QA: N/A

DOE/RW-0184-R1
Volume 1

CHARACTERISTICS OF POTENTIAL REPOSITORY WASTES

JULY 1992

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

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

DOE/RW-0184-R1
Volume 1

Characteristics of Potential Repository Wastes

July 1992

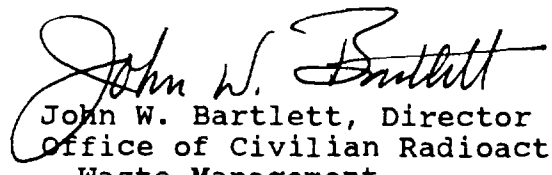
Prepared for the
U.S. DEPARTMENT OF ENERGY
OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
Washington, D.C. 20585

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

FOREWORD

This document, and its associated appendices and microcomputer (PC) data bases, constitutes the reference OCRWM data base of physical and radiological characteristics data of radioactive wastes. This Characteristics Data Base (CDB) system includes data on spent nuclear fuel and high-level waste (HLW), which clearly require geologic disposal, and other wastes which may require long-term isolation, such as sealed radioisotope sources. The data base system was developed for OCRWM by the CDB Project at Oak Ridge National Laboratory. Various principal or official sources of these data provided primary information to the CDB Project which then used the ORIGEN2 computer code to calculate radiological properties. The data have been qualified by an OCRWM-sponsored peer review as suitable for quality-affecting work meeting the requirements of OCRWM's Quality Assurance Program. The wastes characterized include: light-water reactor (LWR) spent fuel, immobilized HLW, non-LWR spent fuel, and miscellaneous wastes (such as wastes arising from the decommissioning of nuclear reactors and other facilities handling radioactive material).

The first edition of this report and data base system was published in 1987, and distributed to OCRWM personnel and contractors involved in waste acceptance, storage, transportation, disposal, and systems engineering work. The current version includes updated information and, by virtue of a formal peer review process involving 29 reviewers in seven technical areas, should provide significantly enhanced utility. Issuance to OCRWM personnel and OCRWM contractors is via controlled distribution. In addition, the reports and PC data bases are also available to other interested persons on request, on an uncontrolled basis.



John W. Bartlett, Director
Office of Civilian Radioactive
Waste Management

PREFACE

This first revision of the OCRWM Characteristics Data Base (CDB) of potential repository wastes includes a 4-year update of all time-dependent data, such as inventories, projected quantities, and planned schedules. It also incorporates significant improvements and enhancements in these areas:

- an improved LWR assembly classification scheme;
- more data on LWR assemblies, especially GE BWR assemblies;
- revised LWR radiological data, including specific inclusion of enrichment, newly recalculated effective cross sections, utility data on cycle- and down-times, built-in interpolation functions for burnup, enrichment, and decay times, and an improved method for integral heats (watt-years of decay heat over a period of time);
- another PC data base, for LWR assembly serial numbers;
- new activation factors for reactor hardware, based on recent experimental determinations;
- the addition of fuel pin data to the assembly data base;
- the addition of neutron source strength to the HLW data base;
- major revision to the Miscellaneous Waste Section, based largely on a recent study by the LLW lead site; and
- improved user interface for all of the PC data bases.

Note that this is a technical document. It does not address policy issues, nor should policy positions or institutional positions be inferred from these physical, chemical, and radiological data.

On the next page you will find an order form for the menu-driven PC data bases. Orders may be placed selectively or for all six. Both 5.25- and 3.5-inch diskettes are available.

Comments and suggestions are welcomed and should be addressed to one of the following:

Tien Nguyen, RW-321
U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Systems Engineering Branch
Washington, DC 20585

Karl J. Notz
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6495
(615) 574-6632

CHARACTERISTICS DATABASE SYSTEM
Order Form for
PC Databases

I have enclosed an UNOPENED* BOX OF PREFORMATTED, HIGH-DENSITY DISKETTES (3.5" or 5.25").

Please send the databases designated below. I understand that any unused diskettes will be returned to me.

____ LWR Radiological PC Database
____ LWR Quantities PC Database
____ LWR Assemblies PC Database
____ LWR NFA Hardware PC Database
____ LWR Serial Number PC Database
____ High-Level Waste PC Database

Name: _____
Title: _____
Program: _____
Organization: _____
Address: _____
City: _____ State: _____ ZIP: _____
Phone: _____ Fax: _____

Send request to:

Characteristics Database System
c/o Dr. Karl J. Notz
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6495

For Further Information
Phone Dr. Notz at (615) 574-6632 or R. Scott Moore at (615) 482-6601

* We are requesting unopened boxes of diskettes to help guard against the spread of computer viruses.

ACKNOWLEDGEMENTS

This document and the supporting data bases were prepared under the direction and sponsorship of the Systems Engineering Branch of the Office of Civilian Radioactive Waste Management. Significant data on LWR spent fuels were provided by DOE's Energy Information Administration. Supporting data were provided by DOE's Integrated Data Base Program, Pacific Northwest Laboratory, and other laboratories and contractors. The work was carried out by the Oak Ridge National Laboratory (ORNL) and the Oak Ridge office of Automated Sciences Group. The following made substantial contributions in the technical areas indicated:

LWR spent fuel and hardware classification
Automated Sciences Group, Inc.

LWR spent fuel quantities data
Energy Information Administration

LWR metal hardware activation
Pacific Northwest Laboratory

LWR fuel assemblies
Advanced Nuclear Fuels
Babcock & Wilcox
Combustion Engineering
Westinghouse Power Systems

High-level waste, defense and commercial
Idaho National Engineering Laboratory
Westinghouse Hanford Co.
Westinghouse Savannah River Co.
West Valley Nuclear Services Co.
Integrated Data Base Program

HTGR and TRIGA spent fuel
GA Technologies

Other non-LWR spent fuels
Integrated Data Base Program

GTCC low-level waste
EG&G/Idaho

PC menu-driven data bases
Automated Sciences Group, Inc.
Dataphile, Inc.

The necessary processing, compilation, evaluation, calculation, and tabulation of data were performed by ORNL staff and subcontractors as follows:

Karl J. Notz, Task Leader
Royes Salmon
Tim D. Welch
William J. Reich
R. Scott Moore (ASG)
Elizabeth A. Dorsey (ASG)
Kathy Jones (Dataphile)

Finally, QA qualification was achieved through a peer review process which involved 29 individual reviewers who collectively covered all technical areas, and whose active participation was essential to the successful culmination of this revision.

CONTENTS — VOLUME 1

LIST OF ACRONYMS

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

LIST OF ACRONYMS (continued)

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

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 (HLW) that will eventually be disposed of in a geologic repository. The purpose of this document, and the information contained in the associated computerized data bases and supporting technical reports, is to provide the technical characteristics of the radioactive waste materials that will (or may) be accepted by DOE for interim storage in an MRS or emplacement in a repository as developed under the Nuclear Waste Policy Act Amendment of 1987. Characterization data will be used by OCRWM for planning purposes, trade-off studies, system optimization, and conceptual design within the Yucca Mountain Site Characterization Project, the transportation program, the MRS program, overall systems integration, and development of waste acceptance criteria.

The primary sources of materials for a geologic repository are light-water reactor (LWR) spent fuel (either intact or consolidated and with associated activated metal) and immobilized HLW from West Valley and the three defense sites. These are the major sources in terms of both volume and radioactive content. Other sources are non-LWR spent fuel and certain miscellaneous wastes. Detailed characterizations are required for the materials in each of these four categories. This includes the intensive characteristics — physical, chemical, and radiological properties. The latter must take into account decay as a function of time. In addition, the extensive characteristics — inventories and projected quantities of the various wastes — are also included. This information is compiled and tabulated in a Characteristics Data Base (CDB) System, of which this document is a major element. The other elements are computerized data bases and various supporting technical documents. The data bases are set up as user-oriented, menu-driven PC data bases written in dBASE-III PLUS. There are presently six of these PC data bases. The supporting technical documents include reports that were prepared during the past three years to improve the content of this document and the PC data bases; these are listed at the end of this chapter.

The CDB serves as a unified, official source of information for the characterization of those materials that will (or may) become the responsibility of OCRWM for transport, storage, and final disposal. This includes clearly defined categories such as LWR spent fuel, other spent fuels, and immobilized HLW. It also includes less clearly defined materials such as Greater-than-Class-C (GTCC) low-level waste (LLW) for which some wastes are near the threshold value. A summary of the estimated quantities and thermal outputs for each of these categories is given in Table 1.1.

1.1.2 Report and Data Base Structure

The CDB System provides data in five formats: hard-copy reports, user-oriented PC data bases, program-level PC data files, mainframe computer files, and other program reports. This report is Revision 1 of the initial hard-copy report (DOE 1987, 1988). This revision includes a 4-year update on inventories and projections, and incorporates significant improvements in (a) classification of LWR assemblies (Moore 1988), (b) descriptions of GE assemblies (Moore 1989), and (c) the calculation of LWR radiological properties (Welch 1992), including the ability to interpolate on enrichment, burnup, and decay time. It also includes an additional PC data base for LWR assembly serial numbers (Reich 1991), new activation factors for assembly hardware based on new work at PNL (Luksic 1989), the addition of fuel pin data to the assembly data base, improved neutron source strength data in the HLW data base (Hermann 1992), and an improved user interface with all of the PC data bases. Other reports have also been issued, dealing with non-LWR spent fuels (Salmon 1990) and the distribution of LWR spent fuel characteristics (Reich 1991a).

The hard-copy reports (DOE 1987, 1988, and this report) provide basic waste characterization descriptions, as well as drawings that are not easily placed in computerized files. The computerized files contain systematic data too extensive to include in a paper report, the radionuclide compositions of each waste for multiple decay times, and the 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. There are six of these data bases:

- LWR Radiological Data Base. Contains radionuclide compositions, heat generation rates, curies and other information as a function of spent fuel type, enrichment, burnup, and decay time.
- LWR Assemblies Data Base. Contains physical descriptions of fuel assemblies and fuel pins.
- High-Level Waste Data Base. Contains physical and radiological descriptions of HLW, 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.
- LWR Serial Numbers Data Base. Contains the individual assembly serial numbers and can be cross-referenced to the other LWR data bases.

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. 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 enrichment, burnup, or decay time after discharge. Another interaction would be to couple a specific assembly (by serial number) with its assembly type (from the Assembly Data Base) and specific enrichment, burnup, and discharge date (from the Quantities Data Base), and obtain its radiological properties (from the Radiological Data Base).

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. These applications are normally performed in-house. Other program reports are prepared as needed to enhance the technical basis of the CDB System or to present selected data in an appropriate format. These are included among the references cited in Section 1.1.6.

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, and the national LLW and TRU waste programs.* The Characteristics Data Base System interacts constructively with these programs, utilizing their files when appropriate and making its data files available to them.

The Energy Information Administration (EIA) is the official source of utility data on LWR spent fuel. Their RW-859 data file provides extensive data which are obtained directly from the utilities on an annual basis. They also provide detailed projection data.

Primary data on HLW are obtained in cooperation with the Integrated Data Base (IDB) directly from the waste generators themselves: the West Valley Demonstration Project, the Savannah River Site (Defense Waste Processing Facility), the Hanford Reservation facilities, and the Idaho National Engineering Laboratory.

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, subcontracts were negotiated with Babcock & Wilcox, Combustion Engineering, Exxon (now Advanced Nuclear Fuels Corporation), GA Technologies, and Westinghouse. Data for GE fuels were obtained from public documents and NRC dockets.

The 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, LLW, remedial action wastes, and mill tailings, in addition to spent fuels and HLW. A report is issued annually by the IDB (DOE 1991).

1.1.3.2 Data Processing System

The CDB System processes data at three levels: user-oriented PC files, program-level PC files, and mainframe files. The initial data, when received (or generated), are entered in 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 isotopes. These are generated in reactors from nuclear fission (fission products), activation of the lighter isotopes (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 is done using the ORIGEN2 code. The appendix to this volume gives a brief overview of ORIGEN2 and lists the input parameters that were employed in this work. These parameters were selected on the basis of rather extensive sensitivity tests (Welch 1992). Data output obtained from use of this code includes:

- quantity of each nuclide (grams or gram-atoms);
- radioactivity, total and by nuclide;
- alpha radioactivity, total and by nuclide;
- thermal power, total and by nuclide;

*Acronyms are defined on pp. ix and x.

- 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. The depletion portion models natural decay. This has been done by using generic BWR and PWR models, plus standard and extended burnup ranges for each. Thus, four sets of effective cross sections are required. The recently published set (Ludwig 1989) was employed. This is a major improvement over the earlier publication, but it should be noted that the resulting values did not change significantly, except for an increase in the BWR neutron source. In particular, the total thermal power was not altered, except for the BWR at high burnup, nor were the major fission products. Another major improvement was in scaling of enrichment to match the burnup, based on actual utility data (Welch 1992). This caused a significant change in the content of higher actinides and therefore in the neutron emission power. The end result is to decrease some neutron emission rates. Thus, the former values are conservatively high in some regions. Other improvements were in the modeling of power level, cycle time, and downtime between cycles. Again, these were based on actual utility data (Welch 1992), and differ between BWRs and PWRs, and for standard cycles and extended cycles. However, these improvements did not greatly affect the computed results for decay times of five years or greater (for short decay times, these factors are significant for some nuclides).

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

ORIGEN2 can calculate decayed values to any desired time. We have selected a spectrum of times out to one millions years, using a logarithmic spread (1, 2, 5, 10; etc.), with additional values during the first thousand years. These values are in the PC data bases. If other decay times are needed, an interpolation function was developed and is incorporated in the PC data bases. Likewise, interpolation functions were developed for enrichment and burnup, and are incorporated in the PC data bases. All of these are menu-driven for the user's convenience.

1.1.4 Menu-Driven PC Data Bases

There are six 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 user's guides published previously (DOE 1987, 1988). Much of the tabular data presented in this volume and in the more detailed supporting technical documents were taken selectively from these PC data bases. These data bases may be ordered on either 5.25 or 3.5 inch diskettes; see page v of this volume for instructions. The PC data bases are estimated to contain about five million data entries; about 1% of these are presented in the hard copy reports.

