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VOLUME 1 of 6
UC-70, -71, and -85

Office of Civilian Radioactive Waste Management

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

DECEMBER 1987

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

DISCLAIMER

PREFACE

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

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

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

The above data bases may be ordered using the form printed on the following page. An introductory information diskette can be found inside the back cover of this report. It provides a brief introduction to each of these five PC data bases. For instructions on reading the information diskette, see Section 1.1.4.

v/91

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

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GA Technologies
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E. R. Johnson Associates, Inc.
Maxima Corporation
PAI Corporation
Battelle-Pacific Northwest Laboratory
Savannah River Plant
Westinghouse Advanced Power Systems
Westinghouse Hanford Co.
West Valley Nuclear Services Co.

LIST OF ACRONYMS

AC	Allis Chalmers
ANF	Advanced Nuclear Fuels Corporation
ANL	Argonne National Laboratory
AP	activation products
APSR	axial power shaping rod
ASTM	American Society for Testing Materials
B-C	Battelle-Columbus
B&W	Babcock and Wilcox
BPRA	burnable poison rod assembly
BWR	boiling-water reactor
CC	complexant concentrate
CDB	Characteristics Data Base
CE	Combustion Engineering
CEA	control element assembly
CEU	Consolidated Edison uranium
CFR	Code of Federal Regulations
CH	contact handled
DHLW	defense high-level waste
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
EIA	Energy Information Administration
EIS	environmental impact statement
EPRI	Electric Power Research Institute
FFTF	Fast Flux Test Facility
FIS	Federal Interim Storage
FP	fission products
FSV	Fort St. Vrain
FWMS	Federal Waste Management System
GAPSR	gray axial power shaping rod
GE	General Electric
GTCC	Greater than Class C
HANF	Hanford
HEDL	Hanford Engineering Development Laboratory
HEPA	high-efficiency particulate air
HLW	high-level waste
HTGR	high-temperature gas-cooled reactor
HWVP	Hanford Waste Vitrification Plant
ICPP	Idaho Chemical Processing Plant
IDB	Integrated Data Base
INEL	Idaho National Engineering Laboratory
LANL	Los Alamos National Laboratory
LER	Licensee Event Report
LLW	low-level waste
LWBR	Light-water Breeder Reactor
LWR	light-water reactor
MOX	mixed oxide
MRS	monitored retrievable storage
MSRE	Molten Salt Reactor Experiment
MTIHM	metric tons of initial heavy metal
MTR	Materials Test Reactor
NCAW	neutralized current acid waste

NFA	nonfuel assembly
NFB	nonfuel bearing
NMMSS	Nuclear Materials Management and Safeguards System
NRC	Nuclear Regulatory Commission
NWTSP	National Waste Terminal Storage Program
O/U	oxygen/uranium atom ratio
OCRWM	Office of Civilian Radioactive Waste Management
OFA	optimized fuel assembly
ORA	orifice rod assembly
ORNL	Oak Ridge National Laboratory
PBI	Peach Bottom Unit I
PC	personal computer
PCI	pellet-clad interaction
PFP	plutonium finishing plant
PIE	postirradiation examination
PNL	Pacific Northwest Laboratory
PNS	primary neutron source
PWR	pressurized-water reactor
QA	quality assurance
QC	quality control
RH	remotely handled
RNS	regenerative neutron source
SAS	Statistical Analysis System
SFD	spent fuel disassembly
SNF	spent nuclear fuel
SRL	Savannah River Laboratory
SRP	Savannah River Plant
SS	stainless steel
SST	single-shell tanks
TMI-2	Three Mile Island 2
TRIGA	Training Research Isotopes - General Atomics
TRU	transuranic (waste)
TRUW	transuranic waste
UN	United Nuclear
WAC	waste acceptance criteria
WAPS	Waste Acceptance Preliminary Specification
WE	Westinghouse
WIPP	Waste Isolation Pilot Plant
WVDP	West Valley Demonstration Project

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2D USER'S GUIDE TO THE LWR QUANTITIES DATA BASE

2E PHYSICAL DESCRIPTIONS OF LWR NFA HARDWARE
2F USER'S GUIDE TO THE LWR NFA HARDWARE DATA BASE

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4D SUPPLEMENTAL DATA FOR PEACH BOTTOM 1 SPENT FUEL

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1. SUMMARY

1.1 INTRODUCTION

1.1.1 Objectives

The Office of Civilian Radioactive Waste Management (OCRWM) is responsible for all spent fuels and high-level wastes that will eventually be disposed of in a geologic repository. The purpose of this report, and the information contained in the associated computerized data bases, is to establish the DOE/OCRWM reference characteristics of the radioactive waste materials that may be accepted by DOE for emplacement in the mined geologic disposal system as developed under the Nuclear Waste Policy Act of 1982. Characterization data will be used by OCRWM for planning purposes, trade-off studies, optimization, standardization, and conceptual design within the various geologic repository projects, the transportation program, the MRS and rod consolidation programs, and overall systems integration.

The primary sources of materials for a geologic repository are LWR spent fuel, either intact or consolidated and with associated activated metal, and immobilized high-level waste from West Valley and the defense sites. These are the major sources in terms of both volume and radioactive materials. Other sources are non-LWR spent fuel and miscellaneous wastes. Detailed characterizations are required for the materials in each of these categories. These characterizations include physical, chemical, radiological, and thermal characteristics which, in the latter two cases, must take into account decay as a function of time. In addition, inventories and projected quantities of the various wastes are also included. This information is tabulated in a Characteristics Data Base, of which this document is a major element. The other elements are computerized data bases, which are set up as user-oriented, menu-driven PC data bases written in dBASE-III PLUS. There are presently five of these PC data bases, and others are to be added later.

The Characteristics Data Base will serve as a unified source of information for the characterization of those materials that will (or may) become the responsibility of OCRWM for transport, storage, and final disposal. It will also provide sufficient information to permit the various wastes to be properly classified even if revisions are made in the definitions of HLW, TRU waste, and LLW in the greater-than-C category. It can also be used in the development of waste acceptance criteria.

1.1.2 Report and Data Base Structure

The Characteristics Data Base uses a four-tiered structure: hard-copy reports, user-oriented PC data bases, program-level PC data files, and mainframe computer files. This report is the initial hard-copy report and the appendices contain user's guides for four menu-driven personal computer data bases for LWR fuel, assemblies, hardware, and quantities and one data base for HLW.

The hard-copy reports provide the basic waste characterization descriptions, as well as the figures and drawings that are not easily placed in computerized files. The computerized files contain systematic data too extensive to include in a paper report, such as the radionuclide compositions of each waste for multiple decay times and derived radiologic data.

The user-oriented PC data bases provide detailed information in a menu-driven system and require no computer programming capabilities by the user. Currently five of these data bases are available:

- LWR Radiological Data Base - Contains radionuclide compositions, heat generation rates, curies and other information as a function of spent fuel type, burnup and decay time.
- LWR Assemblies Data Base - contains physical descriptions of intact assemblies and radiological characteristics of spent fuel disassembly hardware.
- High Level Waste Data Base - Contains physical and radiological descriptions of high level waste, as the interim forms and as the immobilized forms.

- LWR NFA Hardware Data Base - Contains physical and radiological descriptions of non-fuel assembly hardware.
- LWR Quantities Data Base - Contains data on discharged fuel, as historical inventories and as projected quantities.

See Sect. 1.1.4 for more information on these data bases.

The program-level PC files are more versatile than the user-oriented files, but their use requires programming skills with dBase III. Special reports and interactive output can be tabulated from these files. An example of an interactive function is to couple a specific assembly type from the LWR Assemblies Data Base with the radiological properties from the LWR Radiological Data Base to obtain the radiological properties of that assembly for any desired burnup or decay time after discharge.

The mainframe computer files are used to generate the above files and some of the hard-copy reports. Their use requires extensive programming skill in SAS, FORTRAN, and other computer languages.

1.1.3 Methodology

1.1.3.1 Data Sources

Other data bases and data sources, both within and outside of DOE, relate to various facets of spent fuel and radioactive waste, each with its own center of focus. For example, extensive data bases are maintained by EIA, PNL, IDB, EPRI, NMMSS, NRC, and the national LLW and TRU waste programs.* The Characteristics Data Base program interacts constructively with these programs, utilizing their files when appropriate and making our data files available to them.

Primary data on HLW are obtained directly from the waste-generators themselves: The West Valley Demonstration Project, the Savannah River Plant (Defense Waste Processing Facility), the Hanford Reservation facilities, and the Idaho National Engineering Laboratory.

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*Acronyms are defined on pp. ix and x.

The manufacturers of nuclear fuel (i.e., the fuel vendors) are the preferred sources of detailed data on their respective fuel assemblies or elements. For this purpose, contracts were initiated with GA Technologies, Westinghouse, Babcock & Wilcox, Combustion Engineering, and Exxon (now the Advanced Nuclear Fuels Corporation); a contract with General Electric is pending.

The Energy Information Administration (EIA) is a primary source of data on LWR spent fuel inventories and projections. Their RW-859 data file provides extensive data obtained directly from the utilities. They also provide longer-term projection data, in cooperation with Pacific Northwest Laboratory.

The Integrated Data Base (IDB) program, also carried out at ORNL, covers in a less-detailed manner all domestic radioactive wastes and spent fuel. The IDB includes TRU waste, low-level waste, remedial action wastes, and mill tailings, in addition to spent fuels and high-level waste.

1.1.3.2 Data Processing

The Characteristics Data Base processes data at three levels: user-oriented PC files, program-level PC files, and mainframe files. The initial data, when received (or generated), are inputted to the mainframe files or the PC program files. Both of these files, through the use of other programming capabilities such as dBase-III, FORTRAN, and SAS, are used as necessary to create suitable PC program files; these are then used to create the PC user-oriented data bases. The overall data flow is shown schematically in Fig. 1.1. Data manipulation is carried out in a three-tiered structure involving mainframe files, PC program files, and PC user-oriented data bases.

1.1.3.3 Radiological Characteristics

The radiological characteristics derive from the presence of radioactive nuclides that are generated in reactors from nuclear fission (fission products), activation of the lighter isotopes

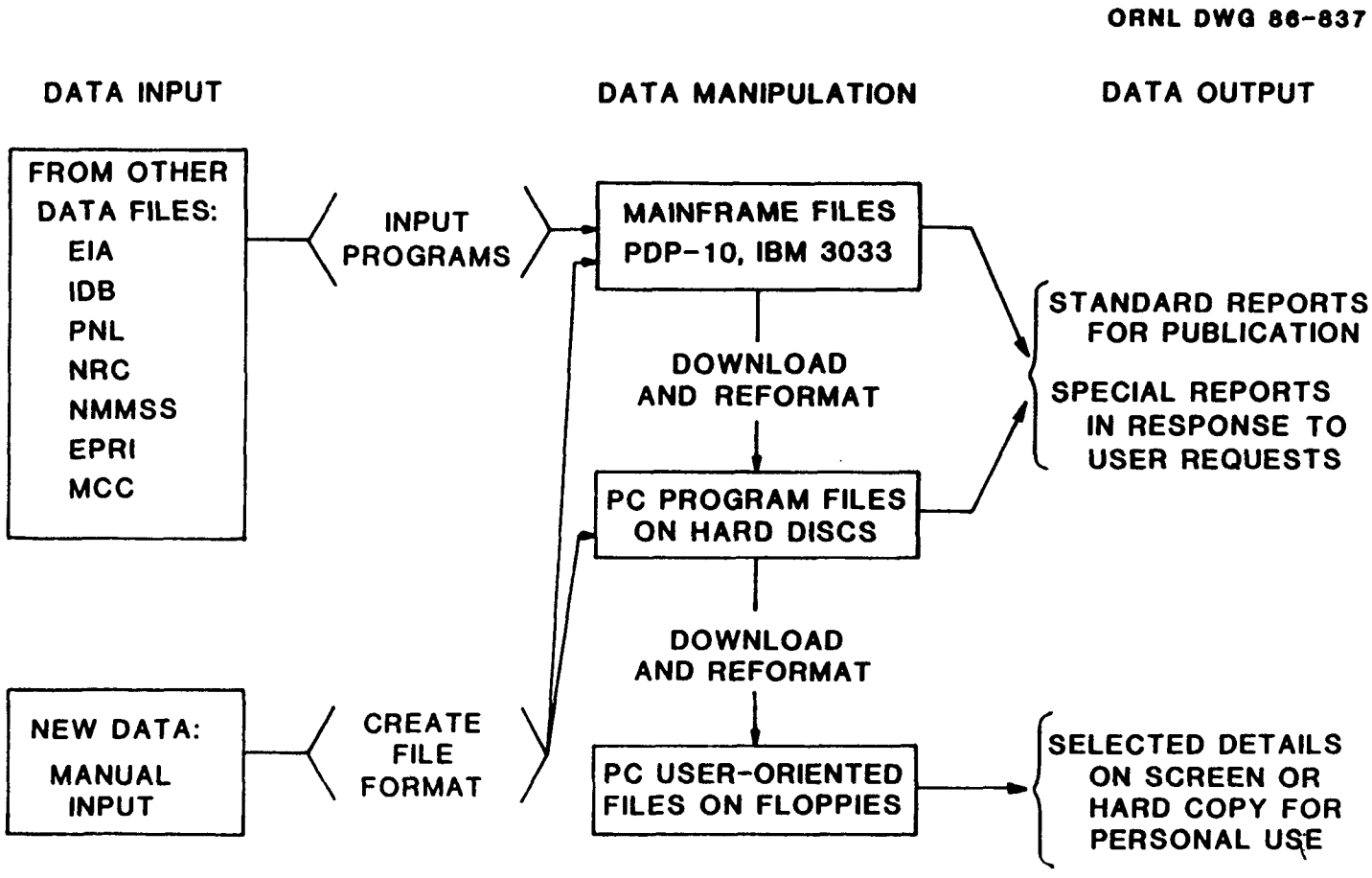

Fig. 1.1. Data processing in the Characterization Data Base.

(activation products), or neutron capture by the heavy metals (actinides). In turn, each of these may undergo further activation, or simply decay to a stable form, in one or more decay steps. Calculation of the quantities generated is a complex process which we carry out using the ORIGEN2 code. Appendix 1A gives a brief overview of ORIGEN2 and a reprint of a technical paper on ORIGEN2. Data output obtained from use of this code includes:

- quantities of each nuclide (grams or gram-atoms);
- radioactivity, total and by nuclide;
- alpha radioactivity, total and by nuclide;
- thermal power, total and by nuclide;
- photon energy spectra, total and by nuclide;
- neutrons from spontaneous fission;
- neutrons from (α,n) reactions; and
- quantity of each element (grams or gram-atoms).

The generation portion of ORIGEN2 requires input data for the specific reactor conditions being modeled. This has been done for PWRs by using both standard and high burnups and for BWRs by using standard burnups. A BWR high-burnup model has just been developed as part of the System Modeling Assessment Task of the Waste System Data and Development Program at ORNL. This assessment task is also undertaking improved verification and validation testing of ORIGEN2. Improved models are also being developed for the calculation of activation products generated outside the immediate reactor core region. As these improved models become available, they will be used to provide improved characteristics data in future updates of this report.

Making ORIGEN2 computations requires several input libraries, such as decay constants (half-lives) and effective cross sections (for the reactor scenario being calculated). These are described briefly in Appendix 1B.

ORIGEN2 can calculate decayed values to any desired time; however, if a time not previously calculated (and stored) is desired, another ORIGEN2 computer run is required. To permit use of the user-oriented files for any desired decay time, we have developed a standard interpolation function (Appendix 1C). This function can also be used to interpolate between different burnup levels, or both time and burnup.

1.1.4 Menu-Driven PC Data Bases

There are five user-oriented, menu-driven PC data bases available at this time. These were described very briefly in Sect. 1.1.2. For detailed descriptions, please see the five user's guides in the appendices. These data bases may be ordered on either floppy disks or Bernoulli cartridges; see page v for instructions. For two of the larger data bases demonstration diskettes are also available. These provide an in-depth overview of the data contents to assist the potential user in deciding if the full data base would be useful.

An information diskette is enclosed with this report, inside the back cover. It provides a very cursory overview to illustrate the menu-driven approach and a few of the available data outputs. This diskette will run on an IBM PC-compatible computer and some version of DOS. To use the diskette insert it in either floppy disk drive and call for that drive (A or B), then type

INFO

and press the enter key. This calls the program up and additional screens are called for by striking any key, except where the instructions on the screen direct selection from a specified set of characters. The display can be on either a monchromatic or a color monitor.

1.1.5 QA Plan/Accuracy and Reliability of Data

This work is being done under the overall requirements of NQA-1, as it applies to data and software. A QA assessment/evaluation was done (QAA 1987) and a QA plan written (QAP 1987). The key elements of this plan revolve around operational procedures. These have, in fact, been factored into this program since its inception, even in the absence of a documented QA plan, and cover these operations:

- Data Input: obtained from the primary sources, and references to identify these sources are provided.
- Data Processing: this is reviewed internally, and the output is then reviewed by the primary sources prior to publication. Only one major computational code, ORIGEN2, is utilized; it has already undergone extensive testing over the past 10 years, and is presently the subject of a formal verification, validation, and benchmarking program.
- Distribution of Hard-Copy Reports: this is controlled by using defined category distribution (from TIC-4500), a published distribution list for additional copies, and a written record of requested copies.
- PC Data Bases and Software: these are programmed in dBASE-III PLUS, which is a thoroughly documented commercial product. A file is kept listing all recipients.
- Future Updating: both hard-copy reports and PC data bases will be identified by date whenever they are updated or revised.

The broad nature of the data encompassed by this program renders it impossible to make a generic statement about the accuracy and reliability of the contained data. A few examples will illustrate this:

- Where a numerical count is made, e.g. the number of discharged LWR fuel assemblies, the count should be 100% accurate and totally reliable.
- Where projections are involved, e.g. of future LWR discharges, accuracy is secondary to reliability, while the latter is a function of both technical aspects (such as cycle time between reloads) and institutional factors, with the latter clearly the overriding factor. This particular situation is handled by utilizing alternative projection bases.

- Where measured quantities are involved, e.g. volume or mass of HLW, both accuracy and reliability depend directly on the original data sources. Again, projections are distinctive from historical data, both for technical reasons (such as borosilicate glass vs tailored ceramic) and institutional factors.
- Where computations are involved, e.g. the calculation of radiological properties using the ORIGEN2 code, two factors are involved - the input data and the calculations themselves. For commercial spent fuel it is safe to assume that both enrichment and burnup are as accurate as the utilities can define these quantities, since neutron economics is a key factor in their operations. ORIGEN2 output is generally taken to be accurate within 5 to 10% on thermal output and for many nuclides, with better accuracy than this on some fission products, but poorer accuracy on some activation products and higher actinides. For non-fuel bearing components (NFBC), both the input data and the computations are less accurate and less reliable, perhaps only within a factor of two. For ORIGEN2 itself, an active program for verification, validation, and benchmarking is underway elsewhere at ORNL, under OCRWM sponsorship. For improved input data on NFBC, experimental work is underway at INEL, PNL, and other sites.

It is an objective of this program to "do no harm" in processing data. All of our primary data come from other sources; none are self-generated. The principal computations are done with ORIGEN2, an accepted nuclide generation and depletion code with its own QA plan. This program has an obligation to (1) not downgrade the data we receive, (2) utilize ORIGEN2 correctly, and (3) provide a review and critique function to our data sources. In support of these principles, numbers are reported in the same units as provided and numbers are not rounded off. Thus, the concept of "significant figures" does not apply to these data in a statistical sense. It should be noted that precision is generally much better than accuracy, otherwise small differences between large numbers could easily become distorted.

1.2 LWR SPENT FUEL (see Sect. 2)

1.2.1 Scope

LWR spent fuel from commercial power reactors is characterized in terms of intact fuel assemblies, spent fuel disassembly (SFD) hardware, defective fuel, special fuel forms, and nonfuel assembly (NFA) hardware. The differences between BWR and PWR spent fuel are sufficient to maintain this distinction throughout. The primary basis for characterization is the assembly type and model, for each of which detailed descriptions are provided. Secondary data are based either on burnup (for the fuel itself), activation of materials of construction (for SFD and NFA hardware), or special handling that may be required (for defective fuel and special fuel forms).

Fuel assemblies are described for each vendor, type, and model. Detailed data and descriptive drawings show the size and location of the various components, the materials of construction, and the mass of each component. Minor constituents and impurities present in the structural materials are identified. The in-core neutron exposure zone of each component was calculated. Each type of assembly is also characterized in terms of inventory-related information, such as the method of manufacture, the date of manufacture, and the reactor in which they were used. For intact assemblies, radiological and thermal data are tabulated and made available based on burnup and reactor type.

The detailed assembly data are coupled with special activation calculations made with ORIGEN2 to estimate the radioactivities of the various SFD and NFA hardware components. The results provide a basis for classifying these components in terms of four LLW categories: A, B, C, and greater than C. For hardware with a greater-than-C radioactivity classification, the radioactivity is also reported as a multiple of Class C. The estimated volumes deriving from these components are calculated.

Fuel performance data and records were reviewed to identify, describe, and categorize various classes of defective fuel. This includes leakers, deformation (bowing, warping, and twisting), visually

observable defects such as fretting or surface corrosion, and any damaged fuel that has been repackaged or encapsulated, such as fuel from Three Mile Island.

1.2.2 Assemblies

Detailed descriptive material was tabulated for 58 specific assembly models (Table 1.1 and 1.2). The data items listed in Table 1.3 were (or are being) collected for each model. These data are then incorporated in "Physical Descriptions of LWR Fuel Assemblies" (see Appendix 2A) and in a user-oriented data base (Appendix 2B). Selected information, for example, the overall physical dimensions of these assemblies, their weights and initial heavy-metal contents, the fuel rod diameters, and the cladding material, can be easily extracted from this data base. Other information can also be extracted, as desired. With minimal programming effort, additional assembly models and new data fields can be added if the need arises.

1.2.3 Spent Fuel Inventory

Inventories and projections are provided by the EIA, IDB, and PNL data bases, and are incorporated in the LWR Quantities Data Base (Appendix 2D). Radiological characteristics, on an MTIHM basis, are calculated using ORIGEN2 and are tabulated in the LWR Radiological Data Base (Appendix 2C). Spent fuel is characterized in terms of reactor type (PWR or BWR), burnup (from 5 to 60 GWd/MT for PWRs and 5 to 40 GWd/MT for BWRs), and decay times (from 1 to 1,000,000 years, in 24 or 38 increments, depending on the data base involved). The types of radiological data provided were listed earlier, in Sect. 1.1.3.3.

The inventory of spent fuel is primarily a function of the number of nuclear reactors in operation and how long they have been operating. Other factors also affect the amount of spent fuel discharged, for example, the on-stream factor and the burnup (service lifetime). The 1987 figures from EIA, which issues annual projections of installed nuclear generating capacity, provide three scenarios and project, for the year 2020:

Table 1.1.

CURRENTLY LISTED ASSEMBLIES

PRESSURIZED WATER REACTOR ASSEMBLIES

Assembly Manufacturer	Array Size	Version
Babcock & Wilcox	14 X 14	
Babcock & Wilcox	15 X 15	Mark B
Babcock & Wilcox	15 X 15	St.Stl.
Babcock & Wilcox	15 X 15	Mark BZ
Babcock & Wilcox	17 X 17	Mark C
Combustion Engineering	14 X 14	Std
Combustion Engineering	14 X 14	Ft.Cal.
Combustion Engineering	15 X 15	Palis.
Combustion Engineering	16 X 16	Onofre
Combustion Engineering	16 X 16	Lucie 2
Combustion Engineering	16 X 16	ANO2
Combustion Engineering	16 X 16	SYS80
Combustion Engineering	15 X 16	Yankee
Exxon/ANF	14 X 14	WE
Exxon/ANF	14 X 14	CE
Exxon/ANF	14 X 14	Top Rod
Exxon/ANF	14 X 14	Ft.Cal.
Exxon/ANF	15 X 15	WE
Exxon/ANF	15 X 15	CE
Exxon/ANF	15 X 16	WE
Exxon/ANF	17 X 17	WE
Westinghouse	13 X 13	
Westinghouse	14 X 14	Std/ZCA
Westinghouse	14 X 14	OFA
Westinghouse	14 X 14	Std/ZCB
Westinghouse	14 X 14	Std/SC
Westinghouse	14 X 14	Model C
Westinghouse	15 X 15	Std/ZC
Westinghouse	15 X 15	OFA
Westinghouse	15 X 15	Std/SC
Westinghouse	15 X 16	
Westinghouse	17 X 17	Std
Westinghouse	17 X 17	OFA
Westinghouse	17 X 17	Vant 5
Westinghouse	17 X 17	XLR

Table 1.2.

CURRENTLY LISTED ASSEMBLIES

BOILING WATER REACTOR ASSEMBLIES

Assembly Manufacturer	Array Size	Version
Allis Chalmers	10 X 10	
Exxon/ANF	6 X 6	GE
Exxon/ANF	6 X 6	HUM.BAY
Exxon/ANF	7 X 7	GE
Exxon/ANF	8 X 8	JP-3
Exxon/ANF	8 X 8	JP-4,5
Exxon/ANF	9 X 9	JP-3
Exxon/ANF	9 X 9	JP-4,5
Exxon/ANF	9 X 9	BRP
Exxon/ANF	10 X 10	AC
Exxon/ANF	11 X 11	GE
General Electric	6 X 6	DRES-1
General Electric	6 X 6	HUM.BAY
General Electric	7 X 7	/2,3:V1
General Electric	7 X 7	/2,3:V2
General Electric	7 X 7	/4,5
General Electric	7 X 7	HUM.BAY
General Electric	8 X 8	/2,3
General Electric	8 X 8	/4,5:V1
General Electric	8 X 8	/4,5:V2
General Electric	9 X 9	BRP
General Electric	11 X 11	BRP
Westinghouse	8 X 8	QUAD+

Table 1.3. Technical data for each fuel assembly model

Fuel assembly
- Designation
- Transverse dimension
- Overall length
- Total weight
- Weight heavy metal
- Number of fuel rods
- Rod pitch

Requirements for disassembly
- Cutting required
- Mechanical disassembly
- Single pins replaceable
- Underwater consolidation
- Drawing showing main features
- Disassembly Drawing No.

Fuel Rods
- Diameter
- Length
- Clad material
- Clad thickness/weight
- Spring material/weight
- Heavy-metal content, U/other
- Burnable poison/weight
- Fabrication parameters
- Initial rod pressurization

Assembly hardware
- Incore hardware
  - Grids, spacers, guide tubes
  - Material/weight
- Top end fittings
  - Nozzles, springs, material/weight
- Bottom end fittings
  - Other peripheral or special hardware, channels, flux wires, etc.

Inventory information
- Number of assemblies fabricated
- Serial numbers
- Batch sequences/enrichments
- Reactor customers
- Initial load/reload number
- Shipment date

Fuel performance
- Enrichment (range)
- Maximum burnup

- No New Orders Case — 51 GW(e)
- Lower Reference Case — 130 GW(e)
- Upper Reference Case — 199 GW(e)

As widely as these cases and projections differ, the projected cumulative spent fuel discharged by 2020 differs by less than 25 percent because most of the additional capacity (or the shutdown capacity) occurs late in time:

- No New Orders Case — 77,800 MTIHM;
- Lower Reference Case — 87,500 MTIHM;
- Upper Reference Case — 98,300 MTIHM; and

Based on the EIA Upper Reference Case, the quantities of spent fuel discharged in 2020 will be as follows:

	Number of Assemblies		Weight (MTIHM)	
Reactor type	Annual rate	Cumulative	Annual rate	Cumulative
BWR	6,600	195,000	1,200	35,000
PWR	6,200	149,000	2,600	63,300
Totals	12,800	344,000	3,800	98,300

1.2.4 Defective Fuel

This category, although not yet rigorously defined, is of considerable interest because these fuels may require special handling. They are expected to contribute only a small fraction of the total. Examination of the major data sources for this category, in light of existing classification schemes, indicates that the 10 CFR 961-based approach (with three failed fuel categories) can provide a workable basis.

Defects generally result from waterside corrosion or crud buildup, pellet-clad interaction (PCI), radiation-induced stressing, vibration-or debris-caused physical damage in-core, and mechanical damage during out-of-core handling. These defects can cause leaks, deformation of rods

and assemblies, or even breakage of rods, although the latter is now historical except for major reactor malfunctions. On occasion, a utility may seal a leaker (or broken rod pieces) into another tube (encapsulation).

The poolside test methods used on spent fuel rods and assemblies include:

- visual examination,
- gamma scan,
- sipping,
- dimensional measurements,
- eddy current test, and
- ultrasonic testing.

Of these, ultrasonic testing appears to be the best approach for identifying leakers via wholesale examination.

Available data are difficult to analyze statistically because of underlying uncertainties; however, it is clear that defects have decreased markedly during the past 15 years. Methods have been developed to deal with radiation-induced elongation and bowing. Improvements in fuel fabrication and in reactor operation and water chemistry have greatly reduced the number of leakers. Current operations generally achieve rod failure rates of 0.01 to 0.02%. Those assemblies containing leakers have an average of about two failed rods per assembly. Approximately 1 to 2% of the assemblies contain one or more failed fuel rods.

1.2.5 Special Fuel Forms

This category is for LWR fuels that are distinctive in some special way and, therefore, may require special handling. This could include fuel rods consolidated at the reactor site; fuel rods disassembled for testing or postirradiation examination (PIE); fuel rods fabricated with nonstandard cladding, of nonstandard dimensions, or with a nonstandard fuel form (such as Shippingport); and grossly damaged fuel such as that from TMI-2. Deformed assemblies, which might require special packaging, might also be included.

1.2.6 Spent Fuel Disassembly (SFD) Hardware

This data base characterizes specific non-fuel hardware items which will be by-products of spent fuel disassembly and consolidation (Appendix 2B). This hardware contains only activation products (no fission products or actinides unless contaminated by leakers or during handling). Some of this hardware is expected to qualify as low-level waste Class C or, at worst, greater-than-C (within the upper limits for greater-than-C, presently assumed to be 30 times the Class C limits). To characterize this material requires the following information:

- the composition of the alloy, including trace impurities,
- the neutron flux zone in which exposed, and
- the burnup of the spent fuel.

Seven primary materials of construction are employed in fuel assembly fabrication (disregarding two high-cobalt alloys, Stellite-3 and Haynes-25, used for cruciform bearings):

- Zircaloy-2,
- Zircaloy-4,
- Stainless steel-302,
- Stainless steel-304,
- Inconel-718,
- Inconel X-750, and
- Nicrobraze 50.

The near-core neutron fluxes and the effective cross sections of key elements comprising the above alloys were modeled for four axial zones, in both PWRs and BWRs:

- top end plate region,
- gas plenum region,
- core zone, and
- bottom end plate region.

The flux decreases significantly in the two zones adjacent to the core zone and falls off drastically beyond that. The effective cross sections outside the core zone increase up to 570%, depending on the element (Co, Ni, Nb, or N), the zone, and the reactor type. This increase

is presumably due to resonance and a higher fraction of thermalized neutrons outside the core zone.

To simplify the data base, only two burnups for each reactor type were used:

- standard (27.5 GWd/MT for BWR; 33 GWd/MT for PWR); and
- high (40 GWd/MT for BWR; 60 GWd/MT for PWR).

These burnups will provide limiting values for activation. Additional burnups can easily be calculated, should there be interest. As it is, all possible combinations of materials, neutron zone, and reactor/burnup total 112; however, in practice, the needed number is less than half of that because not all alloys are used in all zones of both reactor types. For example, Zircaloy-2 is used in BWRs, and Zircaloy-4 is used in PWRs.

By combining assembly data on materials of construction, weight of each component, and relative location, it is possible to calculate the radioactivity (and thermal power, if desired) of each SFD hardware component. This value can then be compared with the Class C limit and a factor calculated. Examples of this are given in Sect. 2.7.

1.2.7 Nonfuel Assembly (NFA) Hardware

This data base is similar to the Spent Fuel Disassembly (SFD) Hardware Data Base in many key aspects: activation products are the primary source of radioactivity (neutron sources providing the one exception); the materials of construction are virtually identical to SFD hardware; the degree of activation depends on the neutron zone where exposure occurred and the amounts of trace impurities. The major factors which distinguish these two classes of hardware are: NFA hardware is not an integral part of an assembly (although these components are sometimes stored in assemblies in the pool), and the in-core exposure cycles are usually longer than assembly cycles, sometimes much longer.

Physical descriptions of NFA hardware are given in Appendix 2E. The user's guide for the LWR NFA Hardware Data Base is Appendix 2F.

1.3 HIGH-LEVEL WASTE (see Sect. 3)

1.3.1 Scope

This includes HLW from domestic fuel reprocessing plants, both commercial and defense-related. The ultimate waste outputs are the individual canisters of solidified HLW, which are characterized by site (West Valley, SRP, Hanford, Idaho) and, ultimately, by time and specific composition for each site. Specific detailed compositions generally cannot be assigned yet because detailed schedules have not been defined. However, certain broad categories can be defined in a relatively straightforward manner for characterization, such as alkaline or acidic wastes and some tank farm groups (e.g., double-shell tanks).

The HLW characterization data include descriptions of the canisters, chemical and isotopic compositions, and age, from which radioactivity and thermal power are calculated. Base-line solidification processes are identified for each site in order to calculate the projected output of HLW canisters, plus any associated transuranic (TRU) waste and LLW in the greater-than-C category for commercial sites.

The West Valley and Savannah River HLW are generally quite similar, and both will be vitrified for final immobilization. The Hanford HLW are distinctive because the cesium and strontium have been stripped out (which concentrates much of the fission product activity in the CsCl and SrF_2 capsules). This practice has now been discontinued. The Idaho HLW are unique because they are not neutralized and are subsequently calcined to an oxide-type ash, which may be more amenable to conversion to a dense ceramic rather than a glass form.

The detailed HLW data are available in a user-oriented PC data base, structured similarly to those for spent fuel and LWR assemblies. The HLW PC Data Base, which is described in Appendix 3C, covers both the immobilized waste in canisters and the interim waste forms.

Table 1.4 summarizes HLW data for all four source sites, both as the interim forms and the immobilized forms in canisters.

Table 1.4. Summary data for high-level waste

	WVDP	DWPF	Hanford[a]	INEL
Interim forms and volumes, m^3				
1986				
Liquid	2,145	72,900	28,300	6,500
Sludge	170	13,800	46,000	-
Salt cake	-	41,200	93,000	-
Slurry	-	-	65,000	-
Calcine	-	-	-	3,000
2020				
Liquid	-	39,440	7,200	1,700
Sludge	-	1,160	46,000	-
Salt cake	-	24,200	93,000	-
Slurry	-	-	52	-
Calcine	-	-	-	10,300[a]
Immobilized forms[b]				
No. of canisters in 2020	275	6,810	1,860	8,800
Kilocuries/canister[c]	125	234	416	143
Watts/canister[c]	380	710	1,160	450
Future annual rate, canisters/year	- [d]	92[e]	- [f]	1,000[g]

[a]At Hanford, the interim forms listed as liquid, sludge, and salt cake represent the total contents of single-shell tanks; slurry represents the contents of double-shell tanks. Hanford's current reference plan is to vitrify only the contents of the double-shell tanks; however, a large portion of the liquid now in single-shell tanks will be transferred to double-shell tanks and vitrified.

[b]Borosilicate glass for WVDP, DWPF, and Hanford; high-density ceramic for INEL. Canisters are assumed to be 2 ft in diameter by 10 ft long.

[c]At the time of immobilization. Maximum values are shown; many canisters will be much lower.

[d]A 2-year campaign scheduled for 1990-91.

[e]Projected for the year 2020.

[f]The backlog will be worked off by 2010, as reported by Hanford. The rate after 2010 depends on future plans for reprocessing.

[g]This includes 650 from then-current operations plus 350 from the backlog, which is projected to be worked off at some later time.

1.3.2 West Valley Demonstration Project

The interim form of this HLW is primarily neutralized liquid and sludge from Purex-type reprocessing. There is also some acidic liquid from Thorex-type reprocessing. These two source streams will be combined prior to vitrification into borosilicate glass. Vitrification is scheduled for the 1990-1991 period.

1.3.3 Defense Waste Processing Facility

The interim waste form at the Savannah River Plant is neutralized liquid and sludge, plus a large amount of salt cake. The liquid and salt cake will be processed to precipitate the soluble cesium, which will be combined with the sludge for vitrification into borosilicate glass. The decontaminated liquid and salt cake will be converted to saltcrete, a low-level waste form. Vitrification of the HLW is scheduled to begin in 1990.

1.3.4 Hanford Operations

The interim waste form is neutralized reprocessing liquor and includes liquid, sludge, salt cake, and slurry. During past years, much of the ^{90}Sr and ^{137}Cs were removed, solidified as SrF_2 and CsCl, and sealed into capsules for use as radiation sources. These capsules incorporate a large amount of radioactivity. The Hanford Waste Vitrification Plant (HWVP) is now in preliminary conceptual design and is scheduled to start producing canisters of borosilicate glass waste in 1996.

1.3.5 Idaho National Engineering Laboratory

The Idaho Chemical Processing Plant produces a distinctive waste form in that the acidic liquid waste resulting from fuel reprocessing is calcined directly to an oxide-type granular calcine. The nitrates are destroyed in the process. In the Fluorinel process, fluoride is first converted to CaF_2, in order to control corrosion and convert the fluoride to a non-hazardous material. The calcine is stored underground in concrete vaults and will eventually be immobilized for final disposal. The calcine contains a large fraction of Al_2O_3, ZrO_2, and CaF_2

from dissolution of the fuel. Should vitrification be selected, a relatively large volume of borosilicate glass would be produced; therefore, other alternatives are being considered. One of these, a ceramic based on CaF_2 and ZrO_2, would have about 40% the volume of glass. The canister production rates for INEL in this report are based on the ceramic form.

1.4 NON-LWR SPENT FUELS (see Sect. 4)

1.4.1 Scope

This category includes spent fuels from research, test, and experimental reactors as well as HTGRs. The various fuel types include carbide-based material in graphite elements, uranium-zirconium hydride, U-Al alloy plate-type, UO_2-polyethylene, U-Mo alloy, aqueous liquid fuel, solidified fluoride salts, sodium-bonded metal, and others. These fuels embrace the spectrum of enrichments, and those which are highly enriched require attention to criticality and safeguards. Where reasonable to do so, they will be reprocessed (at SRP or INEL); in many cases, however, reprocessing will be difficult because of their unique chemical form or content. Characterization is done in terms of fuel element descriptions, quantities, and burnup, from which radiological and thermal properties can be calculated. The fuel element descriptions include physical dimensions and descriptions, chemical compositions, and isotopic enrichments. A summary of these non-LWR fuels is given in Table 1.5.

1.4.2 Fort St. Vrain Reactor

This HTGR reactor has been in operation since 1979, but functioned at reduced power during the earlier years (due to a core vibration problem that has since been resolved). The fuel elements are large graphite blocks, in the shape of hexagonal prisms, containing uranium and thorium carbide microspheres inside a protective coating. Reprocessing of these blocks, which are now being stored at the INEL in an engineered surface structure, is not planned at this time. The quantities shown in Table 1.5 include the graphite matrix material of the fuel elements.

1.4.3 Peach Bottom I Reactor

This HTGR reactor was operated from 1966 to 1974 with fuel elements in the shape of long, slender prisms or cylinders. Two cores were

Table 1.5. Summary of non-LWR spent fuels

Reactor or site	Estimated quantities		
	1987	Annual rate	2020
HTGR Reactors			
Fort St. Vrain (elements)	725	80[a]	3936[b]
Peach Bottom I			
Core I (elements)	804	0	804
Core II (elements)	804	0	804
Research and Test Reactors[c]			
MTR Plate	-	-	20,000[d]
TRIGA	-	-	4,500
UO_2/Polyethylene	-	-	87
PULSTAR	-	-	971
FFTF (assemblies)	170	30-45	677[e]
Others	f	f	f
Miscellaneous Fuels[g]			
ANL	311		
Babcock & Wilcox	54		
Battelle-Columbus	1505		
Battelle-PNL	2251		
HEDL	70[h]		
INEL	38,060[i]		
LANL	127		
ORNL	1276		
SRP	19,020		

[a]Assuming an average operating factor of 35% of full power.

[b]Assuming 7 more reloads of 1/6 core each, plus final discharge of full core.

[c]Total through 2020, including fuels in reactors at that time. Quantities shown are numbers of individual fuel elements, except for the FFTF.

[d]Will be reprocessed and disposed of as defense HLW.

[e]Through year 2003; does not include final core discharge.

[f]Not determined yet.

[g]Reported as kg of heavy metal (U plus Pu plus Th).

[h]Includes some FFTF and TRIGA fuels.

[i]Not including Shippingport LWBR fuel (770 kg U, mostly U-233, and 47,208 kg Th), 17 Turkey Point 3 assemblies and 69 VEPCO assemblies being used for dry consolidation testing, HTGR fuel, and some Pulstar and TRIGA fuel.

discharged: Core I, which is stored in underground dry wells at the INEL, and Core II, which is stored in the facility with Fort St. Vrain spent fuel. The quantities in Table 1.5 include the graphite matrix of the fuel elements.

1.4.4 Research and Test Reactor Fuel

These fuels are categorized into seven basic types that are employed in reactors used at universities or educational facilities, privately owned research and development (R&D) facilities, DOE-owned laboratories, and government-owned (non-DOE) laboratories. The number of reactors in each category is given in Table 1.6. Most of them are either MTR-plate type or hydride-fueled TRIGA reactors. The existing and estimated future quantities of these fuels is given in Section 4 of this report, along with their physical and chemical descriptions.

1.4.5 Miscellaneous Fuels

This category includes a variety of fuels from a wide assortment of reactors. Most of these are at DOE-owned national laboratories; small amounts are at Babcock & Wilcox facilities in Lynchburg and the Battelle-Columbus laboratories. Table 1.7 summarizes the amounts of contained uranium, plutonium, and thorium at each site. A detailed description of the various fuel elements, their chemical form, and cladding materials is given in Sect. 4 of this report.

Of the total quantities listed in Table 1.7 for Idaho, some is sodium-bonded fuel from the Fermi blanket fuel. These may be unacceptable for emplacement in a repository because of the chemically reactive metal. If this is the case, removal of the sodium or NaK might require decladding, in which case these fuels could simply be reprocessed.

Table 1.6. Number of research and test reactors in each fuel type category

Fuel type	University/ educational	Private research and test	Government-owned (DOE)	Government-owned (non DOE)
MTR-plate type, U-Al alloy, high enrichment	19	4	16	1
TRIGA (U-ZrH_2 fuel)	18	5	2	3
UO_2-polyethylene disks or blocks	8	0	0	0
PULSTAR and other low-enriched pin type	3	2	1	0
Liquid fuels (aqueous solutions)	2	0	1	0
U-Mo alloy, high-enriched (93.2%)	0	0	4	2
FFTF (UO_2- PuO_2)	0	0	1	0
Miscellaneous	0	0	27	0
	50	11	52	6

Table 1.7. Inventory of other fuels (as of December 31, 1986)

Storage site	Total candidate materials (kg)	Uranium content (kg)			Total plutonium content (kg)	Total thorium content (kg)
		Total	^{235}U	^{233}U		
Argonne National Laboratory West Idaho Falls, ID	311	302	20		9.00	
Babcock & Wilcox, Lynchburg, VA	54	53	1.2		0.38	
Battelle Memorial Institute, Columbus, OH	1,505	1,492	12		13.12	
Hanford Engineering Development Laboratory	70	60	10.4		10.20	
Idaho Chemical Processing Plant - INEL	136,016	77,790	1,330	862	251.68	58,000
Los Alamos National Laboratory Los Alamos, NM	127	97	54	0.13	30.97	
Oak Ridge National Laboratory Oak Ridge, TN	1,276	1,258	804	280	0.80	17
Pacific Northwest Laboratory Richland, WA	2,251	2,218	17.9		26.77	7
Savannah River Plant Aiken, SC	19,020	10,330	746	31	42.31	8,648
Total	160,631	93,600	2,995	1,174	385.2	66,645

1.5 MISCELLANEOUS WASTES (see Sect. 5)

1.5.1 Scope

These wastes are neither spent fuel nor conventional high-level waste (as presently defined) but may not be appropriate for shallow-land burial for various reasons. Although most of them would probably be suitable for intermediate-depth disposal or greater confinement disposal, the absence of such facilities may destine these materials for a geologic repository. The disposal requirements for these wastes has not yet been defined; hence, their status remains undefined.

1.5.2 OCRWM-Generated Wastes

Operation of the Civilian Radioactive Waste Management System will result in the generation of radioactive wastes from a number of operations, including spent fuel transportation, packaging, and consolidation. All of these are projected wastes, since none of these operations are being carried out. There is, however, some experience in similar areas, and a design study of dry rod consolidation has been made. Indications are that all of these operations wil generate LLW, but only consolidation will generate TRU waste or GTCC waste. Whether dry rod consolidation is done at an MRS or at the repository, the resulting waste will be about the same. The TRU and/or GTCC waste has been estimated at 60 to 260 m^3 per year, depending on the assumptions made regarding useage and handling of HEPA filters. However, if dry consolidation at a central facility is not done, the HEPA filter portion of this waste stream will not be produced, since reactor site consolidation, if done at all, will be done under water.

1.5.3 Commercial TRU Waste

This category of waste is generally characterized by relatively low radioactivity levels but contains enough actinides to be classed as TRU waste. Commercial sources, other than reprocessing, include decommissioning of mixed oxide (uranium plus plutonium) fuel fabrication facilities and the West Valley Plant, major core disruptive incidents

such as TMI-2 (which cause contamination by the release of TRU materials), other abnormal reactor operations, and industrial sources involving transuranics such as ^{241}Am and ^{252}Cf.

The TRU waste generated by the decommissioning of the West Valley Plant at West Valley, New York, has been estimated at 300 m^3. This material is mainly spent resins and filters. It is a mixture of remotely handled (RH) and contact-handled (CH) TRU waste.

A number of nuclear-related facilities will also require decommissioning in the future, with expected generation of TRU waste. These are mainly facilities which have handled plutonium, such as mixed oxide fuel fabrication plants.

Reactor operations sometimes have abnormalities that lead to production of TRU wastes, as at the Oyster Creek reactor and at TMI-2. It is estimated that there are about 100 m^3 of this material at present, with projected future average production rates of 10 to 30 m^3/year.

1.5.4 Reactor Decommissioning

Decommissioning of LWR reactors gives rise to activated metal components from inside the reactor. The degree of activation of these components and the pressure vessel itself has been calculated, using radial flux models extending beyond the core region. Based on these calculations, only the PWR core shroud exceeds the Class C limit, by a narrow margin, while BWR shroud is within the limit. In practice, depending on the actual exposure received and the actual activation of nickel and niobium, either shroud might be Class C or GTCC. The core barrel is calculated to be well below the Class C limit, and also the pressure vessel. The average volume of a PWR core shroud is estimated at 11 m^3, assuming that packaging is 15% efficient (i.e., 85% void volume), and a BWR core shroud is about 47 m^3, under the same assumption.

1.5.5 Radioisotope Capsules

This category includes ^{90}Sr, ^{137}Cs, ^{60}Co, ^{252}Cf, and possibly others. The major potential contributors are the ^{90}Sr and ^{137}Cs capsules, because of their large number: 640 and 1576, respectively. For

both of these, the short-lived daughter nuclides, ^{90}Y and ^{137m}Ba, essentially double the curie content and contribute over three-fourths of the thermal power. It has been estimated that four ^{90}Sr or ^{137}Cs capsules could be placed in a HLW-sized canister, which leads to a total of about 550 canisters. Aging these capsules would, of course, allow a higher loading per canister.

Capsules of ^{60}Co may be potential candidates for a repository but have not yet been reviewed in this light. The relatively short half-life of this nuclide (5.3 years) makes decay time a more promising possibility for dealing with disposal of this nuclide as LLW.

Neutron sources of ^{252}Cf are being used for a variety of applications, which can be categorized for our purposes as industrial, reactor start-up, and medical. The industrial applications capsules are generally massive neutron sources that are used for neutron radiography and activation analyses. These are returned to the DOE supplier, either for reuse or for recovery of ^{248}Cm which has grown in from alpha decay of the ^{252}Cf. The sources which are used for reactor startup stay with the reactors and "burn out" within a few refueling cycles. Thus, they are part of LWR (or HTGR) wastes. The medical applications sources are usually very small — too small to justify processing for recovery of the ^{248}Cm. These are, therefore, candidates for disposal as TRU waste.

1.5.6 Routine Reactor Operations

Routine operation of LWR reactors leads to small quantities of GTCC wastes: These materials are largely filter sludge (from BWRs) or evaporator bottoms (from PWRs). On average, about 3 m^3/GW(e)-yr has been generated, based on historical data.

1.5.7 Summary of Miscellaneous Wastes

Table 1.8 summarizes the estimated projected volumes (in m^3) in the year 2020 and the estimated annual rate at that time. If these are disposed of in HLW-type canisters, 2 ft in diameter by 10 or 12 ft long, one canister could hold up to 1 m^3.

Table 1.8. Projected volumes of miscellaneous wastes[a]

	Estimated total in 2020 (m^3)	Est. annual rate in 2020 (m^3)
OCRWM-generated TRU waste	TBD[b]	60-260[c]
Commercial TRU waste		
West Valley decommissioning	300	0
Other decommissioning	680	TBD
Abnormal reactor operations	70-200[d]	10-30
Industrial/institutional	TBD	10-40
Reactor decommissioning	1560[e]	29[f]
Radioisotope capsules	500[g]	0
Routine reactor operations[h]	TBD	150-600
Totals	3110-3240+	276-976+

[a]Data are given in m^3. One 2-ft by 12-ft canister holds about 1 m^3. "TBD" means to be determined.

[b]Depends on startup date for these facilities.

[c]From dry rod consolidation. The upper limit is a conservative (high) estimate of HEPA filter usage.

[d]Quantity estimated from two abnormal reactor operations (at Oyster Creek and TMI-2).

[e]Assumes 65 have been decommissioned.

[f]Assumes 2 per year, 1 PWR(11 m^3) and 1 BWR (47 m^3).

[g]Assumes that 90% of existing capsules are packaged in canisters by 1995; later packaging would result in fewer canisters because of the decreased thermal output per capsule.

[h]Based on an estimated quantity of 3 m^3 per GW(e)-yr being GTCC, and an EIA projection of 50 to 200 GW(e) installed capacity in 2020.

2. LWR SPENT FUEL

2.1 INTRODUCTION

In the absence of domestic reprocessing of commercial spent fuel, LWR spent fuel will be the predominant source of radioactivity and thermal load to geological repositories. This chapter characterizes intact spent fuel three ways and describes four other categories of wastes associated with LWR spent fuel. Intact spent fuel is characterized in terms of physical descriptions, quantitative information, and radiological properties; the other wastes discussed are defective fuel, special fuel forms, spent fuel disassembly hardware, and nonfuel assembly hardware.

Physical descriptions are presented in Section 2.2. Individual assembly types are grouped together by similar design characteristics. The similarities and major differences are described. The LWR Assemblies Data Base contains detailed physical description data. Physical Description Reports containing these data are given in Appendix 2A, Physical Descriptions of LWR Fuel Assemblies. Appendix 2B is the user's guide to the LWR Assemblies Data Base.

Quantitative information is presented three ways in Section 2.3 -- a broad overview, a reactor- and assembly type-specific basis for historical inventories, and a reactor-specific basis for projections. The LWR Quantities Data Base contains this detailed reactor- and assembly type-specific information. Appendix 2D is the user's guide to this data base.

Radiological properties of intact spent fuel are presented in Section 2.4. Summary information on the isotopes that contribute most to the radioactivity, thermal output, neutron emission, and photon spectra from spent fuel is given, as well as changes in the most significant isotopes with respect to decay time. In-depth radiological properties of intact spent fuel are available through the LWR Radiological Data Base. Appendix 2C is the user's guide to this data base.

Defective fuel is discussed in Section 2.5. Defective fuel is a subset of the total inventory of spent fuel but represents a category that may require special handling. A scheme for the classification of defective fuel is introduced, and types of fuel defects are described. Inspection methods for identifying defects and a statistical categorization of defects are presented.

Special fuel forms are discussed in Section 2.6. Special fuel forms include disassembled or consolidated fuel, nonstandard fuel, and uniquely degraded fuel.

If fuel rods from spent fuel are consolidated, spent fuel disassembly (SFD) hardware is a concern. Section 2.7 discusses the quantities of SFD hardware associated with particular assembly types and the radiological properties of the hardware. The LWR Assemblies Data Base provides detailed radiological characterization of SFD hardware. Appendix 2B is the user's guide to this data base.

Nonfuel assembly (NFA) hardware includes control elements, neutron poison, neutron sources, BWR fuel channels, in-core instrumentation, and orifice rods. This hardware is described in Section 2.8. It is also the subject of the LWR NFA Hardware Data Base. Appendixes 2E and 2F are Physical Descriptions of Nonfuel Assembly Hardware and User's Guide to the LWR NFA Hardware Data Base, respectively.

2.2 ASSEMBLY DESCRIPTIONS

2.2.1 Overview

A light-water reactor (LWR) fuel rod consists essentially of a stack of uranium oxide (UO_2) pellets encapsulated within a metal tube that is sealed on both ends. Early versions of LWR fuel rods used stainless steel for the tubing and sealed the tubes without regard to the gases enclosed or pressurization. As experience with LWR's has grown, changes in fuel rods have been dictated. Atmospheric gases (primarily nitrogen) are evacuated prior to sealing to reduce the production of 14C inside the fuel rod. Fuel rods are prepressurized with helium to reduce potential fuel rod cladding collapse in the plenum (unfueled) region. Zircaloy-2 and Zircaloy-4 have replaced stainless steel as the cladding material for most fuel rods because of their low neutron absorption cross sections and because of their improved resistance to localized corrosion.

Fuel assemblies are constructed from a number of individual fuel rods arranged together, generally in square arrays. These arrays have been of many different sizes. Pressurized-water reactor (PWR) designs have 13 x 14, 14 x 14, 15 x 15, 15 x 16, 16 x 16, and 17 x 17 fuel rod arrays. Boiling-water reactor (BWR) designs have had 6 x 6, 7 x 7, 8 x 8, 9 x 9, 10 x 10, and 11 x 11 fuel rod arrays. Several of these array configurations have had very limited use (one reactor only), while others have been used much more widely. Some of the older designs have all been reprocessed at West Valley, and will be disposed of as commercial, high-level waste.

In this section, a brief description of the major design models of LWR fuel assemblies in existence in the United States is given, followed by a description of the differences between versions of these models. Some manufacturers have made reactor reload fuel using the designs of other manufacturers. These reload versions are listed under the design of the original fuel, not by the manufacturer of the reload version.

Detailed descriptions of the different assembly types are given in Appendix 2A, Physical Descriptions of LWR Fuel Assemblies. This appendix is a listing of the Physical Description Reports from the LWR

Assemblies Data Base. The data contained in these Physical Description Reports were obtained via subcontracts with Advanced Nuclear Fuels, Babcock & Wilcox, Combustion Engineering, and Westinghouse. The reports submitted by the vendors are listed in the references to this section and are not referred to throughout the text. All dimensions and measurements are for unirradiated fuel. Table 2.2.1 is a sample Physical Description Report (for a Babcock & Wilcox 15 x 15 Mark BZ fuel assembly). The user's guide is included with this report as Appendix 2B, User's Guide to the LWR Assemblies Data Base.

2.2.2 Fuel Assemblies of Pressurized-water Reactors

Pressurized-water reactor fuel assemblies are currently manufactured by four vendors - Advanced Nuclear Fuels (which was formerly Exxon Nuclear), Babcock & Wilcox, Combustion Engineering, and Westinghouse. Advanced Nuclear Fuels (ANF) only supplies reload fuel; hence, no unique designs of PWR fuel are attributed to them. Two models of Babcock & Wilcox (B&W) fuel are described - 15 x 15 and 17 x 17 fuel-rod arrays. Three models of Combustion Engineering (CE) fuel are described - 14 x 14, 15 x 15, and 16 x 16 fuel-rod arrays. Four models of Westinghouse (WE) fuel are described - 14 x 14, 15 x 15, 15 x 16, and 17 x 17 fuel-rod arrays. Schematic drawings of these models are shown in Figure 2.2.1 (B&W), Figure 2.2.2 (CE), and Figure 2.2.3 (WE). Only one drawing is shown when the array size is the primary design difference between the models.

2.2.2.1 Babcock & Wilcox 15 x 15 Array Design

The 15 x 15 model of Babcock & Wilcox (B&W) fuel incorporates 208 fuel-rod positions in a square array, and the rods are supported at intervals by eight spacer grids. This model has 16 guide tubes and a centrally located instrument tube. The manner in which B&W positions spacer grids in their fuel assemblies is different from other manufacturers. Grids in B&W fuel are not welded into place on the instrument tube but are supported by a series of Zircaloy-4 grid sleeves that surround the instrument tube. This method establishes the position

of the spacer grids but allows them to move to reduce stresses on the fuel rods, grids, and guide tubes. The 15 x 15 model uses a single coil spring of Inconel-718 as a hold-down device. The spring is held in place by a stainless steel spring retainer that has its position fixed by a spot-welded plug. The overall length of the B&W 15 x 15 model is 165.63 inches. Two versions of the 15 x 15 model have been manufactured by B&W. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

Babcock & Wilcox 15 x 15 Mark B

The original version of the Babcock & Wilcox 15 x 15 model, the Mark B, has Inconel-718 spacer grids.

