Class Est. Number of BAA	WE 15 X 15 Class Number of FA	WE 15 X 15 Class Est.Number of BAA	WE 14 X 14 Class Number of FAA	WE 14 X 14 Class Est Number of BAA
70-85	20%	2793	559	5026	1005	2618	524
86-89	95%	4231	4019	1612	1531	971	922
90-94	95%	6945	6698	1883	1585	1124	1068
95 on	10%	45258	4526	7322	732	4073	407
Total		59227	15802	15843	4853	8786	2921
Percent of Total			27%		31%		33%

Overall Percent of Total 223576 / 83856 = 28%

* It is not unexpected that the percentage of assemblies with burnable absorber assemblies from the WE 17 X 17 Class is less than the corresponding percentage for WE 14 X 14 and WE 15 X 15 Class. This is true even though current usage may be significantly higher in WE 17 X 17 Class reactors. This is the result of the assumption that the use of burnable absorber assemblies in WE-built reactors will decrease dramatically after 1994 and the fact that nearly 75% of the fuel from WE 17 X 17 Class reactors will be discharged after that date.

Table 2.8.7. Neutron Flux Scaling Factor for Different Neutron Exposure Regions

Exposure Region	Abbreviation	BWR Flux Scaling Factor	PWR Flux Scaling Factor
Above Top End Fitting Region	ATEF	0.01*	0.01*
Top End Fitting Region	TEF	0.10	0.10
Gas Plenum Region	GSP	0.20	0.20
Incore Region	INC	1.00	1.00
Bottom End Fitting Region	BEF	0.15	0.20
Below Bottom End Fitting Region	BBEF	0.015*	0.02*

* The flux in these zones is estimated as an order of magnitude less than the flux in the top and bottom end fittings, as given in reference 4.

Table 2.8.8 Summary of estimated exposure lifetime and locations of NFA hardware.

Reactor Vendor	Type of NFA Hardware	Exposure Location ATEF	TEF	GSP	INC	BEF	BBEF	Weighted Scaling Factor*
B&W	Burnable Absorber Assemblies		14%	5%	81%			0.834
	Axial Power Shaping Assemblies		14%	5%	81%			0.834
	Control Rod Assemblies	83%	5%	5%	7%			0.093
	Orifice Rod Assemblies		50%	50%				0.150
	Neutron Source Assemblies		17%	4%	79%			0.815
	Instrumentation			Unknown				
CE	Control Element Assemblies	70%	15%	15%				0.052
	Neutron Sources		10%	2%	88%			0.894
	Instrument Strings	64%	6%	2%	28%			0.354
WE	Control Rod Assemblies	90%	5%	5%				0.020
	Burnable Absorber Assemblies		15%	7%	78%			0.809
	Thimble Plug Assemblies		50%	50%				0.150
	Neutron Source Assemblies		14%	6%	80%			0.826
	Instrumentation			Unknown				
GE	Cruciform Control Blades				15%	5%	80%	0.170
	Local Power Range Monitors.				10%	5%	85%	0.120
	Other Instrumentation				10%	5%	85%	0.120
	Fuel Channels		1%	2%	96%	1%		0.967

* Weighted Scaling Factor (WSF) is defined as the sum of the products of the scaling factors from Table 2.8.7 and the percentage of the component in that exposure location as given in this table. For example, for WE neutron source assemblies, this is:

WSF = 0.14*0.10 (TEF) + 0.06*0.20 (GSP) + 0.80*1.00 (INC) = 0.014 + 0.012 + 0.80 = 0.826

Table 2.8.9 Summary of estimated exposure lifetimes and flux exposure factors for NFA hardware.

Reactor Vendor	Type of NFA Hardware	Estimated Exposure Lifetime (in Cycles)	Cycles per Assembly Lifetime	Weighted Scaling Factor	Flux Exposure Factor
B&W	Burnable Absorber Assemblies	1	3	0.834	0.278
	Axial Power Shaping Assemblies	5	3	0.834	1.390
	Control Rod Assemblies	10	3	0.093	0.310
	Orifice Rod Assemblies	20	3	0.150	1.000
	Neutron Source Assemblies	7	3	0.815	1.902
CE	Control Element Assemblies	10	3	0.052	0.173
	Neutron Sources	10	3	0.894	2.980
	Instrument Strings	4	3	0.354	0.472
WE	Control Rod Assemblies	10	3	0.020	0.067
	Burnable Absorber Assemblies	1	3	0.809	0.270
	Thimble Plug Assemblies	20	3	0.150	1.000
	Neutron Source Assemblies	7	3	0.826	1.927
GE	Cruciform Control Blades	10	4	0.170	0.425
	Local Power Range Monitors.	3	4	0.120	0.090
	Other Instrumentation	3	4	0.120	0.090
	Fuel Channels	4	4	0.967	0.967

3. IMMOBILIZED HIGH-LEVEL WASTE

3.1 SUMMARY

Canisters of high-level waste (HLW) immobilized in borosilicate glass or glass-ceramic mixtures are to be produced at West Valley Demonstration Project (WVDP), Savannah River Site (SRS), Hanford (HANF), and Idaho National Engineering Laboratory (INEL) for ultimate disposal at a geologic repository. The vitrification plants at SRS and HANF have been named the Defense Waste Processing Facility (DWPF) and the Hanford Waste Vitrification Plant (HWVP), respectively. At INEL, the conversion to an immobilized form will be done at the Idaho Chemical Processing Plant (ICPP). Data are presented in this chapter on the estimated physical, chemical, and radiological characteristics and production schedules of the canisters of immobilized waste through the year 2020. Sections 3.2, 3.3, 3.4, and 3.5 give the data for WVDP, SRS, HANF, and INEL, respectively.

3.1.1 Canister Dimensions and Weights

Table 3.1.1 summarizes the estimated dimensions and weights of the canisters for the four sites. Three of the sites (WVDP, SRS, and HANF) plan to use cylindrical stainless steel canisters, 61 cm (24 in.) in diameter and 300 cm (118 in.) high, filled with borosilicate glass to about 85% of the canister volume. The 85% figure refers to the glass volume at filling temperature, which is about 825°C (average) in the canister as filled. Observations at SRS have shown that cooling the canister to ambient temperature does not reduce the glass level in the canister appreciably. The canister designs for SRS and HANF are identical. The WVDP canister has the same outside diameter and length but has a smaller wall thickness and a wider filler neck.

For INEL, neither the canister dimensions nor the waste form have been fixed. Borosilicate glass and hot-isostatic-pressed (HIP) glass-ceramic forms are being considered; the glass-ceramic form permits the use of a considerably smaller number of canisters than the glass form for a given amount of waste. The estimates in Table 3.1.1 for INEL are based on preliminary information from INEL (Berreth 1987), which, in turn, is based on the assumptions that the glass-ceramic form will be used for immobilization and that the external dimensions of the canister will be the same as those used for WVDP, SRS, and HANF.

The weights of loaded canisters for the four sites range from about 2,150 to 2,325 kg, of which about 1,650 to 1,900 kg is HLW glass or glass-ceramic.

Table 3.1.1 also summarizes the estimated maximum radioactivity and thermal power (curies and watts) per canister at the time of filling (indexed to the end of 1991 for WVDP), based on the most highly radioactive immobilized waste composition currently envisaged at each site. However, future changes in plans could result in the production of canisters of higher radioactivity and thermal output than those shown here.

Subsequent sections on the individual sites give further details on immobilized waste characteristics, including tables showing radioactivity and thermal power as functions of decay time after filling the canisters.

3.1.2 Canister Production Schedules

The total number of HLW canisters to be produced at each site by the year 2020 is not yet firmly established. The estimate for the WVDP site is about 255 to 300 canisters; the estimate used in this report is 275 canisters, which is based on the most recent material balance (Crocker 1989a) plus an allowance for operational variations. The 300-canister estimate represents a conservative upper limit. The estimates for WVDP can be established more accurately than those for the three defense sites because the amount of waste at WVDP is a fixed quantity. For the defense sites, there are several possible scenarios and options that can lead to different total numbers of canisters. This report will present one such scenario, based on the 1990 Integrated Data Base submittals from the defense sites (Garvin 1990, Turner 1990, Berreth 1990), which will be referred to as the 1990 Base Case.

Based on the 1990 IDB submittals, initial vitrification was to begin at SRS in 1992, at WVDP in 1993, at HANF in 1999, and at INEL in 2012. Tables 3.1.2 and 3.1.3 show the estimated annual production rate and cumulative production schedule of canisters at each site in the 1990 Base Case. The two tables show the annual number and the cumulative number of canisters produced for each year through the year 2020. The tables show that in the case presented here, an estimated total of about 15,300 canisters will have been produced by the end of 2020. Table 3.1.4 summarizes the assumptions on which Tables 3.1.2 and 3.1.3 are based.

For INEL, the actual strategy, process, and schedule for disposal of HLW have not yet been firmly established. However, based on the 1990 IDB submittal, it was assumed that the glass-ceramic waste form will be used for immobilization and that immobilization will start in 2012. The immobilized waste generation schedule for INEL shown in Tables 3.1.2 and 3.1.3 is based on the assumption that during the first three years of operation the immobilization plant will operate at a reduced rate (500 to 700 canisters per year), which is consistent with the annual fuel reprocessing rate. After the third year, a production

rate of 1,000 canisters per year is assumed to allow for working off the backlog of stored calcine over a period of about 30 years (Berreth 1987, 1989, 1990).

The production schedules detailed in Tables 3.1.2 and 3.1.3 are shown graphically in Figs. 3.1.1 and 3.1.2. Figure 3.1.1 shows the cumulative numbers of canisters produced at each of the four individual sites through the year 2020, and Fig. 3.1.2 shows the cumulative number of canisters summed for all sites. Because of scheduling requirements and the need for consistency throughout this report, it was necessary to freeze data inputs as of the time of the 1990 IDB submittals. More recent data, received too late for incorporation in the tables of this report, revised the estimated start-up dates for SRS, WVDP, HANF, and INEL to 1993, 1996, 2000, and 2014, respectively.

3.1.3 Radiological Properties

To the extent possible, data were obtained from the sites on the projected radionuclide contents of the canisters at the time of filling. The first focus was on the expected maximum radioactivity and thermal power per canister, since this information would define limiting values in repository and transportation system design. Data were obtained indicating maximum and average values of radioactivity per canister for WVDP, maximum values for SRS, and maximum and "nominal" values for neutralized current acid waste (NCAW) from HANF. For INEL, current information on radionuclide composition was unavailable; the estimate used here was based on previously published data. Based on these maximum radionuclide compositions, ORIGEN2 decay calculations were made to determine the mass (grams), radioactivity (curies), and thermal power (watts) of each nuclide, and the total per canister, for decay times up to 10^6 years after filling. This information is summarized in this chapter for each of the sites and is presented in more detailed form in Appendix 3A. These data are also available in magnetic diskette form. The diskette data also show alpha curies, photon energy distributions, and neutron production rates.

Cumulative average radioactivities per canister have also been estimated for SRS and INEL based on projections of cumulative radioactivity in glass or glass-ceramic and cumulative number of canisters produced. Similar projections of cumulative average radioactivity per canister are not yet available for HANF. These cumulative average radioactivities per canister are, in general, considerably lower than the maximum radioactivities per canister previously discussed and should be more useful than the maximum values for the calculation of total repository thermal loadings. However, these averages should not be used for detailed short-term calculations without first verifying actual processing schedules and compositions.

Estimates of the rates of production of neutrons from canisters of HLW glass from spontaneous fission and from (alpha, n) reactions are also shown. Neutron production from (alpha, n) reactions takes place as a result of the action of energetic alpha particles from radionuclides such as Pu-238, Am-241, and Cm-244 on various light nuclides in the borosilicate glass such as Si-29, B-11, and O-18. The rate of neutron production from (alpha, n) reactions is significant in vitrified HLW and must be considered when estimating neutron shielding requirements. In this report, shielding requirements are not calculated; the data shown are source terms only, i.e., the rates of neutron production within the glass (Hermann 1992, Salmon 1992).

3.1.4 Assessment of Data

At WVDP, the radionuclide content per canister is fairly well established. Reprocessing of fuel was discontinued in 1972; thus the waste to be immobilized is a fixed quantity, and its composition is known within the limits of sampling and analytical capabilities. The radiological properties per canister can therefore be readily calculated, and the only variation of these properties with time is that due to the process of radioactive decay. These calculations require only a single ORIGEN2 decay run, which starts with a single fixed composition and tracks the resulting grams, curies, and watts for any desired series of decay times.

At the three defense sites, however, the situation is more complex. Plants at these sites continue to process fuel, so the wastes in storage are a mixture of old, well-aged waste and newly processed waste of much higher radioactivity. When immobilization begins, it might be thought desirable to try to work off the older waste first; however, this may not be feasible because of tankage constraints (for example, the need to have tanks available to receive current production). Thus, the proportions of old and new waste fed to the melter will vary from year to year. In addition, the composition of the freshly produced waste may undergo changes. However, even if this latter variation does not occur, the radiological properties of a canister from a defense site will vary depending on the melter feed composition in the year in which the canister was filled as well as on the decay time elapsed since filling. A complete characterization of the canisters produced from such a site would require a schedule of melter feed composition versus time and a separate decay calculation for each melter feed composition and decay time.

Thus far, SRS is the only site that has presented a forecast of the radionuclide content of the glass on a year-by-year basis. This forecast covers the period 1992-2008 (Garvin 1990). HANF has presented four compositions showing the possible variation of radionuclide contents between 1996 and 2000 (Mitchell 1986); more recently, these data have been supplanted by estimates of maximum and "nominal" radionuclide compositions of NCAW waste glass (Mitchell and Nelson 1988). INEL, because of security restrictions, has released no recent data on

radionuclide compositions. To provide preliminary estimates of maximum radioactivity and decay heat per canister as a function of decay time for INEL HLW, a composition based on 1982 data was used.

Assessment of the data presented in this report for the defense sites pinpoints the variation of radionuclide compositions of melter feeds with time as an area requiring additional information and analysis.

Of equal importance in the assessment of the data in this report is the fact that various strategies and processing alternatives for immobilized waste production are still under consideration. Also, requirements for future defense production may change. As already noted, changes in vitrification start-up dates and canister production schedules have been announced. Further changes in canister production schedules, radiological properties of canisters, and the total number of canisters produced by the year 2020 are still possible. It must also be kept in mind that this report does not present any information on the number of defense HLW canisters produced after the year 2020. Based on the quantities of HLW remaining uncanistered at the end of 2020, it is clear that an appreciable number of additional canisters will be required after that time. As a specific example, the 1990 IDB submittal from INEL (Berreth 1990) shows an estimated production of 19,000 canisters from 2021 to 2039.

3.1.5 Interim Forms of High-Level Waste

At present, the high-level wastes stored at the sites are in various interim forms such as liquids, slurries, sludges, calcine, etc. The quantities and compositions of these interim forms and their conversion to final forms are discussed in Appendix 3B, which thus serves to provide the detailed backup data for the information presented in this section, as well as additional details on the processing of the waste.

3.1.6 Order of Presentation

The remainder of this section is arranged according to site location and is presented in the following order: WVDP, SRS, HANF, and INEL. For each site, the data are presented in a fixed order, as follows: (1) types of waste produced, (2) canister dimensions and weights, (3) canister production schedule, (4) radionuclide content per canister at time of filling (or at a specified time point for WVDP), (5) radiological properties (curies and watts) per canister as a function of time after filling, (6) chemical composition of waste form, and (7) assessment of data.

3.1.7 References for Section 3.1

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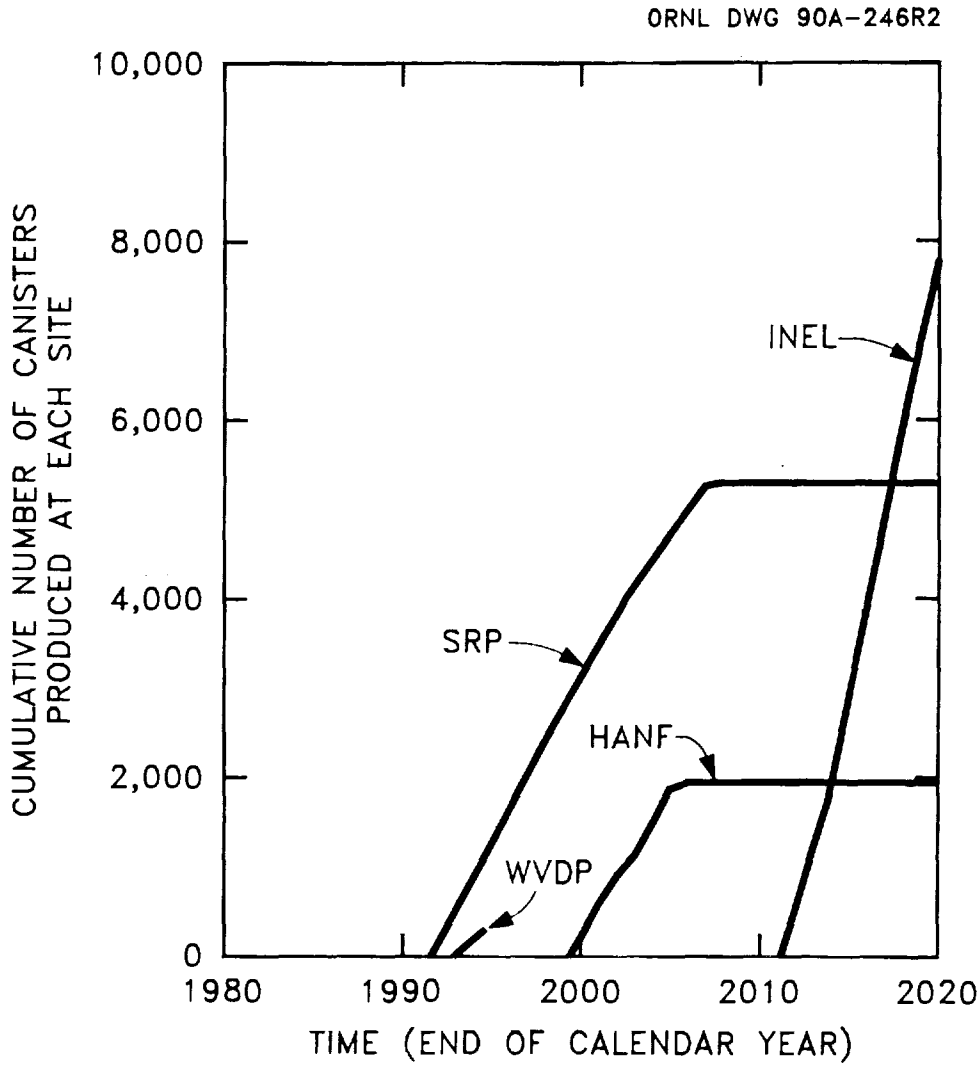

Fig. 3.1.1. Cumulative number of canisters of HLW produced at each individual site in 1990 Base Case. (See data in Tables 3.1.2 and 3.1.3.) Sources: Crocker 1989a, Garvin 1990, Turner 1990, Berreth 1990.

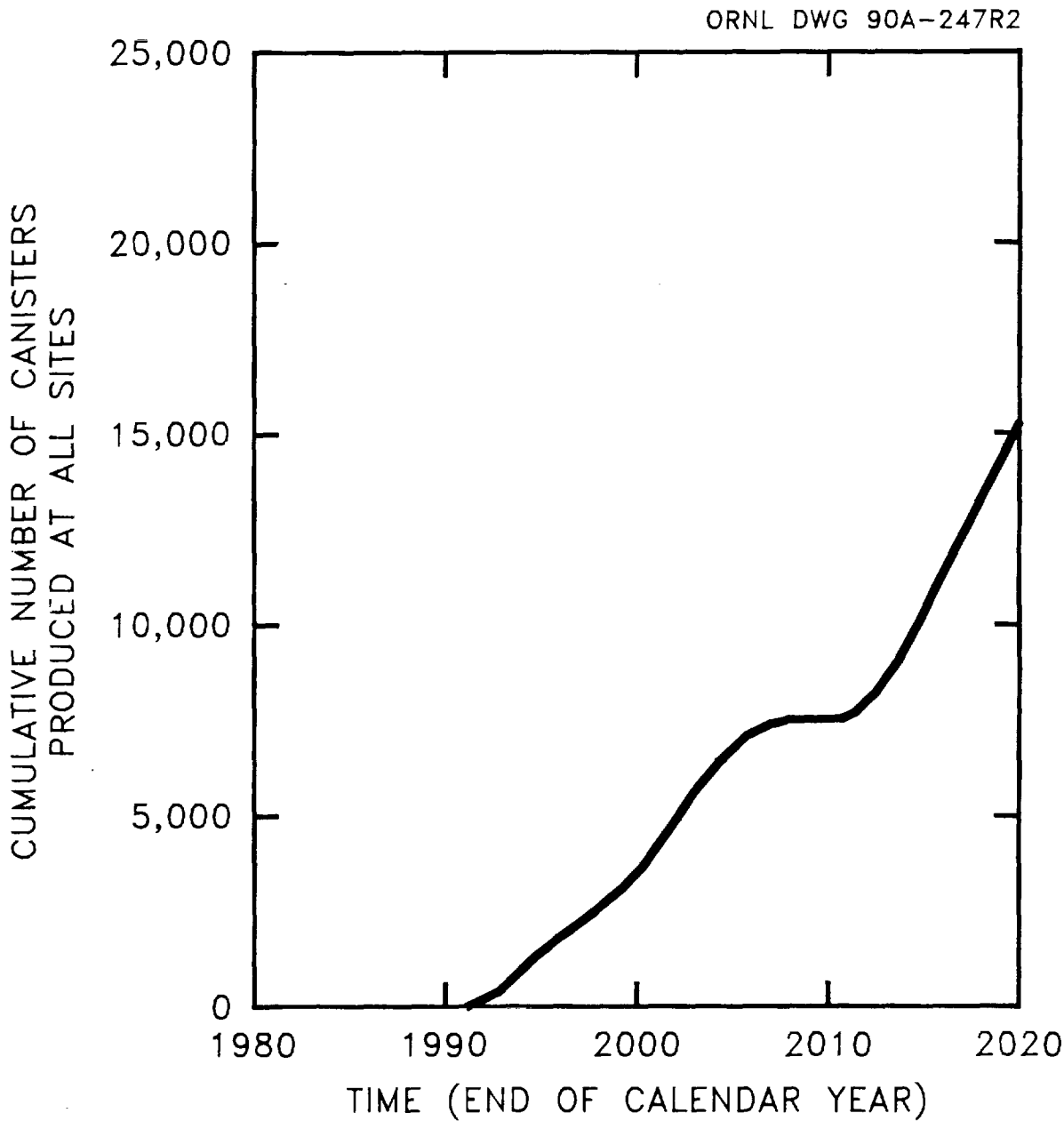

Fig. 3.1.2. Cumulative number of canisters of HLW produced, total of all four sites, in 1990 Base Case. (See data in Tables 3.1.2 and 3.1.3.) Sources: Crocker 1989a, Garvin 1990, Turner 1990, Berreth 1990.

Table 3.1.1. Dimensions, weights, and radioactivity of canisters (summary)

	West Valley Demonstration Project (WVDP)	Savannah River Site (SRS)	Hanford Site (HANF)	Idaho National Engineering Laboratory (INEL)
Outside diameter, cm	61	61	61	61
Overall height, cm	300	300	300	300
Material	SS	SS	SS	SS
Nominal wall thickness				
mm	3.4	9.5	9.5	9.5
in.	0.134	0.375	0.375	0.375
Weights, kg				
Canister	252	500	500	500
Glass or ceramic	1,900	1,682	1,650	1,825
Total	2,152	2,182	2,150	2,325
Curies per canister[a]	114,700	234,400	298,000[b]	109,000
Watts per canister[a]	342	709	869[c]	339

[a]These are estimated maximum values from ORIGEN2 calculations based on radionuclide compositions supplied by the sites. Curies and watts shown are at time of filling the canister, except for West Valley Demonstration Project where the values shown are for the end of year 1991. For West Valley Demonstration Project, maximum values are assumed to be 110% of average values, and average values are based on the Revision 7 mass balance (Crocker 1989a). Maximum values for the defense sites do not necessarily represent initial operations.

[b]Based on Mitchell and Nelson 1988, maximum case.

[c]Based on ORIGEN2 calculations using Mitchell and Nelson maximum case.

Table 3.1.2. Projected annual number of canisters of immobilized HLW produced at each site[a,b]

Calendar year	WVDP	SRS	HANF	INEL	Total
1989	0	0	0	0	0
1990	0	0	0	0	0
1991	0	0	0	0	0
1992	0	136	0	0	136
1993	25	308	0	0	333
1994	200	376	0	0	576
1995	50	410	0	0	460
1996	0	410	0	0	410
1997	0	383	0	0	383
1998	0	369	0	0	369
1999	0	369	0	0	369
2000	0	342	240	0	582
2001	0	342	370	0	712
2002	0	342	345	0	687
2003	0	342	185	0	527
2004	0	302	370	0	672
2005	0	273	370	0	643
2006	0	273	80	0	353
2007	0	273	0	0	273
2008	0	32	0	0	32
2009	0	0	0	0	0
2010	0	0	0	0	0
2011	0	0	0	0	0
2012	0	0	0	500	500
2013	0	0	0	600	600
2014	0	0	0	700	700
2015	0	0	0	1,000	1,000
2016	0	0	0	1,000	1,000
2017	0	0	0	1,000	1,000
2018	0	0	0	1,000	1,000
2019	0	0	0	1,000	1,000
2020	0	0	0	1,000	1,000
Total	275	5,282	1,960	7,800	15,317

[a]Sources: WVDP — Crocker 1989, 1989a, 1990; SRS — Garvin 1990; HANF — Turner 1990; and INEL — Berreth 1990.

[b]For assumptions used in compiling this table see Table 3.1.4. This table represents the 1990 Base Case for this report. Canisters produced after 2020 are not included here. Canister production figures represent most likely estimates rather than maximum potential.

Table 3.1.3. Projected cumulative number of canisters of immobilized HLW produced at each site[a,b]

Calendar year	WVDP	SRS	HANF	INEL	Total
1991	0	0	0	0	0
1992	0	136	0	0	136
1993	25	444	0	0	469
1994	225	820	0	0	1,045
1995	275	1,230	0	0	1,505
1996	275	1,640	0	0	1,915
1997	275	2,023	0	0	2,298
1998	275	2,392	0	0	2,667
1999	275	2,761	0	0	3,036
2000	275	3,103	240	0	3,618
2001	275	3,445	610	0	4,330
2002	275	3,787	955	0	5,017
2003	275	4,129	1,140	0	5,544
2004	275	4,431	1,510	0	6,216
2005	275	4,704	1,880	0	6,859
2006	275	4,977	1,960	0	7,212
2007	275	5,250	1,960	0	7,485
2008	275	5,282	1,960	0	7,517
2009	275	5,282	1,960	0	7,517
2010	275	5,282	1,960	0	7,517
2011	275	5,282	1,960	0	7,517
2012	275	5,282	1,960	500	8,017
2013	275	5,282	1,960	1,100	8,617
2014	275	5,282	1,960	1,800	9,317
2015	275	5,282	1,960	2,800	10,317
2016	275	5,282	1,960	3,800	11,317
2017	275	5,282	1,960	4,800	12,317
2018	275	5,282	1,960	5,800	13,317
2019	275	5,282	1,960	6,800	14,317
2020	275	5,282	1,960	7,800	15,317

[a]Sources: WVDP — Crocker 1989, 1989a, 1990; SRS — Garvin 1990; HANF — Turner 1990; and INEL — Berreth 1990.

[b]For assumptions used in compiling this table, see Table 3.1.4. This table represents the 1990 Base Case for this report. Quantities given represent cumulative number of canisters of waste produced by the end of a given calendar year. Canisters produced after 2020 are not included here. Canister production figures represent most likely estimates rather than maximum potential. The reader is cautioned that this table is subject to future updates.