1.1.5 QA Program

This work was originally done under the overall requirements of NQA-1, as it was interpreted to apply to data and software under the original 18 sections. A QA assessment/evaluation was done in 1987 and a QA plan written at that time. That plan focused on operational procedures and has been followed since it was written. This plan was submitted to OCRWM for review but never officially approved. More recently, additional guidance has been provided by DOE/HQ in the form of Procurement QA Controls Specifications for OCRWM-Managed Contractors (DOE 1990), as directed by QAAP 4.2 (DOE 1989), under the OCRWM QA Requirements document (DOE 1989a). The primary effects of the recent (graded) guidance pertinent to this data base have been to add peer review and controlled distribution requirements. The peer review was carried out under an OCRWM-approved Peer Review Plan (ORNL 1991).

1.1.6 Supporting Technical Documents

During preparation of the original CDB report, it became obvious that additional work in certain technical areas would be highly beneficial. This was done and the various studies were documented as other publications by this program. The results of this work were incorporated in this revision. A brief synopsis of these publications is given below, listed in chronological order.

"A Classification Scheme for LWR Fuel Assemblies" (Moore 1988). This report documents the class and model concept, where class is determined by reactor core configuration, and a variety of models are usually applicable to a given class. The nomenclature has been fully coordinated with the EIA's RW-859 form.

"Physical Characteristics of GE BWR Fuel Assemblies" (Moore 1989). Descriptive details for GE-manufactured assemblies, based on various literature sources, including NRC dockets.

"Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal" (Luksic 1989). A report of actual measurements of the activation of various assembly components, in different neutron density zones, and a comparison of these measurements to values calculated using ORIGEN2.

"Non-LWR and Special LWR Spent Fuels: Characteristics and Criticality Aspects of Packaging and Disposal" (Salmon 1990). A preliminary estimate of the size and number of canisters required for this waste category, including consideration of the neutron poisoning

requirements for fuels that are often highly enriched and with a low burnup.

"Distribution of Characteristics of LWR Spent Fuel" (Reich 1991a). A comparison of the major features of present and future LWR assemblies versus the ATM (Approved Test Materials) assemblies examined in detailed hot cell studies by PNL.

"Analysis of Assembly Serial Number Usage in Domestic Light-Water Reactors" (Reich 1991). A quantitative examination of the serial numbers of the 70,971 assemblies permanently discharged at that time, to define the necessary conditions so that each serial number is, in fact, unique.

"Borosilicate Glass (α,n) Sources Used with ORIGEN-Type Calculations" (Hermann 1992). Improved procedure for calculating neutron source strength of HLW glass from the interaction of alpha particles with target material in the glass itself.

"ORIGEN2 Sensitivity to Enrichment and Other Factors" (Welch 1992). Determined that enrichment must be properly matched to burnup to obtain correct calculated compositions, especially for the higher actinides; also tested other input variables.

1.1.7 References for Section 1.1

DOE 1987. *Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation*, Volumes 1-6, DOE/RW-0184, December 1987.

DOE 1988. Ibid., Volumes 7-8, June 1988.

DOE 1989. *OCRWM QA Administrative Procedures*, DOE/RW-0197 (February 1989 and updates).

DOE 1989a. *OCRWM QA Requirements Document*, DOE/RW-0214 (February 1989 and updates).

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

Hermann 1992. O. W. Hermann and R. Salmon, *Borosilicate Glass (α, n) Sources used with ORIGEN-Type Calculations, Proceedings*, 1992 International High-Level Radioactive Waste Management Conference, Las Vegas, April 1992.

Ludwig 1989. S. B. Ludwig and J. P. Renier, *Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code*, ORNL/TM-11018, December 1989.

Luksic 1989. A. Luksic, *Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal*, PNL-6906, Vol. 1, June 1988.

Moore 1988. R. S. Moore, D. A. Williamson, and K. J. Notz, *A Classification Scheme for LWR Fuel Assemblies*, ORNL/TM-10901, November 1988.

Moore 1989. R. S. Moore and K. J. Notz, *Physical Characteristics of GE BWR Fuel Assemblies*, ORNL/TM-10902, June 1989.

ORNL 1991. *Peer Review Plan for Revision 1 of DOE/RW-0184*, SI-PR-001, February 15, 1991.

Reich 1991. W. J. Reich and R. S. Moore, *Analysis of Assembly Serial Number Usage in Domestic Light-Water Reactors*, ORNL/TM-11841, May 1991.

Reich 1991a. W. J. Reich, R. S. Moore, and K. J. Notz, *Distribution of Characteristics of LWR Spent Fuel*, ORNL/TM-11670, January 1991.

Salmon 1990. R. Salmon and K. J. Notz, *Non-LWR and Special LWR Spent Fuels: Characteristics and Criticality Aspects of Packaging and Disposal*, ORNL/TM-11016, January 1990.

Welch 1992. T. D. Welch, K. J. Notz, and R. J. Andermann, *ORIGEN2 Sensitivity to Enrichment and Other Factors*, ORNL/TM-11333, in preparation.

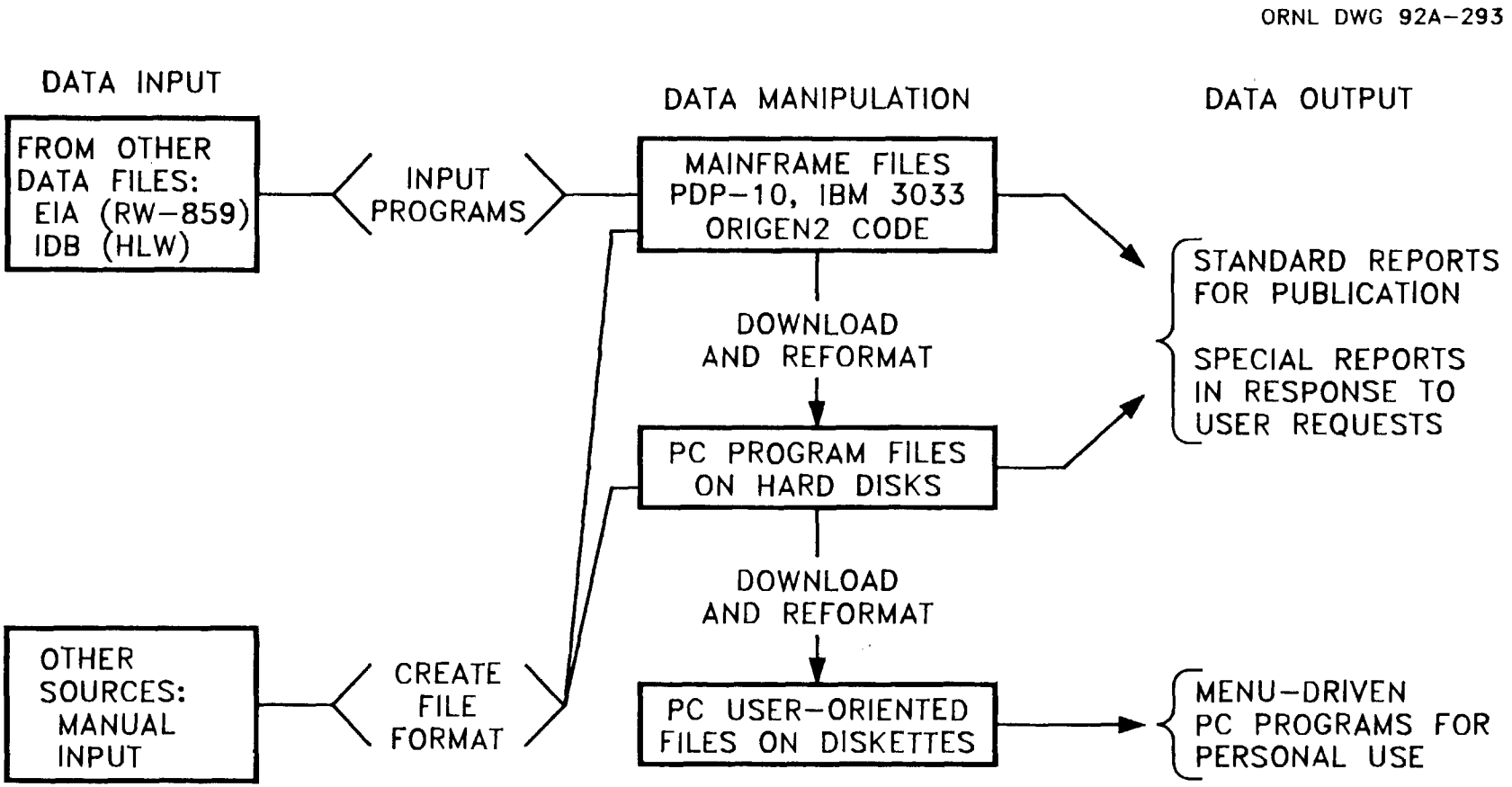


Fig. 1.1. Data processing in the Characteristics Data Base System.

Table 1.1. Estimated projected quantities to 2020 and thermal outputs, by source category

Source category	Number of canisters[a]	Watts per canister	Total megawatts
LWR spent fuel	38,000	1,500-3,500	100
Immobilized HLW	15,000	340-870	9
Non-LWR spent fuel	1,400	50-200	0.2
GTCC hardware	6,000	10-200	0.6
Total	60,400	—	110

[a]For LWR spent fuel, based on a hypothetical canister containing 4 BWR assemblies and 3 PWR assemblies. For immobilized HLW, see Chapter 3. For non-LWR spent fuel, see Chapter 4 and Salmon 1990. For GTCC hardware, based on a hypothetical canister containing 1 m^3 and projected to 2035.

1.2 LWR SPENT FUEL (see Chapter 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 non-fuel 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. These are aggregated into assembly classes, which are based on reactor core configuration. Radiological data are based on burnup and enrichment (for the fuel itself) or activation of materials of construction (for SFD and NFA hardware). Some LWR spent fuels may require special handling (defective fuel with gross defects and special fuel forms). A flow chart of LWR spent data is given in Fig. 1.2.

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 is provided. Each type of assembly is also characterized in terms of inventory-related information, such as the manufacturer, 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 enrichment and burnup for BWRs and PWRs.

The detailed assembly data are coupled with special activation calculations made with ORIGEN2 to estimate the radioactivities of the various SFD hardware components. The results provide a basis for classifying these components in terms of the LLW categories, particularly with regard to the greater-than-class-C category (GTCC). For hardware with a GTCC radioactivity classification, the radioactivity is also reported as a multiple of the Class C limit. The estimated masses and volumes deriving from these components are also calculated.

Fuel performance data and records were reviewed to identify, describe, and categorize various classes of defective fuel. This includes leakers and any damaged fuel that has been repackaged or encapsulated. Special fuels include those that require special treatment, such as fuel from Three Mile Island Unit 2.

1.2.2 Assemblies

Detailed descriptive material was tabulated for over 120 specific assembly models, which have been grouped into 23 classes (Tables 1.2 and 1.3). The data items listed in Table 1.4 were collected for each model, to the extent possible. These data were then incorporated in "Physical Descriptions of LWR Fuel Assemblies" (see Appendix 2A) and in a user-oriented data base (the LWR Assemblies Data Base). 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 and Projections

Inventories and projections are provided by the EIA and are incorporated in the LWR Quantities Data Base. A closely related PC data base, the LWR Serial Numbers Data Base, was added since the earlier publication.

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. The latest figures from EIA, which issues annual projections of installed nuclear generating capacity, are based on the no-new-orders case (EIA 1991):

Year	GW(e)	Year	GW(e)	Year	GW(e)
1991	100	2001	105	2011	99
1992	101	2002	105	2012	94
1993	102	2003	106	2013	86
1994	102	2004	106	2014	77
1995	103	2005	106	2015	71
1996	103	2006	106	2016	65
1997	103	2007	106	2017	62
1998	104	2008	106	2018	59
1999	104	2009	104	2019	59
2000	105	2010	101	2020	57

Based on this case, the quantities of spent fuel discharged in 2020 are projected to be as follows:

	Number of assemblies		Weight, MTIHM	
Reactor type	Annual rate	Cumulative	Annual rate	Cumulative
BWR	2,000	145,900	400	26,200
PWR	1,700	115,600	800	49,700
Totals	3,700	261,500	1,200	75,900

The EIA also makes other projections, e.g., the lower and upper reference cases, which allow for the possible addition of new nuclear plants. However, these new plants cannot be brought on-line for many years, and will thus have only a minor impact on the total accumulated fuel by 2020, but they would have a major impact on the annual discharge rate at that time. They could thus significantly increase the total inventory on-hand after 2020. Another variant among the EIA scenarios is one that allows for plant life extension of 20 years, which would delay the decline in fuel discharge rates starting in 2009. Under this scenario, the last fuel could be discharged as late as 2055.

1.2.4 Radiological Properties

Radiological characteristics, on an MTHM basis, were calculated using ORIGEN2 and are tabulated in the LWR Radiological Data Base. Spent fuel is characterized in terms of reactor type (PWR or BWR), burnup (from 10 to 60 GWd/MT for PWRs and 7.5 to 50 GWd/MT for BWRs), and decay times (from 1 to 1,000,000 years, in 24 increments). The types of radiological data provided were listed earlier, in Sect. 1.1.3.3.

The radiological properties were recalculated this year, using new cross-section data for ORIGEN2 and adjusting the enrichments to match the burnups based on actual utility data. These changes had the greatest effects on actinide content and neutron source strength values at higher burnups. Cycle details were also redefined to better match actual utility operational experience. In addition, interpolation capabilities were incorporated, to allow interpolation on burnup, enrichment, and decay time.

1.2.5 Defective Fuel

This category, although not 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 spent fuel. Examination of the major data sources for this category indicates that the three categories defined in 10 CFR 961 provide a useful basis. These categories are visual damage (F-1), radioactive leakage (F-2), and encapsulated (F-3). Very few assemblies have been classed as F-1 or F-3. Leakage can be minor (e.g., pin-hole gas leaks) or gross (e.g., extreme cladding degradation with loss of fuel). The latter presents an obvious handling problem.