Babcock & Wilcox 15 x 15 Mark BZ

A refinement of the Mark B version, the Mark BZ, replaces the intermediate six Inconel spacer grids with Zircaloy-4 spacers. These Zircaloy grids are somewhat larger than their Inconel counterparts, but the decreased density of the Zircaloy makes the total weight approximately equivalent.

2.2.2.2 Babcock & Wilcox 17 x 17 Mark C Array Design

The Mark C model of Babcock & Wilcox (B&W) fuel incorporates 264 fuel-rod positions in a 17 x 17 square array, and the rods are supported at intervals by 8 spacer grids. This model has 24 guide tubes and a centrally located instrument tube. The manner in which B&W positions spacer grids is different from other manufacturers. Grids in B&W fuel are not welded into place on the instrument tube but are supported by a series of grid sleeves that surround the instrument tube. This method establishes the position of the spacer grids but allows them to move to reduce stresses on the fuel rods, grids, and guide tubes. The Mark C uses four coil springs of Inconel X-750 as a hold-down device. The springs are held in place by a stainless steel retainer. The overall length of the B&W 17 x 17 model is 165.72 inches. Only one version of the 17 x 17 model has been manufactured by B&W, and it is not yet being used commercially. Four of these assemblies have been tested at the Oconee reactors. Design parameters are given in Appendix 2A.

2.2.2.3 Combustion Engineering 14 x 14 Array Design

The 14 x 14 model of Combustion Engineering (CE) fuel incorporates 176 fuel-rod positions and 5 extra large guide tubes in a square array. Each of the guide tubes is approximately 1 inch in diameter and displaces 4 fuel-rod positions. Because of the small number of these large guide tubes, CE was the first designer of PWR fuel to incorporate burnable poisons as integral parts of the fuel assemblies rather than outside the assembly as Nonfuel Assembly (NFA) hardware (see Section 2.8). These burnable poisons were incorporated into the assembly as nonfueled rods. The use of nonfueled poison rods in CE-designed arrays causes the number of fuel rods used in each version to vary from 164 to 176. This model uses 9 spacer grids and is typically 157.24 inches long. The design of the top end fitting is also unusual. It consists of two separate plates connected to the guide tubes by locking posts. Five helical Inconel X-750 springs are utilized as a hold-down device. The locking posts can be removed when a torque is applied to them. This feature makes reconstitution of fuel assemblies or replacement of defective fuel rods relatively simple. Five assembly types of this model have been fabricated by three different vendors. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

Combustion Engineering 14 x 14 Standard

This version features eight Zircaloy-4 spacer grids in the core and plenum regions of the assembly and an Inconel-625 spacer grid in the lower end of the assembly. It typically has 164 fuel rods and 12 nonfueled burnable poison rods which use boron carbide as the neutron poison.

Combustion Engineering 14 x 14 Ft. Calhoun

This version, supplied only to the Fort Calhoun reactor, is shorter (146 in.) than the standard 14 x 14 model. It also uses eight Zircaloy-4 and one Inconel-625 spacer grids. It typically has 168 fuel rods and 8 nonfueled burnable poison rods which use boron carbide as the neutron poison.

Westinghouse 14 x 14 "Model C"

This version was supplied by Westinghouse to the Millstone 2 reactor. All of the spacer grids are constructed of Inconel-718. The spacer grids are brazed to stainless steel grid sleeves to fix their position along the guide tubes. The helical hold-down springs on this version are also constructed of Inconel-718. Westinghouse used no neutron poisons in the fabrication of this assembly; thus, it has 176 fuel rods.

Advanced Nuclear Fuels 14 x 14 CE

The ANF version of Combustion Engineering's 14 x 14 model has bimetallic (89% Zircaloy-4, 11% Inconel-718) spacer grids. It typically has 172 fuel rods and 4 nonfueled burnable poison rods which use approximately 650 grams of boron carbide per rod.

Advanced Nucelar Fuels 14 x 14 Ft. Calhoun

ANF has also manufactured a version of the CE fuel for the Fort Calhoun reactor. It is also shorter than the standard version and uses bimetallic (89% Zircaloy-4, 11% Inconel-718) spacer grids. No other information is available at this time.

2.2.2.4 Combustion Engineering 15 x 15 Array Design

The Combustion Engineering 15 x 15 array for the Palisades reactor and the Westinghouse design used at Yankee Rowe were the two earliest PWR fuel designs and differ in many ways from current PWR fuel designs. The CE 15 x 15 fuel incorporates 216 fuel-rod positions in a square array. The Palisades reactor uses cruciform blades for control elements so versions of this model have no guide tubes for control elements. They do have a single, centrally located instrument tube. Structural support is provided by eight guide bars, two on each side of the assembly. These guide bars are solid pieces of Zircaloy-4 weighing over two kgs each. Major differences between these assembly type are listed below. Detailed differences and parameters are given in Appendix 2A.

Combustion Engineering 15 x 15

Combustion Engineering's version of the 15 x 15 model has nine Zircaloy-4 grid spacers in the core and plenum regions and one Inconel-625 grid spacer in the lower end region. It typically has 204 fuel rods and 12 nonfueled burnable poison rods which use boron carbide as the neutron poison.

Advanced Nuclear Fuels 15 x 15 CE

ANF's version of the 15 x 15 model has bimetallic (79% Zircaloy-4, 21% Inconel-718) spacer grids. It does not use nonfueled burnable poison rods but incorporates a neutron poison in up to eight fuel rods; each rod contains about 84 grams of a gadolinia poison. It typically has 216 fuel rods.

2.2.2.5 Combustion Engineering 16 x 16 Array Design

The 16 x 16 model of Combustion Engineering (CE) fuel incorporates 236 fuel-rod positions and 5 extra large guide tubes in a square array. Each of the guide tubes is approximately 1 inch in diameter and displaces 4 fuel-rod positions. Because of the small number of these large guide tubes, CE was the first designer of PWR fuel to incorporate burnable poisons as integral parts of the fuel assemblies rather than outside the assembly as Nonfuel Assembly (NFA) hardware (see Section 2.8). These burnable poisons were incorporated into the assembly as nonfueled rods. The use of nonfueled poison rods in CE-designed arrays causes the number of fuel rods used in each version to vary from 220 to 232. This model uses 11 spacer grids and is typically 177 inches long. The design of the top end fitting is also unusual. It consists of two separate plates connected to the guide tubes by locking posts. Five helical Inconel X-750 springs are utilized as a hold-down device. The locking posts can be removed when a torque is applied to them. This feature makes reconstitution of fuel assemblies or replacement of defective fuel rods relatively simple. CE has made four assembly types of this model. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

Combustion Engineering 16 x 16 St. Lucie 2

This was the first version of CE's 16 x 16 model fuel. It is shorter (158.1 in.) than the later versions of this model. It also has only ten spacer grids, nine of which are made of Zircaloy-4; they are located in the core and plenum regions of the assembly. The tenth spacer grid, made of Inconel 625, is located in the lower end region. It typically has 224 fuel rods and 12 nonfueled burnable poison rods that use boron carbide as the neutron poison.

Combustion Engineering 16 x 16 Ark. Nucl. 2

This version, supplied to the Arkansas Nuclear One, Unit 2 and Waterford 3 reactors, is the first of the longer 16 x 16 assemblies. It features a total of 12 spacer grids, 11 of which are made of Zircaloy-4 and are located in the core and plenum regions. The remaining spacer grid, made of Inconel 625, is located in the lower end region. This version typically has 232 fuel rods and 4 nonfueled burnable poison rods that use boron carbide as the neutron poison.

Combustion Engineering 16 x 16 San Onofre

This version, supplied to the San Onofre Units 2 and 3, features ten Zircaloy-4 spacer grids in the core and plenum regions. The remaining spacer grid, made of Inconel-625, is located in the lower end region. This version typically has 224 fuel rods and 12 nonfueled burnable poison rods that use boron carbide as the neutron poison.

Combustion Engineering 16 x 16 System 80

Combustion Engineering's latest entry to the 16 x 16 model line, the System 80 version, is more than a new fuel. System 80 is CE's newest reactor system design. The System 80 fuel is not very different from that of the other 16 x 16 versions. It has ten Zircaloy-4 grid spacers in the core and plenum regions and one Inconel-625 grid spacer in the lower end region. It typically has 220 fuel rods and 16 nonfueled burnable poison rods that use boron carbide as the neutron poison.

2.2.2.6 Westinghouse 14 x 14 Array Design

The 14 x 14 model of the Westinghouse fuel incorporates 179 fuel-rod positions in a square array. It has 16 guide tubes and a

centrally located instrument tube. The fuel rods are structurally supported at intervals by 7 spacer grids attached to the instrument tube. The spacer grids are brazed to stainless steel sleeves that are held in place by bulges in the metal of the instrument tube. The fuel rods in the Westinghouse assemblies are held down by Inconel-718 leaf springs. The overall length of the first Westinghouse version was 137.06 in., but all subsequent versions have been about 159.7 in. in length. Seven different assembly types of this model have been fabricated by three different vendors. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

Westinghouse 14 x 14 Standard/SC

In the first version of this model type, the assembly did not have an instrument tube and had 180 fuel rods. The fuel-rod cladding and guide tubes were fabricated of stainless steel, and the grid assemblies were fabricated of Inconel-718.

Westinghouse 14 x 14 Standard/ZCA

In this version, Zircaloy-4 was used for the fuel-rod cladding, but Westinghouse continued to use stainless steel for the guide tubes and instrument tubes and Inconel 718 for the spacer grids. This was the first version of the 160-in. model.

Westinghouse 14 x 14 Standard/ZCB

In addition to Zircaloy-4 fuel-rod cladding, this assembly also utilized Zircaloy-4 for the construction of the guide tubes and instrument tubes. It continued to use Inconel-718 for the grid assemblies.

Westinghouse 14 x 14 OFA

In the optimized fuel assembly, Zircaloy-4 replaced Inconel-718 as the material of construction for the five intermediate grid assemblies. Because less neutron-absorbing material is available in the core, OFA assemblies typically have smaller-diameter fuel rods and slightly less fuel than standard assemblies.

Babcock & Wilcox 14 x 14 Ginna

In 1973, B&W supplied the Ginna reactor with two fuel assemblies of the Westinghouse design. These assemblies use B&W's method for positioning grid assemblies; no other information on this particular version is available at this time.

Advanced Nuclear Fuel 14 x 14 WE

ANF's reload fuel for the Westinghouse 14 x 14 array design is similar to Westinghouse's OFA version. It features bimetallic (93% Zircaloy-4, 7% Inconel-718) grid assemblies at eight (rather than seven) locations along the assembly length. It is somewhat longer than the Westinghouse versions (160.1 in.)

Advanced Nuclear Fuel 14 x 14 Top Rod

ANF's Top Rod reload fuel has been manufactured for the Prairie Island reactor. Like the Exxon 14 x 14 WE, it has eight bimetallic grid assemblies and is somewhat longer than the Westinghouse-made fuel. It features gadolinia-poisoned fuel rods (a maximum of four per assembly) and a grappling rod that runs across the upper end fitting.

2.2.2.7 Westinghouse 15 x 15 Array Design

The 15 x 15 model of the Westinghouse fuel incorporates 204 fuel-rod positions in a square array. It has 20 guide tubes and a centrally located instrument tube. The fuel rods are structurally supported at intervals by seven spacer grids brazed to stainless steel sleeves that are held in place by bulges in the metal of the instrument tube. The fuel rods in Westinghouse assemblies are held down by Inconel-718 leaf springs. The overall length of the first Westinghouse version was 137.06 in., but all subsequent versions have been 159.7 in. in length. Five different assembly types of this model have been fabricated by three different vendors. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

Westinghouse 15 x 15 Standard/SC

In the first version of this model type, the assemblies featured stainless steel 304 fuel-rod cladding and guide tubes and Inconel-718 spacer grids. These assemblies were 137.06 in. in length.

Westinghouse 15 x 15 Standard/ZC

In this version, Zircaloy-4 was use for the fuel-rod cladding and guide tubes, but Westinghouse retained the use of Inconel-718 for the spacer grids. This was the first of the 160-inch versions of this model.

Westinghouse 15 x 15 OFA

In the optimized fuel assembly, Zircaloy-4 replaced Inconel-718 as the material of construction for the five intermediate grid assemblies. OFA assemblies typically have smaller-diameter fuel rods and slightly less fuel than standard assemblies.

Advanced Nuclear Fuel 15 x 15 WE

ANF's reload fuel for the Westinghouse 15 x 15 array design is similar to Westinghouse's OFA version. It features bimetallic (93% Zircaloy-4, 7% Inconel-718) grid assemblies at eight (rather than seven) locations along the assembly length.

Babcock & Wilcox 15 x 15 St. Steel

Babcock & Wilcox's version of this model of Westinghouse fuel has been used only at the Haddam Neck reactor. It, like the Westinghouse 15 x 15 Standard/SC, has stainless steel fuel-rod cladding and guide tubes and uses Inconel-718 for the spacer grids. It is 137.06 inches in length.

2.2.2.8 Westinghouse 15 x 16 Array Design

The 15 x 16 model of Westinghouse fuel was designed for the Yankee-Rowe reactor, the first commercial PWR reactor. The design is unique to Yankee-Rowe and Indian Point 1. These are the only two nonsquare arrays. The Indian Point 1 reactor used a 13 x 14 array, which has not been described by any vendor. Reload fuel for Yankee-Rowe has been manufactured by ANF and CE and possibly by other vendors. Information on reload fuel from ANF and CE serve as the basis for the following description. The 15 x 16 model has an A version and a B version. The A version is essentially a 15 x 15 array, with 15 additional fuel-rod positions on both sides of one corner. The B version is essentially a 16 x 16 array, with 17 fuel-rod positions

removed from both sides of one corner. When the A and B versions are alternated in rows and between rows, as shown in Figure 2.2.4, the result is an opening for a cruciform blade as wide as the 16-element side of the assembly. The A version has 240 fuel-rod positions; the B version has 239. Both use eight Zircaloy-4 solid guide bars and a single, centrally located instrument tube. Both versions are approximately 111.8 in. in length. The fuel rods are held down by large hold-down springs made of Inconel X-750. At least three different assembly types of this model have been fabricated by three different vendors. Major differences between these assembly types are given below. Detailed differences and parameters are given in Appendix 2A.

Westinghouse 15 x 16 Yankee-Rowe

No information is available at this time.

Combustion Engineering 15 x 16 Yankee-Rowe

This version uses five Zircaloy-4 spacer grids in the incore and plenum regions and a single Inconel-625 grid in the lower end.

Advanced Nuclear Fuels 15 x 16 Yankee-Rowe

This version uses six bimetallic spacer grids (86% Zircaloy-4, 16% Inconel-718), one of which is located in the gas plenum region.

2.2.2.9 Westinghouse 17 x 17 Array Design

The 17 x 17 model is the most recent of the Westinghouse fuel designs. It incorporates 264 fuel-rod positions in a square array. It has 24 guide tubes and a centrally located instrument tube. The fuel rods are structurally supported at intervals by seven spacer grids attached to the instrument tube. The spacer grids are brazed to stainless steel sleeves that are held in place by bulges in the metal of the instrument tube. The fuel rods in Westinghouse assemblies are held down by Inconel-718 leaf springs. The overall length of the first Westinghouse version was 159.76 inches. Six different assembly types of this model have been fabricated by three different vendors. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

Westinghouse 17 x 17 Standard

This assembly was the first of the 17 x 17 array design. It features grid assemblies of Inconel-718 and larger diameter fuel rods than the other versions of this model.

Westinghouse 17 x 17 OFA

This assembly features grid assemblies of Zircaloy-4 in the five intermediate grid assembly positions and fuel rods of a smaller diameter.

Westinghouse 17 x 17 Vantage 5

The Vantage 5 is Westinghouse's newest entry in its fuel assembly line. Although not yet in commercial use, four assemblies have been tested at the V.C. Summer reactor. The Vantage 5 features natural uranium axial blankets, three intermediate flow mixers (in addition to eight grid assemblies) to increase turbulence, integral burnable poisons, and a removable top nozzle to aid in fuel-rod replacement or assembly reconstitution. It is approximately 0.3 inch longer than the OFA version. The six intermediate grid assemblies are also made of Zircaloy-4.

Westinghouse 17 x 17 XLR

The XLR version of the 17 x 17 array design has been manufactured specifically for the South Texas reactors. The design mimics the OFA design, but the fuel assembly has an overall length of 199 inches.

Advanced Nuclear Fuel 17 x 17 WE

ANF's reload fuel for Westinghouse's 17 x 17 reactors is similar to the Westinghouse's OFA version. It features 10 bimetallic (86% Zircaloy-4, 14% Inconel-718) grid spacers.

Babcock & Wilcox 17 x 17 Mark BW

Babcock & Wilcox has recently entered the market for supplying reload fuel to Westinghouse reactors by the introduction of the Mark BW fuel. It features larger diameter fuel rods, Zircaloy grids throughout,

and a removable top end fitting. It does not have integral burnable poisons.

2.2.3 Fuel Assemblies of Boiling-Water Reactors

The BWR fuel assemblies are currently manufactured by three vendors - Advanced Nuclear Fuels (formerly Exxon Nuclear), General Electric, and Westinghouse. Allis-Chalmers built one BWR reactor plant at LaCrosse and supplied the initial fuel for it. Other than LaCrosse, all BWR plants have been designed and built by GE; Advanced Nuclear Fuels and Westinghouse only supply reload fuel to existing reactors. Much of the data on BWR fuel is proprietary; efforts are currently under way to characterize BWR fuels from information in the open literature and the Federal Docket. Many features of BWR assemblies are the same regardless of the model or version. For example, BWR fuel assemblies use spacer grids at intervals along the assembly length to provide support for the fuel rods. These spacer grids are normally held in position by metal tabs attached to a spacer capture rod. Spacer capture rods are one type of nonfueled rod in BWR assemblies. Spacer capture rods, inert rods, and water rods are all hollow tubes of Zircaloy-2 that provide additional water inside the fuel assembly for better neutron moderation. BWR assemblies use fueled tie rods to provide axial structural support. These fuel rods are slightly longer than standard fuel rods and are threaded on each end. They either screw into the top and bottom tie plates or are attached to these plates via an external nut. Each fuel rod is held in place against the bottom tie plate by a separate compression spring. These compression springs encircle the top end of the fuel rod. Nine designs of BWR fuel have been identified - 1 ANF design, 1 Allis-Chalmers design, 6 GE designs, and 1 Westinghouse design. Schematic drawings of these designs are shown in Figure 2.2.5 (GE & ANF) and Figure 2.2.6 (AC & WE).

2.2.3.1 Advanced Nuclear Fuels 9 x 9 Array Design

The 9 x 9 model is ANF's first independent design of light-water reactor (LWR) fuel. It is designed to provide reload fuel at GE

BWR/3,4,5,6 plants. It incorporates 81 fuel-rod positions in a square array. The fuel bundle uses 7 bimetallic (84% Zircaloy-4, 16% Inconel-718) spacer grids. The fuel-rod compression springs are made of Inconel X-750. Two versions of this model have been fabricated to fit different GE plant designs. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

Advanced Nuclear Fuels 9 x 9 JP-3

Apparently designed to fit GE's BWR/3 plants, this version is 171.29 inches in length and uses up to 8 poisoned fuel rods that typically contain 77 grams of gadolinia per rod.

Advanced Nuclear Fuels 9 x 9 JP-4,5

Apparently designed to fit GE's BWR/4,5,6 plants, this version is 176.05 in length and uses up to 7 poisoned fuel rods that typically contain 95 grams of gadolinia per rod.

2.2.3.3 Allis-Chalmers 10 x 10 Array Design

Descriptions of the original Allis-Chalmers fuel are not available since Allis-Chalmers discontinued its nuclear reactor activities soon after building the LaCrosse plant. Since that time, Advanced Nuclear Fuels has supplied reload fuel for the plant. The following information is in regard to ANF's reload assemblies. This model incorporates 100 fuel-rod positions in a square array and uses 3 bimetallic (79% stainless steel, 21% Zircaloy-4) spacer grids. It has four nonfueled positions - three inert rods and a spacer capture rod. The fuel rod compression springs are made of Inconel X-750. The overall length of the ANF reload assembly is 102.45 inches. Detailed design parameters are given in Appendix 2A.

2.2.3.4 General Electric 6 x 6 Dresden Array Design

The 6 x 6 Dresden model was one of General Electric's first designs for BWR fuel. It was designed for the BWR/1 plant Dresden-1.

It incorporates 36 fuel-rod positions in a square array and uses 7 spacer grids. The fuel-rod compression springs are made of Inconel X-750. The overall length of the model is 134.32 inches. Versions of this model have been fabricated by three different vendors. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

General Electric 6 x 6

No information is available at this time.

Advanced Nuclear Fuels 6 x 6

This version uses bimetallic (84% Zircaloy-4, 16% Inconel-718) spacer grids. It has one nonfueled inert rod.

United Nuclear 6 x 6

No information is available at this time.

2.2.3.4 General Electric 6 x 6 Humboldt Bay Array Design

The 6 x 6 Humboldt Bay model was one of General Electric's first designs for BWR fuel. It was designed for the BWR/1 plant Humboldt Bay. It incorporates 36 fuel-rod positions in a square array and uses 7 spacer grids. The fuel-rod compression springs are made of Inconel X-750. The overall length of the model is about 85 inches. Versions of this model have been fabricated by GE and ANF vendors. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

General Electric 6 x 6

No information is available at this time.

Advanced Nuclear Fuels 6 x 6

No information is available at this time.

2.2.3.5 General Electric 7 x 7 Array Design

The 7 x 7 model was the backbone of General Electric's BWR/2-5 plants. It incorporates 49 fuel-rod positions in a square array and uses seven spacer grids. The fuel-rod compression springs are made of Inconel X-750. The overall length of the model for BWR/2,3 plants is 171 in. The overall length of the model for BWR/4,5,6 plants is 176 in. The fuel rods for all these versions have been backfilled with helium,

but only one version has been prepressurized. At least three versions of this model have been fabricated by GE and ANF. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

General Electric 7 x 7 BWR/2,3; Ver. 1

The fuel rod diameter of this version was 0.570 in. The fuel pellets had sharp corners and were longer than later versions of this assembly. The spacer grids for this version were made of Zircaloy-4 with Inconel-718 spring fingers. No further information is available.

General Electric 7 x 7 BWR/2,3; Ver. 2

This version features fuel pellets with chamfered corners. It also uses Zircaloy-4 spacer grids with Inconel-718 spring fingers and has a hydrogen getter. The fuel rod diameter was 0.563 in.

General Electric 7 x 7 BWR/4,5

This version was designed for BWR/4,5 plants. It has a 0.563 in. diameter fuel rod.

Advanced Nuclear Fuels 7 x 7

Apparently used only in BWR/2 reactors, this version uses bimetallic spacer grids (84% Zirc-aloy-4, 16% Inconel-718). It has a single inert rod and may use up to 4 poisoned fuel rods that typically contain 45 grams of gadolinia per rod.

2.2.3.6 General Electric 8 x 8 Array Design

The 8 x 8 model of General Electric fuel was introduced for the BWR/6 reactor design. Reload fuel has been supplied to many BWR/2-5 plants also. The fuel incorporates 64 fuel-rod positions in a square array and uses 7 spacer grids. One or two of the fuel-rod positions are taken by water rods, hollow Zircaloy tubes that increase the amount of water available for neutron moderation. The fuel-rod compression springs are made of Inconel X-750. BWR/2,3 versions of this array design are 171 in. in length. BWR/4-6 version of this array design are

176 in. in length. At least four versions of this model have been fabricated by GE and ANF. Major differences between these assembly types are listed below. Detailed differences and parameters are given in Appendix 2A.

General Electric 8 x 8 BWR/2,3

These assemblies are designed for BWR/2 and BWR/3 plants. No further information is available at this time.

General Electric 8 x 8 BWR/4-6; Ver. 1

Developed for BWR/6 reactors, these assemblies are also used in BWR/4,5 plants. They use Zircaloy-4 spacer grids with Inconel-718 spring fingers. They have one water rod.

General Electric 8 x 8 BWR/4-6, Ver. 2

Similar to version 1, these assemblies have two water rods. No further information is available at this time.

Advanced Nuclear Fuels 8 x 8 JP-3

Developed as a reload for BWR/3 reactors, these assemblies are 171.79 inches in length, have only one spacer capture rod, and use bimetallic (84% Zircaloy-4, 16% Inconel-718) spacer grids. They may have up to 6 poisoned fuel rods that typically contain 99 grams of gadolinia per rod.

Advanced Nuclear Fuels 8 x 8 JP-4,5

Developed as a reload for BWR/4,5 reactors, these assemblies are 176.79 inches in length, have an inert rod and a spacer capture rod, and use bimetallic (84% Zircaloy-4, 16% Inconel-718) spacer grids. They may have up to 8 poisoned fuel rods that typically contain 69 grams of gadolinia per rod.

2.2.3.7 General Electric 9 x 9 Array Design

The 9 x 9 model was designed for the BWR/1 plant at Big Rock Point. It incorporates 81 fuel rod positions in a square array. On the basis of the features of the ANF 11 x 11 array design, the overall length of

the model is about 84 inches. It is the the widest BWR design at 6.515 inches. Both GE and ANF have made versions of this model. The names of these versions are given below, although no further information on any of them is available at this time.

General Electric 9 x 9

No further information is available at this time.

Advanced Nuclear Fuels 9 x 9 Big Rock Point

No further information is available at this time.

2.2.3.8 General Electric 11 x 11 Array Design

The 11 x 11 model of GE fuel is designed as a reload for the Big Rock Point reactor. Both GE and ANF have made 11 x 11 fuel for Big Rock Point. No information on the GE version is available at this time. The ANF version incorporates 121 fuel-rod positions in a square array and uses only 3 bimetallic (77% Zircaloy-4, 23% Inconel-718) spacer grids. It has three nonfueled inert rods and one nonfueled spacer capture rod. It may have up to 4 poisoned fuel rods with typically 19 grams of gadolinia per rod. Detailed design parameters are given in Appendix 2A.

2.2.3.9 Westinghouse 8 x 8 Array Design

Westinghouse entered the BWR reload market in 1982 with the QUAD+, an adaptation of a design by ASEA-ATOM in Sweden. The QUAD+ design has an 8 x 8 fuel-rod array subdivided in 4 x 4 subarrays, or minibundles, which are separated by a hollow Zircaloy cross filled with nonboiling water. The fuel assembly has no water rods since the water is provided in the water cross. It has 64 fuel rods and is 175.5 inches in length.

The channel assembly is the most novel mechanical design feature of the QUAD+. The channel assembly forms a basket and offers stronger structural support for the fuel bundle than other designs. The channel assembly consists of the Zircaloy channel welded to the water cross. The channel is attached mechanically to the lower nozzle by three Inconel screws per side and to the upper nozzle by four rectangular Zircaloy bars welded to the inside surface of the channel and bolted to the top nozzle. The top nozzle has a standard bail for lifting the

assembly. The water cross is made of Zircaloy-4 sheet with the walls spaced by welded dimples to produce a 0.240-inch water gap. The bottom of the cross is sealed by appropriately shaped end plugs that are welded to the walls of the cross. The channel, water cross, and upper and lower nozzle assembly form a basket that accommodates the four minibundles. Each minibundle is an independently removable subassembly that consists of 14 regular fuel rods, 2 tie rods, an upper and lower tie plate, and 6 spacers.

The fuel can be disassembled by detaching the upper nozzle from the four posts and lifting the upper nozzle off with a standard tool. A special handling tool is needed to grapple and move the minibundles out of the channel assembly. The tie rods are attached mechanically to the top and bottom tie plates and serve the normal functions of spacing the fuel rods. The bottom of the water cross has a special orifice to control bypass flow into the water cross. Flow communication passages exist between minibundles at various locations at the tips of the water cross. The spacers, made of Zircaloy-4 with integral Zircaloy-4 springs, are captured by cylindrical Zircaloy tabs welded to the Zircaloy cladding of a fuel rod. Detailed design parameters are given in Appendix 2A.

2.2.4 Future Fuel Assembly Designs

The commercial fuel assembly designs have evolved through numerous design changes since the first reactors, and the process continues. The focus of new designs has been, and will continue to be, on increasing burnup, increasing neutron economy, and developing increased resistance to pellet-clad interactions (PCI). Other developments that have occurred involved the fuel rod connections with the assembly body. Newer designs permit easy fuel-rod removals and replacements (either partially or entirely), which permit improved fuel management and easy disassembly for reprocessing or consolidation. It is difficult to foresee any further design changes in the fuel assemblies that will have a large effect from a waste disposal

viewpoint. For existing reactors and those under construction, the overall dimensions of the fuel assembly cannot be changed significantly without redesign of the core, and commercial development of any new core concept is in the distant future for the United States. Small changes in the rod diameter and use of additional water rods in the newer BWR designs can have little effect on canisters packed with fuel rods or intact fuel assemblies.

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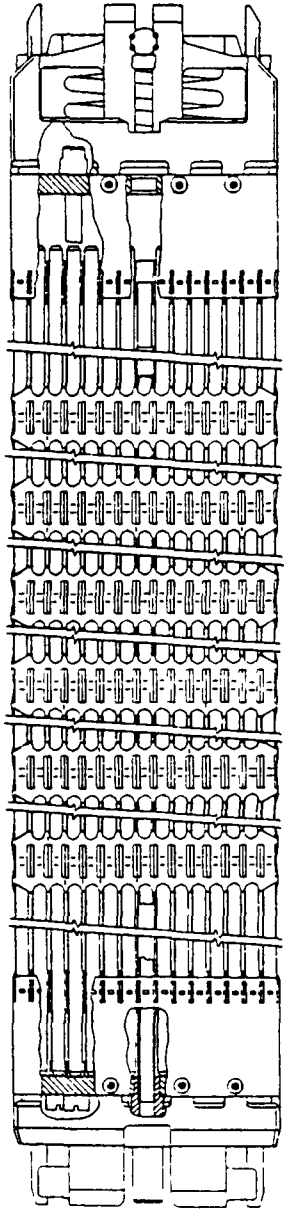

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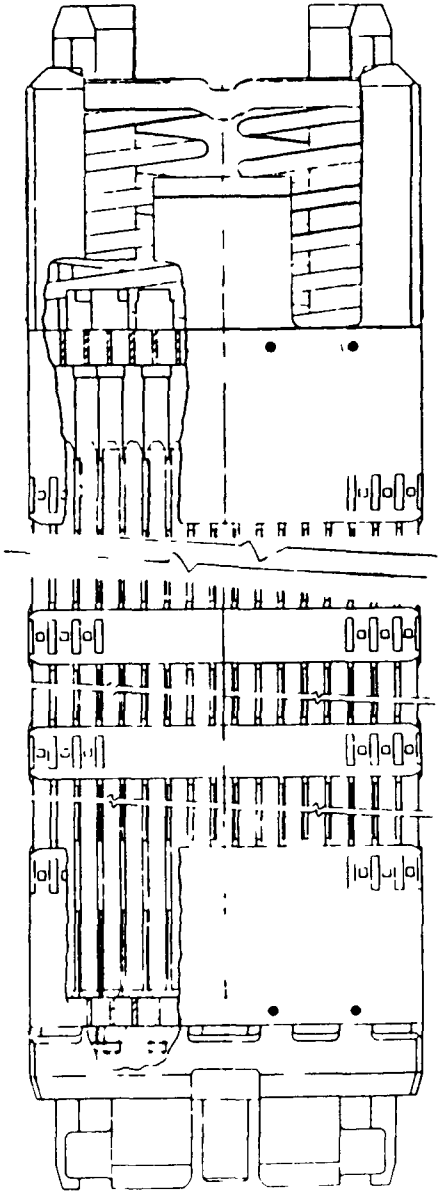

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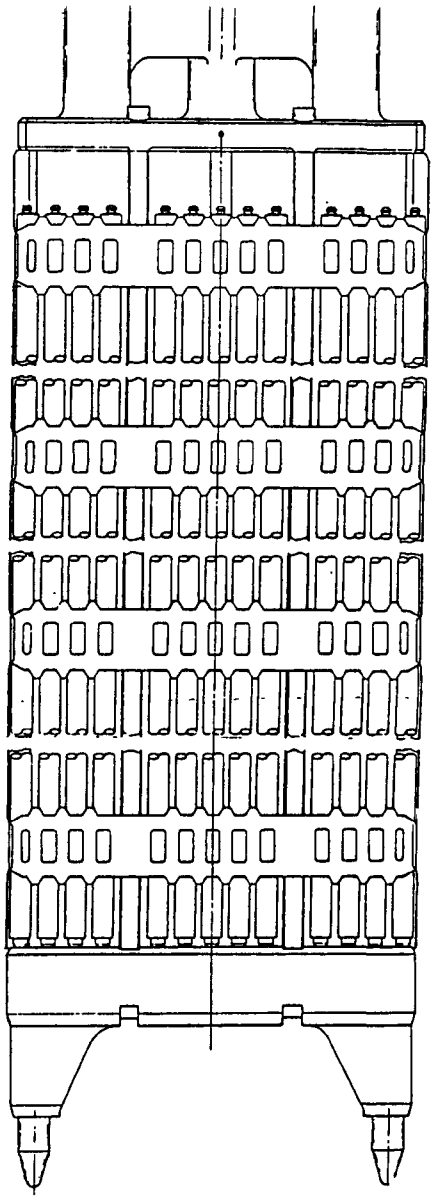
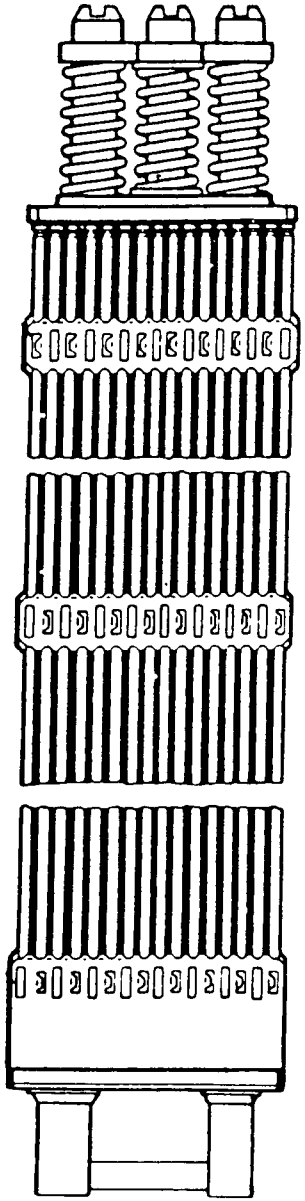
Figure 2.2.1 Schematic drawings of Babcock & Wilcox Fuel Assembly Designs

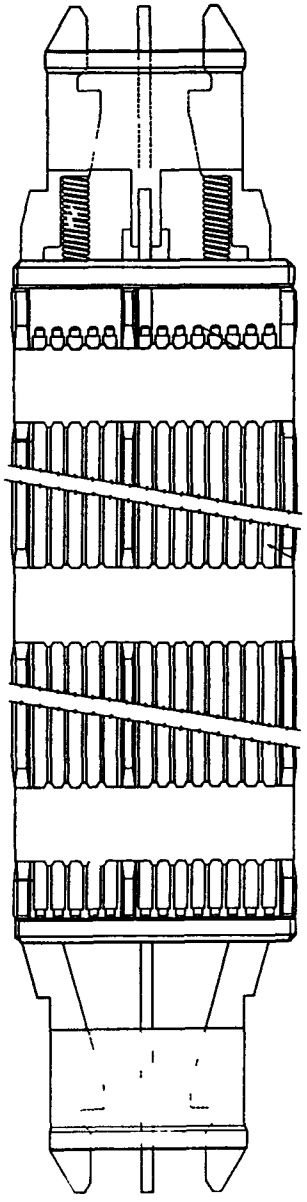


15 X 15

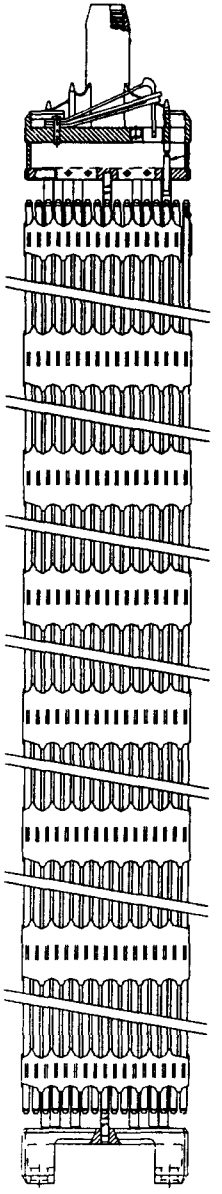


14 X 14, 16 X 16

Figure 2.2.2 Schematic drawings of Combustion Engineering Fuel Assembly Designs



15 X 16



14 X 14, 15 X 15, 17 X 17

Figure 2.2.3 Schematic drawings of Westinghouse Fuel Assembly Designs

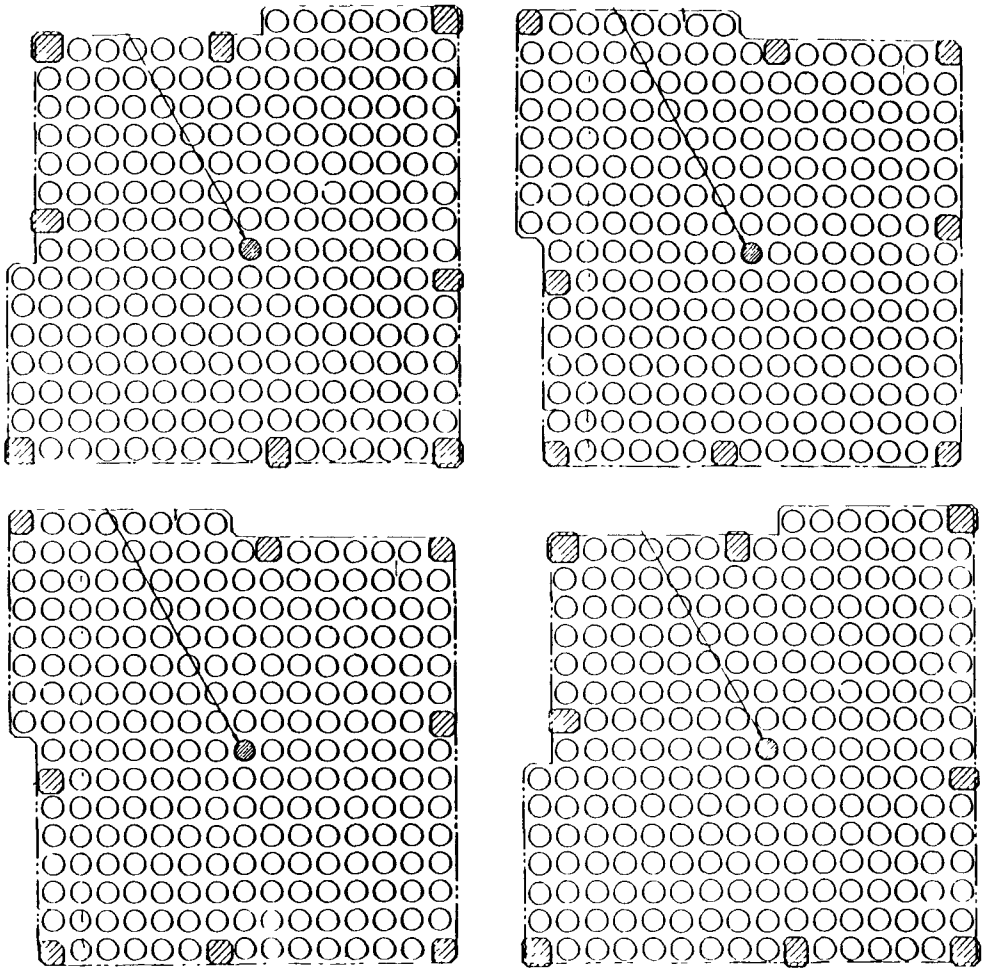
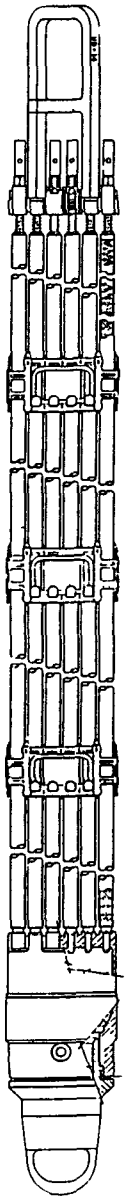
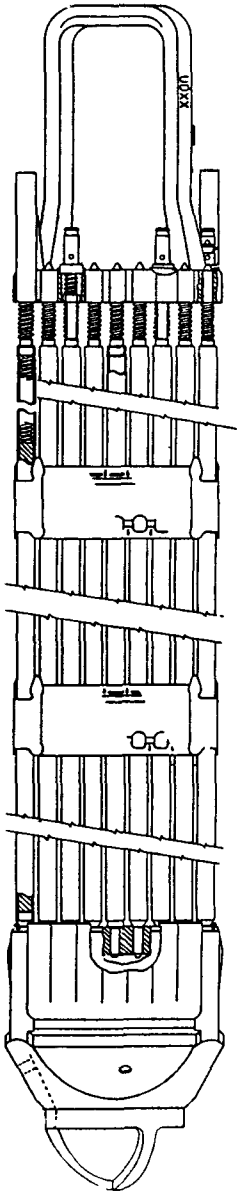


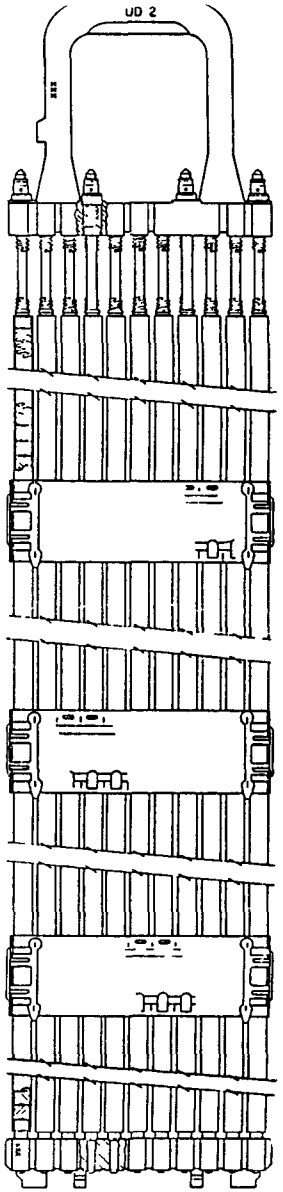
Figure 2.2.4 Schematic drawings of Westinghouse 15 X 16 Fuel Assemblies, Types A and B



6 X 6

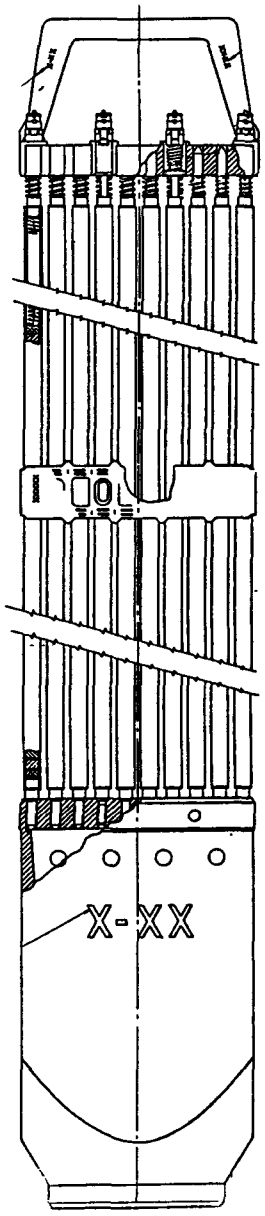


7 X 7, 8 X 8, ANF 9 X 9

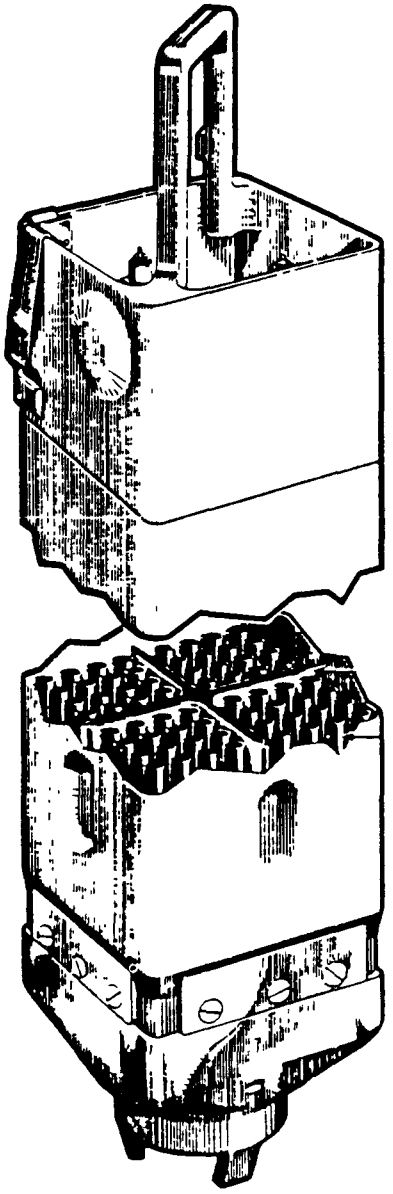


11 X 11

Figure 2.2.5 Schematic drawings of General Electric Fuel Assembly Designs



10 X 10



8 X 8 QUAD+

Figure 2.2.6 Schematic drawings of Allis Chalmers and Westinghouse BWR Fuel Assembly Designs

Table 2.2.1 Example of Physical Description Report.

PHYSICAL DESCRIPTION REPORT PAGE: 1

Babcock & Wilcox 15 X 15 Mark BZ PWR
OVERALL ASSEMBLY CHARACTERISTICS

Initial Year of Manufacture	1984
Final Year of Manufacture	
Total Number Fabricated to Date	3764
Assembly Width (inches)	8.536
Assembly Length (inches)	165.625
with Control Rod Inserted	
including Holddown Device, etc	
Rod Pitch (inches)	0.568
Total Assembly Weight (lbs)	1515.0
Weight of Heavy Metal (lbs)	1022.12
Metric Tons Initial Heavy Metal (metric tons)	0.46363
Enrichment Range (% U235)	2.0-4.0
Average Design Burnup (MWd/MTIHM)	35000
Maximum Design Burnup (MWd/MTIHM)	50200
Linear Heat Rating (KW/foot)	6.30

Difficulty Indexes (0-not required, 1-simple,..,6-impossible)

for Cutting	3
for Mechanical Disassembly in Air	5
for Underwater Cosolidation	3
for Underwater Rod Replacement	5

Comments:

Table 2.2.1 Example of Physical Description Report. (cont.)

PHYSICAL DESCRIPTION REPORT PAGE: 2

Babcock & Wilcox 15 X 15 Mark BZ PWR

FUEL ASSEMBLY HARDWARE PARTS AND MATERIALS

Part Name	Parts/ Assembly	Weight(kg)/ Assembly	Zone	Material Name	Material Fraction
TOP NOZZLE	1	7.4800	TOP	St.Steel CF3M	1.00000
BOTTOM NOZZLE	1	8.1600	BOTTOM	St.Steel CF3M	1.00000
GUIDE TUBES	16	8.0000	IN CORE	Zircaloy-4	1.00000
INSTRUMENT TUBE	1	0.6400	IN CORE	Zircaloy-4	1.00000
SPACER-PLENUM	1	1.0400	GAS PLENUM	Inconel-718	1.00000
SPACER-BOTTOM	1	1.3000	BOTTOM	Inconel-718	1.00000
SPACER-INCORE	6	4.9000	IN CORE	Zircaloy-4	1.00000
SPRING RETAINER	1	0.9100	TOP	St.Steel CF3M	1.00000
HOLDDOWN SPRING	1	1.8000	TOP	Inconel-718	1.00000
UPPER END PLUG	2	0.0600	TOP	St.Steel 304	1.00000
UPPER NUT	15	0.5100	TOP	St.Steel 304L	1.00000
LOWER NUT	16	0.1500	BOTTOM	St.Steel 304	1.00000
GRID SUPPORTS	7	0.6400	IN CORE	Zircaloy-4	1.00000

Drawing Numbers Associated With Assembly:

02-32958F
11-55248F

Table 2.2.1 Example of Physical Description Report. (cont.)

PHYSICAL DESCRIPTION REPORT PAGE: 3

Babcock & Wilcox 15 X 15 Mark BZ PWR

FUEL ROD DESCRIPTION TABLE

Type of Rod	Fuel Rod
Fuel Rod Positions per Assembly	225
Typical Number of Fueled Rods per Assembly	208
Rod Diameter (inches)	0.430
Rod Length (inches)	153.68
Active Length (inches)	141.8
Weight per Rod (lbs)	7.00
Clad Material	Zircaloy-4
Clad Thickness (inches)	0.0265
Clad Final Conditioning	SRA
Fuel-Clad Gap (inches)	0.0042
Fill Gas Used	He
Initial Gas Pressure (psig)	415
Nitrogen Content of Fill Gas (percent)	3.0

Table 2.2.1 Example of Physical Description Report. (cont.)

PHYSICAL DESCRIPTION REPORT PAGE: 4

Babcock & Wilcox 15 X 15 Mark BZ PWR

FUEL ROD DESCRIPTION TABLE continued

Fuel Pellet Material	Uranium Oxide
Fuel Pellet Shape	Dished, Chamfered
Fuel Pellet Diameter (inches)	0.3686
Fuel Pellet Length (inches)	0.435
Fuel Pellet Weight per Rod (lbs)	5.58
Open Porosity (percent)	< 1%
Grain Size (microns)	10-14
Fuel Density (% theoretical)	95
O/U Ratio	2-2.02:1
Smear Density(gr/cm3)	9.75
Spacer Pellet Material	Zircaloy-4
Spacer Pellet Length (inches)	
Plenum Spring Material	St.Steel 302
Plenum Spring Weight per Assembly (lbs)	0.042
Plenum Length (inches)	11.720
Plenum Volume (cubic inches)	1.308

Comments:

2.3 QUANTITIES OF INTACT SPENT FUEL

2.3.1 Overview

Historical inventories and projections of LWR spent fuel are reported on an annual basis by the Energy Information Administration (EIA 1986) and reported by the DOE Integrated Data Base Program (DOE/IDB 1986). The basis for the projections is the future installed nuclear generating capacity. This information is based on EIA's annual survey of the nuclear utilities and is treated according to several possible projection scenarios:

- No New Orders Case: Includes only those reactors currently in operation or those which are more than 40% complete.
- Lower Reference Case: Includes those reactors currently in operation or those which are more than 40% complete plus a limited number of new reactors after the year 2000.
- Upper Reference Case: Includes all existing reactors, either completed or under construction, plus additional new reactors after the year 2000.

Table 2.3.1, summarized below, gives the EIA projections based on these three scenarios. These values were reported in September 1986. The 1987 values are slightly different; for 2020 they are 51, 130, and 199, respectively. Our data base is in terms of the 1986 values.

Installed GW(e)

End of CY	No new orders	Lower ref. case	Upper ref. case
1985	80	80	80
1990	105	105	105
2000	105	105	111
2010	101	113	167
2020	55	125	219

On the basis of these projected installed capacities, the spent fuel discharged can be estimated (DOE/IDB 1986). This estimation allows for an on-stream capacity factor of about 60%. It also allows for gradually increasing burnup from 1986 to 1998 (at a rate of 2% per year), and after 1998, an average fuel burnup of 31,200 MWd/MTIHM for BWR's and 40,000 MWd/MTIHM for PWR's. The projections of cumulative discharged spent fuel are given in Table 2.3.2, summarized below, for the three EIA cases. These are based on the September 1986 report. The 1987 values are slightly different; for 2020 they are 77,800, 87,500, and 98,300, respectively. Our data base is in terms of the 1986 values.

Cumulative spent fuel discharged (MTIHM)[a]

End of CY	No new orders	Lower ref. case	Upper ref. case
1985	12,400	12,400	12,400
1990	20,900	20,900	21,200
2000	40,800	40,800	41,600
2010	60,900	61,200	66,600
2020	79,300	86,900	105,800

[a]1986 EIA data for CY-1985.

The IDB Program used the EIA Upper Reference Case for more detailed calculations and tabulations. Table 2.3.3 shows the projected number of BWR and PWR assemblies discharged annually and the accumulated totals. These figures are based on permanently discharged fuel, and do not list temporarily discharged fuel elements awaiting reinsertion. The latter do, of course, occupy space in the storage pools.

In order to supplement this industry-wide composite and provide reactor- and assembly-type-specific information on the historical and projected quantities of spent fuel assemblies in an easily accessible form, the LWR Quantities Data Base was developed. This data base is a user-oriented, menu-driven IBM PC-compatible system which contains quantitative information on discharged fuel assemblies. Appendix 2D is the

user's guide to this data base. This detailed information is needed as input to in-depth transportation, consolidation, and interim storage studies. The following subsections describe the historical inventories and the two projection cases that the data base encompasses.

2.3.2 Assembly Type/Reactor Specific Inventories and Projections

The EIA's annual survey of nuclear utilities via the RW-859 form has been used as the basis for the historical portion of the LWR Quantities Data Base. The current version of this data base contains data on assemblies permanently discharged prior to December 31, 1985. Figure 2.3.1 shows the types of data available in the historical portion of the LWR Quantities Data Base. Each historical report lists the number of assemblies, the number of defective assemblies, the average burnup of the assemblies, and the average weight of heavy metal. The data may also be broken down by reactor type, utility, reactor, assembly type, or storage pool. Additional breakdowns by discharge year and burnup bin are available.

A sample report, a listing, by reactor and discharge year, of the number of assemblies discharged, is given in Table 2.3.4. The total number of discharged assemblies in this table, 45814, differs from the number reported by IDB, 46352, in Table 2.3.3. This difference of 538 assemblies is a result of IBD's inclusion of temporarily discharged assemblies in their count and the exclusion of these assemblies by the LWR Quantities Data Base. The exclusion of these assemblies stems from the fact that the data base uses different sources for its historical and projected data. Exclusion of these assemblies from the projected data would have been much more difficult than excluding them from the historical data. Also, the burnup attributed to the temporarily discharged assemblies could present a false picture when included with permanently discharged assemblies.

The assembly type (see Section 2.2) of each batch of assemblies in the EIA RW-859 Data Base (Disbrow 1986) was identified, if possible.

This identification was based on, but not limited to, data contained in the RW-859 Data Base and the vendor submittals. The data fields used for identification included the array size, the array suffix, the drawing number, the date of usage, and the weight of heavy metal used in the assembly. A listing from the LWR Quantities Data Base of the discharged fuel assemblies, by assembly type, discharge year, and burnup bin, is given in Table 2.3.5 for the Prairie Island 1 reactor.

2.3.3 Reactor-Specific Projections

The LWR Quantities Data Base has used the work of Heeb, et al. (Heeb 1986) as the basis for its projected data. It uses two of the cases studied by Heeb, the No New Orders Case with Extended Burnup and the Upper Reference Case with Extended Burnup. Figure 2.3.2 shows the types of data available in the projections portion of the LWR Quantities Data Base. Each projections report lists the number of assemblies and the average weight of heavy metal. The data may be broken down by reactor type, utility, or reactor. An additional breakdown by discharge year is available.

Heeb et al. base their reactor specific estimates of spent fuel discharges on information supplied by the utilities via the RW-859 form, EIA energy projections, and EIA discharge projections. The steps involved in the adjustment procedure from the utility supplied information are: 1) shift utility supplied startup and shutdown dates, 2) calculate electric energy generation from utility supplied data, 3) adjust utility-supplied discharge data to match EIA energy projections, and 4) adjust burnups to match EIA discharge projections. The adjusted data base generally matchs the electrical energy generation to within 1 percent. Cumulative spent fuel discharges are also generally within 1 percent of the EIA forecast value.

Table 2.3.6 gives a sample output from the projections portion of the LWR Quantities Data Base. It shows, for the reactors Farley 1 and Farley 2, the number of discharged assemblies per year. Note that in the year 2017, Farley 1 will discharge 157 assemblies, corresponding to

reactor shutdown. Table 2.3.7 projects the total number of fuel assemblies predicted to be discharged through the year 2020, by reactor type.

2.3.4 References for Section 2.3

Disbrow 1986. J.A. Disbrow, Energy Information Administration, RW-859 Data Base, (magnetic tape No. FE9451), July 10, 1986.

EIA 1986. Energy Information Administration, Commercial Nuclear Power: Prospects for the United States and the World, DOE/EIA-0438(86), September 1986.

Heeb 1986. C.M. Heeb, R.A. Libby, R.C. Walling, and W.L. Purcell, Reactor Specific Spent Fuel Discharge Projections: 1985 to 2020, PNL-5833, September 1986.

DOE/IDB 1986. Department of Energy, Integrated Data Base for 1986: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/RW-0006, Rev. 2, September 1986.

Figure 2.3.1 Overview of LWR Quantities Data Base Historical Data.