Table 3.1.4. Assumptions used in 1990 Base Case of this report

1. Canister dimensions 61 cm diam by 300 cm length; 85% fill assumed at filling temperature

2. Maximum immobilization throughputs of the various sites, in canisters per year, are as follows: WVDP - 200; SRS - 410; HANF - 370; and INEL - 1,000

3 Production of canisters of HLW starts at SRS in 1992, at WVDP in 1993, at HANF in 1999, and at INEL in 2012. Canister production is shown through the end of the year 2020 and does not include any waste canistered after 2020[a]

4. WVDP canister production is based on 484,000 kg of total glass loaded at 1,900 kg/canister (Crocker 1989a). An allowance of 20 canisters for operational variations was added, bringing the total number of canisters to 275. Other WVDP data are based on the WVDP Integrated Data Base submittals for 1990 (Maestas 1990, Crocker 1990)

5. SRP canister production is based on Garvin 1990. The DWPF is assumed to reach full production in 3QFY 1992. It is assumed that sludge and liquid inventories will reach steady state by 2009 and that salt cake will reach steady state in 2011. Steady-state volumes must be maintained to allow waste to age before it is processed. Because of a lack of adequately aged feedstock, it is assumed that there will be no production of vitrified HLW during the period 2009–2020. It is assumed that three reactors will be restarted by the end of 1990 and will remain operating through FY 2004. It is also assumed that the current reactors will be replaced by a New Production Reactor (NPR), which will begin operating in FY 2000

6. HANF vitrification plant production is based on Turner 1990; the total consists of 480 canisters of NCAW, 100 canisters of PFP, 400 canisters of NCRW, and 980 canisters of CC (complexant concentrate). Melter production rate is 370 canisters/year. It is assumed that the N Reactor does not restart and the Purex plant operates through the year 1996

7. INEL canister production is based on the schedule given in Berreth 1990. The ceramic-based waste form is used for immobilization. No removal of inerts prior to immobilization was assumed. A canister load is 1,825 kg of ceramic, equivalent to 1,277 kg of calcine. Density of ceramic is 3,200 kg/m^3. The maximum production rate is 1,000 canisters/year; this permits working off the stored calcine over about a 30-year period

[a]More recent information (see Sect. 1.3.1) shows estimated startup dates of 1996 for WVDP, 1993 for SRS, 1999 for HANF (this was unchanged), and 2014 for INEL. However, the assumptions used in the tables of this report are as listed above.

3.2 WEST VALLEY DEMONSTRATION PROJECT (WVDP)

3.2.1 Introduction

Approximately 660 metric tons of irradiated fuel were processed at the commercial fuel reprocessing plant at West Valley, New York, from 1966 to 1972; the reprocessing plant was then shut down. A brief summary of the fuel that was reprocessed is given in Appendix 3B and is further discussed in Section 2.6. The West Valley Demonstration Project (WVDP), jointly funded by the U.S. DOE and the New York State Energy Research and Development Agency, was started in 1982 with the objective of solidifying the HLW remaining from the commercial reprocessing operations into a form suitable for transportation to and disposal in a federal repository.

3.2.2 Types of HLW Produced

Only one type of immobilized HLW will be produced at WVDP, namely HLW immobilized in borosilicate glass encased in stainless steel canisters.

3.2.3 Physical Description

Figure 3.2.1 and Table 3.2.1 show details of the HLW glass canister planned for use at the WVDP vitrification facility. The canister is approximately 0.61 m in diameter and 3.0 m in height with a wall thickness of 0.34 cm and is welded shut after filling. The top closure is a cap made of flat plate about 0.95 cm thick. The expected fill volume is 85% of capacity ±5%. The empty canister weighs about 252 kg. When filled to 85% of capacity, each canister will contain 0.70 m^3 (about 1,900 kg) of vitrified waste and will weigh about 2,152 kg. The density of the solidified waste glass is approximately 2.7 g/cm^3 at 25°C (Rykken 1986, Eisenstatt 1986).

3.2.4 Inventory and Production Schedule

The vitrification schedule for WVDP has been delayed by about 3 years, compared with the date given in our previous report (DOE 1987). Hot startup of the vitrification plant is now scheduled to begin in October 1993, and vitrification is scheduled to be completed in 1995. According to the most recent material balance (revision 7, Crocker 1989a), the total quantity of vitrified waste produced during this period will be 484,000 kg. At 1,900 kg per canister, this would represent 255 canisters (Crocker 1989, 1989a). Earlier material balances have ranged from 471,000 kg to 520,000 kg, corresponding to 249-275 canisters at a fill level of 85%. It is anticipated that there will be some operational variations and clean-up requirements, and on this basis we have estimated that the number of canisters produced will be about 275. Table 3.2.2 shows the estimated schedule of production; this is based on a single campaign with a duration of 18 months, starting in October 1993 and ending in 1995. The 18-month period allows about 20% off-stream time for scheduled and unscheduled shutdowns.

3.2.5 Radioactivity Per Canister

In the previous edition of this report (DOE 1987), the initial radionuclide contents per canister of glass were taken from Eisenstatt 1986, which gave average, maximum, and minimum values, calculated to allow for decay to the year 1990. The present edition of this report uses the Revision 7 material balance (Crocker 1989a), which is based on more recent data, but which gives only average values of radionuclide contents per canister. These compositions, expressed as curies of each radionuclide per canister, are shown in Table 3.2.3. Data are for the start of year 1990. The new data show a calculated average radioactivity for a canister containing 1,900 kg of glass of 109,600 Ci at the end of year 1989, which represents about a 2.7% reduction from the 112,700 Ci/canister used in the previous edition of this report.

Calculated rates of neutron production per canister of HLW glass from spontaneous fission and from (alpha, n) reactions are shown in Table 3.2.4. Calculations were based on the average radionuclide content per canister at the end of year 1989 as given by the Revision 7 Mass Balance and shown in Table 3.2.3.

3.2.6 Radioactivity and Thermal Power vs Time

Table 3.2.5 shows calculated radioactivity and thermal power per canister as a function of decay time for periods up to 1,000,000 years. The decay calculations were made with the ORIGEN2 code using compositional data for the end of year 1989 as the starting point; thus the end of year 1989 represents the zero point for decay time. As shown in Table 3.2.5, an average-activity canister produced at the end of the year 1989 would have a radioactivity of 109,600 Ci and a thermal power of 326 W at that time. The same canister at the end of year 1991 (as an example), or a new canister filled at the end of 1991 from this same batch of waste, would have a radioactivity of 104,300 Ci and a thermal power of 311 W at the end of 1991. These calculations were based on the average canister composition shown in the Revision 7 mass balance (Crocker 1989a).

More detailed tables, showing the contributions of individual radionuclides, are given in Appendix 3A (Table 1-24 WVDP).

3.2.7 Chemical Composition

Table 3.2.6 shows the expected chemical composition of the HLW glass to be produced at WVDP, based on chemical analyses reported in the Revision 7 mass balance

(Crocker 1989a). Estimated ranges of variation of individual components were not shown in Crocker 1989a but were given in an earlier report (Eisenstatt 1986).

3.2.8 Assessment of Data

The radionuclide composition and quantity of the waste at WVDP and of the glass made from that waste are well established. The most recent estimate of the quantity of waste to be vitrified corresponds to 255 canisters at 85% fill level. Based on this, we estimated that a total of 275 canisters would be produced. Previous estimates of the number of canisters to be produced have ranged from 275 (Rykken 1987) to 300 (Bixby 1987); WVDP has estimated 300 as the upper limit of the number of canisters. In this report, we used an estimate of 255 canisters, rather than 275 canisters, for radioactivity and thermal power calculations. This gives a more conservative (i.e., higher) estimate of radioactivity and thermal power per canister. However, the 275-canister estimate is more conservative from the standpoint of space requirements. The storage facility design at WVDP provides space for 312 canisters.

Melter feed batches are prepared individually and thus may have some variation in composition. The fill level of individual canisters also may vary. For these reasons, it is expected that the maximum initial activity per canister might exceed the average by as much as 10%, as indicated in Table 3.2.1.

3.2.9 References for Section 3.2

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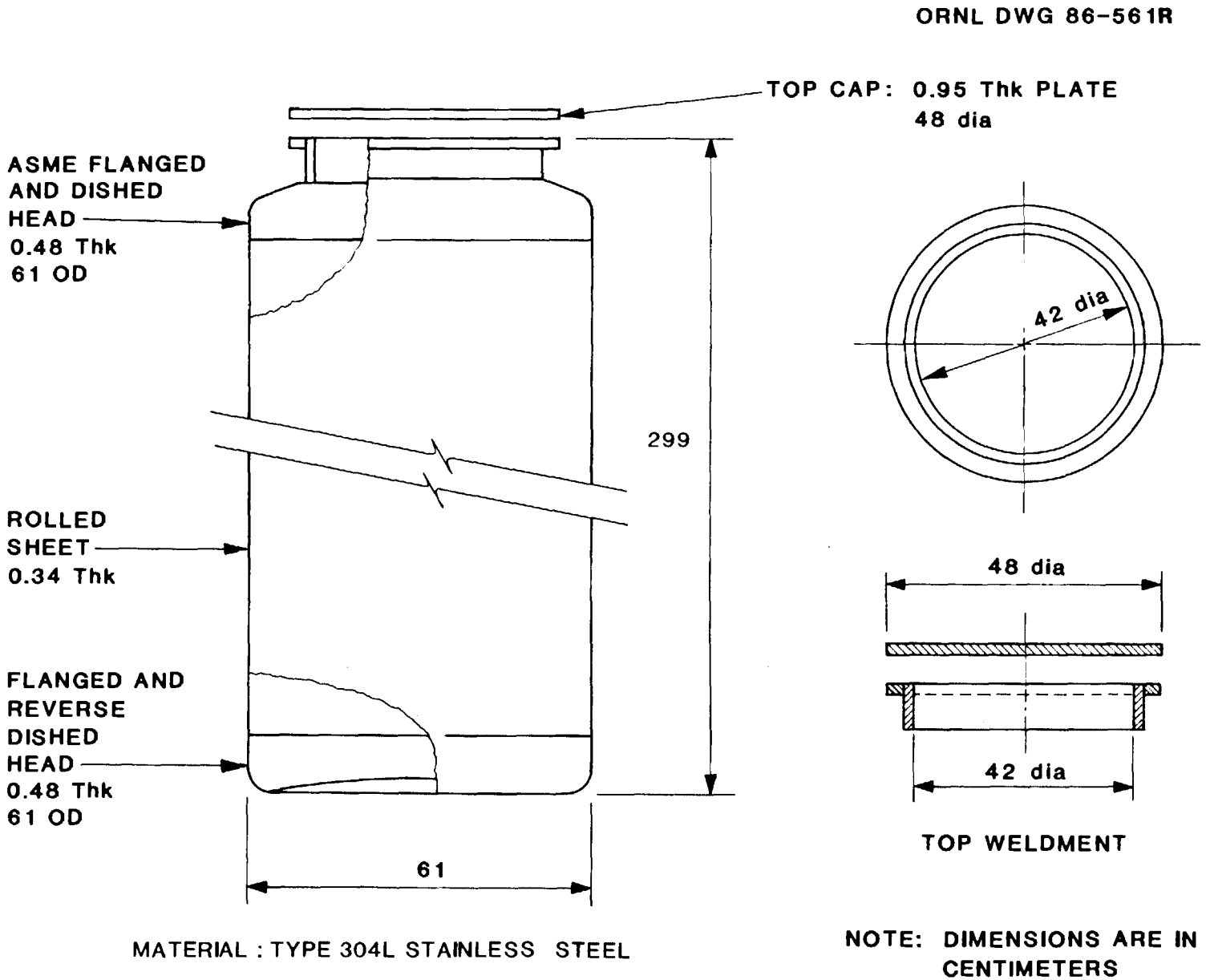

Fig. 3.2.1. West Valley Demonstration Project HLW canister. Source: Rykken 1987.

Table 3.2.1. West Valley Demonstration Project: high-level waste form and canister characteristics[a]

Waste form	Borosilicate glass in sealed canister
Canister material	Stainless steel type 304L
Borosilicate glass density, g/cm^3 at 25°C	2.7
Weights per canister, kg:	
Empty canister	234
Cover	18
Borosilicate glass	1,900
Total loaded weight	2,152
Canister dimensions, cm:	
Outside diameter	61
Height overall	300
Wall thickness	0.34
Radionuclide content, curies per canister (end of 1991)[b]	
Average	104,300
Maximum[c]	114,700
Thermal power, watts per canister (end of 1991)[b]	
Average	311
Maximum[c]	342

[a]Source: Crocker 1989a and ORNL calculations based on Rev. 7 mass balance.

[b]Quantities shown are at 85% fill. Curies and watts per canister are given as of the end of year 1991.

[c]Maximum activity is assumed to be 110% of average.

Table 3.2.2. West Valley Demonstration Project: estimated production schedule of canisters of HLW glass[a]

End of calendar year	Number of canisters produced during year	Cumulative total number of canisters produced	Cumulative total glass produced (kg)
1993	25	25	44,000
1994	200	225	396,000
1995	50	275	484,000

[a]Based on Crocker 1989, 1989a, and 1990. Canister fill volume is assumed to be 85%. The schedule of production is estimated based on a 4Q 1993 startup and a maximum capacity of 200 canisters/year. Production during the last quarter of 1993 has been estimated as one-half of a normal quarter's production in order to allow for start-up difficulties. The column showing kg of glass produced is based on a total glass production of 484,000 kg, which is the total shown on the latest WVDP flowsheet (the Revision 7 mass balance). Based on 484,000 kg of glass and a nominal loading of 1,900 kg per canister, the total number of canisters would be about 255. WVDP has estimated that the actual number of canisters could be between 255 and 300, allowing for operational variations and final clean-up. In this report the most likely final number of canisters is estimated as 275, and the average glass loading per canister is estimated to be 484,000/275 or 1,760 kg/canister.

Table 3.2.3. West Valley Demonstration Project: estimated radionuclide content per HLW canister[a]

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Fe-55	0.1104E-02	0.2760E+01	0.9313E-04
Co-60	0.2679E-02	0.3030E+01	0.4666E-01
Ni-59	0.5491E+01	0.4160E+00	0.1650E-04
Ni-63	0.4895E+00	0.3020E+02	0.3039E-02
Se-79	0.1980E+00	0.1380E-01	0.3431E-05
Sr-90	0.1928E+03	0.2630E+05	0.3048E+02
Y-90	0.4833E-01	0.2630E+05	0.1456E+03
Zr-93	0.4257E+03	0.1070E+01	0.1242E-03
Nb-93m	0.2529E-02	0.7150E+00	0.1265E-03
Tc-99	0.2524E+02	0.4280E+00	0.2144E-03
Ru-106	0.1655E-04	0.5540E-01	0.3290E-05
Rh-106	0.1556E-10	0.5540E-01	0.5307E-03
Pd-107	0.8416E+02	0.4330E-01	0.2563E-05
Cd-113m	0.3845E-01	0.8340E+01	0.1402E-01
Sn-121m	0.1160E-02	0.6860E-01	0.1373E-03
Sn-126	0.1441E+02	0.4090E+00	0.5095E-03
Sb-125	0.2769E-01	0.2860E+02	0.8929E-01
Sb-126	0.6852E-06	0.5730E-01	0.1057E-02
Sb-126m	0.5206E-08	0.4090E+00	0.5201E-02
Te-125m	0.3885E-03	0.7000E+01	0.5876E-02
Cs-134	0.1569E-01	0.2030E+02	0.2063E+00
Cs-135	0.5505E+03	0.6340E+00	0.2113E-03
Cs-137	0.3252E+03	0.2830E+05	0.3126E+02
Ba-137m	0.4981E-04	0.2680E+05	0.1051E+03
Ce-144	0.8023E-06	0.2560E-02	0.1696E-05
Pr-144	0.3387E-10	0.2560E-02	0.1879E-04
Pm-146	0.9566E-04	0.4260E-01	0.2146E-03
Pm-147	0.3721E+00	0.3450E+03	0.1236E+00
Sm-151	0.1258E+02	0.3310E+03	0.3876E-01
Eu-152	0.8267E-02	0.1430E+01	0.1080E-01
Eu-154	0.1389E+01	0.3750E+03	0.3350E+01
Eu-155	0.2014E+00	0.9370E+02	0.6806E-01
Tl-207	0.1690E-09	0.3220E-01	0.9444E-04
Tl-208	0.4312E-10	0.1270E-01	0.2985E-03
Pb-209	0.1815E-09	0.8250E-03	0.9475E-06
Pb-211	0.1308E-08	0.3230E-01	0.9666E-04
Pb-212	0.2540E-07	0.3530E-01	0.6712E-04
Bi-211	0.7718E-10	0.3230E-01	0.1287E-02
Bi-212	0.2409E-08	0.3530E-01	0.5995E-03

Table 3.2.3 (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Bi-213	0.4265E-10	0.8250E-03	0.3464E-05
Po-212	0.1274E-18	0.2260E-01	0.1196E-02
Po-213	0.6231E-19	0.7860E-03	0.3972E-04
Po-215	0.1095E-14	0.3230E-01	0.1440E-02
Po-216	0.1013E-12	0.3530E-01	0.1443E-02
At-217	0.5124E-15	0.8250E-03	0.3516E-04
Rn-219	0.2482E-11	0.3230E-01	0.1339E-02
Rn-220	0.3826E-10	0.3530E-01	0.1339E-02
Fr-221	0.4653E-11	0.8250E-03	0.3180E-04
Fr-223	0.1117E-10	0.4320E-03	0.1120E-05
Ra-223	0.6306E-06	0.3230E-01	0.1149E-02
Ra-224	0.2216E-06	0.3530E-01	0.1210E-02
Ra-225	0.2104E-07	0.8250E-03	0.5778E-06
Ra-228	0.2550E-04	0.5970E-02	0.4595E-06
Ac-225	0.1421E-07	0.8250E-03	0.2878E-04
Ac-227	0.4464E-05	0.3230E-03	0.1562E-06
Ac-228	0.2662E-08	0.5970E-02	0.5153E-04
Th-227	0.1034E-05	0.3180E-01	0.1159E-02
Th-228	0.4306E-04	0.3530E-01	0.1153E-02
Th-229	0.3877E-02	0.8250E-03	0.2521E-04
Th-230	0.1169E-01	0.2360E-03	0.6670E-05
Th-231	0.6657E-09	0.3540E-03	0.1984E-06
Th-232	0.5880E+05	0.6450E-02	0.1559E-03
Th-234	0.1356E-06	0.3140E-02	0.1271E-05
Pa-231	0.1264E+01	0.5970E-01	0.1796E-02
Pa-233	0.4422E-05	0.9180E-01	0.2081E-03
Pa-234m	0.4571E-11	0.3140E-02	0.1550E-04
U-232	0.1270E-02	0.2720E-01	0.8721E-03
U-233	0.3666E+01	0.3550E-01	0.1031E-02
U-234	0.2640E+01	0.1650E-01	0.4746E-03
U-235	0.1637E+03	0.3540E-03	0.9259E-05
U-236	0.1700E+02	0.1100E-02	0.2976E-04
U-238	0.9337E+04	0.3140E-02	0.7954E-04
Np-236	0.2823E+01	0.3720E-01	0.7494E-04
Np-237	0.1302E+03	0.9180E-01	0.2802E-02
Np-239	0.5861E-05	0.1360E+01	0.3283E-02
Pu-236	0.6209E-05	0.3300E-02	0.1147E-03
Pu-238	0.1904E+01	0.3260E+02	0.1079E+01
Pu-239	0.1028E+03	0.6390E+01	0.1967E+00

Table 3.2.3 (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Pu-240	0.2053E+02	0.4680E+01	0.1455E+00
Pu-241	0.3076E+01	0.3170E+03	0.9815E-02
Pu-242	0.1668E+01	0.6370E-02	0.1879E-03
Am-241	0.6117E+02	0.2100E+03	0.6967E+01
Am-242	0.1435E-05	0.1160E+01	0.1315E-02
Am-242m	0.1204E+00	0.1170E+01	0.4616E-03
Am-243	0.6820E+01	0.1360E+01	0.4366E-01
Cm-242	0.2912E-03	0.9630E+00	0.3544E-01
Cm-243	0.1021E-01	0.5270E+00	0.1931E-01
Cm-244	0.3707E+00	0.3000E+02	0.1048E+01
Cm-245	0.2015E-01	0.3460E-02	0.1147E-03
Cm-246	0.1279E-02	0.3930E-03	0.1285E-04
Total	0.7029E+05	0.1096E+06	0.3260E+03

[a]This table represents the radionuclide content of a canister containing 1,900 kg of HLW glass having the radionuclide composition described in the WVDP Mass Balance Revision 7 (Crocker 1989). Radioactivity shown is as of the end of year 1989.

Table 3.2.4. West Valley Demonstration Project: rates of neutron production per canister of vitrified high-level waste from (alpha, n) reactions and from spontaneous fission[a]

	Neutron production rates per canister, neutrons/s		
Actinide	Alpha, n	Spontaneous fission	Total
U-238	3.981E+01	1.185E+02	1.580E+02
Pu-238	1.259E+06	5.059E+03	1.264E+06
Pu-239	1.951E+05	2.812E+00	1.951E+05
Pu-240	1.434E+05	1.869E+04	1.621E+05
Pu-242	1.586E+02	2.812E+03	2.971E+03
Am-241	8.011E+06	7.586E+01	8.011E+06
Cm-242	5.347E+04	6.274E+03	5.974E+04
Cm-244	1.403E+06	4.122E+06	5.525E+06
Totals	1.107E+07	4.155E+06	1.523E+07

[a]Calculated from average canister composition at the end of year 1989 based on the WVDP Revision 7 material balance. Calculations were made by the ALPHN code (Salmon 1992) for the alpha, n neutron production and by the ORIGEN2 code for spontaneous fission. Canister contains 1,900 kg of glass. Neutron production rates shown are uniform source terms in the glass rather than dose rates at the exterior, i.e., the shielding effects of the glass and the canister wall have not been calculated. Actinides that make a negligible contribution have been omitted; for these see Appendix 3A, Table 9 - WVDP.

Table 3.2.5. West Valley Demonstration Project: calculated radioactivity and thermal power per HLW canister as a function of decay time[a]

Decay time after end of 1989 (years)	Calculated radioactivity per canister (Ci)	Calculated thermal power per canister (W)
0	109,600	326
1	106,900	319
2	104,300	311
5	97,080	290
10	86,230	258
15	76,660	230
20	68,180	205
30	53,970	164
50	33,890	105
100	10,730	37
200	1,260	8.9
300	291	5.4
350	202	4.8
500	128	3.7
1,000	63	1.9
1,050	60	1.8
2,000	26.7	0.70
5,000	15.8	0.33
10,000	13.2	0.26
20,000	10.3	0.18
50,000	6.8	0.08
100,000	5.2	0.04
500,000	3.8	0.04
1,000,000	3.1	0.03

[a]Calculations made with ORIGEN2 code based on data supplied by WVDP (Crocker 1989). Canister contains 1,900 kg of HLW glass. Initial time point (0 years) is at the end of year 1989. The material balance used by WVDP for this case (Revision 7, October 1989) shows 484,000 kg of total glass and a total radioactivity of 27.9×10^6 Ci at the end of year 1989 in the HLW to be vitrified. Data are for a canister containing 1,900 kg of glass of the composition shown in the Revision 7 mass balance and therefore do not take into account possible variations in melter feed composition and fill level.

Table 3.2.6. West Valley Demonstration Project: chemical composition of reference HLW glass[a]

Component	Nominal composition (wt %)
Ag_2O	0.0001
Al_2O_3	6.4240
AmO_2	0.0056
BaO	0.0104
$BaSO_4$	0.0677
B_2O_3	9.9095
CaO	0.4911
CdO	0.0002
Ce_2O_3	0.0629
Cm_2O_3	0.0001
CoO	0.0003
Cr_2O_3	0.1378
Cs_2O	0.0840
CuO	0.0635
Eu_2O_3	0.0016
Fe_2O_3	11.9051
Gd_2O_3	0.0003
Hg	0.0041
In_2O_3	0.0001
K_2O	3.6638
La_2O_3	0.0351
Li_2O	3.0941
MgO	0.8840
MnO	0.9823
MoO_3	0.0116
NaCl	0.0133
NaF	0.1666
Na_2O	11.2175
Na_2SO_4	0.0825
Nd_2O_3	0.1000
NiO	0.2482
NpO_2	0.0073
P_2O_5	2.3181
PaO_2	0.0001
PdO	0.0070
Pm_2O_3	0.0003
Pr_6O_{11}	0.0324
PuO_2	0.0077

Table 3.2.6 (continued)

Component	Nominal composition (wt %)
Rb_2O	0.0008
RhO_2	0.0138
RuO_2	0.0779
Sb_2O_3	0.0001
SeO_2	0.0001
SiO_2	42.5812
Sm_2O_3	0.0271
SnO_2	0.0005
SrO	0.0046
$SrSO_4$	0.0478
Tc_2O_7	0.0020
ThO_2	3.5338
TeO_2	0.0007
TiO_2	0.8929
UO_2	0.5682
Y_2O_3	0.0184
ZnO	0.0226
ZrO_2	0.1671
Total	99.9999

[a]Based on Revision 7 mass balance (Crocker 1989a). This glass composition is referred to as WVDP Reference 4 glass.

3.3 SAVANNAH RIVER SITE (SRS)

3.3.1 Introduction

Interim forms of high-level waste stored at SRS have been produced since 1954 by the reprocessing of defense reactor fuels and are held in large underground storage tanks. Neutralization and settling of the HLW have resulted in the formation of sludge and supernatant liquid (supernate). Subsequent evaporation of the supernate, which contains almost all of the Cs-137 activity, has produced a saturated salt solution and a salt cake consisting of the salts crystallized out of the saturated solution. Starting in 1992, the sludge and most of the radioactivity in the salt solution and salt cake will be processed in the Defense Waste Processing Facility (DWPF) to produce canisters of borosilicate glass in which the HLW is dispersed and immobilized. Processing of decontaminated salt solution into saltstone started in 1990; the saltstone is low-level waste and is sent to on-site engineered storage.

As of March 1991, design of the DWPF was more than 99% complete, and construction was 99% complete. The facility includes storage buildings where the filled canisters of HLW can be stored on-site until a repository becomes available.

3.3.2 Types of HLW Produced

The glass to be produced at the DWPF is referred to as sludge-precipitate glass and will be made from a blend of (1) washed sludge, (2) washed precipitate made by treating the salt solution to precipitate cesium together with smaller quantities of other radionuclides, and (3) glass frit. The salt solution will include salts redissolved out of the salt cake phase; thus the washed precipitate will contain essentially all of the radioactivity in the salt cake and all of the radioactivity originally in the supernate. A more complete description of the feed preparation process is given in Appendix 3B.

3.3.3 Physical Description

Design details of the DWPF canister are shown in Figs. 3.3.1 and 3.3.2 and Table 3.3.1. The main body of the canister is made of schedule 20 type 304L stainless steel pipe with an outside diameter of 61 cm and a nominal wall thickness of about 0.95 cm. The overall length of the canister is 300 cm (9 ft 10 in.). The nominal inside volume is about 0.74 m^3, and the weight of the empty canister is about 500 kg (1,100 lb). Each canister will contain 0.626 m^3 of glass, or about 1,680 kg (3,710 lb), when loaded to about 85% of its volume. The density of the reference glass is about 2.73 g/m^3 at a temperature of 25°C. The total weight of a loaded canister is therefore about 2,180 kg (4,810 lb) (Baxter 1988, Plodinec 1990).

A temporary seal plug is shrunk-fit into the neck of the canister after the canister is filled. Decontamination of the exterior of the canister is then accomplished by blasting with glass frit. A weld plug is then inserted and the canister is welded shut and transferred to a storage building (Mellen, Burke, and Kitchen 1989).

3.3.4 Inventory and Production Schedule

Table 3.3.2 shows a preliminary projection of glass production from 1992 to 2020 estimated by SRS (Garvin 1990). As shown in the table, the initial production of glass at SRS is scheduled to start in 1992, and it is estimated that about 136 canisters will be produced in that year. The maximum expected vitrification rate is 410 canisters/year. A total of about 5,280 canisters will have been produced by the end of 2020. All canisters produced will be stored on site until a repository becomes available.

Recent information, received too late to be incorporated in the tables in this report, indicate that vitrification at SRS will begin in 1993 rather than 1992 (Hsu 1991). Quantities given in this report are based on the 1992 startup date.

3.3.5 Maximum Radioactivity Per Canister

The radionuclide composition estimated by SRS to represent the most highly radioactive glass likely to be made from sludge-supernate processing is shown in Table 3.3.3; this was based on data in Baxter 1988 and is the best current estimate of maximum activity per canister. Table 3.3.3 is based on sludge aged an average of 5 years and a cesium-containing precipitate derived from supernate aged an average of 15 years. The radionuclide content of sludge-precipitate glass is shown in terms of curies and grams per canister; this was based on 1,682 kg (3,710 lb) of sludge-precipitate glass at the reference-case fill level of 85%.

Table 3.3.4 shows calculated rates of neutron production per canister of HLW glass at the time of filling based on the radionuclide composition given in Table 3.3.3. Table 3.3.4 shows that α, n reactions account for over 80% of the total neutron production. The neutron production rates shown are source terms only, i.e., the rates of neutron production within the glass (Hermann 1992, Salmon 1992).

3.3.6 Radioactivity and Thermal Power vs Time

The maximum expected values of radioactivity and thermal power per canister as a function of decay time after filling were determined by ORIGEN2 calculations based on the radionuclide content per canister shown in Table 3.3.3. The results are shown in Table 3.3.5 in summary form. The total activity and decay heat at the time of filling are 234,400 Ci and 709 W per canister. Detailed tables

showing the contributions of individual radionuclides to total curies and watts per canister over a time span of 0 to 10^6 years are given in Appendix 3A.