Defects generally result from waterside corrosion or crud buildup, pellet-clad interaction (PCI), radiation-induced stressing, vibration-induced 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 extremely rare. On occasion, a utility may seal a leaker (or broken rod pieces) into another tube (encapsulation). Leakers are usually noted by a utility by increased radioactivity during operation, and are then identified by poolside examination. 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 only 0.01 to 0.02% or less. Those assemblies containing leakers have an average of about one or two failed rods per assembly. Thus, approximately 1 to 2% of the assemblies contain fuel rods with some cladding defects; most of these may not require special handling.

1.2.6 Special Fuel Forms

This category is for LWR fuels that are distinctive in some special way and may, therefore, require special handling. This includes consolidated fuel; 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. Fuel which has been reprocessed is also included here, in order to allow closure on a material balance of all discharged fuel.

1.2.7 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. 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 LLW Class C. However, some will be GTCC LLW and therefore not be eligible for shallow-land burial. The NRC has ruled that GTCC LLW must be sent to a deep geologic repository unless alternatives are approved. To fully characterize this material requires the following information:

- the composition of the alloy,
- 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 only for cruciform bearings in the past and no longer used):

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

Very recently, Westinghouse has introduced Zirlo cladding for high burnup fuel; this variant of Zircaloy has a higher niobium content.

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 (the reference zone), and
- bottom end plate region.

The flux decreases significantly in the two zones adjacent to the core zone and falls off even more beyond that. The effective cross sections outside the core zone may increase significantly, depending on the element (Co, Ni, Nb, or N), the zone, and the reactor type. This increase is presumably due to a higher fraction of thermalized neutrons outside the core zone.

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

- standard (30 GWd/MT for both BWR and PWR); and
- high (50 GWd/MT for BWR; 60 GWd/MT for PWR).

These burnups will provide limiting values for activation. Additional burnups can 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 of each SFD hardware component. This value is then compared with the Class C limit and a factor calculated.

In addition to those nuclides that control the GTCC classification, the Co-60 content is also important because it is a source of a very energetic gamma radiation. The bottom end fitting, in particular, can have troublesome amounts of Co-60.

1.2.8 Non-Fuel 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; and 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.

1.2.9 References for Section 1.2

EIA 1991. U.S. DOE, Energy Information Administration, Nuclear Fuel Data Form RW-859, data as of December 31, 1990 (tabulated data available in August 1991).

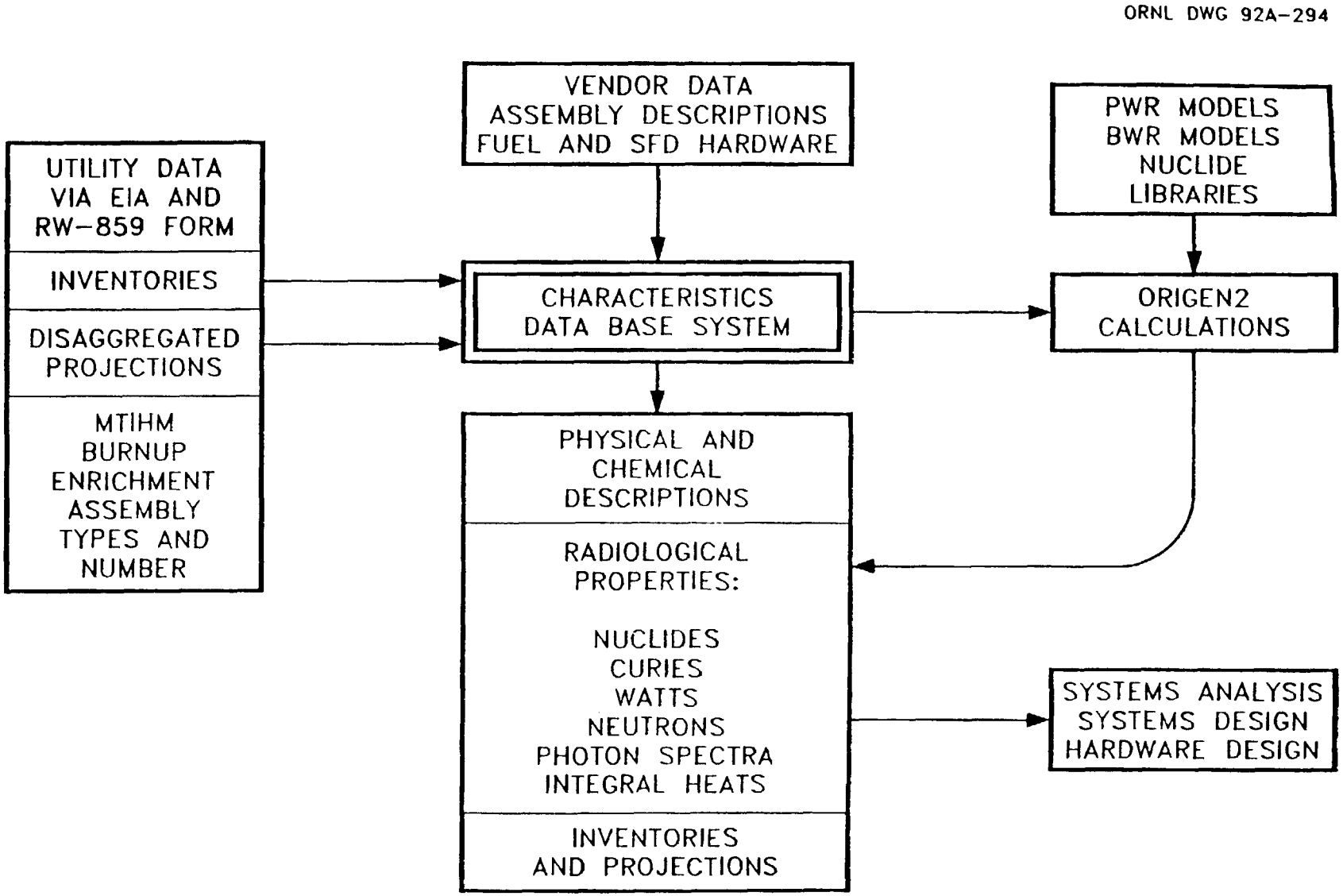


Fig. 1.2. Data flow for LWR spent fuel.

Table 1.2. BWR assembly classes

Class	Reactors using	Assembly types
Multi-reactor classes		
GE BWR/2, 3	9	18
GE BWR/4, 5, 6	27	19
Single-reactor classes		
Big Rock Point	1	8
Dresden 1	1	9
Elk River	1	1
Humboldt Bay	1	4
Lacrosse	1	2

Table 1.3. PWR assembly classes

Class	Reactors using	Assembly types
Multi-reactor classes		
B&W 15×15	8	12
B&W 17×17	3[a]	1
CE 14×14	5	3
CE 16×16	4	2
CE System 80	3	1
WE 14×14	6	7
WE 15×15	10	7
WE 17×17	32	7
South Texas	2	1
Single-reactor classes		
Fort Calhoun	1	3
Haddam Neck	1	7
Indian Point	1	2
Palisades	1	2
St. Lucie 2	1	1
San Onofre 1	1	2
Yankee Rowe	1	4

[a]Not yet in operation.

Table 1.4. Technical data for each fuel assembly model

- Fuel assembly
  - Designation
  - Width
  - Length
  - Total weight
  - Weight of heavy metal
  - Number of fuel rods
  - Rod pitch
- Fuel rods
  - Diameter
  - Length
  - Clad material
  - Clad thickness/weight
  - Spring material/weight
  - Heavy-metal content
  - Burnable poison/weight
  - Fill gas used
  - Initial rod pressurization
  - Pellet description
- Assembly hardware
  - Incore hardware
    - Grids, spacers, guide tubes
    - Material/weight
  - Top end fittings
    - Nozzles, springs, material/weight
  - Bottom end fittings
  - Channels
  - Other peripheral or special hardware
- Inventory information
  - Number of assemblies fabricated
  - Serial numbers
- Fuel performance
  - Enrichment (range)
  - Maximum design burnup
  - Linear heat rating

1.3 HIGH-LEVEL WASTE (see Chapter 3)

1.3.1 Scope

This section includes HLW from domestic fuel reprocessing plants, both commercial and defense-related. The ultimate waste forms for repository disposal are individual canisters of immobilized HLW, which are characterized by site (West Valley, SRS, Hanford, Idaho), by radionuclide composition per canister, and by a year-by-year schedule of the number of canisters produced at each site. Canister compositions on a year-by-year basis generally cannot be assigned yet because detailed composition-specific immobilization schedules have not yet been defined. The compositions that are given represent estimated maximum radioactivity and thermal power per canister for each site. By means of decay calculations, radioactivity and thermal power per canister have been calculated for these compositions as functions of time after immobilization (or, in the case of West Valley, after the end of 1989) for periods up to one million years. Summary tables showing these quantities are included in Chapter 3. More detailed tables, showing the contributions of each radionuclide, are given in Appendix 3A. These tables also show alpha curies, photon energy spectra, and neutron source terms for spontaneous fission and (α, n) reactions.

Appendix 3B gives the quantities and characteristics of interim HLW forms (i.e. liquids, slurries, sludges, calcines, etc.) now in storage at each site and projected to the year 2020. The compositions of the interim forms are quite diverse; however, certain broad categories can be defined in a relatively straightforward manner for purposes of characterization, such as alkaline or acidic liquids, slurries in double-shell tanks, etc. Baseline solidification processes are described for each site to provide background information on the production of HLW canisters and of any associated transuranic (TRU) waste and low-level waste (LLW) in the greater-than-class-C category.

The West Valley and Savannah River HLW are generally quite similar, and both will be vitrified for final immobilization. The Hanford HLW are distinctive in that the older wastes have had most of their cesium and strontium (the major fission products) stripped out, converted to solid salts, and placed in sealed capsules. Thus, much of the fission product activity is concentrated in the CsCl and SrF_2 capsules. This practice was discontinued around 1985. The eventual disposition of these capsules has not yet been decided; one option is to recombine the contents with Hanford HLW. The Idaho HLW are unique because they are not neutralized and are subsequently calcined to a granular solid, 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 covers both the immobilized waste in canisters and the interim waste forms.

Table 1.5 summarizes HLW data for all four source sites, both as the interim forms and the immobilized forms in canisters. According to the projections used in this report, a total of about 15,300 canisters of immobilized HLW will have been produced by the end of year 2020, and additional amounts will be canistered after that time. These projections were based on immobilization schedules submitted by the four sites to the 1990 Integrated Data Base (IDB 1990). At that time, it was estimated that the immobilization startup dates would be 1993 for WVDP, 1992 for SRS, 1999 for HANF, and 2012 for INEL. More recent information (IDB 1991) shows estimated startup dates of 1996 for WVDP, 1993 for SRS, 1999 for HANF (this was unchanged), and 2014 for INEL. It was not possible to incorporate these recent schedule changes into the detailed projections presented in Chapter 3 of this report. When reading Chapter 3, the reader should be aware that changes in vitrification schedules have occurred.

1.3.2 West Valley Demonstration Project (WVDP)

The interim form of this HLW is primarily alkaline liquid and sludge from Purex-type reprocessing. There is also some acidic liquid from Thorex-type reprocessing. These two source streams, after pretreatment to concentrate their radioactivity, will be combined and vitrified into borosilicate glass. Vitrification is now scheduled to start in 1996. An on-site storage facility will provide space for the entire production of canisters of HLW glass until a repository becomes available.

1.3.3 Savannah River Site (SRS)

The interim waste forms at SRS are alkaline liquid and sludge, plus a large amount of salt cake. The liquid and dissolved salt cake will be processed to precipitate the cesium. The precipitate will be combined with the sludge for vitrification into borosilicate glass at the Defense Waste Processing Facility (DWPF). The decontaminated liquid and salt cake will be converted to saltcrete, a LLW form.

As of March 1991, design of the DWPF was more than 99% complete, and construction was 99% complete. Vitrification of the HLW is now scheduled to begin in 1993. The facility includes storage buildings where the filled canisters of HLW can be stored on-site until a repository becomes available.

1.3.4 Hanford Site (HANF)

The interim wastes at Hanford consist of alkaline liquid, slurry, sludge, and salt cake. Most of the old waste has been treated to remove radioactive strontium and cesium. The Sr-90 and Cs-137 have been converted to

solid strontium fluoride and cesium chloride and, together with their daughters, have been placed in double-wall steel capsules and stored in water basins. The liquid, sludge, slurry, and salt cake wastes are stored in underground concrete-encased carbon-steel tanks. The Hanford Waste Vitrification Plant (HWVP) is now in preliminary conceptual design and is scheduled to start up in December 1999, with the first production of canisters of borosilicate glass waste to take place in 2000. The current plan is that the vitrification plant will have four feeds: neutralized current acid waste (NCAW), neutralized cladding removal waste (NCRW), complexant concentrate (CC), and Plutonium Finishing Plant waste (PFP).

It has been determined that the contents of the Sr-90 and Cs-137 capsules will go to a repository, but no decision has been made as to whether they will go as overpacked capsules or be opened and blended with the liquid HLW prior to vitrification.

An on-site storage facility will be constructed at Hanford with sufficient capacity to store 2,000 canisters of HLW glass.

1.3.5 Idaho National Engineering Laboratory (INEL)

The Idaho Chemical Processing Plant produces a distinctive waste form in that the acidic liquid waste resulting from fuel reprocessing is calcined directly to a granular solid composed principally of oxides. The nitrates are destroyed in the process. Fluoride is added during dissolution of the fuel to dissolve the Zircaloy cladding. During calcination, this fluoride is converted to insoluble CaF_2. The calcine also contains large fractions of Al_2O_3 and ZrO_2.

The calcine is stored underground in concrete vaults and will eventually be immobilized for final disposal. The final choice of the immobilization process has not yet been made. Should vitrification in the form of borosilicate glass be selected, a relatively large volume of glass would be produced. For this reason, other alternatives are being considered. One of these, a high-density glass-ceramic composition based on CaF_2 and ZrO_2, would have about 40% the volume of glass for the same quantity of HLW. The canister production rates for INEL in this report are based on this glass-ceramic form. Production of immobilized HLW in a form suitable for repository disposal is currently scheduled to start around the year 2014.