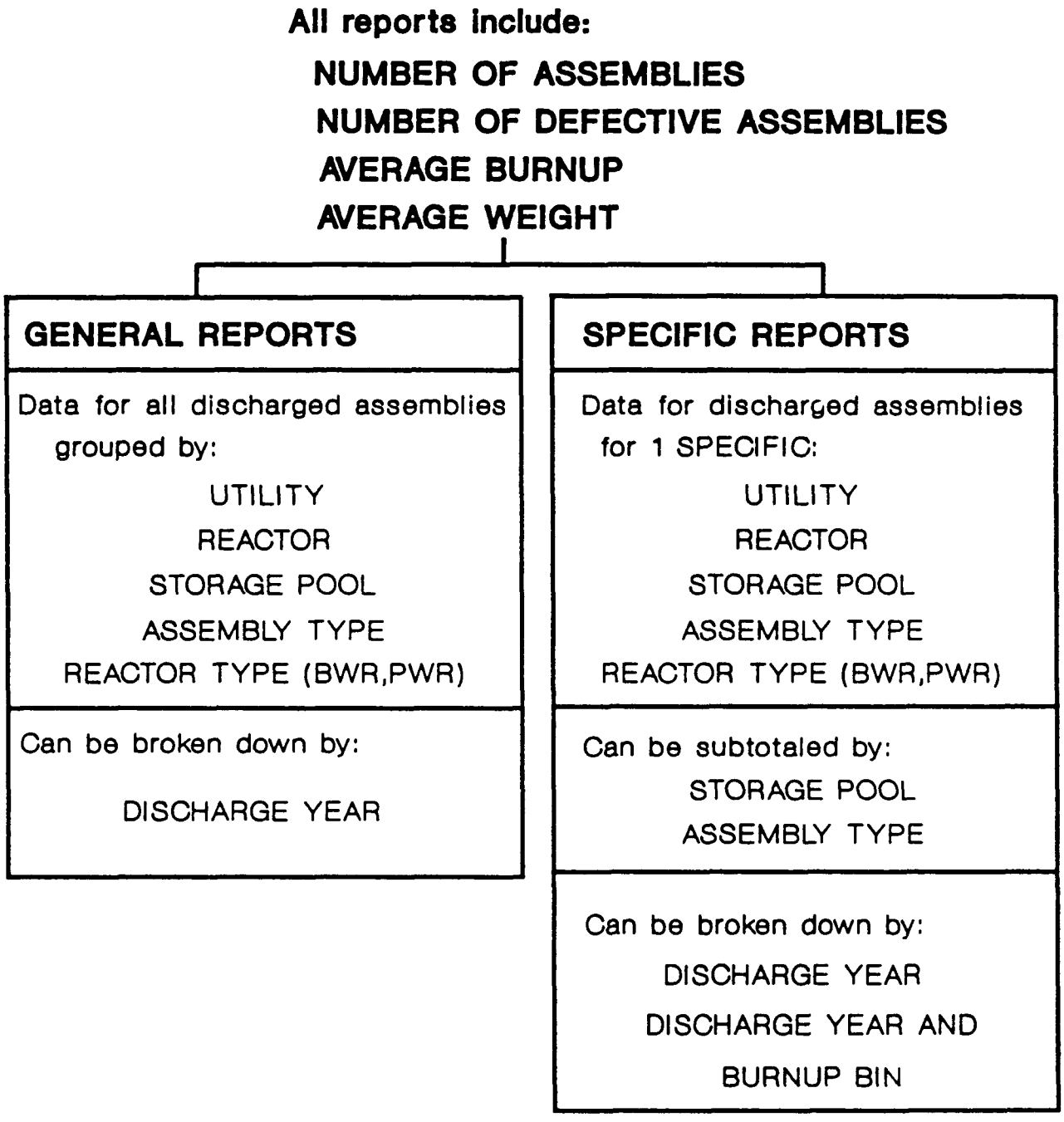


Figure 2.3.2 Overview of LWR Quantities Data Base Projected Data.

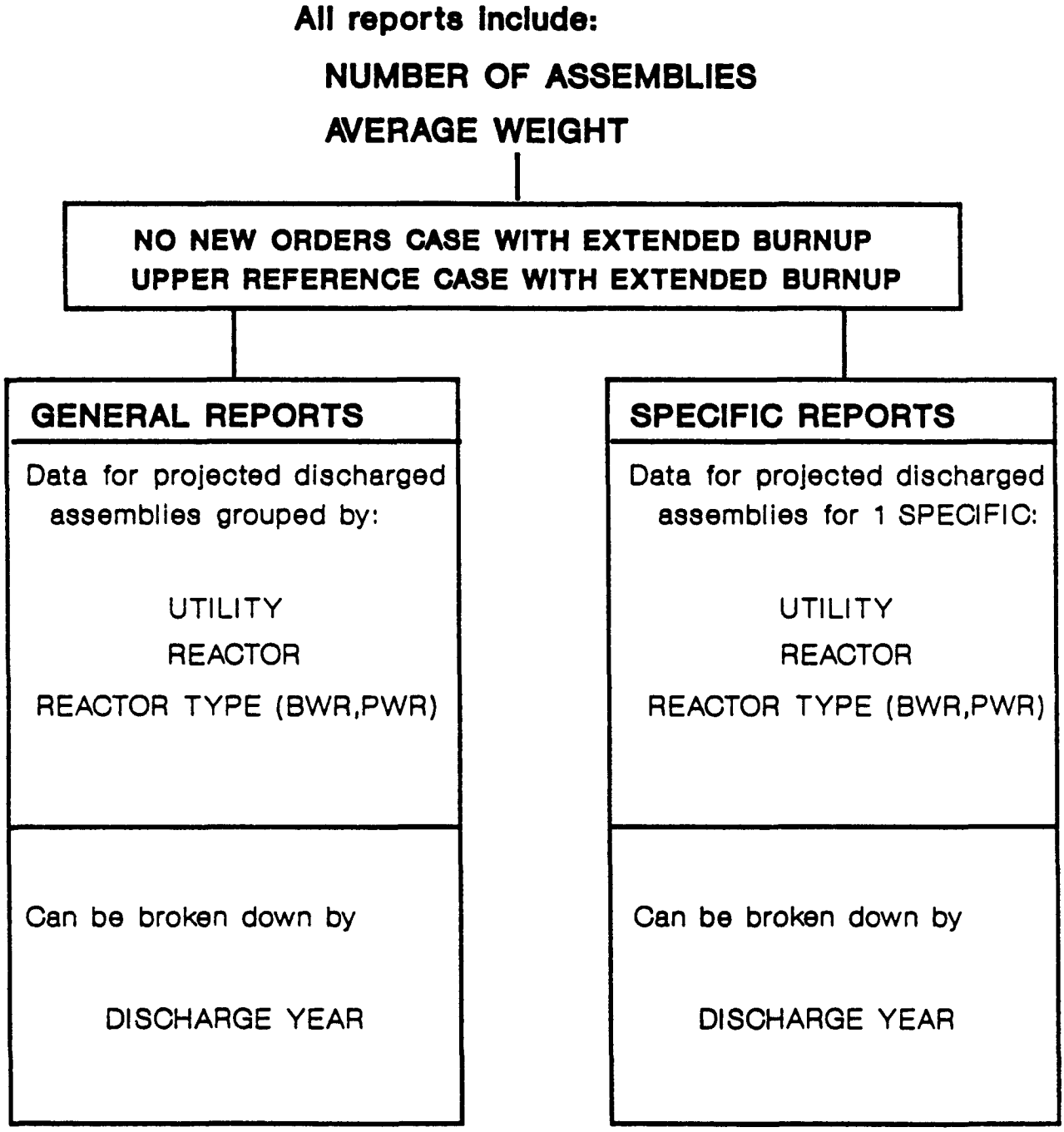


Table 2.3.1 Projected installed nuclear electric power capacity for alternative DOE/EIA scenarios

End of calendar year	End-of-year net capacity, GW(e)		
	No New Orders Case[c]	Lower Reference Case[d]	Upper Reference Case[e]
1985[b]	80	80	80
1986	86	86	92
1987	95	95	100
1988	101	101	104
1989	104	104	105
1990	105	105	105
1991	105	105	105
1992	105	105	107
1993	105	105	107
1994	105	105	109
1995	105	105	111
1996	106	106	111
1997	106	106	111
1998	106	106	112
1999	106	106	112
2000	105	105	111
2001	107	107	117
2002	106	106	123
2003	106	106	129
2004	106	106	134
2005	106	106	140
2006	106	109	145
2007	105	110	151
2008	105	113	156
2009	103	113	162
2010	101	113	167
2011	97	116	173
2012	92	119	178
2013	83	118	183
2014	71	116	189
2015	68	120	194
2016	61	116	199
2017	58	117	204
2018	56	119	209
2019	56	123	214
2020	55	125	219

[a]Projections includes LWRs, Fort St. Vrain HTGR, and the Hanford "N" reactor.

[b]Actual data.

[c]Includes only those reactors currently in operation or greater than 40% complete.

[d]Includes those reactors currently in operation or greater than 40% complete plus a limited number of new reactors beyond the year 2000.

[e]Includes all existing reactors (either completed or under construction) plus additional new reactors beyond the year 2000.

Table 2.3.2 Projected cululative mass of commercial spent fuel discharges for alternative DOE/EIA scenarios

End of calendar year	Cumulative spent fuel discharged,[a,b] MTIHM		
	No New Orders Case	Lower Reference Case	Upper Reference Case
1985[c]	12,400	12,400	12,400
1986	13,751	13,751	13,751
1987	15,273	15,273	15,273
1988	16,981	16,981	17,071
1989	18,910	18,910	19,153
1990	20,928	20,928	21,183
1991	23,006	23,006	23,232
1992	25,145	25,145	25,420
1993	27,220	27,220	27,459
1994	29,188	29,188	29,455
1995	31,181	31,181	31,529
1996	33,152	33,152	33,499
1997	34,994	34,994	35,511
1998	36,907	36,907	37,552
1999	38,858	38,858	39,549
2000	40,782	40,782	41,611
2001	42,721	42,721	43,605
2002	44,650	44,650	45,598
2003	46,616	46,616	47,808
2004	48,594	48,594	50,094
2005	50,445	50,445	52,364
2006	52,446	52,446	54,980
2007	54,558	54,574	57,660
2008	56,470	56,515	60,289
2009	58,591	58,723	63,365
2010	60,883	61,171	66,552
2011	63,059	63,518	69,816
2012	65,268	65,973	73,392
2013	67,616	68,622	77,062
2014	70,211	71,642	81,216
2015	72,348	74,428	85,410
2016	74,030	76,975	89,382
2017	75,717	79,662	93,489
2018	77,031	81,959	97,566
2019	78,104	84,223	101,553
2020	79,286	86,895	105,757

[a]These cumulative values are the sum of projected annual spent fuel discharges that have been smoothed by EIA using a 2-year moving average technique. In addition to LWRs, they include spent fuel from the Fort St. Vrain HTGR and the Hanford "N" reactor.

[b]The projections assume that LWR spent fuel burnup increases from 1986 to 1998 at the rate of 2.0% per year. From 1998 onward, BWR fuel burnup is 31,200 MWd/MTIHM and PWR fuel burnup is 40,000 MWd/MTIHM.

[c]Actual data.

Table 2.3.3. Projected number of spent fuel assemblies by reactor type for the DOE/EIA Upper Reference Case

End of calendar year	BWR[b]		PWR[c]		Total	
	Annual	Cumulative	Annual	Cumulative	Annual	Cumulative
1985	2,605	28,033	2,063	18,319	4,668	46,352
1986	2,255	30,288	2,391	20,710	4,646	50,998
1987	3,812	34,100	2,188	22,898	6,000	56,998
1988	4,433	38,534	2,759	25,657	7,192	64,191
1989	4,362	42,895	3,297	28,954	7,659	71,849
1990	4,213	47,109	2,606	31,560	6,819	78,669
1991	5,044	52,153	3,097	34,656	8,141	86,809
1992	4,054	56,207	3,330	37,986	7,384	94,193
1993	4,747	60,954	2,518	40,505	7,265	101,459
1994	4,246	65,200	3,047	43,551	7,293	108,751
1995	4,807	70,008	2,858	46,410	7,665	116,418
1996	4,356	74,364	2,514	48,923	6,870	123,287
1997	4,455	78,820	3,207	52,131	7,662	130,951
1998	3,960	82,780	2,818	54,949	6,778	137,729
1999	4,840	87,620	2,835	57,783	7,675	145,403
2000	3,845	91,465	3,174	60,958	7,019	152,423
2001	4,180	95,646	2,797	63,754	6,977	159,400
2002	4,345	99,991	2,950	66,705	7,295	166,696
2003	4,213	104,205	3,807	70,512	8,020	174,717
2004	4,730	108,935	3,144	73,655	7,874	182,590
2005	4,499	113,435	3,606	77,262	8,105	190,697
2006	5,809	119,243	4,319	81,581	10,128	200,824
2007	5,275	124,518	3,578	85,159	8,853	209,677
2008	5,347	129,865	4,277	89,435	9,624	219,300
2009	7,756	137,620	4,621	94,057	12,377	231,677
2010	6,689	144,309	4,225	98,281	10,914	242,590
2011	7,134	151,443	5,254	103,535	12,388	254,978
2012	7,514	158,957	5,341	108,876	12,855	267,833
2013	7,585	166,542	5,502	114,378	13,087	280,920
2014	10,710	177,252	6,254	120,632	16,964	297,884
2015	6,029	183,281	6,363	126,995	12,392	310,276
2016	8,933	192,213	5,969	132,964	14,902	325,177
2017	7,948	200,162	6,174	139,138	14,122	339,300
2018	8,476	208,638	6,025	145,163	14,501	353,801
2019	7,657	216,295	5,872	151,036	13,529	367,331
2020	9,428	225,723	6,644	157,679	16,072	383,402

[a]Based on 111 GW(e) installed in the year 2000 and 219 GW(e) installed in the year 2020.

[b]Number of BWR assemblies estimated, based on 0.1818 MTIHM/assembly (historical average).

[c]Number of PWR assemblies estimated, based on 0.4237 MTIHM/assembly (historical average).

Table 2.3.4 Sample Report from LWR Quantities Data Base

PAGE 1

LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	AVG BURNUP (MWd/MTIHM)	AVG WEIGHT (MTIHM)
Alabama Power Co.	1979	46	1	17453	0.459
	1980	53		27244	0.460
	1981	28		28980	0.460
	1982	52		16946	0.458
	1983	130	15	26715	0.459
	1984	78	3	27867	0.458
	1985	150		28125	0.459
*** SUB TOTALS		537	19	25707	0.459
Arkansas Power & Light	1977	50		16533	0.464
	1978	61		26040	0.462
	1979	65		30134	0.463
	1981	98	22	21943	0.450
	1982	130	7	27526	0.437
	1983	64		28948	0.415
	1984	68		31659	0.464
	1985	68		32516	0.427
*** SUB TOTALS		604	29	27018	0.446
Baltimore G&E	1977	32		17973	0.392
	1978	195		22415	0.379
	1979	136		27194	0.383
	1980	71		30644	0.387
	1981	85		32159	0.386
	1982	154		30842	0.388
	1983	99		31875	0.379
	1984	101		31878	0.374
	1985	129		36482	0.387
*** SUB TOTALS		1002		29326	0.383
Boston Edison Co.	1973	20	16	5997	0.194
	1976	132	128	11308	0.193
	1977	428	1	16480	0.193
	1980	92		20506	0.184
	1981	232		20192	0.184
	1983	224		21459	0.184
*** SUB TOTALS		1128	145	17769	0.189
Carolina Power & Light C	1973	53		15858	0.456
	1974	103		25079	0.441
	1975	52		23417	0.454
	1976	56		21640	0.436
	1977	140		7078	0.188
	1978	53		29955	0.456
	1979	219	5	19266	0.240

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

PAGE 2

LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1980	431	13	19362	0.217
	1982	433	2	22848	0.213
	1984	250	2	26002	0.249
	1985	184		24819	0.184
*** SUB TOTALS		1974	22	21254	0.255
Commonwealth Edison Co.	1969	94	32	16762	0.102
	1971	356	98	6940	0.165
	1972	509	2	4454	0.192
	1973	103	97	13094	0.148
	1974	350	173	13000	0.183
	1975	205	134	15558	0.166
	1976	770	261	16565	0.210
	1977	551	68	21855	0.233
	1978	948	187	21026	0.183
	1979	649	109	26832	0.249
	1980	483	64	25661	0.223
	1981	552	127	27866	0.252
	1982	428	47	28167	0.217
	1983	693	15	29130	0.235
	1984	460	64	30132	0.222
	1985	785		25531	0.234
*** SUB TOTALS		7936	1478	21606	0.213
Consolidated Edison Co.	1972	40		25247	0.195
	1974	120		13550	0.190
	1976	72		17677	0.455
	1978	60		28927	0.450
	1979	63		33971	0.449
	1980	54		30692	0.452
	1982	75		32237	0.451
	1984	72		33819	0.451
*** SUB TOTALS		556		25710	0.377
Consumers Power Co.	1974	18	15	13244	0.136
	1975	205		11331	0.411
	1977	20		19955	0.133
	1978	90		13953	0.337
	1979	94		26641	0.319
	1980	22		24746	0.128
	1981	68		30915	0.412
	1982	24		25974	0.126
	1983	90		31304	0.324
	1984	18	1	16630	0.129
	1985	88	2	32489	0.329
*** SUB TOTALS		737	18	21664	0.331

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
Dairyland Power Coop	1972	6	6	11362	0.120
	1973	48	40	13280	0.120
	1975	25	11	15530	0.120
	1977	32	26	16459	0.120
	1979	28	17	13966	0.121
	1980	12	1	15885	0.121
	1982	30	1	16198	0.110
	1983	22		17373	0.109
	1985	28		17995	0.109
*** SUB TOTALS		231	102	15473	0.116
Duke Power Co.	1974	56		11559	0.468
	1976	102		17343	0.467
	1977	191	16	24632	0.465
	1978	185	2	26842	0.464
	1979	131		24932	0.463
	1980	136		27111	0.464
	1981	140	3	29811	0.463
	1982	72		28699	0.464
	1983	137	3	29338	0.464
	1984	166	1	28252	0.462
	1985	240	7	28838	0.461
*** SUB TOTALS		1556	32	26292	0.464
Duquesne Light Co.	1979	35		17554	0.461
	1981	53		26872	0.460
	1983	53		32695	0.459
	1984	77		29222	0.457
*** SUB TOTALS		218		27622	0.459
Florida Power Corporatio	1978	4		9614	0.464
	1979	56		14596	0.464
	1980	48		20786	0.464
	1981	65		25757	0.464
	1983	68		24736	0.468
	1985	65		28443	0.463
*** SUB TOTALS		306		23067	0.465
Florida Power & Light	1974	46		13897	0.453
	1975	81		18573	0.451
	1976	63		28289	0.443
	1977	74		27421	0.451
	1978	112		20503	0.426
	1979	200		27137	0.425
	1980	124		29333	0.400

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1981	170		31024	0.431
	1982	41		33558	0.459
	1983	146	9	30979	0.412
	1984	142		21118	0.418
	1985	137		34594	0.411
*** SUB TOTALS		1336	9	27116	0.426
GPU Nuclear	1971	24		9167	0.193
	1972	136		12220	0.195
	1973	148		16604	0.195
	1974	72		19270	0.195
	1975	112		22765	0.196
	1976	86		18180	0.289
	1977	181		23497	0.273
	1978	241		25268	0.269
	1979	52		26144	0.464
	1980	153		23325	0.180
	1983	207		26637	0.176
*** SUB TOTALS		1412		21689	0.229
Georgia Power Company	1977	4	2	11762	0.188
	1978	24	3	16323	0.187
	1979	188	4	19500	0.187
	1980	76		2724	0.184
	1981	260	29	22013	0.185
	1982	156	25	18357	0.183
	1983	201	10	22947	0.184
	1984	442	25	22932	0.184
	1985	181	3	20801	0.184
*** SUB TOTALS		1532	101	20504	0.184
Iowa Elec. Light & Power	1975	2		4651	0.188
	1976	80		6946	0.188
	1977	74	1	15752	0.188
	1978	120	3	18339	0.188
	1980	88		20641	0.188
	1981	84	1	24318	0.184
	1983	128		26812	0.184
	1985	120		28178	0.184
*** SUB TOTALS		696	5	20982	0.186
Maine Yankee Atomic Powe	1974	152		13591	0.381
	1975	72		11511	0.365
	1977	69		18042	0.390
	1978	129		21777	0.380
	1980	73		30271	0.360

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1981	73		31972	0.386
	1982	73		33018	0.385
	1984	73		33585	0.376
	1985	73		35662	0.375
*** SUB TOTALS		787		24089	0.378
Nebraska Public Power Di	1976	120		10051	0.196
	1977	12		11645	0.196
	1978	60		21923	0.190
	1979	164		25353	0.190
	1980	152		25578	0.188
	1981	112		27379	0.187
	1982	112		29166	0.186
	1983	120		31212	0.185
*** SUB TOTALS		852		24396	0.189
Niagara Mohawk Power Cor	1971	17	17	5701	0.193
	1972	31	29	7970	0.194
	1973	104	104	12682	0.194
	1974	148	31	16807	0.194
	1975	200	55	18870	0.194
	1977	160	11	20713	0.192
	1979	168		26078	0.185
	1981	200		24318	0.184
	1984	216		27274	0.182
*** SUB TOTALS		1244	247	21201	0.189
Norhteast Utilities Serv	0	1		0	0.177
	1972	28	28	8349	0.194
	1974	208	29	14329	0.195
	1975	144	39	16171	0.196
	1976	124	8	19257	0.196
	1977	45		15910	0.396
	1978	124	9	22419	0.194
	1979	220		24150	0.248
	1980	241		27535	0.245
	1981	73		31347	0.388
	1982	192		27105	0.184
	1983	88	3	31476	0.392
	1984	200		26678	0.179
	1985	277		28902	0.241
*** SUB TOTALS		1965	116	23908	0.231
Northern States Power Co	1973	13	13	8066	0.194
	1974	122	86	12672	0.194
	1975	349	126	16818	0.194

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1976	80		18966	0.400
	1977	90		28618	0.351
	1978	89		34094	0.378
	1979	41		29564	0.400
	1980	223		27676	0.262
	1981	185		29027	0.279
	1982	242		28707	0.252
	1983	81		38030	0.380
	1984	235		28902	0.219
	1985	56		37864	0.364
*** SUB TOTALS		1806	225	25660	0.266
Omaha Public Power Distr	1975	25		8601	0.373
	1976	36		21518	0.356
	1977	52		28254	0.364
	1978	44		24013	0.372
	1980	40		30206	0.368
	1981	40		31480	0.365
	1982	19		35261	0.364
	1984	26		36817	0.361
	1985	65		35274	0.358
*** SUB TOTALS		347		28539	0.364
Pacific Gas & Electric C	1965	390		14771	0.074
*** SUB TOTALS		390		14771	0.074
Pennsylvania Power & Lig	1985	192		9035	0.184
*** SUB TOTALS		192		9035	0.184
Philadelphia Electric Co	1976	376	27	10643	0.193
	1977	172	12	18547	0.187
	1978	512	3	21138	0.188
	1979	272	15	24574	0.190
	1980	276		26272	0.185
	1981	216		25804	0.187
	1982	276		25741	0.183
	1983	284		27932	0.183
	1984	292		28566	0.182
	1985	284	94	29615	0.182
*** SUB TOTALS		2960	151	23416	0.186
Portland General Elec.	1978	3	3	16887	0.459
	1980	67	2	27883	0.460
	1981	67		33424	0.460
	1982	55	17	24556	0.459

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1983	39		29577	0.459
	1984	52		31665	0.459
	1985	35		33423	0.459
*** SUB TOTALS		318	22	29807	0.459
Power Auth. of State of	1977	132		9103	0.196
	1978	200		19444	0.274
	1979	76		29875	0.456
	1980	160		22650	0.187
	1981	188		25688	0.187
	1982	76		34203	0.456
	1983	200		26869	0.183
	1985	272		28477	0.260
*** SUB TOTALS		1304		24182	0.247
Public Service of Colora	1979	240		174	0.003
	1981	240		363	0.003
	1984	240		658	0.003
*** SUB TOTALS		720		398	0.003
Public Serv. Elec. & Gas	1979	34		16176	0.460
	1980	64		24919	0.460
	1982	90	2	34253	0.460
	1983	54		18420	0.459
	1984	143	2	26683	0.458
*** SUB TOTALS		385	4	26073	0.459
Rochester Gas & Elec. Co	1972	80	1	19852	0.395
	1974	12		25135	0.383
	1975	25		24013	0.393
	1976	37		25632	0.393
	1977	41		28832	0.391
	1978	41		28586	0.393
	1979	40		29429	0.392
	1980	36		30719	0.393
	1981	28		31257	0.383
	1982	23		32297	0.373
	1983	16		35634	0.373
	1984	29		36640	0.374
	1985	32		37327	0.374
*** SUB TOTALS		440	1	28457	0.388
Sacramento Municipal Uti	1978	48		26906	0.463
	1980	65		32860	0.464
	1981	41		27469	0.461

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1983	53		33367	0.463
	1985	58		31422	0.464
*** SUB TOTALS		265		30734	0.463
South Carolina Elec. & G	1984	44		17600	0.458
	1985	68		26466	0.458
*** SUB TOTALS		112		22983	0.458
Southern California Edis	1970	48		18075	0.366
	1972	49	2	25212	0.367
	1973	56	4	29057	0.360
	1975	53		28875	0.364
	1976	53		31941	0.363
	1978	52	1	31902	0.369
	1980	52		30460	0.369
	1984	65		14743	0.427
*** SUB TOTALS		428	7	26062	0.375
Tennessee Valley Authori	1977	168	8	10479	0.187
	1978	161	22	11038	0.187
	1979	253	4	18334	0.186
	1980	1090	14	22947	0.186
	1981	540	11	22800	0.186
	1982	316	31	23321	0.244
	1983	568	14	26623	0.215
	1984	444		29034	0.270
	1985	260		27954	0.182
*** SUB TOTALS		3800	104	23198	0.205
Toledo Edison Co.	1982	53		23442	0.472
	1983	79		27056	0.472
	1984	65		27962	0.470
*** SUB TOTALS		197		26383	0.471
Virginia Electric & Powe	1974	18	3	14963	0.454
	1975	99	15	20155	0.453
	1976	163	29	22380	0.447
	1977	81		21952	0.450
	1978	43	6	26997	0.456
	1979	103	2	24631	0.458
	1980	135		26603	0.458
	1981	53	3	32180	0.458
	1982	111	4	22874	0.459
	1983	206	37	29120	0.458
	1984	232	28	30428	0.458

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1985	110	4	35229	0.458
*** SUB TOTALS		1354	131	26704	0.456
Wisconsin Electric Power	0	1		0	0.401
	1972	34	33	18612	0.397
	1974	111	22	24453	0.391
	1975	15	3	23862	0.398
	1976	66	2	27154	0.397
	1977	85		32840	0.393
	1978	77	1	31918	0.400
	1979	55		32476	0.401
	1980	40		35013	0.401
	1981	60		32852	0.402
	1982	53	1	35195	0.402
	1983	89	11	35107	0.401
	1984	30	2	35565	0.401
	1985	68	6	34678	0.401
*** SUB TOTALS		784	81	31019	0.398
Wisconsin Public Service	1976	11		18724	0.398
	1977	45		27213	0.392
	1978	41		34162	0.391
	1979	13		33883	0.400
	1980	33		33243	0.401
	1981	41		31776	0.401
	1982	37		33752	0.391
	1983	29		33123	0.394
	1984	57		32008	0.383
	1985	45		34119	0.379
*** SUB TOTALS		352		31934	0.391
Yankee Atomic Electric C	1972	36		23864	0.273
	1974	37		25833	0.273
	1975	40		27970	0.241
	1977	36		28157	0.239
	1978	40		27330	0.235
	1981	36		29913	0.235
	1982	40		28963	0.234
	1984	36		27455	0.234
	1985	40		29396	0.233
*** SUB TOTALS		341		27684	0.244
Connecticut Yankee Atomi	1970	51		18748	0.420
	1971	52		26220	0.421
	1972	53		30799	0.417
	1973	55		27241	0.408

Table 2.3.4 Sample Report from LWR Quantities Data Base (cont.)

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

UTILITY	DISCHARGE YEAR	NUMBER ASSEMBLIES	DEFECTIVE ASSEMBLIES	BURNUP (MWd/MTIHM)	WEIGHT (MTIHM)
	1975	49		32819	0.411
	1976	53		31074	0.411
	1977	53		33434	0.412
	1979	49	36	32741	0.412
	1980	53	7	34276	0.412
	1981	53		33296	0.412
	1983	53		33715	0.413
	1984	53		35236	0.412
*** SUB TOTALS		627	43	30809	0.413
Indiana and Michigan Ele	1976	63	2	19026	0.453
	1978	64		29029	0.455
	1979	137	3	25016	0.456
	1980	65		31899	0.429
	1981	156	3	29449	0.447
	1982	145	9	31842	0.445
	1983	66		31373	0.427
	1984	91		34249	0.459
	1985	94	10	30003	0.427
*** SUB TOTALS		881	27	29264	0.445
Vermont Yankee Nuclear P	1973	50		3706	0.189
	1974	328		9197	0.193
	1977	112		18924	0.184
	1978	106		18880	0.184
	1979	182		19958	0.184
	1980	92		24509	0.184
	1981	120		25694	0.184
	1983	106		28912	0.183
	1984	106		28564	0.183
*** SUB TOTALS		1202		18624	0.186
*** GRAND TOTALS		45814	3119	23302	0.272

Table 2.3.5 Sample Report from LWR Quantities Data Base*

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LWR QUANTITIES DATABASE
Historical Data
Data Broken Down By: Assembly Type, Discharge Year, and Burnup Bin
Discharged Assemblies for Reactor: PRAIRIE ISLAND 1

ASSEMBLY TYPE	DISCHARGE YEAR	BURNUP BIN	NUMBER ASSMB	AVG BURNUP (MWd/MITHM)	AVG WEIGHT (MTIHM)
West. 14 x 14 ZCB	1976	15000-20000	40	18676	0.399
	1977	25000-30000	35	29567	0.394
	1978	35000-40000	41	35218	0.393
	1979	25000-30000	40	29361	0.400
		35000-40000	1	37686	0.396
	1980	35000-40000	40	35705	0.400
	1981	30000-35000	39	33810	0.400
		40000-45000	1	42669	0.402
	1982	35000-40000	1	39388	0.401
	1983	35000-40000	1	37699	0.394
ANF 14 x 14 WE	1982	35000-40000	40	37079	0.379
	1983	35000-40000	40	37353	0.379
ANF 14 x 14 TOP ROD	1984	25000-30000	6	26000	0.365

* Prairie Island 1 has also used West. 14 x 14 OFA; none of these had been permanently discharged through December 31, 1985, so they are not included in the historical data.

Table 2.3.6 Sample Report from LWR Quantities Data Base

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Projected Assemblies by Reactor through 2020

REACTOR	DISCHARGE YEAR	NUMBER ASSEMBLIES	AVG WEIGHT (MTIHM)
FARLEY 1	1986	59	0.460
	1988	60	0.461
	1989	58	0.461
	1991	69	0.461
	1992	61	0.461
	1994	60	0.461
	1995	61	0.461
	1997	58	0.461
	1998	56	0.461
	2000	62	0.461
	2001	54	0.461
	2003	65	0.461
	2004	58	0.461
	2006	63	0.461
	2007	61	0.461
	2009	60	0.461
	2010	61	0.461
	2012	58	0.461
	2013	60	0.461
	2015	48	0.461
	2016	52	0.461
	2017	157	0.461
*** SUB TOTALS		1401	0.461
FARLEY 2	1986	63	0.459
	1987	53	0.462
	1989	61	0.462
	1990	58	0.461
	1992	63	0.461
	1993	57	0.461
	1995	63	0.461
	1996	51	0.461
	1998	58	0.461
	1999	54	0.461
	2001	58	0.461
	2002	48	0.461
	2004	60	0.461
	2005	48	0.461
	2007	63	0.461
	2008	47	0.461
	2010	63	0.461
	2011	58	0.461
	2013	60	0.461
	2014	47	0.461
	2016	52	0.461
	2017	54	0.461
	2019	46	0.461
	2020	55	0.461
*** SUB TOTALS		1340	0.461

Table 2.3.7 Sample Report from LWR Quantities Data Base

PAGE 1

LWR QUANTITIES DATABASE
Projected Data: Upper Reference Case with Extended Burnup
Projected Assemblies by Reactor Type through 2020

REACTOR TYPE	DISCHARGE YEAR	NUMBER ASSEMBLIES	AVG WEIGHT (MTIHM)
BWR	1986	2342	0.179
	1987	3016	0.182
	1988	3581	0.181
	1989	4121	0.181
	1990	3476	0.181
	1991	4017	0.180
	1992	4264	0.179
	1993	3602	0.179
	1994	4607	0.180
	1995	3526	0.179
	1996	4304	0.179
	1997	3783	0.180
	1998	3929	0.174
	1999	4032	0.179
	2000	4078	0.178
	2001	3656	0.179
	2002	4277	0.179
	2003	4501	0.180
	2004	4184	0.178
	2005	4271	0.180
	2006	5581	0.179
	2007	4464	0.179
	2008	5071	0.179
	2009	7083	0.180
	2010	6072	0.178
	2011	6932	0.179
	2012	7332	0.179
	2013	6236	0.181
	2014	9590	0.180
	2015	8222	0.181
	2016	7092	0.181
	2017	7749	0.181
	2018	6607	0.181
	2019	6870	0.182
	2020	8655	0.181
*** SUB TOTALS		181123	0.180

Table 2.3.7 Sample Report from LWR Quantities Data Base (cont.)

PAGE 2

LWR QUANTITIES DATABASE
Projected Data: Upper Reference Case with Extended Burnup
Projected Assemblies by Reactor Type through 2020

REACTOR TYPE	DISCHARGE YEAR	NUMBER ASSEMBLIES	AVG WEIGHT (MTIHM)
PWR	1986	2266	0.433
	1987	2315	0.432
	1988	2793	0.427
	1989	3116	0.435
	1990	3156	0.435
	1991	2956	0.432
	1992	3262	0.438
	1993	3114	0.435
	1994	2703	0.433
	1995	3358	0.438
	1996	2837	0.433
	1997	3030	0.435
	1998	3029	0.431
	1999	2930	0.436
	2000	3144	0.436
	2001	3062	0.440
	2002	2827	0.436
	2003	3193	0.436
	2004	3571	0.441
	2005	3431	0.441
	2006	3666	0.437
	2007	4310	0.441
	2008	3859	0.438
	2009	4204	0.434
	2010	4824	0.439
	2011	4751	0.433
	2012	5290	0.433
	2013	5871	0.438
	2014	5598	0.442
	2015	6212	0.436
	2016	6210	0.438
	2017	6126	0.440
	2018	6635	0.437
	2019	6359	0.433
	2020	6051	0.435
*** SUB TOTALS		140059	0.436
*** GRAND TOTALS		321182	0.292

2.4 RADIOLOGICAL PROPERTIES OF INTACT SPENT FUEL

2.4.1 Overview

The long-term disposal of LWR spent fuel in a mined geologic repository requires specific knowledge concerning the radioactive components in discharged fuel assemblies as a function of time. ORIGEN2 (Croff 1980) was used to generate the data presented in this section. Two reference LWR's were modeled, a PWR (Westinghouse 1972) and a BWR (General Electric 1973), and results were obtained for several burnups (5000-MWd increments to 60,000 MWd for the PWR and 40,000 MWd for the BWR). All values were predicated on the burnup occurring over a 3- to 4-year irradiation period with normal downtimes for refueling. Although the physical characteristics and structural material distribution vary from vendor to vendor, the radiological characteristics of the spent fuel are not very different. (This is not true for the hardware components; see Sections 2.7 and 2.8.)

The composition (in grams), total radioactivity (in curies), and thermal power (in watts) of the significant nuclides in one MTIHM have been tabulated for decay periods from 1 to 1 million years (Roddy 1986). These data have been downloaded from the mainframe computer to a personal computer data base, the LWR Radiological Data Base. Also included in the data base are the neutron and photon energy spectra emitted by the assembly. The addition of the photon and neutron spectra is the primary difference between the LWR Radiological Data Base and Roddy's National Waste Terminal Storage Program Data Base. A few decay times have also been added. The data base is described in greater detail in Appendix 2C, User's Guide to the LWR Radiological Data Base. For quick reference and to summarize this information, a series of tables (Tables 2.4.1 to 2.4.24) and illustrations (Figures 2.4.1 to 2.4.4) have been included.

Three separate and distinct categories for the radioactivity produced have been included in these calculations. Activation products include low-atomic number impurities in the fuel and structural

materials. The actinides include the heavy isotopes, their decay daughters, and the final stable nuclides. Fission products comprise all nuclides that have a significant fission-product yield (binary or ternary) plus some nuclides resulting from neutron capture by fission products. The tables included list all isotopes that contribute more than 0.1% to the total for each specified time since discharge. (The PC data base lists all isotopes, with optional cutoffs at 1%, 0.1%, 0.01%, and 0.001%.)

2.4.2 Major Contributors to Radioactivity

Although the three categories display minor differences in the radioactivity for the various reactor types and burnups, major variations occur with decay time. The fission products (Tables 2.4.1 to 2.4.4) dominate the total radioactivity for the first 100 years after discharge; during an interim period (100 to 300 years), both fission products and actinides contribute to the total; the long-lived actinides (Tables 2.4.5 to 2.4.8) dominate after 300 years.

The major contributors to the total radioactivity one year after discharge include four decay chains, 90Sr → 90Y, 106Ru → 106Rh, 137Cs → 137Ba, and 144Ce → 144Pr; one additional fission product, 134Cs; and one actinide, 241Pu. After 100 years, the total activity will have decreased by a factor of 40, with the fission products (90Sr, 90Y, 137Cs, and ^{137m}Ba) supplying about 80% of the total. The long-lived actinides control the activity after 1 (>98%) and 10 (>94%) millennia. The dominant nuclides include 239Pu, 240Pu, and 241Am after 1000 years; 239Np, 239Pu, 240Pu, and 243Am dominate after 10,000 years. Following extremely long storage (100,000 years), one major fission product, 99Tc, one reactor-produced actinide, 239Pu, and the naturally occurring radioactive isotopes present in the uranium decay chain generate the major quantities of radioactivity.

2.4.3 Major Contributors to Thermal Power

The heat generated by a fuel assembly is an important factor in the design of repositories and storage/shipping casks. The thermal power (Tables 2.4.9 to 2.4.12) generated by a discharged fuel assembly initially comes from fission products. The heat output from the

actinides is approximately equivalent to that from the fission products after a decay period of 60 to 70 years. The contribution from the activation products is small at all decay times.

The initial loadings (1 year) placed on a storage facility stem primarily from three fission products, 106Rh, 134Cs, and 144Pr, all of which exhibit short half-lives. The thermal power of spent fuel decreases by a factor of 6 after the initial 10 years of aging. The major sources of thermal power at this point are 90Y, 137Cs, and $137^{m}Ba$ for all cases plus 238Pu and 244Cm for extended-burnup cases. The power output decreases by an additional factor of 5 after 100 years of cooling. The effects from fission products decrease significantly after discharge of the fuel and contribute 1% or less after about 300 years. During the intermediate storage periods (100 to 1000 years), the actinide isotopes of importance are 238Pu, 239Pu, and 241Am; 239Pu and 250Pu are the major sources of heat in the 10,000-year timeframe; 240Pu is the major heat generator at 100,000 years.

2.4.4 Neutron Sources

Neutrons are generated from a discharged fuel assembly by two mechanisms, spontaneous fission (Tables 2.4.13 to 2.4.16) and alpha interactions (Tables 2.4.17 to 2.4.20) with an isotope. Spontaneous fission produces more than 80% of the neutrons for all but the intermediate decay periods, when its contribution is reduced to about 60%. The curium nuclides, 242Cm and 244Cm, dominate this production during the first 10 years; plutonium isotopes (the specific isotopes depending on the reactor type and burnup) are the major contributors in the 10,000- to 100,000-year timeframe. With a half-life of nearly 400,000 years, 242Pu is the only isotope of consequence after 100,000 years of storage. A mixture of plutonium and curium nuclides, along with 241Am, produces the neutrons at 1000 years.

2.4.5 Photon Production

The ORIGEN2 photon data base supplies the number of photons per decay of an isotope in an 18-energy-group structure. Primary gamma rays, X-rays, conversion photons, (α,n) gamma rays, prompt and fission

product gamma rays from spontaneous fission, and bremsstrahlung radiations have been included in this compilation (Tables 2.4.21 to 2.4.24).

The number of photons produced by the activation products never exceeds 3% of the total, and, as might be expected, 60Co is the major contributor immediately after discharge. Minor contributions at 5 and 10 years are 95Zr, 95Nb, and 54Mn. Nickel-63 and 94Nb are the chief nuclides after a century; 94Nb and 93Zr are the only isotopes of consequence after 1000 years. At 100,000 years, 93Zr is the only activation product contributing to the photon spectra.

Several fission products produce photons in the first few years after fuel discharge. The percentage of their contribution drops from about 99% at 1 year to 90% at 100 years. Ultimately, their contribution drops to less than 1% at decay times greater than 1000 years. The major B-emitting isotopes at 1 year include 106Rh, 144Pr, and 134Cs. After one decade, 90Sr, 90Y, and 137Ba become the isotopes of importance. Strontium-90 becomes relatively inconsequential after 100 years.

The actinides and their daughters are relatively poor photon generators and exceed the output of the fission products only after about 200 years of storage. After 1000 years, two americium isotopes, 241Am and 243Am, and two plutonium isotopes, 239Pu and 240Pu, predominate in varying amounts up to 100,000 years.

2.4.6 References for Section 2.4

Croff 1980. A.T. Croff, ORIGEN2 - a Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, ORNL-5621, July 1980.

General Electric 1973. General Electric Standard Safety Analysis Report, BWR/6, DOCKET STN 50-447, 1973.

Roddy 1986. J.W. Roddy, et al., Physical and Decay Characteristics of Commercial LWR Spent Fuel, ORNL/TM-9591/V1&R1, January 1986.

Westinghouse 1972. Westinghouse Nuclear Energy Systems, RESAR-3, Reference Safety Analysis Report, DOCKET STN 50-480, 1972.

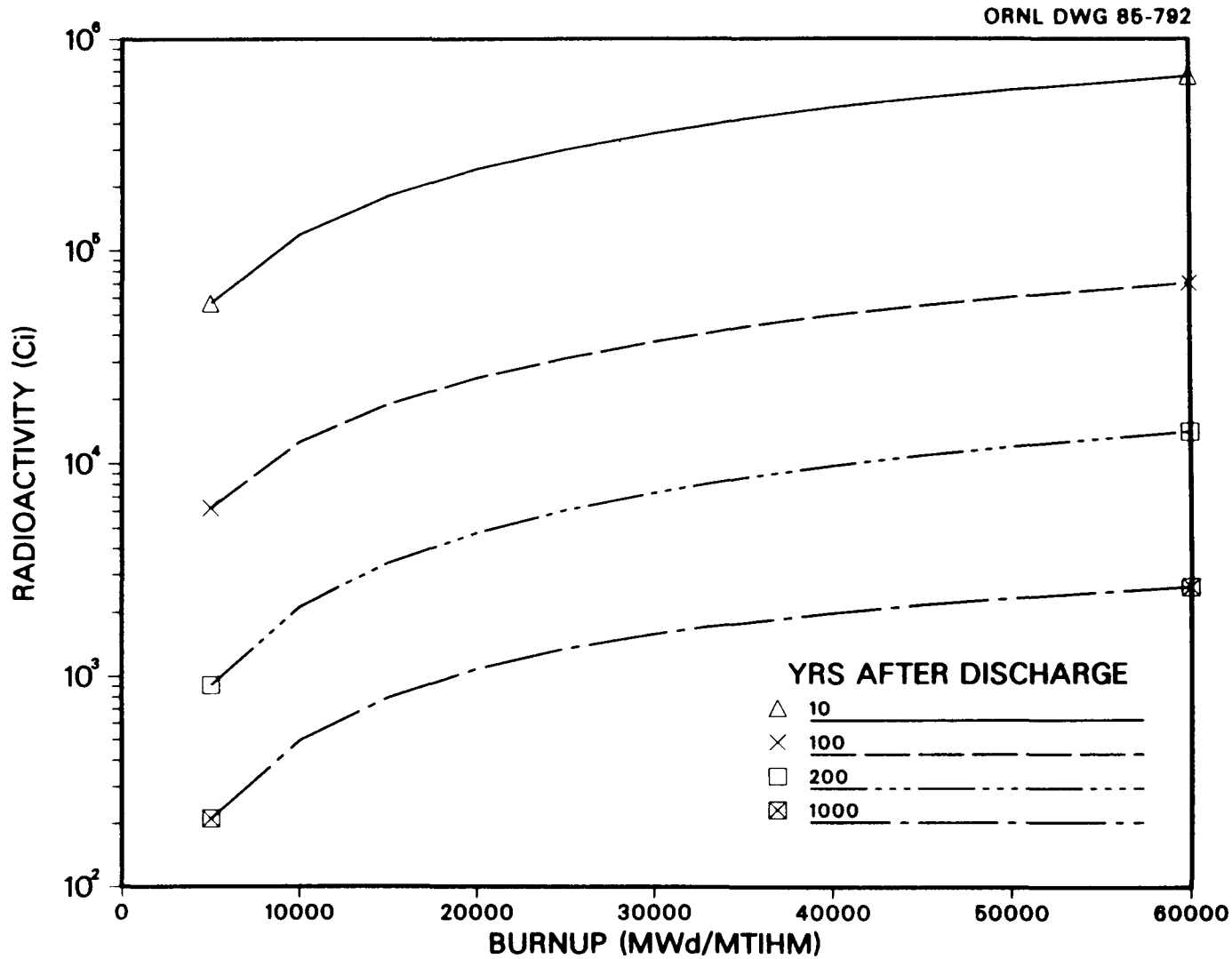

Figure 2.4.1. Radioactivity produced by 1 metric ton of initial heavy metal for a PWR (Source: Roddy 1986)

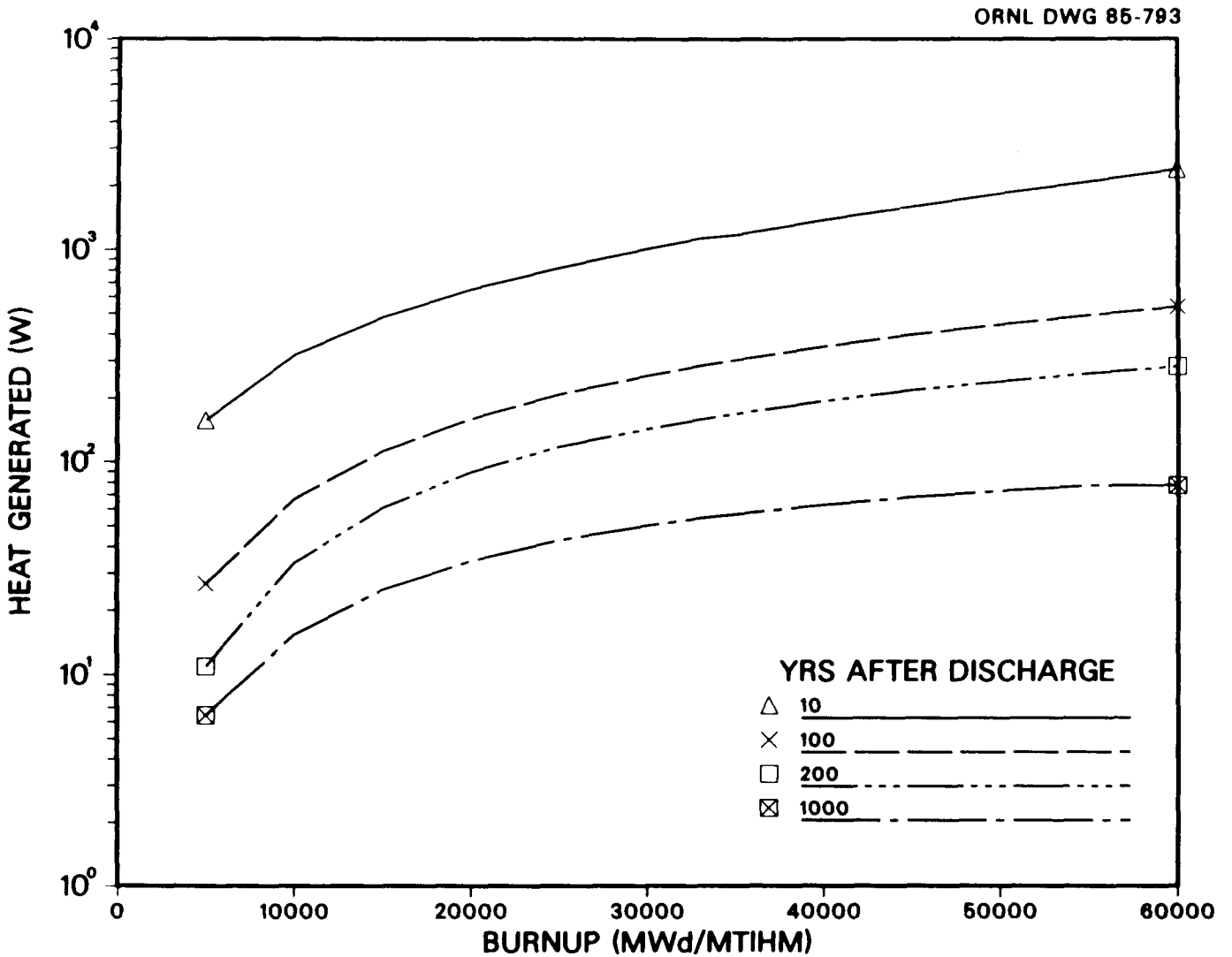

Figure 2.4.2. Heat generated by 1 metric ton of initial heavy metal for a PWR (Source: Roddy 1986)

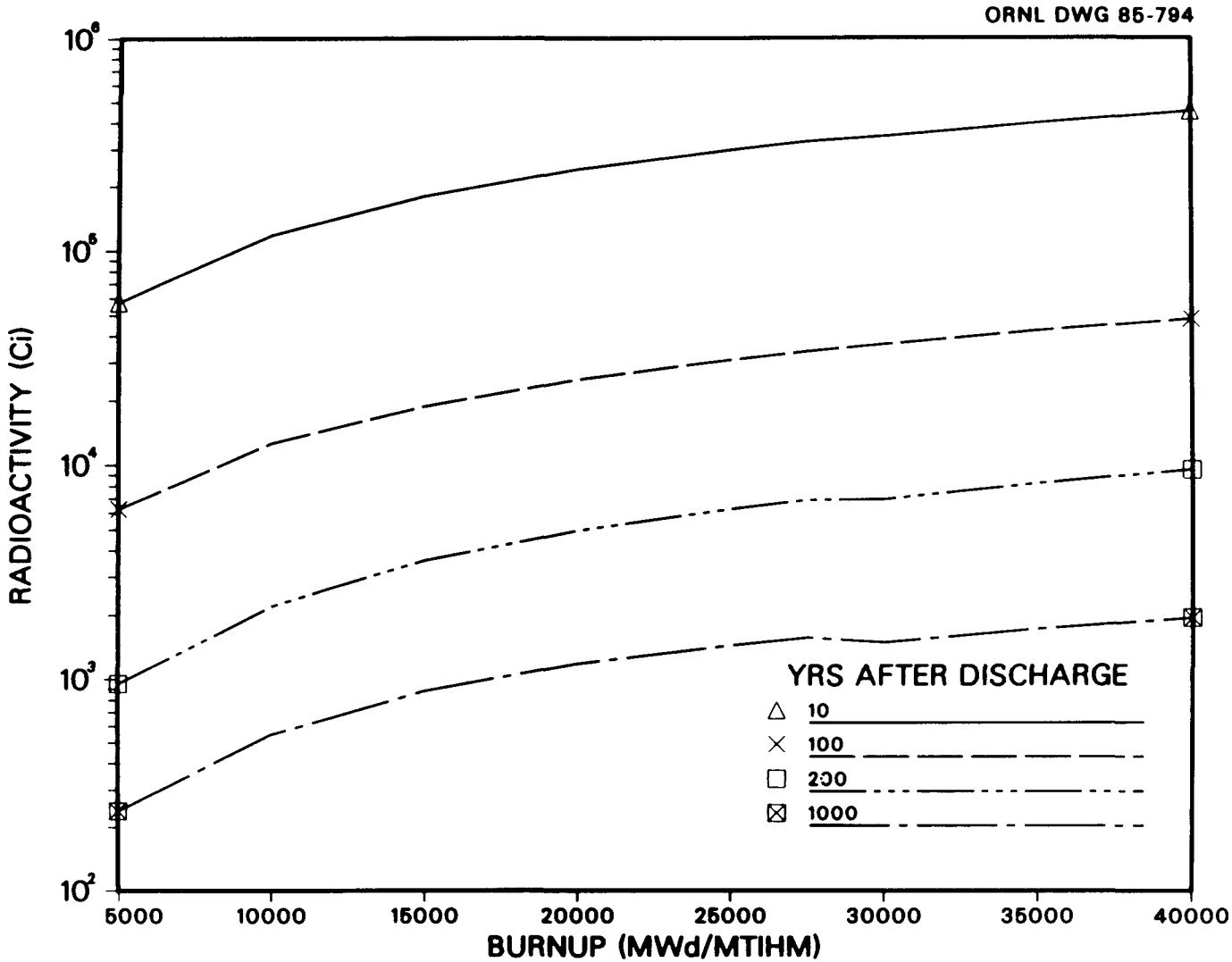

Figure 2.4.3. Radioactivity produced by 1 metric ton of initial heavy metal for a BWR (Source: Roddy 1986)

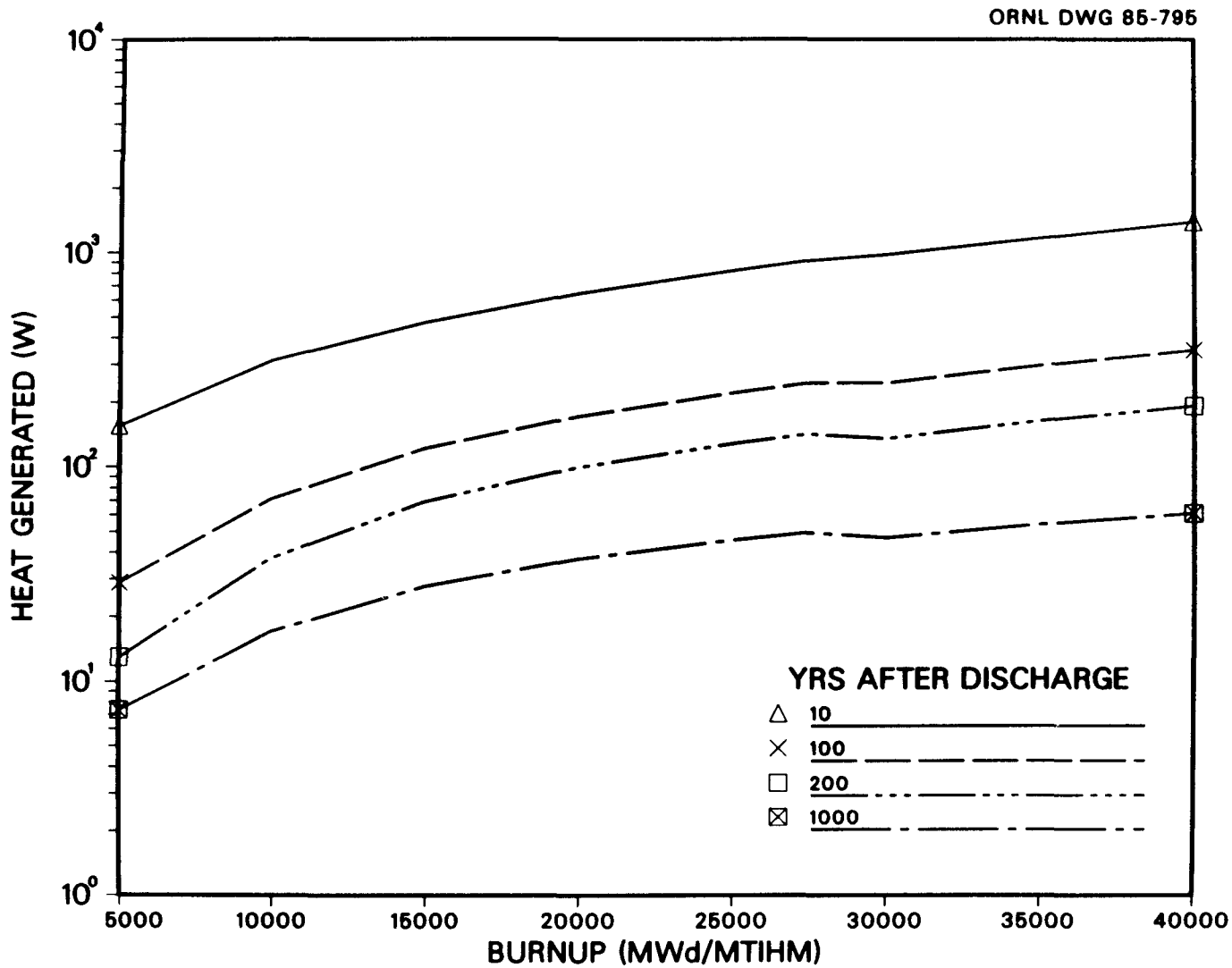

Figure 2.4.4. Heat generated by 1 metric ton of initial heavy metal for a BWR (Source: Roddy 1986)

Table 2.4.1 Variation of radioactivity (Ci/MTIHM) for significant activation- and fission- product nuclides as a function of time since discharge from a 60,000 MWd/MTIHM PWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	1.17E+3	7.09E+2	4.54E+0	-	-	-
C-14[c]	2.44E+0	2.44E+0	2.41E+0	2.16E+0	7.27E-1	-
Mn-54[c]	4.59E+2	-	-	-	-	-
Fe-55[c]	5.24E+3	4.76E+2	-	-	-	-
Co-58[c]	2.13E+2	-	-	-	-	-
Co-60[c]	9.54E+3	2.92E+3	-	-	-	-
Ni-59[c]	6.40E+0	6.40E+0	6.39E+0	6.34E+0	5.87E+0	2.69E+0
Ni-63[c]	1.05E+3	9.83E+2	4.98E+2	-	-	-
Zn-65[c]	4.78E+1	-	-	-	-	-
Se-79	-	-	-	-	6.45E-1	2.47E-1
Kr-85	1.34E+4	7.48E+3	2.22E+1	-	-	-
Sr-89	4.53E+3	-	-	-	-	-
Sr-90	1.14E+5	9.16E+4	1.08E+4	-	-	-
Y-90	1.14E+5	9.16E+4	1.08E+4	-	-	-
Y-91	1.22E+4	-	-	-	-	-
Zr-93[b]	3.32E+0	3.32E+0	3.32E+0	3.32E+0	3.30E+0	3.17E+0
Zr-95[b]	2.93E+4	-	-	-	-	-
Nb-93m[b]	-	-	3.14E+0	3.15E+0	3.14E+0	3.01E+0
Nb-94[c]	-	-	-	2.18E+0	1.61E+0	7.43E-2
Nb-95[b]	6.59E+4	-	-	-	-	-
Tc-99	2.11E+1	2.11E+1	2.11E+1	2.10E+1	2.04E+1	1.52E+1
Ru-103	2.84E+3	-	-	-	-	-
Ru-106	3.84E+5	7.88E+2	-	-	-	-
Rh-106	3.84E+5	7.88E+2	-	-	-	-
Pd-107	-	-	-	2.43E-1	2.43E-1	2.41E-1
Ag-110m	3.72E+3	-	-	-	-	-
Sn-119m[b]	2.47E+3	-	-	-	-	-
Sn-126	1.47E+0	1.47E+0	1.47E+0	1.46E+0	1.37E+0	7.35E-1
Sb-125[b]	1.80E+4	1.89E+3	-	-	-	-
Sb-126	-	-	-	2.04E-1	1.92E-1	1.03E-1
Sb-126m	-	-	-	1.46E+0	1.37E+0	7.35E-1
Te-125m[b]	4.38E+3	4.62E+2	-	-	-	-
I-129	5.68E-2	5.68E-2	5.68E-2	5.68E-2	5.68E-2	5.66E-2
Cs-134	2.62E+5	1.27E+4	-	-	-	-
Cs-135	-	-	-	7.66E-1	7.64E-1	7.43E-1
Cs-137	1.78E+5	1.44E+5	1.80E+4	-	-	-
Ba-137m	1.68E+5	1.37E+5	1.71E+4	-	-	-
Ce-144	4.29E+5	1.42E+2	-	-	-	-
Pr-144	4.29E+5	1.42E+2	-	-	-	-
Pr-144m	5.14E+3	1.70E+0	-	-	-	-
Pm-147	9.39E+4	8.71E+3	-	-	-	-
Sm-151	5.30E+2	4.95E+2	2.47E+2	2.42E-1	-	-
Eu-154	2.33E+4	1.13E+4	7.99E+0	-	-	-
Eu-155	1.42E+4	4.05E+3	-	-	-	-
OTHER	7.55E+3	2.29E+2	1.22E+1	2.40E+0	9.89E-1	7.63E-2
SUBTOTAL						
A.P.[d]	2.59E+4	4.79E+3	5.11E+2	1.18E+1	8.71E+0	3.24E+0
F.P.[e]	2.75E+6	5.14E+5	5.70E+4	3.22E+1	3.10E+1	2.38E+1
TOTAL	2.79E+6	5.18E+5	5.75E+4	4.40E+1	3.98E+1	2.71E+1

[a] Nuclides contributing >0.1% are listed.