SRS has made a forecast of the radionuclide content of the glass produced during each year of vitrification plant operation, and has also made a year-by-year forecast of the cumulative radioactivity and thermal power of the glass (Garvin 1990). Table 3.3.6, which was derived from these data, shows the estimated average radioactivity and thermal power per canister on a cumulative year-by-year basis. The average radioactivity of canisters produced through the year 2020 is considerably less than the maximum radioactivity per canister shown in Table 3.3.3, decayed to the year 2020. For example, at the end of year 2020 the total cumulative radioactivity in glass is about 348×10^6 Ci. If this total cumulative radioactivity is divided by the total number of canisters produced (5,282), the resulting average is 65,900 Ci/canister. The average thermal power, determined in the same way, is 203 W/canister at the end of year 2020. These long-term cumulative averages should be useful for preliminary repository calculations, since it is clear that the averages should give better estimates of overall repository heat loads than would be obtained by multiplying the total number of canisters by the maximum heat load per canister.

Firm estimates of the detailed radionuclide compositions of individual feed batches are not expected to be available until about one year before the start of vitrification of each batch. In 1986 it was estimated that the glass produced during the first five years of operation would not exceed an activity of about 154,000 Ci/canister and a heat generation rate of about 460 W/canister (Baxter 1986). More recent data indicate that the glass produced in 1992 could average about 166,000 Ci and 418 W per canister (Garvin 1990).

Table 3.3.7 shows the current forecast of fission-product radioactivity in the HLW glass produced each year at SRS. Most of this activity is from Sr-90, Y-90, Cs-137, Ba-137m, Tc-99, and Pm-147. Actinides are not shown but would add about 1% to the total curies. Estimates of actinide contents and other details are given in Garvin 1990.

3.3.7 Chemical Composition

Table 3.3.8 shows the approximate range of chemical compositions of borosilicate glass from SRS (Plodinec 1990).

3.3.8 Assessment of Data

The data in Baxter 1988 are the best available at present for the estimation of maximum radioactivity and thermal power per canister. Based on these data and a canister loading of 1,682 kg, it appears that the maximum values of radioactivity and thermal power per canister will not exceed those shown in Table 3.3.5.

SRS has also presented an estimated schedule of the radionuclide content of the vitrification plant feed as a function of the year of operation. Information of this type should be very useful in repository design and overall waste management system planning. However, although these projections were based on the most recent available operating forecasts, actual processing schedules and tankage allocations may change, and the radioactivity and thermal power of canisters produced each year could change accordingly.

3.3.9 References for Section 3.3

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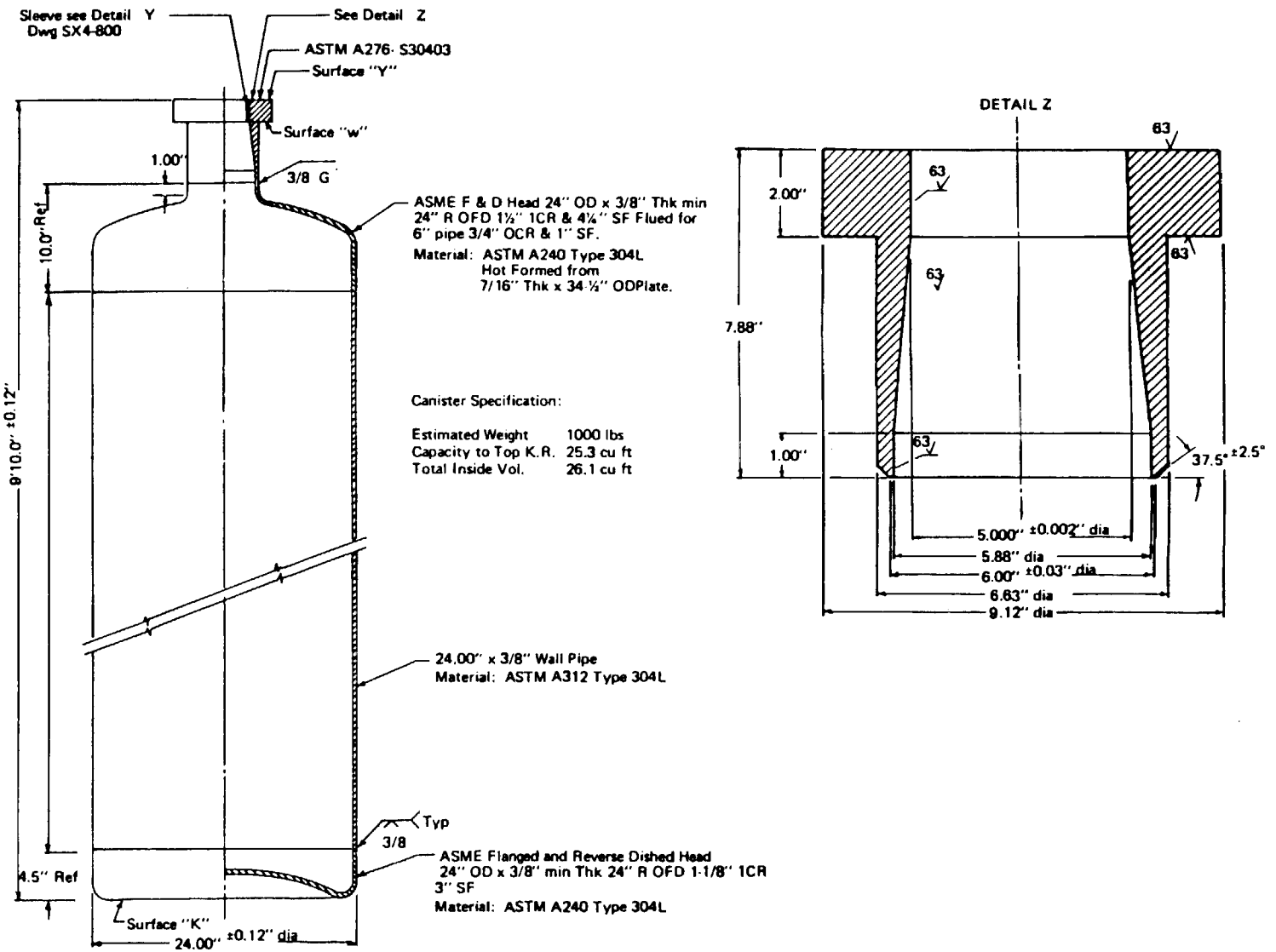

Fig. 3.3.1. Savannah River Site HLW canister. Source: Kelker 1986.

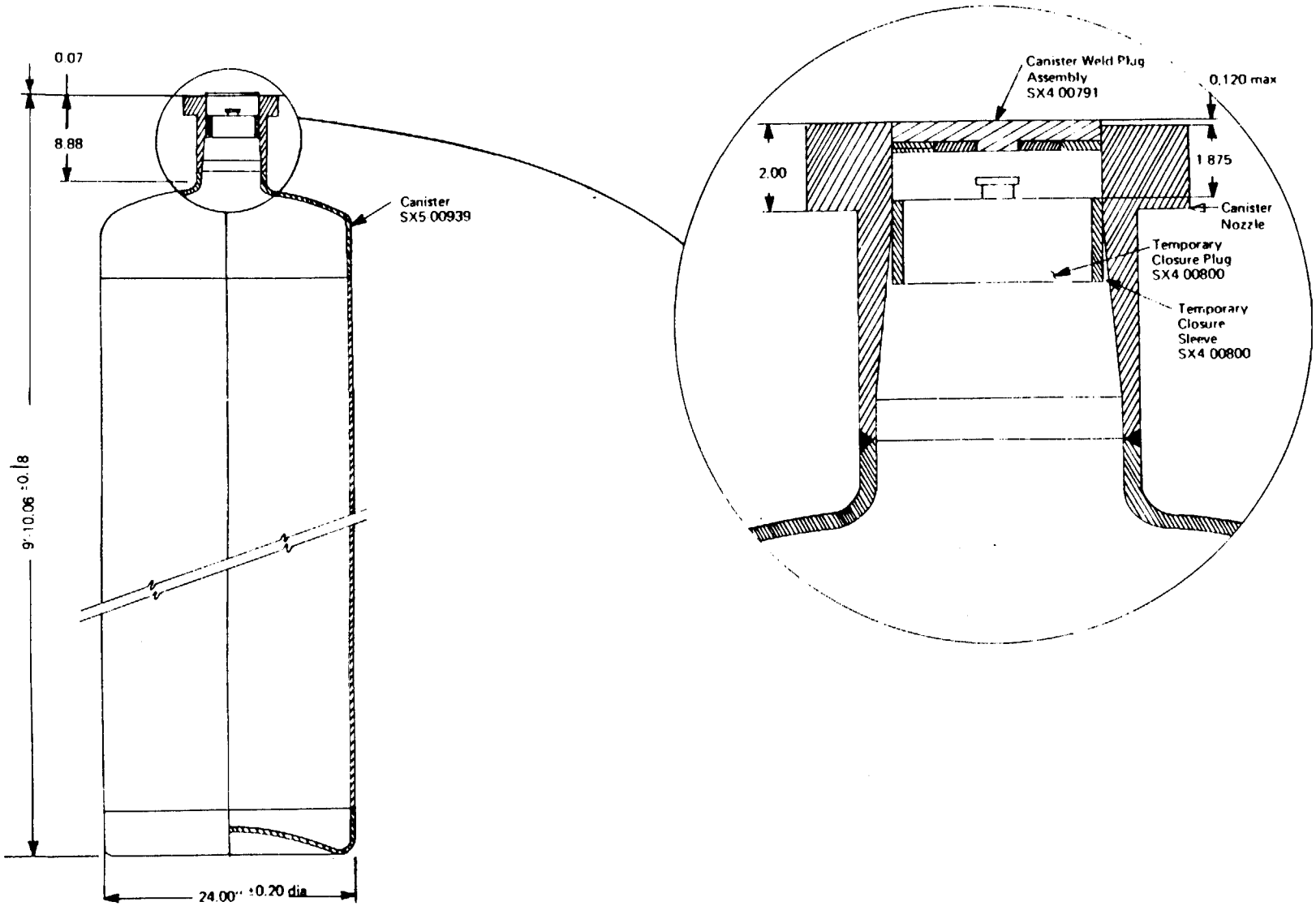

Fig. 3.3.2. Savannah River Site HLW canister closure. Source: Kelker 1986.

Table 3.3.1. Savannah River Site: high-level waste form and canister characteristics[a]

Canister inside volume, m^3	0.736
Canister loading, % of volume	85
Glass volume at 85% fill, nominal (see note b), m^3	0.626
Glass density at 25°C, g/cm^3	2.73
Glass weight, kg	1,682
Canister weight, kg	500
Gross weight, kg	2,182
Total radioactivity, Ci[c]	234,000
Decay heat, W[c]	690

[a]Source: Baxter 1988, Plodinec 1990.

[b]The canister is assumed to be filled to 85% of its volume.

[c]These figures are the ones given in Baxter 1988. The corresponding figures calculated by ORIGEN2 are 234,000 Ci and 709 W, as shown in Table 3.3.4. Activity and decay heat (thermal power) are at the time of filling the canister and are based on the maximum case, i.e., 5-year-old sludge and 15-year-old supernate.

Table 3.3.2. Savannah River Site (DWPF): estimated production schedule of canisters of HLW glass[a]

End of calendar year	Number of canisters produced during year	Cumulative number of canisters produced	Cumulative volume of glass produced, m^3
1992	136	136	85
1993	308	444	278
1994	376	820	513
1995	410	1,230	769
1996	410	1,640	1,025
1997	383	2,023	1,264
1998	369	2,392	1,495
1999	369	2,761	1,726
2000	342	3,103	1,939
2001	342	3,445	2,153
2002	342	3,787	2,367
2003	342	4,129	2,581
2004	302	4,431	2,769
2005	273	4,704	2,940
2006	273	4,977	3,111
2007	273	5,250	3,281
2008	32	5,282	3,301
2009	0	5,282	3,301
2010	0	5,282	3,301
2011	0	5,282	3,301
2012	0	5,282	3,301
2013	0	5,282	3,301
2014	0	5,282	3,301
2015	0	5,282	3,301
2016	0	5,282	3,301
2017	0	5,282	3,301
2018	0	5,282	3,301
2019	0	5,282	3,301
2020	0	5,282	3,301

[a]Production shown is based on a glass melt rate of 228 lb/h and 75% attainment. Canisters (2-ft diam × 10-ft long) are assumed to contain 1,682 kg of glass, which represents filling to 85% capacity. Source: Baxter 1988, Garvin 1990.

Table 3.3.3. Savannah River Site: radionuclide content per HLW canister[a]

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Cr-51	0.1008E-20	0.9312E-16	0.1996E-19
Co-60	0.1502E+00	0.1699E+03	0.2619E+01
Ni-59	0.3163E+00	0.2397E-01	0.9519E-06
Ni-63	0.4824E-01	0.2975E+01	0.3000E-03
Se-79	0.2439E+01	0.1699E+00	0.4232E-04
Rb-87	0.9961E+01	0.8719E-06	0.7278E-09
Sr-89	0.1470E-08	0.4267E-04	0.1473E-06
Sr-90	0.3426E+03	0.4675E+05	0.5426E+02
Y-90	0.8795E-01	0.4786E+05	0.2653E+03
Y-91	0.3085E-07	0.7568E-03	0.2715E-05
Zr-93	0.4443E+03	0.1117E+01	0.1298E-03
Zr-95	0.4680E-06	0.1005E-01	0.5084E-04
Nb-94	0.5147E-03	0.9646E-04	0.9830E-06
Nb-95	0.5407E-06	0.2115E-01	0.1013E-03
Nb-95m	0.3272E-09	0.1247E-03	0.1730E-06
Tc-99	0.1816E+03	0.3079E+01	0.1545E-02
Ru-103	0.5217E-12	0.1684E-07	0.5827E-10
Ru-106	0.6729E+00	0.2252E+04	0.1339E-00
Rh-103m	0.5028E-15	0.1636E-07	0.3761E-11
Rh-106	0.6346E-06	0.2259E+04	0.2167E+02
Pd-107	0.2863E+02	0.1473E-01	0.8732E-06
Ag-110m	0.2647E-04	0.1258E+00	0.2098E-02
Cd-113	0.1472E+00	0.5009E-13	0.8420E-16
Cd-115m	0.4763E-13	0.1213E-08	0.4518E-11
Sn-121m	0.1336E-02	0.7902E-01	0.1581E-03
Sn-123	0.3101E-04	0.2549E+00	0.7951E-03
Sn-126	0.1556E+02	0.4415E+00	0.5508E-03
Sb-124	0.4071E-11	0.7123E-07	0.9445E-09
Sb-125	0.8226E+00	0.8496E+03	0.2656E+01
Sb-126	0.7365E-06	0.6159E-01	0.1138E-02
Sb-126m	0.5619E-08	0.4415E+00	0.5622E-02
Te-126m	0.1532E-01	0.2760E+03	0.2320E+00
Te-127	0.4555E-07	0.1202E+00	0.1622E-03

Table 3.3.3 (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Te-128m	0.1302E-04	0.1228E+00	0.6597E-04
Te-129	0.1457E-18	0.3053E-11	0.1089E-13
Te-129m	0.1576E-15	0.4749E-11	0.8316E-14
Cs-134	0.2606E+00	0.3372E+03	0.3433E+01
Cs-135	0.8633E+02	0.9943E-01	0.3319E-04
Cs-136	0.1068E-43	0.7838E-39	0.1066E-41
Cs-137	0.4989E+03	0.4341E+05	0.4802E+02
Ba-136m	0.3195E-49	0.8607E-38	0.1040E-40
Ba-137m	0.7724E-04	0.4155E+05	0.1632E+03
Ba-140	0.1404E-40	0.1024E-35	0.2853E-38
La-140	0.7734E-42	0.4304E-36	0.7205E-38
Ce-141	0.1260E-14	0.3591E-10	0.5250E-13
Ce-142	0.4005E+03	0.9609E-05	0.0000E+00
Ce-144	0.3093E+01	0.9869E+04	0.6547E+01
Pr-143	0.1780E-38	0.1198E-33	0.2291E-37
Pr-144	0.1306E-03	0.9869E+04	0.7255E+02
Pr-144m	0.6545E-06	0.1187E+03	0.4063E-01
Nd-144	0.4110E+03	0.4860E-09	0.0000E+00
Nd-147	0.1570E-48	0.1261E-43	0.3038E-46
Pm-147	0.2609E+02	0.2419E+05	0.8679E+01
Pm-148	0.4243E-15	0.6975E-10	0.5364E-12
Pm-148m	0.4722E-13	0.1009E-08	0.1277E-10
Sm-147	0.8796E+02	0.2000E-05	0.2738E-07
Sm-148	0.1916E+02	0.5788E-11	0.6901E-13
Sm-149	0.7420E+01	0.1781E-11	0.0000E+00
Sm-151	0.9418E+01	0.2478E+03	0.2906E-01
Eu-152	0.2132E-01	0.3688E+01	0.2790E-01
Eu-154	0.2295E+01	0.6196E+03	0.5543E+01
Eu-155	0.1021E+01	0.4749E+03	0.3455E+00
Eu-156	0.9489E-36	0.5231E-31	0.5392E-33
Tb-160	0.9923E-10	0.1120E-05	0.9110E-08
Tl-208	0.3829E-11	0.1128E-02	0.2645E-04

Table 3.3.3 (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
U-232	0.6256E-03	0.1339E-01	0.4301E-03
U-233	0.1636E-03	0.1584E-05	0.4605E-07
U-234	0.5485E+01	0.3428E-01	0.9875E-03
U-235	0.7278E+02	0.1573E-03	0.4122E-05
U-236	0.1742E+02	0.1128E-02	0.3054E-04
U-238	0.3122E+05	0.1050E-01	0.2663E-03
Np-236	0.1323E-05	0.1744E-07	0.3514E-10
Np-237	0.1263E+02	0.8904E-02	0.2722E-03
Pu-236	0.2297E-03	0.1221E+00	0.4249E-02
Pu-237	0.7401E-15	0.8941E-11	0.3292E-14
Pu-238	0.8667E+02	0.1484E+04	0.4919E+02
Pu-239	0.2076E+03	0.1291E+02	0.3979E+00
Pu-240	0.3809E+02	0.8681E+01	0.2704E+00
Pu-241	0.1620E+02	0.1670E+04	0.5176E-01
Pu-242	0.3206E+01	0.1224E-01	0.3616E-03
Am-241	0.3210E+01	0.1102E+02	0.3661E+00
Am-242	0.1776E-07	0.1436E-01	0.1628E-04
Am-242m	0.1488E-02	0.1447E-01	0.5709E-05
Am-243	0.2902E-01	0.5788E-02	0.1860E-03
Cm-242	0.1057E-04	0.3495E-01	0.1288E-02
Cm-243	0.1078E-03	0.5565E-02	0.2039E-03
Cm-244	0.1329E+01	0.1076E+03	0.3763E+01
Cm-245	0.3910E-04	0.6715E-05	0.2225E-06
Cm-246	0.1739E-05	0.5342E-06	0.1747E-07
Cm-247	0.7116E-08	0.6604E-12	0.2107E-13
Cm-248	0.1614E-09	0.6864E-12	0.8533E-13
Totals	0.3427E+05	0.2344E+06	0.7093E+03

[a]Quantities shown are for sludge-precipitate glass and are based on Baxter 1988, assuming sludge aged an average of 5 years and supernate aged an average of 15 years, with a canister load of 3,710 lb of glass (1,682 kg). Radionuclide contents are at time of filling canister.

Table 3.3.4. Savannah River Site: rates of neutron production per canister of vitrified high-level waste from (alpha, n) reactions and from spontaneous fission[a]

	Neutron production rates per canister, neutrons/s		
Actinide	Alpha, n	Spontaneous fission	Total
U-238	1.331E+02	3.961E+02	5.292E+02
Pu-238	5.791E+07	2.303E+05	5.814E+07
Pu-239	3.969E+05	5.685E+00	3.969E+05
Pu-240	2.680E+05	3.468E+04	3.027E+05
Pu-242	3.064E+02	5.404E+03	5.710E+03
Am-241	4.248E+05	3.981E+00	4.248E+05
Cm-242	1.975E+03	2.278E+02	2.203E+03
Cm-244	5.105E+06	1.478E+07	1.989E+07
Totals	6.411E+07	1.505E+07	7.916E+07

[a]Calculated from canister composition of sludge-precipitate glass at time of pouring. Calculations were made by the ALPHN code (Salmon 1992) for the alpha, n neutron production and by the ORIGEN2 code for spontaneous fission. Canister contains 1,682 kg of glass. Neutron production rates shown are uniform source terms in the glass rather than dose rates at the exterior, i.e., the shielding effects of the glass and the canister wall have not been calculated. Actinides that make a negligible contribution have been omitted; for these see Appendix 3A, Table 9-SRS.

Table 3.3.5. Savannah River Site: calculated radioactivity and thermal power per HLW canister[a]

Decay time (years)[b]	Radioactivity per canister (Ci)[c]	Thermal power per canister (W)[c]
0	234,400	709
1	208,500	627
2	193,800	586
5	169,300	527
10	145,800	467
15	128,400	418
20	113,900	374
30	90,000	301
50	56,500	198
100	17,900	75
200	2,100	17
300	390	7.2
350	227	5.2
500	95	2.7
1,000	42	1.1
1,050	41	1.1
2,000	29	0.72
5,000	24	0.54
10,000	20	0.43
20,000	16	0.30
50,000	11	0.16
100,000	9.2	0.11
500,000	4.8	0.05
1,000,000	2.4	0.02

[a]Based on 5-year cooled sludge and 15-year cooled supernate. Calculations made by ORIGEN2 code based on data supplied from Baxter 1988. Canister is filled to 85% of capacity and contains 1,682 kg of glass.

[b]Years after vitrification.

[c]Radioactivity and thermal power include contributions of actinides and activation products, as well as fission products.

Table 3.3.6. Savannah River Site: estimated cumulative average radioactivity and thermal power per canister of HLW glass[a]

End of calendar year	Cumulative number of canisters produced	Cumulative radioactivity		Cumulative thermal power	
		Total (10^6 Ci)	Per canister (Ci)	Total (10^3 W)	Per canister (W)
1992	136	22.6	116,200	56.9	418.4
1993	444	52.0	117,100	132.6	298.6
1994	820	87.7	107,000	223.1	272.1
1995	1,230	150.0	122,000	412.5	335.4
1996	1,640	211.0	128,700	603.0	367.7
1997	2,023	263.6	130,300	763.0	377.2
1998	2,392	310.8	129,900	908.0	379.6
1999	2,761	345.6	125,200	1,018.8	369.0
2000	3,103	372.2	120,000	1,103.5	355.6
2001	3,445	393.3	114,200	1,170.7	339.8
2002	3,787	410.8	108,500	1,227.4	324.1
2003	4,129	425.7	103,100	1,276.2	309.1
2004	4,431	438.3	98,900	1,318.1	297.5
2005	4,704	449.3	95,500	1,355.0	288.1
2006	4,977	458.7	92,200	1,387.2	278.7
2007	5,250	467.4	89,000	1,415.8	269.7
2008	5,282	459.4	87,000	1,397.3	264.5
2009	5,282	449.8	85,200	1,368.1	259.0
2010	5,282	439.5	83,200	1,339.9	253.7
2011	5,282	429.3	81,300	1,310.3	248.1
2012	5,282	419.4	79,400	1,281.5	242.6
2013	5,282	409.7	77,600	1,253.2	237.3
2014	5,282	399.2	75,600	1,225.6	232.0
2015	5,282	390.9	74,000	1,198.8	227.0
2016	5,282	381.9	72,300	1,172.5	222.0
2017	5,282	373.1	70,600	1,146.5	217.1
2018	5,282	364.5	69,000	1,121.9	212.4
2019	5,282	356.0	67,400	1,097.5	207.8
2020	5,282	347.8	65,900	1,073.7	203.3

[a]Source: Garvin 1990. Year-by-year radioactivity and thermal power per canister do not necessarily represent actual processing schedules and tankage allocations and should not be used for design purposes.

Table 3.3.7. Projected radioactivity of canisters of HLW produced each year at SRS[a]

End of calendar year	Canisters produced per year	Radioactivity per canister produced in that year, Ci[b]
1992	136	164,600
1993	308	96,500
1994	376	97,400
1995	410	156,400
1996	410	156,500
1997	383	149,600
1998	369	143,900
1999	369	113,600
2000	342	101,100
2001	342	86,700
2002	342	77,900
2003	342	71,500
2004	302	74,700
2005	273	77,600
2006	273	72,900
2007	273	71,000
2008	32	121,000

[a]Source: Garvin 1990.

[b]Radioactivity shown is for fission products only. When actinides are included, the radioactivities shown will increase by about 1%.

Table 3.3.8. Savannah River Site: projected DWPF waste glass compositions[a]

Major glass components	Amount in constituent sludge type, wt %						
	Blend	Batch 1	Batch 2	Batch 3	Batch 4	HM	Purex
Al_2O_3	3.98	4.87	4.46	3.25	3.32	7.08	2.89
B_2O_3	8.01	7.69	7.70	7.69	8.11	6.94	10.21
$BaSO_4$	0.27	0.22	0.24	0.26	0.38	0.18	0.29
CaO	0.97	1.17	1.00	0.93	0.83	1.00	1.02
$CaSO_4$	0.077	0.12	0.11	0.10	0.003	Trace	0.12
Cr_2O_3	0.12	0.10	0.12	0.13	0.14	0.086	0.14
CuO	0.44	0.40	0.41	0.40	0.46	0.25	0.42
Fe_2O_3	6.95	8.39	7.11	7.48	7.59	4.95	8.54
FeO	3.11	3.72	3.15	3.31	3.36	2.19	3.78
Group A[b]	0.14	0.099	0.14	0.10	0.20	0.20	0.078
Group B[c]	0.36	0.22	0.44	0.25	0.60	0.89	0.084
K_2O	3.86	3.49	3.50	3.47	3.99	2.14	3.58
Li_2O	4.40	4.42	4.42	4.42	4.32	4.62	3.12
MgO	1.35	1.36	1.35	1.35	1.38	1.45	1.33
MnO	2.03	2.06	1.62	1.81	3.08	2.07	1.99
Na_2O	8.73	8.62	8.61	8.51	8.88	8.17	12.14
Na_2SO_4	0.10	0.10	0.12	0.096	0.13	0.14	0.12
NaCl	0.19	0.31	0.23	0.22	0.090	0.093	0.26
NiO	0.89	0.75	0.90	1.07	1.09	0.40	1.21
SiO_2	50.20	49.81	50.17	49.98	49.29	54.39	44.56
ThO_2	0.19	0.36	0.63	0.77	0.24	0.55	0.011
TiO_2	0.90	0.66	0.67	0.66	1.02	0.55	0.65
U_3O_8	2.14	0.53	2.30	3.16	0.79	1.01	2.89

[a]Source: Plodinec 1990.

[b]Group A: radionuclides of Tc, Se, Te, Rb, and Mo.

[c]Group B: radionuclides of Ag, Cd, Cr, Pd, Tl, La, Ce, Pr, Pm, Nd, Sm, Tb, Sn, Sb, Co, Zr, Nb, Eu, Np, Am, and Cm.

3.4 HANFORD SITE (HANF)

3.4.1 Introduction

The HLW currently stored at HANF was generated by the reprocessing of irradiated fuel from production reactors for the recovery of uranium, plutonium, and other elements. The Hanford Waste Vitrification Plant (HWVP) is now in the design stage. The plant will vitrify pretreated HLW in a borosilicate glass which will be cast into stainless steel canisters. Vitrification of high-level waste is currently (1990) scheduled to begin in 1999. Maximum use will be made of existing technology, such as that developed in the design of the Defense Waste Processing Facility at SRS.

Current HWVP design activities are important to meet the milestone for the initiation of HWVP construction (July 1991) that was specified in the Tri-Party Agreement of May 1989 (TPA 1989). This agreement, entered into by DOE, EPA, and the Washington State Department of Ecology, also lists milestones for the completion of HWVP construction (June 1998) and the initiation of vitrification operations (December 1999). The HWVP is designed with a 40-year life, which should allow for the vitrification of single-shell tank (SST) waste if a decision is made to vitrify some or all of this waste (HANF 1989).

3.4.2 Types of HLW Produced

Most of the wastes stored in tanks at Hanford are mixtures and cannot be classified simply as HLW, LLW, or TRU waste. Many of these wastes will require some degree of pretreatment and/or separation using processes currently under development and evaluation. The result of these pretreatment and separation operations will be products that can be classified as HLW, LLW, or TRU wastes. It is expected that some of these products will be acceptable for near-surface disposal and others will require geologic repository disposal. This section is concerned primarily with those waste streams that are possible candidates for vitrification and geologic disposal; additional data on the characteristics and treatment of interim wastes are given in Appendix 3B.