1.3.6 References for Section 1.3

IDB 1990. *Integrated Data Base for 1990: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, prepared for U.S. Department of Energy by Oak Ridge National Laboratory, DOE/RW-0006, Rev. 6, October 1990.

IDB 1991. *Integrated Data Base for 1991: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, prepared for U.S. Department of Energy by Oak Ridge National Laboratory, DOE/RW-0006, Rev. 7, October 1991.

Table 1.5. Summary data for high-level waste

	WVDP	SRS	HANF[a]	INEL
	Interim forms			
Volumes, m^3, end of year 1989				
Liquid	1,800	53,300	26,500	8,500
Sludge	50	13,800	46,000	0
Salt cake	0	54,800	93,000	0
Slurry	0	0	79,300	0
Calcine	0	0	0	3,500
Precipitate	30[b]	100	0	0
Volumes, m^3, end of year 2020				
Liquid	0	42,900	11,300	300
Sludge	0	10,400	46,000	0
Salt cake	0	21,400	93,000	0
Slurry	0	0	115,800	0
Calcine	0	0	0	8,600
Precipitate	0	1,100	0	0
	Immobilized forms[c]			
Total number of canisters at end of year 2020	275	5,282	1,960	7,800
Kilocuries per canister[d]	115	234	298	109
Watts per canister[d]	342	709	869	339
Maximum expected vitrification rate, canisters per year	200	410	370	1,000[e]

[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. Because of unavailability of data, the volumes of waste shown for Hanford for 2020 do not include the effect of converting interim waste forms to glass.

[b]This represents wastes collected on ion-exchange resins during pretreatment.

[c]Borosilicate glass for WVDP, SRP, and HANF; high-density glass-ceramic for INEL. Canisters are assumed to be 2 ft in diam by 10 ft long.

[d]At the time of immobilization. For SRS, HANF, and INEL, estimated maximum values are shown; many canisters will be much lower. WVDP values are for end of year 1991 and include a 10% safety factor.

[e]This includes 650 from then-current operations plus 350 from the backlog.

1.4 NON-LWR SPENT FUELS (see Chapter 4)

1.4.1 Scope

This category includes spent fuels from research, test, and experimental reactors as well as two non-LWR power station reactors. 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 a wide spectrum of enrichments and materials. Some of the fuels are expected to be reprocessed at SRS or INEL. In other cases, however, the fuels are of types that are difficult to reprocess because of their unique chemical form or content. In this report, characterization is done in terms of fuel element descriptions, quantities, and estimated 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 1.6. Additional information on quantities and characteristics is given in Salmon 1990.

DOD-owned Navy fuels and other classified fuels are also being considered for disposal in a geologic repository (Harrison-Gieslcr 1991). These are not included in this report. Those which have been reprocessed for DOD at INEL provide the source material for INEL HLW.

1.4.2 Fort St. Vrain Reactor

The 330-MW(e) Fort St. Vrain high-temperature, gas-cooled (HTGR) reactor achieved initial criticality in 1974 and was permanently shut down in August 1989. The fuel elements are large graphite blocks, in the shape of hexagonal prisms, containing uranium and thorium carbide microspheres inside a protective coating. The fuel elements have all been discharged and are now being stored in part at INEL and in part at the reactor site in engineered surface structures. Reprocessing of these fuel elements is not planned at this time. The quantities shown in Table 1.6 include the graphite matrix material. Further information is given in Appendix 4C.

1.4.3 Peach Bottom 1 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 discharged and shipped to INEL for storage: Core I, which is stored in underground dry wells, and Core II, which is stored in the engineered surface facility with Fort St. Vrain spent fuel. The quantities in Table 1.6 include the graphite matrix of the fuel elements. Additional information is given in Appendix 4D.

1.4.4 Research and Test Reactor Fuels

These fuels are categorized into eight basic types that are employed in reactors used at universities or other educational facilities, privately owned research and development (R&D) facilities, DOE-owned laboratories, and government-owned, non-DOE facilities. The number of reactors in each category is given in Table 1.7. Most of them are either MTR-plate-type or uranium-zirconium-hydride fueled TRIGA reactors. The existing and estimated future quantities of these fuels is given in Chapter 4 of this report, along with their physical and chemical descriptions. Further details are given in Appendices 4A, 4B, and 4E.

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. Table 1.8 summarizes the amounts of contained uranium, plutonium, and thorium at each site (IDB 1990). A detailed description of the various fuel elements, their chemical form, and cladding materials is given in Chapter 4 of this report.

Of the total quantities listed in Table 1.8 for Idaho, some is sodium-bonded fuel from the Fermi reactor blanket. Such fuel may be deemed unacceptable for emplacement in a repository because of its content of chemically reactive metal. If this is the case, removal of the sodium or NaK might require decladding, in which case the alternative of reprocessing the fuel might be considered.

1.4.6 References for Section 1.4

Harrison-Giesler 1991. Private communication, D. J. Harrison-Giesler, September 1991.

IDB 1990. *Integrated Data Base for 1990: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics,* prepared for U.S. Department of Energy by Oak Ridge National Laboratory, DOE/RW-0006, Rev. 6, October 1990.

Salmon 1990. R. Salmon and K. J. Notz, *Non-LWR and Special LWR Spent Fuels: Characteristics and Criticality Aspects of Packaging and Disposal,* ORNL/TM-11016, January 1990.

Table 1.6. Summary of non-LWR spent fuels

Reactor or site	Estimated quantities		
	End of 1989	Annual rate	End of 2020
HTGR reactors			
Fort St. Vrain (elements)	732	0[a]	2,208[b]
Peach Bottom 1			
Core I (elements)	819	0[a]	819
Core II (elements)	820	0[a]	820
Research and test reactors[c]			
MTR plate (elements)	–	–	20,000[d]
TRIGA (elements)	–	–	4,500
UO_2/polyethylene (elements)	–	–	87
PULSTAR (elements)	–	–	170
FFTF (assemblies)	170	30-45	677[e]
Miscellaneous (kg HM)[f]			
ANL West	311		
Babcock & Wilcox	88		
PNL	2,348		
HEDL	263[g]		
INEL	39,508[h]		
LANL	38		
ORNL	1,254		
SRS	19,110		

[a]These reactors are permanently shut down. Fort St. Vrain was shut down in 1989 and Peach Bottom 1 in 1974.

[b]Includes discharge of full core following shutdown.

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

[d]This is expected to be reprocessed and disposed of as defense HLW.

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

[f]Reported as kg of heavy metal (U + Pu + Th). Data are from IDB 1990. See Sect. 4.5 for details.

[g]Includes some FFTF and TRIGA fuels.

[h]Not including Shippingport LWBR fuel (982 kg U, mostly U-233, and 56,167 kg Th), 17 Turkey Point-3 assemblies and 69 VEPCO assemblies being used for dry consolidation testing, HTGR fuel, PULSTAR and TRIGA fuel, and TMI-2 spent fuel and core debris.

Table 1.7. Number of operational research and test reactors in each fuel type category[a]

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

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

[b]These reactors have all been shut down. They are discussed in Chapter 4, Sect. 4.4, and are mentioned here for completeness.

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

Table 1.8. Inventory of other miscellaneous spent fuels stored at various sites as of December 31, 1989[a]

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

[a]Source: IDB 1990.

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

[c]Many of the fuels at INEL have a lower uranium enrichment than that of fuels normally processed. These fuels could be reprocessed in a special campaign, if required. Quantity shown includes TMI-2 spent fuel and core debris, Shippingport LWBR fuel, and various other fuels. For details, see Sect. 4.5.

1.5 MISCELLANEOUS WASTES (see Chapter 5)

1.5.1 Scope

Chapters 2, 3, and 4 discuss the principal sources of repository wastes: light-water-reactor spent fuels, high-level wastes, and non-LWR spent fuels. These sources account for over 99% of all the wastes (on a curie basis) expected to go to the repository. The miscellaneous wastes discussed in Chapter 5 are mainly in the greater-than-Class C low-level waste (GTCC LLW) category. GTCC LLW is defined as low-level waste generated by a licensee of the Nuclear Regulatory Commission (NRC) or Agreement States that exceeds the radionuclide concentration limits established by NRC for Class C LLW, and is therefore not acceptable for near-surface disposal. The NRC has recently ruled that GTCC LLW must be disposed of in a geologic repository unless disposal elsewhere has been approved by the NRC (NRC 1989). Typical examples of wastes that might require disposal as GTCC LLW are activated metal hardware (such as control rods), spent fuel disassembly hardware, ion exchange resins, filters, evaporator residues, sealed sources used in medical and industrial applications (such as oil-well logging), moisture and density gauges, scrap, and contaminated trash. Such wastes are generated by routine operations at nuclear power plants, by reactor fuel research facilities, and by manufacturers of radiopharmaceuticals and sealed sources, and will be generated in the future by the decommissioning of nuclear reactors, and possibly by the decommissioning of other facilities.

The wastes discussed in Chapter 5 as possible GTCC LLW are divided into the following general categories:

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

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

1.5.2 Waste Volume Projections

Because of the wide differences in previous estimates and in other data, the DOE National LLW Management Program (NLLWMP) initiated activities to develop best estimates of volumes and radioactivities of GTCC LLW for use in planning for the disposal of this waste. The NLLWMP issued a report on this work in August 1991 (NLLWMP 1991). Volume projections from NLLWMP 1991 and other sources are used in this report. Table 1.9 is a summary of the estimated volumes of miscellaneous wastes expected to be generated by the year 2035.

1.5.3 GTCC LLW from Routine LWR Operations

Typical GTCC LLW from routine LWR operations includes control rod blades and dry tubes from BWRs, local power range monitors, decontamination resins, filters, in-core detectors, instrument strings, and thimble plug assemblies.

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

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

1.5.4 Sealed Radioisotope Sources

Sealed radioisotope sources, also known as radioactive sealed sources or simply sealed sources, are manufactured for use in a wide variety of industrial, medical, and other applications, including oil-well logging, nuclear reactor start-up, density, thickness, and level gauges, radiography, and numerous others. The activities of sealed sources can range from a few curies to several thousand curies. The radioisotopes used in the manufacture of sealed sources in the United States can be purchased from a number of suppliers, the largest of which is DOE. Some of the major radioisotopes, particularly the transuranics, can be obtained for United States use only from DOE. The major radioisotopes that have been distributed thus far in industrial quantities are Co-60, Sr-90, Cs-137, Pu-238, Am-241, Cm-244, and Cf-252.

The number of sealed sources now in existence that may eventually be disposed of as GTCC LLW was recently evaluated in a 1989 survey by NRC. Based on an analysis of the NRC survey, the NLLWMP report estimated that there were about 27,000 sealed sources currently in the possession of specific licensees that would qualify as GTCC LLW. The total base-case and high-case packaged volumes of sealed sources to be disposed of as GTCC LLW by year 2035 were estimated to be 6 and 18 m^3, respectively. Additional investigations into quantities and volumes of sealed sources are planned by both NRC and NLLWMP during FY 1992, so these estimates should be considered as subject to change.

1.5.5 GTCC LLW from Decommissioning LWRs

There are 111 operable, commercial LWR power reactors in the United States as of December 1991. Three additional reactors shown are expected to be operational by 1995. Commercial nuclear reactors, as well as other nuclear facilities, must be decommissioned at the end of their useful life. Decommissioning means the steps taken at the end of a facility's life to retire it safely from service. This may be accomplished by any one of three methods, as defined by NRC. These methods are referred to as SAFSTOR (also known as mothballing), ENTOMB (in-place entombment), and DECON (dismantling and removal to permit unrestricted use of the property).

In this report, as well as in the 1991 NLLWMP report (NLLWMP 1991), it was assumed that reactors will operate for 40 years after initial criticality. Reactor life extensions of 20 years were also considered. The two reports were based on similar assumptions and obtained similar results. The bulk of GTCC LLW from LWR decommissioning will consist of remote-handled, activated metal components removed from the reactor core region. The volumes of decommissioning waste estimated by the NLLWMP report and this report are considerably lower than those estimated in earlier studies. Earlier studies usually considered four PWR core components (core shroud, core barrel, thermal shields, and lower grid plate) to be GTCC LLW. Newer information from in-core measurements of neutron fluences, updated material compositions, and more detailed calculations of activation levels have determined that only the PWR core shroud is GTCC LLW, with the core barrel being GTCC LLW in a few cases. Only the core shroud was calculated to be GTCC LLW for BWRs.

1.5.6 GTCC LLW from Other Sources

Potential GTCC LLW from miscellaneous sources, including the decommissioning of non-LWRs and other facilities, are discussed in Sect. 5.6. Also discussed are wastes that could potentially be classified as GTCC LLW but whose classification at present is uncertain for various reasons. This includes wastes from the decommissioning of facilities owned or operated by licensees of NRC or Agreement States in connection with DOE defense-related or research-related activities. The classification of some of these wastes may require the resolution of legal questions.

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

1.5.7 Other DOE Wastes

Other DOE wastes that may require repository disposal, but which do not fall into any of the categories already mentioned, are discussed in Sect. 5.7. The principal reason for listing this category is to allow for any wastes that might be generated within the OCRWM waste management system. At present, it does not seem likely that significant quantities of such wastes will be generated.

1.5.8 References for Section 1.5

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

Table 1.9. Summary of miscellaneous waste volume projections[a]

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

[a]These are packaged waste volumes. Projections are discussed in Chapter 5. Most of the projections are based on data in NLLWMP 1991. The base case and high case for year 2035 are as defined in NLLWMP 1991 (see Sect. 5.2.3 for definitions).

[b]See Table 5.2.1.

[c]See Sect. 5.3.

[d]See Table 5.4.3.

[e]See Table 5.5.1.

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

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 (a) physical descriptions of intact assemblies, (b) quantities of spent fuel, both historical and projected, and (c) radiological properties. LWR-related wastes discussed are defective fuel, special fuel forms, spent fuel disassembly hardware, and non-fuel assembly (NFA) hardware.