[b] Both activation and fission products contribute to this nuclide.

[c] Only activation products contribute to this nuclide.

[d] A.P. = Activation products.

[e] F.P. = Fission products.

Table 2.4.2 Variation of radioactivity (Ci/MTIHM) for significant activation- and fission- product nuclides as a function of time since discharge from a 33,000 MWd/MTIHM PWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	7.69E+2	4.64E+2	2.97E+0	-	-	-
C-14[c]	1.55E+0	1.55E+0	1.53E+0	1.38E+0	4.63E-1	-
Mn-54[c]	3.91E+2	-	-	-	-	-
Fe-55[c]	4.28E+3	3.89E+2	-	-	-	-
Co-58[c]	1.92E+2	-	-	-	-	-
Co-60[c]	6.97E+3	2.12E+3	-	-	-	-
Ni-59[c]	5.15E+0	5.15E+0	5.15E+0	5.11E+0	4.72E+0	2.17E+0
Ni-63[c]	6.97E+2	6.52E+2	3.31E+2	3.76E-1	-	-
Zn-65[c]	4.72E+1	-	-	-	-	-
Se-79	-	-	-	-	3.67E-1	1.41E-1
Kr-85	8.69E+3	4.85E+3	1.44E+1	-	-	-
Sr-89	5.72E+3	-	-	-	-	-
Sr-90	7.08E+4	5.72E+4	6.71E+3	-	-	-
Y-90	7.08E+4	5.72E+4	6.71E+3	-	-	-
Y-91	1.49E+4	-	-	-	-	-
Zr-93[b]	1.93E+0	1.93E+0	1.93E+0	1.93E+0	1.92E+0	1.84E+0
Zr-95[b]	3.14E+4	-	-	-	-	-
Nb-93m[b]	-	-	-	1.83E+0	1.83E+0	1.75E+0
Nb-94[c]	-	-	-	1.24E+0	9.10E-1	4.21E-2
Nb-95[b]	7.07E+4	-	-	-	-	-
Tc-99	1.31E+1	1.31E+1	1.30E+1	1.30E+1	1.26E+1	9.43E+0
Ru-103	2.59E+3	-	-	-	-	-
Ru-106	2.68E+5	5.50E+2	-	-	-	-
Rh-106	2.68E+5	5.50E+2	-	-	-	-
Pd-107	-	-	-	1.12E-1	1.12E-1	1.11E-1
Ag-110m	1.52E+3	-	-	-	-	-
Sn-119m[b]	2.14E+3	-	-	-	-	-
Sn-126	7.76E-1	7.76E-1	7.76E-1	7.71E-1	7.24E-1	3.88E-1
Sb-125[b]	1.22E+4	1.29E+3	-	-	-	-
Sb-126	-	-	-	1.08E-1	1.01E-1	5.44E-2
Sb-126m	-	-	-	7.71E-1	7.24E-1	3.88E-1
Te-125m[b]	2.98E+3	3.14E+2	-	-	-	-
I-129	3.15E-2	3.15E-2	3.15E-2	3.15E-2	3.15E-2	3.14E-2
Cs-134	1.08E+5	5.22E+3	-	-	-	-
Cs-135	-	-	-	3.45E-1	3.44E-1	3.35E-1
Cs-137	1.01E+5	8.21E+4	1.03E+4	-	-	-
Ba-137m	9.56E+4	7.77E+4	9.71E+3	-	-	-
Ce-144	4.51E+5	1.49E+2	-	-	-	-
Pr-144	4.51E+5	1.49E+2	-	-	-	-
Pr-144m	5.41E+3	1.79E+0	-	-	-	-
Pm-147	1.02E+5	9.48E+3	-	-	-	-
Sm-151	3.55E+2	3.31E+2	1.66E+2	1.62E-1	-	-
Eu-154	9.69E+3	4.69E+3	3.32E+0	-	-	-
Eu-155	5.62E+3	1.60E+3	-	-	-	-
OTHER	6.81E+3	3.80E+1	8.70E+0	9.90E-1	6.70E-2	5.60E-2
SUBTOTAL						
A.P.[d]	1.95E+4	3.48E+3	3.40E+2	8.38E+0	6.36E+0	2.46E+0
F.P.[e]	2.16E+6	3.04E+5	3.36E+4	1.92E+1	1.86E+1	1.42E+1
TOTAL	2.18E+6	3.07E+5	3.39E+4	2.76E+1	2.49E+1	1.67E+1

[a]Nuclides contributing >0.1% are listed.

[b]Both activation and fission products contribute to this nuclide.

[c]Only activation products contribute to this nuclide.

[d]A.P. = Activation products.

[e]F.P. = Fission products.

Table 2.4.3 Variation of radioactivity (Ci/MTIHM) for significant activation- and fission- product nuclides as a function of time since discharge from a 40,000 MWd/MTIHM BWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	8.43E+2[a]	5.09E+2	3.26E+0	-	-	-
C-14[c]	2.05E+0	2.05E+0	2.02E+0	1.82E+0	6.11E-1	-
Mn-54[c]	1.49E+2	-	-	-	-	-
Fe-55[c]	2.54E+3	2.31E+2	-	-	-	-
Co-58[c]	3.75E+1	-	-	-	-	-
Co-60[c]	2.62E+3	8.01E+2	-	-	-	-
Ni-59[c]	1.39E+0	1.39E+0	1.39E+0	1.38E+0	1.27E+0	5.84E-1
Ni-63[c]	2.08E+2	1.94E+2	9.84E+1	-	-	-
Zn-65[c]	3.56E+1	-	-	-	-	-
Se-79	-	-	-	4.80E-1	4.36E-1	1.67E-1
Kr-85	9.52E+3	5.32E+3	1.58E+1	-	-	-
Sr-89	3.59E+3	-	-	-	-	-
Sr-90	8.20E+4	6.62E+4	7.77E+3	-	-	-
Y-90	8.20E+4	6.62E+4	7.77E+3	-	-	-
Y-91	9.41E+3	-	-	-	-	-
Zr-93[b]	2.56E+0	2.56E+0	2.56E+0	2.56E+0	2.55E+0	2.45E+0
Zr-95[b]	2.18E+4	-	-	-	-	-
Nb-93m[b]	-	-	-	2.44E+0	2.43E+0	2.33E+0
Nb-95[b]	4.89E+4	-	-	-	-	-
Tc-99	1.56E+1	1.56E+1	1.56E+1	1.56E+1	1.51E+1	1.13E+1
Ru-103	1.86E+3	-	-	-	-	-
Ru-106	2.28E+5	4.67E+2	-	-	-	-
Rh-106	2.28E+5	4.67E+2	-	-	-	-
Pd-107	-	-	-	1.40E-1	1.40E-1	1.39E-1
Ag-110m	1.63E+3	-	-	-	-	-
Sn-119m[b]	3.83E+3	-	-	-	-	-
Sn-126	8.88E-1	8.88E-1	8.87E-1	8.82E-1	8.28E-1	4.44E-1
Sb-125[b]	1.25E+4	1.31E+3	-	-	-	-
Sb-126	-	-	-	1.24E-1	1.16E-1	6.22E-2
Sb-126m	-	-	-	8.82E-1	8.28E-1	4.44E-1
Te-125m[b]	3.04E+3	3.20E+2	-	-	-	-
I-129	3.73E-2	3.73E-2	3.73E-2	3.73E-2	3.73E-2	3.72E-2
Cs-134	1.27E+5	6.15E+3	-	-	-	-
Cs-135	-	-	-	5.66E-1	5.64E-1	5.49E-1
Cs-137	1.19E+5	9.66E+4	1.21E+4	-	-	-
Ba-137m	1.12E+5	9.14E+4	1.14E+4	-	-	-
Ce-144	3.06E+5	1.01E+2	-	-	-	-
Pr-144	3.06E+5	1.01E+2	-	-	-	-
Pr-144m	3.67E+3	-	-	-	-	-
Pm-147	8.80E+4	8.20E+3	-	-	-	-
Sm-151	3.80E+2	3.55E+2	1.78E+2	1.73E-1	-	-
Eu-154[b]	1.30E+4	6.31E+3	4.42E+0	-	-	-
Eu-155[b]	7.46E+3	2.12E+3	-	-	-	-
OTHER	4.95E+3	2.15E+1	3.52E+1	2.12E-1	8.14E-2	2.10E-2
SUBTOTAL						
A.P.[d]	1.94E+4	1.84E+3	1.04E+2	4.15E+0	2.71E+0	1.35E+0
F.P.[e]	1.81E+6	3.52E+5	3.93E+4	2.30E+1	2.22E+1	1.71E+1
TOTAL	1.83E+6	3.53E+5	3.94E+4	2.72E+1	2.50E+1	1.85E+1

[a] Nuclides contributing >0.1% are listed.

[b] Both activation and fission products contribute to this nuclide.

[c] Only activation products contribute to this nuclide.

[d] A.P. = Activation products.

[e] F.P. = Fission products.

Table 2.4.4 Variation of radioactivity (Ci/MTIHM) for significant activation- and fission- product nuclides as a function of time since discharge from a 27,500 MWd/MTIHM BWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
H-3[b]	6.63E+2	4.00E+2	2.56E+0	-	-	-
C-14[c]	1.53E+0	1.53E+0	1.52E+0	1.36E+0	4.57E-1	-
Mn-54[c]	1.45E+2	-	-	-	-	-
Fe-55[c]	2.23E+3	2.02E+2	-	-	-	-
Co-58[c]	3.71E+1	-	-	-	-	-
Co-60[c]	2.18E+3	6.66E+2	-	-	-	-
Ni-59[c]	1.07E+0	1.07E+0	1.07E+0	1.06E+0	9.82E-1	4.50E-1
Ni-63[c]	1.57E+2	1.47E+2	7.47E+1	-	-	-
Zn-65[c]	3.51E+1	-	-	-	-	-
Se-79	-	-	-	3.34E-1	3.04E-1	1.16E-1
Kr-85	7.02E+3	3.92E+3	1.16E+1	-	-	-
Sr-89	3.90E+3	-	-	-	-	-
Sr-90	5.82E+4	4.70E+4	5.52E+3	-	-	-
Y-90	5.82E+4	4.70E+4	5.52E+3	-	-	-
Y-91	1.01E+4	-	-	-	-	-
Zr-93[b]	1.80E+0	1.80E+0	1.80E+0	1.80E+0	1.80E+0	1.72E+0
Zr-95[b]	2.24E+4	-	-	-	-	-
Nb-93m[b]	-	-	-	1.71E+0	1.71E+0	1.64E+0
Nb-95[b]	5.04E+4	-	-	-	-	-
Tc-99	1.11E+1	1.11E+1	1.11E+1	1.11E+1	1.08E+1	8.04E+0
Ru-103	1.81E+3	-	-	-	-	-
Ru-106	1.97E+5	4.04E+2	-	-	-	-
Rh-106	1.97E+5	4.04E+2	-	-	-	-
Pd-107	-	-	-	9.46E-2	9.45E-2	9.36E-2
Ag-110m	1.05E+3	-	-	-	-	-
Sn-119m[b]	3.77E+3	-	-	-	-	-
Sn-126	6.25E-1	6.24E-1	6.24E-1	6.20E-1	5.83E-1	3.12E-1
Sb-125[b]	1.05E+4	1.10E+3	-	-	-	-
Sb-126	-	-	-	8.68E-2	8.16E-2	4.37E-2
Sb-126m	-	-	-	6.20E-1	5.83E-1	3.12E-1
Te-125m[b]	2.56E+3	2.69E+2	-	-	-	-
I-129	2.64E-2	2.64E-2	2.64E-2	2.64E-2	2.64E-2	2.63E-2
Cs-134	7.65E+4	3.71E+3	-	-	-	-
Cs-135	-	-	-	3.59E-1	3.58E-1	3.49E-1
Cs-137	8.37E+4	6.80E+4	8.49E+3	-	-	-
Ba-137m	7.91E+4	6.43E+4	8.03E+3	-	-	-
Ce-144	3.10E+5	1.02E+2	-	-	-	-
Pr-144	3.10E+5	1.02E+2	-	-	-	-
Pr-144m	3.72E+3	1.23E+0	-	-	-	-
Pm-147	8.68E+4	8.05E+3	-	-	-	-
Sm-151	3.20E+2	2.98E+2	1.49E+2	1.46E-1	-	-
Eu-154[b]	7.63E+3	3.70E+3	2.61E+0	-	-	-
Eu-155[b]	4.49E+3	1.28E+3	-	-	-	-
OTHER	5.82E+3	9.30E+1	-	1.53E-1	5.40E-2	4.16E-2
SUBTOTAL						
A.P.[d]	1.81E+4	1.58E+3	7.92E+1	3.14E+0	2.06E+0	1.02E+0
F.P.[e]	1.58E+6	2.50E+5	2.78E+4	1.63E+1	1.57E+1	1.21E+1
TOTAL	1.60E+6	2.51E+5	2.78E+4	1.94E+1	1.78E+1	1.31E+1

[a] Nuclides contributing >than 0.1% are listed.

[b] Both activation and fission products contribute to this nuclide.

[c] Only activation products contribute to this nuclide.

[d] A.P. = Activation products.

[e] F.P. = Fission products.

Table 2.4.5 Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 60,000 MWd/MTIHM PWR

(Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	3.32E-5	5.81E-3	2.68E-1	2.12E+0
U-234	-	-	-	4.08E+0	3.99E+0	3.16E+0
Np-237	-	-	-	1.74E+0	2.03E+0	1.97E+0
Np-239	7.22E+1	7.21E+1	7.15E+1	6.57E+1	2.82E+1	-
Pu-238	8.56E+3	8.10E+3	3.98E+3	3.60E+0	-	-
Pu-239	3.67E+2	3.67E+2	3.66E+2	3.59E+2	2.87E+2	2.24E+1
Pu-240	6.78E+2	6.90E+2	7.13E+2	6.49E+2	2.50E+2	-
Pu-241	1.88E+5	1.22E+5	1.61E+3	1.74E+0	-	-
Pu-242	-	-	-	4.53E+0	4.47E+0	3.80E+0
Am-241	5.77E+2	2.76E+3	5.98E+3	1.43E+3	-	-
Am-243	7.22E+1	7.21E+1	7.15E+1	6.57E+1	2.82E+1	-
Cm-242	2.75E+4	1.40E+1	9.25E+0	-	-	-
Cm-243	9.13E+1	7.34E+1	8.22E+0	-	-	-
Cm-244	1.55E+4	1.10E+4	3.51E+2	-	-	-
OTHER	6.47E+1	4.16E+1	3.03E+1	5.84E+0	-	3.07E+1[b]
TOTAL	2.42E+5	1.45E+5	1.32E+4	2.59E+3	6.13E+2	6.20E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 2.12 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.64 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 2.4.6 Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 33,000 MWd/MTIHM PWR

(Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	2.66E-5	3.12E-3	1.34E-1	1.07E+0
U-234	-	-	-	2.03E+0	1.99E+0	1.61E+0
Np-237	-	-	-	9.99E-1	1.18E+0	1.14E+0
Np-239	1.71E+1	1.71E+1	1.69E+1	1.56E+1	6.68E+0	-
Pu-238	2.45E+3	2.33E+3	1.15E+3	1.08E+0	-	-
Pu-239	3.13E+2	3.13E+2	3.12E+2	3.05E+2	2.37E+2	1.80E+1
Pu-240	5.26E+2	5.27E+2	5.26E+2	4.78E+2	1.84E+2	-
Pu-241	1.20E+5	7.76E+4	1.02E+3	-	-	-
Pu-242	-	-	-	1.72E+0	1.69E+0	1.44E+0
Am-241	3.08E+2	1.69E+3	3.75E+3	8.93E+2	-	-
Am-243	1.71E+1	1.71E+1	1.69E+1	1.56E+1	6.68E+0	-
Cm-242	1.04E+4	5.72E+0	3.78E+0	-	-	-
Cm-243	2.06E+1	1.66E+1	1.86E+0	-	-	-
Cm-244	1.86E+3	1.32E+3	4.21E+1	-	-	-
OTHER	2.74E+2	2.60E+1	1.56E+1	2.68E+0	4.30E+0	1.68E+1[b]
TOTAL	1.36E+5	8.39E+4	6.85E+3	1.72E+3	4.44E+2	3.90E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 1.07 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.37 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 2.4.7 Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 40,000 MWd/MTIHM BWR

(Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	2.94E-5	3.85E-3	1.70E-1	1.35E+0
U-234	-	-	-	2.58E+0	2.52E+0	2.02E+0
Np-237	-	-	-	1.21E+0	1.42E+0	1.38E+0
Np-239	2.83E+1	2.83E+1	2.80E+1	2.58E+1	1.11E+1	-
Pu-238	4.06E+3	3.85E+3	1.90E+3	1.82E+0	-	-
Pu-239	3.06E+2	3.06E+2	3.06E+2	2.98E+2	2.34E+2	1.79E+1
Pu-240	5.63E+2	5.65E+2	5.67E+2	5.16E+2	1.98E+2	-
Pu-241	1.37E+5	8.87E+4	1.17E+3	-	-	-
Pu-242	-	-	-	2.37E+0	2.33E+0	1.98E+0
Am-241	4.36E+2	2.02E+3	4.36E+3	1.04E+3	-	-
Am-243	2.83E+1	2.83E+1	2.80E+1	2.58E+1	1.11E+1	-
Cm-242	1.60E+4	1.09E+1	7.22E+0	-	-	-
Cm-243	3.64E+1	2.92E+1	3.28E+0	-	-	-
Cm-244	3.75E+3	2.66E+3	8.48E+1	-	-	-
OTHER	1.08E+2	6.23E+1	1.27E+1	3.56E+0	5.33E+0	2.06E+1[b]
TOTAL	1.62E+5	9.83E+4	8.47E+3	1.92E+3	4.66E+2	4.38E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 1.35 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.45 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 2.4.8 Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 27,500 MWd/MTIHM BWR

(Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Ra-226	-	-	2.32E-5	2.60E-3	1.11E-1	8.86E-1
U-234	-	-	-	1.68E+0	1.64E+0	1.34E+0
Np-237	-	-	-	8.64E-1	1.02E+0	9.95E-1
Np-239	1.29E+1	1.29E+1	1.28E+1	1.18E+1	5.06E+0	-
Pu-238	1.86E+3	1.78E+3	8.77E+2	8.87E-1	-	-
Pu-239	3.00E+2	3.00E+2	3.00E+2	2.92E+2	2.27E+2	1.72E+1
Pu-240	4.78E+2	4.78E+2	4.76E+2	4.33E+2	1.67E+2	-
Pu-241	1.07E+5	6.95E+4	9.13E+2	-	-	-
Pu-242	-	-	-	1.42E+0	1.39E+0	1.19E+0
Am-241	3.15E+2	1.56E+3	3.39E+3	8.07E+2	-	-
Am-243	1.29E+1	1.29E+1	1.28E+1	1.18E+1	5.06E+0	-
Cm-242	9.42E+3	6.87E+0	4.54E+0	-	-	-
Cm-243	1.67E+1	1.34E+1	1.50E+0	-	-	-
Cm-244	1.25E+3	8.86E+2	2.83E+1	-	-	-
OTHER	3.05E+1	2.29E+1	1.61E+1	2.00E+0	3.90E+0	1.44E+1[b]
TOTAL	1.21E+5	7.45E+4	6.03E+3	1.56E+3	4.12E+2	3.51E+1

[a]Nuclides contributing >0.1% are listed.

[b]The following isotopes contribute 0.89 Ci each: Pb-210, Pb-214, Bi-210, Bi-214, Po-210, Po-214, Po-218, and Rn-222. Others contributing 0.33 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 2.4.9 Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 60,000 MWd/MTIHM PWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	1.47E+2	4.50E+1	-	-	-	-
Kr-85	2.00E+1	1.12E+1	-	-	-	-
Sr-89	1.57E+1	-	-	-	-	-
Sr-90	1.32E+2	1.06E+2	-	-	-	-
Y-90	6.29E+2	5.08E+2	5.96E+1	-	-	-
Y-91	4.38E+1	-	-	-	-	-
Zr-95[c]	1.48E+2	-	-	-	-	-
Nb-95[c]	3.16E+2	-	-	-	-	-
Ru-106	2.28E+1	-	-	-	-	-
Rh-106	3.68E+3	7.56E+0	-	-	-	-
Ag-110m	6.21E+1	-	-	-	-	-
Sb-125[c]	5.63E+1	5.34E+0	-	-	-	-
Cs-134	2.66E+3	1.29E+2	-	-	-	-
Cs-137	1.97E+2	1.60E+2	2.00E+1	-	-	-
Ba-137m	6.60E+2	5.36E+2	6.71E+1	-	-	-
Ce-144	2.84E+2	-	-	-	-	-
Pr-144	3.15E+3	-	-	-	-	-
Pm-147	3.37E+1	3.12E+0	-	-	-	-
Eu-154[c]	2.09E+2	1.01E+2	-	-	-	-
U-233	-	-	-	-	-	2.05E-2
U-234	-	-	-	1.18E-1	1.15E-1	9.10E-2
U-236	-	-	-	-	-	1.55E-2
Np-237	-	-	-	-	-	6.02E-2
Pu-238	2.84E+2	2.68E+2	1.32E+2	-	-	-
Pu-239	1.13E+1	1.13E+1	1.13E+1	1.10E+1	8.84E+0	6.90E-1
Pu-240	2.11E+1	2.15E+1	2.22E+1	2.02E+1	7.78E+0	-
Pu-241	5.84E+0	3.79E+0	-	-	-	-
Pu-242	-	-	-	1.34E-1	1.32E-1	1.12E-1
Am-241	1.92E+1	9.16E+1	1.98E+2	4.74E+1	-	-
Am-243	2.32E+0	2.32E+0	2.30E+0	2.11E+0	9.07E-1	-
Cm-242	1.01E+3	-	-	-	-	-
Cm-243	3.35E+0	2.69E+0	-	-	-	-
Cm-244	5.44E+2	3.85E+2	1.23E+1	-	-	-
OTHER	7.25E+1	7.00E+0	8.50E+0	5.18E-1	3.42E-1	6.44E-1
SUBTOTAL						
A.P.[d]	1.80E+2	4.61E+1	2.23E-1	2.35E-2	1.69E-2	9.54E-4
F.P.[e]	1.23E+4	1.57E+3	1.59E+2	3.62E-2	3.43E-2	2.10E-2
A.+D.[f]	1.90E+3	7.88E+2	3.80E+2	8.14E+1	1.81E+1	1.61E+0
TOTAL	1.44E+4	2.41E+3	5.39E+2	8.15E+1	1.81E+1	1.63E+0

[a] Nuclides contributing >0.1% of total are listed.
[b] Only activation products contribute to this nuclide.
[c] Both activation and fission products contribute to this nuclide.
[d] A.P. = Activation products.
[e] F.P. = Fission products.
[f] A.+D. = Actinides plus daughters.

Table 2.4.10 Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 33,000 MWd/MTIHM PWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	1.07E+2	3.28E+1	-	-	-	-
Kr-85	1.30E+1	7.27E+1	-	-	-	-
Sr-89	1.98E+1	-	-	-	-	-
Sr-90	8.22E+1	6.63E+1	7.79E+0	-	-	-
Y-90	3.93E+2	3.17E+2	3.72E+1	-	-	-
Y-91	5.34E+1	-	-	-	-	-
Zr-95[c]	1.59E+2	-	-	-	-	-
Nb-95[c]	3.39E+2	-	-	-	-	-
Ru-106	1.60E+1	-	-	-	-	-
Rh-106	2.57E+3	5.28E+0	-	-	-	-
Ag-110m	2.54E+1	-	-	-	-	-
Sb-125[c]	3.82E+1	4.02E+0	-	-	-	-
Cs-134	1.10E+3	5.31E+1	-	-	-	-
Cs-137	1.12E+2	9.08E+1	1.14E+1	-	-	-
Ba-137m	3.76E+2	3.05E+2	3.81E+1	-	-	-
Ce-144	2.99E+2	-	-	-	-	-
Pr-144	3.31E+3	-	-	-	-	-
Pm-147	3.67E+1	3.40E+0	-	-	-	-
Eu-154[c]	8.67E+1	4.20E+1	-	-	-	-
U-233	-	-	-	-	-	1.19E-2
U-234	-	-	-	5.84E-2	5.72E-2	4.64E-2
U-236	-	-	-	-	-	1.09E-2
Np-237	-	-	-	-	-	3.49E-2
Pu-238	8.13E+1	7.74E+1	3.71E+1	-	-	-
Pu-239	9.65E+0	9.64E+0	9.62E+0	9.39E+0	7.32E+0	5.54E-1
Pu-240	1.64E+1	1.64E+1	1.64E+1	1.49E+1	5.73E+0	-
Pu-241	3.71E+0	2.41E+0	-	-	-	-
Pu-242	-	-	-	5.08E-2	5.00E-2	4.25E-2
Am-241	1.02E+1	5.63E+1	1.24E+2	2.97E+1	-	-
Am-243	5.49E-1	5.49E-1	5.44E-1	5.00E-1	2.15E-1	-
Cm-242	3.83E+2	-	-	-	-	-
Cm-243	7.56E-1	6.08E-1	-	-	-	-
Cm-244	6.51E+1	4.62E+1	1.47E+0	-	-	-
OTHER	4.96E+1	4.70E+0	1.60E+0	1.65E-1	1.40E-1	3.57E-1
SUBTOTAL						
A.P.[d]	1.30E+2	3.35E+1	1.46E-1	1.34E-2	9.66E-3	5.64E-4
F.P.[e]	9.04E+3	8.96E+2	9.46E+1	2.01E-2	1.91E-2	1.18E-2
A.+D.[f]	5.71E+2	2.10E+2	1.91E+2	5.47E+1	1.35E+1	1.03E+0
TOTAL	9.74E+3	1.14E+3	2.86E+2	5.47E+1	1.35E+1	1.05E+0

[a]Nuclides contributing >0.1% of total are listed.
[b]Only activation products contribute to this nuclide.
[c]Both activation and fission products contribute to this nuclide.
[d]A.P. = Activation products.
[e]F.P. = Fission products.
[f]A.+D. = Actinides plus daughters.

Table 2.4.11 Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 40,000 MWd/MTIHM BWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	4.04E+1	1.24E+1	-	-	-	-
Kr-85	1.43E+1	7.97E+0	-	-	-	-
Sr-89	1.24E+1	-	-	-	-	-
Sr-90	9.51E+1	7.68E+1	9.01E+0	-	-	-
Y-90	4.54E+2	3.67E+2	4.30E+1	-	-	-
Y-91	3.38E+1	-	-	-	-	-
Zr-95[c]	1.10E+2	-	-	-	-	-
Nb-95[c]	2.35E+2	-	-	-	-	-
Ru-106	1.35E+1	-	-	-	-	-
Rh-106	2.18E+3	4.48E+0	-	-	-	-
Ag-110m	2.72E+1	-	-	-	-	-
Sb-125[c]	3.90E+1	4.10E+0	-	-	-	-
Cs-134	1.29E+3	6.26E+1	-	-	-	-
Cs-137	1.32E+2	1.07E+2	1.34E+1	-	-	-
Ba-137m	4.42E+2	3.59E+2	4.49E+1	-	-	-
Ce-144	2.03E+2	-	-	-	-	-
Pr-144	2.25E+3	-	-	-	-	-
Pm-147	3.17E+1	2.94E+0	-	-	-	-
Eu-154[c]	1.17E+2	5.64E+1	-	-	-	-
U-233	-	-	-	-	-	1.44E-2
U-234	-	-	-	7.43E-2	7.26E-2	5.83E-2
U-236	-	-	-	-	-	1.23E-2
Np-237	-	-	-	-	-	4.22E-2
Pu-238	1.34E+2	1.28E+2	6.29E+1	-	-	-
Pu-239	9.44E+0	9.44E+0	9.41E+0	9.20E+0	7.22E+0	5.51E-1
Pu-240	1.75E+1	1.76E+1	1.76E+1	1.60E+1	6.18E+0	-
Pu-241	4.24E+0	2.75E+0	-	-	-	-
Pu-242	-	-	-	6.99E-2	6.88E-2	5.85E-2
Am-241	1.45E+1	6.71E+1	1.45E+2	3.45E+1	-	-
Am-243	9.10E-1	9.09E-1	9.02E-1	8.28E-1	3.56E-1	-
Cm-242	5.91E+2	-	-	-	-	-
Cm-243	1.34E+0	1.07E+0	-	-	-	-
Cm-244	1.31E+2	9.30E+1	2.97E+0	-	-	-
OTHER	1.24E+1	7.85E+0	8.00E-1	2.96E-1	1.75E-1	4.25E-1
SUBTOTAL						
A.P.[d]	8.28E+1	1.40E+1	4.18E-2	1.20E-3	6.64E-4	1.64E-4
F.P.[e]	7.66E+3	1.05E+3	1.10E+2	2.34E-2	2.22E-2	1.38E-2
A.+D.[f]	9.05E+2	3.20E+2	2.39E+2	6.09E+1	1.40E+1	1.15E+0
TOTAL	8.65E+3	1.38E+3	3.50E+2	6.09E+1	1.41E+1	1.16E+0

[a]Nuclides contributing >0.1% of total are listed.
[b]Only activation products contribute to this nuclide.
[c]Both activation and fission products contribute to this nuclide.
[d]A.P. = Activation products.
[e]F.P. = Fission products.
[f]A.+D. = Actinides plus daughters.

Table 2.4.12 Variation in thermal power (W/MTIHM) for significant nuclides as a function of time since discharge from a 27,500 MWd/MTIHM BWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Co-60[b]	3.36E+1	1.03E+1	-	-	-	-
Kr-85	1.05E+1	5.88E+0	-	-	-	-
Sr-89	1.35E+1	-	-	-	-	-
Sr-90	6.76E+1	5.45E+1	6.40E+0	-	-	-
Y-90	3.23E+2	2.60E+2	3.06E+1	-	-	-
Y-91	3.63E+1	-	-	-	-	-
Zr-95[c]	1.14E+2	-	-	-	-	-
Nb-95[c]	2.42E+2	-	-	-	-	-
Ru-106	1.17E+1	-	-	-	-	-
Rh-106	1.89E+3	3.87E+0	-	-	-	-
Ag-110m	1.76E+1	-	-	-	-	-
Sb-125[c]	3.28E+1	3.45E+0	-	-	-	-
Cs-134	7.78E+2	3.78E+2	-	-	-	-
Cs-137	9.25E+1	7.52E+1	9.40E+0	-	-	-
Ba-137m	3.11E+2	2.52E+2	3.16E+1	-	-	-
Ce-144	2.06E+2	-	-	-	-	-
Pr-144	2.28E+3	-	-	-	-	-
Pm-147	3.12E+1	2.89E+0	-	-	-	-
Eu-154[c]	6.83E+1	3.31E+1	-	-	-	-
U-233	-	-	-	-	-	1.04E-2
U-234	-	-	-	4.83E-2	4.73E-2	3.87E-2
U-236	-	-	-	-	-	9.42E-3
Np-237	-	-	-	-	-	3.04E-2
Pu-238	6.18E+1	5.90E+1	2.91E+1	-	-	-
Pu-239	9.26E+0	9.26E+0	9.23E+0	9.01E+0	7.00E+0	5.29E-1
Pu-240	1.49E+1	1.49E+1	1.48E+1	1.35E+1	5.19E+0	-
Pu-241	3.32E+0	2.15E+0	-	-	-	-
Pu-242	-	-	-	4.18E-2	4.12E-2	3.50E-2
Am-241	1.05E+1	5.17E+1	1.12E+2	2.68E+1	-	-
Am-243	4.16E-1	4.15E-1	4.12E-1	3.78E-1	1.62E-1	-
Cm-242	3.47E+2	-	-	-	-	-
Cm-243	6.12E-1	4.92E-1	-	-	-	-
Cm-244	4.37E+1	3.10E+1	9.89E-1	-	-	-
OTHER	2.47E+1	6.32E+0	6.00E-1	1.25E-1	1.14E-1	2.92E-1
SUBTOTAL						
A.P.[d]	7.42E+2	1.19E+1	3.18E-2	8.92E-4	5.02E-4	1.24E-4
F.P.[e]	6.50E+3	7.30E+2	7.80E+1	1.65E-2	1.57E-2	9.78E-3
A.+D.[f]	4.92E+2	1.69E+2	1.68E+2	4.99E+1	1.25E+1	9.35E-1
TOTAL	7.07E+3	9.11E+2	2.46E+2	4.99E+1	1.26E+1	9.45E-1

[a]Nuclides contributing >0.1% of total are listed.
[b]Only activation products contribute to this nuclide.
[c]Both activation and fission products contribute to this nuclide.
[d]A P. = Activation products.
[e]F P. = Fission products.
[f]A.+D. = Actinides plus daughters.

Table 2.4.13 Variation in neutron production (neutrons/s·MTIHM) by spontaneous fission as a function of time since discharge from a 60,000 MWd/MTIHM PWR (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	1.16E+4	1.16E+4	1.16E+4	1.16E+4
Pu-238	-	-	6.18E+5	5.59E+2	-	-
Pu-240	2.71E+6	2.76E+6	2.85E+6	2.59E+6	9.98E+5	7.17E+1
Pu-242	2.00E+6	2.00E+6	2.00E+6	2.00E+6	1.97E+6	1.68E+6
Cm-242	1.79E+8	9.11E+4	6.02E+4	-	-	-
Cm-244	2.14E+9	1.51E+9	4.83E+7	-	-	-
Cm-246	2.11E+7	2.11E+7	2.08E+7	1.82E+7	4.88E+7	9.15E+0
Cm-248	-	-	1.62E+5	1.62E+5	1.59E+5	1.32E+5
Cf-252	9.45E+6	8.88E+5	-	-	-	-
TOTAL	2.35E+9	1.54E+9	7.48E+7	2.30E+7	8.02E+6	1.82E+6

[a]Nuclides contributing >0.1% are listed.

Table 2.4.14 Variation in neutron production (neutrons/s·MTIHM) by spontaneous fission as a function of time since discharge from a 33,000 MWd/MTIHM PWR (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	-	1.20E+4	1.20E+4	1.20E+4
Pu-238	3.80E+5	3.62E+5	1.78E+5	1.68E+2	-	-
Pu-240	2.10E+6	2.10E+6	2.10E+6	1.91E+6	7.35E+5	5.27E+1
Pu-242	7.60E+5	7.60E+5	7.60E+5	7.59E+5	7.47E+5	6.36E+5
Cm-242	6.78E+7	3.72E+4	2.46E+4	-	-	-
Cm-244	2.56E+8	1.81E+8	5.79E+6	-	-	-
Cm-246	9.06E+5	9.04E+5	8.92E+5	7.82E+5	2.09E+5	-
Cm-248	-	-	-	1.93E+3	1.89E+3	1.57E+3
TOTAL	3.28E+8	1.86E+8	9.76E+6	3.46E+6	1.70E+6	6.49E+5

[a]Nuclides contributing >0.1% are listed.

Table 2.4.15 Variation in neutron production (neutrons/s·MTIHM) by spontaneous fission as a function of time since discharge from a 40,000 MWd/MTIHM BWR (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	1.19E+4	1.19E+4	1.19E+4	1.19E+4
Pu-238	6.29E+5	5.98E+5	2.94E+5	2.82E+2	-	-
Pu-240	2.25E+6	2.26E+6	2.26E+6	2.06E+6	7.93E+5	5.70E+1
Pu-242	1.04E+6	1.04E+6	1.04E+6	1.04E+6	1.03E+6	8.75E+5
Cm-242	1.04E+8	7.11E+4	4.70E+4	7.76E+2	-	-
Cm-244	5.15E+8	3.65E+8	1.16E+7	-	-	-
Cm-246	2.58E+6	2.58E+6	2.55E+6	2.32E+6	5.97E+5	-
Cm-248	-	-	8.58E+3	8.56E+3	8.41E+3	7.00E+3
TOTAL	6.27E+8	3.72E+8	1.79E+7	5.36E+6	2.44E+6	8.94E+5

[a]Nuclides contributing >0.1% are listed.

Table 2.4.16 Variation in neutron production (neutrons/s·MTIHM) by spontaneous fission as a function of time since discharge from a 27,500 MWd/MTIHM BWR (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
U-238	-	-	1.21E+4	1.21E+4	1.21E+4	1.21E+4
Pu-238	2.89E+5	2.76E+5	1.36E+5	1.38E+2	-	-
Pu-240	1.91E+6	1.91E+6	1.90E+6	1.73E+6	6.66E+5	4.77E+1
Pu-242	6.26E+5	6.26E+5	6.26E+5	6.25E+5	6.15E+5	5.24E+5
Cm-242	6.14E+7	4.47E+4	2.96E+4	4.89E+2	-	-
Cm-244	1.72E+8	1.22E+8	3.88E+6	-	-	-
Cm-246	5.01E+5	5.01E+5	4.94E+5	4.33E+5	1.16E+5	-
Cm-248	-	-	-	8.70E+2	8.54E+2	7.10E+2
TOTAL	2.36E+8	1.25E+8	7.08E+6	2.80E+6	1.41E+6	5.37E+5

[a]Nuclides contributing >0.1% are listed.

Table 2.4.17 Variation in neutron production (neutrons/s·MTIHM) by the (α,n) reaction as a function of time since discharge from a 60,000 MWd/MTIHM PWR
(Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	2.26E+2	1.79E+3
Po-213	-	-	-	-	-	1.72E+3
Po-214	-	-	-	-	6.33E+2	5.00E+3
Po-218	-	-	-	-	3.42E+2	2.70E+3
At-217	-	-	-	-	-	1.28E+3
Rn-222	-	-	-	-	2.55E+2	2.01E+3
Fr-221	-	-	-	-	-	9.88E+2
Ra-226	-	-	-	-	1.50E+2	1.19E+3
Ac-225	-	-	-	-	-	7.32E+2
Th-229	-	-	-	-	-	4.56E+2
Th-230	-	-	-	-	1.78E+2	1.08E+3
U-233	-	-	-	-	-	4.07E+2
U-234	-	-	-	2.27E+3	2.22E+3	1.76E+3
U-236	-	-	-	-	2.14E+2	2.43E+2
U-238	-	-	-	-	-	9.67E+1
Np-237	-	-	-	1.23E+3	1.43E+3	1.39E+3
Pu-238	8.14E+6	7.71E+6	3.79E+6	3.43E+3	-	-
Pu-239	2.67E+5	2.67E+5	2.67E+5	2.61E+5	2.09E+5	1.63E+4
Pu-240	5.14E+5	5.23E+5	5.40E+5	4.92E+5	1.89E+5	-
Pu-242	-	-	-	2.79E+3	2.75E+3	2.34E+3
Am-241	5.53E+5	2.65E+6	5.73E+6	1.37E+6	8.03E+2	-
Am-243	6.16E+4	6.15E+4	6.10E+4	5.60E+4	2.41E+4	-
Cm-242	3.69E+7	1.88E+4	1.24E+4	-	-	-
Cm-243	1.22E+5	9.77E+4	1.10E+4	-	-	-
Cm-244	1.77E+7	1.26E+7	4.01E+5	1.66E+3	7.98E+2	-
TOTAL	6.43E+7	2.39E+7	1.08E+7	2.19E+6	4.33E+5	4.17E+4

[a]Nuclides contributing >0.1% are listed.

Table 2.4.18 Variation in neutron production (neutrons/s·MTIHM) by the (α,n) reaction as a function of time since discharge from a 33,000 MWd/MTIHM PWR (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	4.40E+1	9.00E+2
Po-213	-	-	-	-	-	9.99E+2
Po-214	-	-	-	-	3.16E+2	2.52E+3
Po-218	-	-	-	-	1.71E+2	1.36E+3
At-217	-	-	-	-	-	7.40E+2
Rn-222	-	-	-	-	1.27E+2	1.01E+3
Fr-221	-	-	-	-	-	5.73E+2
Ra-226	-	-	-	-	7.52E+1	5.98E+2
Ac-225	-	-	-	-	-	4.24E+2
Th-229	-	-	-	-	-	2.64E+2
Th-230	-	-	-	-	8.87E+1	5.45E+2
U-233	-	-	-	-	-	2.36E+2
U-234	-	-	-	1.13E+3	1.10E+3	8.94E+2
U-236	-	-	-	-	1.50E+2	1.72E+2
U-238	-	-	-	-	-	9.99E+1
Np-237	-	-	-	7.04E+2	8.30E+2	8.06E+2
Pu-238	2.33E+6	2.12E+6	1.09E+6	1.03E+3	-	-
Pu-239	2.28E+5	2.28E+5	2.27E+5	2.22E+5	1.73E+5	1.31E+4
Pu-240	3.99E+5	4.00E+5	3.98E+5	3.62E+5	1.40E+5	-
Pu-242	-	-	-	1.06E+3	1.04E+3	8.86E+2
Am-241	2.95E+5	2.23E+6	3.59E+6	8.57E+5	-	-
Am-243	1.46E+4	1.46E+4	1.44E+4	1.33E+4	5.70E+3	-
Cm-242	1.40E+7	7.45E+3	5.08E+3	-	-	-
Cm-243	2.74E+4	2.20E+4	2.47E+3	-	-	-
Cm-244	2.12E+6	1.51E+6	4.81E+4	-	-	-
TOTAL	1.94E+7	6.03E+6	5.38E+6	1.46E+6	3.22E+5	2.63E+4

[a]Nuclides contributing >0.1% are listed.

Table 2.4.19 Variation in neutron production (neutrons/s·MTIHM) by the (α,n) reaction as a function of time since discharge from a 40,000 MWd/MTIHM BWR (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	1.43E+2	1.14E+3
Po-213	-	-	-	-	-	1.21E+3
Po-214	-	-	-	-	4.01E+2	3.18E+3
Po-218	-	-	-	-	2.17E+2	1.72E+3
At-217	-	-	-	-	-	8.93E+2
Rn-222	-	-	-	-	1.61E+2	1.28E+3
Fr-221	-	-	-	-	-	6.92E+2
Ra-226	-	-	-	-	9.53E+1	7.56E+2
Ac-225	-	-	-	-	-	5.12E+2
Th-229	-	-	-	-	-	3.19E+2
Th-230	-	-	-	-	1.12E+2	6.88E+2
U-233	-	-	-	-	-	2.85E+2
U-234	-	-	-	1.43E+3	1.40E+3	1.12E+3
U-236	-	-	-	-	1.71E+2	1.94E+2
U-238	-	-	-	-	-	9.92E+1
Np-237	-	-	-	8.56E+2	1.00E+3	9.73E+2
Pu-238	3.86E+6	3.67E+6	1.80E+6	1.73E+3	-	-
Pu-239	2.23E+5	2.23E+5	2.22E+5	2.17E+5	1.70E+5	1.30E+4
Pu-240	4.26E+5	4.28E+5	4.30E+5	3.91E+5	1.50E+5	-
Pu-242	-	-	-	1.46E+3	1.43E+3	1.22E+3
Am-241	4.19E+5	1.94E+6	4.18E+6	9.96E+5	1.42E+2	-
Am-243	2.42E+4	2.41E+4	2.39E+4	2.20E+4	9.44E+3	-
Cm-242	2.15E+7	1.46E+4	9.69E+3	-	-	-
Cm-243	4.84E+4	3.89E+4	4.36E+3	-	-	-
Cm-244	4.28E+6	3.03E+6	9.68E+4	-	-	-
TOTAL	3.08E+7	9.37E+6	6.77E+6	1.63E+6	3.36E+5	2.95E+4

[a]Nuclides contributing >0.1% are listed.

Table 2.4.20 Variation in neutron production (neutrons/s·MTIHM) by the (α,n) reaction as a function of time since discharge from a 27,500 MWd/MTIHM BWR (Source: Roddy 1986)

	Time since discharge (years)					
Isotope[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
Po-210	-	-	-	-	9.35E+1	7.48E+2
Po-213	-	-	-	-	-	8.70E+2
Po-214	-	-	-	-	2.62E+2	2.09E+3
Po-218	-	-	-	-	1.41E+2	1.13E+3
At-217	-	-	-	-	-	6.44E+2
Rn-222	-	-	-	-	1.05E+2	8.42E+2
Fr-221	-	-	-	-	-	4.99E+2
Ra-226	-	-	-	-	6.21E+1	4.97E+2
Ac-225	-	-	-	-	-	3.70E+2
Th-229	-	-	-	-	-	2.30E+2
Th-230	-	-	-	-	7.34E+1	4.53E+2
U-233	-	-	-	-	-	2.06E+2
U-234	-	-	-	9.31E+2	9.12E+2	7.47E+2
U-236	-	-	-	-	1.29E+2	1.48E+2
U-238	-	-	-	-	1.01E+2	1.01E+2
Np-237	-	-	-	6.10E+2	7.23E+2	7.02E+2
Pu-238	1.77E+6	1.69E+6	8.35E+5	8.44E+2	-	-
Pu-239	2.19E+5	2.19E+5	2.18E+5	2.13E+5	1.66E+5	1.25E+4
Pu-240	3.62E+5	3.62E+5	3.61E+5	3.28E+5	1.26E+5	-
Pu-242	-	-	-	8.71E+2	8.57E+2	7.30E+2
Am-241	3.02E+5	1.49E+6	3.25E+6	7.74E+5	3.98E+1	-
Am-243	1.10E+4	1.10E+4	1.09E+4	1.00E+4	4.31E+3	-
Cm-242	1.26E+7	9.22E+3	6.10E+3	-	-	-
Cm-243	2.22E+4	1.78E+4	2.00E+3	-	-	-
Cm-244	1.43E+6	1.01E+6	3.22E+4	-	-	-
TOTAL	1.68E+7	4.82E+6	4.72E+6	1.33E+6	3.00E+5	2.37E+4

[a]Nuclides contributing >0.1% are listed.

Table 2.4.21 Variation in photon production (photons/s·MTIHM) as a function of time since discharge from a 60,000 MWd/MTIHM PWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	2.97E+16	3.21E+15	4.22E+14	1.92E+13	3.50E+12	4.09E+11
2.50E-2	6.75E+15	6.71E+14	7.50E+13	1.41E+12	5.83E+10	4.35E+10
3.75E-2	6.72E+15	8.75E+14	8.68E+13	2.97E+11	8.72E+10	2.47E+10
5.75E-2	6.15E+15	6.33E+14	1.46E+14	1.97E+13	5.84E+10	2.71E+10
8.50E-1	4.31E+15	3.84E+14	3.97E+13	1.91E+12	8.58E+11	9.75E+10
1.25E-1	4.83E+15	4.09E+14	2.55E+13	1.22E+12	5.26E+11	1.61E+10
2.25E-1	3.78E+15	3.12E+14	3.20E+13	8.12E+11	3.56E+11	4.55E+10
3.75E-1	2.10E+15	1.45E+14	1.33E+13	1.64E+11	1.24E+11	9.78E+10
5.75E-1	2.50E+16	5.94E+15	6.58E+14	1.32E+11	1.29E+11	1.10E+11
8.50E-1	1.28E+16	6.32E+14	2.40E+12	1.57E+11	1.18E+11	1.90E+10
1.25E+0	2.28E+15	4.70E+14	8.49E+11	1.78E+09	4.97E+09	2.77E+10
1.75E+0	1.15E+14	7.21E+12	5.85E+10	7.84E+07	2.77E+09	2.21E+10
2.25E+0	1.42E+14	8.42E+10	2.09E+07	2.28E+07	8.45E+08	6.66E+09
2.75E+0	3.16E+12	7.76E+09	6.47E+08	3.07E+06	1.57E+07	1.16E+08
3.50E+0	3.99E+11	9.77E+08	7.72E+06	2.41E+06	3.62E+06	2.20E+07
5.00E+0	1.04E+08	6.76E+07	3.30E+06	1.01E+06	3.72E+05	1.00E+05
7.00E+0	1.20E+07	7.80E+06	3.79E+05	1.16E+05	4.28E+04	1.16E+04
9.50E+0	1.37E+06	8.96E+05	4.35E+04	1.33E+04	4.92E+03	1.33E+03
TOTAL	1.05E+17	1.37E+16	1.50E+15	4.50E+13	5.82E+12	9.47E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceeding and following value.

Table 2.4.22 Variation in photon production (photons/s·MTIHM) as a function of time since discharge from a 33,000 MWd/MTIHM PWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	2.46E+16	1.91E+15	2.50E+14	1.14E+13	1.96E+12	2.41E+11
2.50E-2	5.55E+15	4.13E+14	4.61E+13	8.68E+11	2.86E+10	2.45E+10
3.75E-2	5.69E+15	4.95E+14	5.14E+13	1.17E+11	2.47E+10	1.41E+10
5.75E-2	5.11E+15	3.82E+14	9.07E+13	1.23E+13	2.14E+10	1.53E+10
8.50E-1	3.61E+15	2.24E+14	2.37E+13	4.76E+11	2.26E+11	5.51E+10
1.25E-1	4.26E+15	2.12E+14	1.52E+13	2.95E+11	1.27E+11	9.33E+09
2.25E-1	3.15E+15	1.86E+14	1.95E+13	1.94E+11	8.77E+10	2.46E+10
3.75E-1	1.72E+15	9.09E+13	8.22E+12	6.58E+10	5.76E+10	5.32E+10
5.75E-1	1.33E+16	3.29E+15	3.74E+14	7.00E+10	6.82E+10	5.69E+10
8.50E-1	7.55E+15	2.65E+14	1.46E+12	8.85E+10	6.64E+10	1.00E+10
1.25E+0	1.36E+15	2.64E+14	4.97E+11	9.65E+08	2.56E+09	1.40E+10
1.75E+0	7.70E+13	3.13E+12	3.55E+10	4.49E+07	1.39E+09	1.12E+10
2.25E+0	1.42E+14	7.27E+10	5.96E+06	1.07E+07	4.22E+08	3.36E+09
2.75E+0	2.25E+12	4.78E+09	1.76E+08	6.96E+05	7.58E+06	5.85E+07
3.50E+0	2.79E+11	5.91E+08	1.12E+06	4.68E+05	1.60E+06	1.10E+07
5.00E+0	1.46E+07	8.19E+06	4.74E+05	1.86E+05	9.35E+04	3.67E+04
7.00E+0	1.69E+06	9.44E+05	5.40E+04	2.12E+04	1.07E+04	4.22E+03
9.50E+0	1.94E+05	1.08E+05	6.17E+03	2.43E+03	1.23E+03	4.86E+02
TOTAL	7.61E+16	7.73E+15	8.80E+14	2.59E+13	2.67E+12	5.33E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceding and following value.

Table 2.4.23 Variation in photon production (photons/s·MTIHM) as a function of time since discharge from a 40,000 MWd/MTIHM BWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	1.95E+16	2.22E+15	2.94E+14	1.33E+13	2.26E+12	2.86E+11
2.50E-2	4.48E+15	4.73E+14	5.35E+13	1.01E+12	3.41E+10	2.90E+10
3.75E-2	4.48E+15	5.85E+14	6.00E+13	1.56E+11	3.72E+10	1.69E+10
5.75E-2	4.04E+15	4.44E+14	1.05E+14	1.43E+13	2.75E+10	1.83E+10
8.50E-1	2.83E+15	2.63E+14	2.77E+13	7.68E+11	3.55E+11	6.61E+10
1.25E-1	3.23E+15	2.60E+14	1.78E+13	4.82E+11	2.08E+11	1.12E+10
2.25E-1	2.48E+15	2.17E+14	2.27E+13	3.20E+11	1.42E+11	3.01E+10
3.75E-1	1.38E+15	1.03E+14	9.53E+12	8.56E+10	7.12E+10	6.39E+10
5.75E-1	1.40E+16	3.86E+15	4.40E+14	8.01E+10	7.83E+10	6.77E+10
8.50E-1	7.11E+15	3.25E+14	1.60E+12	7.18E+09	6.70E+09	8.87E+09
1.25E+0	1.06E+15	2.00E+14	5.87E+11	1.10E+09	3.13E+09	1.77E+10
1.75E+0	6.70E+13	4.11E+12	4.14E+10	5.29E+07	1.76E+09	1.41E+10
2.25E+0	9.85E+13	5.42E+10	8.17E+06	1.33E+07	5.35E+08	4.24E+09
2.75E+0	1.89E+12	4.26E+09	2.32E+08	9.64E+05	9.64E+06	7.39E+07
3.50E+0	2.37E+11	5.24E+08	1.96E+06	6.72E+05	2.05E+06	1.40E+07
5.00E+0	2.79E+07	1.64E+07	8.33E+05	2.70E+05	1.28E+05	5.04E+04
7.00E+0	3.21E+06	1.89E+06	9.54E+04	3.09E+04	1.48E+04	5.80E+03
9.50E+0	3.69E+05	2.17E+05	1.09E+04	3.54E+03	1.70E+03	6.67E+02
TOTAL	6.48E+16	8.96E+15	1.03E+15	3.05E+13	3.23E+12	6.34E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceding and following value.