There are several types of interim waste stored at Hanford that are possible candidates for vitrification. These wastes, and the degree to which they are currently committed to vitrification, are listed in Table 3.4.1. This table also shows the number of canisters potentially producible from each type of waste, as currently estimated (Turner 1990).

Present plans are that four types of waste, all of which are contained in double-shell tanks (DSTs), will be vitrified. These are (1) neutralized current acid waste (NCAW), (2) neutralized cladding removal waste (NCRW), (3) Plutonium Finishing Plant waste (PFP), and (4) complexant concentrate (CC). At present, it appears that these will be vitrified separately; however, it is possible that NCRW and PFP might be combined. The NCAW has the highest radioactivity and is expected to be the initial feed to the vitrification plant. The remaining feeds are expected to follow in the order NCRW, PFP, and CC. According to these assumptions, a total of 1960 canisters of vitrified HLW will be produced at Hanford. It is possible that volume-reduction processes currently under development may reduce this total somewhat (HANF 1989). Also, Hanford has noted that the 500 canisters produced from PFP and NCRW waste meet the definition of transuranic waste and may be candidates for disposal at the Waste Isolation Pilot Plant; however, no decision on this has been made (Turner 1990, Watrous 1992).

3.4.2.1 Sources of Potential Vitrification Plant Feeds

The NCAW is a two-phase (solid-liquid) waste produced by neutralizing high-level first-cycle raffinate from the reprocessing of N-Reactor fuels at the PUREX plant. It can be stored in any of the four double-shell tanks (called "aging waste tanks") that are specifically designed to contain this waste. The solid phase, consisting of sludge-like solids, contains the bulk of the insoluble radionuclides, mainly in the form of hydroxides or hydrated oxides. The supernatant liquid contains most of the cesium and technetium. Pretreatment of the NCAW will provide the principal feed to the vitrification plant.

The NCRW is an alkaline solid-liquid waste resulting from chemical dissolution of the zirconium-alloy N-Reactor fuel cladding material, followed by addition of sodium hydroxide to the decladding waste. The solid phase consists mainly of a hydrated oxide precipitate of zirconium plus various actinides and fission products. The supernatant liquid is a dilute aqueous solution of nitrates and hydroxides.

The CC contains organic compounds and their degradation products resulting from the use of chelating agents in strontium recovery processing. To separate CC waste into feed streams acceptable for immobilization as grout or glass would most likely require solid/liquid separation, solids dissolution, and TRUEX processing (Jain and Barnes 1989).

The PFP waste is a solid/liquid mixture originating from solvent extraction, ion exchange, plutonium nitrate-to-metal conversion, scrap stabilization, and laboratory activities at the Plutonium Finishing Plant (Z-plant). The settled sludge contains the bulk of the TRU contaminants (Jain and Barnes 1989, HANF 1988).

The only other HLW forms produced at HANF of interest to the repository are strontium and cesium capsules. These are discussed in Appendix 3B. It is possible that these capsules may be opened at Hanford and their contents combined with other feeds to vitrification; at present, however, there has been no decision to do this. It has been definitely decided that the strontium and cesium will go to the repository, but whether they will go as

capsules enclosed in overpacks or in vitrified form has not been decided (HANF 1989).

3.4.3 Physical Description

The HLW canisters are made of type 304L stainless steel pipe with an outside diameter of 61 cm, a length of 300 cm, and a thickness of 0.95 cm. Figures 3.4.1 and 3.4.2 are sketches of the canister and neck detail showing relevant dimensions. The canister is identical to that planned for use at the Savannah River DWPF. Additional information on the canister and HLW glass is given in Table 3.4.2. The fill level of the HWVP canister is approximately 85% of the available internal canister fill volume, resulting in a canister glass volume of 0.62 m^3 (22 ft^3). A 15% void volume minimizes the potential of canister overfill.

The density of the HWVP glass is 2.64 g/cm^3 (165 lb/ft^3) at 100°C. A glass volume of 0.62 m^3 (22 ft^3) corresponds to a glass weight of 1,650 kg (3,630 lb). The total weight of the filled canister is approximately 2,100 kg (4,630 lb). Fill height to the 85% level is approximately 2.3 m (7.5 ft) measured from the bottom of the canister (Mitchell and Nelson 1988).

3.4.4 Inventory and Production Schedule

Estimated annual canister production rates for the vitrified waste are shown in Table 3.4.3. The HWVP is planned to start up in December 1999. The HWVP design throughput is 370 canisters per year. After every three years of HWVP operations, there is a six-month shutdown for melter change-out. Table 3.4.3 indicates about half of a normal year's production in the years in which melter change-out occurs. Hanford is currently planning a total production of 1960 canisters of vitrified HLW consisting of 480 canisters of NCAW glass, 400 canisters of NCRW glass, 100 canisters of PFP glass, and 980 canisters of CC glass. Future additions to or revisions of this schedule are of course possible. Volumetric reduction processes currently under development may reduce the total. All canisters of vitrified HLW glass are expected to be stored on-site; a storage facility is planned with a capacity of 2,000 canisters (HANF 1989).

It is assumed in the vitrification schedule shown here that the HANF reference plan will be followed. In this plan, the SST wastes are not vitrified but are immobilized in place. The decision as to whether the SST wastes are vitrified is dependent on the outcome of the Environmental Impact Statement (EIS) process. If the SST wastes are vitrified, these canisters might also go to a repository, but no decision has been reached on this. It has not been determined that SST waste and certain DST wastes are HLW, although these wastes are managed as HLW for convenience (HANF 1989).

3.4.5 Maximum Radioactivity Per Canister

Radioisotopic data describing the compositions of "maximum" and "nominal" canisters of glass made from NCAW were supplied by Hanford (Mitchell and Nelson 1988) and are shown in Tables 3.4.4 and 3.4.5. Maximum values of activity and thermal power represent the most active waste expected to be fed to the vitrification plant, according to present plans.

There are no plans at HANF to produce any more Sr and Cs capsules; none of these have been produced since 1985. A decision as to the precise method of disposal of the existing capsules has not yet been made. One possibility is to place the capsules in overpacks for repository emplacement, as described in Appendix 3B. Another possibility is to open the capsules and combine the Sr and Cs with one of the low-heat feeds to the HWVP for vitrification. If this is done, the maximum radioactivity and thermal output per canister could conceivably exceed the quantities shown in Table 3.4.5. At the time of the previous issue of this report (1987) there were 640 Sr fluoride and 1,576 Cs chloride capsules. Some of these have been dismantled, and at present (1990) it is projected that 597 Sr and 1,350 Cs capsules will require repository disposal (Wojtasek 1989, Turner 1990).

Table 3.4.6 gives calculated neutron production rates per canister from spontaneous fission and from (alpha, n) reactions. These are for the "maximum" canister composition given in Table 3.4.4 based on NCAW glass and are at the time of filling the canister. Neutron production rates shown are source terms within the glass rather than dose rates at the exterior (Salmon 1992).

3.4.6 Radioactivity and Thermal Power vs Time

Based on the nominal and maximum isotopic data supplied by Hanford, ORIGEN2 calculations were made to determine the estimated radioactivity and thermal power per canister of HLW glass made from NCAW. Table 3.4.7 shows the calculated radioactivity and thermal power per canister for decay times ranging from 0 to 10^6 years.

More detailed tables showing the contributions of individual radionuclides to the radioactivity and thermal power of the "maximum" NCAW glass on a per-canister basis for decay times from 0 to 10^6 years are given in Appendix 3A.

Because radionuclide compositions of the glasses produced from CC, PFP, and NCRW are not available, no calculations of radioactivity and thermal power per canister as functions of decay time have been made for glasses produced from those streams.

In 1987, based on the schedule in use at that time, Hanford provided estimates of annual and cumulative radioactivities of the vitrified waste on a year-by-year basis from 1996 to 2020. These estimates were shown in Tables 3.4.5 and 3.4.6 of the 1987 edition of this report. In 1987,

Table 3.4.5 showed the average radioactivity per canister on an annual "as produced" basis, and Table 3.4.6 showed the average radioactivity per canister on a cumulative basis; the amounts of radioactivity per canister were calculated by dividing Hanford's estimates of annual or cumulative radioactivity in vitrified form by the annual or cumulative number of canisters. Similar estimates of cumulative average radioactivity per canister are not provided in the present report, since the vitrification schedule is different and new estimates of average radioactivity per canister on a year-by-year basis are not yet available.

3.4.7 Chemical Composition

The latest NCAW glass composition, designated HW39-4, is shown in Table 3.4.8. The final glass composition is based on 25 wt % waste oxides and 75 wt % glass frit. The frit composition will be modified as necessary to accommodate variations in NCAW composition (Mitchell 1986, Mitchell and Nelson 1988).

3.4.8 Assessment of Data

Maximum and nominal values of the radioactivity and thermal power of NCAW glass have been established to the extent possible at the present time. Additional information on the glasses made from other feeds to the plant would also be useful, since approximately 1,500 canisters of these glasses will be produced. It is expected that the radioactivity per canister of these glasses will be low relative to NCAW glass, but an exception to this could take place if it is decided to blend the contents of the Sr and Cs capsules into one of the low-heat feeds to the vitrification plant. Estimates of average thermal power per canister of glass on a year-by-year basis would be useful in estimating repository thermal loadings and temperatures.

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Wolfe 1985b. Personal communication, B. A. Wolfe, Rockwell Hanford Operations, to R. Salmon, ORNL, November 12, 1985.

ORNL DWG 90-421

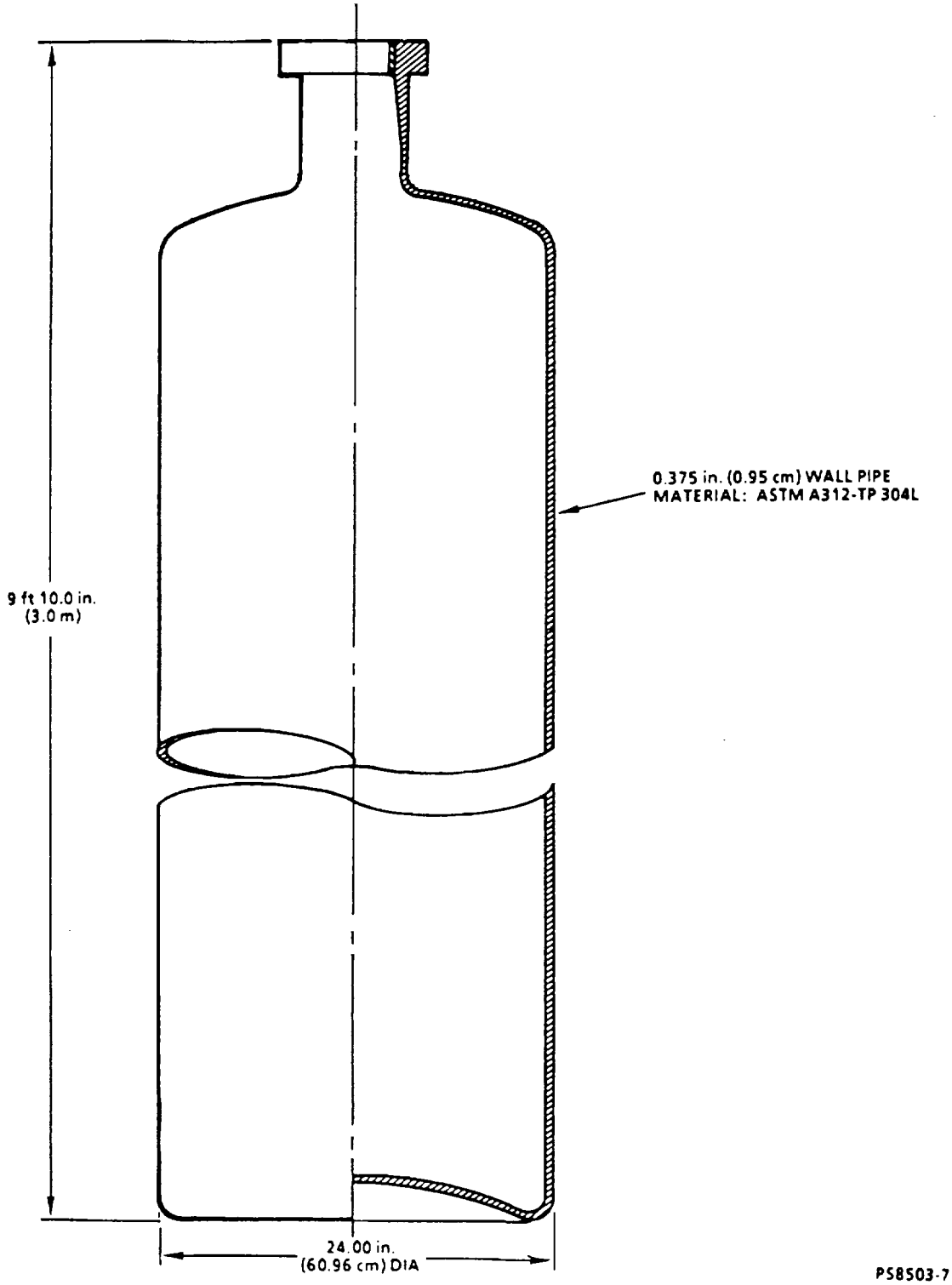

Fig. 3.4.1. Hanford HLW canister. Source: Mitchell and Nelson 1988.

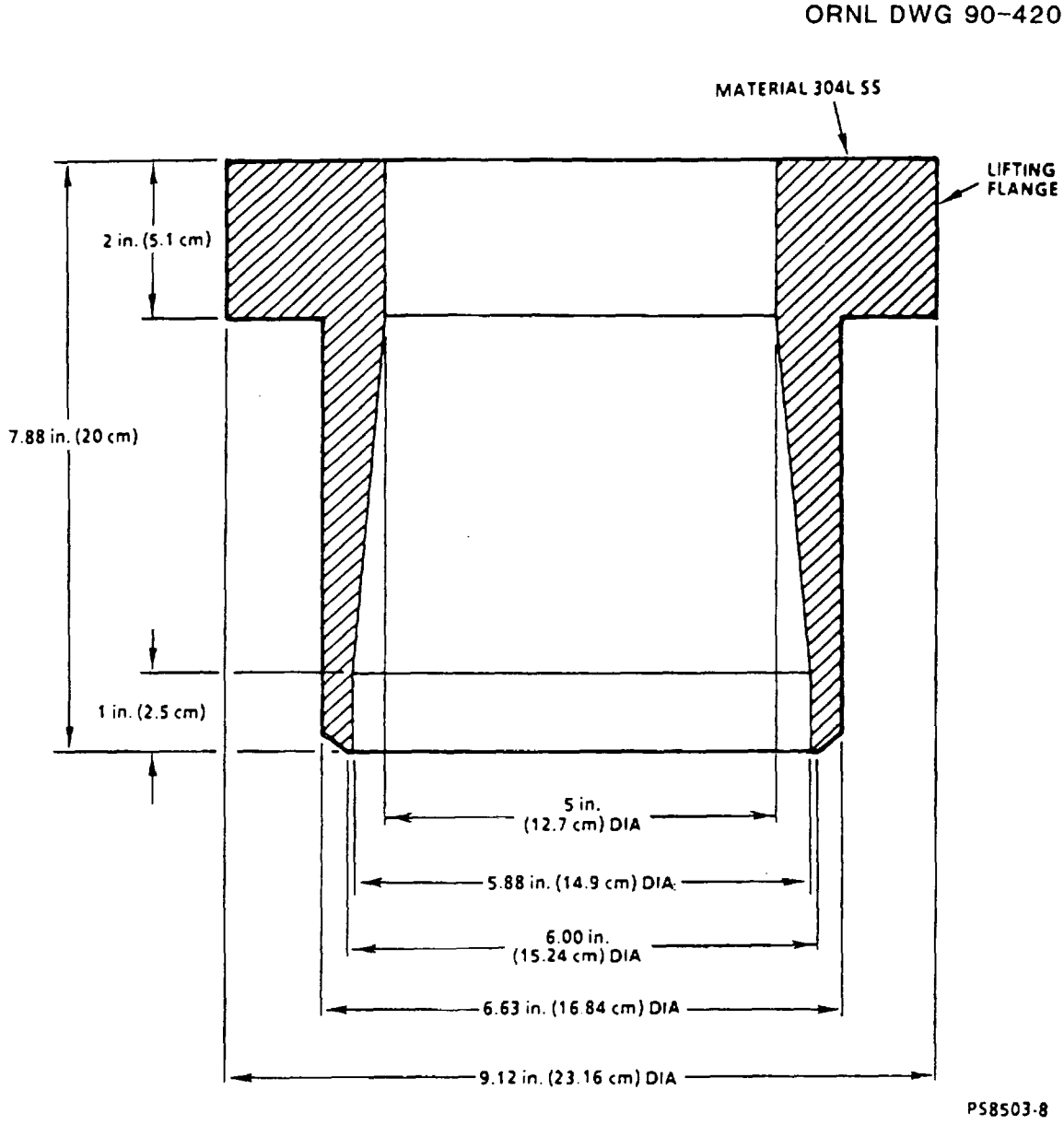

Fig. 3.4.2. Hanford HLW canister neck detail. Source: Mitchell and Nelson 1988.

Table 3.4.1. Hanford Waste Vitrification Plant: current (1990) estimates of canister production from committed and potential feedstocks[a]

Interim waste type	Degree of commitment of this waste to vitrification	Number of canisters potentially producible
Slurry from double-shell tanks:		
NCAW[b]	Committed	480
CC - existing	High potential	150–580
CC - future	Some potential	0–400
PFP	High potential[c]	100
NCRW	High potential[c]	400
Total		1,130–1,960
Liquid, sludge, and salt cake from single-shell tanks	Not currently included[d]	7,600

[a]Source: Hanford submittal to IDB 1989, updated in accordance with HANF 1989a and Turner 1990.

[b]NCAW = neutralized current acid waste; CC = complexant concentrate; PFP = Plutonium Finishing Plant waste; and NCRW = neutralized cladding removal waste.

[c]PFP and NCRW were not originally included in the mission plan of the Hanford Waste Vitrification Plant, but are now expected to be vitrified. Hanford states that these 500 canisters will meet the definition of transuranic waste (Turner 1990).

[d]It is estimated that 7,600 canisters would be produced if the waste in all 149 single-shell tanks were retrieved and processed by the TRUEX process; however, the decision to exhume and process this waste has not been made.

Table 3.4.2. Hanford Site: high-level waste form and canister characteristics[a]

Waste form	Borosilicate glass in sealed steel canister
Canister material	Type 304L stainless steel
Weights per canister, kg	
Empty canister	500
Borosilicate glass	1,650
Total loaded weight	2,150
Canister dimensions	
Outside diameter, cm	61
Height overall, cm	300
Wall thickness, cm	0.95
Inside volume, m^3	0.736
Glass volume at average fill temperature, m^3	0.626[b]
Radionuclide content, curies per canister[c]	
Nominal	137,000
Maximum	298,000
Thermal power, watts per canister[c]	
Nominal	389
Maximum	869

[a]Sources: Wolfe 1985, White 1986, Mitchell and Nelson 1988.

[b]Canister is filled to 85% of volume at average fill temperature of 825°C.

[c]All values shown are based on NCAW reference feed (neutralized current acid waste) with 25 wt % waste oxide in glass. Activities and thermal power are at time of filling canister. Range of values shown is from Mitchell and Nelson 1988 in which estimated activities and radionuclide compositions were given for two NCAW feeds referred to as nominal and maximum. Radionuclide compositions are shown in Table 3.4.4.

Table 3.4.3. Hanford Site: estimated production schedule of canisters of HLW glass[a]

End of calendar year	Number of canisters produced	Cumulative number of canisters
1990	0	0
1991	0	0
1992	0	0
1993	0	0
1994	0	0
1995	0	0
1996	0	0
1997	0	0
1998	0	0
1999	0	0
2000	240	240
2001	370	610
2002	345	955
2003	185	1,140
2004	370	1,510
2005	370	1,880
2006	80	1,960
2007	0	1,960
2008	0	1,960
2009	0	1,960
2010	0	1,960
2011	0	1,960
2012	0	1,960
2013	0	1,960
2014	0	1,960
2015	0	1,960
2016	0	1,960
2017	0	1,960
2018	0	1,960
2019	0	1,960
2020	0	1,960

[a]Sources: HANF 1989a, Turner 1990. The total canister production of 1,960 canisters represents 480 canisters of NCAW glass, 400 canisters of NCRW glass, 100 canisters of PFP glass, and 980 canisters of complexant concentrate glass. Hot startup is assumed to occur in December 1999, with zero production of glass in that month. Production of overpack canisters of Sr and Cs capsules (if any) is not included in this table, since such canisters would not involve the vitrification plant. The 500 canisters of PFP and NCRW waste meet the definition of transuranic waste and are candidates for disposal at the Waste Isolation Pilot Plant (Turner 1990).

Table 3.4.4. Hanford Site: radionuclide content per HLW canister, NCAW glass, maximum case[a]

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Fe-55	5.64E-02	1.41E+02	4.765E-03
Ni-59	1.80E+00	1.36E-01	5.402E-06
Co-60	3.79E-03	4.29E+00	6.615E-02
Ni-63	2.54E-01	1.57E+01	6.236E-03
Se-79	5.60E-02	3.90E-03	9.711E-07
Sr-89	2.24E-06	6.52E-02	2.254E-04
Sr-90	3.06E+02	4.18E+04	4.852E+01
Y-90	7.68E-02	4.18E+04	2.317E+02
Y-91	2.96E-05	7.26E-01	2.608E-03
Nb-93m	2.04E-03	5.77E-01	1.023E-04
Zr-93	5.13E+02	1.29E+00	1.499E-04
Zr-95	1.28E-04	2.76E+00	1.398E-02
Nb-95	1.45E-04	5.67E+00	2.720E-02
Tc-99	5.51E+02	9.35E+00	4.689E-03
Ru-103	9.23E-09	2.98E-04	9.971E-07
Rh-103m	8.27E-12	2.69E-04	6.192E-08
Ru-106	1.49E+00	4.99E+03	2.967E-01
Rh-106	1.40E-06	4.99E+03	4.786E+01
Pd-107	7.91E+01	4.07E-02	2.413E-06
Ag-110m	3.35E-04	1.59E+00	2.655E-02
Cd-113m	6.73E-02	1.46E+01	2.458E-02
In-113m	1.51E-09	2.52E-02	5.871E-05
Sn-113	2.51E-06	2.52E-02	4.198E-06
Cd-115m	3.45E-10	8.78E-06	3.275E-08
Sn-119m	1.21E-03	5.42E+00	2.802E-03
Sn-121m	1.79E-03	1.06E-01	2.124E-04
Sn-123	3.52E-04	2.89E+00	9.027E-03
Sn-126	1.62E+01	4.60E-01	5.738E-04
Sb-124	2.00E-09	3.50E-05	4.648E-07
Sb-126	7.75E-07	6.48E-02	1.197E-03
Sb-126m	5.86E-09	4.60E-01	5.858E-03
Sb-125	1.70E+00	1.76E+03	5.503E+00
Te-125m	2.38E-02	4.29E+02	3.606E-01

Table 3.4.4 (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Te-127	1.12E-06	2.95E+00	3.984E-03
Te-127m	3.18E-04	3.00E+00	1.614E-03
Te-129	1.79E-14	3.75E-07	1.340E-09
Te-129m	1.91E-11	5.77E-07	1.012E-09
I-129	9.23E-02	1.63E-05	7.541E-09
Cs-134	9.27E-01	1.20E+03	1.221E+01
Cs-135	2.18E+02	2.51E-01	8.378E-05
Cs-137	5.86E+02	5.10E+04	5.642E+01
Ba-137m	8.96E-05	4.82E+04	1.893E+02
Ce-141	3.97E-10	1.13E-05	1.655E-08
Ce-144	9.34E+00	2.98E+04	1.977E+01
Pr-144	3.94E-04	2.98E+04	2.191E+02
Pr-144m	1.97E-06	3.58E+02	1.225E-01
Pm-147	4.28E+01	3.97E+04	1.424E+01
Pm-148m	6.18E-10	1.32E-05	1.674E-07
Sm-151	3.18E+01	8.36E+02	9.803E-02
Eu-152	1.58E-02	2.74E+00	2.073E-02
Gd-153	3.26E-06	1.15E-02	1.039E-05
Eu-154	1.24E+00	3.36E+02	3.006E+00
Eu-155	8.83E-01	4.11E+02	2.990E-01
Tb-160	9.74E-09	1.10E-04	8.961E-07
U-234	7.71E-01	4.82E-03	1.388E-04
U-235	9.11E+01	1.97E-04	5.160E-06
U-236	7.34E+00	4.75E-04	1.287E-05
U-238	1.11E+04	3.72E-03	9.437E-05
Np-237	2.82E+02	1.99E-01	6.083E-03
Pu-238	4.48E-02	7.68E-01	2.546E-02
Pu-239	2.27E+01	1.41E+00	4.346E-02
Pu-240	2.38E+00	5.42E-01	1.688E-02
Pu-241	2.50E-01	2.58E+01	7.999E-04
Pu-242	3.43E-02	1.31E-04	3.869E-06
Am-241	1.68E+02	5.77E+02	1.917E+01
Am-242	5.12E-07	4.14E-01	4.700E-04

Table 3.4.4 (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Am-243	3.39E-01	6.76E-02	2.173E-03
Cm-242	1.51E-04	4.99E-01	1.839E-02
Cm-244	1.54E-01	1.25E+01	4.374E-01
Total	1.40E+04	2.98E+05	8.687E+02

[a]This table identifies the maximum expected activity of HWVP canisters at the time of vitrification. The maximum is principally based on close-coupling the final accumulated tank of NCAW (21 months from fuel discharge to HWVP). Canister contains 1,650 kg of HLW glass (85% fill). Source: Mitchell and Nelson 1988.

Table 3.4.5. Hanford Site: radionuclide content per HLW canister, NCAW glass, nominal case[a]

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Fe-55	7.20E-03	1.80E+01	6.083E-04
Ni-59	1.44E+00	1.09E-01	4.330E-06
Co-60	1.33E-03	1.50E+00	2.313E-02
Ni-63	1.96E-01	1.21E+01	4.806E-03
Se-79	4.52E-02	3.15E-03	7.843E-07
Sr-89	1.84E-17	5.35E-13	1.850E-15
Sr-90	2.18E+02	2.98E+04	3.459E+01
Y-90	5.48E-02	2.98E+04	1.652E+02
Y-91	5.63E-15	1.38E-10	4.957E-13
Nb-93m	2.18E-03	6.16E-01	1.092E-04
Zr-93	4.18E+02	1.05E+00	1.220E-04
Zr-95	1.36E-13	2.92E-09	1.479E-11
Nb-95	1.72E-13	6.73E-09	3.229E-11
Tc-99	4.43E+02	7.51E+00	3.767E-03
Ru-103	1.04E-22	3.37E-18	1.126E-20
Rh-103m	9.34E-26	3.04E-18	6.988E-22
Ru-106	1.25E-02	4.18E+01	2.486E-03
Rh-106	1.17E-08	4.18E+01	4.009E-01
Pd-107	5.87E+01	3.02E-02	1.790E-06
Ag-110m	4.67E-07	2.22E-03	3.708E-05
Cd-113m	3.93E-02	8.53E+00	1.436E-02
In-113m	6.04E-15	1.01E-07	2.353E-10
Sn-113	1.01E-11	1.01E-07	1.683E-11
Cd-115m	1.26E-22	3.20E-18	1.192E-20
Sn-119m	1.52E-06	6.80E-03	3.516E-06
Sn-121m	1.31E-03	7.76E-02	1.555E-04
Sn-123	4.44E-09	3.65E-05	1.140E-07
Sn-126	1.29E+01	3.65E-01	4.553E-04
Sb-124	6.57E-19	1.15E-14	1.527E-16
Sb-126	6.10E-07	5.10E-02	9.424E-04
Sb-126m	4.65E-09	3.65E-01	4.648E-03
Sb-125	2.46E-01	2.54E+02	7.942E-01
Te-125m	3.44E-03	6.20E+01	5.212E-02
Te-127	2.48E-12	6.55E-06	8.846E-09
Te-127m	7.06E-10	6.66E-06	3.583E-09
Te-129	1.49E-30	3.14E-23	1.120E-25

Table 3.4.5 (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Te-129m	1.60E-27	4.82E-23	8.440E-26
I-129	7.31E-02	1.29E-05	5.968E-09
Cs-134	7.19E-02	9.31E+01	9.476E-01
Cs-135	1.75E+02	2.02E-01	6.742E-05
Cs-137	4.15E+02	3.61E+04	3.994E+01
Ba-137m	6.32E-05	3.40E+04	1.335E+02
Ce-141	1.03E-26	2.93E-22	4.284E-25
Ce-144	2.51E-02	8.00E+01	5.307E-02
Pr-144	1.06E-06	8.00E+01	5.881E-01
Pr-144m	5.29E-09	9.60E-01	3.285E-04
Pm-147	5.62E+00	5.21E+03	1.869E+00
Pm-148m	2.92E-23	6.23E-19	7.889E-21
Sm-151	2.65E+01	6.98E+02	8.185E-02
Eu-152	8.09E-03	1.40E-00	1.059E-02
Gd-153	3.83E-09	1.35E-05	1.220E-08
Eu-154	5.37E-01	1.45E+02	1.297E+00
Eu-155	2.94E-01	1.37E+02	9.965E-02
Tb-160	8.41E-17	9.49E-13	7.730E-15
U-234	7.31E-01	4.57E-03	1.316E-04
U-235	8.83E+01	1.91E-04	5.003E-06
U-236	6.51E+00	4.21E-04	1.141E-05
U-238	1.04E+04	3.51E-03	8.904E-05
Np-237	2.21E+02	1.56E-01	4.769E-03
Pu-238	2.59E-02	4.43E-01	1.468E-02
Pu-239	1.88E+01	1.17E+00	3.606E-02
Pu-240	1.72E+00	3.93E-01	1.224E-02
Pu-241	1.22E-01	1.26E+01	3.907E-04
Pu-242	1.99E-02	7.61E-05	2.248E-06
Am-241	8.27E+01	2.84E+02	9.436E+00
Am-242	2.73E-07	2.21E-01	2.509E-04
Am-243	1.90E-01	3.79E-02	1.218E-03
Cm-242	5.50E-05	1.82E-01	6.707E-03
Cm-244	6.22E-02	5.03E+00	1.760E-01
Total	1.26E+04	1.37E+05	3.892E+02

[a]This table identifies the nominal expected activity of HWVP canisters at the time of vitrification. Canister contains 1,650 kg of HLW glass (85% fill). Source: Mitchell and Nelson 1988.