Physical descriptions are presented in Sect. 2.2. Individual assembly types are grouped together by similar design characteristics. Physical Description Reports containing these data are given in Appendix 2A, Physical Descriptions of LWR Fuel Assemblies. Dimensions are given for the unirradiated fuel. The LWR Assemblies Data Base is a menu-driven PC data base that contains detailed physical description data. It also includes spent fuel disassembly hardware physical descriptions and radiological properties.

Quantities information is presented three ways in Sect. 2.3 — a broad overview, a reactor-specific and assembly-type-specific basis for historical inventories, and a reactor-specific basis for projections. The corresponding PC data base, the LWR Quantities Data Base, contains this detailed reactor-specific and assembly-type-specific information.

Radiological properties of intact spent fuel are presented in Sect. 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. These properties are given at various decay times ranging from one year to one million years. In-depth and detailed radiological properties of intact spent fuel are available through the LWR Radiological Data Base, the corresponding PC data base.

Defective fuel is discussed in Sect. 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. The Quantities Data Base also includes data on defective assemblies.

Special fuel forms are discussed in Sect. 2.6. Special fuel forms include degraded fuel, nonstandard fuel, consolidated fuel, Shippingport LWR fuel, reprocessed fuel, and fuel rod crud.

If spent fuel assemblies are consolidated, then spent fuel disassembly (SFD) hardware becomes a waste stream separate from the fuel rods themselves. 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.

NFA hardware includes control elements, burnable poison assemblies, neutron sources, BWR fuel channels, in-core instrumentation, and other components not integral parts of fuel assemblies. This hardware is described in Sect. 2.8. It is also covered by the LWR NFA Hardware Data Base, the associated PC data base.

In addition to the four PC data bases cited above for LWR spent fuel, there is also a Serial Number Data Base. This lists every discharged LWR assembly by serial number and, by accessing the other LWR data bases, provides detailed data for each assembly.

2.2 PHYSICAL DESCRIPTIONS OF LWR FUEL ASSEMBLIES

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. Fuel assemblies are constructed from a number of individual fuel rods arranged together, generally in square arrays. Significant changes in the design of fuel assemblies have taken place as a result of many factors, including increased reactor size, fuel-cycle economics, improved materials of construction, and a broader experience base. The efforts of the Characteristics Database (CDB) over the past four years have identified more than 120 "assembly types" that have been used in commercial LWRs in the United States. In order to bring some order to this large ensemble of assembly types, the CDB introduced the term "assembly class" in 1988 to describe the physical dimensions of fuel assemblies. More recently, the term "fuel design" has been used to describe the nuclear design and expected performance properties of fuel assembly types, often across more than one assembly class. The first three subsections focus on assembly classes, fuel designs, and assembly types, particularly with respect to how these terms can be used to assist in the characterization of the spent fuel that will eventually be accepted by the Civilian Radioactive Waste Management System (CRWMS). The final subsection attempts to identify current trends in fuel-cycle management, including new fuel designs, increased competition for reload supply contracts, and possible effects of joint ventures with foreign fuel vendors. General descriptions of fuel assembly types, focusing on the distinguishing features, are grouped by assembly class and cross-referenced by fuel design in Appendix 2A. The LWR Assemblies Database (Revision 1) contains more detailed information on individual fuel assembly types, where available.

2.2.1 The Assembly Class Concept and Its Uses

With over 100 nuclear reactors operating nationwide, designed by four major vendors, and with reload fuel manufactured by the reactor vendors and other suppliers, a wide variety of fuel assembly types are in existence. The variety of fuel assembly types with which the CRWMS must deal has been further increased by the evolutionary changes in designs and materials. In order that the methodology for classifying these fuel assembly types be systematic, complete, and compatible, the assembly class concept was developed (Moore 1988).

The assembly class is based primarily upon the physical configuration of the fuel rods into assemblies. Westinghouse, Combustion Engineering, Babcock & Wilcox, and General Electric reactors use assemblies which have distinct configurations. This configuration is designed to be compatible with the internals of the rest of the core (e.g. the upper and lower core plate, the control elements, etc). Reload fuels for each reactor must be compatible with this configuration. Thus, reload fuel made by a fuel vendor other than the reactor supplier continues to look like the fuel supplied by the original vendor. It must have the same physical dimensions, although there may be small differences (typically less than 0.1 in.) in some cases. Because the assembly class in based on compatibility with the reactor core configuration, the assembly class for a reactor does not change over time.

Twenty-three assembly classes have been identified and are outlined in Table 2.2.1. Twelve assembly classes are reactor-specific - that is, only one reactor exists that uses fuel from a particular assembly class. Multiple-reactor assembly classes account for the remaining eleven classes. The multiple-reactor classes are applicable to 115 of the 127 commercial LWRs in the U.S. Several of the reactor-specific assembly classes (San Onofre 1, Haddam Neck, Fort Calhoun, and St. Lucie 2) are similar to multiple-reactor assembly classes (WE 14 X 14, WE 15 X 15, CE 14 X 14, and CE 16 X 16, respectively) except for the fact that the fuel assemblies are significantly shorter (10 - 20 in.). Table 2.2.2 lists the particular reactors in each of the multiple-reactor assembly classes and each of the reactor-specific assembly classes.

The advantage of the assembly class scheme is twofold - it reduces to a manageable number (23) the dimensional possibilities that must be considered in the design of transportation casks and disposal canisters. Secondly, because the assembly class for a given reactor does not change over time, it gives a basis for projections on an assembly-class-specific basis. This means that the impact of the fuels from a particular assembly class can be measured over time. Because within assembly classes, utilities have an increasing assortment of fuel options to select for purchase, projections of the assembly type (or even the fuel vendor) have great uncertainty. A prime example of the predictive power of the assembly class scheme can be seen in looking at BWR fuels. Although the array size for the majority of large BWRs has changed from 7 X 7 to 8 X 8 (and 9 X 9 and 10 X 10 fuels are in use and being marketed), the assembly class for these reactors remains the same.

Three changes to the assembly class scheme have been made recently. First, the SOUTH TEXAS class has been moved from a reactor-specific class to a multiple-reactor class, since there are two reactors at the South Texas site. Second, a new class has been designated for fuels from the old Elk River reactor. Third, a separate class for CE SYSTEM 80 16 X 16 fuel has been designated, since its length (178.3 in.) is longer than other CE 16 X 16 fuels (176.8 in.). Originally, this 1.5 in. difference was not considered to be of great importance, but the CE SYSTEM 80 fuel is longer that the 178 in. limit on "Standard Fuel" specified in 10 CFR 961, Appendix E Standard Contract.

The versatility of the assembly class scheme can best be demonstrated by noting that in 1988, when the assembly class was introduced, only 79 fuel assembly types had been identified. Currently, over 120 fuel assembly types are known and the assembly class scheme accounts for all of them; none of the changes to the assembly class

scheme was the result of the addition of these 41 new assembly types.

2.2.2 Fuel Assembly Types

The term "assembly type" as used by the CDB, describes a group of fuel assemblies with similar physical and design characteristics. The assembly type is, in general, a breakdown of the assembly class into various fuel designs. In cases where other differences between fuel assemblies exist within the same assembly class and fuel design, additional fuel assembly types have been designated. This is most frequent with variations in the active fuel length. Assemblies of a particular assembly type are always of the same assembly class, fuel design, and manufacturer. Typically, they have identical lengths, widths, cladding material, active fuel lengths, hardware (both parts and materials), and weight of heavy metal. Factors that may not be identical include initial enrichment, enrichment distributions, and/or neutron absorbers used. Using these criteria, more than 120 fuel assembly types have been identified.

Not all of these fuel assembly types have been used on a large scale. In fact, many have been lead test assemblies (LTAs) of which few assemblies (less than 10) of a particular assembly type exist. Some fuel assembly types are no longer in existence, having been completely reprocessed at West Valley (see section 2.6.6 for information on reprocessed fuel). Table 2.2.3 lists, by assembly class, the fuel assembly types that have been identified to date, their EIA rod array code identifiers, and information on their usage.

2.2.3 Fuel Design

Independently of the introduction of new reactors (and new assembly classes), the fuel vendors have effected many design changes to their initial core and reload fuels over the years. Some of these changes have been introduced individually; others have been introduced into fuel assemblies in combinations. The term "fuel design" is used to designate particular design changes in fuel which often apply to more that one fuel assembly type or to more than one fuel assembly class. Design changes are often proprietary in nature; therefore this section does not always provide definitive information. Fuel design has had and will continue to have an important role in fuel performance. Table 2.2.4 summarizes the fuel designs that have been used in commercial LWRs.

2.2.3.1 Westinghouse Fuel Designs

Westinghouse has manufactured a significant fraction of the fuel that the CRWMS will accept, both for Westinghouse-built reactors, and, increasingly, for other vendors' reactors. The forerunner of modern WE PWR fuel designs were the stainless steel-clad assemblies used at San Onofre-1 and Haddam Neck. This fuel design was soon replaced with a "Standard" design fuel, which was used in WE 14 X 14 and WE 15 X 15 class reactors. In this context, "Standard" design fuel should not be confused with the Standard and Nonstandard fuel categories of 10 CFR 961. Westinghouse's Standard design fuel has inconel grid spacers and stainless steel guide tubes. In the early 70's, "Pressurized Standard" fuel introduced fuel rod prepressurization. In the mid 70's, LOPAR fuels introduced a low parasitic fuel design. In this design, the stainless steel guide tubes of the Standard design were replaced with Zircaloy guide tubes. LOPAR fuels have been utilized at WE 14 X 14, WE 15 X 15, and WE 17 X 17 class reactors. LOPAR fuels have sometimes been referred to as Standard design fuels because of the similarities between the two designs. The Optimized Fuel Assembly (OFA) design was introduced with Zircaloy intermediate grid spacers and an optimized fuel rod diameter (which is smaller for WE 14 X 14 and WE 17 X 17 class reactors). First introduced in 1984, VANTAGE 5 fuel features the same optimized fuel rod diameter and Zircaloy grids as OFA fuel. In addition, features available on VANTAGE 5 assemblies include 1) integral fuel burnable absorbers, 2) intermediate flow mixer grids, 3) natural uranium axial blankets, 4) increased discharge burnups, and 5) reconstitutable top nozzles. VANTAGE 5 fuel features are available separately or in combination. Currently being used in two demonstration assemblies at North Anna-1, VANTAGE + fuel features ZIRLO cladding. ZIRLO is an advanced zirconium-niobium alloy with additional resistance to corrosion at high temperatures and burnups. Westinghouse's newest offering is the Vantage 5H fuel design. VANTAGE 5H (or Hybrid) fuel combines the features available with VANTAGE 5 fuel with Zircaloy grids spacers, but utilizes the larger fuel rod diameters of the STANDARD and LOPAR designs. Westinghouse also offers a debris-filtering bottom nozzle as an option on its newer fuel designs.

In the early 1980's, WE licensed the right to produce a version of an ASEA ABB BWR fuel assembly. Westinghouse's version, the QUAD+, was scheduled for use as LTAs in Browns Ferry in 1985. Since Browns Ferry has been in an administrative shutdown since that time, other QUAD+ LTAs were demonstrated at the Fitzpatrick reactor. Additionally, GE filed a patent infringement suit against Westinghouse over the use of the barrier fuel concept in the QUAD+. GE and WE settled out-of-court, reaching an agreement that WE would only supply QUAD+ for existing reload contracts (with Shoreham and Nine Mile Point 2). At this point, it appears that the 4 LTAs at FitzPatrick will be the only QUAD+ assemblies irradiated in the U.S., since Shoreham will apparently not operate and WE and Niagara Mohawk, the owner of NMP-2, have apparently reached an agreement to cancel WE's reload contract. ASEA had also reached a licensing agreement with Westinghouse for its SVEA 96 fuel. The SVEA 96 is similar to the QUAD+ in that it uses a water channel and four minibundles, but does not utilize barrier fuel. Each of the minibundles is a 5 X 5 array with a single

water rod, making the SVEA 96 a 10 X 10 array with four water rods and a water cross. LTAs of the SVEA 96 were scheduled for loading in Washington Nuclear Power Plant No. 2 (WNP-2) in 1990, and possibly in other reactors as well. Since ASEA has recently purchased CE, it is not certain if these assemblies were to be manufactured by WE, CE, or ASEA.

2.2.3.2 General Electric Fuel Designs

Like Westinghouse, GE has manufactured a significant fraction of the fuel that the CRWMS will accept. Early fuels for the BWR/1 product line reactors (Big Rock Point, Dresden-1, and Humboldt Bay) are grouped together in the GE-1 fuel design (Moore 1989). The original fuel design for use at BWR/2, BWR/3, BWR/4, and BWR/5 reactors, GE-2 fuel, or original 7 X 7 fuel had unacceptably high failure rates which lead to the development of 8 X 8 fuel. The GE-3 fuel design, introduced as an interim solution to the high failure rates of GE-2 fuels, was a 7 X 7 array which featured thicker cladding and a hydrogen getter in the plenum region. The increased cladding thickness gave rise to a slightly smaller fuel pellet diameter, making these assemblies slightly lighter than the GE-2 fuels. A variation on this design, used in BWR/4 reactors, used a slightly longer (146 in.) active fuel length. GE-3 fuels have often been called improved 7 X 7 fuel. Originally introduced as initial core fuel for BWR/6 reactors, GE-4 fuel design was the first 8 X 8 array. It was quickly accepted by the utilities as reload fuel for earlier BWRs. The additional fuel rods in this design reduced the linear heat generation rate substantially. GE-4 fuel also introduced a water rod in the center of the assembly to provide nonboiling water along the length of the assembly for better neutron moderation. A variation on this design, used in BWR/4 reactors, used a slightly longer (146 in.) active fuel length. GE-4 fuel has often been called simply 8 X 8 fuel. GE-5 fuel design, or retrofit fuel, introduced several new design features. Among these were a second water rod, axial blankets of natural uranium at both ends of the fuel rod, and an increased active fuel length (to 145.24 in. for GE BWR/2,3 class assemblies and to 150 in. for BWR/4-6 class assemblies). Prepressurized fuels were introduced in LTAs used at Peach Bottom in 1977. Essentially, these fuels are GE-5, or retrofit, fuels with the fuel rod prepressurized to 3 atm of helium. This increased prepressurization decreases fission gas release and reduces pellet-clad interaction (PCI) by improving heat transfer within the fuel rod. Barrier fuels are prepressurized fuels with a pure zirconium layer on the interior of the Zircaloy cladding. The softer zirconium effectively inhibits PCI crack formation. Fuels of the GE-8 fuel design feature increased prepressurization (to 5 atm), an increased fuel-pellet diameter, a variable numbers of water rods (2 to 4), single diameter upper end plug shafts, a streamlined upper tie plate to reduce two-phase pressure drop, and axially zoned gadolinia distributions. These fuels were introduced at Brown's Ferry-3 in 1981 and over the past few years, have been GE's standard reload offering. Lead use of GE-9 fuel began in 1987 at the Hatch-1 reactor. GE-9 fuels have ferrule-type spacers (rather than grid-type spacers), a large, centralized water rod that displaces four fuel rod locations, and axially varying enrichment and gadolinia concentrations. LTAs of GE-10 fuel are being irradiated in the Cooper Station reactor. GE-10 fuel features GE-9 features plus an interactive fuel channel. For some plant types, GE-10 fuels offer offset lower tie plates or expanded bundle lattices. GE-11 LTAs have been inserted in the FitzPatrick and WNP-2 reactors. GE-11 fuel is a 9 X 9 array with seven fuel rod locations replaced by two large-diameter water rods. Additionally, GE-11 fuel uses 9 part-length fuel rods to reduce the two phase pressure drop. GE has also announced GE-12 and GE-13 fuels. GE-12 is a 10 X 10 fuel rod array with part-length rods and a low-pressure drop spacer. GE-13 is a 9 X 9 fuel rod array with increased uranium content and a choice of active fuel lengths. GE-13 fuel is specifically designed to improve critical power performance.