Table 2.4.24 Variation in photon production (photons/s·MTIHM) as a function of time since discharge from a 27,500 MWd/MTIHM BWR (Includes all structural material) (Source: Roddy 1986)

	Time since discharge (years)					
Emean[a]	1.0E+0	1.0E+1	1.0E+2	1.0E+3	1.0E+4	1.0E+5
1.00E-2	1.76E+16	1.56E+15	2.08E+14	1.02E+13	1.75E+12	2.12E+11
2.50E-2	4.06E+15	3.40E+14	3.82E+13	7.81E+11	2.23E+10	2.07E+10
3.75E-2	4.08E+15	4.06E+14	4.24E+13	9.89E+10	1.92E+10	1.21E+10
5.75E-2	3.67E+15	3.15E+14	7.88E+13	1.11E+13	1.86E+10	1.31E+10
8.50E-1	2.58E+15	1.83E+14	1.95E+13	3.65E+11	1.75E+11	4.72E+10
1.25E-1	3.01E+15	1.71E+14	1.25E+13	2.25E+11	9.72E+10	8.10E+09
2.25E-1	2.26E+15	1.52E+14	1.60E+13	1.47E+11	6.69E+10	2.09E+10
3.75E-1	1.25E+15	7.51E+13	6.76E+12	5.32E+10	4.74E+10	4.48E+10
5.75E-1	9.95E+15	2.69E+15	3.10E+14	5.64E+10	5.49E+10	4.64E+10
8.50E-1	5.39E+15	1.96E+14	1.13E+12	5.15E+09	4.82E+09	5.99E+09
1.25E+0	7.79E+14	1.33E+14	4.06E+11	7.94E+08	2.12E+09	1.17E+10
1.75E+0	5.57E+13	2.47E+12	2.91E+10	3.94E+07	1.15E+09	9.28E+09
2.25E+0	9.80E+13	5.11E+10	4.74E+06	8.88E+06	3.49E+08	2.79E+09
2.75E+0	1.64E+12	3.46E+09	1.02E+08	5.69E+05	6.27E+06	4.86E+07
3.50E+0	2.04E+11	4.32E+08	8.31E+05	3.88E+05	1.32E+06	9.18E+06
5.00E+0	1.06E+07	5.53E+06	3.52E+05	1.54E+05	7.84E+04	3.04E+04
7.00E+0	1.22E+06	6.37E+05	4.00E+04	1.76E+04	8.98E+03	3.49E+03
9.50E+0	1.41E+05	7.32E+04	4.57E+03	2.01E+03	1.03E+03	4.02E+02
TOTAL	5.48E+16	6.23E+15	7.33E+14	2.30E+13	2.26E+12	4.55E+11

[a]Energy is given in MeV and covers a range which is equal distance between the preceding and following value.

2.5 DEFECTIVE FUEL

2.5.1 Introduction

Prior to 1983, nuclear fuel performance was an interest of the fuel vendors and the owner-operators of power reactors primarily from an economic viewpoint. The NRC was concerned with in-core fuel performance and out-of-core fuel storage from the viewpoint of radiation protection and nuclear materials safeguard. The interest of the U.S. Department of Energy was primarily in the area of research and development. Spent nuclear fuel (SNF), and particularly defective SNF in storage pools, did not receive the same degree of attention as in-service fuel performance. However, with the advent of the Nuclear Waste Policy Act (NWPA) of 1982 and the requirement for eventual disposal of SNF, including defective SNF, the systematic characterization of these materials has become important because they may require special handling during storage, transport, or emplacement, or special analysis for post-closure performance

This section characterizes and categorizes defective SNF and analyzes the currently available data from the perspective of establishing a data base containing pertinent information on defective Light Water Reactor fuel to support the programs of the OCRWM.

2.5.2 Description of Irradiated Fuel Defects

Defective SNF consists of both assembly and rod failures and defects.

2.5.2.1 Defective Fuel Assemblies

A defective fuel assembly is one that has damage to the assembly hardware or that contains one or several defective fuel rods. Fuel assemblies may be damaged in several ways, including bowing, mechanical parts failure, and handling damage.

Handling is the most common cause of fuel assembly damage. The grid spacers of fuel assemblies may be damaged on the corners when the assembly is being shuffled during refueling. Alignment pins at the bottom of an assembly may be sheared off or bent. The holddown springs at the top of the assemblies may break due to vibration or other causes. Damage to the grid spacers or other portions of the assembly cage usually results in flow-induced fretting or vibration damage. At times a broken piece can lodge in the fuel channels and damage two or three rods.

2.5.2.2 Defective Fuel Rods

A defective fuel rod is one that suffers cladding failure (failed fuel) or becomes flawed through some physical or chemical damage.

A fuel rod fails when the cladding is breached, resulting in release of fission products from inside the fuel cladding. The cladding may fail from pellet-clad interaction (PCI) which is the differential movement of the fuel and the cladding following a rapid power transient. PCI is now well understood and has been practically eliminated by improved fuel pellet design. Also, some vendors now use a thin layer of pure zirconium metal on the interior of the cladding to further minimize this source of failure.

The cladding may also fail from the inside due to the release of water vapor or fission products from the fuel pellet during power operation. Water vapor released from the fuel when in service reacts with the zircaloy cladding, causing hydride embrittlement; this problem has been solved by using a higher fired fuel with a lower water content. Gaseous fission products released into the gap between the fuel pellet and the cladding cause a decrease in the thermal conductivity of the initially helium-filled gap.

The cladding may fail from the outside due to corrosion caused by reactor water and the impurities it carries. The initial corrosion of the cladding results in an oxide coating which protects the base material from further rapid attack. But additional crud will deposit inhomogeneously on the outside of this coating while the fuel rod is in service. The composition of the crud depends on the primary system hardware constitution and reactor water chemistry. When the crud is copper-rich, such as in the case of BWR reactors using brass condenser tubes, a phenomenon called crud-induced localized corrosion (CILC) can be quite pronounced, particularly at

locations with high radiation but low heat fluxes, such as with UO_2/GdO_2 burnable poison fuel rods (Marlowe 1985, Bailey 1985, Cheng 1985).

Other clad failure modes include mechanical effects such as rubbing of metallic parts due to flow-induced vibrations, debris lodging in the fuel channels, or water jetting due to certain flow imbalances or blockages. Welding defects, dropping, and excessive stress in handling are also frequent causes of clad failure.

In addition, under the effects of radiation, temperature, and pressure, fuel cladding undergoes a decrease in diameter and an increase in length. Quantitative data have been obtained for this phenomenon. Nonuniform neutron fluxes also cause the fuel rod to deform or bow because of differential changes in dimension. The deformation or bowing may become excessive and result in damage to the fuel rod or neighboring rods (Franklin 1983, Bailey 1985, Marlowe 1985).

2.5.3 Methods for Detecting Defective Fuel

Both indirect and direct methods may be used for detecting defective fuel.

2.5.3.1 Indirect Fuel Monitoring Methods

During reactor operation, the fuel rod reliability is typically monitored indirectly by measuring the activity levels of certain fission products in the reactor coolant or in the off-gases. These measured activities are compared against standard levels to infer the number of fuel rods that have leaks.

All three PWR reactor manufacturers measure iodine-131 activity (and other fission product activities) in the circulating reactor coolant to infer fuel reliability. This method takes into account fuel burnup, power transients, radioactive decay, and actions of the reactor coolant cleanup system. The General Electric Company also uses activities of noble gases (Xe-138, Kr-87, Kr-85m, Xe-135, and Xe-133) in the off-gases to characterize the fuel failure type and whether the failure is stable or increasing in severity. This approach for indirect monitoring of fuel performance provides an overall indication of fuel integrity and, when a threshold is exceeded, serves as a signal to investigate causes of failure and methods for improvement. It cannot monitor defects other than leaks, and it cannot

differentiate whether high activity is caused by one large leak, several small leaks, or by "tramp" uranium that adheres to the outside surface of the fuel cladding.

2.5.3.2 Direct Fuel Inspection Methods

Direct fuel inspection methods are employed during refueling or shutdown maintenance periods and are often referred to collectively as "poolside inspection."

Remote viewing, television, underwater periscopic examination, and photography are visual inspection methods; as a group they are most widely used to examine fuel assemblies and fuel rods. Damaged assembly hardware, excessive surface corrosion, surface cracking, excessive bowing, and cracked end cap welds can be observed during refueling operations. However, the fuel rods in an assembly are visible only to the second or third row and none of the rods is fully visible.

The existence of a leaking fuel rod within a fuel assembly may be confirmed by a process called "sipping," which may be performed in the core or in the storage pool. In this procedure the test fuel assembly is isolated in a can filled with clean water and a count with a scintillation counter is taken. If a gas is to be sampled, the can is sealed and flushed completely with clean water and a portion of the liquid is removed from the bottom of the can; the water may be boiled and the gas taken as a vapor and sampled to determine the isotopic concentration of volatile fission products. If water is to be sampled, the can is filled with clean water, sealed, and the water is allowed to heat up; a sample is removed from the bottom of the can and counted for activity in a multichannel analyzer.

Ultrasonic scanning has become one of the more important techniques for inspecting fuel assemblies. This method is now being used by some utilities to routinely inspect all fuel rods that are removed for storage or for fuel shuffling at the end of a cycle. Ultrasonic scanning is an effective and reliable method for determining the presence of water inside the fuel cladding (and therefore a leaked fuel rod); it is rapid, can be done without removing the fuel rods from the assemblies, produces a storable record, and can be coupled with TV cameras; and is not sensitive to cooling time (as is sipping).

The eddy-current method is used to test for fuel rod integrity and incipient failure of the clad. This method requires the fuel bundle be disassembled in order to test each rod separately. The rod is passed through a coil and acts as the core of a magnet. An alternating current passing through the coil generates a magnetic field within the fuel rod and counteracting eddy currents in the clad. When the clad has a flaw, a change is produced in the counteracting eddy current. While this method is very accurate and can detect flaws other than leaks, it is not of common use since it requires dismantling the assemblies.

Dimensional measurements on fuel bundles and rods are usually made by the vendors to verify calculation methods and to resolve unusual fuel performance problems. These measurements are made at the poolside or in hot cells. The measurements are usually time consuming and are of little interest to the utilities. The vendors are working at automating dimensional measurements to decrease time requirements.

2.5.4 Data Sources

The data used in this study include utilities' and fuel vendors' data on nuclear fuel performance, the Nuclear Regulatory Commission's data based on licensee event reports (LERs) and topical reports, nuclear fuel service companies' data on inspection of discharged fuel, and the Department of Energy/Energy Information Administration's (DOE/EIA) data base RW-859. Research and development data from the Electric Power Research Institute (EPRI) were also examined where they relate to spent fuel defects.

The majority of the information in fuel performance reports and LERs deals with fuel rod failure while in service; it has not been standardized and is primarily concerned with regulatory requirements. The information in the DOE/EIA-RW-859 was submitted by the utilities in response to EIA questionnaires. While these data are neither uniform nor complete, they are the result of direct poolside observations. The data from the DOE/EIA-RW-859 file are in terms of fuel assemblies, while the data from the fuel vendors and NRC files are mainly in terms of fuel rods.

2.5.5 Approaches to Categorization of Defective SNF

Basically, there are two principal reasons for the systematic study of defective nuclear fuel: to improve future nuclear fuel in-core performance and to facilitate the ultimate disposal of SNF. This leads to two general approaches to descriptive categorization of defects in SNF, as outlined below.

2.5.5.1 Categorization in Terms of Fuel Performance

This approach is commonly used by research and development projects which attempt to determine the nature of the defect in order to improve fuel performance.

1. Categories by nature of defect:
   - o visually observed abnormal degradation (e.g., color, shape, dimension, handling difficulties)
   - o pin-hole leak (detectable by ultrasonic, eddy current, sipping, or other means)
   - o circumferential crack
   - o longitudinal split
   - o gross cladding failure

2. Categories by operational cause of defect:
   - o water chemistry
   - o flow-induced vibration
   - o jetting
   - o manufacturing defects
   - o handling defects

3. Categories by physical/chemical mechanisms:
   - o water corrosion
   - o hydriding
   - o localized crud-induced corrosion
   - o pellet-clad interaction
   - o radiation-induced bowing
   - o fretting
   - o excessive stress

4. Categories by detection techniques:
   - o coolant/off-gas radioactivity analysis
   - o poolside sipping
   - o poolside gamma-ray scanning
   - o poolside eddy-current testing
   - o poolside ultrasonic scanning
   - o refueling and poolside visual inspection

2.5.5.2 Categories Defined in 10CFR961

The original DOE categorization system for SNF was published in 1983 as part of regulation 10CFR961, which establishes the procedures for the transfer of spent nuclear fuel to the federal government.

10CFR961 mentions three classes of failed fuel as follows:

Class F-1	Visual Failure or Damage
Class F-2	Radioactive "Leakage"
Class F-3	Encapsulated

From the viewpoint of characterization, this system has a number of shortcomings:

- The definition of encapsulated fuel, Class F-3, as those that were encapsulated prior to 1983, is overly restrictive and precludes use of the category for post-1983 SNF.
- There is no definition of "special handling" although the term is used many times in the document. The distinction between "failed fuel" and "defective fuel" is unclear, although failed fuel often means a leaker that exceeded NRC release limits.

2.5.5.3 Categories Defined in DOE/EIA Form RW-859

Another classification system for defective fuel is provided in EIA form RW-859 which must be filled out annually by reactor owner-operators (EIA 1983). Among other information, this form asks the respondent to fill out the Defective Assembly Section, "if known," and to use up to three of the following defect codes:

Code 1. Visually observed failure or damage
Code 2. Encapsulated or other remedial action taken
Code 3. Requires special handling
Code 4. Cannot be consolidated
Code 5. Physically deformed
Code 6. Does not fit in pool rack
Code 7. Clad damage (mechanical, chemical, or other--possibly detectable by ultrasonic means)
(Code 8.) (Not listed)
Code 9. Other

This system also has some shortcomings from the characterization viewpoint:

- Code 7 implies a leaker, but not necessarily so.
- While RW-859 is mandatory, the section on defective fuel assemblies is not because of the qualification "if known." Much of the information is, in fact, not known by the utilities.
- The codes are not mutually exclusive. For example, Code 1, Code 5, and Code 6 are almost redundant.
- Some codes are subjective. For example, Code 3 and Code 4 require subjective judgment that depends on the observer, on the state of fuel handling technology, and on the utility programs.

The RW-859 form is presently (1987) undergoing review and revision, including the defective assemblies section. The above codes may or may not be revised in this process.

2.5.5.4 Reconciliation of the Categories

Characterization of defective fuel must work with existing data. In the statistical treatment presented later in this section, data from the RW-859 form are used in their original codified format, and also by collapsing the data into the three categories of 10CFR961. In such consolidation, the F-1 category has been assumed to include fuel with RW-859 defect codes 1, 3, 4, 5, and 6 (however, codes 4 and 6 have not yet been used). Category F-2 includes data having defect code 7, and category F-3 includes data having defect code 2. For this treatment, the F-1, 2, 3 categories are taken as mutually exclusive.

In addition, a reconciliation must be made between DOE/EIA RW-859 data on defective fuel assemblies and NRC data on failed fuel rods. Again, in the statistical data section, we have not only examined the data separately but have also attempted to correlate the data using (a) information on "failed" assemblies among "defective" assemblies, (b) information on the average number of failed rods per failed assembly, and (c) the average number of fuel rods within a fuel assembly.

2.5.6 Statistical Information

The various sources for statistical information on LWR fuel performance and defective fuel in the United States obtain their data from the same basic source: power reactors that have been or are in operation. However, for a variety of reasons, the nature and quality of data vary from source to source. Following is a summary of the statistical information available from the sources that are of interest to this study.

2.5.6.1 Data from Nuclear Utilities

Utilities have a keen interest in defective fuel while it is in the core because of their concern in keeping radiation levels as low as possible, to comply with regulatory restrictions, and maintaining "good practices" in a peer environment promoted by the Institute of Nuclear Power Operations. However, utilities have little incentive to determine which or how many of the total discharged fuel assemblies already in the storage pool are defective, for two reasons: (a) so far defective spent fuel does not present any special handling problems out-of-core and (b) inspecting for defects and types of defects requires time and money but results in no immediate benefits.

Every nuclear utility has a nuclear fuel group that has the function of buying the fuel, obtaining fuel warranty, optimizing fuel utilization, following fuel performance, and storing SNF. Tracking and maintaining a record on failed fuel is important to improving plant capacity factor, meeting as-low-as-reasonably achievable (ALARA) occupational burdens, reconstituting defective assemblies, and obtaining warranty claims. However, such records are generally proprietary in nature.

A statutory requirement imposed on the nuclear utilities by the Nuclear Regulatory Commission is the filing of a licensee event report (LER) whenever there is an off-normal event that results in a violation of technical specifications or in an unanalyzed condition (NRC 1982). Pertinent data from the LERs have been studied by the NRC and are discussed in section 2.5.6.5.

Another statutory requirement is imposed on the nuclear utilities by the DOE/EIA by authority of the NWPA: nuclear utilities must fill out the Nuclear Fuel Data Form RW-859 annually. However, utilities do not always have all of the data required by the form. In addition, they sometimes

omit data for which the RW-859 form allows voluntary submission (e.g., on an "if known" basis). Data from RW-859 are discussed in section 2.5.6.6.

2.5.6.2 Data from Nuclear Fuel Vendors

Nuclear fuel vendors are reluctant to make available their fuel reliability data, mostly for proprietary reasons. However, on occasion fuel vendors give papers at technical or trade meetings. Pertinent data from these papers are presented in Table 2.5.1. Since there is not a common basis for the reported data, one should be cautious in drawing conclusions or making comparisons.

o Advanced Nuclear Fuel Corporation (formerly Exxon Nuclear Company)

As of January 1987, Advanced Nuclear Fuel (ANF) had 10,564 fuel assemblies containing over 1,350,000 fuel rods irradiated in commercial power reactors. ANF reported the fuel failure rates in two parts: (a) the part that is definitely traceable to nuclear fuel design, manufacture, or warranty and (b) the part that is related to nuclear plant operation such as power transients, coolant chemistry, and fuel handling. The details of ANF statistics are reported by Sofer 1985, Sofer 1987.

o Babcock & Wilcox (B&W) Fuels

As of April 1985 B&W had fabricated more than 5,600 fuel assemblies containing over 1.2 million fuel rods. B&W also manufactured the 32,000 TMI-2 fuel rods, which are not included in the statistics of this section (these are characterized in section 2.6.1.).

The B&W calculations account for the release of volatile fission products to the coolant and the removal of those products by the reactor coolant cleanup system (Mayer 1980). B&W follows the industry convention of using iodine-131 as the marker isotope. It reports fuel performance data in terms of a measure for the circulating activity in the reactor coolant (Mayer 1980, Matheson 1985, Pyecha 1985). This measure is called "Failed Fuel Index" and is the ratio of the circulating activity to the product of rod number and burnup.

o Combustion Engineering (CE)

Similar to B&W, CE monitors the circulating iodine-131 activity in its reactors and infers the percentage of failed fuel rods. The reported failure rates are the same as those reported to the Nuclear Regulatory Commission.

o General Electric Company (GE)

As of December 31, 1984, General Electric Company has fabricated over 55,000 fuel assemblies containing approximately 3.2 million fuel rods. No information on the overall performance of these fuel rods is available directly from GE. However, GE reported that data for 1983 indicated only 0.007% failure for 8 x 8 array fuel and no failure for Zr-barrier 8 x 8 array fuel (Baily 1985).

Burnups as high as 58,000 MWD/T have been achieved by several test BWR fuel assemblies with no noticeable life-limiting effects on the fuel rod. However, at higher burnups increased production of helium is observed (due to ternary fissions and alpha decay of transuranic products) and also a decrease in fuel rod diameter (Baily 1985, Marlowe 1985, Cheng 1985).

o Westinghouse Electric Corporation

The Westinghouse Electric Corporation does not release information on defective SNF on grounds this is utility proprietary data (Miller 1987). There are no available statistics on Westinghouse fuel defect rates, except those that can be inferred from the NRC and DOE/EIA data. Westinghouse uses the indirect method of monitoring the fission product levels in the reactor primary system (Skaritka 1983, 1985).

2.5.6.3 Data from Nuclear Fuel Service Companies

The Brown Boveri Company (BBC) has provided data on the failure rates of BBC-inspected fuel and on the accuracy of the ultrasonic scanning method as compared to other methods. BBC markets the Failed Fuel Rod Detection System (FFRDS), which uses the ultrasonic method to inspect all PWR fuel rods within an intact fuel assembly. Because the method is based on the presence of water inside the fuel cladding, the FFRDS identifies leaked fuel pins.

As of December 1986, BBC has inspected 1,117 PWR assemblies involving 265,923 fuel rods in the United States and 2,022 PWR assemblies involving 406,471 fuel rods overseas. The U.S. data are presented in Table 2.5.2 (Snyder 1987).

2.5.6.4 Data from the Electric Power Research Institute (EPRI)

EPRI has a specialized fuel performance data base which is not, however, available for external use. The data base includes reactor-specific radioactivity of the reactor coolant and off-gases, fuel cycle history,

coolant chemistry data, and end-of-cycle examination data for failed assemblies (Franklin 1983, 1985; Bailey 1986b; Rumble 1980; Michaels 1985; Lawson 1986b). Communication with EPRI indicates that this data base is of little value to the present study (Franklin 1987).

An EPRI document, EPRI NP-4561, documented and interpreted LERs related to fuel handling and pool storage as of 1986 (Bailey 1986b). It was found that there was no evidence of further deterioration of SNF in storage in wet pools at the reactor sites.

2.5.6.5 Data from the Nuclear Regulatory Commission (NRC)

The data on defective fuel from the NRC comes from the nuclear utilities in the form of licensee event reports (LERs) and from the nuclear fuel vendors. These reports serve their own specific purposes, are selective, and are therefore difficult to use. Since 1978 the NRC has sponsored the Pacific Northwest Laboratory in compiling an annual report on fuel performance with the purpose of providing integrated information for licensing decisionmaking (Houston 1979, Tokar 1981, Bailey 1981b, 1982, 1984, 1985, 1986a).

Tables 2.5.3 and 2.5.4 show data on BWR and PWR fuel rod failures that have been abstracted from the available reports in the above series. Some of the data were calculated. Only failed fuel rod data are presented; other fuel defects are reported in the NRC report series but are more in the nature of isolated observations and are not included here. Since these data were not designed to fit into a unified data base structure, there may be omissions or inconsistencies; one must therefore be cautious in drawing conclusions or making comparisons.

Data from the LER data base that is maintained for the NRC by the Nuclear Operations Analysis Center (NOAC) of Oak Ridge National Laboratory (ORNL) were also reviewed. The nature and causes of off-normal fuel-related events that resulted in an LER were extracted. These data are presented in Table 2.5.5 (Cletcher 1987).

"Fuel failure" as defined in the LERs appears to be subjective. It may include events that did not result in fuel rod failure. As there were 43 reported assembly failures and 95 reported rod failures, the average number of rod failures per assembly failure is 2.2 (1.7 for BWRs and 2.3 for PWRs). This ratio compares favorably with direct inspection observations such as

those made by the Brown Boveri Fuel Service Company (Snyder 1987) using the ultrasonic method.

2.5.6.6 Data From DOE/EIA

The following data were extracted from the RW-859 data file, Section 5, dated Nov. 14, 1986 (Andress 1986). This includes all data extant as of Dec. 31, 1985, and possibly some data "to date" (terminology used in RW-859). It was indicated that the number of 1986-discharged SNF assemblies that were included in the data file for 1985 is very small (Andress 1987).

o Total Discharged and Total Defective Fuel Assembly Populations

As of Dec. 31, 1985, there were 28 BWRs and 52 PWRs that had discharged spent fuel into their spent fuel storage pools. An additional 3 BWRs and 8 PWRs were in operation but had not discharged any spent fuel. Furthermore, there were 10 BWRs and 14 PWRs that were in the process of fuel loading or construction. The total discharged SNF population and the defective SNF population are as follow:

	Total Number	Defective Number	Defective %
BWR fuel assemblies	27,446	3,374	12.3
Long-cooled (before 1981)	18,340	2,953	16.1
Short-cooled (1981-1985)	9,106	421	4.6
PWR fuel assemblies	18,123	1,317	7.3
Long-cooled (before 1981)	10,588	861	8.1
Short-cooled (1981-1985)	7,535	456	6.1

o Distribution of Total and Defective Discharged Fuel Assemblies by Reactor

The number of reactors reporting data is summarized as follows:

	BWRs	PWRs
Total number of reporting reactors	28	53
No. that also reported defective fuel	22(79%)	24(45%)

A higher fraction of BWR sites reported defective fuel than did the PWR sites. This difference may be explained by functional differences, namely:

- BWRs practice reconstitution to a greater degree;
- the channels on BWRs facilitate in-core sipping.

o Distribution of Total and Defective Fuel Assemblies by Rod Array

Many assembly designs have been developed for BWR and PWR fuel over the years. The designs were originally based on core physics, core configuration, manufacturing capability, and economics. They were subsequently

improved with new knowledge on in-service material behavior, new core cooling requirements (such as the emergency core cooling criteria), and higher burnup possibilities. More recently, zircaloy has almost completely replaced stainless steel and inconel as cladding material and grid spacers; the fuel pin diameter has decreased; fuel assemblies contain more fuel rods; fuel pellets are manufactured with higher density and with dished ends; and a zirconium metal barrier is added between the fuel and the zircaloy cladding. Quality assurance and improved manufacturing technologies have allowed burnup to increase from the original design values to higher than 36,000 MWD/T for BWRs and to higher than 40,000 MWD/T for PWRs. Along with these changes, in-service fuel failure has also drastically decreased.

Table 2.5.6 shows the statistics on defective rates of various assembly arrays. Since defective fuel data were not reported for 4254 discharged BWR and 8450 discharged PWR assemblies, the actual numbers with defective fuel may be larger than the values shown.

The average burnup of defective BWR assemblies is generally lower than the average burnup of intact assemblies of the same type. On the other hand, the average burnup of defective PWR assemblies is about the same as that of intact assemblies of the same type. This may be a reflection of the fact that leaky BWR fuel assemblies were easier to detect during refueling by the sipping method and were reconstituted or discharged early, whereas leaky PWR fuel assemblies tended to stay in the core for the entire burnup duty unless the leak was such that technical specifications or occupational radiation burdens were of concern.

o Distribution of Total and Defective Fuel by Defect Categories

Table 2.5.7 shows the statistics for defective fuel assemblies by RW-859 defect codes and by the three 10CFR961 defect categories. The reconciliation method presented in Section 2.5.5.4 has been used.

o Distribution of Total and Defective Discharged Assemblies by Burnup and by Year

The number of annual discharged spent fuel assemblies steadily increased between 1970 and 1985. This of course is a reflection of the increase in the number of reactors and the amount of nuclear electricity being generated by these reactors during the period. The number of defective discharged spent fuel assemblies, on the other hand, is showing a

down trend. This, in spite of the larger number of total discharged fuel assemblies, clearly indicates that fuel failure rates have been significantly reduced.

The average burnup also shows a clear upward trend, from 5000-6000 MWD/T to 25,000-30,000 MWD/T for BWR fuel and from 18,000-20,000 MWD/T to 30,000-35,000 MWD/T for PWR fuel. This trend continues as more reliable fuel and better reactor operation are achieved.

o Discharged Fuel Assembly Defect Rates

Because of incomplete reporting on the number of defective SNF assemblies in RW-859, data from this source have some uncertainties. Two estimates for the defect rates have been made, using two different assumptions.

Table 2.5.8 shows the "low" estimate. It is considered low because it includes the total discharged SNF population of those reactors that reported defective SNF at least once (but may have omitted some defective fuel). The defect rates are 9.9% and 10.5% for BWR and PWR discharged spent nuclear fuel, respectively. Table 2.5.9 shows the "high" estimate. It is considered high because only batches of discharged SNF that contained known defective fuel are included. The defective rates are 14.4% and 23.8% for BWR and PWR discharged spent nuclear fuel, respectively.

2.5.6.7 Reconciliation of Fuel Rod Failure Rates and Fuel Assembly Defect Rates

A reconciliation of fuel rod failure rates and fuel assembly defect rates has been approximated by using (a) the data on assembly defect category to convert defect rate to leaker rate (90% for BWRs and 54% for PWR), (b) the LER and Brown Boveri data for the approximate number of failed fuel rods in a failed assembly (1.6 for BWRs and 2.1 for PWRs), and (c) assuming 49 fuel rods in a BWR fuel assembly and 205 fuel rods each in a PWR fuel assembly. The results are:

	BWRs	PWRs
Fuel rod failure rates as determined from LERs, NRC reports, and vendors' reports (a)		
Prior to 1981	0.04-0.76%	0.01-0.05%
1981-1985	0.01-0.02%	0.01-0.05%
Fuel assembly defect rates as determined from EIA-RW-859 data file (b)		
Prior to 1981	2-41%	3-51%
1981-1985	1-20%	4-30%
Derived fuel rod failure rates from fuel assembly defect rates		
Prior to 1981	0.06-1.13%	0.02-0.28%
1981-1985	0.03-0.55%	0.02-0.17%

(a) From tables 2.5.3 and 2.5.4
(b) From tables 2.5.8 and 2.5.9. The value of about 94% for BWR fuel defect rate for 1973 was considered an outlier and therefore excluded.

The above reconciliation indicates that the number of leaky fuel rods in storage may be more than twice more numerous than can be inferred from the NRC-associated fuel reliability data. The difference is not surprising since the LER data would not be expected to include all known leakers, for a number of reasons, such as : (a) not all leakers trigger an LER and (b) some leakers are not identified until after final discharge.

2.5.7 Conclusions

Examination and analysis of defective fuel data from the various industry and government sources leads to the following conclusions:

1. The majority of data supplied by the nuclear utilities, fuel vendors, and nuclear fuel service companies deals with leaked fuel while the reactor is in operation. Fuel rod failure rates are between 0.02% and 0.07% yon a cumulative basis. Instances of rod failure a factor of 10 higher have been reported for certain fuel lots. Newer fuel is claimed to have a reliability factor up to 10 lower. While these data give confidence in the ability to operate reactors with very low fuel leakage, they do not provide

all the information on the defect rates of discharged SNF that is required for characterization.

2. Other data from the nuclear fuel vendors or from industry organizations, such as the Electric Power Research Institute, deal mostly with fuel research, development, and marketing and are of limited use in terms of overall characterization.

3. The major source of data that is of significance to the OCRWM mission has been collected since 1984 and updated annually by the DOE/EIA. The data collected for calendar year 1985 have shown significant improvement over that for 1984 and further improvements are expected in the future.

4. The percentage of PWR respondents that reported defective discharged fuel assemblies is less than that of BWR respondents. This may be due to the fact that (a) PWR fuel assemblies were not routinely checked for fuel defects in the same manner as BWR fuel assemblies because such a check is more expensive and yet may not lead to assembly reconstitution or fuel warranty claim and (b) the RW-859 form does not require reporting on fuel defects if such information is not available.

5. Discharged fuel assemblies between 1970 and 1985 show a clear trend of increase in number and in burnup. Defective assemblies, however, show a clear trend of decrease for BWRs, but a not-so-clear trend for PWRs.

6. A reconciliation of the fuel rod failure rates and fuel assembly defect rates can roughly be made by modifying the latter using the fraction of leaks among defects, the number of leaked fuel rods in a defective assembly, and the number of fuel rods within a fuel assembly. This reconciliation leads to a number of leaked fuel rods now in storage more than twice the number that can be calculated from NRC-associated fuel reliability data.

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Table 2.5.1. DATA FROM NUCLEAR FUEL VENDORS

Fuel Vendor	Irradiated Fuel Rods	Average Rod Failure Rate
Advanced Nuclear Fuel (Formerly Exxon)	(cumulative as of 1/1987)	
BWR	365,938	0.027%
PWR	1,002,495	0.014%
Babcock & Wilcox		
1982	287,872	0.006%
1983	336,128	0.009%
1984	310,000	0.011%
Combustion Engineering		
1982	not available	0.01%
1983	not available	0.02%
1984	333,883	0.02%
General Electric		
1983	not available	0.007%
Westinghouse	not available	not available

Table 2.5.2. DATA FROM POOLSIDE ULTRASONIC INSPECTION
(Source: Snyder 1987)

	Fuel Assemblies			Fuel Rods		
Fuel Type	Total Number	No. with Leaked Rods	Percent Leaked	Total[c] Number	No. of Leaked Rods	Percent Leaked
B&W 15[a]	117	14	12.0	26,325	18	0.07
CE 14	7	4	57.1	1,372	4	0.29
CE 16	269	35	13.0	68,864	124	0.18
Exxon 16	36	2	5.6	9,216	2	0.02
W 14CE[b]	65	42	64.6	12,740	51	0.40
W 14	117	19	16.2	22,932	39	0.17
W 15	340	86	25.3	76,500	149	0.19
W 17	166	14	8.4	47,947	29	0.06
Total	1,117	216	19.3	265,923	416	0.16

(a) B&W = Babcock and Wilcox, CE = Combustion Engineering, Exxon = Exxon Nuclear Company (changed to Advanced Nuclear Fuel Corporation as of 1987), W = Westinghouse.

(b) W 14 CE is 14 x 14 fuel assemblies manufactured by W for a CE reactor.

(c) Brown Boveri has counted the rod positions available in an assembly as the number of fuel rods. This is a high estimate of fuel rods because there are also control rods in some of those positions.

Table 2.5.3. ABSTRACT OF BWR FUEL ROD FAILURE DATA REPORTED IN NRC ANNUAL REPORTS (a)

Fuel Vendors and Dates	Incore and Discharged Fuel Rods		Failure Rates		Calculated No. of Defective Rods	
	Annual	Cumulative	Annual	Cumulative	Annual	Cumulative
Exxon Nuclear Co.(b)						
12/81	--	116,891	--	0.012%	--	14
12/82	32,391	142,471	0.130%	--	42	56
12/83	45,801	170,710	0.012%	--	6	62
12/83	53,324	172,176	0.022%	0.043%	12	74
GE 7x7 Array(c)						
9/71	--	>440,000	--	0.200%	--	880
9/74	--	>810,000	--	0.760%	--	6156
12/76	--	1,040,000(e)	--	Proprietary	--	--
12/76	--	110,000I, R(d)	--	0.043%	--	47R
1/80	--	285,376I, R	--	0.066%R	--	188
1/80	504,161	--	1.010%	--	5092	--
GE 8x8 Array						
5/79	--	676,053	--	0.028%R	--	189(e)
	--	117,676I	--	0.	--	0
1/80	--	758,016	--	0.016%R	--	121(e)
	--	268,398I	--	0.002%R	--	5
12/80	--	1,239,000	--	0.020%R	--	248
12/81	--	1,489,000	--	<0.020%R	--	<298
12/82	--	1,821,338	--	0.020%R	--	364
12/83	1,300,000R	--	0.007%R	--	91	455
12/84	1,300,000R	>2,460,840R	0.019%R	0.024%R	130	585

(a) Source: Compiled from Houston 1979; Tokar 1981; Bailey 1981b, 1982, 1984, 1985, 1986a.

(b) Note that the numbers given here are slightly different from those reported by Exxon Nuclear Company (Sofer, 1985).

(c) Counting of 7x7 arrays no longer reported beyond 1980.

(d) I = improved (rearranged) array; R = reported (not calculated) data.

(e) The reported numbers are duly reported; however, they may be inconsistent with other reported or calculated numbers. Efforts to resolve the inconsistencies have not been successful. At any rate, the accuracy of the individual reported reliability should not be of much importance.

Table 2.5.4. ABSTRACT OF PWR FUEL ROD FAILURE DATA REPORTED IN NRC ANNUAL REPORTS (a)

Fuel Vendors and Dates	Incore and Discharged Fuel Rods(b,c) Annual	Incore and Discharged Fuel Rods(b,c) Cumulative	Failure Rates (c) Reported Annual	Failure Rates (c) Calculated Cumulative	Calculated (c,d) No. of Defective Rods Annual	Calculated (c,d) No. of Defective Rods Cumulative
Babcock & Wilcox						
12/80	--	586,034R	--	0.037%	--	216R
12/81	--	647,728R	--	0.037%	24R	240
12/82	--	686,608R	--	0.037%	30R	256
12/83	--	769,184R	--	0.037%	30R	286
12/84	--	797,472R	--	0.040%	33R	319
Combustion Engineering						
12/79 to 12/81	--	579,201R	--	0.047%	--	274R
12/82	266,000R	680,613R	0.010%R	0.044%	27	301
12/83	333,700R	780,872				
12/84	333,883R	839,980R	0.020%R	0.052%	67	435
Exxon Nuclear						
12/81	--	410,965R	--	0.013%R	--	52R
12/82	160,856R	547,725R	0.110%R(e)	0.042%	177	229
12/83	237,807R	623,572R	0.013%R	0.042%	31	260
12/84	339,013R	798,737R	0.002%R	0.033%R	7	267
Westinghouse	No data available except a plot of reactor coolant activity of Westinghouse reactors between 1972-1984					

(a) Source: Complied from Houston 1979; Tokar 1981; Bailey 1981b, 1982, 1984, 1985, 1986a.

(b) Note that annual numbers cannot be added to obtain the cumulative number because fuel rods stay in the core for more than one year.

(c) R = reported (not calculated)

(d) The calculated cumulative failure rate was obtained by dividing the calculated cumulative number of defective fuel rods by the reported number of cumulative fuel rods. The calculated number of defective fuel rods was obtained by multiplying the reported failure rate by the population of exposed fuel rods.

(e) This figure appears to be high by a factor of 10 in the source document.

Table 2.5.5 LICENSEE EVENT REPORT (LER) DATA [a]
(Cletcher 1987)

	Fuel Rods		Fuel Assemblies	
	BWR	PWR	BWR	PWR
Total number of failures	10	85	6	37
Nature of failure				
Leaks	9	73	4	20
Deformation/maladjustment	1	11	2	16
Others	0	1	0	1
Cause of failure				
Cladding degradation	80%		19%	
Vibration	6%		-	
Component failure	5%		56%	
Wear, age	2%		7%	
Drop, impact	-		2%	
Others	7%		16%	

[a] Data analysis by Cletcher 1987 for LER data taken from computerized data base maintained by the Nuclear Operations Analysis Center of the Oak Ridge National Laboratory.

Table 2.5.6 DEFECT RATES BY ARRAY
(Data corrected for those reactors that did not report defects as of 12/31/1985)
Source: DOE/EIA RW-859

Array	Total No. Discharged Assemblies	No. Discharged That Reported Defects	No. Of Defective Assemblies	Rate (%)
		Boiling Water Reactors		
Unknown	12,709	10,623	1,362	12.8
6 x 6	888	888	168	18.9
7 x 7	5,308	4,200	260	6.2
7 x 7NP[a]	484	484	10	2.1
8 x 8	4,851	2,917[d]	32[d]	1.1
8 x 8NP[b]	920	920	81	8.8
8 x 8R [c]	1,859	1,531	128	8.4
9 x 9	78	78	24	30.8
10 x 10	235	235	102	43.4
11 x 11	114	114	5	4.4
TOTAL	27,446	21,990	2,172	9.9
		Pressurized Water Reactors		
Unknown	4,573	3,568	687	19.3
4 x 4	1	0	-	-
14 x 14	3,482	363	6	1.7
14 x 14[e]	1	0	-	-
15 x 15	6,704	3,661	200	5.5
16 x 16	719	226[f]	5[f]	2.2
17 x 17	2,643	1,514	78	5.2
TOTAL	18,123	9,332	976	10.5

BWR notes: [a] At Hatch 1; [b] at Hatch 1 & 2; [c] at Browns Ferry 1, 2, & 3; Peach Bottom 2 & 3; Fitzpatrick; and Susquehanna 1; [d] excludes 1,202 at Vermont Yankee.

PWR notes: [e] A special assembly at Calvert Cliff; [f] excludes 341 at Yankee Rowe.

Table 2.5.7. DEFECTIVE FUEL ASSEMBLIES BY DEFECT CATEGORY AND/OR CODE

Defect Category or Code[a]	Boiling Water Reactors			Pressurized Water Reactors		
	Number	Weight(T)	Average Burnup (MWD/T)	Number	Weight(T)	Average Burnup (MWD/T)
			Data from RW-859			
F-1	135	24.752	22,670	398	181.249	31,144
F-2	1,860	344.729	12,894	206	91.468	24,501
F12	-	-	-	60	27.246	35,585
F-3	46	8.765	1,134	-	-	-
LAT(ER)[a]	(691)	(126.962)	23,350	(341)	(83.159)	27,605
1	7	1.271	19,182	14	5.885	25,193
1 7	151	27.897	24,123	-	-	-
15	48	5.760	14,168	28	12.724	28,995
137	-	-	-	2	0.734	26,406
17	8	1.459	13,986	63	25.676	25,705
157	-	-	-	13	5.950	9,633
3	11	2.020	13,609	5	1.795	20,744
5	2	0.304	17,854	-	-	-
7	406	71.296	15,697	184	81.664	25,193
71	7	1.282	22,339	3	1.388	19,064
715	1	0.183	23,753	-	-	-
751	1	0.183	23,416	-	-	-
TOTAL	3,374	616.863	16,162	1,276	435.78	28,105
			Reconciliation with 10CFR961[b]			
F-1	203 (7%)	34.107	20,525	445 (46%)	201.653	30,742
F-2	2,434 (91%)	445.576	14,123	531 (54%)	234.126	25,760
F-3	46 (2%)	8.765	1,134	-	-	-
LAT(ER)	691	126.962	23,350	341	83.159	27,605
TOTAL	3,374	615.410	16,162	1,317	518.938	28,105

[a] The definition of categories and codes has been given in the text.
LAT(ER) means that the defective fuel will be reported at a later date.

[b] Reconciliation is achieved by assigning all items having a defect codes F12 and 7 to category F-2; defect code 2 (there is none) to category F-3; and defect codes 1, 3, 4, 5, 6, (but not F12, 2 or 7) to category F-1.

Table 2.5.8. DEFECT RATES OF ASSEMBLIES: LOW ESTIMATE

(All Discharged SNF from Reactors that Reported Defective Fuel One Time or More Are Included [a])

	Boiling Water Reactors			Pressurized Water Reactors		
	No. of Assemblies	No. Ass. Defective	Defect Rate (%)	No. of Assemblies	No. Ass. Defective	Defect Rate (%)
Before 1971	125	32	25.6	160	1	0.6
1971	373	115	30.8	52	-	-
1972	574	65	11.3	136	25	18.4
1973	288	270	93.8	111	3	2.7
1974	846	334	39.5	231	31	13.4
1975	925	367	39.7	253	17	6.7
1976	1,459	388	26.6	646	78	12.1
1977	1,662	107	6.4	682	35	5.1
1978	1,769	110	6.2	655	131	20.0
1979	1,764	91	5.2	943	103	10.9
1980	2,768	43	1.6	836	57	6.8
1981	2,039	41	2.0	979	155	15.8
1982	1,849	56	3.0	732	70	9.6
1983	1,957	25	1.3	990	99	10.0
1984	1,935	31	1.6	867	97	11.2
1985	1,515	97	6.4	863	33	3.8
TEMP[b]	142	-	-	196	41	-
TOTAL	21,990	2,172	9.9	9,332	976	10.5

[a] Data from Vermont Yankee (BWR) and Yankee Rowe (PWR) were excluded.
[b] TEMP = Temporary discharge; may be reinserted into the reactor.

Table 2.5.9. DEFECT RATES OF ASSEMBLIES: HIGH ESTIMATE
(Only those SNF Lots that Reported Defects are Included)

	Boiling Water Reactors			Pressurized Water Reactors		
	Pop. W/Def. Assemblies	Defective Assemblies	Defect Rate (%)	Pop. W/Def. Assemblies	Defective Assemblies	Defect Rate (%)
Before 1971	94	32	34.0	60	1	1.7
1971	373	115	30.8	-	-	-
1972	574	65	11.3	83	25	30.1
1973	288	270	93.8	56	3	5.4
1974	809	334	41.3	150	31	20.5
1975	925	367	39.7	114	17	14.9
1976	1,455	388	26.7	341	78	22.9
1977	1,408	107	7.6	110	35	31.8
1978	1,501	110	7.3	315	131	41.6
1979	1,325	91	6.9	405	103	25.4
1980	2,746	43	1.6	112	57	50.9
1981	814	41	5.0	511	155	30.3
1982	593	56	9.4	495	70	14.1
1983	692	25	3.6	479	99	20.7
1984	843	31	3.7	461	97	21.0
1985	479	97	20.3	272	33	12.1
TEMP[a]	142	0	-	144	41	28.5
TOTAL	15,061	2,172	14.4	4,108	976	23.8

Note: Only those discharged SNF batches that also contained known defective assemblies were selected. The data for Vermont Yankee and Yankee Rowe were not used.

[a] TEMP = Temporary discharge; may be reinserted into the reactor.

2.6 SPECIAL LWR FUEL FORMS

Most, but not all, LWR fuel assemblies are currently being stored intact and have relatively standard dimensions. This section provides information on fuels that are different from most LWR fuel assemblies because they have been disassembled or highly degraded in some fashion, their design parameters are radically different in some way, or their fuel rods have been consolidated. Fuel rods and assemblies that have been cut apart and disassembled for testing, evaluation, and research are covered in Section 4.5, Miscellaneous Fuels.

2.6.1 Degraded Fuel from TMI-2

LWR fuel at TMI-2 is highly degraded and will require special handling. It can be assumed that the entire core will be handled as debris and placed in special canisters. The core loading at the time of the accident included 82,023 kg of uranium, of which 2,064.4 kg is 235U. This debris, which includes Zircaloy cladding and other assembly structural materials, is being shipped to DOE's Idaho facility for storage. Three styles of containers are being used - a fuel canister, a knockout canister, and a filter canister (Childress 1986). These are specially designed for different modes of loading, depending on the physical state of the degraded fuel. All three have the same external dimensions - 14-in. diameter, 150-in. overall length, with dished bottoms and flat tops. The internal designs differ as well as the manner in which neutron poisons (for criticality control) are placed in the canisters.

As of December 31, 1986, a total of 13.9 MT of debris had been shipped to Idaho (Ball 1987). This material had a volume of 13.2 m3 and contained about 700,000 Ci of radioactivity. This work will continue until the entire core has been removed.

2.6.2 Nonstandard Fuel Assemblies

Certain fuel assemblies may require special handling because of nonstandard dimensions, unique designs, extremely high burnups, or

differences in the fuel pellets. Westinghouse 17 x 17 assemblies for the South Texas plants are more than 3 feet longer than standard 17 x 17 fuel assemblies. Early fuel assemblies for both PWR's and BWR's were shorter than current fuel assemblies. The fuel assemblies at the Dresden-1 and Humboldt Bay were 6 x 6 arrays; at the Big Rock Point reactor, the original fuel assembly was a 12 x 12 array but has been replaced with both 9 x 9 and 11 x 11 arrays. The assemblies from the Indian Point 1 and Yankee-Rowe reactors are non-square arrays. Several assemblies at various reactors have been exposed to extremely high burnups; annular fuel pellets have been used in others; all manufacturers test a new change on a few test assemblies before implementing the change in all assemblies - the properties of these unique assemblies may need special characterization. Differences in many of these areas are covered in the LWR Assemblies Data Base, but it is important to make a special note of the difficulties they may present. Any of these factors may require specialized equipment for the safe storage, transportation, consolidation, and/or disposal of these assemblies.

2.6.3 Consolidated LWR Fuel

Consolidated LWR spent fuel must also be characterized in terms of physical descriptions (length, width, and weight of cans), quantitative information (how many cans of PWR fuel? BWR fuel? When?), and radiological properties (radioactivity, thermal output, neutron production, photon spectra). Consolidation studies and demonstrations are ongoing at several sites. An early demonstration of wet consolidation was done at West Valley. Northeast Utilities have recently done an underwater test on six assemblies using Combustion Engineering equipment; they achieved a 2-to-1 volume reduction at the end of the test. Northern States Power has just finished an underwater test with 40 assemblies, using Westinghouse equipment. Dry consolidation tests are planned at INEL. Data from these tests will be incorporated in the data base as this information becomes available.

2.6.4 Shippingport LWBR Fuel

The Shippingport PWR was subsequently converted to serve as a test unit for a Light Water Breeder Reactor (LWBR) and was fueled with a mixture of 233U and thorium oxide. This spent fuel has been shipped to DOE's Idaho facility for storage (Schreiber 1987). A total of 65 assemblies were involved with a uranium content of about 700 kg (mostly 233U) and a thorium content of 47 MT as a mixed (Th,U) dioxide.

2.6.5 References for Section 2.6

Ball 1987. Letter from L.J. Ball, EG&G Idaho, to S.N. Storch, Oak Ridge National Laboratory, dated April 9, 1987.

Childress 1986. P.C. Childress, et al., "TMI-2 Defueling Canisters", in High-Level Nuclear Watse Disposal, H.C. Burkholder, ed., 1986.

Schreiber 1987. Letter from J.J. Schreiber, Shippingport Station Decomissioning Project Office, to S.N. Storch, Oak Ridge National Laboratory, dated April 14, 1987.

2.7 SPENT FUEL DISASSEMBLY HARDWARE

2.7.1 Overview

Spent Fuel Disassembly (SFD) hardware is defined as the pieces of a fuel assembly left after the fuel rods have been removed. Generally, SFD hardware for PWR fuel assemblies includes guide tubes; instrument tubes; top and bottom nozzles; grid spacers; hold-down springs; and attachment components, such as nuts and locking caps. Guide tubes and instrument tubes are hollow metal cylinders into which control elements, neutron sources and poisons, and/or instrumentation are inserted. Recently, most tubes have been made of Zircaloy, although early tubes were made of stainless steel. The top and bottom nozzles, which are relatively large solid pieces of stainless steel, direct the flow of water around the fuel rods and provide structural support. Most vendors make nozzles from stainless steel 304, although other similar alloys (SS304L, SS348, CF3M) have been used. Grid spacers, which historically were made of a springy material like Inconel, have recently been made of Zircaloy because of the low neutron absorption cross section. They are attached in some manner to the instrument and/or guide tubes at various locations throughout the assembly to provide both positioning and support for the fuel rods. Together these items make up the skeleton of the fuel assembly. Different vendors use different methods to attach these various components - spot welding, bolting in place, and crimping are all used. The nuts, locking caps, and grid sleeves used in these different methods of attachment are also SFD hardware. The hold-down springs are typically made of a nickel-base alloy and are used to hold the fuel rods down against the bottom nozzle, opposed to the upward flow of water through the assembly.

For BWR fuel assemblies, SFD hardware includes the top and bottom tie plates, compression springs for individual fuel rods, grid spacers, and water rods. In a BWR assembly, structural support is provided by the grid spacers, the top and bottom tie plates, fueled tie rods, and water rods. The tie plates have typically been made of stainless steel

304 and the grid spacers of Zircaloy. The position of the grid spacers is usually determined by welded tabs on the water rod. These tabs are welded onto the water rod so as to not damage the integrity of the fuel-rod cladding. The water rod(s) is similar to guide tubes in that it is a hollow Zircaloy tube. Its purpose is to provide additional water for neutron moderation rather than a location for control elements, etc. BWR assemblies typically use a separate compression spring for each individual fuel rod. These springs, located in the gas plenum region, hold the fuel rod against the bottom tie plate.

Both PWR and BWR assemblies contain some unique pieces of SFD hardware. Structural support for the fuel assemblies for the Palisades and Yankee-Rowe plants is provided by solid bars of Zircaloy rather than guide tubes. The top end fittings of Combustion Engineering's 14 x 14 and 16 x 16 fuel assembly designs are not solid pieces of metal but rather are two flat plates separated by five large metal posts surrounded by Inconel hold-down springs. Westinghouse's new entry into the BWR reload market, the QUAD+, has a unique channel assembly that consists of a water cross welded to a relatively standard outside channel. It is mechanically attached to the bottom nozzle.

With the exception of the QUAD+ channel assembly, BWR fuel channels are not included with SFD hardware in this data base. This is simply a reflection of the fact that reuse of fuel channels is becoming more prevalent. The radioactive properties of these reused channels can be more accurately described by using the format of the LWR NFA Hardware Data Base, where items may be subject to irradiation for more than 1 assembly lifetime.

Nonfueled burnable poison rods in PWR assemblies are also not included as SFD hardware since these rods would probably not be separated from the fueled rods during consolidation but would be included in the consolidated canister with the fuel rods. The extra

handling operations involved with identifying, separating, and disposing of the nonfueled rods safely greatly offset the disadvantage of a slight increase in volume of consolidated fuel-rod canisters.

SFD hardware is not a major technical issue unless fuel assemblies are consolidated prior to emplacement in a repository. Without consolidation, SFD hardware would remain with the assembly where the radioactivity of the hardware is small compared to the radioactivity of the fuel and fission products. If fuel assemblies are consolidated, several concerns arise with regard to the remaining hardware. These include: How much of it is there going to be? and What are the radiological characteristics of it? The issues addressed by this study are the quantitative and radiological characterization of SFD hardware.

2.7.2 Quantitative Characterization

A general description of SFD hardware for the different assembly types is included in the descriptions of the versions in section 2.2 of this report. Specific pieces of SFD hardware are described in as much detail as possible in the LWR Assemblies Data Base. This description includes the name of the specific pieces, the number of pieces per assembly, the weights, and the construction materials. For each assembly, these parts are listed on page 2 of the Physical Description Report. Physical Description Reports for all assemblies for which data are available are given in Appendix 2A, Physical Descriptions of LWR Fuel Assemblies.

Summary quantitative information on how much SFD hardware is associated with PWR fuel assemblies is given in Table 2.7.1, and for BWR assemblies, in Table 2.7.2. The number of assemblies of each type is a question that is addressed by both the LWR Quantities Data Base (Appendix 2D) and, to some extent, the shipping records supplied by the various vendors (Appendix 2G).

2.7.3 Methodology for Radiological Characterization

The disposal of radioactive wastes is primarily regulated by two sections of the Code of Federal Regulations (CFR). The disposal of high-level wastes and spent nuclear fuel in a deep geologic repository is governed by 10 CFR 60, whereas the disposal of low-level wastes in near-surface burial is governed by 10 CFR 61. Although SFD hardware [and Nonfuel Assembly (NFA) hardware] is not specifically included in either set of these regulations, neither is it specifically excluded. 10 CFR 61 puts low-level wastes into four categories -- Class A, Class B, Class C, and Greater than Class C. Inclusion in any one of these categories is based on concentrations of radioactive isotopes in the material to be disposed of. Because SFD hardware does not contain uranium or other actinides, activation products are the sole source of radioactivity. In particular, 10 CFR 61 puts limits on the concentrations of ^{14}C, ^{59}Ni, ^{63}Ni, and ^{94}Nb that are permitted to be present.

The computer code ORIGEN2 has been used to estimate the concentrations of these isotopes that are present in SFD hardware. ORIGEN2 is a revised edition of the widely used code ORIGEN - the Oak Ridge Isotope Generation and Depletion Code (Croff 1980). It is a versatile code for use in the simulation of the conditions occurring in the nuclear fuel cycle and in the calculation of the nuclide compositions and characteristics of the materials contained therein. In particular, ORIGEN2 can model the effects of irradiation on a material.

ORIGEN2 calculates the concentration of all isotopes present in an activated material at a given time on the basis of the initial isotopic composition of the material, the intensity and duration of the neutron flux to which it has been exposed, the cross sections for neutron activation, and the half-lives and decay products of the radioisotopes involved.

Because of the severity of conditions to which materials are exposed in the core of a nuclear power reactor, relatively few materials have been used in the fabrication of fuel assemblies. These materials are alloys of zirconium (Zircaloy-2 and Zircaloy-4), alloys of nickel (Inconel-625, Inconel-718, and Inconel X-750), and stainless steels (stainless steel 304). These materials were chosen for the resistance to corrosion, retention of structural strength after intense irradiation, and low neutron absorption cross sections.

The elemental composition of these materials is determined by standards set by the American Society for Testing and Materials (ASTM). Because they are the precursors to the isotopes that determine low-level waste categories, the initial amounts of nitrogen, nickel, cobalt, and niobium are of particular interest. If these elements are included in the material specifications, it is generally as an upper limit for an impurity. Often they are not included at all, although they are present in trace quantities. For niobium in particular, trace quantities may be sufficient for the irradiated material to exceed the Class C limits. Niobium as an impurity does not affect the physical and chemical characteristics of the material. Thus, as long as the niobium concentrations are below some reasonable level, the actual amount is not of concern to the ingot manufacturer or the fuel assembly vendor. The input to ORIGEN2 of the elemental concentrations of these materials has been based on reasonable upper limits and ASTM specifications. The values used for nitrogen, cobalt, nickel, and niobium are given below.