Table 3.4.6. Hanford Site: rates of neutron production per canister of vitrified high-level waste from (alpha, n) reactions and from spontaneous fission[a]

	Neutron production rates per canister, neutrons/s		
Actinide	Alpha, n	Spontaneous fission	Total
U-238	5.130E+01	1.404E+02	1.917E+02
Pu-238	3.236E+04	1.192E+02	3.248E+04
Pu-239	4.688E+04	6.209E-01	4.688E+04
Pu-240	1.809E+04	2.165E+03	2.026E+04
Pu-242	3.549E+00	5.782E+01	6.137E+01
Am-241	2.402E+07	2.088E+02	2.402E+07
Cm-242	3.033E+04	3.251E+03	3.358E+04
Cm-244	6.389E+05	1.718E+06	2.358E+06
Totals	2.479E+07	1.724E+06	2.651E+07

[a]Calculated from "maximum" canister composition at time of pouring, using the ALPHN code (Salmon 1992) for alpha, n neutron production and the ORIGEN2 code for spontaneous fission. Canister contains 1,650 kg of HLW glass. Neutron production rates shown are uniform source terms in the glass rather than dose rates at the exterior, i.e., the shielding effects of the glass and the canister wall have not been calculated. Actinides that make a negligible contribution to totals have been omitted; for these see Appendix 3A, Table 9 - HANF (MAX). Glass composition is from Mitchell and Nelson 1988.

Table 3.4.7. Hanford Site: calculated radioactivity and thermal power per HLW canister[a]

Decay time (years)[b]	Radioactivity per canister (Ci)		Thermal power per canister (W)	
	Nominal	Maximum	Nominal	Maximum
0	136,900	298,300	389	869
1	132,600	243,600	380	683
2	128,500	214,600	370	595
5	118,200	177,100	344	502
10	104,200	149,400	306	439
15	92,500	131,000	273	391
20	82,300	116,100	243	349
30	65,200	91,900	194	279
50	41,000	57,800	125	181
100	13,100	18,500	44	67
200	1,570	2,310	10	19
300	375	621	6.2	12
350	260	454	5.6	11
500	157	295	4.3	8.7
1,000	70	133	2.0	3.9
1,050	66	123	1.8	3.6
2,000	24	39	0.44	0.86
5,000	12	16	0.06	0.08
10,000	12	15	0.05	0.06
20,000	11	14	0.04	0.05
50,000	10	13	0.03	0.04
100,000	9.2	12	0.03	0.04
500,000	5.3	7.0	0.05	0.07
1,000,000	3.6	4.9	0.04	0.07

[a]Calculations made by ORIGEN2 code based on data supplied by HANF (Mitchell and Nelson 1988). Canister is filled to 85% of capacity and contains 1,650 kg of HLW glass made from neutralized current acid waste (NCAW). Data are shown for two cases, the nominal case and the maximum case. The maximum case is based on a 21-month cooling time from fuel reprocessing to HWVP.

[b]Years after vitrification.

Table 3.4.8. Hanford Site: chemical composition of HWVP reference HLW glass type HW39-4[a]

Waste oxide	Nominal waste wt % oxide (dry basis)	Frit wt % oxide	Reference nominal glass HW39-4 wt % oxide
Ag_2O	1.0E-02	--	2.5E-03
As_2O_3	4.3E-05	--	1.1E-05
Al_2O_3	9.0E+00	--	2.3E-00
Am_2O_3	2.0E-02	--	5.0E-03
B_2O_3	1.0E-01	1.4E+01	1.1E+01
BaO	4.0E-01	--	1.0E-01
BeO	1.0E-01	--	2.5E-02
CaO	3.0E-01	1.0E+00	8.3E-01
CdO	3.0E+00	--	7.6E-01
CeO_2	6.0E-01	--	1.5E-01
Cr_2O_3	5.0E-01	--	1.3E-01
Cs_2O	6.0E-01	--	1.5E-01
CuO	6.0E-01	--	1.5E-01
Dy_2O_3	1.0E-04	--	2.6E-05
Er_2O_3	3.1E-06	--	7.7E-07
Eu_2O_3	2.0E-02	--	5.0E-03
F	1.2E+00	--	3.0E-01
Fe_2O_3	2.8E+01	--	7.0E+00
Gd_2O_3	1.0E-02	--	2.5E-03
GeO_2	1.6E-04	--	3.9E-05
Ho_2O_3	5.3E-06	--	1.3E-06
I	4.5E-06	--	1.1E-06
In_2O_3	1.3E-03	--	3.3E-04
K_2O	5.0E-02	--	1.3E-02
La_2O_3	2.9E+00	--	7.3E-01
Li_2O	--	5.0E+00	3.7E+00
MgO	2.0E-01	1.0E+00	8.0E-01
MnO_2	6.0E-01	--	1.5E-01
MoO_3	1.2E+00	--	3.0E-01
Na_2O	1.8E+01	9.0E+00	1.1E+01
Nb_2O_3	1.0E-02	--	2.5E-03
Nd_2O_3	1.6E+00	--	3.9E-01
NiO	2.3E+00	--	5.8E-01
NpO_2	1.0E-01	--	2.5E-02
P_2O_5	8.7E-01	--	2.2E-01

Table 3.4.8 (continued)

Waste oxide	Nominal waste wt % oxide (dry basis)	Frit wt % oxide	Reference nominal glass HW39-4 wt % oxide
PbO_2	5.0E-02	--	1.3E-02
PdO	2.0E-01	--	5.0E-02
Pm_2O_3	1.0E-01	--	2.5E-02
Pr_2O_3	4.0E-01	--	1.0E-01
PuO_2	2.0E-02	--	5.0E-03
Rb_2O_3	2.0E-01	--	5.0E-02
Rh_2O_3	2.0E-01	--	5.0E-02
Ru_2O_3	6.0E-01	--	1.5E-01
SO_3	6.6E-01	--	1.6E-01
Sb_2O_3	5.9E-03	--	1.5E-03
SeO_2	3.0E-02	--	7.6E-03
SiO_2	4.0E+00	7.0E+01	5.4E+01
Sm_2O_3	2.0E-01	--	5.0E-02
SnO	4.0E-02	--	1.0E-02
SrO	4.0E-01	--	1.0E-01
Ta_2O_5	3.0E-02	--	7.6E-03
Tb_2O_3	2.3E-04	--	5.6E-05
Tc_2O_7	4.0E-01	--	1.0E-01
TeO_2	1.0E-01	--	2.5E-02
TiO_2	1.0E-02	--	2.5E-03
Tm_2O_3	1.7E-10	--	4.2E-11
U_3O_8	4.7E+00	--	1.2E+00
Y_2O_3	2.0E-01	--	5.0E-02
ZrO_2	1.5E+01	--	3.8E+00

[a]Data are from Mitchell and Nelson and represent the chemical composition of borosilicate glass-type HW39-4 for neutralized current acid waste. Glass composition is based on 25 wt % waste oxides and 75 wt % glass frit.

3.5 IDAHO NATIONAL ENGINEERING LABORATORY (INEL)

3.5.1 Introduction

The Idaho Chemical Processing Plant (ICPP), which is located at INEL, has as its primary purpose the reprocessing of DOE and naval fuels for the recovery of uranium and other elements. Fuels routinely processed include aluminum-, stainless steel-, and zirconium-based fuels, the latter comprising the majority of fuel. The acidic high-level liquid waste resulting from dissolution and organic solvent extraction of these fuels is temporarily stored in stainless steel tanks and is subsequently dehydrated and solidified by a fluidized-bed calcining process. The granular calcine resulting from this process is stored retrievably on-site in stainless steel bins located in below-ground concrete vaults. Thus far, about 5.6 million gallons of liquid HLW have been calcined, resulting in an average volume reduction of about 7:1.

3.5.2 Types of HLW Produced

Various alternatives for the immobilization of HLW are being studied at INEL; both glass and hot-isostatic-pressed glass-ceramic (also referred to as "ceramic-based") compositions are being considered for possible use as final waste forms. A final decision on the waste form has not yet been made. Volumetric considerations favor the glass-ceramic form, which has only about 40% of the volume of the glass form for an equivalent amount of waste (Staples, Knecht, and Berreth 1986). The terminology "glass-ceramic" is used here rather than "ceramic" because the solid is a mixture of an amorphous glass phase and a crystalline ceramic phase.

3.5.3 Physical Description

Regardless of whether glass-ceramic or vitrified HLW is produced, it appears likely that the waste will be contained in canisters similar in dimensions to those planned for use at WVDP, SRS, and HANF; that is, 61 cm in diameter by 300 cm high. If glass-ceramic blocks are to be placed in such canisters, the canister could be designed with a wide-mouth opening and several blocks could be placed in each canister. Table 3.5.1 gives estimated physical characteristics of the canister and its contents based on the assumption that the glass-ceramic form of HLW is used. This physical description should be considered preliminary at this time.

3.5.4 Inventory and Production Schedule

Under the Defense Waste Management Plan, construction of a HLW immobilization facility will be started at INEL in 2002. The immobilized HLW production schedule shown in this report is based on the assumption that immobilization starts in 2012 (Berreth 1990). If the glass-ceramic form is chosen, the maximum rate of immobilized HLW production would be approximately 1,000 canisters per year, as shown in Table 3.5.2. This is based on an estimated 650 canisters per year required to handle the waste from anticipated annual fuel reprocessing operations, plus an additional 350 canisters per year to work off the backlog of stored calcine from past operations (Knecht 1986a). Other assumptions are that (1) there is no pretreatment of the calcine to remove inerts prior to immobilization, (2) the usable waste volume per canister is 0.57 m^3, (3) the waste loading (calcine in glass-ceramic) is 70 wt%, (4) the density of the glass-ceramic is 3,200 kg/m^3, and (5) during the first three years of operation, the immobilization plant runs at a reduced rate (500 to 700 canisters/year) sufficient to keep up with current production (Berreth 1987 and 1989). Final decisions on processing options for INEL have not yet been made, so the schedule and canistered waste characteristics presented here should be considered as preliminary (Berreth and Knecht 1986, Berreth 1987a). As mentioned previously, the 1990 IDB submittal from INEL (Berreth 1990) shows an estimated production of 19,000 canisters during the period 2021-2039.

3.5.5 Maximum Radioactivity Per Canister

Table 3.5.3 shows the estimated radionuclide composition of a canister based on the assumptions that the calcined HLW is converted to glass-ceramic form and that each canister contains 1,825 kg of glass-ceramic, which is the equivalent of 1,277 kg of calcine (Berreth 1986c). The radionuclide composition of the calcine for these calculations represents 3-year-old calcine and was taken from an INEL report (IDO 1982). In practice, the feed to the immobilization plant could include calcine with an age greater than three years, and the activity per canister would accordingly be lower. The composition given is intended to represent the maximum activity per canister. Because of security restrictions, no radionuclide composition data have been officially released by INEL; therefore, the estimates presented in Table 3.5.3 should be considered preliminary.

Neutron production rates per canister have not been calculated for INEL HLW glass/ceramic because the composition of the immobilized waste form has not yet been completely defined.

3.5.6 Radioactivity and Thermal Power vs Time

Table 3.5.4 shows the calculated radioactivity and thermal power per canister as functions of decay time ranging from 0 to 10^6 years. These calculations were made by the ORIGEN2 program using the radionuclide composition shown in Table 3.5.3 and hence carry the same caveats as those mentioned for Table 3.5.3; however, they

are intended to represent the maximum radioactivity per canister that could be encountered.

Appendix 3A presents detailed decay tables showing the contributions of individual radionuclides to total curies and watts per canister for decay times ranging from 0 to 10^6 years; these are for the maximum activity canister only.

Table 3.5.5 shows estimated year-by-year projections of cumulative average radioactivity and thermal power per canister. These were calculated from projected estimates of total curies and watts for calcined waste from INEL's FY 1990 Integrated Data Base submittal (Berreth 1990). The 1990 estimates of waste generated, total radioactivity, and total thermal power are lower than the 1989 estimates because of projected decreases in reprocessing rates. The cumulative averages shown in Table 3.5.5 were calculated from the IDB submittal based on the assumption that the canisters produced in a given year would have the same radionuclide composition as the average calcine in storage that year. Obviously, this may not correspond to the actual scheduling of feeds to the immobilization plant; however, the average values shown should be more useful than maximum values for estimation of total repository radioactivity and thermal loads.

3.5.7 Chemical Composition

Table 3.5.6 shows the compositions of typical calcines produced at INEL by the calcination of high-level liquid wastes. These calcines can be densified and immobilized by hot isostatic pressing with added components that convert sodium and boron oxides in the waste to an interstitial glass phase and stabilize the ceramic-based product. The chemical composition of the final ceramic-based product has not been completely decided and will depend on the type of calcine fed to the plant. Table 3.5.7 gives approximate chemical compositions of five ceramic-based products that have been produced during process development studies. These studies are continuing, and it should be understood that the composition of the actual immobilized high-level waste produced at INEL may not necessarily be typified by the developmental results shown here (Baker 1986: Staples, Knecht, and Berreth 1986).

3.5.8 Assessment of Data

Because the strategy and processing for disposal of INEL high-level waste will not be decided until sometime in the 1990s, estimates of canister production and radioactivity given here are preliminary. These estimates also are based on incomplete information on immobilized waste radionuclide compositions. The data contained in the most recent Integrated Data Base submittal (Berreth 1990) give projections of total curies and watts for liquid waste and calcined waste inventories from 1989 to 2035. However, these data cannot be used to estimate the maximum radioactivity per canister, since the cumulative average radioactivity gives no indication of the maximum radioactivity in a given year of production. Because of security limitations, no data were furnished by INEL on the radionuclide compositions of glass or ceramic immobilized wastes made from interim wastes. Our estimates were based on an assumed radionuclide composition of 3-year aged calcine from a 1982 report. Repository calculations will require information on the maximum expected radioactivity and thermal power per canister and on the decay of these quantities as a function of time.

3.5.9 References for Section 3.5

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Berreth and Knecht 1986. J. R. Berreth and D. A. Knecht, "Potential Process Options to Minimize Immobilized HLW Volume at the Idaho Chemical Processing Plant," WINCO-M-10079, presented at the ANS International Symposium on Waste Management, Niagara Falls, September 14-18, 1986.

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Berreth 1990. Letter from J. R. Berreth, INEL, to M. J. Bonkowski, DOE, March 28, 1990, with revisions by D. Knecht, June 15, 1990.

Knecht 1986a. Letter from D. A. Knecht, INEL, to R. Salmon, ORNL, DAK-14-86, dated April 11, 1986.

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IDO 1982. *Environmental Evaluation of Alternatives for Long-Term Management of Defense High-Level Radioactive Wastes at the ICPP*, Report IDO-10105, U. S. DOE Idaho Operations Office, September 1982.

Staples, Knecht, and Berreth 1986. B. A. Staples, D. A. Knecht, and J. R. Berreth, *Technology for the Long-Term Management of Defense HLW at the ICPP*, Westinghouse-Idaho Report WINCO-1038, June 1986.

Table 3.5.1. Idaho National Engineering Laboratory: high-level waste form and canister characteristics[a]

Waste form	Glass-ceramic blocks in sealed canister
Canister material	Stainless steel type 304L
Glass-ceramic density, g/cm^3	3.2
Weights per canister, kg	
Empty canister	500
Glass-ceramic	1,825
Total loaded weight	2,325
Waste loading in glass-ceramic, wt %	70[b]
Glass-ceramic volume per canister, m^3	0.57[b]
Canister dimensions, cm	
Outside diameter	61
Height overall	300
Wall thickness	0.95
Radionuclide content, Ci/canister	108,900[c]
Heat generation rate, W/canister	339[c]

[a]Based on the following assumptions:

1. Glass-ceramic form is chosen for HLW immobilization. The term "glass-ceramic" denotes an immobilized waste form consisting of a glass phase dispersed in a ceramic phase.
2. Canister load is equivalent to 1,277 kg calcine.
3. Calcine is 3 years old at time of immobilization.
4. Canister is similar in dimensions to DWPF canister.
5. Radionuclide content of calcine is as shown in IDO-1982 (see Table 3.5.3).

[b]Reference: Berreth 1987.

[c]At time of immobilization. Quantities shown are estimated maximum values; average values are expected to be considerably less.

Table 3.5.2. Idaho National Engineering Laboratory: estimated production schedule of canisters of HLW glass-ceramic[a]

Calendar year	Number of canisters produced during year	Cumulative number of canisters produced
2012	500	500
2013	600	1,100
2014	700	1,800
2015	1,000	2,800
2016	1,000	3,800
2017	1,000	4,800
2018	1,000	5,800
2019	1,000	6,800
2020	1,000	7,800

[a]This assumes that a glass-ceramic form (density 3.2 g/cm^3) is selected for HLW disposal and that each canister contains 1,277 kg of calcine (1,825 kg of glass-ceramic). Waste loading is 70 wt %. Canister production will continue after 2020 but is not shown. Source: Berreth 1990.

Table 3.5.3. Idaho National Engineering Laboratory: radionuclide content per HLW canister[a]

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal power (W/canister)
Se-79	0.1173E+01	0.8173E-01	0.2035E-04
Rb-87	0.5252E+02	0.4597E-05	0.3843E-08
Sr-90	0.1217E+03	0.1660E+05	0.1927E+02
Y-90	0.3051E-01	0.1660E+05	0.9204E+02
Zr-93	0.1575E+03	0.3959E+00	0.4600E-04
Nb-93m	0.3387E-03	0.9577E-01	0.1697E-04
Tc-99	0.1582E+03	0.2682E+01	0.1346E-02
Ru-106	0.3701E+00	0.1239E+04	0.7365E-01
Rh-106	0.3479E-06	0.1239E+04	0.1188E+02
Pd-107	0.4965E+01	0.2554E-02	0.1514E-06
Sn-126	0.1440E+01	0.4086E-01	0.5097E-04
Sb-126m	0.5201E-09	0.4086E-01	0.5203E-03
Sb-126	0.4887E-06	0.4086E-01	0.7552E-03
Cs-134	0.3256E+01	0.4214E+04	0.4290E+02
Cs-135	0.8316E+02	0.9577E-01	0.3197E-04
Cs-137	0.1908E+03	0.1660E+05	0.1837E+02
Ba-137m	0.2848E-04	0.1532E+05	0.6017E+02
Ce-144	0.3282E+01	0.1047E+05	0.6947E+01
Pr-144	0.1386E-03	0.1047E+05	0.7700E+02
Pm-147	0.1653E+02	0.1532E+05	0.5499E+01
Sm-151	0.8250E+01	0.2171E+03	0.2546E-01
Eu-154	0.8513E+00	0.2299E+03	0.2056E+01
U-233	0.1583E-06	0.1532E-08	0.4456E-10
U-234	0.8785E-04	0.5491E-06	0.1582E-07
U-235	0.1063E+01	0.2299E-05	0.6020E-07
U-236	0.1973E+00	0.1277E-04	0.3459E-06
U-237	0.7507E-13	0.6130E-08	0.1158E-10
U-238	0.3797E-04	0.1277E-10	0.3235E-12
Np-237	0.8693E-01	0.6130E-04	0.1874E-05
Pu-238	0.5221E+01	0.8939E+02	0.2963E+01
Pu-239	0.1437E+02	0.8939E+00	0.2754E-01
Pu-240	0.3642E+01	0.8300E+00	0.2585E-01
Pu-241	0.1983E+01	0.2043E+03	0.6336E-02
Pu-242	0.6018E+00	0.2299E-02	0.6788E-04
Am-241	0.3385E+00	0.1162E+01	0.3861E-01
Am-243	0.5315E-01	0.1060E-01	0.3407E-03
Cm-242	0.2510E-03	0.8300E+00	0.3059E-01
Cm-244	0.8201E-02	0.6640E+00	0.2322E-01
Total	0.8315E+03	0.1088E+06	0.3393E+03

[a]Quantities are at time of filling canister and are based on 3-year-old calcine immobilized in glass-ceramic with a load of 1,277 kg of calcine per canister (1,825 kg of glass-ceramic per canister). Based on IDO 1982 and Berreth 1986c.

Table 3.5.4. Idaho National Engineering Laboratory: calculated radioactivity and thermal power per HLW canister[a]

Decay time after immobilization (years)	Total radioactivity per canister (Ci)	Total thermal power per canister (W)
0	108,900	339
1	89,400	267
2	78,600	230
5	64,100	185
10	53,600	157
15	46,900	138
20	41,500	123
30	32,800	97
50	20,500	61
100	6,430	20
200	680	2.6
300	98	0.67
350	48	0.45
500	16	0.24
1,000	7.2	0.11
1,050	7.0	0.10
2,000	5.6	0.06
5,000	5.0	0.04
10,000	4.6	0.033
20,000	4.2	0.023
50,000	3.6	0.012
100,000	3.1	0.008
500,000	1.4	0.003
1,000,000	0.71	0.001

[a]Results of ORIGEN2 calculations based on glass-ceramic form, assuming 1,277 kg of calcine per canister (1,825 kg of glass-ceramic per canister), with the initial radionuclide composition shown in Table 3.5.3. The years 350 and 1,050 were included at the request of the project sponsor, because of the possible need for calculations at emplacement age plus 300 or 1,000 years after repository closure.

Table 3.5.5. Idaho National Engineering Laboratory: estimated cumulative average radioactivity and thermal power per canister of HLW glass-ceramic[a]

End of calendar year	Cumulative number of canisters produced	Cumulative radioactivity		Cumulative thermal power	
		Total (10^6 Ci)	Per canister (Ci)	Total (10^3 W)	Per canister (W)
2012	500	8	16,000	23	46
2013	1,100	18	16,300	52	47
2014	1,800	29	16,100	84	47
2015	2,800	45	16,200	132	47
2016	3,800	62	16,400	180	47
2017	4,800	79	16,400	230	48
2018	5,800	96	16,500	280	48
2019	6,800	110	16,200	320	47
2020	7,800	129	16,300	370	47

[a]Calculated from estimates given in Berreth 1990, using the simplifying assumption that the glass-ceramic produced in a given year has the same composition as the average calcine in storage that year. Each canister contains 1,277 kg of calcine, which is equivalent to 0.91 m^3 of calcine in bulk form. The term "glass-ceramic" denotes a ceramic-based immobilized waste. Cumulative radioactivity per canister means cumulative immobilized radioactivity divided by cumulative number of canisters produced. These estimates are based on incomplete information and simplifying assumptions and are therefore subject to change.

Table 3.5.6. Composition of typical HLW calcines produced at INEL[a]

Component	Type of calcine and composition, wt %			
	Alumina	Zirconia	Fluorinel	Zirconia-sodium
Al_2O_3	82–95	13–17	6	12–14
Na_2O	1–3	–	--	0–5
ZrO_2	–	21–27	23	20–26
CaF_2	–	50–56	56	48–53
CaO	–	2–4	4	2–4
Nitrate	5–9	0.5–2	0.5–2	0.5–4
B_2O_3	0.5–2	3–4	4	3–4
CdO	–	–	6	–
Fission products and actinides	≤1	≤1	≤1	≤1

[a]Sources: Staples, Knecht, and Berreth, 1986; Knecht 1991.

Table 3.5.7. Compositions of typical ceramic-based waste forms developed for immobilization of INEL calcined HLW[a]

Formulation number	SiO_2 (wt %)	Na_2O (wt %)	Li_2O (wt %)	B_2O_3 (wt %)	Waste (wt %)
12	8.6	1.1	0.5	2.6	87.2
11	16.0	0.0	0.0	1.4	82.6
17	30.3	0.0	0.0	2.3	67.5
6	28.6	2.1	0.9	3.5	64.9
1	14.2	2.6	1.2	1.7	80.3

[a]Source: Baker 1986.

APPENDIX 1A. ORIGEN2 OVERVIEW

APPENDIX 1A. ORIGEN2 OVERVIEW

ORIGEN2 is a computer code developed to model the composition and characteristics of various kinds of spent nuclear fuels as a function of burnup and age (Croff 1983). To do this, the code performs two major computational functions, isotope generation and isotope depletion, both within the core of an operating reactor and after shutdown. The name derives from "Oak Ridge Isotope Generation and Depletion Code." The original version was called ORIGEN and a later, improved version is called ORIGEN2. There is also another improved version called ORIGEN-S. All three versions perform the two basic functions cited above and described below.

Generation. The generation of individual nuclides resulting from neutron-induced fission or from neutron capture reactions or other transmutation reactions.

Depletion. The depletion (and concurrent buildup) of nuclides resulting from natural decay processes.

Both of these functions can deal with the 1,400 or so potentially relevant nuclides. Many of these nuclides have very short half-lives and exist only within the reactor core or as transient intermediates in a decay chain. Both generation and depletion can be very complex. Neutron-induced fission and transmutation are highly dependent on both neutron flux and neutron energy. Neutron flux itself depends on the number of induced fissions. Neutron energy depends on the moderators that are present. The cross sections (propensity for neutron interaction) are highly sensitive to neutron energy. Decay processes are governed by invariant constants: the half-lives and branching ratios. In a real-time situation, some transmutation may occur while decay is also taking place.

Thus, during generation there is a competition between transmutation and decay which is governed by the relative magnitudes of the cross sections and the decay constants, with the former highly sensitive to the neutron moderation within the particular system being modeled. This is handled mathematically by calculating a set of effective cross sections which are applicable to the reactor scenario being modeled.

This is accomplished by first computing a set of effective one-group cross sections; i.e., a weighted-average value for each nuclide that is appropriate for the moderated neutron energies in that particular reactor model. Once this effective cross section library has been developed, it can be used for variations on that particular reactor model. For example, ORIGEN2 cross section libraries have been developed specifically for LWRs. The latter requires different models and libraries for PWRs and BWRs, and also for standard burnup and high burnup fuel designs. For a given LWR case, the appropriate model must be used. These cross sections have recently been recalculated (Ludwig 1989) for these conditions:

PWR standard burnup: 33,000 MWd/MTIHM;
PWR extended burnup: 50,000 MWd/MTIHM;
BWR standard burnup: 27,500 MWd/MTIHM; and
BWR extended burnup: 40,000 MWd/MTIHM.

For the above cross section libraries, initial enrichments and cycle conditions can be varied. Also, variations in structural materials (both quantities and compositions) may be used if appropriate. These factors have been examined in a series of sensitivity studies (Welch 1992). It was clearly shown that enrichment is a major factor in these calculations, especially for the actinides (and their derivative properties, such as neutron emission strength). For this reason, enrichments were handled in a very definitive manner, with enrichment tailored to burnup, and with ranges on either side to allow for normal variations. Cycle conditions were modeled after average conditions derived from utility data.

The neutron fluxes and energies, and the effective cross sections, have been developed for the reactor core region since this is where the fuel resides, along with the fuel cladding and some of the assembly hardware. However, much of the hardware is outside the core region, where the flux falls off very rapidly with distance, and where the moderation is also quite different. These latter effects have a profound influence on the quantities of activation products formed within these hardware components.