2.2.3.3 Babcock & Wilcox Fuel Designs

Differences in fuel designs from Babcock & Wilcox have not been as dramatic or as noticeable as new designs from other vendors. B&W has, however, continued to introduce fuel design improvements. Apparently, Mark B2 and Mark B3 fuel were used as the initial core loadings for most B&W 15 X 15 plants. The Mark B2 fuel used a corrugated flexible spacer grids and ZrO_2 solid spacer pellets in the fuel rods. This fuel typically had an active fuel length of 144 in. and a density of 93.5% of theoretical. Mark B3 fuel replaced the corrugated flexible spacer with a spring spacer and solid spacer material changed to Zircaloy-4. The active fuel length (by some accounts) decreased to 142 in. and the density increased to >94.5% of theoretical. It is possible that these differences were the result of the intention to irradiate Mark B2 fuel for 2 cycles and Mark B3 fuel for 3 cycles, although this is not known. Mark B4 fuel apparently added a prepressurization feature, as did many fuel designs introduced about this time. Reloads submittals generally indicated that the prepressurization level is proprietary, but early reload (prior to Mark B4 loadings) do not discuss prepressurization, and at least one submittal indicates that the prepressurization in Mark B3 fuel is less that Mark B4 fuel. Additionally, Mark B4 assemblies seem to have a modified end fitting to reduce fuel assembly pressure drop and to increase the holddown margin. A revised upper end fitting eliminated the use of retainers for burnable poison rod assembly holddown in Mark B5 fuel. This fuel also utilizes a redesigned holddown spring made of Inconel 718. Zircaloy intermediate spacers grids (replacing the Inconel spacers grids used on previous designs) were introduced on both Mark B4 and B5 fuels and designated Mark B4Z and B5Z. Zircaloy spacer grids continue to be used in Mark B6, B7, and B8 fuels, but without the 'Z' designation. In

addition, Mark B6 fuel uses a skirtless Inconel upper end grid and has a removable upper end fitting. Mark B7 fuel uses the Mark B6 features and adds a shorter lower end fitting, longer fuel rods, and an increased plenum volume. Mark B8 incorporates a debris-fretting resistent fuel rod design featuring a longer lower end plug on the fuel rods, which extend below the bottom end grid. Because this longer lower end plug shortens the plenum region, the fuel rod prepressurization has also been reduced.

2.2.3.4 Combustion Engineering Fuel Designs

CE has always utilized Zircaloy for the core components of its fuel assemblies (cladding, grid spacers, and guide tubes). Detailed information on CE fuel designs is limited to those data provided in the assembly descriptions (Appendix 2A). In addition to fuel for its own reactors, CE has also manufactured fuel for the Yankee Rowe reactor. In 1989, CE was acquired by ASEA ABB, a Swedish conglomerate. The effect of this acquisition on CE fuel designs is as yet unknown.

2.2.3.5 ANF Fuel Designs

Siemans Nuclear Power Corporation (which has also been called Advanced Nuclear Fuels, Exxon Nuclear Company, and Jersey Nuclear), was purchased from The Exxon Corporation by Siemans KWU in 1987. Because Advanced Nuclear Fuels was the name used by this entity for the majority of the period during which this report was prepared, we attribute all fuel assemblies manufactured by them to Advanced Nuclear Fuels (ANF). ANF has manufactured reload assemblies for reactors built by Westinghouse, Combustion Engineering, General Electric, and Allis-Chalmers. ANF has designed fuel for many multiple-reactor classes, including WE 14 X 14, WE 15 X 15, and WE 17 X 17, CE 14 X 14, GE BWR/2,3 and GE BWR/4-6 class reactors. ANF has also supplied fuel to many single-reactor classes, including Fort Calhoun, Palisades, Dresden-1, Big Rock Point, Humboldt Bay, and LaCrosse. Many of these designs have provided innovative solutions to unique situations. At one reactor, H.B. Robinson-2, the core supports at the core periphery were in danger of embrittlement from excessive neutron exposure. In order to reduce the neutron flux to these core supports ANF developed Partial Length Shield Assemblies. These assemblies are mechanically identical to the axially blanketed fuel manufactured by ANF for WE 15 X 15 class reactors, except that the bottom 42 inches of the fuel column contain stainless steel inserts. Inserted on the periphery of the core, the inserts in these assemblies effectively shield the core supports from additional neutron exposure. The degree of activation of cobalt in the incore stainless steel in these assemblies is not known at this time. ANF's Toprod fuel design for WE 14 X 14 class reactors, took advantage of ANF's experience with the use of gadolinia in BWR fuels and extended the technology to PWR fuels. Fuel rods with integral gadolinia have also been used in assemblies for WE 15 X 15 and CE 14 X 14 (and possibly WE 17 X 17) class reactors. In 1983, BWR assemblies featuring 9 X 9 arrays and two water rods were introduced. BWR fuel designs introduced since ANF's purchase by KWU include the 9 X 9-5, a fuel design using five water rods; the 9 X 9-9X, using a square water channel that displaces 9 fuel rod positions; and the 9 X 9-IX, featuring the water channel and zirconium barrier fuel. Current licensing submittals indicate plans to introduce the 9 X 9-IX+, an assembly with a high-performance spacer, presumably in addition to 9 X 9-IX features.

2.2.4 Trends in Fuel Management and Its Effect on Fuel Design

A number of factors, driven primarily by fuel cycle economics and increased competition for reload fuel contracts, may have a significant effect on the characteristics of spent fuel from future discharges. Many nuclear power plants are moving from annual refueling cycles to fuel cycles of 18 to 24 months. These longer cycle times will probably not be used at all reactors, but a significant number of reactors are moving towards them. The effect of these longer cycles seems to be toward increasing the final discharge burnup of the fuel assemblies. Increasing burnups, especially in PWRs, have begun to be evidenced in discharge data supplied to the EIA (see section 2.3). These longer fuel cycles and increased discharge burnups require that the initial enrichments of current reload fuel assemblies are higher than previous reloads. The data supplied on incore fuel to EIA and data obtained from the federal docket confirm a strong trend toward higher enrichments and higher expected burnups.

In both BWRs and PWRS, higher enrichments give rise to higher core reactivities. The continuing trend in BWRs towards more and larger water rods and the introduction of water crosses and channels to provide additional nonboiling water for neutron moderation also increases the core reactivity. To offset these increased reactivities, both BWR and PWR fuel vendors are increasing their use of burnable neutron absorbers. While BWRs have used integral gadolinia for many years, both the number of fuel rods containing gadolinia and the concentration of gadolinia continue to increase. Additionally, PWR fuel vendors are increasingly using integral neutron absorbers in their fuel. All PWR fuel vendors have at least experimented with burnable absorbers that are integral to the assembly and WE's use of zirconium diboride coatings on selected fuel rods in VANTAGE 5 fuel has been steadily increasing. The use of burnable absorbers is discussed in more detail in section 2.6.5.

Since GE introduced it as a feature of its 8 X 8 fuel in 1972, the use of water rods and similar concepts in BWR fuels has increased significantly. Current LTAs from GE have two large-diameter water rods that displace seven fuel pins in a 9 X 9 array; ANF LTAs use a water

channel which replaces nine fuels pins in a 9 X 9 array; ASEA's current prototype (SVEA 96) uses a water cross that divides the assembly into four minibundles and each minibundle is a 5 X 5 fuel rod array with an additional water rod in it. The trend toward more, smaller fuel pins in BWR assemblies (from 49 in original BWR fuel to 96 in the SVEA 96) and the increasing use of water rods has reduced the weight of uranium in BWR fuel assemblies significantly. The first GE BWR/2,3 and GE BWR/4-6 assemblies contained upwards of 190 kg of uranium per assembly. Most current LTA prototypes seem to contain about 177 kg of uranium, although some have contained less. A similar trend in PWR assemblies has also occurred. WE 17 X 17 WE LOPAR fuel contains nearly 460 kg, while WE 17 X 17 WE OFA and WE 17 X 17 WE VANTAGE 5 fuel assembly types contain only approximately 426 kg of uranium. Similar trends in CE and ANF fuels can be seen. This trend may have the effect of increasing the number of fuel assemblies the CRWMS must receive in order to meet a specified rate of waste acceptance if the rate is based on metric tons of uranium.

Fuel vendors continue to be sensitive to utility concerns regarding defective fuels. According to vendor-supplied reports, fuel pin failures "due to other than external causes" have been reduced to about the 0.005% level for individual fuel rods. External causes, such as debris fretting and baffle jetting, are now the dominant fuel failure mechanisms. Improvements in fuel design, such as debris-filtering bottom nozzles (WE) and debris-fretting resistant fuel rods (B&W), are being implemented to reduce fuel failure rates even more. A more detailed examination of defective fuel in presented in section 2.5.

In the past several years, perhaps as a result of the slowdown in new reactor construction, nuclear fuel vendors have expended more effort on improving their fuel designs. Of the 49 fuel designs listed in Table 2.2.4, 17 have been introduced since 1984. Additionally, nuclear fuel vendors have aggressively marketed their abilities to provide fuels for other vendors' reactors. GE and WE have been the most visible targets of this effort, presumably because the market for these types of fuels is relatively large.

The final development that impacts future fuels is the acquisition of U.S. fuel vendors by foreign firms. Since the acquisition of ANF by Siemans KWU in 1987, at least three new ANF fuel designs (the 9 X 9-5, 9 X 9-IX, and 9 X 9-9X) have been introduced, based on KWU designs. B&W formed the Babcock & Wilcox Fuel Company in a joint venture with Framatome, a long-time Westinghouse licensee. The impacts of this relationship as B&W enters the Westinghouse reload market have not yet been determined. ASEA ABB, a large electric power conglomerate with a strong nuclear engineering emphasis, has recently acquired CE. ASEA has had some measure of success in selling its services and its BWR fuels in the U.S. Prior to the acquisition of CE, the marketing and production of ASEA fuel designs was accomplished through a licensing agreement with Westinghouse. This licensing agreement has been canceled and CE will begin to manufacture assemblies for ASEA in the US.

2.2.5 References for Section 2.2

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

Exxon 1986. Exxon Nuclear Company, Inc. *Fuel Assembly Characterization Data*, December 1986.

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

Moore 1988. R.S. Moore, D.A. Williamson, and K.J. Notz, *A Classification Scheme for LWR Fuel Assemblies*, ORNL/TM-10901, November 1988.

Moore 1989. R.S. Moore and K.J. Notz, *Physical Characteristics of GE BWR Fuel Assemblies*, ORNL/TM-10902, June 1989.

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

Table 2.2.1. Characteristics of CDB Assembly Classes*

Assembly Class	Reactor Type	Assembly Length	Assembly Width	Array Size(s) Used
		MULTIPLE REACTOR CLASSES		
GE BWR/2,3	BWR	171.2	5.44	7 X 7, 8 X 8, 9 X 9, 10 X 10
GE BWR/4-6	BWR	176.2	5.44	7 X 7, 8 X 8, 9 X 9, 10 X 10
B&W 15 X 15	PWR	165.7	8.54	15 X 15
B&W 17 X 17	PWR	165.7	8.54	17 X 17
CE 14 X 14	PWR	157.	8.1	14 X 14
CE 16 X 16	PWR	176.8	8.1	16 X 16
CE SYSTEM 80	PWR	178.3	8.1	16 X 16
WE 14 X 14	PWR	159.8	7.76	14 X 14
WE 15 X 15	PWR	159.8	8.44	15 X 15
WE 17 X 17	PWR	159.8	8.44	17 X 17
SOUTH TEXAS	PWR	199.	8.43	17 X 17
		SINGLE REACTOR CLASSES		
Big Rock Point	BWR	84.0	6.52	12 X 12, 11 X 11, 9 X 9, 8 X 8, 7 X 7
Dresden 1	BWR	134.4	4.28	6 X 6, 7 X 7, 8 X 8
Elk River	BWR	81.6	3.5	5 X 5
Humboldt Bay	BWR	95.	4.67	6 X 6, 7 X 7
Lacrosse	BWR	102.5	5.62	10 X 10
Ft. Calhoun	PWR	146.	8.1	14 X 14
Haddam Neck	PWR	137.1	8.42	15 X 15
Indian Point	PWR	138.8	6.27	13 X 14 (14 X 14)
Palisades	PWR	147.5	8.2	15 X 15
St. Lucie 2	PWR	158.2	8.1	16 X 16
San Onofre 1	PWR	137.1	7.76	14 X 14
Yankee Rowe	PWR	111.8	7.62	15 X 16 (16 X 16), 17 X 18 (18 X 18)

*Dimensions are nominal values before irradiation. Definitive data on changes in dimensions due to irradiation are not available. Limited data indicates that assemblies may grow about 1 to 2 inches in length and the effective cross section may increase due to distortion such as bowing or twisting. All dimensions are in inches. Lengths are rounded to the next higher tenth of an inch. Lengths of some newer fuel assemblies use slightly (0.1 in.) longer fuel designs. Widths are rounded to the next higher hundredth of an inch. Fuel assembly widths for GE BWR/2,3 and GE BWR/4-6 Classes include 80 mil fuel channels. Assemblies with thicker channels (100 and 120 mil) have larger widths.