Material	Nitrogen	Cobalt	Nickel	Niobium
Inconel-625	---	1.00%	57.9%	3.65%
Inconel-718	1300 ppm	4700 ppm	52.0%	5.55%
Inconel X-750	1300 ppm	6490 ppm	72.2%	0.9%
St. Steel 304	1300 ppm	800 ppm	8.92%	100 ppm
Zircaloy-2	80 ppm	10 ppm	500 ppm	120 ppm
Zircaloy-4	80 ppm	10 ppm	20 ppm	120 ppm

The neutron flux intensity and exposure used by ORIGEN2 are a function of the input variables of type of reactor, burnup, initial enrichment, and neutron exposure zone. ORIGEN2 has been used to model the conditions in the core of several different reactor types, including high-temperature gas-cooled reactors, PWRs, and BWRs. Activation of the materials used in SFD hardware has been calculated for a standard burnup and a high burnup. For PWR's, concentrations of isotopes in materials exposed to standard burnup (33,000 MWd/MTIHM) were calculated using the standard PWR model; concentrations in materials exposed to high burnup (60,000 MWd/MTIHM) were calculated using the PWR extended burnup model. An initial enrichment of 3.2% was used for the former and 4.15% for the latter. The concentrations of isotopes in materials exposed to standard burnup (27,500 MWd/MTIHM) in a BWR were calculated using the standard BWR model. A high burnup run for the BWR case was not made; the development of a BWR extended burnup model has just recently been completed. An initial enrichment of 2.75% was used for the BWR case.

Much of the hardware is outside the active core region, and a correction is required for this. The flux is a maximum in the core region and drops off rapidly outside the core. The neutron flux has been modeled in four exposure zones - the top end region, the gas plenum region, the core region, and the bottom end region. The neutronic flux in each region has been calculated (Luksic 1986a, Croff 1978) on the basis of the materials present in each region. These relative neutron flux factors, which were incorporated into the ORIGEN2 runs, are listed below.

		Top End	Gas Plenum	In Core	Bottom End
Luksic	PWR	0.011	0.083	1.000	0.063
	BWR	0.0063	0.065	1.000	0.071
Croff	PWR	0.011	0.042	1.000	0.011
	BWR	0.131	0.500	1.000	0.131

The later values, calculated by Luksic, were used for the present calculations.

Neutronic flux calculations were based on an average PWR and an average BWR assembly. Currently, efforts are under way to perform the same modeling for different array designs to determine if these relative flux factors and effective cross sections are assembly model dependent. Differences between the array designs of different manufacturers are expected to be minor in terms of effect on neutron flux.

To simplify the computational algorithm and conserve storage space, ORIGEN2 uses a single energy-averaged neutron absorption cross section for each isotope rather than a set of individual neutron energy dependent cross sections. New effective cross sections were calculated in the different zones for the precursors of the isotopes of particular interest because the neutron energy distribution changes from the in-core neutron zone to the gas plenum, top end and bottom end zones. These isotopes were 14C, 50Ni, 63Ni, and 94Nb. These isotopes are primarily the result of the following reactions:

$^{14}N(n,p)^{14}C$
$^{59}Co(n,\beta)^{60}Co$; $^{59}Co(n,\beta)^{60}Co^*$
$^{58}Ni(n,\beta)^{59}Ni$
$^{62}Ni(n,\beta)^{63}Ni$
$^{93}Nb(n,\beta)^{94}Nb$; $^{93}Nb(n,\beta)^{94}Nb^*$

The neutron spectra and the individual neutron energy dependent cross sections were energy averaged to obtain spectral flux factors for each neutron zone. These spectral flux factors are multiplied by the cross section supplied by the appropriate ORIGEN2 library to give the effective energy averaged cross section for these reactions in a particular zone. The neutron flux factors and the spectral flux factors obtained by Croff (Croff 1978) differ from the results obtained by Luksic (Luksic 1986a). Experimental studies on actual samples of SFD hardware are ongoing and are expected to reconcile these differences. The spectral flux factors obtained by Luksic (Luksic 1986a) and incorporated in the ORIGEN2 runs are given below.

Reactor Type	Region	Nitrogen	Cobalt	Nickel	Niobium
PWR	Top End	4.3	3.1	4.6	1.6
PWR	Gas Plenum	6.3	4.4	6.7	2.1
PWR	In Core	1.0	1.0	1.0	1.0
PWR	Bottom End	4.3	3.2	4.6	1.7
BWR	Top End	1.8	2.1	1.8	1.3
BWR	Gas Plenum	3.6	3.2	3.7	2.1
BWR	In Core	1.0	1.0	1.0	1.0
BWR	Bottom End	2.3	2.1	2.3	1.5

Data on radioactive half-lives, decay products, relative natural abundances of isotopes, and effective cross sections for other isotopes were taken directly from the appropriate ORIGEN2 libraries. A portion of these data is reproduced in Appendix 1B, ORIGEN2 Data Libraries.

2.7.4 Results of Radiological Characterization

Thirty eight ORIGEN2 runs have been made with different materials being irradiated in different exposure zones of different reactor types. Cases that have been run are listed in Table 2.7.3. The output of these ORIGEN2 runs consists of the concentrations of an isotope in one kilogram of material. These concentrations are expressed in grams, curies, and watts. The overall photon spectra from the irradiated material are also an output. These results have been downloaded from the ORIGEN2 output to data files in the LWR Assemblies Data Base. Isotopic concentrations have been subjected to a set of cutoffs to limit the space required to store these induced radioactivity data. These cutoffs indicate that the concentration of an isotope at a given time after discharge is significant if the radioactivity (in curies) or the thermal power (in watts) resulting from the isotope is greater than 1 part in 100,000 of the total radioactivity or thermal power of all isotopes in the material at that time. Even if the isotope does not meet these criteria at time A, it is included in the induced radioactivity data in

the data base if it meets these criteria at some later time B. Thus, although 94Nb is not generally a significant contributor to the total radioactivity after short decay times, it is included in the data base for all times because of its significance at longer decay times. If either the radioactivity or the thermal output of an isotope causes it to be significant at a particular time, the curies, watts, and grams of the isotope are saved in a data file. The photon spectra from the ORIGEN2 runs are downloaded to another file without any cutoffs being imposed.

Two Radiological Description Reports are available from the data base. The first is the Material Report. This report gives the radiological description of a particular material that has been exposed to a specified (Standard or High) burnup in a given zone. The user must also specify the time after discharge for which he/she wants this radiological characterization. An example of the Material Report is given in Tables 2.7.4, 2.7.5, and 2.7.6 for Zircaloy-2, stainless steel 304 and Inconel-718, respectively. These reports are for materials exposed to standard burnup in the core of a BWR, 15 years after discharge.

Downloaded ORIGEN2 output is combined with the physical information given on page 2 of the Physical Description Report to produce a second Radiological Description Report, the Assembly Report. This report gives the radiological characteristics of the SFD hardware associated with a particular assembly type. This report is available for a particular piece of SFD hardware, for all parts within a specified zone, or for all the SFD hardware associated with the particular assembly. An example of the Radiological Description Report for a Babcock & Wilcox 15 x 15 Mark BZ fuel assembly is given in Table 2.7.7.

These results generally show that stainless steel and Zircaloy may be acceptable for low-level waste disposal, depending on the actual niobium content of the initial material and the regulatory status of Greater than Class C low-level wastes. Since 94Nb is the primary isotope that routinely exceeds the Class C limits in these materials,

niobium concentrations in the materials that are far below the ASTM limits may allow for disposal of SFD hardware made from these materials as Class C low-level wastes. Even if the niobium content is near the maximum limits, some SFD hardware may be acceptable for near surface burial if the NRC should approve a specific proposal for disposal of these Greater than Class C wastes. A particular problem with the 94Nb concentration lies in the difficulty of accurately observing and quantifying its decay. The primary photon associated with its decay has an energy of 871 keV. The photopeak associated with this relatively small decay is completely obscured by Compton backscattering from the high energy 60Co gamma rays. If SFD hardware made of stainless steel and Zircaloy is disposed of as low-level wastes, the heat output of the 60Co may create stiff surcharges at low-level waste disposal areas. As taken from Tables 2.7.4 and 2.7.5, the concentration of 60Co in Zircaloy-2 and stainless steel 304 is 880 and 88,000 Ci/m^3, respectively. The Class A limit for 60Co is 700 Ci/m^3. No Class B or Class C limit is specified for 60Co. The initial concentrations of nickel and niobium in Inconel alloys preclude the possibility that SFD hardware made from Inconel alloys will be acceptable for near-surface burial.

Experimental verification of these results is being pursued by a series of in-depth experiments currently being conducted by Pacific Northwest Laboratories (PNL) on pieces of SFD hardware with known histories. These experiments will determine the initial concentration of nitrogen, cobalt, nickel, and niobium, as well as the concentrations of the radionuclides present after irradiation. These results will provide more accurate input to ORIGEN2 and will allow verification of the flux factors and effective cross sections used in the radiological characterization of SFD hardware.

2.7.5 References for Section 2.7

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Table 2.7.1 Spent Fuel Disassembly Hardware for Different PWR Assembly Types. Listing by Array Design.

Babcock & Wilcox 15 x 15 Array Design	
Babcock & Wilcox 15 x 15 Mark B	35.6 kg
Babcock & Wilcox 15 x 15 Mark BZ	35.6 kg
Babcock & Wilcox 17 X 17 Array Design	
Babcock & Wilcox 17 x 17 Mark C	42.3 kg
Combustion Engineering 14 x 14 Array Design	
Combustion Engineering 14 x 14 Standard	29.8 kg
Combustion Engineering 14 x 14 Ft. Calhoun	27.6 kg
Advanced Nuclear Fuels 14 x 14 CE	33.3 kg
Westinghouse Electric 14 x 14 Model C	34.1 kg
Westinghouse Electric 14 x 14 Ft. Calhoun	N/A
Combustion Engineering 15 x 15 Array Design	
Combustion Engineering 15 x 15 Palisades	39.7 kg
Exxon/ANF 15 x 15 CE	31.4 kg
Combustion Engineering 16 x 16 Array Design	
Combustion Engineering 16 x 16 St. Lucie 2	34.8 kg
Combustion Engineering 16 x 16 Ark. Nucl. 2	40.1 kg
Combustion Engineering 16 x 16 San Onofre	42.6 kg
Combustion Engineering 16 x 16 System 80	44.0 kg
Westinghouse 14 x 14 Array Design	
Westinghouse 14 x 14 Std/SC	28.9 kg
Westinghouse 14 x 14 Std/ZCA	32.0 kg
Westinghouse 14 x 14 Std/ZCB	31.8 kg
Westinghouse 14 x 14 OFA	32.1 kg
Babcock & Wilcox 14 x 14 Ginna	N/A
Exxon/ANF 14 x 14 WE	28.4 kg
Exxon/ANF 14 x 14 Top Rod	24.6 kg
Westinghouse 15 x 15 Array Design	
Westinghouse 15 x 15 Std/SC	33.4 kg
Westinghouse 15 x 15 Std/ZC	35.8 kg
Westinghouse 15 x 15 OFA	32.6 kg
Babcock & Wilcox 15 x 15 St. Steel	28.4 kg
Exxon/ANF 15 x 15 WE	27.3 kg
Westinghouse 15 x 16 Array Design	
Westinghouse 15 x 16	N/A
Combustion Engineering 15 x 16 Yankee-Rowe	35.0 kg
Exxon/ANF 15 x 16 WE	30.4 kg
Westinghouse 17 x 17 Array Design	
Westinghouse 17 x 17 Std	29.6 kg
Westinghouse 17 x 17 OFA	32.3 kg
Westinghouse 17 x 17 Vantage 5	N/A
Westinghouse 17 x 17 X-long	N/A
Babcock & Wilcox 17 x 17 Mark BW	N/A
Exxon/ANF 17 x 17 WE	34.6 kg

Table 2.7.2 Spent Fuel Disassembly Hardware for Different BWR Assembly Types. Listing by Array Design.

Advanced Nuclear Fuels 9 x 9 Array Design	
Exxon/ANF 9 x 9 JP-3	9.3 kg
Exxon/ANF 9 x 9 JP-4,5	9.3 kg
Allis Chalmers 10 x 10 Array Design	
Allis Chalmers 10 x 10	N/A
Exxon/ANF 10 x 10	16.8 kg
General Electric 6 x 6 Dresden-1 Array Design	
General Electric 6 x 6	N/A
Exxon/ANF 6 x 6 GE	9.4 kg
United Nuclear 6 x 6	N/A
General Electric 6 x 6 Hum. Bay Array Design	
General Electric 6 x 6	N/A
Exxon/ANF 6 x 6	N/A
General Electric 7 x 7 Array Design	
General Electric 7 x 7 /2,3:V.1	8.0 kg
General Electric 7 x 7 /2,3:V.2	8.0 kg
General Electric 7 x 7 /4,5	8.0 kg
Exxon/ANF 7 x 7 GE	13.6 kg
General Electric 8 x 8 Array Design	
General Electric 8 x 8 /2,3	N/A
General Electric 8 x 8 /4-6:V.1	13.1 kg
General Electric 8 x 8 /4-6:V.2	14.8 kg
Exxon/ANF 8 x 8 JP-3	8.1 kg
Exxon/ANF 8 x 8 JP-4,5	9.0 kg
General Electric 9 x 9 Array Design	
General Electric 9 x 9	N/A
Exxon/ANF 9 x 9 Big Rock Point	N/A
General Electric 11 x 11 Array Design	
General Electric 11 x 11	N/A
Exxon/ANF 11 x 11	10.5 kg
Westinghouse 8 x 8 Array Design	
Westinghouse 8 x 8 QUAD+	N/A

Table 2.7.3 ORIGEN2 Runs Made for SFD Hardware Characterization.

Reactor Type	Material	Neutron Exposure Zone	Burnup
PWR	Zircaloy-4	Top End	Std.,High
PWR	Zircaloy-4	Gas Plenum	Std.,High
PWR	Zircaloy-4	Incore	Std.,High
PWR	Inconel 625	Incore	Std.,High
PWR	Inconel 718	Top End	Std.,High
PWR	Inconel 718	Gas Plenum	Std.,High
PWR	Inconel 718	Incore	Std.,High
PWR	Inconel 718	Bottom End	Std.,High
PWR	Inconel X-750	Top End	Std.,High
PWR	Inconel X-750	Incore	Std.,High
PWR	St. Steel 304	Top End	Std.,High
PWR	St. Steel 304	Incore	Std.,High
PWR	St. Steel 304	Bottom End	Std.,High
BWR	Zircaloy-2	Top End	Std.
BWR	Zircaloy-2	Incore	Std.
BWR	Zircaloy-2	Bottom End	Std.
BWR	Zircaloy-4	Top End	Std.
BWR	Zircaloy-4	Incore	Std.
BWR	Inconel 718	Top End	Std.
BWR	Inconel 718	Incore	Std.
BWR	Inconel X-750	Top End	Std.
BWR	Inconel X-750	Incore	Std.
BWR	St. Steel 304	Top End	Std.
BWR	St. Steel 304	Incore	Std.
BWR	St. Steel 304	Bottom End	Std.

Table 2.7.4 Material Radiological Description Report for Zircaloy-2.

Radiological Report (Curies)
for
Zircaloy-2

Boiling Water Reactor
In Core Zone
Standard (27,500) 15 Years After Discharge

ALL VALUES ARE PER KILOGRAM OF IRRADIATED MATERIAL

Isotope	Curies
C 14	1.500E-03
Fe 55	1.256E-02
Co 60	1.357E-01
Ni 59	2.279E-04
Ni 63	3.123E-02
Sr 90	3.184E-06
Y 90	3.185E-06
Zr 93	5.915E-04
Nb 93m	3.242E-04
Nb 94	1.768E-04
Tc 99	1.702E-08
Sn121m	2.010E-03
Sb125	1.378E-01
Te125m	3.361E-02

Table 2.7.5 Material Radiological Description Report for St. Steel 304.

Radiological Report (Curies)
for
Stainless Steel 304

Boiling Water Reactor
In Core Zone
Standard (27,500) 15 Years After Discharge

ALL VALUES ARE PER KILOGRAM OF IRRADIATED MATERIAL

Isotope	Curies
Be 10	2.304E-09
C 14	2.436E-02
Cl 36	9.060E-09
Fe 55	5.699E+00
Co 60	1.094E+01
Ni 59	4.065E-02
Ni 63	5.565E+00
Zr 93	8.639E-10
Nb 93m	4.735E-10
Nb 94	2.122E-04

Table 2.7.6 Material Radiological Description Report for Inconel-718.

Radiological Report (Curies)
for
Inconel-718

Boiling Water Reactor
In Core Zone
Standard (27,500) 15 Years After Discharge

ALL VALUES ARE PER KILOGRAM OF IRRADIATED MATERIAL

Isotope	Curies
Be 10	1.152E-09
C 14	2.436E-02
Fe 55	1.712E+00
Co 60	6.394E+01
Ni 59	2.368E-01
Ni 63	3.242E+01
Zr 93	4.197E-07
Nb 93m	2.300E-07
Nb 94	9.808E-02
Mo 93	1.895E-03
Tc 99	3.586E-04

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly.

Spent Fuel Disassembly Hardware Page: 1
for a
Babcock & Wilcox 15 X 15 Mark BZ

Top Zone Standard Burnup 10 years after discharge Units are Curies

Component Name	Inconel-718	St.Steel 304	Isotope Total
HOLDDOWN SPRING	(1.8000 kg)		
C 14	2.146E-03		2.146E-03
Fe 55	1.501E-01		1.501E-01
Co 60	8.573E+00		8.573E+00
Ni 59	2.882E-02		2.882E-02
Ni 63	3.271E+00		3.271E+00
Zr 93	1.216E-08		1.216E-08
Nb 93m	5.125E-09		5.125E-09
Nb 94	3.289E-03		3.289E-03
Mo 93	3.825E-05		3.825E-05
Tc 99	1.780E-06		1.780E-06
	1.202E+01		
SPRING RETAINER	(0.9100 kg)		
Be 10		3.180E-11	3.180E-11
C 14		1.085E-03	1.085E-03
Mn 54		1.681E-04	1.681E-04
Fe 55		2.423E-01	2.423E-01
Co 60		7.378E-01	7.378E-01
Ni 59		2.499E-03	2.499E-03
Ni 63		2.836E-01	2.836E-01
Zr 93		1.059E-11	1.059E-11
Nb 93m		4.463E-12	4.463E-12
Nb 94		2.996E-06	2.996E-06
		1.267E+00	
TOP NOZZLE	(7.4800 kg)		
Be 10		2.614E-10	2.614E-10
C 14		8.916E-03	8.916E-03
Mn 54		1.382E-03	1.382E-03
Fe 55		1.992E+00	1.992E+00
Co 60		6.065E+00	6.065E+00
Ni 59		2.054E-02	2.054E-02
Ni 63		2.332E+00	2.332E+00
Zr 93		8.707E-11	8.707E-11
Nb 93m		3.668E-11	3.668E-11
Nb 94		2.462E-05	2.462E-05
		1.042E+01	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware for a Babcock & Wilcox 15 X 15 Mark BZ

Page: 2

Top Zone | Standard Burnup | 10 years after discharge | Units are Curies

TOP NOZZLE	Inconel-718	St.Steel 304	Isotope Total
UPPER NUT	(0.5100 kg)		
Be 10		1.782E-11	1.782E-11
C 14		6.079E-04	6.079E-04
Mn 54		9.420E-05	9.420E-05
Fe 55		1.358E-01	1.358E-01
Co 60		4.135E-01	4.135E-01
Ni 59		1.400E-03	1.400E-03
Ni 63		1.590E-01	1.590E-01
Zr 93		5.936E-12	5.936E-12
Nb 93m		2.501E-12	2.501E-12
Nb 94		1.679E-06	1.679E-06
		7.104E-01	
UPPER END PLUG	(0.0600 kg)		
Be 10		2.096E-12	2.096E-12
C 14		7.152E-05	7.152E-05
Mn 54		1.108E-05	1.108E-05
Fe 55		1.598E-02	1.598E-02
Co 60		4.865E-02	4.865E-02
Ni 59		1.648E-04	1.648E-04
Ni 63		1.870E-02	1.870E-02
Zr 93		6.984E-13	6.984E-13
Nb 93m		2.942E-13	2.942E-13
Nb 94		1.975E-07	1.975E-07
		8.357E-02	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 3
for a
Babcock & Wilcox 15 X 15 Mark BZ

Top Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 10.760000 kg. Total zone volume: 0.0013370 cu m.

Top Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
Be 10	3.131E-10	2.910E-11	2.342E-07		
C 14	1.283E-02	1.192E-03	9.596E+00	80.00	1.200E-01
Mn 54	1.655E-03	1.538E-04	1.238E+00		
Fe 55	2.536E+00	2.357E-01	1.897E+03		
Co 60	1.584E+01	1.472E+00	1.185E+04		
Ni 59	5.342E-02	4.965E-03	3.996E+01	220.00	1.816E-01
Ni 63	6.065E+00	5.637E-01	4.536E+03	7000.00	6.480E-01
Zr 93	1.227E-08	1.140E-09	9.177E-06		
Nb 93m	5.169E-09	4.804E-10	3.866E-06		
Nb 94	3.319E-03	3.085E-04	2.482E+00	0.20	1.241E+01
Mo 93	3.825E-05	3.555E-06	2.861E-02		
Tc 99	1.780E-06	1.654E-07	1.331E-03		

Top Zone Totals by material:

Inconel-718	St.Steel 304
1.202E+01	1.248E+01

Top Zone Totals by material in kilograms:

Inconel-718	St.Steel 304
1.8000	8.9600

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 4
for a
Babcock & Wilcox 15 X 15 Mark BZ

Gas Plenum Zone Standard Burnup 10 years after discharge Units are Curies

Component Name	Inconel-718	Isotope Total
SPACER-PLENUM	(1.0400 kg)	
C 14	1.369E-02	1.369E-02
Fe 55	6.541E-01	6.541E-01
Co 60	5.195E+01	5.195E+01
Ni 59	1.805E-01	1.805E-01
Ni 63	2.056E+01	2.056E+01
Zr 93	5.296E-08	5.296E-08
Nb 93m	2.231E-08	2.231E-08
Nb 94	1.873E-02	1.873E-02
Mo 93	1.667E-04	1.667E-04
Tc 99	7.698E-06	7.698E-06
	7.338E+01	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 5
for a
Babcock & Wilcox 15 X 15 Mark BZ

Gas Plenum Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 1.040000 kg. Total zone volume: 0.0001270 cu m.

Gas Plenum Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
C 14	1.369E-02	1.316E-02	1.078E+02	80.00	1.348E+00
Fe 55	6.541E-01	6.289E-01	5.151E+03		
Co 60	5.195E+01	4.995E+01	4.091E+05		
Ni 59	1.805E-01	1.736E-01	1.421E+03	220.00	6.461E+00
Ni 63	2.056E+01	1.977E+01	1.619E+05	7000.00	2.313E+01
Zr 93	5.296E-08	5.092E-08	4.171E-04		
Nb 93m	2.231E-08	2.145E-08	1.757E-04		
Nb 94	1.873E-02	1.801E-02	1.475E+02	0.20	7.375E+02
Mo 93	1.667E-04	1.603E-04	1.313E+00		
Tc 99	7.698E-06	7.402E-06	6.062E-02		

Gas Plenum Zone Totals by material:

Inconel-718
7.338E+01

Gas Plenum Zone Totals by material in kilograms:

Inconel-718
1.0400

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 6
for a
Babcock & Wilcox 15 X 15 Mark BZ

In Core Zone Standard Burnup 10 years after discharge Units are Curies

Component Name	Zircaloy-4	Isotope Total
GRID SUPPORTS	(0.6400 kg)	
C 14	9.901E-04	9.901E-04
Mn 54	3.475E-05	3.475E-05
Fe 55	5.004E-02	5.004E-02
Co 60	1.784E-01	1.784E-01
Ni 59	6.733E-06	6.733E-06
Ni 63	8.819E-04	8.819E-04
Sr 90	3.114E-06	3.114E-06
Y 90	3.115E-06	3.115E-06
Zr 93	3.558E-04	3.558E-04
Nb 93m	1.500E-04	1.500E-04
Nb 94	1.357E-04	1.357E-04
Sn119m	4.768E-04	4.768E-04
Sn121m	1.410E-03	1.410E-03
Sb125	3.396E-01	3.396E-01
Te125m	8.288E-02	8.288E-02
	6.554E-01	
GUIDE TUBES	(8.0000 kg)	
C 14	1.238E-02	1.238E-02
Mn 54	4.344E-04	4.344E-04
Fe 55	6.254E-01	6.254E-01
Co 60	2.230E+00	2.230E+00
Ni 59	8.416E-05	8.416E-05
Ni 63	1.102E-02	1.102E-02
Sr 90	3.893E-05	3.893E-05
Y 90	3.894E-05	3.894E-05
Zr 93	4.448E-03	4.448E-03
Nb 93m	1.874E-03	1.874E-03
Nb 94	1.697E-03	1.697E-03
Sn119m	5.960E-03	5.960E-03
Sn121m	1.762E-02	1.762E-02
Sb125	4.245E+00	4.245E+00
Te125m	1.036E+00	1.036E+00
	8.192E+00	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 7
for a
Babcock & Wilcox 15 X 15 Mark BZ

In Core Zone Standard Burnup 10 years after discharge Units are Curies

Component Name	Zircaloy-4	Isotope Total
INSTRUMENT TUBE	(0.6400 kg)	
C 14	9.901E-04	9.901E-04
Mn 54	3.475E-05	3.475E-05
Fe 55	5.004E-02	5.004E-02
Co 60	1.784E-01	1.784E-01
Ni 59	6.733E-06	6.733E-06
Ni 63	8.819E-04	8.819E-04
Sr 90	3.114E-06	3.114E-06
Y 90	3.115E-06	3.115E-06
Zr 93	3.558E-04	3.558E-04
Nb 93m	1.500E-04	1.500E-04
Nb 94	1.357E-04	1.357E-04
Sn119m	4.768E-04	4.768E-04
Sn121m	1.410E-03	1.410E-03
Sb125	3.396E-01	3.396E-01
Te125m	8.288E-02	8.288E-02
	6.554E-01	
SPACER-INCORE	(4.9000 kg)	
C 14	7.580E-03	7.580E-03
Mn 54	2.661E-04	2.661E-04
Fe 55	3.831E-01	3.831E-01
Co 60	1.366E+00	1.366E+00
Ni 59	5.155E-05	5.155E-05
Ni 63	6.752E-03	6.752E-03
Sr 90	2.384E-05	2.384E-05
Y 90	2.385E-05	2.385E-05
Zr 93	2.724E-03	2.724E-03
Nb 93m	1.148E-03	1.148E-03
Nb 94	1.039E-03	1.039E-03
Sn119m	3.650E-03	3.650E-03
Sn121m	1.079E-02	1.079E-02
Sb125	2.600E+00	2.600E+00
Te125m	6.345E-01	6.345E-01
	5.018E+00	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 8
for a
Babcock & Wilcox 15 X 15 Mark BZ

In Core Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 14.180000 kg. Total zone volume: 0.0021616 cu m.

In Core Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
C 14	2.194E-02	1.547E-03	1.015E+01	80.00	1.269E-01
Mn 54	7.699E-04	5.429E-05	3.562E-01		
Fe 55	1.108E+00	7.814E-02	5.126E+02		
Co 60	3.952E+00	2.787E-01	1.828E+03		
Ni 59	1.492E-04	1.052E-05	6.902E-02	220.00	3.137E-04
Ni 63	1.953E-02	1.377E-03	9.035E+00	7000.00	1.291E-03
Sr 90	6.899E-05	4.865E-06	3.192E-02		
Y 90	6.901E-05	4.867E-06	3.193E-02		
Zr 93	7.884E-03	5.560E-04	3.647E+00		
Nb 93m	3.322E-03	2.343E-04	1.537E+00		
Nb 94	3.008E-03	2.121E-04	1.392E+00	0.20	6.958E+00
Sn119m	1.056E-02	7.447E-04	4.885E+00		
Sn121m	3.123E-02	2.202E-03	1.445E+01		
Sb125	7.525E+00	5.307E-01	3.481E+03		
Te125m	1.836E+00	1.295E-01	8.494E+02		

In Core Zone Totals by material:

Zircaloy-4
1.452E+01

In Core Zone Totals by material in kilograms:

Zircaloy-4
14.1800

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware for a Babcock & Wilcox 15 X 15 Mark BZ

Page: 9

Bottom Zone | Standard Burnup | 10 years after discharge | Units are Curies

Component Name	Inconel-718	St.Steel 304	Isotope Total
SPACER-BOTTOM	(1.3000 kg)		
C 14	8.869E-03		8.869E-03
Fe 55	6.207E-01		6.207E-01
Co 60	3.622E+01		3.622E+01
Ni 59	1.181E-01		1.181E-01
Ni 63	1.345E+01		1.345E+01
Zr 93	5.027E-08		5.027E-08
Nb 93m	2.118E-08		2.118E-08
Nb 94	1.440E-02		1.440E-02
Mo 93	1.582E-04		1.582E-04
Tc 99	7.320E-06		7.320E-06
	5.043E+01		
BOTTOM NOZZLE	(8.1600 kg)		
Be 10		1.633E-09	1.633E-09
C 14		5.567E-02	5.567E-02
Mn 54		8.625E-03	8.625E-03
Fe 55		1.244E+01	1.244E+01
Co 60		3.871E+01	3.871E+01
Ni 59		1.272E-01	1.272E-01
Ni 63		1.449E+01	1.449E+01
Zr 93		5.434E-10	5.434E-10
Nb 93m		2.290E-10	2.290E-10
Nb 94		1.630E-04	1.630E-04
		6.583E+01	
LOWER NUT	(0.1500 kg)		
Be 10		3.001E-11	3.001E-11
C 14		1.023E-03	1.023E-03
Mn 54		1.585E-04	1.585E-04
Fe 55		2.287E-01	2.287E-01
Co 60		7.116E-01	7.116E-01
Ni 59		2.338E-03	2.338E-03
Ni 63		2.664E-01	2.664E-01
Zr 93		9.988E-12	9.988E-12
Nb 93m		4.209E-12	4.209E-12
Nb 94		2.995E-06	2.995E-06
		1.210E+00	

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 10
for a
Babcock & Wilcox 15 X 15 Mark BZ

Bottom Zone Standard Burnup 10 years after discharge Units are Curies

Total zone weight: 9.610000 kg. Total zone volume: 0.0011949 cu m.

Bottom Zone Totals by isotope:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
Be 10	1.663E-09	1.730E-10	1.392E-06		
C 14	6.556E-02	6.822E-03	5.487E+01	80.00	6.858E-01
Mn 54	8.784E-03	9.140E-04	7.351E+00		
Fe 55	1.329E+01	1.383E+00	1.112E+04		
Co 60	7.564E+01	7.871E+00	6.330E+04		
Ni 59	2.476E-01	2.576E-02	2.072E+02	220.00	9.419E-01
Ni 63	2.821E+01	2.935E+00	2.361E+04	7000.00	3.373E+00
Zr 93	5.082E-08	5.288E-09	4.253E-05		
Nb 93m	2.141E-08	2.228E-09	1.792E-05		
Nb 94	1.456E-02	1.515E-03	1.219E+01	0.20	6.093E+01
Mo 93	1.582E-04	1.646E-05	1.324E-01		
Tc 99	7.320E-06	7.617E-07	6.126E-03		

Bottom Zone Totals by material:

Inconel-718	St.Steel 304
5.043E+01	6.704E+01

Bottom Zone Totals by material in kilograms:

Inconel-718	St.Steel 304
1.3000	8.3100

Table 2.7.7 Assembly Radiological Description Report for a Babcock & Wilcox 15 X 15 Mark BZ Fuel Assembly. (cont.)

Spent Fuel Disassembly Hardware Page: 11
for a
Babcock & Wilcox 15 X 15 Mark BZ

All Zones Standard Burnup 10 years after discharge Units are Curies

Total zones weight: 35.590000 kg. Total zones volume: 0.0048204 cu m.

Totals by isotope for all zones:

Isotope	Curies	Curies/kg	Curies/cu m	Class C Limit	Ratio
Be 10	1.976E-09	5.552E-11	4.099E-07		
C 14	1.140E-01	3.203E-03	2.365E+01	80.00	2.956E-01
Mn 54	1.121E-02	3.150E-04	2.326E+00		
Fe 55	1.759E+01	4.942E-01	3.649E+03		
Co 60	1.474E+02	4.142E+00	3.058E+04		
Ni 59	4.816E-01	1.353E-02	9.991E+01	220.00	4.541E-01
Ni 63	5.485E+01	1.541E+00	1.138E+04	7000.00	1.626E+00
Sr 90	6.899E-05	1.938E-06	1.431E-02		
Y 90	6.901E-05	1.939E-06	1.432E-02		
Zr 93	7.884E-03	2.215E-04	1.636E+00		
Nb 93m	3.322E-03	9.334E-05	6.891E-01		
Nb 94	3.962E-02	1.113E-03	8.219E+00	0.20	4.110E+01
Mo 93	3.631E-04	1.020E-05	7.532E-02		
Tc 99	1.680E-05	4.720E-07	3.485E-03		
Sn119m	1.056E-02	2.967E-04	2.191E+00		
Sn121m	3.123E-02	8.775E-04	6.479E+00		
Sb125	7.525E+00	2.114E-01	1.561E+03		
Te125m	1.836E+00	5.159E-02	3.809E+02		

Assembly totals by material:

Zircaloy-4	Inconel-718	St.Steel 304
1.452E+01	1.358E+02	7.952E+01

Assembly grand total in Curies	Curies/kg	Curies/cu m
2.298E+02	6.458E+00	4.768E+04

*Some radiological data based on a slightly different elemental composition than actual material.

2.8 NONFUEL ASSEMBLY HARDWARE

2.8.1 Overview

This category deals with reactor hardware that is not necessarily tied in with fuel assemblies on a one-to-one basis, as SFD hardware is. Nonfuel Assembly (NFA) hardware is generally used within or between assemblies but is not permanently attached to an assembly. These NFA hardware components are usually retired from service on a schedule that is different from that of the fuel assemblies. Physical and radiological characterization of these components is being done as part of the LWR NFA Hardware Data Base. Appendix 2E contains physical description reports for pieces of NFA hardware as described by the vendors. Table 2.8.1 gives an example of a physical description report from the LWR NFA Hardware Data Base. Appendix 2F is the user's guide to the LWR NFA Hardware Data Base. Because NFA hardware may remain in the reactor for many cycles, radiological characterization of the NFA hardware will require somewhat different treatment than that of SFD hardware. The methodology of using four neutron exposure zones (see Section 2.7.3) is continued with NFA hardware; likewise, the structure of the data file describing the materials used, weight, etc., remains essentially unchanged. Generating a different set of ORIGEN2 outputs is required.

The major contributors to this waste category are BWR fuel channels, BWR control blades, PWR control rods, and PWR burnable poison assemblies. Other contributors to NFA hardware are neutron sources, in-core instrumentation, and guide-tube thimbles or orifice rods.

Historically, BWR fuel channels have been discharged on a one-to-one basis with the BWR assemblies. They are normally attached to the assemblies and could be considered as SFD hardware except that actual fuel disassembly is not required.

The BWR control blades (cruciforms) and PWR control rods also contribute significantly to the volume of NFA hardware waste. The BWR cruciforms have an estimated lifetime of 3 to 25 years, depending on their service mode. The PWR control rods have an estimated lifetime of

up to the full reactor lifetime because they are normally in a withdrawn position. In the past, they have been replaced halfway through reactor lifetime.

Burnable poison assemblies, especially from B&W reactors, are the third large contributor to the NFA hardware waste category. They are typically used for only one cycle and then changed out. Since they are used in the core of the reactor, their levels of neutron activation are high.

Guide-tube thimbles or orifice rods should be a minor contributor to NFA hardware quantities. Typically stainless steel tubes designed to inhibit the flow of water through otherwise empty guide tubes, these rods are generally outside the core zone. They are hollow and short; so they are relatively lightweight. They are not routinely replaced except when damaged; so the number being changed out is small.

In-core power instrumentation should also be a minor contributor to NFA hardware quantities. The detectors/emitters on these pieces of hardware are in the reactor core and may be highly activated, but the majority of the mass of these pieces is outside the in-core zone and may be far less activated.

Neutron sources may also pose a problem from the standpoint of TRU wastes. Primary neutron sources are typically polonium- or plutonium-beryllium alloys, or californium. Secondary neutron sources are typically antimony-beryllium alloys. Quantities of these wastes should be small.

2.8.2 Babcock and Wilcox Non-Fuel Assembly Hardware

2.8.2.1 Control Rod Assemblies

Control rod assemblies 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 control rod assembly. The spider assembly consists of seven 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 18.38 pounds. The nuts, upper end plugs, and spring spacers weigh 7.5 pounds. The Ag-In-Cd alloy weighs 95 pounds, and the total weight of the assembly is 130 pounds.

Axial power shaping assemblies have a similar spider assembly that weighs 7.8 pounds. The axial power shaping rods 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 weigh 24.7 pounds. The Ag-In-Cd absorber must be shorter, is assumed to be at the bottom of the rod, and weighs 23.4 pounds. The overall assembly weight is 57 pounds.

Gray axial power shaping rod assemblies also have a similar spider assembly that weighs 7.5 pounds. The gray axial power shaping rods are 155.56 inches long and 0.440 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 63.25 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.

2.8.2.2 Neutron Sources

Primary neutron source clusters

The description is incomplete. Weights are given for three of the ten components, and 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. It is not known if it is part of a control assembly or an orifice rod assembly.

Regenerative neutron source clusters

These clusters 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. 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 weighs 26.4 pounds. The source composition is B&W proprietary. The total weight of the cluster is 46.3 pounds.

2.8.2.3 Burnable Poison Assemblies

The spider for these assemblies is made of CF3M, 304, and 316 stainless steel and weighs 7.8 pounds. The 16 burnable poison 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 burnable poison is a B&W proprietary mixture of Al_2O_3 and B_4C. The poison weighs 20.8 pounds, the Zircaloy 25.2 pounds, and the springs 2.1 pounds. The overall weight of the assembly is 57 pounds.

2.8.2.4 Orifice Rod Assembly

The spider assembly 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 is 0.3 pound heavier than the sum of the other components.

2.8.2.5 In-Core Instrumentation

B&W provides in-core instrumentation but none was described in the information supplied.

2.8.3 Combustion Engineering Non-Fuel Assembly Hardware

2.8.3.1 Control Element Assemblies

Control element assemblies (CEA's) provide a means of controlling core reactivity. CE designed control element assemblies consist of

tubes or "fingers" filled with neutron absorbing materials. The geometry of the fingers allows them to fit inside the guide tubes of fuel assemblies. During normal operation, the fingers are fully withdrawn from the fueled zone into the upper guide structure of the reactor vessel. In this position the lower tips of the CEA fingers are approximately two inches above the fueled zone. It is sometimes necessary to control the shape of the power distribution by using the CEA's. This is accomplished by inserting designated blanks of CEA's several inches into the fueled zone. This technique is used for relatively short-term power shaping. CEA's that have reached the end of their usable life (4000 FPD) are housed in spent fuel assemblies which are in the utility's spent fuel pool.

Palisades is a special case; the control 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 weighs 62.2 pounds and the absorber 151.8 pounds.

For the remaining reactors, the CEA's consist of four or five fingers fastened to a spider at their upper ends. The control rod drive mechanism engages the spider assembly to withdraw and position the CEA. The three Palo Verde reactors also have 48 CEA's that consist of twelve fingers and their associated spider. There are eight variations of the number of CEA's in the 14 reactors built by CE involving totals of 45, 81, 89, and 91. Of these totals, 4, 8, 12, or 13 CEA's may have part-length rods with the remainder having full-length rods. Six of the reactors use 14 x 14 fuel. The CEA's for these reactors use Inconel 625 clad control rods 0.948 inch in diameter with a 0.040-inch wall thickness. The pellet diameter is 0.86 inch in the CEA's for all six

reactors. One reactor uses CEA's 152 inches long; this reactor uses 128 inches of B_4C in all five fingers of the full-length CEA's and the center finger of the part-length CEA's. The four outside fingers of the part-length CEA's 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. The other five reactors use 161 inch CEA's and 134 inches of B_4C in the center fingers of the full-length rods. The outside fingers have 2.6 inches of Inconel on the tip, 8.0 inches of Ag-In-Cd alloy and 124 inches of B_4C.

The part-length CEA's 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; three stainless steel rods and two rods with 8 inches of Ag-In-Cd and 124 inches of B_4C; one rod of Al_2O_3 and four rods of silver alloy and B_4C; one rod of Al_2O_3, two rods of silver alloy and B_4C, and two rods having 10 inches of stainless steel and 124 inches of Al_2O_3; and one rod of B_4C, two rods of silver alloy and B_4C, and two rods of stainless steel and Al_2O_3. 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 silver alloy. The total weight of CEA's for 14 x 14 fuel ranges from 63 to 105 pounds.

The CEA's for the eight reactors with 16 x 16 fuel are also of varying length - one at 162.8 inches, one at 180.8 inches and three each at 181.3 and 253.0 inches. The 253-inch CEA's and four of the 181.3-inch CEA's have only four fingers, but all others have five fingers. All are 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, although the Ag-In-Cd alloy may be annular. The poison arrangement also varies. The full-length control rods may be all B_4C; 12.5 inches of Ag-In-Cd, or 9.2 inches of Inconel 625 may be substituted for some of the B_4C near the tip. In all cases, a 0.5-inch stainless steel spacer is used above and below the poison and to separate the poisons when there are two. The end plug and spacer hold the poison 1.125 to 1.25 inches from the tip. The part-length CEA's may

have 68.5 to 75 inches of Inconel 625 near the tip and 14 to 16 inches of B_4C above it or 16 inches of B_4C near the tip with 75 inches of Inconel above it. The plenum springs are 302 stainless steel. The 12-finger CEA's at the three Palo Verde units have 19.5 pounds of stainless steel, 123 pounds of Inconel, and 49.7 pounds of B_4C. The total weight is 192.2 pounds. The full-length CEA's for the other five reactors have 8 pounds of stainless steel, 31.8 to 37.1 pounds of Inconel, 17.4 to 19.3 pounds of B_4C, and 6.9 to 8.6 pounds of silver alloy. The total weight ranges from 65.8 to 72 pounds. The total weight of the part-length CEA's ranges from 83 to 95 pounds. The part-length CEA's consist of about 8 pounds of stainless steel, 2.0 to 2.3 pounds of B_4C, and the balance Inconel. It is assumed that through December 1987 CE has manufactured only the original set of CEA's.

2.8.3.2 Neutron Source Assemblies

Two neutron source assemblies, placed in guide tubes of perimeter assemblies on opposite sides of the core, are used in each core. Their life expectancy is 3500 FPD for the 16 x 16 fuel and 4000 FPD for the other fuel. They are stationary fixtures with an upper shoulder resting on a post of the fuel assembly and held down by a plunger and spring.

Palisades is again a special case and uses two startup sources and two sustaining 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 of antimony and nearly 0.1 pound of polonium. The total weight of each is 4.5 pounds.

The assemblies for 14 x 14 fuel and 16 x 16 fuel are quite similar. The assembly for 16 x 16 fuel is described as follows. The assembly 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, 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 0.812 inch. The assemblies contain 7.2 pounds of stainless steel, 0.4 pound of beryllium, and 0.2 pound each of nickel-based alloy springs, plutonium, and antimony. The assembly for 14 x 14 fuel is 0.875 inch in diameter and about 20 to 30 percent heavier.

2.8.3.3 Burnable Poison Assemblies

Combustion Engineering uses integral burnable poisons; they do not manufacture burnable poison NFA Hardware.

2.8.3.4 Orifice Rod Assemblies

Combustion Engineering does not describe any Orifice Rod Assemblies in CEND-428.

2.8.3.5 In-Core Instrumentation

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 aluminum oxide insulator and sheathed in Inconel 600. The emitter is 0.018 inches in diameter and 15.75 inches long. The sheath is 0.064 inches in diameter and 30 to 116 feet long. Four or five emitters, a calibration tube, a background detector, and an outlet thermocouple are enclosed in a housing tube which is 0.45 inch maximum diameter for Palo Verde and 0.35 inch for all other reactors. The data sheets state that the housing material is Inconel 600 for all reactors, however, it may be stainless steel for those reactors using 14x14 fuel.

The 128 to 150 inches of each assembly in the active core includes 0.01 pounds of rhodium and 0.1 to 0.3 pounds of Al_2O_3. The assemblies for 14x14 fuel have 0.3 to 0.4 pounds of Inconel and 2.0 to 2.9 pounds of stainless steel. The assemblies for 16x16 fuel have 1.7 to 3.1 pounds of Inconel and no stainless steel. There are 28 to 61 assemblies per reactor.

2.8.4 General Electric Nonfuel Assembly Hardware

Preliminary data of GE NFA hardware have been obtained from Safety Analysis Reports for selected reactors. Efforts to obtain more complete data are ongoing.

2.8.4.1 Control Element Assemblies

General Electric use cruciform blades made of stainless steel 304 with a boron carbide absorber contained in 84 stainless steel 304 tubes for control elements in their reactors. The boron carbide is packed into the tubes at about 70% of its theoretical density. The cruciform blades, which are 9.75 inches wide and have a control length of 143 inches, are positioned below the active fuel zone. The weights of the cruciforms have not been given in the Safety Analysis Reports, but estimates based on the materials of construction indicate that cruciforms weigh about 225 pounds. BWR cruciforms have an estimated lifetime of 3 to 25 years. The cruciforms have a unique radiological feature: their end bearings are made of an alloy high in cobalt (Stellite-3 or Haynes-25). The natural cobalt activates to ^{60}Co to a level high enough that these bearings are sometimes removed and packaged for use as ^{60}Co sources. The heat output from these bearings can present a distorted picture of the heat output from the cruciform as a whole.

2.8.4.2 Neutron Sources

General Electric uses five to seven antimony-beryllium neutron source rods per reactor. Each source rod consists of two irradiated 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.

2.8.4.3 Burnable Poison Assemblies

Temporary poison curtains made of borated (3800 to 5400 ppm boron) stainless steel sheets were used to control reactivity in the initial core of early reactors. These curtains, which 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 poisons to provide this reactivity control.

2.8.4.4 Orifice Rod Assemblies

Orifice rod assemblies are not applicable to GE reactors.

2.8.4.5 In-Core Instrumentation

General Electric uses three types of in-core instrumentation: source range monitors, intermediate range monitors, and local power range monitors. Most reactors seem to have four source range monitors and four intermediate range monitors. The number of local power range monitors in dependent of the number of control-rod groups. One four-element local power range monitor is at the center of each four-control-rod group. No weights or dimensions are available at this time.

2.8.4.6 BWR Channels

BWR fuel channels for all BWR/2, 3, 4, and 5 reactors have a square cross section with a 5.278-inch inside width. The nominal length is 162 inches for BWR/2 and 3 reactors and 167 inches for BWR/4 and 5 reactors. Three channel thicknesses have been produced: 80, 100, and 120 mil. Estimates of the weights for these channels for BWR/4 and 5 reactors are 67, 83, and 100 pounds, respectively.

Recently, suggestions have been made that a BWR channel might be reused with a second fuel assembly. This approach could decrease the number of BWR channels by up to 50%; however, the channel would almost

surely have to be thicker. Thus the total mass of the channels might not be greatly affected. Higher burnup fuels may also reduce the utility of this approach.

2.8.5 Westinghouse Non-Fuel Assembly Hardware

2.8.5.1 Control Rod Assemblies

Control rod assemblies for reactors using 17 x 17 fuel consist of 24 individual rods fastened on their upper ends to a spider assembly. Reactors using 15 x 15 fuel have 20 control rods, and reactors using 14 x 14 fuel have 16. The control-rod-drive mechanism engages the spider assembly to withdraw and position the control rod assembly. The spider assembly consists of 4.8 to 7.6 pounds of 304 and 308 stainless steel and 1.65 pounds of Inconel 718 springs for use with 17 x 17 fuel. The assembly for 15 x 15 fuel uses 7.6 pounds of 304 and 308 stainless steel and 1.61 pounds of Inconel X-750. The assemblies for 14 x 14 fuel use 6.25 pounds of 304 and 308 stainless steel for the longer assemblies and 4.25 pounds for the short ones. All use 1.61 pounds of Inconel X-750.

The control rods for 17 x 17 fuel are 151.885 inches long and 0.385 inch in diameter. The cladding is 304 stainless steel with 308 stainless steel end plugs. The cladding is 0.0185 inch thick except for hybrid rods, which have a 0.038 inch thick cladding. The full-length rods contain 142 inches of Ag-In-Cd alloy or hafnium. Part-length rods contain 36 inches of Ag-In-Cd alloy with 106 inches of aluminum oxide spacer above the absorber. Hybrid rods contain 40 inches of Ag-In-Cd alloy with 102 inches of B_4C above it. The part-length and full-length rods contain hold-down springs of 302 stainless steel, whereas the hybrid rods may use Inconel 718. The cladding and end plugs for the hybrid assembly weigh 45 pounds, whereas for the other assemblies they weigh 24 or 25 pounds. The assembly of part-length rods contains 29 pounds of Ag-In-Cd, whereas the hybrid contains 25 pounds and that of the full-length contains 114 pounds. The hybrid assembly also contains about 14 pounds of B_4C. The hafnium version of the full-length assembly

contains 144 pounds of hafnium. The stainless steel springs weigh 1.2 pounds, whereas the Inconel springs weigh 0.6 pound. The total assembly weights range from 93 to 180 pounds. The overall length of the assemblies is 161 inches.

The control rods for 15 x 15 fuel are 150.92 to 152.75 inches long and 0.443 inch in diameter. The 304 stainless steel cladding is 0.020 inch thick and the end plugs are 308 stainless steel. The cladding and end plugs weigh 28 pounds. The absorber is 142 inches of Ag-In-Cd alloy and adds 129 pounds to the weight of the assembly. The hold-down springs in the rods are carbon steel and weigh 0.3 pound. The total assembly weight is 165 pounds and the overall length is 156.6 to 158.5 inches.

The control rods for 14 x 14 fuel are 120 to 153 inches long and 0.435 inch in diameter. The 304 stainless steel cladding is 0.0185 inch thick and the end plugs are 308 stainless steel. The cladding and end plugs weigh 18 to 21 pounds. The absorber is 118 to 142 inches of Ag-In-Cd weighing from 83 to 100 pounds. The hold-down springs in the longer rods are carbon steel weighing 0.3 pound whereas the short rods have 0.4 pound of Inconel X-750 springs. The total assembly weights are 109 to 128 pounds, and the overall length is 134 to 158 inches.

Through June 30, 1986, Westinghouse has manufactured 2,215 control rod assemblies compared to 18,032 fuel assemblies.

2.8.5.2 Neutron Sources

Primary Source Assemblies

The fuel assemblies that do not have control assemblies may be fitted with primary source assemblies, although presumably only one or two per reactor and then probably only for the first cycle or two. (Westinghouse has only manufactured 92, although they describe 106 configurations in WSTD-TME-148.) All these configurations have only one primary source rod. They indicate that six configurations have 20 thimble plugs but no other rods in addition to the source rod. There are 23 configurations with a primary source rod and one, three, or four

secondary source rods and 0, 12, or 16 burnable poison rods with the balance thimble plugs. All others had a primary source rod and 12 to 23 burnable poison rods, with the balance, if any, being thimble plugs. The overall weight ranges from 15 to 52 pounds. The 16 to 24 rods and plugs are attached to a spider pack or hold-down assembly that is held down by the same hold-down plate that keeps the fuel assemblies in place. The hold-down assemblies and spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless and 0.4 pound to 1.4 pounds of Inconel springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pound of Inconel X-750. Overall length ranged from 116.2 inches to 158.8 inches.

Primary source rods for 17 x 17 fuel are clad with 304 stainless tubing and 308 stainless end plugs. The tubing weighs 0.92 pound to 1.07 pounds and the end plugs weigh 0.07 pound. The diameter is 0.385 inch, 0.019-inch wall thickness, 153.7 inches long. The source is californium with a strength of 2×10^8 to 1.2×10^9 Ci. The source rests on 32.5 inches of aluminum oxide spacer and the plenum length ranges from 0.6 inch to 2.65 inches. The total weight of a rod is about three pounds. For the other fuels, the cladding may be PDS 10708BN or 304 stainless steel with PDS 10708BN or 308 stainless steel end caps. The tubing weighs 0.83 pound to 1.1 pounds and the end plugs weigh 0.07 to 0.12 pound. The diameter is 0.371 to 0.466 inch and the wall thickness is 0.016 to 0.024 inch. The length is 110.4 to 152.5 inches. The sources are plutonium-beryllium, polonium-beryllium or californium. Plutonium-beryllium source strengths were 50 Ci, polonium-beryllium 200 Ci, and californium 4×10^8 Ci. The plutonium-beryllium sources rest on 8.3 inches of antimony-beryllium secondary source and has 96.3 inches of antimony-beryllium above it. The polonium-beryllium sources rest on 11.8 inches of Al_2O_3 or 14.8 inches of antimony-beryllium with 100.9 inches of Al_2O_3 or 101.0 inches of antimony-beryllium above it. The californium sources rest on 32.5 to 32.9 inches of Al_2O_3 or 304 stainless steel spacer and some have a spacer above them. The plenum length is 1.25 to 2.65 inches. The total weight of a rod ranges from 2.7 to 3.6 pounds.

Secondary source rods for 17 x 17 fuel are also clad in 304 stainless with 308 stainless end plugs. The tubing weighs 0.92 pound to 1.07 pounds, the end plugs weigh 0.07 pound, and the length is 152.3 inches. The source material is an alloy of 22% beryllium and 77% antimony. It is said to be 88 inches long with no spacer material and a plenum of 1.06 inches. The overall weight of the rod is 2.05 pounds. The 23 configurations that contain both primary and secondary sources might be left in the reactor more cycles than those containing only primary source rods. For the other fuels, the cladding may be PDS 10708BN or 304 stainless steel with PDS 10708BN or 308 stainless steel end caps. The tubing weighs 0.83 pound to 1.06 pounds and the end caps weigh 0.18 pound where data are given. The diameter ranges from 0.431 to 0.474 inch. The source material is antimony-beryllium and ranges in length from 67.1 to 121.65 inches. Again the data indicates that there is no spacer material, but the plenum space indicated that the end plugs must be 3 to 9 inches long. Rod weights range from 2.4 to 3.1 pounds.

Burnable poison rods for 17 x 17 fuel are clad with 0.92 pounds of 304 stainless with 0.08 pound of 308 stainless end plugs. The length is 152.4 inches, and the diameter is 0.385 inch. There is also 304 stainless radial spacer material 142.28 inches long weighing 0.11 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 142 inches long, 0.073 inch thick and weighs 0.70 pound. For the other fuel, 304 stainless steel cladding weighs from 1.10 to 1.18 pounds, and the 308 stainless end plugs weigh from 0.08 to 0.17 pound. The diameter ranges from 0.437 to 0.445 inch and the length from 150.4 to 152.8 inches. The 304 stainless steel spacer material is 142.95 inches long and weighs 0.19 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 0.072 to 0.076 inch thick, 141.6 to 142.7 inches long, and weighs 0.83 pound.

The thimble plugs range in length from 5.15 to 9.16 inches, from 0.424 to 0.502 inch in diameter, and 0.16 to 0.49 pound in weight.

Secondary Source Assemblies

Westinghouse manufactured 120 secondary source assemblies of 95 different configurations. One had three secondary source rods, 12 burnable poison rods and left 5 guide tubes vacant. One configuration had 4 secondary source rods and 16 burnable poison rods, leaving 4 guide tubes vacant. One configuration had 4 secondary source rods; 12 burnable poison rods, and three thimble plugs again leaving 5 guide tubes vacant. Two configurations had 4 secondary source rods; 16 guide tubes were left vacant. Fifty-five configurations had four secondary source rods and all the remaining locations filled with thimble plugs. Seven configurations consisted of four secondary source rods with the remaining locations filled with burnable poison rods. The remaining 28 configurations had 4 secondary source rods and 7 to 16 burnable poison rods; the remaining 4 to 12 locations were thimble plugs. The 16 to 24 rods and plugs are attached to a spider pack or hold-down assembly which is held down by the same hold-down plate which keeps the fuel assemblies in place. The hold-down assemblies and spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless and 0.4 pound to 1.4 pounds of Inconel 718 springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pounds of Inconel X-750.

Secondary source rods for 17 x 17 fuel are clad in 304 stainless with 308 stainless end plugs. The tubing weighs 0.92 pound to 1.07 pounds, the end plugs weigh 0.07 pound, and the length is 152.3 inches. The source material is an alloy of 22% beryllium and 77% antimony. It is said to be 88 inches long with no spacer material and a plenum of 1.06 inches. The overall weight of the rod is 2.05 pounds. For the other fuels, the cladding may be PDS 10708BN or 304 stainless steel with PDS 10708BN or 308 stainless steel end caps. The tubing weighs 0.83 pound to 1.06 pounds, the end caps weigh 0.18 pound where data are given, and the diameter ranges from 0.431 to 0.474 inch. The source material, which is antimony-beryllium, ranges in length from 67.1 to 121.65 inches. Again the data indicate that there is no spacer

material, but the plenum space indicated that the end plugs must be 3 to 9 inches long. Rod weights range from 2.4 to 3.1 pounds.