The major steps required in ORIGEN2 modeling and computation are shown on the schematic drawing. The center column, boxes 1, 2, 3, and 4, are the major steps involved in running the ORIGEN2 code. Box 2 requires complex, multidimensional computations which apply to the generic reactor model defined in box 1; once done, the resulting library is used for all specific cases for that generic reactor model. Boxes A and B are independent data libraries that ORIGEN2 utilizes. Box X is the required input data to conduct a run for a specific case; this includes initial enrichment and cycle specifications, and also the desired output data and format. In the past, the near-core region was modeled by an approximation method to estimate hardware activation. Now, with increased interest in activated metal, this aspect was reexamined. Both calculations and experimental measurements at PNL (Luksic 1989) provided new factors, relative to the core region, for the top, plenum, and bottom zones. These new factors were utilized.

The basic data output from ORIGEN2 is in terms of the gram atoms of each nuclide, at specified times. From this, the derived quantities and radiological properties are calculated, including thermal output, alpha activity, beta and gamma activities, neutron production from spontaneous fission and from alpha-induced reaction, and the photon energy spectra. These quantities can be provided for

individual nuclides or elements or for all nuclides or elements within one or more of the three major categories: fission products, activation products, and actinides. Each group also includes decay daughters. The fission products derive from nuclear fission of fissionable isotopes. The activation products derive from neutron activation (transmutation) of structural materials and components. The actinides derive from neutron capture (often multiple or sequential neutron captures) of the initially-present heavy metal isotopes (mainly U-238 and U-235 for LWRs).

References

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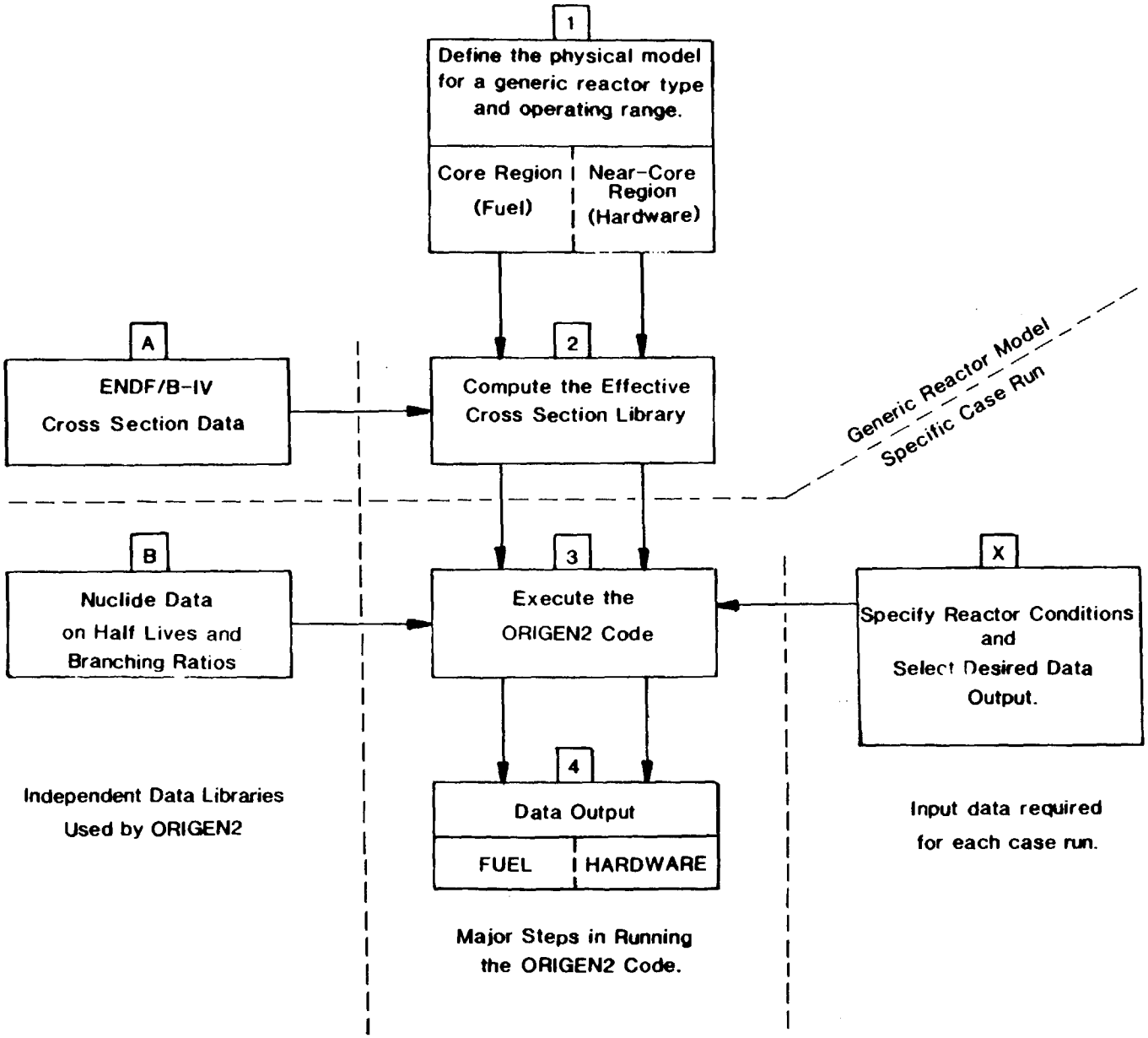
ORNL DWG 87-982

APPLICATION OF THE ORIGEN2 CODE



APPENDIX 1B. ORIGEN2 LIBRARY DATA

Table 1B.1. Nuclide half-lives, decay modes, and recoverable heat (Q) from the ORIGEN2 decay library used to calculate radiological data

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
H-3	3.897E+08 s	Beta	3.367E-05	He-3
H-4	1.000E-03 s	Beta	0.000E+00	He-4
He-6	0.808 s	Beta	9.295E-03	Li-6
Li-8	0.842 s	Beta	3.729E-02	Be-8
Be-8	2.000E-06 s	Alpha	5.632E-04	He-4
Be-10	5.049E+13 s	Beta	1.200E-03	B-10
Be-11	13.6 s	Beta	6.823E-02	B-11
B-12	2.030E-02 s	Beta	7.926E-02	C-12
C-14	1.808E+11 s	Beta	2.933E-04	N-14
C-15	2.45 s	Beta	1.702E-02	N-15
N-13	598. s	POS or EC	8.957E-03	C-13
N-16	7.12 s	Beta	4.334E-02	O-16
O-19	29.0 s	Beta	2.857E-02	F-19
F-20	11.4 s	Beta	4.167E-02	Ne-20
Ne-23	37.2 s	Beta	1.226E-02	Na-23
Na-22	8.211E+07 s	POS or EC	1.415E-02	Ne-22
Na-24	5.400E+04 s	Beta	2.771E-02	Mg-24
Na-24m	1.990E-02 s	IT	2.798E-03	Na-24
Na-25	59.6 s	Beta	1.148E-02	Mg-25
Mg-27	568. s	Beta	9.443E-03	Al-27
Mg-28	7.528E+04 s	Beta	9.088E-03	Al-28
Al-28	134. s	Beta	1.794E-02	Si-28
Al-29	391. s	Beta	1.394E-02	Si-29
Al-30	3.68 s	Beta	3.393E-02	Si-30
Si-31	9.438E+03 s	Beta	3.536E-03	P-31
Si-32	650. y	Beta	1.245E-03	P-32
P-32	14.3 d	Beta	1.014E-02	S-32
P-33	25.0 d	Beta	1.470E-03	S-33
P-34	12.4 s	Beta	3.023E-02	S-34
S-35	88.0 d	Beta	9.924E-04	Cl-35
S-37	5.06 min	Beta	2.845E-02	Cl-37
Cl-36	9.499E+12 s	POS or EC (1.90%)	1.475E-03	S-36
		Beta (98.1%)		Ar-36
Cl-38	2.233E+03 s	Beta	1.788E-02	Ar-38
Cl-38m	0.716 s	IT	3.980E-03	Cl-38
Ar-37	3.026E+06 s	POS or EC	1.245E-05	Cl-37
Ar-39	269. y	Beta	3.349E-03	K-39
Ar-41	6.577E+03 s	Beta	1.036E-02	K-41
Ar-42	33.0 y	Beta	3.557E-03	K-42

[a]POS – positron; EC – electron capture; IT – internal transition; * – excited state; n – neutron; and SF – spontaneous fission.

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
K-40	4.039E+16 s	POS or EC (10.7%)	3.619E-03	Ar-40
		Beta (89.3%)		Ca-40
K-42	4.450E+04 s	Beta	1.011E-02	Ca-42
K-43	8.136E+04 s	Beta	7.582E-03	Ca-43
K-44	22.0 min	Beta	3.083E-02	Ca-44
Ca-41	81.0 10^3 y	POS or EC	1.601E-05	K-41
Ca-45	1.408E+07 s	Beta	4.576E-04	Sc-45
Ca-47	3.919E+05 s	Beta	8.347E-03	Sc-47
Ca-49	8.80 min	Beta	3.118E-02	Sc-49
Sc-46	7.240E+06 s	Beta	1.258E-02	Ti-46
Sc-46m	18.7 s	IT	8.133E-04	Sc-46
Sc-47	2.895E+05 s	Beta	1.605E-03	Ti-47
Sc-48	1.577E+05 s	Beta	2.115E-02	Ti-48
Sc-49	57.5 min	Beta	1.190E-02	Ti-49
Sc-50	103. s	Beta	2.861E-02	Ti-50
Ti-51	346. s	Beta	7.321E-03	V-51
V-49	2.851E+07 s	POS or E	2.549E-05	Ti-49
V-50	4.000E+16 y	POS or EC (70.0%)	1.103E-02	Ti-50
		Beta (30.0%)		Cr-50
V-52	225. s	Beta	1.490E-02	Cr-52
V-53	96.6 s	Beta	1.208E-02	Cr-53
V-54	55.0 s	Beta	4.328E-02	Cr-54
Cr-51	2.394E+06 s	POS or EC	2.146E-04	V-51
Cr-55	213. s	Beta	6.527E-03	Mn-55
Mn-54	2.700E+07 s	POS or EC	4.978E-03	Cr-54
Mn-56	9.283E+03 s	Beta	1.494E-02	Fe-56
Mn-57	96.6 s	Beta	6.942E-03	Fe-57
Mn-58	65.3 s	Beta	2.368E-02	Fe-58
Fe-55	2.60 y	POS or EC	3.379E-05	Mn-55
Fe-59	45.0 d	Beta	7.742E-03	Co-59
Co-58	6.115E+06 s	POS or EC	5.981E-03	Fe-58
Co-58m	3.294E+04 s	IT	1.462E-04	Co-58
Co-60	1.663E+08 s	Beta	1.542E-02	Ni-60
Co-60m	628. s	IT (99.8%)	3.736E-04	Co-60
		Beta (0.250%)		Ni-60m
Co-61	5.940E+03 s	Beta	3.237E-03	Ni-61
Co-62	90.0 s	Beta	1.897E-02	Ni-62
Co-72	0.123 s	Beta	5.086E-02	Ni-72
Co-73	0.116 s	Beta	4.520E-02	Ni-73
Co-74	0.108 s	Beta	5.654E-02	Ni-74
Co-75	8.016E-02 s	Beta	5.079E-02	Ni-75
Ni-59	80.0 10^3 y	POS or EC	3.972E-05	Co-59
Ni-63	92.0 y	Beta	1.008E-04	Cu-63
Ni-65	9.072E+03 s	Beta	7.001E-03	Cu-65
Ni-66	1.966E+05 s	Beta	3.972E-04	Cu-66
Ni-72	2.42 s	Beta	1.901E-02	Cu-72
Ni-73	0.394 s	Beta	3.186E-02	Cu-73

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Ni-74	0.648 s	Beta	2.519E-02	Cu-74
Ni-75	0.180 s	Beta	3.805E-02	Cu-75
Ni-76	0.268 s	Beta	3.125E-02	Cu-76
Ni-77	0.103 s	Beta	4.381E-02	Cu-77
Ni-78	0.138 s	Beta	3.736E-02	Cu-78
Cu-62	584. s	POS or EC	1.359E-02	Ni-62
Cu-64	4.572E+04 s	POS or EC (62.8%)	1.855E-03	Ni-64
		Beta (37.2%)		Zn-64
Cu-66	306. s	Beta	6.847E-03	Zn-66
Cu-67	2.227E+05 s	Beta	1.606E-03	Zn-67
Cu-72	6.00 s	Beta	2.781E-02	Zn-72
Cu-73	3.95 s	Beta	2.050E-02	Zn-73
Cu-74	0.573 s	Beta	3.323E-02	Zn-74
Cu-75	0.767 s	Beta	2.671E-02	Zn-75
Cu-76	0.221 s	Beta	3.930E-02	Zn-76
Cu-77	0.295 s	Beta	3.273E-02	Zn-77
Cu-78	0.121 s	Beta	4.500E-02	Zn-78
Cu-79	0.147 s	Beta	3.876E-02	Zn-79
Cu-80	9.110E-02 s	Beta	5.654E-02	Zn-80
Cu-81	7.447E-02 s	Beta	5.402E-02	Zn-81
Zn-63	38.5 min	POS or EC	1.196E-02	Cu-63
Zn-65	2.107E+07 s	POS or EC	3.500E-03	Cu-65
Zn-69	3.420E+03 s	Beta	1.902E-03	Ga-69
Zn-69m	4.954E+04 s	IT (100.%)	2.601E-03	Zn-69
		Beta (3.000E-02%)		Ga-69m
Zn-71	2.40 min	Beta	1.663E-02	Ga-71
Zn-71m	3.92 h	IT (5.000E-02%)	1.756E-02	Zn-71
		Beta (99.9%)		Ga-71m
Zn-72	1.674E+05 s	Beta	1.513E-03	Ga-72
Zn-73	23.5 s	Beta	1.455E-02	Ga-73
Zn-74	95.0 s	Beta	6.426E-03	Ga-74
Zn-75	9.00 s	Beta	1.943E-02	Ga-75
Zn-76	5.40 s	Beta	1.304E-02	Ga-76
Zn-77	1.40 s	Beta	2.506E-02	Ga-77
Zn-78	2.43 s	Beta	1.834E-02	Ga-78
Zn-79	0.382 s	Beta	3.119E-02	Ga-79
Zn-80	0.711 s	Beta	2.410E-02	Ga-80
Zn-81	0.129 s	Beta	4.307E-02	Ga-81
Zn-82	0.135 s	Beta	3.989E-02	Ga-82
Zn-83	8.386E-02 s	Beta	4.876E-02	Ga-83
Ga-70	1.266E+03 s	Beta	3.872E-03	Ge-70
Ga-72	5.076E+04 s	Beta	1.901E-02	Ge-72
Ga-72m	3.968E-02 s	IT	7.114E-04	Ga-72
Ga-73	1.757E+04 s	Beta to *	4.523E-03	Ge-73m
Ga-74	486. s	Beta	2.573E-02	Ge-74
Ga-75	114. s	Beta to * (4.00%)	8.187E-03	Ge-75m
		Beta (96.0%)		Ge-75

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Ga-76	27.1 s	Beta	2.886E-02	Ge-76
Ga-77	13.0 s	Beta to * (88.0%)	1.517E-02	Ge-77m
		Beta (12.0%)		Ge-77
Ga-78	4.90 s	Beta	2.717E-02	Ge-78
Ga-79	2.86 s	Beta + n (0.140%)	2.076E-02	Ge-78
		Beta (99.9%)		Ge-79
Ga-80	1.70 s	Beta + n (0.860%)	3.333E-02	Ge-79
		Beta (99.1%)		Ge-80
Ga-81	0.705 s	Beta	2.656E-02	Ge-81
Ga-82	0.154 s	Beta	4.499E-02	Ge-82
Ga-83	0.148 s	Beta	4.232E-02	Ge-83
Ga-84	9.887E-02 s	Beta	5.066E-02	Ge-84
Ga-85	9.197E-02 s	Beta	4.789E-02	Ge-85
Ge-71	11.8 d	POS or EC	5.335E-05	Ga-71
Ge-71m	2.190E-02 s	IT	1.162E-03	Ge-71
Ge-73m	0.530 s	IT	3.972E-04	Ge-73
Ge-75	4.968E+03 s	Beta	2.762E-03	As-75
Ge-75m	48.9 s	IT	8.240E-04	Ge-75
Ge-77	4.068E+04 s	Beta	1.027E-02	As-77
Ge-77m	54.3 s	IT (21.0%)	6.130E-03	Ge-77
		Beta (79.0%)		As-77m
Ge-78	5.220E+03 s	Beta	3.053E-03	As-78
Ge-79	43.0 s	Beta	1.271E-02	As-79
Ge-80	24.0 s	Beta	6.100E-03	As-80
Ge-81	10.1 s	Beta	1.925E-02	As-81
Ge-82	4.60 s	Beta	1.224E-02	As-82
Ge-83	1.90 s	Beta + n (0.160%)	2.988E-02	As-82
		Beta (99.8%)		As-83
Ge-84	1.20 s	Beta + n (9.60%)	2.570E-02	As-83
		Beta (90.4%)		As-84
Ge-85	0.234 s	Beta	3.598E-02	As-85
Ge-86	0.259 s	Beta	3.294E-02	As-86
Ge-87	0.125 s	Beta	4.224E-02	As-87
Ge-88	0.143 s	Beta	3.853E-02	As-88
As-76	9.475E+04 s	Beta	8.857E-03	Se-76
As-77	1.397E+05 s	Beta to * (0.248%)	1.409E-03	Se-77m
		Beta (99.8%)		Se-77
As-78	5.442E+03 s	Beta	1.563E-02	Se-78
As-79	540. s	Beta to *	5.211E-03	Se-79m
As-80	16.5 s	Beta	1.855E-02	Se-80
As-81	32.0 s	Beta	9.894E-03	Se-81
As-82	21.0 s	Beta	2.068E-02	Se-82
As-82m	13.0 s	Beta	3.381E-02	Se-82m
As-83	13.5 s	Beta to * (64.0%)	1.577E-02	Se-83m
		Beta (36.0%)		Se-83
As-84	5.80 s	Beta + n (0.130%)	3.477E-02	Se-83
		Beta (99.9%)		Se-84

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
As-85	2.03 s	Beta + n (20.0%)	2.986E-02	Se-84
		Beta (80.0%)		Se-85
As-86	0.900 s	Beta + n (3.80%)	4.035E-02	Se-85
		Beta (96.2%)		Se-86
As-87	0.300 s	Beta + n (31.0%)	3.496E-02	Se-86
		Beta (69.0%)		Se-87
As-88	0.130 s	Beta	4.687E-02	Se-88
As-89	0.129 s	Beta	4.348E-02	Se-89
As-90	9.009E-02 s	Beta	5.351E-02	Se-90
Se-75	1.035E+07 s	POS or EC	2.406E-03	As-75
Se-77m	17.5 s	IT	1.482E-03	Se-77
Se-79	2.050E+12 s	Beta	2.490E-04	Br-79
Se-79m	233. s	IT	5.632E-04	Se-79
Se-81	1.110E+03 s	Beta	3.634E-03	Br-81
Se-81m	3.438E+03 s	IT	6.106E-04	Se-81
Se-83	1.350E+03 s	Beta	1.779E-02	Br-83
Se-83m	70.0 s	Beta	1.311E-02	Br-83m
Se-84	198. s	Beta	5.566E-03	Br-84
Se-85	39.0 s	Beta	1.988E-02	Br-85
Se-85m	19.0 s	Beta	2.071E-02	Br-85m
Se-86	16.6 s	Beta to * (50.0%)	1.446E-02	Br-86m
		Beta (50.0%)		Br-86
Se-87	5.60 s	Beta + n (0.180%)	2.512E-02	Br-86
		Beta (99.8%)		Br-87
Se-88	1.50 s	Beta + n (0.500%)	2.209E-02	Br-87
		Beta (99.5%)		Br-88
Se-89	0.410 s	Beta + n (5.00%)	3.020E-02	Br-88
		Beta (95.0%)		Br-89
Se-90	0.554 s	Beta	2.721E-02	Br-90
Se-91	0.184 s	Beta	3.880E-02	Br-91
Se-92	0.248 s	Beta	3.302E-02	Br-92
Se-93	0.107 s	Beta	4.454E-02	Br-93
Br-79m	4.86 s	IT	1.245E-03	Br-79
Br-80	1.044E+03 s	POS or EC (8.26%)	4.751E-03	Se-80
		Beta (91.7%)		Kr-80
Br-80m	1.591E+04 s	IT	5.011E-04	Br-80
Br-82	1.271E+05 s	Beta	1.647E-02	Kr-82
Br-82m	368. s	IT (97.6%)	4.631E-04	Br-82
		Beta (2.40%)		Kr-82m
Br-83	8.604E+03 s	Beta to * (100.%)	1.948E-03	Kr-83m
		Beta (3.000E-02%)		Kr-83
Br-84	1.908E+03 s	Beta	1.800E-02	Kr-84
Br-84m	360. s	Beta	2.172E-02	Kr-84m
Br-85	172. s	Beta to *	6.171E-03	Kr-85m
Br-86	55.0 s	Beta	3.019E-02	Kr-86
Br-86m	4.50 s	Beta	2.817E-02	Kr-86m

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Br-87	55.8 s	Beta + n (2.30%)	2.289E-02	Kr-86
		Beta (97.7%)		Kr-87
Br-88	16.3 s	Beta + n (4.60%)	1.499E-02	Kr-87
		Beta (95.4%)		Kr-88
Br-89	4.50 s	Beta + n (8.60%)	2.844E-02	Kr-88
		Beta (91.4%)		Kr-89
Br-90	1.60 s	Beta + n (12.0%)	3.364E-02	Kr-89
		Beta (88.0%)		Kr-90
Br-91	0.600 s	Beta + n (7.00%)	3.196E-02	Kr-90
		Beta (93.0%)		Kr-91
Br-92	0.300 s	Beta + n (26.0%)	3.966E-02	Kr-91
		Beta (74.0%)		Kr-92
Br-93	0.201 s	Beta	3.891E-02	Kr-93
Br-94	0.111 s	Beta	5.013E-02	Kr-94
Br-95	0.117 s	Beta	4.444E-02	Kr-95
Br-96	8.379E-02 s	Beta	5.554E-02	Kr-96
Kr-79	34.9 h	POS or EC	1.669E-03	Br-79
Kr-79m	55.0 s	IT	7.529E-04	Kr-79
Kr-81	6.623E+12 s	POS or EC	1.233E-04	Br-81
Kr-81m	13.3 s	IT	1.126E-03	Kr-81
Kr-83m	6.588E+03 s	IT	2.416E-04	Kr-83
Kr-85	3.383E+08 s	Beta	1.498E-03	Rb-85
Kr-85m	1.613E+04 s	IT (21.1%)	2.449E-03	Kr-85
		Beta (78.9%)		Rb-85m
Kr-87	4.578E+03 s	Beta	1.257E-02	Rb-87
Kr-88	1.022E+04 s	Beta	1.375E-02	Rb-88
Kr-89	190. s	Beta	1.896E-02	Rb-89
Kr-90	32.3 s	Beta to * (12.2%)	1.536E-02	Rb-90m
		Beta (87.8%)		Rb-90
Kr-91	8.70 s	Beta	1.957E-02	Rb-91
Kr-92	1.84 s	Beta + n (4.000E-02%)	1.890E-02	Rb-91
		Beta (100.%)		Rb-92
Kr-93	1.27 s	Beta + n (3.20%)	2.844E-02	Rb-92
		Beta (96.8%)		Rb-93
Kr-94	0.210 s	Beta + n (4.40%)	2.293E-02	Rb-93
		Beta (95.6%)		Rb-94
Kr-95	0.500 s	Beta	3.491E-02	Rb-95
Kr-96	0.440 s	Beta	2.877E-02	Rb-96
Kr-97	0.148 s	Beta	4.147E-02	Rb-97
Kr-98	0.224 s	Beta	3.383E-02	Rb-98
Rb-86	1.612E+06 s	Beta	4.519E-03	Sr-86
Rb-86m	61.1 s	IT	3.320E-03	Rb-86
Rb-87	1.482E+18 s	Beta	8.359E-04	Sr-87
Rb-88	1.068E+03 s	Beta	1.591E-02	Sr-88
Rb-89	912. s	Beta	1.837E-02	Sr-89
Rb-90	153. s	Beta	2.384E-02	Sr-90

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Rb-90m	258. s	IT (4.30%)	2.651E-02	Rb-90
		Beta (95.7%)		Sr-90m
Rb-91	58.2 s	Beta	2.411E-02	Sr-91
Rb-92	4.48 s	Beta + n (1.200E-02%)	2.214E-02	Sr-91
		Beta (100.%)		Sr-92
Rb-93	5.80 s	Beta + n (1.62%)	2.040E-02	Sr-92
		Beta (98.4%)		Sr-93
Rb-94	2.69 s	Beta + n (11.1%)	2.807E-02	Sr-93
		Beta (88.9%)		Sr-94
Rb-95	0.360 s	Beta + n (7.10%)	2.681E-02	Sr-94
		Beta (92.9%)		Sr-95
Rb-96	0.207 s	Beta + n (12.7%)	3.658E-02	Sr-95
		Beta (87.3%)		Sr-96
Rb-97	0.170 s	Beta + n (21.0%)	3.101E-02	Sr-96
		Beta (79.0%)		Sr-97
Rb-98	0.140 s	Beta + n (26.0%)	4.034E-02	Sr-97
		Beta (74.0%)		Sr-98
Rb-99	7.600E-02 s	Beta + n (37.0%)	3.570E-02	Sr-98
		Beta (63.0%)		Sr-99
Rb-100	0.101 s	Beta	5.015E-02	Sr-100
Rb-101	0.113 s	Beta	4.372E-02	Sr-101
Sr-85	5.602E+06 s	POS or EC	3.122E-03	Rb-85
Sr-85m	70.0 min	POS or EC (14.0%)	1.358E-03	Rb-85m
		IT (86.0%)		Sr-85
Sr-87m	1.010E+04 s	POS or EC (0.300%)	2.294E-03	Rb-87m
		IT (99.7%)		Sr-87
Sr-89	4.363E+06 s	Beta	3.457E-03	Y-89
Sr-90	9.190E+08 s	Beta	1.161E-03	Y-90
Sr-91	3.420E+04 s	Beta to * (58.0%)	8.015E-03	Y-91m
		Beta (42.0%)		Y-91
Sr-92	9.756E+03 s	Beta	9.100E-03	Y-92
Sr-93	450. s	Beta	1.515E-02	Y-93
Sr-94	75.6 s	Beta	1.252E-02	Y-94
Sr-95	26.0 s	Beta	1.957E-02	Y-95
Sr-96	4.00 s	Beta	1.465E-02	Y-96
Sr-97	0.200 s	Beta + n (9.500E-02%)	2.482E-02	Y-96
		Beta (99.9%)		Y-97
Sr-98	0.850 s	Beta + n (0.500%)	1.889E-02	Y-97
		Beta (99.5%)		Y-98
Sr-99	0.560 s	Beta	3.085E-02	Y-99
Sr-100	1.05 s	Beta	2.350E-02	Y-100
Sr-101	0.252 s	Beta	3.613E-02	Y-101
Sr-102	0.415 s	Beta	2.893E-02	Y-102
Sr-103	0.139 s	Beta	4.163E-02	Y-103
Sr-104	0.192 s	Beta	3.540E-02	Y-104
Y-89m	16.1 s	IT	5.435E-03	Y-89
Y-90	2.304E+05 s	Beta	5.543E-03	Zr-90