Table 2.2.2. Listing of Commercial LWRs by Assembly Class

MULTIPLE REACTOR CLASSES

GENERAL ELECTRIC BWR/2,3

Dresden 2 (BWR/3)	Dresden 3 (BWR/3)	Millstone 1 (BWR/3)
Monticello (BWR/3)	Nine Mile Point 1 (BWR/2)	Oyster Creek (BWR/2)
Pilgrim (BWR/3)	Quad Cities 1 (BWR/3)	Quad Cities 2 (BWR/3)

GENERAL ELECTRIC BWR/4-6

Browns Ferry 1 (BWR/4)	Browns Ferry 2 (BWR/4)	Browns Ferry 3 (BWR/4)
Brunswick 1 (BWR/4)	Brunswick 2 (BWR/4)	Clinton (BWR/6)
Cooper Station (BWR/4)	Duane Arnold (BWR/4)	Enrico Fermi 2 (BWR/4)
Fitzpatrick (BWR/4)	Grand Gulf 1 (BWR/6)	Grand Gulf 2 (BWR/6) **
Hatch 1 (BWR/4)	Hatch 2 (BWR/4)	Hope Creek (BWR/4)
LaSalle 1 (BWR/5)	LaSalle 2 (BWR/5)	Limerick 1 (BWR/4)
Limerick 2 (BWR/4)	Nine Mile Point 2 (BWR/5)	Peach Bottom 2 (BWR/4)
Peach Bottom 3 (BWR/4)	Perry 1 (BWR/6)	Perry 2 (BWR/6) **
River Bend 1 (BWR/6)	Shoreham (BWR/4) **	Susquehanna 1 (BWR/4)
Susquehanna 2 (BWR/4)	Vermont Yankee (BWR/4)	Washington Nuclear 2 (BWR/5)

BABCOCK & WILCOX 15 X 15

Arkansas Nuclear One, Unit 1	Crystal River 3	Davis-Besse
Oconee 1	Oconee 2	Oconee 3
Rancho Seco *	Three Mile Island 1	Three Mile Island 2 *

BABCOCK & WILCOX 17 X 17

Bellefonte 1 **	Bellefonte 2 **	Washington Nuclear 1 **

COMBUSTION ENGINEERING 14 X 14

Calvert Cliffs 1	Calvert Cliffs 2	Maine Yankee
Millstone 2	St. Lucie 1	

COMBUSTION ENGINEERING 16 X 16

Arkansas Nuclear One, Unit 2	San Onofre 2	San Onofre 3
Waterford 3		

COMBUSTION ENGINEERING SYSTEM 80

Palo Verde 1	Palo Verde 2	Palo Verde 3
Washington Nuclear 3 **		

WESTINGHOUSE 14 X 14

Ginna	Kewaunee	Point Beach 1
Point Beach 2	Prairie Island 1	Prairie Island 2

WESTINGHOUSE 15 X 15

Cook 1	Indian Point 2	Indian Point 3
Robinson 2	Surry 1	Surry 2
Turkey Point 3	Turkey Point 4	Zion 1
Zion 2		

Table 2.2.2. (continued)

MULTIPLE REACTOR CLASSES

WESTINGHOUSE 17 X 17

Beaver Valley 1	Beaver Valley 2	Braidwood 1
Braidwood 2	Byron 1	Byron 2
Callaway	Catawba 1	Catawba 2
Comanche Peak 1	Comanche Peak 2	Cook 2
Diablo Canyon 1	Diablo Canyon 2	Farley 1
Farley 2	Harris	McGuire 1
McGuire 2	Millstone 3	North Anna 1
North Anna 2	Salem 1	Salem 2
Seabrook 1	Sequoyah 1	Sequoyah 2
Summer	Trojan	Vogtle 1
Vogtle 2	Watts Bar 1 **	Watts Bar 2 **
Wolf Creek		

SOUTH TEXAS

South Texas 1	South Texas 2

SINGLE REACTOR CLASSES

Big Rock Point	Dresden 1 *	Elk River *
Fort Calhoun	Humboldt Bay *	Haddam Neck
Indian Point 1 *	LaCrosse *	Palisades
St. Lucie 2	San Onofre 1	Yankee-Rowe

* Reactors are permanently shutdown.

** Reactor completion (or continued commercial use) is uncertain at this time.

Table 2.2.3. Listing of assembly types by assembly class.

Assembly Type	EIA Rod Array Codes Revised	Original	Status	Comments
MULTIPLE REACTOR CLASSES				
GENERAL ELECTRIC BWR/2,3				
GE BWR/2,3 7 X 7 GE-2a	G2307G2A	07G14	Discharged	
GE BWR/2,3 7 X 7 GE-2b	G2307G2B	07G21	Discharged	
GE BWR/2,3 7 X 7 GE-3	G2307G3	07G31	Discharged	
GE BWR/2,3 7 X 7 ANF	G2307A	07EGE	Discharged	
GE BWR/2,3 8 X 8 GE-4	G2308G4	08G41	Discharged	
GE BWR/2,3 8 X 8 GE-5	G2308G5	08G51	Incore	
GE BWR/2,3 8 X 8 GE Prepres.	G2308GP	08G61,08G71	Incore	
GE BWR/2,3 8 X 8 GE Barrier	G2308GB	08G61,08G71	Incore	
GE BWR/2,3 8 X 8 GE-8a	G2308G8A	08G81	Incore	
GE BWR/2,3 8 X 8 GE-9	G2308G9	08G91	Lead Assembly	
GE BWR/2,3 8 X 8 GE-10	G2308G10	-	Lead Assembly	
GE BWR/2,3 9 X 9 GE-11	G2309G11	-	Lead Assembly	
GE BWR/2,3 10 X 10 GE-12	G2310G12	-	Future	
GE BWR/2,3 9 X 9 GE-13	G2309G13	-	Future	
GE BWR/2,3 8 X 8 ANF	G2308A	08EG3	Incore	
GE BWR/2,3 8 X 8 ANF Prepres.	G2308AP	08EG3	Incore	
GE BWR/2,3 9 X 9 ANF	G2309A	09EG3	Incore	
GE BWR/2,3 9 X 9 ANF 9-5	G2309A5	-	Lead Assembly	
GE BWR/2,3 9 X 9 ANF IX	G2309AIX	-	Lead Assembly	
GE BWR/2,3 9 X 9 ANF 9X	G2309A9X	-	Lead Assembly	
GENERAL ELECTRIC BWR/4-6				
GE BWR/4-6 7 X 7 GE-2	G4607G2	07G22	Discharged	
GE BWR/4-6 7 X 7 GE-3a	G4607G3A	07G32	Discharged	
GE BWR/4-6 7 X 7 GE-3b	G4607G3B	07G33	Discharged	
GE BWR/4-6 8 X 8 GE-4a	G4608G4A	08G42	Discharged	
GE BWR/4-6 8 X 8 GE-4b	G4608G4B	08G43	Discharged	
GE BWR/4-6 8 X 8 GE-5	G4608G5	08G52	Incore	
GE BWR/4-6 8 X 8 GE Prepres.	G4608GP	08G62,08G72	Incore	
GE BWR/4-6 8 X 8 GE Barrier	G4608GB	08G62,08G72	Incore	
GE BWR/4-6 8 X 8 GE-8	G4608G8	08G82	Incore	
GE BWR/4-6 8 X 8 GE-9	G4608G9	08G92	Incore	
GE BWR/4-6 8 X 8 GE-10	G4608G10	-	Incore	
GE BWR/4-6 9 X 9 GE-11	G4609G11	-	Lead Assembly	
GE BWR/4-6 10 X 10 GE-12	G4610G12	-	Future	
GE BWR/4-6 9 X 9 GE-13	G4609G13	-	Future	
GE BWR/4-6 8 X 8 ANF	G4608A	08EG4	Incore	
GE BWR/4-6 8 X 8 ANF Prepres.	G4608AP	08EG4	Incore	
GE BWR/4-6 9 X 9 ANF	G4609A	09EG4	Incore	
GE BWR/4-6 9 X 9 ANF 9-5	G4609A5	-	Lead Assembly	
GE BWR/4-6 9 X 9 ANF IX	G4609AIX	-	Lead Assembly	
GE BWR/4-6 9 X 9 ANF 9X	G4609A9X	-	Lead Assembly	
GE BWR/4-6 8 X 8 WE	G4608W	08WQ4	Lead Assembly	
GE BWR/4-6 10 X 10 SVEA 96	G4610C	-	Lead Assembly	

Table 2.2.3. (continued)

Assembly Type	EIA Rod Array Codes Revised	Original	Status	Comments
MULTIPLE REACTOR CLASSES				
BABCOCK & WILCOX 15 X 15				
B&W 15 X 15 B&W Mark B	B1515B4	15BMB	Discharged	
B&W 15 X 15 B&W Mark B2	B1515B2	15BMB	Discharged	
B&W 15 X 15 B&W Mark B3	B1515B3	15BMB	Discharged	
B&W 15 X 15 B&W Mark B4	B1515B4	15BMB	Incore	
B&W 15 X 15 B&W Mark B4Z	B1515B4Z	15BBZ	Incore	
B&W 15 X 15 B&W Mark B5	B1515B5	15BMB	Incore	
B&W 15 X 15 B&W Mark B5Z	B1515B5Z	15BBZ	Incore	
B&W 15 X 15 B&W Mark B6	B1515B6	15BBZ	Incore	
B&W 15 X 15 B&W Mark B7	B1515B7	15BBZ	Incore	
B&W 15 X 15 B&W Mark B8	B1515B8	15BBZ	Incore	
B&W 15 X 15 B7W Mark BEB	B1515BEB	-	Discharged	Lead Assembly
B&W 15 X 15 B&W Mark BGd	B1515BG	15BMB	Discharged	Lead Assembly
B&W 15 X 15 WE	B1515W	-	Projected	Lead Assembly
BABCOCK & WILCOX 17 X 17				
B&W 17 X 17 B&W Mark C	B1717B	17BMC	Discharged	Lead Assembly
COMBUSTION ENGINEERING 14 X 14				
CE 14 X 14 CE	C1414C	14CST	Incore	
CE 14 X 14 ANF	C1414A	14ECE	Incore	
CE 14 X 14 WE	C1414W	14WMC	Incore	
COMBUSTION ENGINEERING 16 X 16				
CE 16 X 16 CE	C1616C	16CSD	Incore	
COMBUSTION ENGINEERING SYSTEM 80				
CE SYSTEM 80 16 X 16 CE	C8016C	16CS8	Incore	
WESTINGHOUSE 14 X 14				
WE 14 X 14 WE Standard	W1414W	14WZS	Incore	
WE 14 X 14 WE LOPAR	W1414WL	14WZZ	Incore	
WE 14 X 14 WE OFA	W1414WO	14WOF	Incore	
WE 14 X 14 WE Vantage 5	W1414WV5	-	Future	
WE 14 X 14 ANF	W1414A	14EWE	Incore	
WE 14 X 14 ANF Top Rod	W1414ATR	14ETR	Incore	
WE 14 X 14 B&W	W1414B	14BST	Discharged	
WESTINGHOUSE 15 X 15				
WE 15 X 15 WE Standard	W1515W	15WZS	Incore	
WE 15 X 15 WE LOPAR	W1515WL	15WZZ	Incore	
WE 15 X 15 WE OFA	W1515WO	15WOF	Incore	
WE 15 X 15 WE Vantage 5	W1515WV5	-	Future	
WE 15 X 15 ANF	W1515A	15EWE	Incore	
WE 15 X 15 ANF Part Length	W1515APL	-	Incore	
WE 15 X 15 B&W Mark BW	W1515B	-	Future	

Table 2.2.3. (continued)

Assembly Type	EIA Rod Array Codes Revised	Original	Status	Comments
MULTIPLE REACTOR CLASSES				
WESTINGHOUSE 17 X 17				
WE 17 X 17 WE LOPAR	W1717WL	17WST	Incore	
WE 17 X 17 WE OFA	W1717WO	17WOF	Incore	
WE 17 X 17 WE Vantage 5	W1717WV5	17WVA	Incore	
WE 17 X 17 WE Vantage +	W1717WV+	-	Lead Assembly	
WE 17 X 17 WE Vantage 5H	W1717WVH	-	Lead Assembly	
WE 17 X 17 ANF	W1717A	17EWE	Incore	
WE 17 X 17 B&W Mark BW	W1717B	17BBW	Incore	
SOUTH TEXAS				
SOUTH TEXAS 17 X 17 WE	WST17W	17WXL	Incore	
SINGLE REACTOR CLASSES				
Big Rock Point				
Big Rock Point 12 X 12 GE	XBR12G	-	Reprocessed	
Big Rock Point 11 X 11 GE	XBR11G	11GBR	Discharged	
Big Rock Point 9 X 9 GE	XBR09G	09GBR	Discharged	
Big Rock Point 7 X 7 GE	XBR07G	07GBR	Discharged	Lead Assembly
Big Rock Point 8 X 8 GE	XBR08G	08GBR	Discharged	Lead Assembly
Big Rock Point 9 X 9 ANF	XBR09A	09EBR	Discharged	
Big Rock Point 11 X 11 ANF	XBR11A	11EBR	Incore	
Big Rock Point 11 X 11 NFS	XBR11N	11NBR	Discharged	
Dresden 1				
Dresden 1 6 X 6 GE Type I	XDR06G	06G11	Reprocessed	
Dresden 1 7 X 7 GE Type II	XDR07G	07GTS	Reprocessed	
Dresden 1 6 X 6 GE Type III-B	XDR06G3B	06G3B	Discharged	
Dresden 1 6 X 6 GE Type III-F	XDR06G3F	06G3F	Discharged	
Dresden 1 6 X 6 GE Type V	XDR06G5	06GE5	Discharged	
Dresden 1 7 X 7 GE SA-1	XDR07GS	07GSA	Discharged	
Dresden 1 8 X 8 GE PF Fuels	XDR08G	08GZ4	Discharged	
Dresden 1 6 X 6 UNC	XDR06U	06UGD	Discharged	
Dresden 1 6 X 6 ANF	XDR06A	06EGD	Discharged	
Elk River				
Elk River 5 X 5 AC	-	-	Discharged	Remainder stored at Savannah River.
Fort Calhoun				
Ft. Calhoun 14 X 14 CE	XFC14C	14CFC	Incore	
Ft. Calhoun 14 X 14 ANF	XFC14A	14EFC	Incore	
Ft. Calhoun 14 X 14 WE	XFC14W	-	Future	