Burnable poison rods for 17 x 17 fuel are clad with 0.92 pound of 304 stainless with 0.08 pound of 308 stainless end plugs. The length is 152.4 inches and the diameter is 0.385 inch. The 304 stainless steel radial spacer material is 142.28 inches long and weighs 0.11 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 142 inches long, 0.073 inch thick and weighs 0.70 pound. For the other fuel, the 304 stainless steel cladding weighs from 1.10 to 1.18 pounds and the 308 stainless end plugs weigh from 0.08 to 0.17 pound. The diameter ranges from 0.437 to 0.445 inch and the length from 150.4 to 152.8 inches. The 304 stainless steel spacer material is 142.95 inches long and weighs 0.19 pound. The tubing is 0.072 to 0.076 inch thick, 141.6 to 142.7 inches long, and weighs 0.83 pound.

The thimble plugs range in length from 5.15 to 9.16 inches, from 0.424 to 0.502 inch in diameter and 0.16 to 0.49 pound in weight.

The secondary source assemblies range in weight from 15 to 52 pounds and 91.1 to 158.8 inches in overall length.

2.8.5.3 Neutron Poisons

Burnable Poison Assemblies

Westinghouse has fabricated 5003 burnable poison assemblies in 329 different configurations. They had from 1 to 24 burnable poison rods with the remaining locations occupied by thimble plugs. The 16 to 24 rods and plugs are attached to a spider pack or hold-down assembly which is held down by the same hold-down plate that keeps the fuel assemblies in place. The hold-down assemblies and spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless steel and 0.4 pound to 1.4 pounds of Inconel 718 springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pounds of Inconel X-750.

Burnable poison rods for 17 x 17 fuel are clad with 0.92 pound of 304 stainless steel with 0.08 pound of 308 stainless steel end plugs. The length is 152.4 inches and the diameter is 0.385 inch. The 304 stainless steel radial spacer material is 142.28 inches long and weighs 0.11 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 142 inches long, 0.073 inch thick and weighs 0.70 pound. For the other fuel, 304 stainless steel cladding weighs from 1.10 to 1.18 pounds and the 308 stainless end plugs weigh from 0.08 to 0.17 pound. The diameter ranges from 0.437 to 0.445 inch and the length from 150.4 to 152.8 inches. The 304 stainless steel spacer material is 142.95 inches long and weighs 0.19 pound. The absorber material is borosilicate glass tubing with 12.5% natural B_2O_3. The tubing is 0.072 to 0.076 inch thick, 141.6 to 142.7 inches long, and weighs 0.83 pound.

The thimble plugs range from 5.15 to 9.16 inches in length, from 0.424 to 0.502 inch in diameter, and from 0.16 to 0.49 pound in weight.

The overall weight of the assemblies ranges from 11 to 54 pounds and the overall length ranges from 151.6 to 156.9 inches.

Wet Annular Burnable Absorber Assemblies

Westinghouse described 143 configurations using 3 to 24 wet annular burnable absorber rods. All these configurations except one used thimble plugs in the remaining locations. The hold-down assemblies use 3.75 to 5.10 pounds of 304 stainless steel and 0.62 to 0.92 pound of Inconel 718 springs. The thimble plugs are 5.15 to 8.08 inches long and 0.424 to 0.498 inch in diameter. Their weight ranges from 0.16 to 0.30 pound. The absorber rods are 143.1 to 150.1 inches long and contain 105.5 to 134 inches of B4C-Al203 as annular pellets. The pellet wall thickness is 0.070 inch and is clad inside and outside with Zircaloy-4. Rods with shorter absorber lengths have up to 13.5 inches of Zircaloy-4 spacer. The inner cladding has a wall thickness of 0.020 inch, whereas the wall thickness of the outer tube is 0.026 inch. The cladding diameter is 0.381 inch. The top connector is made of 304 stainless steel. Each rod weighs 1.9 pounds and the assembly weighs from 15 to 52 pounds. If any of these assemblies have been built, they must be included in the 5003 burnable poison assemblies described above.

2.8.5.4 Thimble Plug Assemblies

Westinghouse has fabricated 8208 thimble plug assemblies from the thimble plugs and hold-down assemblies described above. The overall weight is expected to range from 9 to 19 pounds. These assemblies may stay in the reactor for more than one cycle. The 16 to 24 plugs are attached to a spider pack or hold-down assembly which is held down by the same hold-down plate which keeps the fuel assemblies in place. The hold-down assemblies and the spider packs consist of 3.7 to 7.8 pounds of 304 or 308 stainless and 0.4 pound to 1.4 pounds of Inconel 718 springs. Alternate springs included 0.02 pound of carbon steel, 0.8 pound of 302 stainless steel and 0.5 pound to 1.1 pounds of Inconel X-750.

The thimble plugs range from 5.15 to 9.16 inches in length, from 0.424 to 0.502 inch in diameter, and from 0.16 to 0.49 pound in weight.

2.8.5.5 In-Core Instrumentation

Westinghouse provides in-core instrumentation, but none was described in WSTD-TME-148.

2.8.6 Radiological Characterization

Because most nonfuel assembly hardware components remain in use for more than one assembly lifetime, they are expected to be more highly activated than corresponding pieces of spent fuel disassembly hardware, for example. Notable exceptions may be BWR fuel channels (if not reused), PWR burnable poison assemblies that are used for only one cycle, and BWR poison curtains. These components, depending on the concentration of niobium in the materials of construction, may qualify as Class C low-level waste. Other components, such as instrumentation, may also qualify as Class C waste because the majority of the component is far removed from the core of the reactor. The LWR NFA Hardware Data Base provides radiological characterization of nonfuel hardware components. A sample radiological description report is shown in Table 2.8.2. Preliminary estimates of the amounts of Greater than Class C Waste from both NFA and SFD hardware are given in section 5.1.1.

2.8.7 References for Section 2.8

Commonwealth Edison. Commonwealth Edison Company and Iowa-Illinois Gas and Electric Company, Quad-Cities Station Units 1 and 2, Safety Analysis Report, DOCKET 50-254-18.

Cooper 1986. Letter from R. G. Cooper, Babcock & Wilcox, to A. R. Irvine, Oak Ridge National Laboratory, dated August 28, 1986.

General Electric 1976. General Electric Corporation, BWR Fuel Channel Mechanical Design and Deflection, NEDO-21354, September 1976.

Hayduk 1987. D. M. Hayduk, Reference Manual of Core Components Fabricated by Combustion Engineering, Combustion Engineering Document CEND-428, March 1987.

Illinois Power. Illnois Power Company, Clinton Power Station Units 1 & 2, Preliminary Safety Analysis Report, DOCKET 50-461-3.

Long Island Lighting. Long Island Lighting Co., Shoreham Nuclear Power Station Unit 1, Final Safety Analysis Report, DOCKET 50-322-110.

Luksic 1987. Letter from A. T. Luksic, Pacific Northwest Laboratory, to K. J. Notz, Oak Ridge National Laboratory, dated June 25, 1987.

Public Service. Public Service Electric and Gas Company, Newbold Island Nuclear Generating Station, Preliminary Safety Analysis Report, DOCKET 50-354-2.

Westinghouse 1986. Westinghouse Electric Corporation, Nuclear Fuel Data, Westinghouse Electric Document WTSD-TME-148, September 1986.

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

Physical Description Report Page: 1

Combustion Enigneering SYSTEM80 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

Poison:

Primary Material:	Boron Carbide (CE)	
Poison 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 EFPD

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

Physical Description Report Page: 2

Combustion Enigneering SYSTEM80 12-Rod Full-Length Control Element

Composition:

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

Used at the Following Reactors:

Reactor	Number in Core
Palo Verde 1	48
Palo Verde 2	48
Palo Verde 3	48

Used with the Following Fuel Assembly Types:

Vendor	Array	Version
Combustion Engineering	16 x 16	System 80

Table 2.8.2. Sample Radiological Description Report from the LWR NFA Hardware Data Base.

Radiological Description Report Page 1

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

ISOTOPIC COMPOSITION

Used for 7 cycles (77,000 MWd/MTIHM) 5 years after discharge
Weight: 97.170 kg Volume of metal: 0.013289 Cu. Meters

Isotope	Grams	Watts	Curies	Curies/m3	Class C Limit	Class C Ratio
C-14	5.348E-04	6.994E-07	2.384E-05	5.311E-03	80	0.6
Ni-59	2.474E-01	7.447E-07	1.876E-02	4.179E+02	220	1.9
Ni-63	3.583E-02	2.227E-04	2.211E+02	4.926E+04	7000	7.0
Co-60	9.512E-03	1.659E-03	1.068E+01	2.397E+03	N/A	N/A
Nb-94	9.760E-03	1.865E-05	1.831E+00	4.097E+02	0.2	220
Total	5.490E+00	1.535E+00	8.349E+03	2.465E+06	N/A	N/A

Used for 10 cycles (111,000 MWd/MTIHM) 5 years after discharge
Weight: 97.170 kg Volume of metal: 0.013289 Cu. Meters

Isotope	Grams	Watts	Curies	Curies/m3	Class C Limit	Class C Ratio
C-14	5.348E-04	6.994E-07	2.384E-05	5.311E-03	80	0.6
Ni-59	2.474E-01	7.447E-07	1.876E-02	4.179E+02	220	1.9
Ni-63	3.583E-02	2.227E-04	2.211E+02	4.926E+04	7000	7.0
Co-60	9.512E-03	1.659E-03	1.068E+01	2.397E+03	N/A	N/A
Nb-94	9.760E-03	1.865E-05	1.831E+00	4.097E+02	0.2	220
Total	5.490E+00	1.535E+00	8.349E+03	2.465E+06	N/A	N/A

NOTE: The data presented here is only for the purpose of illustrating the form of the Radiological Description Report. It is not intended to be used for any purpose other than that illustration.

Table 2.8.2. Sample Radiological Description Report from the LWR NFA Hardware Data Base (cont.).

Radiological Description Report Page 2

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

PHOTON SPECTRA

Mean Energy(MeV)	Photons/second (77,000 MWd/MTIHM)	Photons/second (110,000 MWd/MTIHM)
0.0100	2.162E+10	3.569E+10
0.0250	3.674E+09	6.063E+09
0.0375	2.088E+09	3.444E+09
0.0575	2.397E+09	3.874E+09
0.0850	9.237E+08	1.524E+09
0.1250	3.548E+08	7.851E+08
0.2250	1.167E+08	1.925E+08
0.3750	3.272E+07	5.396E+07
0.5750	1.879E+06	3.099E+06
0.8500	6.411E+08	9.650E+08
1.2500	7.960E+11	1.313E+12
1.7500	2.253E+01	2.768E+01
2.2500	4.219E+06	6.956E+06
2.7500	1.306E+04	2.152E+04

METALLIC COMPOSITION
(Materials modeled to obtain this report)

Material	Total Weight (kg)	Zone
Inconel 625	53.620	Top
Boron Carbide	20.900	Top
Stainless Steel 304	8.170	Top
Inconel 625	2.200	Gas Plenum
Boron Carbide	1.600	Gas Plenum
Stainless Steel 304	0.680	Gas Plenum

NOTE: The data presented here is only for the purpose of illustrating the form of the Radiological Description Report. It is not intended to be used for any purpose other than that illustration.

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 Plant (SRP), Hanford (HANF), and Idaho National Engineering Laboratory (INEL) for shipment to one or more geologic repositories. Data are presented in this section on the estimated physical characteristics and production schedules of the canisters of immobilized waste through the year 2020.

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, SRP, 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. According to SRP experiments, cooling the canister to ambient temperature does not reduce the glass level in the canister appreciably. The designs are similar but not identical; wall thickness and filler neck dimensions vary. The weight of a loaded canister is about 2150 to 2180 kg, of which about 1650 to 1895 kg is HLW glass.

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 requires 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 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, SRP, and HANF.

Table 3.1.1 also summarizes the estimated maximum radioactivity and thermal power (curies and watts) per canister at the time of filling,

based on the most highly radioactive immobilized waste composition at each site. Curies and watts as functions of decay time after filling are given in subsequent sections on the individual sites.

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 completely established. The estimate for the WVDP site is about 275 to 300 canisters; this estimate should be fairly accurate, because the amount of waste at WVDP is a known quantity. For the three 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, which will be referred to as the base case. Other possible cases that give larger numbers of canisters are also discussed, and the assumptions and processing options that go into each case are described.

Table 3.1.2 shows the estimated production schedule of canisters at each site in the base case. Both the annual number and the cumulative number of canisters are shown for each year through the year 2020. In this scenario, based on the assumptions used in this report, it is estimated that a total of about 18,000 canisters will have been produced by the end of 2020. Table 3.1.3 summarizes the assumptions on which the estimates in Table 3.1.2 are based. All projections were obtained from the respective sites. Startup dates are based on current plans, which call for initial vitrification to begin at WVDP and SRP in 1990 and at HANF in 1996. For INEL, the actual strategy and process for disposal of HLW will not be decided until the 1990s; however, for planning purposes it is assumed that the glass-ceramic waste form will be used for immobilization and that immobilization will start in 2011. The immobilized waste generation schedule for INEL shown in Table 3.1.2 assumes 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 1000 canisters/year is assumed to allow for working off the backlog of stored calcine over a period of about 30 years (Berreth 1987).

The production schedules detailed in Table 3.1.2 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 Figure 3.1.2 shows the cumulative number of canisters summed for all sites.

3.1.2.1 Comparison with Previous Projections

A recent report on defense HLW repository fee calculation (DOE/RL-86-10, 1986) gave estimated defense HLW canister production for three cases referred to (in order of increasing number of canisters) as the base case, augmented case, and maximum case. These cases were described as follows: The base case gave the quantity of defense HLW from SRP, HANF, and INEL that was expected to go to the geologic repositories, assuming that INEL waste was in the low-volume glass-ceramic form with removal of inerts prior to immobilization. The augmented case was the same as the base case except that the INEL waste was assumed to be converted to glass-ceramic form without removal of inerts prior to immobilization. The maximum case was the augmented case plus vitrified HANF single-shell tank waste and overpacked strontium and cesium capsules; in this case, each overpack was counted as a canister. The number of canisters produced in each of these cases is shown in Table 3.1.4, and the year-by-year production schedule of canisters for the base case is shown in Table 3.1.5. Table 3.1.6 summarizes the assumptions that were used to produce the data in Tables 3.1.4 and 3.1.5.

In DOE/RL-86-10, the total number of canisters produced by a given year was stated in terms of "equivalent" canisters. This means that all of the HLW produced by that year was included in the calculation of the number of canisters, although this total quantity of waste would not actually be canistered until several years later. Thus the cumulative equivalent numbers of canisters shown in Table 3.1.4 for a given year are not readily comparable to those calculated in the present report (Table 3.1.2), which are based on cumulative actual canisters produced by a given year. However, Table 3.1.5, which shows the DOE/RL-86-10 base-case canister production schedule, can be directly compared with

the base case schedule in our Table 3.1.2, since both are in terms of actual canisters. The total number of defense HLW canisters produced through the year 2020 is about 17,500 in this report and about 14,000 in the base case of DOE/RL-86-10. Most of the difference is in the production at INEL: 8,800 canisters in this report compared with 4,350 in DOE/RL-86-10. This in turn was due to the assumption of inerts removal prior to immobilization in DOE/RL-86-10, which gave an immobilization rate of 335 canisters/yr vs 1,000 canisters/yr in this report. INEL considers that the assumption of inerts removal prior to immobilization is based on unproven technology and gives no cost advantage, and therefore INEL did not use the assumption of inerts removal in making their own estimates of the number of canisters produced (Berreth and Knecht 1986, Berreth 1987a). Thus the augmented case in DOE/RL-86-10 (no inerts removal prior to immobilization) is more comparable to the base case in the present report; the augmented case would have an INEL production rate of about 1,000 canisters/yr for a total production at INEL of 13,000 canisters by 2020, and an overall total DHLW production of about 22,600 canisters by 2020. This is about 5,000 canisters more than the 17,500 estimated in this report. Most of this difference is accounted for by the recently revised INEL startup schedule (Berreth 1987), which shows a total production of 8,800 canisters by the end of 2020 based on starting up in year 2011, rather than the 13,000 canisters that would be produced in the DOE/RL-86-10 augmented case based on start-up at full rate in year 2008. Taking this adjustment into account, the remaining difference in total canister production in the two reports is about 900 canisters by year 2020. This is accounted for by differences in SRP and HANF production in the two reports. For SRP, the present report shows 6,800 canisters versus 7,500 in DOE/RL-86-10, and for HANF the corresponding estimates were 1900 canisters in this report and 2100 in DOE/RL-86-10. The DOE/RL-86-10 projections include "future" production of 1100 canisters at SRP and HANF, split about equally between the two sites. In the present report there is no category of "future" production; updated projections from the defense sites through the year 2020 were used directly. As mentioned previously, the totals presented here do not include any canisters produced after the year 2020.

In DOE/RL-86-10, it was estimated that vitrification of HANF single-shell tank waste for repository emplacement, if required, would result in an additional 21,500 canisters. Based on the current design throughput of the HANF vitrification plant (145 canisters/yr), production of this number of canisters would take many years. Referring to the base case production schedule shown in Table 3.1.2, the maximum additional number of canisters that could be produced at HANF from year 2011 to year 2020 would be about 1500 canisters. Thus the maximum case for this report (i.e., including vitrification of single-shell tank waste) would be the base case plus 1500 canisters, or a total of 19,300 canisters by the end of year 2020. This assumes that the present HANF vitrification plant design throughput is not increased; however, it appears likely that this throughput would be revised upward if vitrification of single-shell tank waste became necessary. Also, it appears likely that the requirement of 21,500 canisters for this waste could be considerably reduced by pretreatment.

3.1.3 Radiological Properties

To the extent possible, data were obtained from the sites on the projected radionuclide content of the canisters at the time of filling. These data represent estimated maximum radioactivities per canister for the three defense sites and both maximum and average radioactivity per canister for WVDP. Where current information from the sites was unavailable, estimates were made based on previously published data. Using this information, ORIGEN2 decay calculations were made to determine the grams, 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 and photon energy distributions.

Cumulative average radioactivities per canister have also been estimated for the defense sites based on their projections of cumulative radioactivity in glass and cumulative number of canisters produced.

These cumulative average radioactivities per canister are in general much 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 loading at time points subsequent to the year 2020. However, the year-by-year cumulative average radioactivities and thermal power per canister have been calculated from preliminary data that cannot take into account actual tankage allocations and detailed processing schedules; thus these averages should not be used for detailed short-term calculations.

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. 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 vitrification begins, it might be desirable to try to work off the older waste first; however, this may not be feasible because of tankage constraints. 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 depend 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.

Little data are available on estimated melter feed compositions and on the estimated variation of these compositions following startup of the vitrification plants. Two defense sites (SRP and HANF) have each released compositions representing feeds of estimated maximum activity. HANF has also issued four compositions representing the estimated variation of radionuclide contents between 1996 and 2000 (Mitchell 1986). INEL, because of security restrictions, has released no 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 pinpoints the variation of radionuclide compositions of melter feeds with time as an area requiring additional information and analysis.

Equally important in the assessment of the data in this report, it should be recognized that various strategies and processing alternatives for immobilized waste production are still under consideration. Also, future defense production requirements may change. This report presents a scenario that appears likely at this time; however, changes in the canister production schedule, the radiological properties of canisters, and the total number of canisters produced 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 several thousand additional canisters will be required after that time.

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, SRP, 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, (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

Berreth 1987. Letters from J. Berreth, INEL, to J. E. Solecki, DOE/IDO, March 19, 1987 and April 1, 1987.

Chandler 1987. Letter from R. L. Chandler to M. W. Shupe, HLW Lead Office, Richland, transmitting SRP input to DHLW Integrated Data Base, April 1, 1987.

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

DOE/RL-86-10, 1986. Defense High-Level Waste Technology Program Office, Perspective on Methods to Calculate a Fee for Disposal of Defense High-Level Waste in Combined (Civilian/Defense) Repositories, December 1986.

McDonell and Goodlett 1984. W. R. McDonell and C. B. Goodlett, Systems Costs for Disposal of Savannah River High-Level Waste Sludge and Salt, DP-MS-83-121, August 1984.

Mitchell 1986. D. E. Mitchell, Hanford Waste Vitrification Plant, Preliminary Description of Waste Form and Canister, RHO-RE-SR-55P, August 1986.

Rykken 1987. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, March 25, 1987.

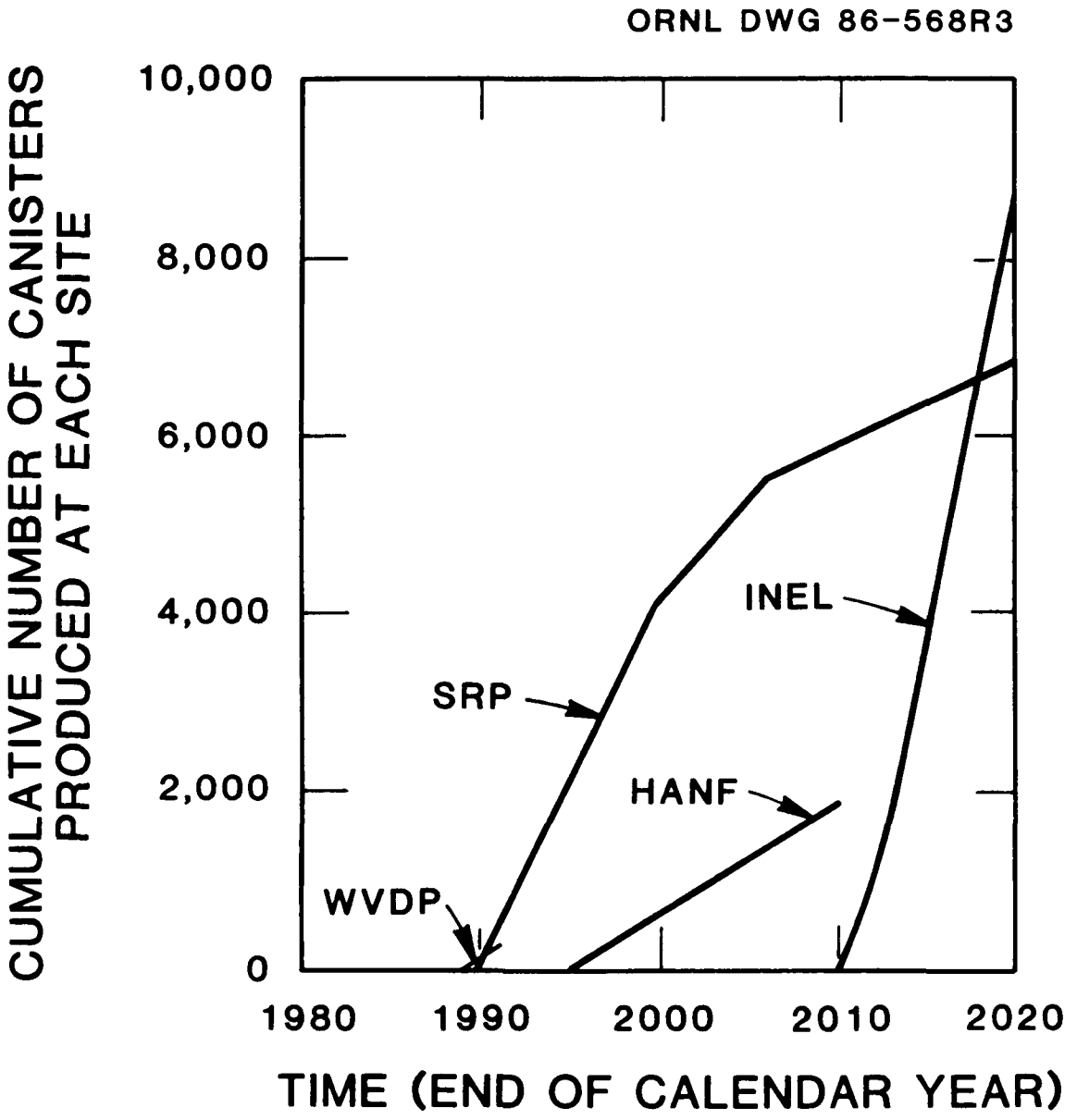


Fig. 3.1.1. Cumulative number of canisters of HLW produced at each individual site. Base case, as shown in Table 3.1.2.

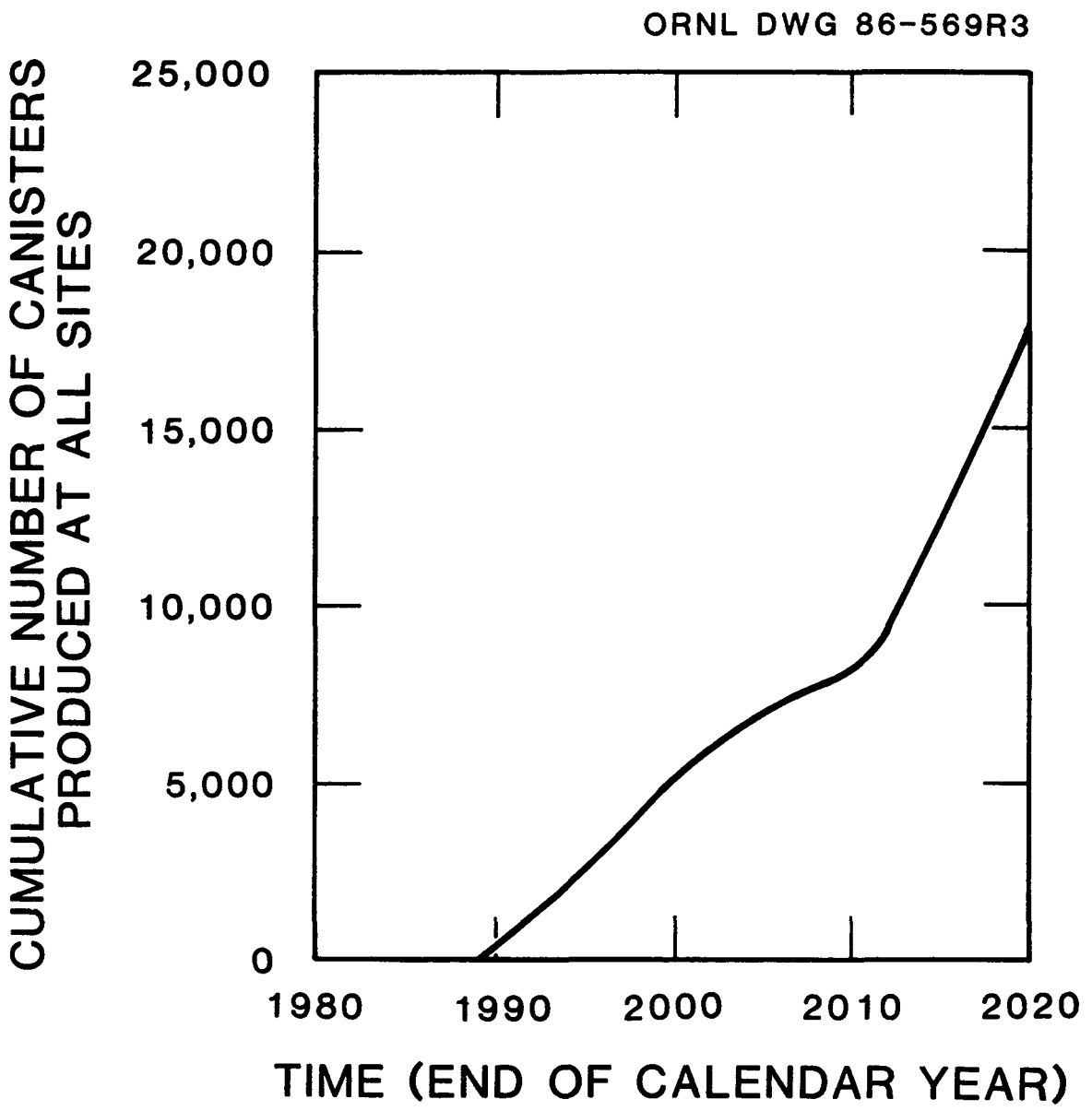


Fig. 3.1.2. Cumulative number of canisters of HLW produced, total for all four sites. Base case, as shown in Table 3.1.2.

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

	West Valley Demonstration Project	Savannah River Plant	Hanford	Idaho National Engineering Laboratory
Outside diameter, cm	61	61	61	61
Overall height, cm	300	300	300	300
Material	SS	SS	SS	SS
Wall thickness, cm	0.34	0.95	0.95	0.95
Weights (kg)				
Canister	252	500	500	500
Glass or ceramic	1895	1682	1650	1825
Total	2147	2182	2150	2325
Curies per canister[a]	125,200	234,400	416,000	143,000
Watts per canister[a]	382	709	1158	446

[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 1990. Maximum values for the defense sites do not represent initial operations; canisters of maximum activity will not be produced until after several years of operation.

Table 3.1.2. Number of canisters of immobilized HLW produced at all sites[a,b]

End of calendar year	West Valley Demonstration Project		Savannah River Plant		Hanford Operations		Idaho National Engineering Laboratory		Total for all four sites	
	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced	Number of canisters produced during year	Cumulative number of canisters produced
1988	0	0	0	0	0	0	0	0	0	0
1989	0	0	0	0	0	0	0	0	0	0
1990	130	130	102	102	0	0	0	0	232	232
1991	145	275	410	512	0	0	0	0	555	787
1992	0	275	410	922	0	0	0	0	410	1197
1993	0	275	410	1332	0	0	0	0	410	1607
1994	0	275	410	1742	0	0	0	0	410	2017
1995	0	275	410	2152	0	0	0	0	410	2427
1996	0	275	410	2562	145	145	0	0	555	2982
1997	0	275	410	2972	145	290	0	0	555	3537
1998	0	275	410	3382	145	435	0	0	555	4092
1999	0	275	410	3792	73	508	0	0	483	4575
2000	0	275	410	4202	145	653	0	0	555	5130
2001	0	275	220	4422	145	798	0	0	365	5495
2002	0	275	220	4642	72	870	0	0	292	5787
2003	0	275	220	4862	145	1015	0	0	365	6152
2004	0	275	220	5082	145	1160	0	0	365	6517
2005	0	275	220	5302	145	1305	0	0	365	6882
2006	0	275	220	5522	73	1378	0	0	293	7175
2007	0	275	92	5614	145	1523	0	0	237	7412
2008	0	275	92	5706	145	1668	0	0	237	7649
2009	0	275	92	5798	72	1740	0	0	164	7813
2010	0	275	92	5890	120	1860	0	0	212	8025
2011	0	275	92	5982	0[c]	1860	500	500	592	8617
2012	0	275	92	6074	0	1860	600	1100	692	9309
2013	0	275	92	6166	0	1860	700	1800	792	10101
2014	0	275	92	6258	0	1860	1000	2800	1092	11193
2015	0	275	92	6350	0	1860	1000	3800	1092	12285
2016	0	275	92	6442	0	1860	1000	4800	1092	13377
2017	0	275	92	6534	0	1860	1000	5800	1092	14469
2018	0	275	92	6626	0	1860	1000	6800	1092	15561
2019	0	275	92	6718	0	1860	1000	7800	1092	16653
2020	0	275	92	6810	0	1860	1000	8800	1092	17745

[a]Sources: WVDP - Rykken 1987.
SRP - Chandler 1987 (IDB submittal).
HANF - Coony 1987.
INEL - Berreth 1987.

[b]For assumptions used in compiling this table see Table 3.1.3. This table represents the base case for this report. Canisters produced after 2020 are not included here.

[c]The Hanford schedule is based on the assumption that there will be no fuel reprocessing operations after calendar year 2001. Some planning scenarios do project such operations beyond CY 2001. Each additional year of fuel reprocessing would generate about 50 canisters after CY 2010 (Coony 1987).

Table 3.1.3. Assumptions used in base case of this report

1. Canister dimensions 61 cm diameter 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; SRP, 410; HANF, 145; INEL, 1000.

3. Production of canisters of HLW starts at WVDP and SRP in 1990, at HANF in 1996, and at INEL in 2011. Canister production is shown through the end of year 2020 and does not include any waste canistered after 2020.

4. WVDP canister production is based on 520,000 kg of total glass loaded at 1890 kg/canister (Rykken 1987).

5. SRP canister production is based on SRP 1987 IDB submittal (Boore 1987). Waste production rates are based on operation of three reactors throughout the projection period (2020); last year's forecast was based on a four-reactor case. The DWPF is assumed to reach full production in 4QFY 1990. It is assumed that sludge and liquid inventories will reach steady state in 2006. Steady-state volumes must be maintained to allow waste to age before it is processed.

6. HANF canister production is based on HANF "reference alternative," in which single-shell tank waste is not vitrified but is immobilized in place. About 21,000 m^3 of single-shell tank liquid is transferred to double-shell tanks between 1985 and 1996. HANF strontium/cesium capsules are not reprocessed and vitrified but are overpacked for emplacement in a repository; overpacks are not included in canister production figures. It is assumed that the N Reactor operates through the year 2000 and the Purex plant operates through the year 2001. It is assumed in this projection that there will be no fuel reprocessing operations after year 2001. However, there are some planning scenarios that do project such operations beyond year 2001.

7. INEL canister production is based on the schedule given in Berreth 1987a. The ceramic-based waste form is used for immobilization. No removal of inerts prior to immobilization was assumed. A canister load is 1825 kg of ceramic, equivalent to 1277 kg of calcine. Density of ceramic is 3200 kg/m^3. The maximum production rate is 1000 canisters/year; this permits working off the stored calcine over about a 30-year period.

Table 3.1.4. Defense high-level waste quantities and characteristics used in report DOE/RL-86-10.[a]

Site/waste form	Cumulative number of equivalent canisters to: 1986	2000	2020
Base case			
Savannah River Plant - glass	4,900	7,000	7,000
Hanford - glass	800	1,500	1,500
Future production (2000-2020)[b] - glass	-	-	1,500
Idaho - ceramic (with inerts removal)[c]	900	2,500	6,000
Total of base case	6,600	11,000	16,000
Augmented			
Base case without Idaho	5,700	8,500	10,000
Idaho - ceramic (without inerts removal)	3,000	9,000	22,000
Total augmented case	8,700	17,500	32,000
Maximum			
Augmented quantity	8,700	17,500	32,000
Hanford additional - glass[d]	21,500	21,500	21,500
Hanford Cs and Sr capsules - overpacked	500	500	500
Total maximum case	30,700	39,500	54,000

[a]This table is excerpted from Table 4.1 of DOE/RL-86-10, December 1986; cumulative radioactivity and thermal power have been deleted. All values shown are totals for the year shown; that is, they are not additive in the horizontal direction. "Equivalent canisters" means the number of canisters that would result if all of the waste produced by a given year were immobilized.

[b]Assumed equivalent to two Savannah River reactors on sites to be determined.

[c]This case is based on a postulated process for removing inerts prior to immobilization.

[d]Quantity if all single-shell-tank (SST) waste must go to geologic disposal.

Table 3.1.5. Defense high-level waste canister production schedule used in base case of report DOE/RL-86-10[a]

Year	Savannah River		Hanford		Future production[b]		Idaho[c]		DHLW cumm. total
	Rate	Total	Rate	Total	Rate	Total	Rate	Total	
1990	405	405							405
1991	405	810							810
1992	405	1,215							1,215
1993	405	1,620							1,620
1994	405	2,025							2,025
1995	405	2,430							2,430
1996	405	2,835							2,835
1997	405	3,240	135	135					3,375
1998	405	3,645	135	270					3,915
1999	405	4,050	135	405					4,455
2000	405	4,455	135	540					4,995
2001	405	4,860	135	675					5,535
2002	405	5,265	135	810					6,075
2003	105	5,370	135	945					6,315
2004	105	5,475	135	1,080					6,555
2005	105	5,580	135	1,215					6,795
2006	105	5,685	135	1,350	75	75			7,110
2007	105	5,790	150	1,500	75	150			7,440
2008	105	5,895			75	225	335	335	7,955
2009	105	6,000			75	300	335	670	8,470
2010	105	6,105			75	375	335	1,005	8,985
2011	105	6,210			75	450	335	1,340	9,500
2012	105	6,315			75	525	335	1,675	10,015
2013	105	6,420			75	600	335	2,010	10,530
2014	105	6,525			75	675	335	2,345	11,045
2015	105	6,630			75	750	335	2,680	11,560
2016	105	6,735			75	825	335	3,015	12,075
2017	105	6,840			75	900	335	3,350	12,590
2018	105	6,945			75	975	335	3,685	13,105
2019	55	7,000			75	1,050	335	4,020	13,570
2020					75	1,125	335	4,355	13,980
Total canisters produced		7,000		1,500		1,125		4,355	13,980
Total canisters from waste generated through 2020		7,000		1,500		1,500		6,000	16,000

[a]This table is the same as Table 7.1 of DOE/RL-86-10, December 1986. This schedule is representative of the many possible production scenarios.

[b]A level rate for 20 years was assumed.

[c]Processing for Future and Idaho will continue after 2020 to allow for cooling of wastes produced from 2016 to 2020. The additional waste that will be produced at Idaho after 2020 is not considered here.

Table 3.1.6. Assumptions used in estimating number of defense HLW canisters in report DOE/RL-86-10, December 1986

1. Definitions of cases:

   Base case: The quantity of immobilized defense HLW expected to go to geologic repositories, with INEL waste in the low-volume ceramic form with inerts removed prior to immobilization.

   Augmented case: Same as base case except that INEL waste volume is increased because inerts are not removed.

   Maximum case: Same as augmented case except that HANF single-shell tank waste is vitrified and HANF strontium-cesium capsules are overpacked for shipment to repository.

2. (All cases) canister dimensions 61 cm diameter x 300 cm overall length.

3. (All cases) production of HWL canisters begins at SRP in 1990, at HANF in 1997, and at INEL in 2008.

4. (All cases) canister production in Table 3.1.4 is given as number of equivalent canisters; this means the number of canisters that would be required if all of the waste produced through a given year were immobilized.

5. (Base case) SRP production is based on McDonell and Goodlett, 1984.

6. (Base case) HANF production is based on the HANF "reference alternative," in which single-shell tank waste is not vitrified, but is immobilized in place.

7. (Base case) Future defense production (i.e. production from 2000 to 2020) is shown separately; but it is likely that this production will be split between SRP and HANF. Future production is assumed to be equivalent to two SRP reactors and is assumed to give a constant rate of canister production of 75 canisters/year for 20 years.

8. (Maximum case) HANF single-shell tank waste is vitrified, requiring an additional 21,500 canisters. The strontium and cesium capsules produced at HANF are placed in overpacks for disposal at the repository. The total number of overpacks is 500; each of these is counted as a canister.

3.2 WEST VALLEY DEMONSTRATION PROJECT

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. 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 and disposal in a federal repository.

3.2.2 Types of HLW Produced

Only one type of immobilized HLW will be produced at WVDP, i.e. 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 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 $\pm$5%. The empty canister weighs about 234 kg. When filled to 85% of capacity, each canister will contain 0.70 m^3 (about 1895 kg) of vitrified waste and will weigh about 2147 kg. The density of the solidified waste glass is approximately 2.7 g/cm^3 at 25°C (Rykken 1986a,b,c, Eisenstatt 1986).

3.2.4 Inventory and Production Schedule

Cold operations at the vitrification plant are scheduled to start in 1989 and be completed in April 1990. Vitrification of waste is scheduled to begin in April 1990 and to be completed about October 1991. A total of about 520,000 kg of vitrified waste (about 275 canisters) will be produced during this period (Rykken 1987). Table 3.2.2 shows the estimated schedule of production; this schedule is based on a single

campaign with a duration of 18 months, starting April 1990 and ending in October 1991. This allows about 20% offstream time for scheduled and unscheduled shutdowns, producing a total of 109 batches at 100 onstream hours per batch. On this basis, 130 canisters of waste will be produced in 1990 and 145 canisters in 1991. This will account for the entire quantity of HLW at WVDP.

3.2.5 Radionuclide Content per Canister

The initial radionuclide contents per canister of the glass were taken from Eisenstatt 1986, which gives average, maximum, and minimum values. These compositions, expressed as curies of each radionuclide per canister, are shown in Table 3.2.3. Data are for the year 1990.

3.2.6 Radioactivity and Thermal Power

Table 3.2.4 shows calculated radioactivity and thermal power per canister as a function of decay time for maximum and average canisters for periods up to 1,000,000 years. The decay calculations were made with the ORIGEN2 code using compositional data for the year 1990 as the starting point; thus the year 1990 represents the zero point for decay time. As shown in Table 3.2.4, a maximum-activity canister produced in 1990 would have a radioactivity of 125,200 Ci and a thermal power of 382 W. The same canister in 1995, or a new canister filled in 1995 from this same batch of waste, would have a maximum radioactivity of 111,100 Ci and a thermal power of 341 W. The corresponding average values of radioactivity and thermal power per canister are 112,700 Ci and 339 W for year 1990 and 100,200 Ci and 303 W for year 1995.

More detailed tables, showing the contributions of individual radionuclides, are given in Appendix 3A (Tables 3A.1-3A.6).

3.2.7 Chemical Composition

Table 3.2.5 shows the expected chemical composition of the HLW glass to be produced at WVDP and the possible range of variation of the concentrations of individual components (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. Estimates of

the number of canisters to be produced range from 275 (Rykken 1987) to 300 (Bixby 1987). We used 275 since that gives a higher value for radioactivity per canister; however, the 300 estimate is more conservative from the standpoint of space requirement for the repository.

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, the maximum initial activity per canister can exceed the average by an amount estimated at 11%, as indicated in Table 3.2.4.

3.2.9 References for Section 3.2

Bixby 1987. Letter from W. W. Bixby, West Valley Project Office, to S. N. Storch, ORNL, February 27, 1987.

Eisenstatt 1986. L. R. Eisenstatt, Description of the West Valley Demonstration Project Reference High-Level Waste Form and Canister, WVDP-056, July 1986.

Rykken 1986a. Letter from L. E. Rykken, WVDP, to R. Salmon, ORNL, April 11, 1986.

Rykken 1986b. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, April 16, 1986.

Rykken 1986c. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, April 23, 1986.

Rykken 1987. Telephone conversation, L. E. Rykken, WVDP, and R. Salmon, ORNL, March 25, 1987.

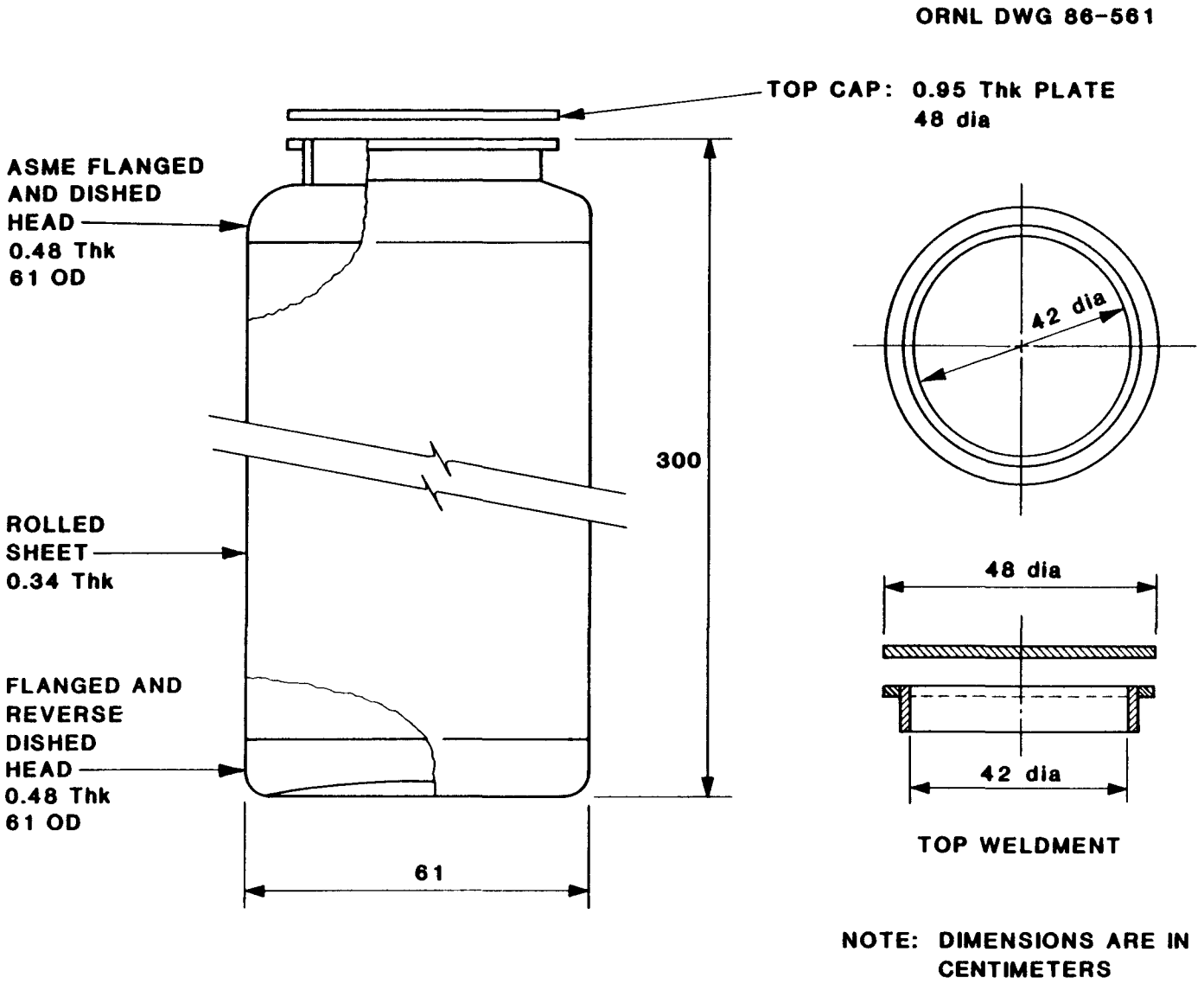


Fig. 3.2.1. West Valley Demonstration Project HLW canister. Source: Rykken 1986b.

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

Waste form	Borosilicate glass in closed canister
Canister material	Stainless steel type 304L
Borosilicate glass density, g/cm^3 at 25°C	2.7
Weights per canister:	
Empty canister, kg	234
Cover, kg	18
Borosilicate glass, kg	1,895
Total loaded weight, kg	2,147
Canister dimensions:	
Outside diameter, cm	61
Height overall, cm	300
Wall thickness, cm	0.34
Radionuclide content, curies per canister (1990)[b]	
Average	112,700
Maximum	125,200
Thermal power, watts per canister (1990)[b]	
Average	339
Maximum	382

[a]Source: Eisenstatt 1986 and ORNL calculations.

[b]Quantities shown are at 85% fill. Curies and watts per canister are for the year 1990.

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)
1989	0	0	0
1990	130	130	246,000
1991	145	275	520,000

[a]Based on Rykken, 1987. Canister fill volume is assumed to be 85%. Each canister contains about 1895 kg of glass.

Table 3.2.3. West Valley Demonstration Project.
Radioisotope content per HLW canister[a]

Radionuclide	Radioactivity (Ci)		
	Nominal	Range	
Fe-55	1.9E+0	1.7E+0	2.1E+0
Ni-59	3.2E-1	2.9E-1	3.6E-1
Ni-63	2.5E+1	2.3E+1	2.7E+1
Co-60	3.2E+0	2.8E+0	3.6E+0
Se-79	1.5E-2	1.3E-2	1.6E-2
Sr-90	2.7E+4	2.4E+4	3.0E+4
Y-90	2.7E+4	2.4E+4	3.0E+4
Zr-93	9.5E-1	8.5E-1	1.1E+0
Nb-93m	7.8E-1	7.0E-1	8.6E-1
Tc-99	6.7E+0	6.0E+0	7.4E+0
Ru-106	3.2E-2	2.9E-2	3.6E-2
Rh-106	3.2E-2	2.9E-2	3.6E-2
Pd-107	4.7E-3	4.2E-3	5.3E-3
Sb-125	8.4E+0	7.5E+0	9.3E+0
Te-125m	1.9E+0	1.7E+0	2.1E+0
Sn-126	1.6E-1	1.4E-1	1.8E-1
Sb-126m	1.6E-1	1.4E-1	1.8E-1
Sb-126	2.2E-1	2.0E-1	2.5E-1
Cs-134	1.5E+1	1.3E+1	1.6E+1
Cs-135	6.3E-1	5.6E-1	7.0E-1
Cs-137	2.9E+4	2.5E+4	3.2E+4
Ba-137m	2.7E+4	2.4E+4	3.0E+4
Ce-144	3.8E-3	3.4E-3	4.3E-3
Pr-144	3.8E-3	3.4E-3	4.3E-3
Pm-147	5.4E+2	5.0E+2	6.3E+2
Sm-151	8.1E+2	7.2E+2	9.0E+2
Eu-152	1.5E+0	1.3E+0	1.6E+0
Eu-154	4.0E+2	3.6E+2	4.5E+2
Eu-155	5.9E+1	5.3E+1	6.5E+1
Th-232	6.3E-3	4.7E-3	8.0E-3
U-233	3.8E-2	3.4E-2	4.2E-2
U-234	1.7E-2	1.6E-2	1.9E-2
U-235	3.9E-4	3.5E-4	4.4E-4
U-236	1.1E-3	9.9E-4	1.2E-3
U-238	3.1E-3	2.8E-3	3.5E-3
Np-237	4.3E-2	2.0E-2	6.9E-2
Np-239	9.4E+0	4.4E+0	1.5E+1
Pu-238	2.7E+1	2.4E+1	3.0E+1
Pu-239	6.8E+0	6.1E+0	7.6E+0
Pu-240	1.5E+1	8.7E+0	1.9E+1
Pu-241	3.0E+2	2.6E+2	3.3E+2
Pu-242	6.8E-3	6.0E-3	7.5E-3
Am-241	3.4E+2	1.7E+2	5.0E+2
Am-242	8.3E-2	3.8E-2	1.3E-1
Am-242m	8.3E-2	3.8E-2	1.3E-1
Am-243	9.4E+0	4.4E+0	1.5E+1
Cm-242	8.3E-2	3.8E-2	1.3E-1
Cm-243	6.2E-1	3.0E-1	1.0E+0
Cm-244	7.8E+1	3.7E+1	1.2E+2
Cm-245	3.9E-2	1.9E-2	6.3E-2
Cm-246	1.7E-2	8.0E-3	2.7E-2

[a]Source: Eisenstatt 1986. Quantities shown are for the year 1990 and are based on a canister containing 1895 kg of HLW glass.

Table 3.2.4. West Valley Demonstration Project. Calculated radioactivity and thermal power per HLW canister.[a]

Decay time after 1990, years	Radioactivity per canister (Ci)		Thermal power per canister (W)	
	Average	Maximum	Average	Maximum
0	112,700	125,200	339	382
1	110,400	122,400	333	375
2	107,700	119,500	325	366
5	100,200	111,100	303	341
10	89,040	98,750	270	305
15	79,190	87,840	241	272
20	70,480	78,180	216	244
30	55,870	62,000	173	196
50	35,220	39,120	112	128
100	11,350	12,690	42	50
200	1,510	1,770	12.6	17.2
300	450	587	8.6	12.3
350	339	457	7.8	11.2
500	226	317	6.2	9.0
1,000	119	168	3.3	4.7
1,050	113	159	3.1	4.4
2,000	59.3	80.8	1.4	1.9
5,000	38.0	50.2	0.72	0.96
10,000	29.5	38.1	0.51	0.69
20,000	20.3	25.3	0.30	0.39
50,000	12.3	14.4	0.11	0.14
100,000	9.2	10.7	0.05	0.06
500,000	4.6	5.6	0.03	0.04
1,000,000	3.0	3.8	0.03	0.04

[a]Calculations made with ORIGEN2 code based on data supplied by WVDP (Eisenstatt 1986). Canister contains 1895 kg of HLW glass. Initial time point (0 years) is at year 1990.

Table 3.2.5. West Valley Demonstration Project. Chemical composition of reference HLW glass[a]

Component	Nominal composition (wt %)	Range (wt %)	
AgO	0.0001	-	-
Al_2O_3	2.8295	1.19	7.15
AmO_2	0.0073	-	-
BaO	0.0540	0.04	0.08
B_2O_3	9.9516	9.33	10.66
CaO	0.5993	0.39	0.93
CdO	0.0003	-	-
CeO_2	0.0670	0.04	0.10
CmO_2	0.0001	-	-
CoO	0.0002	-	-
Cr_2O_3	0.3112	0.21	0.48
Cs_2O	0.0826	0.05	0.13
CuO	0.0001	-	-
Eu_2O_3	0.0014	-	-
Fe_2O_3	12.1573	8.32	18.50
Gd_2O_3	0.0003	-	-
In_2O_3	0.0001	-	-
K_2O	3.5733	3.36	3.84
La_2O_3	0.0337	0.02	0.05
Li_2O	3.0315	2.84	3.25
MgO	1.3032	1.22	1.39
MnO_2	1.3107	0.84	1.96
MoO_3	0.0088	-	0.01
NaCl	0.0183	0.01	0.03
NaF	0.0013	-	-
Na_2O	10.9335	10.25	11.71
Nd_2O_3	0.1209	0.08	0.19
NiO	0.3358	0.22	0.52
NpO_2	0.0224	0.01	0.03
P_2O_5	2.5084	0.21	3.16
PdO	0.0062	-	-
Pm_2O_3	0.0003	-	-
Pr_6O_{11}	0.0321	0.02	0.05
PuO_2	0.0076	-	-
Rb_2O	0.0005	-	-
RhO_2	0.0136	0.01	0.02
RuO_2	0.0759	0.05	0.12
SO_3	0.2164	0.14	0.33
Sb_2O_3	0.0001	-	-
SeO_2	0.0005	-	-
SiO_2	44.8770	42.08	48.10
Sm_2O_3	0.0267	0.02	0.04
SnO_2	0.0006	-	-
SrO	0.0269	0.02	0.04
Tc_2O_7	0.0021	-	-
ThO_2	3.5844	1.83	6.56
TeO_2	0.0028	-	-
TiO_2	0.9800	0.92	1.05
UO_2	0.5605	0.37	0.87
Y_2O_3	0.0177	0.01	0.03
ZnO	0.0010	-	-
ZrO_2	0.2943	0.19	0.45
Insolubles	0.0080	-	-

[a]Source: Eisenstatt 1986. Reference glass composition is WV-205.

3.3 SAVANNAH RIVER PLANT (SRP)

3.3.1 Introduction

Interim forms of high-level waste now in storage at SRP have been produced since 1954 by the reprocessing of defense reactor fuels. Neutralization and settling of the HLW have resulted in the formation of sludge and supernatant liquid. Subsequent evaporation of the supernatant liquid, which contains almost all of the Cs-137 activity, has produced a saturated salt solution and a saltcake consisting of the salts crystallized out of the saturated solution. Starting in 1990, the sludge and most of the radioactivity in the supernatant liquid 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 will be started in 1988; the saltstone is low-level waste and will go to onsite engineered storage.

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 saltcake phase; thus the washed precipitate will contain essentially 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 wall thickness of 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 (1100 lb). Each canister will contain 0.626 m^3 of glass, or about 1680 kg (3710 lb), when loaded to about 85% of its volume at an average glass temperature of 825°C. The density of the reference glass is about 2.7 g/cm^3 at this temperature; the density at 25°C is about 2.85 g/cm^3. The total weight of a loaded canister is therefore about 2180 kg (4810 lb), and the volume of glass in a loaded canister at 25°C, based on density ratio, would be about 0.59 m^3; however, the actual glass level in the canister is essentially unchanged (Kelker 1986; DPSP 80-1033, Rev. 91).

3.3.4 Inventory and Production Schedule

Table 3.3.2 shows a preliminary projection of glass production from 1990 to 2020 estimated by SRP for the 1987 Integrated Data Base submittal (Boore 1987). As shown in the table, the initial production of glass at SRP is scheduled to start in 1990, and it is estimated that about 102 canisters will be produced in that year. A total of about 6800 canisters will have been produced by the end of 2020. All canisters produced will be stored on site until a repository becomes available.

3.3.5 Radionuclide Content per Canister

Existing tanks at SRP contain blends of waste of all ages. From the standpoint of minimizing the radioactivity of the glass, it would be desirable to vitrify the waste in the oldest tanks first. However, this is not feasible because of practical contraints in the waste tank farms. Some of the tanks currently receiving fresh waste from the fuel reprocessing facilities will be nearly full by the time the DWPF begins operation. Since it is essential to have tankage space available to receive current production, it will be necessary to process some of the fresher waste first. It has not yet been possible for SRP to prepare a complete characterization of the feed to the vitrification plant in terms of an estimated schedule of radionuclide content versus time. It is clear, however, that the oldest waste will not be vitrified first.

Although present plans are to prepare the initial feed batch from sludge of lower activity, the activity of the cesium precipitate feed will probably be close to the DWPF flowsheet maximum because of processing constraints in the tank farm.

The radionuclide composition estimated by SRP 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 DPSP 80-1033, Rev. 91, 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 1682 kg (3710 lb) of sludge-precipitate glass at the reference-case fill level of 85%.

3.3.6 Radioactivity and Thermal Power

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 3.3.4 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.

Recent curie balances indicate that the glass produced during the first five years of operation will not exceed an activity of about 154,000 Ci/canister and a heat generation rate of about 460 W/canister (Baxter 1986). However, the detailed radionuclide composition of this glass will not be available until just before it is processed. Washed sludge for the initial feed blend has just been collected but has not been analyzed, and the washed precipitate for initial feed blending has not yet been produced. Estimates of the analysis of the first feed batch should be available by 1988 and about one year prior to feeding for subsequent feed batches.