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Y-90m	1.116E+04 s	IT (99.6%)	4.049E-03	Y-90
		Beta (0.400%)		Zr-90m
Y-91	5.055E+06 s	Beta	3.592E-03	Zr-91
Y-91m	2.983E+03 s	IT	3.306E-03	Y-91
Y-92	1.274E+04 s	Beta	1.007E-02	Zr-92
Y-93	3.636E+04 s	Beta	7.481E-03	Zr-93
Y-94	1.146E+03 s	Beta	1.666E-02	Zr-94
Y-95	630. s	Beta	1.324E-02	Zr-95
Y-96	138. s	Beta	2.294E-02	Zr-96
Y-97	1.11 s	Beta + n (1.60%)	1.836E-02	Zr-96
		Beta (98.4%)		Zr-97
Y-98	0.300 s	Beta + n (0.480%)	2.838E-02	Zr-97
		Beta (99.5%)		Zr-98
Y-99	0.800 s	Beta + n (3.80%)	2.216E-02	Zr-98
		Beta (96.2%)		Zr-99
Y-100	0.756 s	Beta	3.454E-02	Zr-100
Y-101	0.976 s	Beta	2.735E-02	Zr-101
Y-102	0.273 s	Beta	3.992E-02	Zr-102
Y-103	0.366 s	Beta	3.284E-02	Zr-103
Y-104	0.144 s	Beta	4.521E-02	Zr-104
Y-105	0.174 s	Beta	3.940E-02	Zr-105
Y-106	9.292E-02 s	Beta	5.015E-02	Zr-106
Y-107	0.105 s	Beta	4.432E-02	Zr-107
Zr-89	2.824E+05 s	POS or EC (0.161%)	7.481E-03	Y-89
		POS or EC to * (99.8%)		Y-89m
Zr-90m	0.830 s	IT	1.372E-02	Zr-90
Zr-93	4.828E+13 s	Beta to * (95.0%)	1.162E-04	Nb-93m
		Beta (5.00%)		Nb-93
Zr-95	5.528E+06 s	Beta to * (0.700%)	5.066E-03	Nb-95m
		Beta (99.3%)		Nb-95
Zr-97	6.084E+04 s	Beta to * (94.6%)	5.211E-03	Nb-97m
		Beta (5.37%)		Nb-97
Zr-98	31.0 s	Beta	5.353E-03	Nb-98
Zr-99	2.40 s	Beta	1.431E-02	Nb-99
Zr-100	7.10 s	Beta to * (50.0%)	8.116E-03	Nb-100m
		Beta (50.0%)		Nb-100
Zr-101	3.30 s	Beta	1.851E-02	Nb-101
Zr-102	28.6 s	Beta	1.286E-02	Nb-102
Zr-103	1.77 s	Beta	2.452E-02	Nb-103
Zr-104	3.78 s	Beta	1.765E-02	Nb-104
Zr-105	0.559 s	Beta	2.970E-02	Nb-105
Zr-106	0.980 s	Beta	2.348E-02	Nb-106
Zr-107	0.249 s	Beta	3.556E-02	Nb-107
Zr-108	0.408 s	Beta	2.879E-02	Nb-108
Zr-109	0.139 s	Beta	4.067E-02	Nb-109
Nb-91	1.000E+04 y	POS or EC	1.020E-04	Zr-91
Nb-92	10.2 d	POS or EC	8.951E-03	Zr-92

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Nb-93m	4.292E+08 s	IT	1.772E-04	Nb-93
Nb-94	6.406E+11 s	Beta	1.019E-02	Mo-94
Nb-94m	376. s	IT (99.5%)	2.792E-04	Nb-94
		Beta (0.480%)		Mo-94m
Nb-95	3.037E+06 s	Beta	4.797E-03	Mo-95
Nb-95m	3.118E+05 s	IT	1.390E-03	Nb-95
Nb-96	8.406E+04 s	Beta	1.663E-02	Mo-96
Nb-97	4.326E+03 s	Beta	6.657E-03	Mo-97
Nb-97m	60.0 s	IT	4.403E-03	Nb-97
Nb-98	2.80 s	Beta	1.234E-02	Mo-98
Nb-98m	3.090E+03 s	Beta	1.924E-02	Mo-98m
Nb-99	14.3 s	Beta	9.230E-03	Mo-99
Nb-99m	156. s	Beta	1.286E-02	Mo-99m
Nb-100	2.40 s	Beta	2.359E-02	Mo-100
Nb-100m	2.41 s	Beta	2.065E-02	Mo-100m
Nb-101	7.00 s	Beta	1.323E-02	Mo-101
Nb-102	3.00 s	Beta	2.475E-02	Mo-102
Nb-103	15.7 s	Beta	1.849E-02	Mo-103
Nb-104	1.00 s	Beta	3.020E-02	Mo-104
Nb-105	1.80 s	Beta	2.345E-02	Mo-105
Nb-106	0.535 s	Beta	3.560E-02	Mo-106
Nb-107	0.669 s	Beta	2.945E-02	Mo-107
Nb-108	0.222 s	Beta	4.157E-02	Mo-108
Nb-109	0.286 s	Beta	3.495E-02	Mo-109
Nb-110	0.126 s	Beta	4.643E-02	Mo-110
Nb-111	0.156 s	Beta	4.015E-02	Mo-111
Nb-112	8.510E-02 s	Beta	5.101E-02	Mo-112
Mo-93	1.104E+11 s	POS or EC	9.366E-05	Nb-93
Mo-93m	2.466E+04 s	IT	1.399E-02	Mo-93
Mo-99	2.376E+05 s	Beta to * (87.6%)	3.212E-03	Tc-99m
		Beta (12.4%)		Tc-99
Mo-101	877. s	Beta	1.142E-02	Tc-101
Mo-102	666. s	Beta	1.844E-03	Tc-102
Mo-103	60.0 s	Beta	1.360E-02	Tc-103
Mo-104	96.0 s	Beta	6.136E-03	Tc-104
Mo-105	54.0 s	Beta	1.847E-02	Tc-105
Mo-106	9.00 s	Beta	1.064E-02	Tc-106
Mo-107	6.39 s	Beta	2.189E-02	Tc-107
Mo-108	1.50 s	Beta	1.595E-02	Tc-108
Mo-109	1.03 s	Beta	2.722E-02	Tc-109
Mo-110	1.89 s	Beta	2.080E-02	Tc-110
Mo-111	0.392 s	Beta	3.247E-02	Tc-111
Mo-112	0.689 s	Beta	2.569E-02	Tc-112
Mo-113	0.197 s	Beta	3.801E-02	Tc-113
Mo-114	0.322 s	Beta	3.081E-02	Tc-114
Mo-115	0.116 s	Beta	4.279E-02	Tc-115
Tc-97	2.60 10^6 y	POS or EC	1.008E-04	Mo-97

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Tc-97m	90.0 d	IT	5.721E-04	Tc-97
Tc-98	1.325E+14 s	Beta	9.082E-03	Ru-98
Tc-99	6.722E+12 s	Beta	5.015E-04	Ru-99
Tc-99m	2.167E+04 s	IT	8.430E-04	Tc-99
Tc-100	15.8 s	Beta	8.803E-03	Ru-100
Tc-101	852. s	Beta	4.800E-03	Ru-101
Tc-102	5.28 s	Beta	1.031E-02	Ru-102
Tc-102m	261. s	IT (5.00%)	1.898E-02	Tc-102
		Beta (95.0%)		Ru-102m
Tc-103	50.0 s	Beta	7.274E-03	Ru-103
Tc-104	1.092E+03 s	Beta	2.174E-02	Ru-104
Tc-105	480. s	Beta	1.103E-02	Ru-105
Tc-106	37.0 s	Beta	2.305E-02	Ru-106
Tc-107	29.0 s	Beta	1.660E-02	Ru-107
Tc-108	5.20 s	Beta	2.739E-02	Ru-108
Tc-109	51.0 s	Beta	2.207E-02	Ru-109
Tc-110	0.830 s	Beta	3.342E-02	Ru-110
Tc-111	1.34 s	Beta	2.708E-02	Ru-111
Tc-112	0.355 s	Beta	3.883E-02	Ru-112
Tc-113	0.458 s	Beta	3.217E-02	Ru-113
Tc-114	0.173 s	Beta	4.435E-02	Ru-114
Tc-115	0.222 s	Beta	3.748E-02	Ru-115
Tc-116	0.106 s	Beta	4.898E-02	Ru-116
Tc-117	0.135 s	Beta	4.163E-02	Ru-117
Tc-118	7.722E-02 s	Beta	5.292E-02	Ru-118
Ru-97	2.506E+05 s	POS or EC (99.9%)	1.497E-03	Tc-97
		POS or EC to * (7.540E-02%)		Tc-97m
Ru-103	3.394E+06 s	Beta to * (90.1%)	3.346E-03	Rh-103m
		Beta (9.94%)		Rh-103
Ru-105	1.598E+04 s	Beta to * (28.0%)	7.019E-03	Rh-105m
		Beta (72.0%)		Rh-105
Ru-106	3.181E+07 s	Beta	5.946E-05	Rh-106
Ru-107	252. s	Beta	8.608E-03	Rh-107
Ru-108	270. s	Beta	3.059E-03	Rh-108
Ru-109	35.0 s	Beta to * (50.0%)	1.412E-02	Rh-109m
		Beta (50.0%)		Rh-109
Ru-110	16.0 s	Beta	9.123E-03	Rh-110
Ru-111	15.4 s	Beta	1.922E-02	Rh-111
Ru-112	0.700 s	Beta	1.308E-02	Rh-112
Ru-113	2.77 s	Beta	2.397E-02	Rh-113
Ru-114	5.05 s	Beta	1.746E-02	Rh-114
Ru-115	0.729 s	Beta	2.899E-02	Rh-115
Ru-116	1.40 s	Beta	2.211E-02	Rh-116
Ru-117	0.309 s	Beta	3.450E-02	Rh-117
Ru-118	0.616 s	Beta	2.626E-02	Rh-118
Ru-119	0.177 s	Beta	3.892E-02	Rh-119
Ru-120	0.293 s	Beta	3.154E-02	Rh-120

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Rh-102	2.90 y	POS or EC	1.276E-02	Ru-102
Rh-103m	3.367E+03 s	IT	2.302E-04	Rh-103
Rh-104	42.3 s	POS or EC (0.450%)	5.913E-03	Ru-104
		Beta (99.6%)		Pd-104
Rh-104m	260. s	IT (99.9%)	8.299E-04	Rh-104
		Beta (0.130%)		Pd-104m
Rh-105	1.273E+05 s	Beta	1.369E-03	Pd-105
Rh-105m	45.0 s	IT	7.641E-04	Rh-105
Rh-106	29.9 s	Beta	9.592E-03	Pd-106
Rh-106m	7.920E+03 s	Beta	1.906E-02	Pd-106m
Rh-107	1.302E+03 s	Beta	4.783E-03	Pd-107
Rh-108	16.8 s	Beta	1.382E-02	Pd-108
Rh-108m	354. s	Beta	1.885E-02	Pd-108m
Rh-109	90.0 s	Beta to * (50.0%)	7.558E-03	Pd-109m
		Beta (50.0%)		Pd-109
Rh-109m	50.0 s	IT	1.482E-03	Rh-109
Rh-110	29.0 s	Beta	2.142E-02	Pd-110
Rh-110m	3.00 s	Beta	1.504E-02	Pd-110m
Rh-111	63.0 s	Beta to * (0.400%)	1.349E-02	Pd-111m
		Beta (99.6%)		Pd-111
Rh-112	4.70 s	Beta	2.415E-02	Pd-112
Rh-113	0.900 s	Beta	1.786E-02	Pd-113
Rh-114	1.70 s	Beta	2.880E-02	Pd-114
Rh-115	6.02 s	Beta	2.244E-02	Pd-115
Rh-116	0.833 s	Beta	3.401E-02	Pd-116
Rh-117	1.08 s	Beta	2.720E-02	Pd-117
Rh-118	0.295 s	Beta	3.965E-02	Pd-118
Rh-119	0.448 s	Beta	3.151E-02	Pd-119
Rh-120	0.162 s	Beta	4.377E-02	Pd-120
Rh-121	0.221 s	Beta	3.695E-02	Pd-121
Rh-122	0.105 s	Beta	4.725E-02	Pd-122
Rh-123	0.133 s	Beta	4.101E-02	Pd-123
Pd-103	1.465E+06 s	POS or EC	3.824E-04	Rh-103
Pd-107	2.050E+14 s	Beta	5.928E-05	Ag-107
Pd-107m	21.3 s	IT	1.245E-03	Pd-107
Pd-109	4.846E+04 s	Beta to * (99.9%)	2.659E-03	Ag-109m
		Beta (5.000E-02%)		Ag-109
Pd-109m	281. s	IT	1.114E-03	Pd-109
Pd-111	1.320E+03 s	Beta to * (99.2%)	5.318E-03	Ag-111m
		Beta (0.800%)		Ag-111
Pd-111m	1.980E+04 s	Beta to * (32.0%)	3.486E-03	Ag-111
		IT (68.0%)		Pd-111
Pd-112	7.236E+04 s	Beta	9.189E-04	Ag-112
Pd-113	90.0 s	Beta to * (10.0%)	1.177E-02	Ag-113m
		Beta (90.0%)		Ag-113
Pd-114	144. s	Beta	6.954E-03	Ag-114

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Pd-115	38.0 s	Beta to * (27.0%)	1.529E-02	Ag-115m
		Beta (73.0%)		Ag-115
Pd-116	14.0 s	Beta to * (50.0%)	9.301E-03	Ag-116m
		Beta (50.0%)		Ag-116
Pd-117	5.00 s	Beta to * (50.0%)	1.997E-02	Ag-117m
		Beta (50.0%)		Ag-117
Pd-118	3.10 s	Beta to * (50.0%)	1.337E-02	Ag-118m
		Beta (50.0%)		Ag-118
Pd-119	1.71 s	Beta	2.538E-02	Ag-119
Pd-120	4.27 s	Beta	1.749E-02	Ag-120
Pd-121	0.622 s	Beta	2.940E-02	Ag-121
Pd-122	1.27 s	Beta	2.233E-02	Ag-122
Pd-123	0.310 s	Beta	3.373E-02	Ag-123
Pd-124	0.560 s	Beta	2.678E-02	Ag-124
Pd-125	0.183 s	Beta	3.776E-02	Ag-125
Pd-126	0.287 s	Beta	3.161E-02	Ag-126
Ag-106	8.50 d	POS or EC	7.226E-03	Pd-106
Ag-108	142. s	POS or EC (2.35%)	3.723E-03	Pd-108
		Beta (97.6%)		Cd-108
Ag-108m	4.008E+09 s	POS or EC (91.1%)	9.687E-03	Pd-108m
		IT (8.90%)		Ag-108
Ag-109m	39.6 s	IT	5.155E-04	Ag-109
Ag-110	24.6 s	POS or EC (0.300%)	7.185E-03	Pd-110
		Beta (99.7%)		Cd-110
Ag-110m	2.159E+07 s	IT (1.33%)	1.670E-02	Ag-110
		Beta (98.7%)		Cd-110m
Ag-111	6.437E+05 s	Beta	2.240E-03	Cd-111
Ag-111m	65.0 s	IT	3.853E-04	Ag-111
Ag-112	1.127E+04 s	Beta	1.241E-02	Cd-112
Ag-113	1.908E+04 s	Beta to * (1.30%)	6.242E-03	Cd-113m
		Beta (98.7%)		Cd-113
Ag-113m	66.0 s	Beta to * (4.50%)	7.001E-03	Cd-113
		Beta (95.5%)		Cd-113m
Ag-114	4.52 s	Beta	1.215E-02	Cd-114
Ag-115	1.200E+03 s	Beta to * (2.05%)	1.023E-02	Cd-115m
		Beta (98.0%)		Cd-115
Ag-115m	17.0 s	Beta to * (27.0%)	1.131E-02	Cd-115
		Beta (73.0%)		Cd-115m
Ag-116	161. s	Beta	2.030E-02	Cd-116
Ag-116m	10.4 s	Beta	2.377E-02	Cd-116m
Ag-117	73.2 s	Beta to * (20.0%)	1.470E-02	Cd-117m
		Beta (80.0%)		Cd-117
Ag-117m	5.30 s	Beta to * (50.0%)	1.541E-02	Cd-117
		Beta (50.0%)		Cd-117m
Ag-118	3.70 s	Beta	1.888E-02	Cd-118
Ag-118m	2.80 s	IT (41.0%)	1.259E-02	Ag-118
		Beta (59.0%)		Cd-118m

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Ag-119	6.00 s	Beta to * (50.0%)	1.885E-02	Cd-119m
		Beta (50.0%)		Cd-119
Ag-120	1.17 s	Beta	1.523E-02	Cd-120
Ag-121	3.00 s	Beta	2.289E-02	Cd-121
Ag-122	0.100 s	Beta	3.485E-02	Cd-122
Ag-123	0.863 s	Beta	2.786E-02	Cd-123
Ag-124	0.269 s	Beta	3.932E-02	Cd-124
Ag-125	0.382 s	Beta	3.248E-02	Cd-125
Ag-126	0.155 s	Beta	4.308E-02	Cd-126
Ag-127	0.205 s	Beta	3.728E-02	Cd-127
Ag-128	0.102 s	Beta	4.679E-02	Cd-128
Cd-107	2.336E+04 s	POS or EC	7.155E-04	Ag-107
Cd-109	4.009E+07 s	POS or EC to *	1.162E-04	Ag-109m
Cd-111m	2.922E+03 s	IT	2.348E-03	Cd-111
Cd-113m	4.604E+08 s	IT (0.100%)	1.684E-03	Cd-113
		Beta (99.9%)		In-113m
Cd-115	1.925E+05 s	Beta to *	3.174E-03	In-115m
Cd-115m	3.853E+06 s	Beta to * (7.000E-03%)	3.730E-03	In-115
		Beta (100.%)		In-115m
Cd-117	9.360E+03 s	Beta to * (93.0%)	7.203E-03	In-117m
		Beta (7.00%)		In-117
Cd-117m	1.224E+04 s	Beta to * (44.0%)	8.127E-03	In-117
		Beta (56.0%)		In-117m
Cd-118	3.018E+03 s	Beta	2.602E-03	In-118
Cd-119	564. s	Beta to *	1.097E-02	In-119m
Cd-119m	192. s	Beta to * (50.0%)	1.224E-02	In-119
		Beta (50.0%)		In-119m
Cd-120	50.8 s	Beta to * (50.0%)	5.620E-03	In-120m
		Beta (50.0%)		In-120
Cd-121	12.8 s	Beta to * (18.0%)	1.657E-02	In-121m
		Beta (82.0%)		In-121
Cd-122	5.50 s	Beta	8.590E-03	In-122
Cd-123	8.40 s	Beta to * (23.0%)	1.997E-02	In-123m
		Beta (77.0%)		In-123
Cd-124	17.2 s	Beta	1.356E-02	In-124
Cd-125	1.62 s	Beta to * (30.0%)	2.395E-02	In-125m
		Beta (70.0%)		In-125
Cd-126	3.77 s	Beta	1.755E-02	In-126
Cd-127	0.659 s	Beta to * (50.0%)	2.762E-02	In-127m
		Beta (50.0%)		In-127
Cd-128	1.29 s	Beta	2.190E-02	In-128
Cd-129	0.338 s	Beta	3.217E-02	In-129
Cd-130	0.524 s	Beta	2.714E-02	In-130
Cd-131	0.119 s	Beta	4.389E-02	In-131
Cd-132	0.145 s	Beta	3.965E-02	In-132
In-113m	5.969E+03 s	IT	2.330E-03	In-113

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
In-114	71.9 s	POS or EC (3.37%)	4.763E-03	Cd-114
		Beta (96.6%)		Sn-114
In-114m	4.278E+06 s	POS or EC (4.30%)	1.419E-03	Cd-114m
		IT (95.7%)		In-114
In-115	1.577E+22 s	Beta	1.435E-03	Sn-115
In-115m	1.548E+04 s	IT (96.3%)	1.994E-03	In-115
		Beta (3.70%)		Sn-115m
In-116	14.1 s	Beta	8.193E-03	Sn-116
In-116m	3.249E+03 s	Beta	1.646E-02	Sn-116m
In-117	2.640E+03 s	Beta	4.505E-03	Sn-117
In-117m	6.984E+03 s	IT (47.0%)	3.776E-03	In-117
		Beta (53.0%)		Sn-117m
In-118	5.00 s	Beta	1.208E-02	Sn-118
In-118m	267. s	Beta	1.959E-02	Sn-118m
In-119	150. s	Beta to * (5.00%)	7.997E-03	Sn-119m
		Beta (95.0%)		Sn-119
In-119m	1.080E+03 s	IT (5.00%)	8.448E-03	In-119
		Beta (95.0%)		Sn-119m
In-120	44.4 s	Beta	2.316E-02	Sn-120
In-120m	3.08 s	Beta	1.602E-02	Sn-120m
In-121	28.0 s	Beta	1.205E-02	Sn-121
In-121m	198. s	Beta	1.288E-02	Sn-121m
In-122	10.0 s	Beta	2.756E-02	Sn-122
In-122m	1.50 s	Beta	1.962E-03	Sn-122m
In-123	5.97 s	Beta to * (95.0%)	1.388E-02	Sn-123m
		Beta (5.00%)		Sn-123
In-123m	48.0 s	Beta to * (50.0%)	1.612E-02	Sn-123
		Beta (50.0%)		Sn-123m
In-124	3.20 s	Beta	2.641E-02	Sn-124
In-125	2.33 s	Beta to * (70.0%)	1.915E-02	Sn-125m
		Beta (30.0%)		Sn-125
In-125m	12.0 s	Beta to * (92.0%)	1.987E-02	Sn-125
		Beta (8.00%)		Sn-125m
In-126	1.53 s	Beta	3.042E-02	Sn-126
In-127	2.00 s	Beta + n (0.670%)	2.410E-02	Sn-126
		Beta (99.3%)		Sn-127
In-127m	3.64 s	Beta	2.518E-02	Sn-127m
In-128	3.70 s	Beta + n (1.20%)	3.479E-02	Sn-127
		Beta (98.8%)		Sn-128
In-129	0.800 s	Beta to * (51.8%)	2.738E-02	Sn-129m
		Beta + n (3.50%)		Sn-128
		Beta (44.7%)		Sn-129
In-130	0.530 s	Beta + n (4.50%)	3.185E-02	Sn-129
		Beta (95.5%)		Sn-130
In-131	0.300 s	Beta + n (9.50%)	3.212E-02	Sn-130
		Beta (90.5%)		Sn-131
In-132	0.120 s	Beta	5.030E-02	Sn-132

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
In-133	0.114 s	Beta	4.626E-02	Sn-133
In-134	7.754E-02 s	Beta	5.423E-02	Sn-134
Sn-113	9.945E+06 s	POS or EC to *	1.666E-04	In-113m
Sn-113m	20.0 min	POS or EC	4.701E-04	In-113m
Sn-117m	1.210E+06 s	IT	1.855E-03	Sn-117
Sn-119m	2.117E+07 s	IT	5.169E-04	Sn-119
Sn-121	9.648E+04 s	Beta	1.209E-03	Sb-121
Sn-121m	1.577E+09 s	Beta	2.004E-03	Sb-121m
Sn-123	1.116E+07 s	Beta	3.124E-03	Sb-123
Sn-123m	2.405E+03 s	Beta	3.645E-03	Sb-123m
Sn-125	8.329E+05 s	Beta	6.628E-03	Sb-125
Sn-125m	571. s	Beta	6.728E-03	Sb-125m
Sn-126	3.156E+12 s	Beta to *	1.247E-03	Sb-126m
Sn-127	7.560E+03 s	Beta	1.448E-02	Sb-127
Sn-127m	248. s	Beta	9.651E-03	Sb-127m
Sn-128	3.540E+03 s	Beta to *	4.825E-03	Sb-128m
Sn-129	450. s	Beta	1.500E-02	Sb-129
Sn-129m	150. s	Beta	1.593E-02	Sb-129m
Sn-130	223. s	Beta to *	7.795E-03	Sb-130m
Sn-131	63.0 s	Beta	1.786E-02	Sb-131
Sn-132	40.0 s	Beta	1.222E-02	Sb-132
Sn-133	1.47 s	Beta + n (2.100E-02%)	2.897E-02	Sb-132
		Beta (100.%)		Sb-133
Sn-134	0.845 s	Beta	2.451E-02	Sb-134
Sn-135	0.291 s	Beta	3.344E-02	Sb-135
Sn-136	0.413 s	Beta	2.904E-02	Sb-136
Sb-122	2.333E+05 s	POS or EC (2.40%)	5.964E-03	Sn-122
		Beta (97.6%)		Te-122
Sb-122m	252. s	IT	9.604E-04	Sb-122
Sb-124	5.201E+06 s	Beta	1.328E-02	Te-124
Sb-124m	93.0 s	IT (80.0%)	2.571E-03	Sb-124
		Beta (20.0%)		Te-124m
Sb-125	8.741E+07 s	Beta to * (23.0%)	3.126E-03	Te-125m
		Beta (77.0%)		Te-125
Sb-126	1.071E+06 s	Beta	1.848E-02	Te-126
Sb-126m	1.140E+03 s	IT (14.0%)	1.273E-02	Sb-126
		Beta (86.0%)		Te-126m
Sb-127	3.326E+05 s	Beta to * (13.9%)	5.934E-03	Te-127m
		Beta (86.1%)		Te-127
Sb-128	3.244E+04 s	Beta	2.064E-02	Te-128
Sb-128m	624. s	Beta	1.755E-02	Te-128m
Sb-129	1.555E+04 s	Beta to * (13.1%)	1.117E-02	Te-129m
		Beta (86.9%)		Te-129
Sb-130	2.400E+03 s	Beta	2.341E-02	Te-130
Sb-130m	378. s	Beta	2.203E-02	Te-130m
Sb-131	1.380E+03 s	Beta to * (6.80%)	1.448E-02	Te-131m
		Beta (93.2%)		Te-131

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Sb-132	168. s	Beta	2.333E-02	Te-132
Sb-132m	252. s	Beta	2.330E-02	Te-132m
Sb-133	144. s	Beta to * (2.20%)	2.193E-02	Te-133m
		Beta (97.8%)		Te-133
Sb-134	11.0 s	Beta	2.923E-02	Te-134
Sb-134m	10.7 s	Beta + n (8.000E-02%)	2.993E-02	Te-133m
		Beta (99.9%)		Te-134m
Sb-135	1.70 s	Beta + n (8.00%)	2.888E-02	Te-134
		Beta (92.0%)		Te-135
Sb-136	0.231 s	Beta	3.898E-02	Te-136
Sb-137	0.284 s	Beta	3.466E-02	Te-137
Sb-138	0.130 s	Beta	4.416E-02	Te-138
Sb-139	0.172 s	Beta	3.851E-02	Te-139
Te-121	1.469E+06 s	POS or EC	3.470E-03	Sb-121
Te-121m	1.331E+07 s	POS or EC (11.4%)	1.752E-03	Sb-121m
		IT (88.6%)		Te-121
Te-123	3.156E+20 s	POS or EC	1.014E-04	Sb-123
Te-123m	1.034E+07 s	IT	1.457E-03	Te-123
Te-125m	5.011E+06 s	IT	8.406E-04	Te-125
Te-127	3.366E+04 s	Beta	1.350E-03	I-127
Te-127m	9.418E+06 s	IT (97.6%)	5.379E-04	Te-127
		Beta (2.40%)		I-127m
Te-129	4.176E+03 s	Beta	3.573E-03	I-129
Te-129m	2.903E+06 s	IT (65.0%)	1.754E-03	Te-129
		Beta (35.0%)		I-129m
Te-131	1.500E+03 s	Beta	6.752E-03	I-131
Te-131m	1.080E+05 s	IT (22.2%)	9.615E-03	Te-131
		Beta (77.8%)		I-131m
Te-132	2.815E+05 s	Beta	1.981E-03	I-132
Te-133	747. s	Beta	1.035E-02	I-133
Te-133m	3.324E+03 s	IT (13.0%)	1.768E-02	Te-133
		Beta (87.0%)		I-133m
Te-134	2.508E+03 s	Beta	7.007E-03	I-134
Te-135	19.2 s	Beta	1.841E-02	I-135
Te-136	21.0 s	Beta + n (0.500%)	1.684E-02	I-135
		Beta (99.5%)		I-136
Te-137	3.50 s	Beta + n (0.500%)	2.544E-02	I-136
		Beta (99.5%)		I-137
Te-138	1.64 s	Beta	2.127E-02	I-138
Te-139	0.424 s	Beta	3.112E-02	I-139
Te-140	0.752 s	Beta	2.515E-02	I-140
Te-141	0.236 s	Beta	3.563E-02	I-141
Te-142	0.491 s	Beta	2.745E-02	I-142
I-125	59.7 d	POS or EC	3.480E-04	Te-125
I-126	1.125E+06 s	POS or EC (56.0%)	3.553E-03	Te-126
		Beta (44.0%)		Xe-126

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
I-128	1.499E+03 s	POS or EC (6.00%)	4.961E-03	Te-128
		Beta (94.0%)		Xe-128
I-129	4.954E+14 s	Beta	4.626E-04	Xe-129
I-130	4.450E+04 s	Beta	1.440E-02	Xe-130
I-130m	540. s	IT (83.0%)	1.823E-03	I-130
		Beta (17.0%)		Xe-130m
I-131	6.947E+05 s	Beta to * (1.11%)	3.397E-03	Xe-131m
		Beta (98.9%)		Xe-131
I-132	8.280E+03 s	Beta	1.644E-02	Xe-132
I-133	7.488E+04 s	Beta to * (2.88%)	6.005E-03	Xe-133m
		Beta (97.1%)		Xe-133
I-133m	9.00 s	IT	9.669E-03	I-133
I-134	3.156E+03 s	Beta	1.925E-02	Xe-134
I-134m	222. s	IT (98.0%)	2.056E-03	I-134
		Beta (2.00%)		Xe-134m
I-135	2.380E+04 s	Beta to * (15.4%)	1.148E-02	Xe-135m
		Beta (84.6%)		Xe-135
I-136	83.0 s	Beta	2.523E-02	Xe-136
I-136m	46.0 s	Beta	2.267E-02	Xe-136m
I-137	24.6 s	Beta + n (5.40%)	2.100E-02	Xe-136
		Beta (94.6%)		Xe-137
I-138	6.40 s	Beta + n (2.50%)	2.371E-02	Xe-137
		Beta (97.5%)		Xe-138
I-139	2.40 s	Beta + n (10.0%)	2.504E-02	Xe-138
		Beta (90.0%)		Xe-139
I-140	0.860 s	Beta + n (32.0%)	2.976E-02	Xe-139
		Beta (68.0%)		Xe-140
I-141	0.400 s	Beta + n (12.0%)	2.866E-02	Xe-140
		Beta (88.0%)		Xe-141
I-142	0.196 s	Beta	4.052E-02	Xe-142
I-143	0.328 s	Beta	3.267E-02	Xe-143
I-144	0.133 s	Beta	4.278E-02	Xe-144
I-145	0.187 s	Beta	3.691E-02	Xe-145
Xe-125	17.0 h	POS or EC	1.785E-03	I-125
Xe-125m	57.0 s	IT	1.483E-03	Xe-125
Xe-127	3.146E+06 s	POS or EC	1.832E-03	I-127
Xe-127m	70.0 s	IT	1.350E-03	Xe-127
Xe-129m	6.912E+05 s	IT	1.399E-03	Xe-129
Xe-131m	1.028E+06 s	IT	9.621E-04	Xe-131
Xe-133	4.532E+05 s	Beta	1.071E-03	Cs-133
Xe-133m	1.892E+05 s	IT	1.374E-03	Xe-133
Xe-134m	0.290 s	IT	1.129E-02	Xe-134
Xe-135	3.272E+04 s	Beta	3.346E-03	Cs-135
Xe-135m	917. s	IT	3.122E-03	Xe-135
Xe-137	230. s	Beta	1.164E-02	Cs-137
Xe-138	850. s	Beta	1.068E-02	Cs-138
Xe-139	39.5 s	Beta	1.584E-02	Cs-139

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Xe-140	13.6 s	Beta	1.330E-02	Cs-140
Xe-141	1.72 s	Beta + n (5.400E-02%)	2.277E-02	Cs-140
		Beta (99.9%)		Cs-141
Xe-142	1.22 s	Beta + n (0.510%)	1.697E-02	Cs-141
		Beta (99.5%)		Cs-142
Xe-143	0.300 s	Beta + n (1.10%)	2.661E-02	Cs-142
		Beta (98.9%)		Cs-143
Xe-144	1.00 s	Beta	1.899E-02	Cs-144
Xe-145	0.900 s	Beta	2.985E-02	Cs-145
Xe-146	0.937 s	Beta	2.343E-02	Cs-146
Xe-147	0.264 s	Beta	3.389E-02	Cs-147
Cs-131	9.70 d	POS or EC	1.654E-04	Xe-131
Cs-132	5.594E+05 s	POS or EC (98.0%)	4.317E-03	Xe-132
		Beta (2.00%)		Ba-132
Cs-134	6.507E+07 s	Beta	1.018E-02	Ba-134
Cs-134m	1.044E+04 s	IT	8.015E-04	Cs-134
Cs-135	7.258E+13 s	Beta	3.338E-04	Ba-135
Cs-135m	53.0 min	IT	9.598E-03	Cs-135
Cs-136	1.132E+06 s	Beta to * (16.5%)	1.363E-02	Ba-136m
		Beta (83.5%)		Ba-136
Cs-137	9.467E+08 s	Beta to * (94.6%)	1.106E-03	Ba-137m
		Beta (5.40%)		Ba-137
Cs-138	1.932E+03 s	Beta	2.109E-02	Ba-138
Cs-138m	174. s	IT (75.0%)	5.536E-03	Cs-138
		Beta (25.0%)		Ba-138m
Cs-139	564. s	Beta	1.184E-02	Ba-139
Cs-140	63.8 s	Beta	2.408E-02	Ba-140
Cs-141	25.0 s	Beta + n (7.300E-02%)	1.898E-02	Ba-140
		Beta (99.9%)		Ba-141
Cs-142	1.70 s	Beta + n (0.210%)	2.720E-02	Ba-141
		Beta (99.8%)		Ba-142
Cs-143	1.70 s	Beta + n (1.13%)	2.213E-02	Ba-142
		Beta (98.9%)		Ba-143
Cs-144	1.02 s	Beta + n (1.10%)	3.196E-02	Ba-143
		Beta (98.9%)		Ba-144
Cs-145	0.560 s	Beta + n (4.40%)	2.384E-02	Ba-144
		Beta (95.6%)		Ba-145
Cs-146	0.190 s	Beta + n (3.90%)	3.439E-02	Ba-145
		Beta (96.1%)		Ba-146
Cs-147	0.558 s	Beta	2.908E-02	Ba-147
Cs-148	0.202 s	Beta	3.892E-02	Ba-148
Cs-149	0.278 s	Beta	3.391E-02	Ba-149
Cs-150	0.124 s	Beta	4.304E-02	Ba-150
Ba-131	1.020E+06 s	POS or EC	3.055E-03	Cs-131
Ba-131m	15.0 min	IT	1.067E-03	Ba-131
Ba-133	3.389E+08 s	POS or EC	2.623E-03	Cs-133

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Ba-133m	1.400E+05 s	POS or EC (1.100E-02%)	1.690E-03	Cs-133m
		IT (100.%)		Ba-133
Ba-135m	1.033E+05 s	IT	1.580E-03	Ba-135
Ba-136m	0.308 s	IT	1.209E-02	Ba-136
Ba-137m	153. s	IT	3.927E-03	Ba-137
Ba-139	4.962E+03 s	Beta	5.569E-03	La-139
Ba-140	1.105E+06 s	Beta	2.790E-03	La-140
Ba-141	1.096E+03 s	Beta	1.025E-02	La-141
Ba-142	642. s	Beta	8.708E-03	La-142
Ba-143	13.6 s	Beta	1.576E-02	La-143
Ba-144	11.0 s	Beta	1.004E-02	La-144
Ba-145	6.20 s	Beta	1.902E-02	La-145
Ba-146	2.20 s	Beta	1.150E-02	La-146
Ba-147	2.23 s	Beta	2.168E-02	La-147
Ba-148	5.90 s	Beta	1.553E-02	La-148
Ba-149	0.918 s	Beta	2.554E-02	La-149
Ba-150	1.80 s	Beta	2.022E-02	La-150
Ba-151	0.437 s	Beta	3.010E-02	La-151
Ba-152	0.755 s	Beta	2.485E-02	La-152
La-137	1.893E+12 s	POS or EC	1.837E-04	Ba-137
La-138	4.260E+18 s	POS or EC (67.1%)	7.333E-03	Ba-138
		Beta (32.9%)		Ce-138
La-140	1.448E+05 s	Beta	1.676E-02	Ce-140
La-141	1.415E+04 s	Beta	5.873E-03	Ce-141
La-142	5.562E+03 s	Beta	2.118E-02	Ce-142
La-143	840. s	Beta	1.169E-02	Ce-143
La-144	40.0 s	Beta	2.043E-02	Ce-144
La-145	29.0 s	Beta	1.528E-02	Ce-145
La-146	8.30 s	Beta	2.445E-02	Ce-146
La-147	10.0 s	Beta	1.704E-02	Ce-147
La-148	1.30 s	Beta	2.728E-02	Ce-148
La-149	2.86 s	Beta	2.123E-02	Ce-149
La-150	0.649 s	Beta	3.124E-02	Ce-150
La-151	0.954 s	Beta	2.608E-02	Ce-151
La-152	0.309 s	Beta	3.600E-02	Ce-152
La-153	0.437 s	Beta	3.087E-02	Ce-153
La-154	0.175 s	Beta	4.051E-02	Ce-154
La-155	0.221 s	Beta	3.628E-02	Ce-155
Ce-137	3.240E+04 s	POS or EC	3.096E-04	La-137
Ce-137m	1.238E+05 s	POS or EC (1.00%)	1.518E-03	La-137m
		IT (99.0%)		Ce-137
Ce-139	1.189E+07 s	POS or EC	1.141E-03	La-139
Ce-139m	56.2 s	IT	4.457E-03	Ce-139
Ce-141	2.809E+06 s	Beta	1.464E-03	Pr-141
Ce-142	3.311E+18 s	Alpha	0.000E+00	Ba-138
Ce-143	1.188E+05 s	Beta	4.212E-03	Pr-143

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Ce-144	2.456E+07 s	Beta to * (1.20%)	6.634E-04	Pr-144m
		Beta (98.8%)		Pr-144
Ce-145	180. s	Beta	8.803E-03	Pr-145
Ce-146	852. s	Beta	2.603E-03	Pr-146
Ce-147	70.0 s	Beta	1.259E-02	Pr-147
Ce-148	43.0 s	Beta	5.845E-03	Pr-148
Ce-149	1.00 s	Beta	1.490E-02	Pr-149
Ce-150	1.00 s	Beta	9.011E-03	Pr-150
Ce-151	1.00 s	Beta	1.837E-02	Pr-151
Ce-152	14.0 s	Beta	1.326E-02	Pr-152
Ce-153	1.73 s	Beta	2.259E-02	Pr-153
Ce-154	3.59 s	Beta	1.749E-02	Pr-154
Ce-155	0.712 s	Beta	2.710E-02	Pr-155
Ce-156	1.16 s	Beta	2.271E-02	Pr-156
Ce-157	0.362 s	Beta	3.173E-02	Pr-157
Pr-139	4.40 h	POS or EC	9.953E-04	Ce-139
Pr-140	3.39 min	POS or EC	4.150E-03	Ce-140
Pr-142	6.887E+04 s	Beta	5.140E-03	Nd-142
Pr-142m	876. s	IT	1.482E-03	Pr-142
Pr-143	1.172E+06 s	Beta	1.863E-03	Nd-143
Pr-144	1.037E+03 s	Beta	7.351E-03	Nd-144
Pr-144m	432. s	IT (99.9%)	3.422E-04	Pr-144
		Beta (6.000E-02%)		Nd-144m
Pr-145	2.153E+04 s	Beta	4.099E-03	Nd-145
Pr-146	1.452E+03 s	Beta	1.519E-02	Nd-146
Pr-147	720. s	Beta	9.295E-03	Nd-147
Pr-148	138. s	Beta	1.672E-02	Nd-148
Pr-149	138. s	Beta	8.353E-03	Nd-149
Pr-150	12.4 s	Beta	1.904E-02	Nd-150
Pr-151	4.00 s	Beta	1.409E-02	Nd-151
Pr-152	8.32 s	Beta	2.363E-02	Nd-152
Pr-153	7.74 s	Beta	1.864E-02	Nd-153
Pr-154	1.31 s	Beta	2.800E-02	Nd-154
Pr-155	1.89 s	Beta	2.302E-02	Nd-155
Pr-156	0.510 s	Beta	3.266E-02	Nd-156
Pr-157	0.678 s	Beta	2.838E-02	Nd-157
Pr-158	0.263 s	Beta	3.745E-02	Nd-158
Pr-159	0.314 s	Beta	3.396E-02	Nd-159
Nd-141	2.50 h	POS or EC	5.434E-04	Pr-141
Nd-144	6.623E+22 s	Alpha	0.000E+00	Ce-140
Nd-147	9.556E+05 s	Beta	2.413E-03	Pm-147
Nd-149	6.228E+03 s	Beta	5.276E-03	Pm-149
Nd-151	744. s	Beta	8.791E-03	Pm-151
Nd-152	690. s	Beta	3.332E-03	Pm-152
Nd-153	67.5 s	Beta	1.238E-02	Pm-153
Nd-154	40.0 s	Beta	6.396E-03	Pm-154
Nd-155	26.1 s	Beta	1.513E-02	Pm-155

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Nd-156	58.5 s	Beta	1.023E-02	Pm-156
Nd-157	4.15 s	Beta	1.924E-02	Pm-157
Nd-158	7.89 s	Beta	1.503E-02	Pm-158
Nd-159	1.41 s	Beta	2.347E-02	Pm-159
Nd-160	2.12 s	Beta	1.999E-02	Pm-160
Nd-161	0.556 s	Beta	2.886E-02	Pm-161
Pm-145	5.586E+08 s	POS or EC	2.626E-04	Nd-145
Pm-146	5.50 y	POS or EC (63.0%)	5.044E-03	Nd-146
		Beta (37.0%)		Sm-146
Pm-147	8.279E+07 s	Beta	3.587E-04	Sm-147
Pm-148	4.640E+05 s	Beta	7.701E-03	Sm-148
Pm-148m	3.568E+06 s	IT (4.90%)	1.268E-02	Pm-148
		Beta (95.1%)		Sm-148m
Pm-149	1.911E+05 s	Beta	2.234E-03	Sm-149
Pm-150	9.648E+03 s	Beta	1.353E-02	Sm-150
Pm-151	1.022E+05 s	Beta	3.681E-03	Sm-151
Pm-152	246. s	Beta	1.024E-02	Sm-152
Pm-152m	450. s	Beta	1.012E-02	Sm-152m
Pm-153	324. s	Beta	4.446E-03	Sm-153
Pm-154	168. s	Beta	1.568E-02	Sm-154
Pm-154m	108. s	IT (10.0%)	1.515E-02	Pm-154
		Beta (90.0%)		Sm-154m
Pm-155	36.6 s	Beta	1.163E-02	Sm-155
Pm-156	13.1 s	Beta	1.906E-02	Sm-156
Pm-157	68.0 s	Beta	1.557E-02	Sm-157
Pm-158	3.80 s	Beta	2.457E-02	Sm-158
Pm-159	4.23 s	Beta	2.048E-02	Sm-159
Pm-160	0.996 s	Beta	2.901E-02	Sm-160
Pm-161	1.19 s	Beta	2.562E-02	Sm-161
Pm-162	0.400 s	Beta	3.449E-02	Sm-162
Sm-145	2.938E+07 s	POS or EC	5.760E-04	Pm-145
Sm-146	70.0 10^6 y	Alpha	1.506E-02	Nd-142
Sm-147	3.377E+18 s	Alpha	1.369E-02	Nd-143
Sm-148	2.525E+23 s	Alpha	1.194E-02	Nd-144
Sm-149	3.154E+23 s	Alpha	0.000E+00	Nd-145
Sm-151	2.840E+09 s	Beta	1.173E-04	Eu-151
Sm-153	1.681E+05 s	Beta	1.962E-03	Eu-153
Sm-155	1.332E+03 s	Beta	5.833E-03	Eu-155
Sm-156	3.384E+04 s	Beta	2.531E-03	Eu-156
Sm-157	480. s	Beta	9.017E-03	Eu-157
Sm-158	2.639E+03 s	Beta	4.191E-03	Eu-158
Sm-159	162. s	Beta	1.172E-02	Eu-159
Sm-160	349. s	Beta	1.062E-02	Eu-160
Sm-161	12.9 s	Beta	1.628E-02	Eu-161
Sm-162	19.6 s	Beta	1.292E-02	Eu-162
Sm-163	2.56 s	Beta	2.122E-02	Eu-163
Sm-164	4.25 s	Beta	1.741E-02	Eu-164

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Sm-165	0.927 s	Beta	2.602E-02	Eu-165
Eu-149	93.1 d	POS or EC	4.427E-04	Sm-149
Eu-150	36.0 y	POS or EC	9.129E-03	Sm-150
Eu-152	4.292E+08 s	POS or EC (72.1%)	7.564E-03	Sm-152
		Beta (27.9%)		Gd-152
Eu-152m	3.355E+04 s	POS or EC (28.0%)	4.804E-03	Sm-152m
		Beta (72.0%)		Gd-152m
Eu-154	2.714E+08 s	Beta	8.946E-03	Gd-154
Eu-155	1.565E+08 s	Beta	7.274E-04	Gd-155
Eu-156	1.312E+06 s	Beta	1.032E-02	Gd-156
Eu-157	5.472E+04 s	Beta	4.458E-03	Gd-157
Eu-158	2.754E+03 s	Beta	1.263E-02	Gd-158
Eu-159	1.086E+03 s	Beta	9.378E-03	Gd-159
Eu-160	51.0 s	Beta	1.345E-02	Gd-160
Eu-161	42.1 s	Beta	1.231E-02	Gd-161
Eu-162	270. s	Beta	1.985E-02	Gd-162
Eu-163	14.8 s	Beta	1.783E-02	Gd-163
Eu-164	2.17 s	Beta	2.614E-02	Gd-164
Eu-165	2.55 s	Beta	2.241E-02	Gd-165
Gd-152	3.408E+21 s	Alpha	1.303E-02	Sm-148
Gd-153	2.091E+07 s	POS or EC	9.034E-04	Eu-153
Gd-155m	3.100E-02 s	IT	7.203E-04	Gd-155
Gd-159	6.696E+04 s	Beta	3.260E-03	Tb-159
Gd-161	222. s	Beta	7.256E-03	Tb-161
Gd-162	600. s	Beta to * (2.00%)	3.634E-03	Tb-162m
		Beta (98.0%)		Tb-162
Gd-163	92.8 s	Beta	9.882E-03	Tb-163
Gd-164	1.301E+03 s	Beta	6.373E-03	Tb-164
Gd-165	100. s	Beta	1.379E-02	Tb-165
Tb-157	4.734E+09 s	POS or EC	4.802E-05	Gd-157
Tb-160	6.247E+06 s	Beta	8.145E-03	Dy-160
Tb-161	5.979E+05 s	Beta	2.004E-03	Dy-161
Tb-162	448. s	Beta	9.971E-03	Dy-162
Tb-162m	8.028E+03 s	Beta	1.086E-02	Dy-162m
Tb-163	1.170E+03 s	Beta	6.035E-03	Dy-163
Tb-163m	7.00 min	Beta	4.440E-03	Dy-163m
Tb-164	180. s	Beta	1.401E-02	Dy-164
Tb-165	32.8 s	Beta to * (50.0%)	1.010E-02	Dy-165m
		Beta (50.0%)		Dy-165
Dy-157	2.916E+04 s	POS or EC	2.147E-03	Tb-157
Dy-159	144. d	POS or EC	3.432E-04	Tb-159
Dy-165	8.460E+03 s	Beta	4.630E-03	Ho-165
Dy-165m	75.4 s	IT (97.5%)	7.469E-04	Dy-165
		Beta (2.50%)		Ho-165m
Dy-166	2.934E+05 s	Beta	1.168E-03	Ho-166
Ho-163	33.0 y	POS or EC	5.335E-05	Dy-163
Ho-166	9.648E+04 s	Beta	4.287E-03	Er-166

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Ho-166m	3.787E+10 s	Beta	1.108E-02	Er-166m
Er-163	75.0 min	POS or EC	5.928E-05	Ho-163
Er-165	10.3 h	POS or EC	2.709E-04	Ho-165
Er-167m	2.30 s	IT	1.233E-03	Er-167
Er-169	9.40 d	Beta	2.016E-03	Tm-169
Er-171	2.707E+04 s	Beta	4.782E-03	Tm-171
Er-172	49.0 h	Beta	5.395E-03	Tm-172
Tm-170	1.111E+07 s	POS or EC (0.146%)	1.984E-03	Er-170
		Beta (99.9%)		Yb-170
Tm-170m	4.100E-06 s	IT	0.000E+00	Tm-170
Tm-171	6.059E+07 s	Beta	1.551E-04	Yb-171
Tm-172	63.6 h	Beta	1.114E-02	Yb-172
Tm-173	8.24 h	Beta	7.825E-03	Yb-173
Yb-169	2.766E+06 s	POS or EC	2.510E-03	Tm-169
Yb-175	3.620E+05 s	Beta	1.004E-03	Lu-175
Yb-175m	6.700E-02 s	IT	3.041E-03	Yb-175
Yb-177	1.90 h	Beta	8.299E-03	Lu-177
Lu-176	30.0 10^9 y	Beta	6.047E-03	Hf-176
Lu-176m	3.69 h	Beta	7.766E-03	Hf-176m
Lu-177	5.797E+05 s	Beta	1.079E-03	Hf-177
Lu-177m	155. d	IT (22.0%)	8.050E-03	Lu-177
		Beta (78.0%)		Hf-177m
Hf-175	70.0 d	POS or EC	2.460E-03	Lu-175
Hf-178m	4.00 s	IT	6.800E-03	Hf-178
Hf-179m	18.6 s	IT	2.241E-03	Hf-179
Hf-180m	5.50 h	IT	6.770E-03	Hf-180
Hf-181	3.663E+06 s	Beta	4.470E-03	Ta-181
Hf-182	9.00 10^6 y	Beta	2.964E-03	Ta-182
Ta-180	1.600E+13 y	Beta	1.778E-03	W-180
Ta-182	9.936E+06 s	Beta	8.904E-03	W-182
Ta-182m	16.5 min	IT	2.982E-03	Ta-182
Ta-183	5.10 d	Beta	6.343E-03	W-183
W-181	1.047E+07 s	POS or EC	2.863E-04	Ta-181
W-183m	5.20 s	IT	1.541E-03	W-183
W-185	75.1 d	Beta	7.529E-04	Re-185
W-185m	1.67 min	IT	1.168E-03	W-185
W-187	8.604E+04 s	Beta	4.536E-03	Re-187
W-188	5.996E+06 s	Beta	5.981E-04	Re-188
W-189	11.5 min	Beta	1.186E-02	Re-189
Re-186	90.6 h	POS or EC (6.50)	2.138E-03	W-186
		Beta (93.5%)		Os-186
Re-187	50.0 10^9 y	Beta	1.535E-02	Os-187
Re-188	6.113E+04 s	Beta	4.962E-03	Os-188
Re-188m	18.7 min	IT	1.020E-03	Re-188
Re-189	24.3 h	Beta	5.987E-03	Os-189
Os-185	94.0 d	POS or EC	4.263E-03	Re-185
Os-190m	9.90 min	IT	1.011E-02	Os-190

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Os-191	1.331E+06 s	Beta	1.452E-03	Ir-191
Os-191m	13.0 h	IT	4.387E-04	Os-191
Os-193	31.0 h	Beta	6.711E-03	Ir-193
Os-194	6.00 y	Beta	5.750E-04	Ir-194
Ir-192	6.395E+06 s	POS or EC (4.78%)	6.124E-03	Os-192
		Beta (95.2%)		Pt-192
Ir-192m	241. y	IT	9.544E-04	Ir-192
Ir-194	6.894E+04 s	Beta	5.336E-03	Pt-194
Ir-194m	3.200E-02 s	Beta	5.928E-04	Pt-194m
Pt-190	600. 10^9 y	Alpha	1.927E-02	Os-186
Pt-191	3.00 d	POS or EC	2.094E-03	Ir-191
Pt-193	500. y	POS or EC	3.142E-05	Ir-193
Pt-193m	4.30 d	IT	8.774E-04	Pt-193
Pt-195m	2.713E+05 s	IT	1.457E-03	Pt-195
Pt-197	18.0 h	Beta	4.446E-03	Au-197
Pt-197m	80.0 min	IT (97.0%)	2.502E-03	Pt-197
		Beta (3.00%)		Au-197m
Pt-199	30.0 min	Beta	9.959E-03	Au-199
Pt-199m	14.1 s	IT	2.519E-03	Pt-199
Au-198	2.70 d	Beta	8.145E-03	Hg-198
Au-199	2.713E+05 s	Beta	1.368E-03	Hg-199
Au-200	48.4 min	Beta	1.304E-02	Hg-200
Hg-197	65.0 h	POS or EC	7.742E-04	Au-197
Hg-197m	24.0 h	POS or EC (6.00%)	1.789E-03	Au-197m
		IT (94.0%)		Hg-197
Hg-199m	43.0 min	IT	3.160E-03	Hg-199
Hg-203	4.025E+06 s	Beta	1.992E-03	Tl-203
Hg-205	5.50 min	Beta	9.485E-03	Tl-205
Tl-204	3.80 y	POS or EC (2.10%)	1.416E-03	Hg-204
		Beta (97.9%)		Pb-204
Tl-206	4.19 min	Beta	9.034E-03	Pb-206
Tl-207	286. s	Beta	2.937E-03	Pb-207
Tl-208	184. s	Beta	2.353E-02	Pb-208
Tl-209	132. s	Beta	1.662E-02	Pb-209
Pb-204	1.400E+17 y	Alpha	1.541E-02	Hg-200
Pb-205	30.0 10^6 y	POS or EC	2.905E-05	Tl-205
Pb-209	3.30 h	Beta	1.150E-03	Bi-209
Pb-210	7.037E+08 s	Beta	2.317E-04	Bi-210
Pb-211	2.166E+03 s	Beta	2.997E-03	Bi-211
Pb-212	3.830E+04 s	Beta	1.904E-03	Bi-212
Pb-214	1.608E+03 s	Beta	3.189E-03	Bi-214
Bi-208	368. 10^3 y	POS or EC	1.573E-02	Pb-208
Bi-210	4.330E+05 s	Beta	2.306E-03	Po-210
Bi-210m	3.00 10^6 y	Alpha (99.6%)	3.140E-02	Tl-206m
		Beta (0.400%)		Po-210m
Bi-211	128. s	Alpha (99.7%)	3.989E-02	Tl-207
		Beta (0.280%)		Po-211

Table 1B.1 (continued)

Parent	Half-life	Decay mode[a]	Q (W/Ci)	Daughter
Bi-212	3.633E+03 s	Alpha (35.9%)	1.701E-02	Tl-208
		Beta (64.1%)		Po-212
Bi-213	2.739E+03 s	Alpha (2.16%)	4.204E-03	Tl-209
		Beta (97.8%)		Po-213
Bi-214	1.194E+03 s	Alpha (2.100E-02%)	1.282E-02	Tl-210
		Beta (100.%)		Po-214
Po-210	1.196E+07 s	Alpha	3.206E-02	Pb-206
Po-211	0.560 s	Alpha	4.501E-02	Pb-207
Po-211m	25.0 s	Alpha	4.505E-02	Pb-207m
Po-212	3.000E-07 s	Alpha	5.300E-02	Pb-208
Po-213	4.200E-06 s	Alpha	5.061E-02	Pb-209
Po-214	1.643E-04 s	Alpha	4.643E-02	Pb-210
Po-215	1.780E-03 s	Alpha	4.464E-02	Pb-211
Po-216	0.150 s	Alpha	4.094E-02	Pb-212
Po-218	183. s	Alpha (100.%)	3.624E-02	Pb-214
		Beta (2.000E-02%)		At-218
At-217	3.230E-02 s	Alpha	4.268E-02	Bi-213
Rn-218	3.500E-02 s	Alpha	4.304E-02	Po-214
Rn-219	3.96 s	Alpha	4.150E-02	Po-215
Rn-220	55.6 s	Alpha	3.797E-02	Po-216
Rn-222	3.304E+05 s	Alpha	3.314E-02	Po-218
Fr-221	288. s	Alpha	3.860E-02	At-217
Fr-223	1.308E+03 s	Alpha (6.000E-03%)	2.597E-03	At-219
		Beta (100.%)		Ra-223
Ra-222	38.0 s	Alpha	3.960E-02	Rn-218
Ra-223	9.879E+05 s	Alpha	3.561E-02	Rn-219
Ra-224	3.162E+05 s	Alpha	3.432E-02	Rn-220
Ra-225	1.279E+06 s	Beta	7.013E-04	Ac-225
Ra-226	5.049E+10 s	Alpha	2.888E-02	Rn-222
Ra-228	6.70 y	Beta	7.707E-05	Ac-228
Ac-225	8.640E+05 s	Alpha	3.493E-02	Fr-221
Ac-227	6.871E+08 s	Alpha (1.38%)	4.843E-04	Fr-223
		Beta (98.6%)		Th-227
Ac-228	2.207E+04 s	Beta	8.643E-03	Th-228
Th-226	31.0 min	Alpha	3.824E-02	Ra-222
Th-227	1.617E+06 s	Alpha	3.650E-02	Ra-223
Th-228	6.037E+07 s	Alpha	3.271E-02	Ra-224
Th-229	2.316E+11 s	Alpha	3.059E-02	Ra-225
Th-230	2.430E+12 s	Alpha (100.%)	2.830E-02	Ra-226
		SF (5.000E-11%)		fiss. p
Th-231	9.187E+04 s	Beta	5.612E-04	Pa-231
Th-232	4.434E+17 s	Alpha	2.421E-02	Ra-228
Th-233	22.1 min	Beta	2.531E-03	Pa-233
Th-234	2.082E+06 s	Beta to *	4.055E-04	Pa-234m
Pa-231	1.034E+12 s	Alpha (100.%)	3.013E-02	Ac-227
		SF (3.000E-10%)		fiss. p
Pa-232	1.132E+05 s	Beta	6.539E-03	U-232