Table 3.3.5 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. For example, at the end of year 2020 the total cumulative radioactivity in glass is 404.2×10^6 Ci. Dividing this by the total number of canisters produced (6810), the resulting average is 59,400 Ci/canister. The average thermal power, determined in the same way, is 169 W/canister. The SRP projections on which Table 3.3.5 was based were given in Chandler 1987. The reader is cautioned that these projections were not intended to represent actual processing schedules and tankage allocations and therefore should not be used to calculate radioactivity or thermal power per canister in any specific year. However, the long-term cumulative averages shown should be useful for repository calculations, since it is clear that the averages should give better estimates of overall heat loads than would be obtained by multiplying the total number of canisters by the maximum heat load per canister.

3.3.7 Chemical Composition

Table 3.3.6 shows the approximate chemical composition of a simulated average borosilicate glass from SRP (Chandler 1987).

3.3.8 Assessment of Data

The data in DPSP 80-1033, Rev. 91, 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 1682 kg, it appears that the maximum values of radioactivity and thermal power per canister will not exceed those shown in Table 3.3.4. For repository design and other purposes, it would be useful to have an estimated schedule of the radionuclide content of the vitrification plant feed as a function of the year of operation; but, as already indicated, such estimates will not be available until about one year before feeding to

the vitrification plant. It appears likely that the glass produced during the first 5 years of operation will not exceed about 154,000 Ci/canister and 460 W/canister. SRP is continually working to update waste treatment and vitrification process flowsheets, mass balances, and curie balances, so the estimates given here are subject to revision as new data become available.

3.3.9 References for Section 3.3

Baxter 1986. Telephone conversation, R. G. Baxter (SRP) and R. Salmon (ORNL), May 16, 1986.

Baxter 1987. Letter from R. G. Baxter, SRP, to Royes Salmon, ORNL, February 18, 1987.

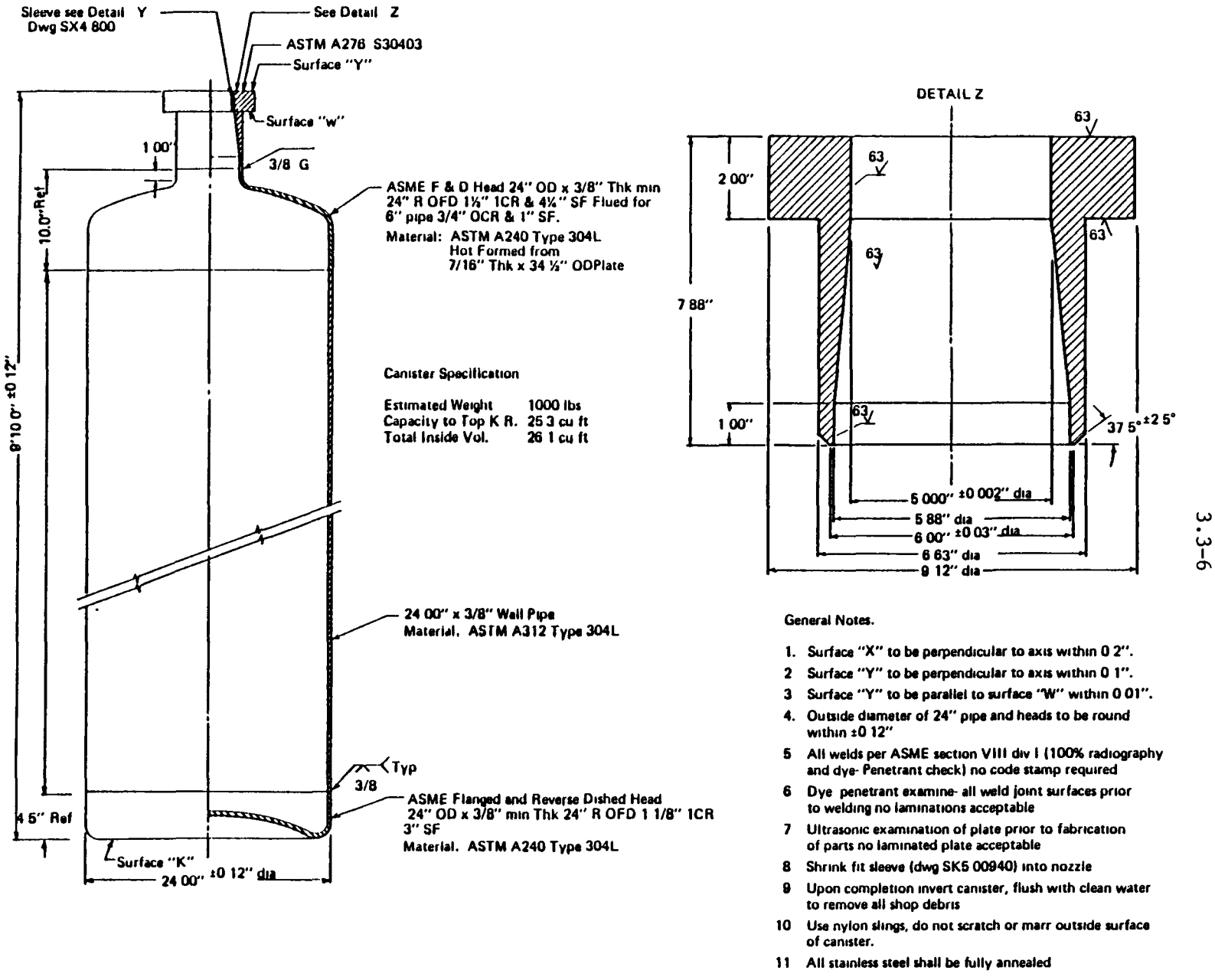
Boore 1987. Letter from W. B. Boore, SRP, to M. G. O'Rear, SRO, March 10, 1987.

Chandler 1987. Letter from R. L. Chandler to M. W. Shupe, HLW Lead Office, Richland, transmitting SRP input to DHLW Integrated Data Base, April 1, 1987.

DPSP 80-1033, Rev. 91. DWPF Basic Data Report, DPSP-80-1033, Rev. 91, April 1985.

Kelker 1986. J. W. Kelker, Jr., Development of the DWPF Canister Temporary Shrink-fit Seal, DP-1720, April 1986.

SRP 1987. Data transmittal at meeting at SRP, March 10, 1987.



General Notes.

1. Surface "X" to be perpendicular to axis within 0 2".
2. Surface "Y" to be perpendicular to axis within 0 1".
3. Surface "Y" to be parallel to surface "W" within 0 01".
4. Outside diameter of 24" pipe and heads to be round within ±0 12"
5. All welds per ASME section VIII div I (100% radiography and dye- Penetrant check) no code stamp required
6. Dye penetrant examine- all weld joint surfaces prior to welding no laminations acceptable
7. Ultrasonic examination of plate prior to fabrication of parts no laminated plate acceptable
8. Shrink fit sleeve (dwg SK5 00940) into nozzle
9. Upon completion invert canister, flush with clean water to remove all shop debris
10. Use nylon slings, do not scratch or marr outside surface of canister.
11. All stainless steel shall be fully annealed

Fig. 3.3.1. Savannah River Plant HLW canister. Source: DuPont report DP-1720, April 1986.

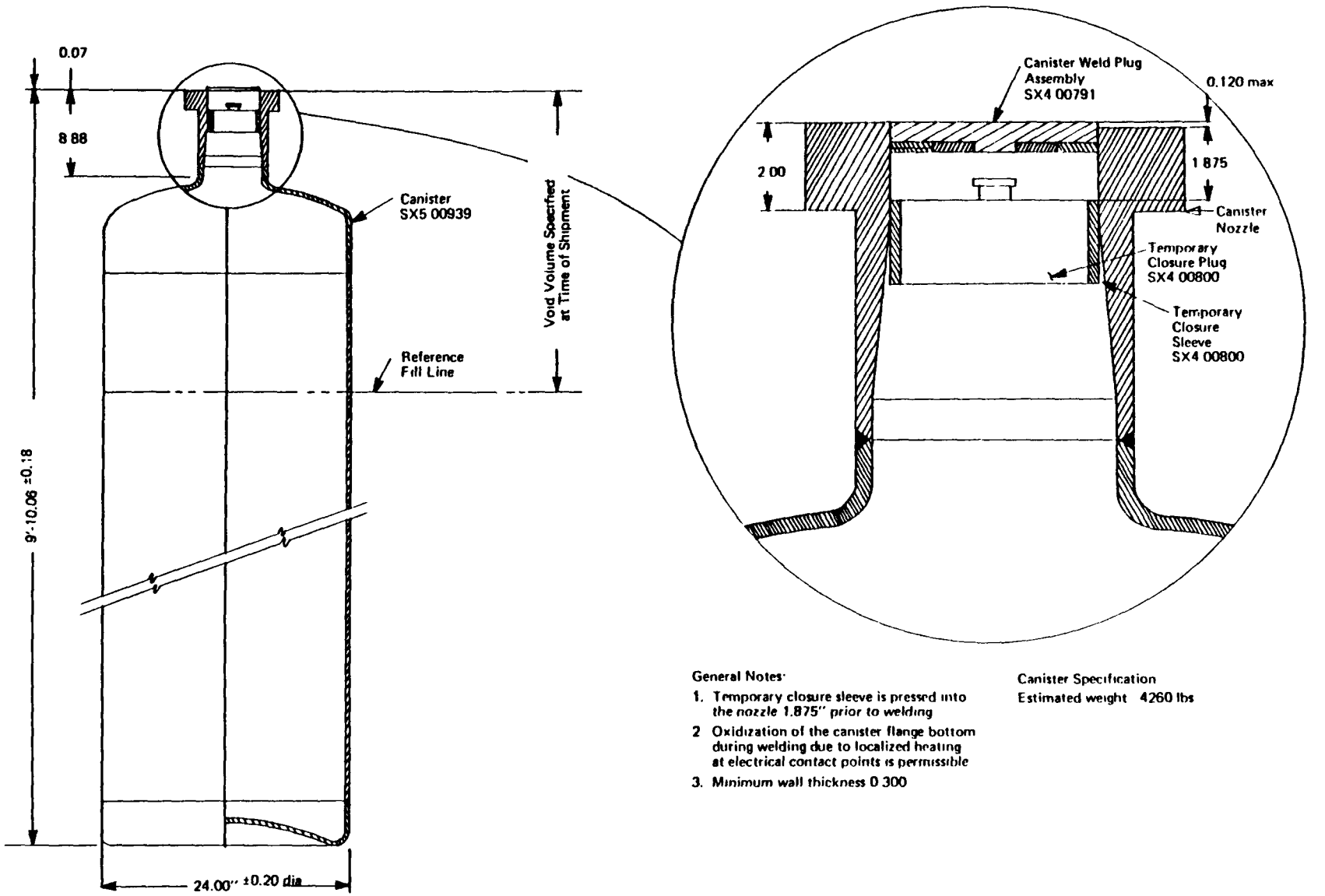


Fig. 3.3.2. Savannah River Plant HLW canister closure. Source: DuPont report DP-1720, April 1986.

Table 3.3.1. Savannah River Plant. High-level waste form and canister characteristics.[a]

	Canister 85% fill
Canister inside volume, m^3	0.736
Glass volume at average fill temperature (see note b), m^3	0.626
Glass density at average fill temperature (see note b), g/cm^3	2.69
Glass weight, kg	1,682
Canister weight, kg	500
Gross weight, kg	2,182
Total activity, curies	234,000[c]
Decay heat, watts	690[c]

[a]Source: DWPF Basic Data Report, DPSP 80-1033, Rev. 91, April 1985.

[b]The average fill temperature (i.e. the average temperature of the glass upon completion of filling to 85% of canister volume) is 825°C. The glass volume per canister when cooled to 25°C is about 0.59 m^3. The density of the glass is about 2.69 g/cm^3 at 825°C and 2.85 g/cm^3 at 25°C (SRP 1987).

[c]These figures are the ones given in DPSP 80-1033, Rev. 91. The corresponding figures calculated by ORIGEN2 are 234,400 Ci and 709W, as shown in Table 3.1. Activity and decay heat (thermal power) are at the time of filling the canister and are based on the maximum case, i.e. 5-yr old sludge and 15-yr old supernate.

Table 3.3.2. DWPF, Savannah River Plant.
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 $10^3 m^3$
1987	0	0	0
1988	0	0	0
1989	0	0	0
1990	102	102	0.06
1991	410	512	0.32
1992	410	922	0.58
1993	410	1332	0.84
1994	410	1742	1.10
1995	410	2152	1.36
1996	410	2562	1.62
1997	410	2972	1.88
1998	410	3382	2.14
1999	410	3792	2.40
2000	410	4202	2.66
2001	220	4422	2.80
2002	220	4642	2.94
2003	220	4862	3.08
2004	220	5082	3.22
2005	220	5302	3.36
2006	220	5522	3.50
2007	92	5614	3.56
2008	92	5706	3.62
2009	92	5798	3.68
2010	92	5890	3.74
2011	92	5982	3.80
2012	92	6074	3.86
2013	92	6166	3.92
2014	92	6258	3.98
2015	92	6350	4.04
2016	92	6442	4.10
2017	92	6534	4.16
2018	92	6626	4.22
2019	92	6718	4.28
2020	92	6810	4.34

[a]Production shown is based on a glass melt rate of 228 lb/hr and 75% attainment. Canisters (2-ft diameter x 10-ft long) are assumed to be filled to 85% capacity with a glass waste form incorporating 28 wt% waste sludge oxides, 8 wt% residues from waste salt, and 64 wt% oxides from a nonradioactive frit. Volumes reported are for the glass waste form and not the canisters. Source: Chandler 1987.

Table 3.3.3. Savannah River Plant. Radioisotope content per HLW canister[a]

	Isotope	Curies/canister	Grams/canister
1	Cr-51	0.9312E-16	0.1008E-20
2	Co-60	0.1699E+03	0.1502E+00
3	Ni-59	0.2397E-01	0.3163E+00
4	Ni-63	0.2975E+01	0.4824E-01
5	Ti-208	0.1128E-02	0.3829E-11
6	U-232	0.1339E-01	0.6256E-03
7	U-233	0.1584E-05	0.1636E-03
8	U-234	0.3428E-01	0.5485E+01
9	U-235	0.1573E-03	0.7278E+02
10	U-236	0.1128E-02	0.1742E+02
11	U-238	0.1050E-01	0.3122E+05
12	Np-236	0.1744E-07	0.1323E-05
13	Np-237	0.8904E-02	0.1263E+02
14	Pu-236	0.1221E+00	0.2297E-03
15	Pu-237	0.8941E-11	0.7401E-15
16	Pu-238	0.1484E+04	0.8667E+02
17	Pu-239	0.1291E+02	0.2076E+03
18	Pu-240	0.8681E+01	0.3809E+02
19	Pu-241	0.1670E+04	0.1620E+02
20	Pu-242	0.1224E-01	0.3206E+01
21	Am-241	0.1102E+02	0.3210E+01
22	Am-242	0.1436E-01	0.1776E-07
23	Am-242m	0.1447E-01	0.1488E-02
24	Am-243	0.5788E-02	0.2902E-01
25	Cm-242	0.3495E-01	0.1057E-04
26	Cm-243	0.5565E-02	0.1078E-03
27	Cm-244	0.1076E+03	0.1329E+01
28	Cm-245	0.6715E-05	0.3910E-04
29	Cm-246	0.5342E-06	0.1739E-05
30	Cm-247	0.6604E-12	0.7116E-08
31	Cm-248	0.6864E-12	0.1614E-09
32	Se-79	0.1699E+00	0.2439E+01
33	Rb-87	0.8719E-06	0.9961E+01
34	Sr-89	0.4267E-04	0.1470E-08
35	Sr-90	0.4675E+05	0.3426E+03
36	Y-90	0.4786E+05	0.8795E-01
37	Y-91	0.7568E-03	0.3085E-07
38	Zr-93	0.1117E+01	0.4443E+03
39	Zr-95	0.1005E-01	0.4680E-06
40	Nb-94	0.9646E-04	0.5147E-03
41	Nb-95	0.2115E-01	0.5407E-06
42	Nb-95m	0.1247E-03	0.3272E-09
43	Tc-99	0.3079E+01	0.1816E+03
44	Ru-103	0.1684E-07	0.5217E-12
45	Ru-106	0.2252E+04	0.6729E+00

Table 3.3.3 (continued)

	Isotope	Curies/canister	Grams/canister
46	Rh-103m	0.1636E-07	0.5028E-15
47	Rb-106	0.2259E+04	0.6346E-06
48	Pd-107	0.1473E-01	0.2863E+02
49	Ag-110m	0.1258E+00	0.2647E-04
50	Cd-113	0.5009E-13	0.1472E+00
51	Cd-115m	0.1213E-08	0.4763E-13
52	Sn-121m	0.7902E-01	0.1336E-02
53	Sn-123	0.2549E+00	0.3101E-04
54	Sn-126	0.4415E+00	0.1556E+02
55	Sb-124	0.7123E-07	0.4071E-11
56	Sb-125	0.8496E+03	0.8226E+00
57	Sb-126	0.6159E-01	0.7365E-06
58	Sb-126m	0.4415E+00	0.5619E-08
59	Te-125m	0.2760E+03	0.1532E-01
60	Te-127	0.1202E+00	0.4555E-07
61	Te-127m	0.1228E+00	0.1302E-04
62	Te-129	0.3053E-11	0.1457E-18
63	Te-129m	0.4749E-11	0.1576E-15
64	Cs-134	0.3372E+03	0.2606E+00
65	Cs-135	0.9943E-01	0.8633E+02
66	Cs-136	0.7828E-39	0.1068E-43
67	Cs-137	0.4341E+05	0.4989E+03
68	Ba-136m	0.8607E-38	0.3195E-49
69	Ba-137m	0.4155E+05	0.7724E-04
70	Ba-140	0.1024E-35	0.1404E-40
71	La-140	0.4304E-36	0.7734E-42
72	Ce-141	0.3591E-10	0.1260E-14
73	Ce-142	0.9609E-05	0.4005E+03
74	Ce-144	0.9869E+04	0.3093E+01
75	Pr-143	0.1198E-33	0.1780E-38
76	Pr-144	0.9869E+04	0.1306E-03
77	Pr-144m	0.1187E+03	0.6545E-06
78	Nd-144	0.4860E-09	0.4110E+03
79	Nd-147	0.1261E-43	0.1570E-48
80	Pm-147	0.2419E+05	0.2609E+02
81	Pm-148	0.6975E-10	0.4243E-15
82	Pm-148m	0.1009E-08	0.4722E-13
83	Sm-147	0.2000E-05	0.8796E+02
84	Sm-148	0.5788E-11	0.1916E+02
85	Sm-149	0.1781E-11	0.7420E+01
86	Sm-151	0.2478E+03	0.9418E+01
87	Eu-152	0.3688E+01	0.2132E-01
88	Eu-154	0.6196E+03	0.2295E+01
89	Eu-155	0.4749E+03	0.1021E+01
90	Eu-156	0.5231E-31	0.9489E-36
91	Tb-160	0.1120E-05	0.9923E-10
	Total	0.2344E+06	0.3427E+05

[a]Quantities shown are for sludge + supernate glass and are based on the DWPF Basic Data Report, DPSP 80-1033, Rev. 91, April 1985, assuming sludge aged an average of 5 years and supernate aged an average of 15 years, with a canister load of 3710 lb of glass (1683 kg).

Table 3.3.4. Savannah River Plant. Calculated radioactivity and thermal power per HLW canister.[a]

Decay time, years[b]	Radioactivity per canister (Ci)	Thermal power per canister (W)
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-yr cooled sludge and 15-yr cooled supernate. Calculations made by ORIGEN2 code based on data supplied by SRP (Basic Data Report, DPSP-80-1033, Rev. 91, April 1985). Canister is filled to 85% of capacity and contains 1683 kg of glass.

[b]Years after vitrification.

Table 3.3.5. Savannah River Plant. 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)
1987	0	0	0	0	0
1988	0	0	0	0	0
1989	0	0	0	0	0
1990	102	6.1	59,800	17.3	170
1991	512	27.0	52,700	78.6	154
1992	922	50.8	55,100	143.3	155
1993	1,332	74.2	55,700	208.4	156
1994	1,742	98.4	56,500	277.3	159
1995	2,152	127.9	59,400	359.9	167
1996	2,562	156.5	61,100	438.5	171
1997	2,972	171.7	57,800	482.7	162
1998	3,382	192.8	57,000	541.9	160
1999	3,792	214.3	56,500	602.6	159
2000	4,202	242.0	57,600	681.1	162
2001	4,422	249.0	56,300	701.6	159
2002	4,642	251.3	54,100	708.2	153
2003	4,862	256.9	52,800	724.8	149
2004	5,082	261.4	51,400	738.3	145
2005	5,302	268.5	50,600	759.7	143
2006	5,522	276.7	50,100	784.1	142
2007	5,614	277.9	49,500	787.6	140
2008	5,706	281.4	49,300	798.1	140
2009	5,798	283.8	49,000	805.2	139
2010	5,890	287.1	48,700	815.3	138
2011	5,982	287.7	48,100	817.1	137
2012	6,074	288.3	47,500	818.9	135
2013	6,166	288.9	46,900	820.7	133
2014	6,258	289.4	46,200	822.2	131
2015	6,350	291.1	45,800	825.4	130
2016	6,442	292.0	45,300	827.9	129
2017	6,534	292.6	44,800	829.6	127
2018	6,626	293.1	44,200	831.1	125
2019	6,718	293.7	43,700	832.9	124
2020	6,810	294.3	43,200	834.6	123

[a]Calculated from estimates given in Chandler 1987. 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. Radioactivity and thermal power shown are for fission products only. Radioactivity will be about 1% higher and thermal power about 6% higher when actinides are included.

Table 3.3.6. Savannah River Plant. Chemical composition of HLW glass[a]

Component	wt %
SiO_2	45.6
Na_2O	11.0
B_2O_3	10.3
Fe_2O_3	7.0
Al_2O_3	4.0
K_2O	3.6
Li_2O	3.2
FeO	3.1
U_3O_8	2.2
MnO	2.0
Other	8.0
Total	100.0

[a]Source: Chandler 1987.

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 preliminary conceptual design stage. Procurement and construction are scheduled to begin in 1989, and hot startup is scheduled for 1996. The plant will vitrify pretreated HLW in a borosilicate glass which will be cast into stainless steel canisters. Maximum use will be made of existing technology, such as that developed in the design of the Defense Waste Processing Facility at SRP.

3.4.2 Types of HLW Produced

Current plans are that the HWVP will produce vitrified waste of three different compositions, corresponding to three different feeds. These are known as neutralized current acid waste (NCAW), complexant concentrate (CC), and plutonium finishing plant waste (PFP). The NCAW has a much higher activity than the CC/PFP and may therefore be considered as the design basis feed for repository purposes. It is possible that the CC and PFP may be combined and run as a single feed. The two or three HLW borosilicate glasses from these operations are the major HLW forms with which the repository will be concerned. The only other HLW forms produced at HANF of possible interest to the repository are strontium and cesium capsules. These are discussed in Section 5.5 and in Appendix 3B.

It is assumed here that the HANF reference plan will be followed. In this plan, the single-shell tank wastes are not vitrified but are immobilized in place. The decision as to whether the single-shell tank (SST) wastes are vitrified is dependent on the outcome of the Environmental Impact Statement (EIS) process. If the SST wastes are vitrified, these canisters would also go to a repository.

3.4.3 Physical Description

The canisters are made of type 304L stainless steel pipe with an outside diameter of 61 cm and a length of 300 cm. Figures 3.4.1 and 3.4.2 are sketches of the canister and neck detail showing relevant dimensions. The canister is essentially identical to that planned for use at the Savannah River DWPF. Additional descriptive information on the canister and HLW glass is given in Table 3.4.1. 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) which is equivalent to a glass height of 2.3 m (7.5 ft). 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). A glass volume of 0.62 m^3 (22 ft^3) corresponds to a glass weight of 1650 kg (3630 lb) (White 1986). The total weight of the filled canister is approximately 2150 kg (4740 lb), assuming that the empty canister weighs 500 kg, in accordance with SRP's estimate.

3.4.4 Inventory and Production Schedule

Estimated annual canister production rates for the vitrified waste are shown in Table 3.4.2. The HWVP is planned to start up in FY 1996. The HWVP design throughput is 145 canisters per year. After every three years of HWVP operations, there is a six-month shutdown for melter change-out. Table 3.4.2 indicates half of a normal year's production in the years in which melter change-out occurs. Rockwell is currently planning a production of 930 canisters of vitrified NCAW, 580 canisters of complexant concentrate, and 350 canisters of plutonium finishing plant waste (Coony 1987).

3.4.5 Radionuclide Content per Canister

Radioisotopic data describing the composition of a canister of glass made from NCAW were supplied by Rockwell Hanford (White 1986). These data, shown in Table 3.4.3, were intended as the upper bound of activity and thermal power and represent the most active waste expected

to be fed to the vitrification plant; this type of feed would not be encountered before 1999 (Watrous, 1986). The radionuclide composition of the NCAW glass (upper bound case) given in the 1987 Integrated Data Base submittal (Coony 1987) is identical to that given in White 1986. Because of the upper-bound conservatism in both the values of maximum curies per canister and number of canisters, the product of these two values will be higher than the values for curies given in Coony 1987.

The current reference plan at HANF is not to produce any more Sr and Cs capsules; none of these have been produced since 1985. The Sr and Cs in the HLW will become part of the NCAW HLW glass. The radionuclide composition shown in Table 3.4.3 is based on this assumption. Currently there are no plans to vitrify any Sr and Cs capsules; current plans provide for enclosing these capsules in overpacks for repository emplacement, as described in Sect. 5.5. At present there are 640 Sr and 1576 Cs capsules.

3.4.6 Radioactivity and Thermal Power

Based on the upper-bound isotopic data supplied by Rockwell Hanford (Coony 1987), ORIGEN2 calculations were made to determine the estimated radioactivity and thermal power per canister of HLW glass made from NCAW. Table 3.4.4 shows the calculated radioactivity and thermal power per canister for decay times ranging from 0 to 10^6 years.

Because radionuclide compositions of the glasses produced from CC and PFP 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.

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

During the first three years of vitrification plant operation, it is not expected that the radioactivity of the NCAW glass will be as high

as that estimated for the upper bound case. At present the best information on the glass produced during initial operations is that decay heat loads per canister will be in the range of 400 to 800 watts rather than the 1150 watts shown for the upper-bound case (Mitchell 1986).

Hanford has also provided estimates of annual and cumulative radioactivities of the vitrified waste on a year-by-year basis from 1996 to 2020. These estimates are shown in Tables 3.4.5 and 3.4.6. Table 3.4.5 shows the average radioactivity per canister on an annual "as produced" basis, and Table 3.4.6 shows 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. The table shows a maximum of 324,000 Ci/canister for NCAW in 1996, about 20% less than the "upper bound" estimate of 416,500 Ci/canister. However, the radioactivities per canister shown for specific years should probably not be taken too literally since it is doubtful that actual melter feed batch scheduling could be projected accurately over the time span indicated. Complexant concentrate and plutonium finishing plant wastes appear to have radioactivities of about 220 and 60 Ci/canister, respectively.

3.4.7 Chemical Composition

The reference NCAW glass composition, designated HW-39, is shown in Table 3.4.7. Because of the radioactive nature of the waste, glass formulation and process development studies were conducted with a simulated or substituted NCAW. The elements that were substituted or deleted, as well as the glass frit composition, are indicated in the table. 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).

3.4.8 Assessment of Data

The upper-bound and average radioactivities of the NCAW glass have been established to the extent possible at the present time. Additional

information on the glass made from plutonium finishing plant waste and complexant concentrate would also be useful, since approximately 900 canisters of this glass will be produced; however, as indicated in Table 3.4.5, the radioactivity per canister is very low for these glasses (about 220 Ci/canister for CC waste and 60 Ci/canister for PFP waste). Maximum thermal power per canister of NCAW glass has been determined, but average thermal power has not. As a rough preliminary approximation, it could be assumed that the average and maximum thermal power are in the same ratio as the average and maximum radioactivities. However, this approximation is not recommended for long decay times because of changes in the relative importance of fission products and actinides.

3.4.9 References for Section 3.4

Coony 1986. Telephone conversation, M. R. Coony (Rockwell Hanford) and R. Salmon, ORNL, June 18, 1986.

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

Mitchell 1986. D. E. Mitchell, Hanford Waste Vitrification Plant, Preliminary Description of Waste Form and Canister, RHO-RE-SR-55P, August 1986.

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

White 1986. Letter from J. D. White, Richland Operations Office, to W. R. Bibb, DOE/ORO, dated July 3, 1986.

Wolfe 1985a. Personal communication, B. A. Wolfe, Rockwell Hanford Operations, to R. Salmon, ORNL, November 8, 1985.

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

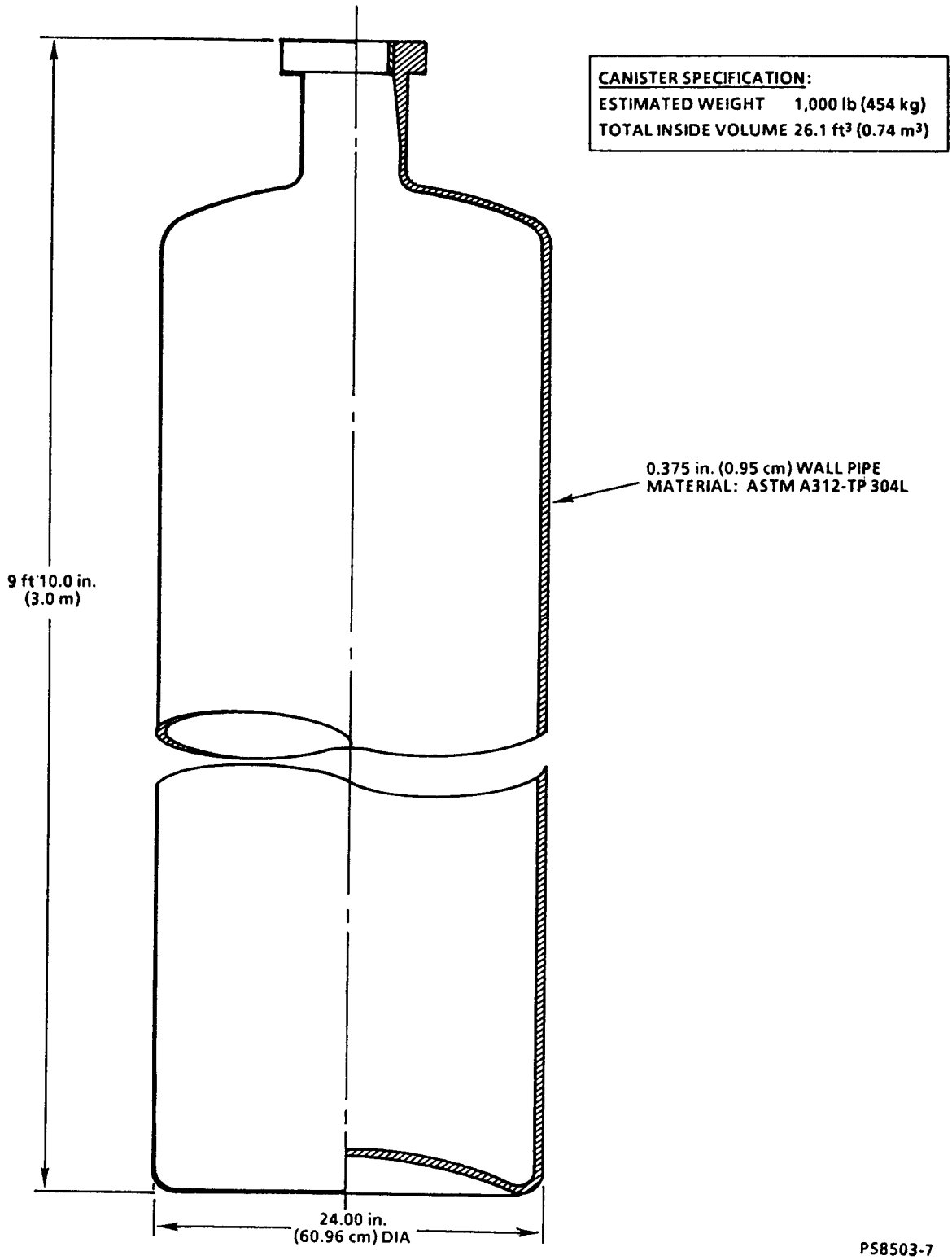


Fig. 3.4.1. Hanford HLW canister. Source: White 1986.

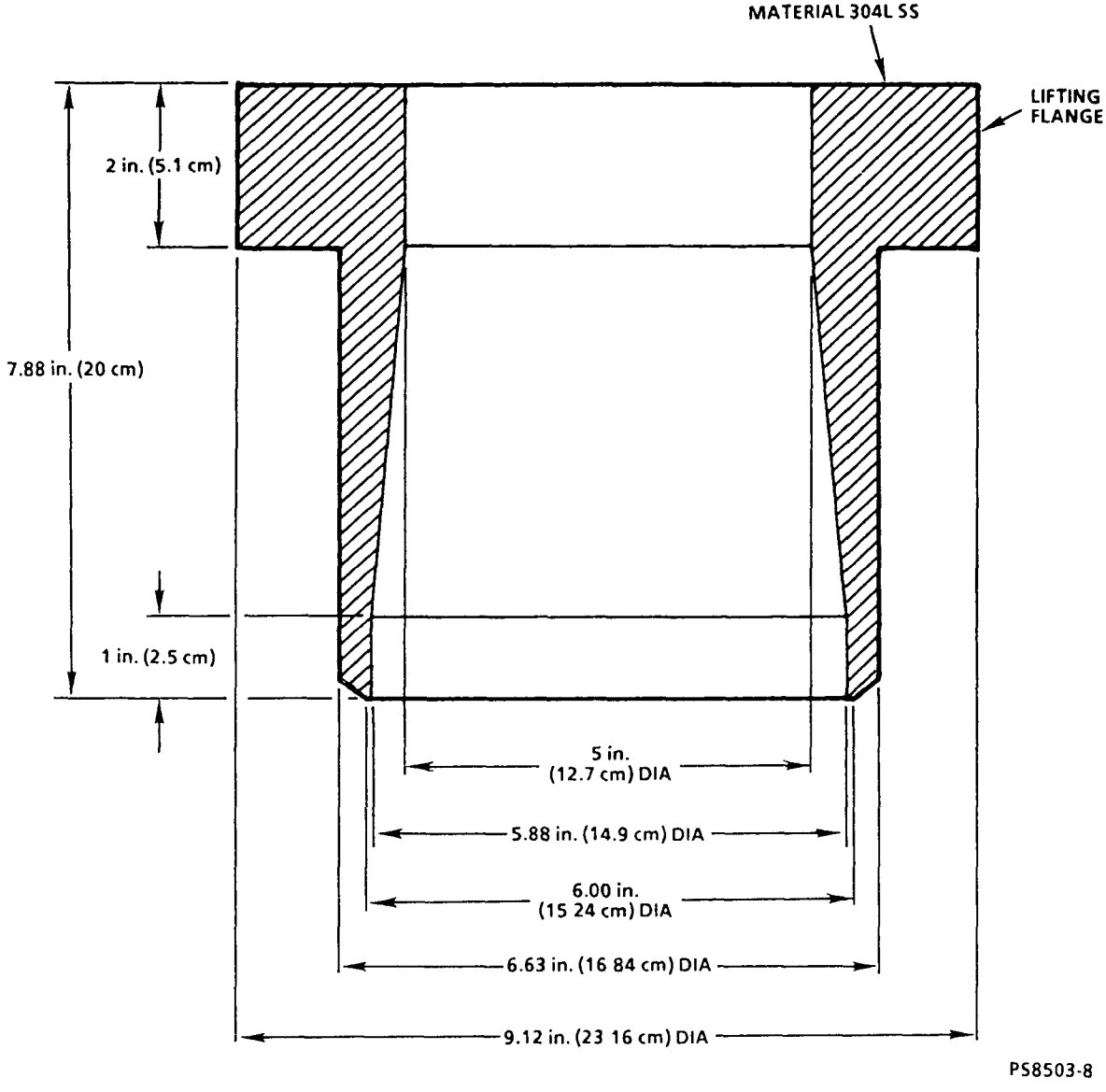


Fig. 3.4.2. Hanford HLW canister neck detail. Source: White 1986.

Table 3.4.1. Hanford Operations. High-level waste form and canister characteristics[a]

Waste form	Borosilicate glass in closed steel canister
Canister material	Type 304L stainless steel
Weights per canister	
Empty canister, kg	500
Borosilicate glass, kg	1650
Total loaded weight, kg	2150
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]	126,000 - 478,000
Thermal power, watts per canister[c]	354 - 1750

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

[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 1986 in which estimated activities and radionuclide compositions were given for four feeds typical of production during the period from 1996 to 2000. Radionuclide compositions are shown in Table 3.20.

Table 3.4.2. Hanford Operations. Estimated production schedule of canisters of HLW glass[a]

End of calendar year	Number of canisters produced during year	Cumulative number of canisters produced
1990	0	0
1991	0	0
1992	0	0
1993	0	0
1994	0	0
1995	0	0
1996	145	145
1997	145	290
1998	145	435
1999	73	508
2000	145	653
2001	145	798
2002	72	870
2003	145	1015
2004	145	1160
2005	145	1305
2006	73	1378
2007	145	1523
2008	145	1668
2009	72	1740
2010	120	1860
2011	0	1860
2012	0	1860
2013	0	1860
2014	0	1860
2015	0	1860
2016	0	1860
2017	0	1860
2018	0	1860
2019	0	1860
2020	0	1860

[a]Source: Coony 1987. It was assumed there that no fuel reprocessing takes place after CY 2001. If reprocessing does extend beyond 2001, each additional year of reprocessing would produce about 50 additional canisters.

Table 3.4.3. Hanford Operations. Radioisotope content per HLW canister (NCAW glass)[a]

	Isotope	Curies/canister	Grams/canister
1	C-14	0.9590E-01	0.2151E-01
2	Fe-55	0.1260E+02	0.5039E-02
3	Ni-59	0.1030E-01	0.1360E+00
4	Ni-63	0.2380E+01	0.3858E-01
5	Co-60	0.3580E+02	0.3166E-01
6	Zr-93	0.3430E-01	0.1365E+02
7	Nb-93m	0.7460E-02	0.2639E-04
8	In-113m	0.2870E-01	0.1716E-08
9	Sn-113	0.2870E-01	0.2858E-05
10	Sn-119m	0.2460E+02	0.5492E-02
11	Sn-121m	0.1310E+00	0.2215E-02
12	Sb-125	0.1100E+03	0.1065E+00
13	Te-125m	0.2690E+02	0.1493E-02
14	U-234	0.1750E-04	0.2800E-02
15	U-235	0.3250E-03	0.1504E+03
16	U-236	0.8190E-03	0.1265E+02
17	U-238	0.5880E-02	0.1749E+05
18	Np-237	0.3120E+00	0.4425E+03
19	Pu-238	0.4110E+00	0.2400E-01
20	Pu-239	0.3600E+01	0.5789E+02
21	Pu-240	0.1180E+01	0.5177E+01
22	Pu-241	0.3740E+02	0.3630E+00
23	Pu-242	0.6500E-04	0.1702E-01
24	Am-241	0.1030E+04	0.3000E+03
25	Am-242m	0.5960E+00	0.6131E-01
26	Am-243	0.4400E+00	0.2227E+01
27	Cm-242	0.1230E+02	0.3719E-02
28	Cm-244	0.9700E+00	0.1198E-01
29	C-14	0.9400E-04	0.2109E-04
30	Se-79	0.5260E+00	0.7550E+01
31	Sr-89	0.1150E-01	0.3961E-06
32	Sr-90	0.7310E+05	0.5357E+03
33	Y-90	0.7310E+05	0.1343E+00
34	Y-91	0.1280E+00	0.5218E-05
35	Zr-93	0.2440E+01	0.9709E+03
36	Zr-95	0.4910E+00	0.2285E-04
37	Nb-93m	0.1220E+01	0.4315E-02
38	Nb-95	0.1080E+01	0.2762E-04
39	Tc-99	0.1760E+02	0.1038E+04
40	Ru-103	0.1040E-03	0.3221E-08
41	Ru-106	0.5960E+04	0.1781E+01
42	Rh-103m	0.1040E-03	0.3196E-11
43	Rh-106	0.5960E+04	0.1674E-05
44	Pd-107	0.6960E-01	0.1353E+03
45	Ag-110	0.1120E-03	0.2686E-13

Table 3.4.3 (continued)

	Isotope	Curies/canister	Grams/canister
46	Ag-110m	0.8430E-02	0.1774E-05
47	Cd-113m	0.2240E+02	0.1033E+00
48	Cd-115m	0.1570E-05	0.6164E-10
49	Sn-119m	0.1980E+01	0.4421E-03
50	Sn-121m	0.1460E+00	0.2469E-02
51	Sn-123	0.6100E+00	0.7422E-04
52	Sn-126	0.8290E+00	0.2921E+02
53	Sb-124	0.6330E-05	0.3618E-09
54	Sb-125	0.2350E+04	0.2275E+01
55	Sb-126	0.1160E+00	0.1387E-05
56	Sb-126m	0.8290E+00	0.1055E-07
57	Te-123m	0.8890E-31	0.1002E-34
58	Te-125m	0.5750E+03	0.3192E-01
59	Te-127	0.5760E+00	0.2183E-06
60	Te-127m	0.5850E+00	0.6200E-04
61	Te-129	0.6170E-07	0.2945E-14
62	Te-127m	0.9810E-07	0.3256E-11
63	I-129	0.2140E-02	0.1212E+02
64	Cs-134	0.1360E+04	0.1051E+01
65	Cs-135	0.4650E+00	0.4038E+03
66	Cs-137	0.8730E+05	0.1003E+04
67	Ba-137m	0.8250E+05	0.1534E-03
68	Ce-141	0.1880E-05	0.6598E-10
69	Ce-144	0.1370E+05	0.4294E+01
70	Pr-143	0.6440E-21	0.9565E-26
71	Pr-144	0.1370E+05	0.1813E-03
72	Pr-144m	0.1960E+03	0.1080E-05
73	Pm-147	0.5270E+05	0.5685E+02
74	Pm-148	0.1100E-06	0.6692E-12
75	Pm-148m	0.2280E-05	0.1067E-09
76	Sm-151	0.1550E+04	0.5891E+02
77	Eu-152	0.3560E+01	0.2058E-01
78	Eu-154	0.4190E+03	0.1552E+01
79	Eu-155	0.6610E+03	0.1421E+01
80	Gd-153	0.4400E-02	0.1247E-05
81	Tb-160	0.2010E-04	0.1780E-08
Total		0.4165E+06	0.2273E+05

[a]Source: Coony 1987. Based on 1650 kg of HLW glass per canister. This is the upper bound case for HANF NCAW glass.

Table 3.4.4. Hanford Operations. Calculated radioactivity and thermal power per HLW canister.[a]

Decay time, years[b]	Radioactivity per canister (Ci)	Thermal power per canister (W)
0	416,500	1,159
1	373,600	1,034
2	346,400	964
5	300,300	861
10	257,000	759
15	226,200	676
20	200,600	604
30	158,900	484
50	100,000	314
100	32,000	117
200	4,040	33
300	1,110	22
350	813	20
500	530	16
1,000	240	7
1,050	224	6.6
2,000	73	1.6
5,000	31	0.18
10,000	29	0.15
20,000	27	0.12
50,000	24	0.08
100,000	22	0.08
500,000	13	0.11
1,000,000	9	0.11

[a]Calculations made by ORIGEN2 code based on data supplied by HANF (White, 1986). Canister is filled to 85% of capacity and contains 1650 kg of HLW glass made from neutralized current acid waste (NCAW). Data shown represent the "upper bound" case, i.e., the maximum expected activity.

[b]Years after vitrification.

Table 3.4.5. Hanford Operations. Estimated annual average radioactivity per canister of HLW glass[a]

End of calendar year	Annual number of canisters	Annual radioactivity curies	Annual average curies per canister
1995	0	0	0
1996	145	4.7E+07	324,100
1997	145	3.8E+07	262,000
1998	145	3.4E+07	234,500
1999	73	1.6E+07	219,200
2000	145	3.0E+07	206,900
2001	145	2.9E+07	200,000
2002	72	1.4E+07	194,400
2003	145	1.1E+07	75,900
2004	145	3.3E+04	228
2005	145	3.2E+04	221
2006	73	1.6E+04	219
2007	145	2.9E+04	200
2008	145	8.8E+03	61
2009	72	4.4E+03	61
2010	120	7.3E+03	61
2011	0	0	0
2012	0	0	0
2013	0	0	0
2014	0	0	0
2015	0	0	0
2016	0	0	0
2017	0	0	0
2018	0	0	0
2019	0	0	0
2020	0	0	0

[a]Calculated from Table 4 of Coony 1987. It was assumed there that no fuel reprocessing takes place after year 2001, and that if reprocessing continues after year 2001, each additional year of fuel reprocessing generates an equivalent of 50 canisters per year and an equivalent borosilicate glass activity of 7.3E+6 curies per year after CY 2010. It was also assumed that the neutralized current acid waste would be canistered first (930 canisters), then the complexant concentrate (580 canisters), and finally the plutonium finishing plant waste (350 canisters). Note that this table does not show the maximum radioactivity per canister, only the average.

Table 3.4.6. Hanford Operations. Estimated cumulative radioactivity and thermal power per canister of HLW glass[a]

End of calendar year	Cumulative number of canisters produced	Cumulative radioactivity and thermal power of HLW glass			
		Cumulative radioactivity		Cumulative thermal power[b]	
		Total (10^6 Ci)	per canister (Ci)	Total (10^3 W)	per canister (W)
1995	0	0	0		
1996	145	47	324,000	130	900
1997	290	76	262,000	212	730
1998	435	100	230,000	278	640
1999	508	110	217,000	305	600
2000	653	130	200,000	359	550
2001	798	160	200,000	439	550
2002	870	170	195,000	470	540
2003	1,015	170	167,000	470	460
2004	1,160	170	146,000	470	405
2005	1,305	160	123,000	450	345
2006	1,378	160	116,000	450	327
2007	1,523	150	98,000	420	276
2008	1,668	150	90,000	420	252
2009	1,740	140	80,000	390	224
2010	1,860	140	75,000	390	210
2011	1,860	140	75,000	390	210
2012	1,860	130	70,000	360	194
2013	1,860	130	70,000	360	194
2014	1,860	130	70,000	360	194
2015	1,860	130	70,000	360	194
2016	1,860	120	65,000	330	177
2017	1,860	120	65,000	330	177
2018	1,860	120	65,000	330	177
2019	1,860	110	60,000	310	167
2020	1,860	110	60,000	310	167

[a]Calculated from Table 4 of Coony 1987. It was assumed there that no fuel reprocessing takes place after year 2001, and that if reprocessing continues after year 2001, each additional year of fuel reprocessing generates an equivalent of 50 canisters per year and an equivalent borosilicate glass activity of 7.3E+6 curies per year after CY 2010. It was also assumed that the neutralized current acid waste would be canistered first (930 canisters), then the complexant concentrate (580 canisters), and finally the plutonium finishing plant waste (350 canisters). Note that this table does not show the maximum radioactivity per canister, only the average.

[b]Thermal power was estimated by ratio from radioactivity.

Table 3.4.7. Hanford Operations. Chemical compositions of HWVP reference HLW (NCAW), substituted NCAW, frit, and borosilicate glass[a]

Component	Reference NCAW waste composition wt %	Substituted NCAW waste composition[b] wt %	Frit composition wt %	Glass composition wt %
SiO_2	2.9	3.0	67.25	51.3
B_2O_3	0.0	0.0	12.75	9.6
Na_2O	10.5	10.7	10.25	10.4
Li_2O	0.0	0.0	5.0	3.8
CaO	0.3	0.3	3.75	2.9
MgO	0.2	0.3	1.0	0.8
Fe_2O_3	44.0	44.4	--	11.1
Al_2O_3	17.0	17.2	--	4.3
Cr_2O_3	5.3	5.3	--	1.3
ZrO_2	2.3	2.4	--	0.6
NiO	2.3	2.4	--	0.6
La_2O_3	2.2	2.2	--	0.5
SO_4	1.8	1.8	--	0.4
Nd_2O_3	1.7	2.1	--	0.5
MoO_3	1.2	1.2	--	0.3
F	1.2	1.2	--	0.3
CuO	0.6	0.6	--	0.1
TOC	0.6	0.6	--	--
MnO_2	0.6	0.7	--	0.2
CeO_2	0.6	0.7	--	0.2
RuO_2	0.6	0.6	--	0.1
U_3O_8	0.6	Sub Nd	--	--
Cs_2O	0.6	1.0	--	0.2
BaO	0.4	0.4	--	0.1
SrO	0.4	0.4	--	0.1
Pr_6O_{11}	0.4	0.4	--	0.1
Tc_2O_7	0.4	Sub Mn	--	--
Rb_2O	0.2	Sub Cs	--	--
Y_2O_3	0.2	0.2	--	0.04
Sm_2O_3	0.2	0.2	--	0.04
PdO	0.2	Del	--	--
Rh_2O_3	0.2	Del	--	--
NpO_2	0.1	Sub Ce	--	--
TeO_2	0.1	Del	--	--
Pm_2O_3	0.1	Sub Nd	--	--
BeO	0.1	Sub Mg	--	--
SeO_2	0.03	Del	--	--
SnO_2	0.02	Del	--	--
CdO	0.02	Del	--	--
Eu_2O_3	0.02	Sub Nd	--	--
PuO_2	0.02	Sub Ce	--	--
Am_2O_3	0.02	Sub Nd	--	--
P_2O_5	0.02	Del	--	--
Ag_2O	0.01	Del	--	--
Nb_2O_5	0.01	Sub Mo	--	--
Gd_2O_3	0.01	0.01	--	0.003
Ta_2O	0.01	Del	--	--
TiO_2	0.01	Del	--	--
Total	100	100	100	100

[a]Source: Mitchell 1986. Reference glass is HW-39. Data given are for a waste oxide loading of 25 wt% and are based on approximately 4-year old waste.

[b]Components marked sub were substituted as indicated. Components marked Del were deleted.

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 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 solidified by a fluidized-bed calcining process. The granular oxide 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 solidified by calcining, 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 (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, SRP, and HANF; that is,

61 cm diameter by 300 cm high. If glass-ceramic blocks are to be placed in such canisters, the canister would be designed with a wide-mouth opening and several blocks could be placed in each canister. Table 3.5.1 gives estimated physical characeristics 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; operation of the facility is scheduled to begin by 2011 (Berreth 1987). If the glass-ceramic form is chosen, the maximum rate of immobilized HLW production would be approximately 1000 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 3200 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). 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).

3.5.5 Radionuclide Content of Canister

Table 3.5.3 shows the estimated radionuclide composition of a canister based on the assumptions that the calcined HLW is converted to ceramic form and that each canister contains 1825 kg of ceramic, which

is the equivalent of 1277 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-10105, 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.

3.5.6 Radioactivity and Thermal Power

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 1987 Integrated Data Base submittal (Berreth 1987). The cumulative averages shown in the table were calculated from the IDB submittal based on the assumption that two-thirds of the canisters produced in a given year would be made from "fresh" calcine (actually aged 3 or more years), and the other one-third would be made from old calcine. 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 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 not be assumed that the compositions of the actual immobilized high-level wastes produced at INEL are 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 prior to 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 1987) give projections of total curies and watts for liquid waste and calcined waste inventories from 1987 to 2020. 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 the interim waste forms, nor on the compositions of glass or ceramic immobilized wastes made from the interim

wastes. Our estimates were based on an assumed radionuclide composition of 3-yr 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

Baker 1986. R. S. Baker, B. A. Staples, and H. C. Wood, Development of a Ceramic-Based Waste Form to Immobilize ICPP HLW, WINCO-1044, September 1986.

Berreth 1987a. Letter from J. R. Berreth, INEL, to Royes Salmon, ORNL, August 6, 1987.

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.

Berreth 1986a. Letter from J. R. Berreth, INEL, to J. E. Solecki, DOE, BERR-21-86, dated March 14, 1986.

Berreth 1986b. Telephone conversation, J. R. Berreth (INEL) and R. Salmon (ORNL), June 17, 1986.

Berreth 1986c. Letter from J. R. Berreth, INEL, to K. J. Notz, ORNL, December 1, 1986.

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

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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 closed canister
Canister material	stainless steel type 304L
Glass-ceramic density, g/cm^3	3.2
Weights per canister:	
Empty canister, kg	500
Glass-ceramic, kg	1825
Total loaded weight, kg	2325
Waste loading in glass-ceramic, wt%	70[b]
Glass-ceramic volume per canister, m^3	0.57[b]
Canister dimensions:	
Outside diameter, cm.	61
Height overall, cm.	300
Wall thickness, cm.	0.95
Radionuclide content, curies/canister	108,900[c]
Heat generation rate, watts/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 1277 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-10105 (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
2010	0	0
2011	500	500
2012	600	1,100
2013	700	1,800
2014	1,000	2,800
2015	1,000	3,800
2016	1,000	4,800
2017	1,000	5,800
2018	1,000	6,800
2019	1,000	7,800
2020	1,000	8,800

[a]This assumes that a glass-ceramic form (density 3.2 g/cm^3) is selected for HLW disposal and that each canister contains 1277 kg of calcine (1825 kg of glass-ceramic). Waste loading is 70 wt%. Canister production will continue after 2020 but is not shown. Source: Berreth 1987.

Table 3.5.3. Idaho National Engineering Laboratory. Radioisotope content per HLW Canister.[a]

	Isotope	Curies/canister	Grams/canister
1	Se-79	0.8173E-01	0.1173E+01
2	Rb-87	0.4597E-05	0.5252E+02
3	Sr-90	0.1660E+05	0.1217E+03
4	Y-90	0.1660E+05	0.3051E-01
5	Zr-93	0.3959E+00	0.1575E+03
6	Nb-93M	0.9577E-01	0.3387E-03
7	Tc-99	0.2682E+01	0.1582E+03
8	Ru-106	0.1239E+04	0.3701E+00
9	Rh-106	0.1239E+04	0.3479E-06
10	Pd-107	0.2554E-02	0.4965E+01
11	Sn-126	0.4086E-01	0.1440E+01
12	Sb-126M	0.4086E-01	0.5201E-09
13	Sb-126	0.4086E-01	0.4887E-06
14	Cs-134	0.4214E+04	0.3256E+01
15	Cs-135	0.9577E-01	0.8316E+02
16	Cs-137	0.1660E+05	0.1908E+03
17	Ba-137M	0.1532E+05	0.2848E-04
18	Ce-144	0.1047E+05	0.3282E+01
19	Pr-144	0.1047E+05	0.1386E-03
20	Pm-147	0.1532E+05	0.1653E+02
21	Sm-151	0.2171E+03	0.8250E+01
22	Eu-154	0.2299E+03	0.8513E+00
23	U-233	0.1532E-08	0.1583E-06
24	U-234	0.5491E-06	0.8785E-04
25	U-235	0.2299E-05	0.1063E+01
26	U-236	0.1277E-04	0.1973E+00
27	U-237	0.6130E-08	0.7507E-13
28	U-238	0.1277E-10	0.3797E-04
29	Np-237	0.6130E-04	0.8693E-01
30	Pu-238	0.8939E+02	0.5221E+01
31	Pu-239	0.8939E+00	0.1437E+02
32	Pu-240	0.8300E+00	0.3642E+01
33	Pu-241	0.2043E+03	0.1983E+01
34	Pu-242	0.2299E-02	0.6018E+00
35	Am-241	0.1162E+01	0.3385E+00
36	Am-243	0.1060E-01	0.5315E-01
37	Cm-242	0.8300E+00	0.2510E-03
38	Cm-244	0.6640E+00	0.8201E-02
	Total	0.1088E+06	0.8315E+03

[a]Quantities are at time of filling canister and are based on 3-yr old calcine immobilized in glass-ceramic with a load of 1277 kg of calcine per canister (1825 kg of glass-ceramic per canister). Based on IDO-10105 (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 1277 kg of calcine per canister (1825 kg of glass-ceramic per canister), with the initial radionuclide composition shown in Table 3.5.3.

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 and thermal power of HLW glass-ceramic			
		Cumulative radioactivity		Cumulative thermal power	
		Total (10^6 Ci)	Per canister (Ci)	Total (kW)	Per canister (W)
2010	0	0	0	0	0
2011	500	20	40,300	56	112
2012	1100	44	40,000	124	112
2013	1800	70	38,900	202	112
2014	2800	107	38,200	313	112
2015	3800	143	37,600	421	111
2016	4800	177	36,900	526	110
2017	5800	210	36,200	624	108
2018	6800	242	35,600	726	107
2019	7800	272	34,900	819	105
2020	8800	301	34,200	908	103

[a]Calculated from estimates given in Berreth 1987, using the assumptions that two-thirds of the glass-ceramic produced in a given year is made from fresh calcine, while the other one-third is made from old calcine, and that each canister contains 1277 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.

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
Ca	---	2-4	4	2-4
NO_3	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]Source: Staples, Knecht, and Berreth, 1986.

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

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.