Table 2.2.3. (continued)

Assembly Type	EIA Rod Array Codes Revised	Original	Status	Comments
SINGLE REACTOR CLASSES				
Haddam Neck				
Haddam Neck 15 X 15 WE	XHN15W	15WSS	Discharged	
Haddam Neck 15 X 15 NUMEC Zr	XHN15MZ	15MWZ	Discharged	Lead Assembly
Haddam Neck 15 X 15 NUMEC SS	XHN15MS	15MWS	Discharged	Lead Assembly
Haddam Neck 15 X 15 GULF Zr	XHN15IZ	15IWZ	Discharged	Lead Assembly
Haddam Neck 15 X 15 GULF SS	XHN15IS	15IWS	Discharged	Lead Assembly
Haddam Neck 15 X 15 B&W SS	XHN15B	15BWH,15BWS	Incore	
Haddam Neck 15 X 15 B&W Zr	XHN15BZ	-	Lead Assembly	
Humboldt Bay				
Humboldt Bay 7 X 7 GE Type I	XHB07G1	-	Reprocessed	
Humboldt Bay 7 X 7 GE Type II	XHB07G2	07G13	Discharged	
Humboldt Bay 6 X 6 GE	XHB06G	06G12	Discharged	
Humboldt Bay 6 X 6 ANF	XHB06A	06EGH	Discharged	
Indian Point 1				
Indian Point 1 13 X 14 B&W	XIP14B	-	Reprocessed	
Indian Point 1 13 X 14 WE	XIP14W	14WIP	Discharged	
LaCrosse				
LaCrosse 10 X 10 AC	XLC10L	10AST	Discharged	
LaCrosse 10 X 10 ANF	XLC10A	10EAC	Discharged	
Palisades				
Palisades 15 X 15 CE	XPA15C	15CPR	Discharged	
Palisades 15 X 15 ANF	XPA15A	15EPR	Incore	
St. Lucie 2				
St. Lucie 2 16 X 16 CE	XSL16C	16CSL	Incore	
San Onofre 1				
San Onofre 1 14 X 14 WE	XSO14W	14WSS	Incore	
San Onofre 1 14 X 14 WE Zr	XSO14WZ	-	Lead Assembly	
Yankee Rowe				
Yankee Rowe 17 X 18 WE	XYR18W	18WYR	Discharged	
Yankee Rowe 15 X 16 UNC	XYR16U	16UYR	Discharged	
Yankee Rowe 15 X 16 ANF	XYR16A	16EYR	Discharged	
Yankee Rowe 15 X 16 CE	XYR16C	16CYR	Incore	

Table 2.2.4. Summary of Fuel Design Usage

Name	Year Introduced	Where Used	Brief Description
		WESTINGHOUSE FUEL DESIGNS Westinghouse Reactors	
St. Steel	First Use:	San Onofre-1 Haddam Neck	Stainless steel cladding and guide tubes, inconel grid spacers.
Standard	First Use:	WE 14 X 14 WE 15 X 15	Zircaloy cladding introduced; inconel grid spacers, stainless steel guide tubes.
LOPAR	First Use:	WE 14 X 14 WE 15 X 15 WE 17 X 17 SOUTH TEXAS	Inconel grid spacers, Zircaloy guide tubes.
OFA	First Use: 1979 Farley 1, Point Beach 2 Beaver Valley 1	WE 14 X 14 WE 15 X 15 WE 17 X 17	Zircaloy intermediate grid spacers, optimized fuel rod diameter.
VANTAGE 5	First Use: 1984 V.C. Summer	WE 14 X 14 WE 15 X 15 WE 17 X 17	OFA fuels features, plus 5 options: 1) integral fuel burnable absorbers, 2) intermediate flow mixer grids, 3) natural uranium axial blankets, 4) increased discharge burnups, and 5) reconstitutable top nozzles. Options are available separately or in combination.
VANTAGE 5H	First Use: Unknown	WE 17 X 17	VANTAGE 5H (or Hybrid) fuel combines the features available with VANTAGE 5 fuel with Zircaloy grid spacers, but utilizes the larger fuel rod diameters of the Standard and LOPAR designs.
VANTAGE +	First Use: 1987 North Anna-1	WE 17 X 17	VANTAGE + fuel features ZIRLO cladding. ZIRLO is an advanced zirconium-niobium alloy with additional resistance to corrosion at high temperatures and burnups.
		WESTINGHOUSE FUEL DESIGNS Other Reactor Vendors	
Model C	First Use: 1980 Millstone 2	CE 14 X 14 Fort Calhoun	Fuel designed for use in CE-built reactors.
B&W	First Use: 1991 Three Mile Island-1	B&W 15 X 15	Fuel designed for use in B&W-built reactors.
QUAD+	First Use: 1987 Fitzpatrick	GE BWR/4-6	BWR fuel design utilizing four 4 X 4 minibundles and a Zircaloy water cross.
		GENERAL ELECTRIC FUEL DESIGNS General Electric Reactors	
Early Fuels	First Use: 1959	Dresden-1	Fuels for BWR/1 reactors: 6 X 6, 7 X 7, 8 X 8, 9 X 9,
(GE-1)	Continuing at Big Rock Point	Humboldt Bay Big Rock Point	11 X 11, and 12 X 12 fuel rod arrays.

Table 2.2.4. (continued)

Name	Year Introduced	Where Used	Brief Description
	GENERAL ELECTRIC FUEL DESIGNS (continued) General Electric Reactors		
GE-2	First Use: 1969 Last Discharge: 1979	GE BWR/2,3 GE BWR/4-6	Original 7 X 7 Array
GE-3	First Use: 1972 Last Discharge: 1983	GE BWR/2,3 GE BWR/4-6	Improved 7 X 7 Array - thicker cladding, hydrogen getter, chamfered pellets.
GE-4	First Use: 1974 Last Discharge: 1986	GE BWR/2,3 GE BWR/4-6	Original 8 X 8 Array - introduction of water rod.
GE-5	First Use: 1975 Last Discharge:	GE BWR/2,3 GE BWR/4-6	8 X 8 Retrofit fuel - two water rods, axial natural uranium blankets, longer active fuel rod length.
Prepressurized (GE-6 and GE-7)	First Use: 1977 Peach Bottom-2	GE BWR/2,3 GE BWR/4-6	Retrofit fuel with fuel rods prepressurized to 3 atm helium.
Barrier (GE-6 and GE-7)	First Use: 1979 Quad Cities-1	GE BWR/2,3 GE BWR/4-6	Pressurized Retrofit fuel with pure zirconium barrier on cladding interior.
GE-8	First Use: 1981 Brown's Ferry-3	GE BWR/2,3 GE BWR/4-6	Increased number of water rods (4), larger diameter fuel pellets, higher stack density, axial gadolinia distribution, improved upper tie plate, prepressurization increased to 5 atms in BWR/3-6 reactors.
GE-9	First Use: 1987 Hatch-1	GE BWR/2,3 GE BWR/4-6	Large water central rod, ferrule-type spacers, axially varying enrichment and gadolinia concentrations.
GE-10	First Use: 1988(?) Cooper Station (?)	GE BWR/2,3 GE BWR/4-6	Interactive fuel channel. For certain plant types, offset lower tie plate or expanded bundle lattice.
GE-11	First Use: 1990 WNP-2, Fitzpatrick	GE BWR/2,3 GE BWR/4-6	9 X 9 fuel rod array with 74 fuel rods, two large-diameter water rods, and 9 part-length fuel rods to reduce two phase pressure drop.
GE-12	First Use: New	GE BWR/2,3 GE BWR/4-6	10 X 10 fuel rod array, part-length fuel rods, low-pressure drop spacer.
GE-13	First Use: New	GE BWR/2,3 GE BWR/4-6	9 X 9 fuel rod with increased uranium content and choice choice of active fuel lengths. Improves critical power performance.
	BABCOCK & WILCOX FUEL DESIGNS Babcock & Wilcox Reactors		
Mark B2	First Use: Oconee 2	B&W 15 X 15	
Mark B3	First Use: Oconee 2	B&W 15 X 15	Increased fuel pellet density and changed spacer from corrugated to spring type.
Mark B4	First Use: 1975 Oconee 1	B&W 15 X 15	Introduced fuel rod prepressurization; modified end fitting reduced fuel assembly pressure drop.

Table 2.2.4. (continued)

Name	Year Introduced	Where Used	Brief Description
		BABCOCK & WILCOX FUEL DESIGNS Babcock & Wilcox Reactors	
Mark B5	First Use: 1982 Rancho Seco	B&W 15 X 15	Eliminated burnable absorbers holddown retainers by using modified end fitting. Inconel 718 holddown spring.
Mark B4Z,B5Z	First Use: 1979 Oconee 1	B&W 15 X 15	Mark B4 and B5 fuels with Zircaloy intermediate grid spacers.
Mark B6	First Use: 1988 Ark. Nuclear One-1	B&W 15 X 15	Zircaloy intermediate spacers grids; skirtless upper end grid, and removable upper end fitting.
Mark B7	First Use: 1988 Oconee 3	B&W 15 X 15	Mark B6 features plus slightly shorter lower end fitting, slightly longer fuel rod, and increased plenum volume.
Mark B8	First Use: 1989 Oconee 3	B&W 15 X 15	Debris fretting resistent fuel rod design, reduced prepressurization.
Mark C	First Use: 1976 Oconee 2	B&W 15 X 15	Four demonstration assemblies of fuel design intended for B&W 17 X 17 class reactors.
		BABCOCK & WILCOX FUEL DESIGNS Other Reactor Vendors	
St. Steel	First Use: Haddam Neck	Haddam Neck	Stainless steel clad assemblies for use at WE-built reactors.
Westinghouse	First Use: 1974 Ginna	WE 14 X 14	Demonstration assemblies for use at WE-built reactors.
Mark BW	First Use: 1989 McGuire 1	WE 15 X 15 WE 17 X 17	Lead test assemblies under irradiation; full core reload at McGuire in 1991.
		COMBUSTION ENGINEERING FUEL DESIGNS Combustion Engineering Reactors	
Standard	First Use:	CE 14 X 14 CE 16 X 16 CE SYSTEM 80 Fort Calhoun St. Lucie 2	
Palisades	First Use: Palisades	Palisades	
Yankee Rowe	First Use: Yankee Rowe	Yankee Rowe	

Table 2.2.4. (continued)

Name	Year Introduced	Where Used	Brief Description
		ADVANCED NUCLEAR FUELS, INC. BWR Fuels	
Early BWR Fuels	First Use: 1972	Dresden 1 Humboldt Bay Big Rock Point	Fuels designed for GE BWR/1 reactors.
7 X 7 Arrays	First Use: 1972 Oyster Creek	GE BWR/2,3	7 X 7 array designed for use at GE-built reactors.
8 X 8 Arrays	First Use: 1975 Oyster Creek	GE BWR/2,3 GE BWR/4-6	8 X 8 array designed for use at GE-built reactors.
9 X 9	First Use: 1983 Dresden-2	GE BWR/2,3 GE BWR/4-6	9 X 9 fuel array for GE-built reactors; 2 water rods.
9 X 9-5	First Use: Unknown	GE BWR/2,3 GE BWR/4-6	9 X 9 fuel array for GE-built reactors; 5 water rods.
9 X 9-IX	First Use: 1989 WNP-2	GE BWR/2,3 GE BWR/4-6	9 X 9 fuel array for GE-built reactors; 72 fuel rods; Zirconium barrier used on all rods except Gadolinia rods; 1.65" square water channel.
9 X 9-9X	First Use: 1989 WNP-2	GE BWR/2,3 GE BWR/4-6	9 X 9 fuel array for GE-built reactors; 72 fuel rods; 1.65" square water channel.
		ADVANCED NUCLEAR FUELS PWR Fuels	
Westinghouse	First Use: 1974 Ginna	WE 14 X 14 WE 15 X 15 WE 17 X 17	Fuels designed for use at WE-built reactors.
Toprod	First Use: 1981 Prairie Island-1	WE 14 X 14	Fuels for use at WE 14 X 14 plants; fueled rods containing gadolinia.
Part Length	First Use: 1986 (?) Robinson-2	WE 15 X 15	Fuel for use at WE-built reactors; bottom 42 in. of fuel rod contains stainless steel inserts.
Comb. Eng.	First Use: 1980 Fort Calhoun	CE 14 X 14 Fort Calhoun	Fuels designed for use at CE-built reactors.
Palisades	First Use: 1975 Palisades	Palisades	Fuel designed for use at Palisades reactor.
Yankee Rowe	First Use: 1975 Yankee Rowe	Yankee Rowe	Fuel designed for use at Yankee Rowe.

2.3 QUANTITIES OF INTACT LWR SPENT FUEL

2.3.1 Introduction

The official OCRWM source of quantitative data on discharged and projected discharges of LWR spent fuel is the Energy Information Administration (EIA). They obtain their data directly from the utilities via the RW-859 data form, which is updated annually. These data are reviewed and QA'd by the EIA and then tabulated on their main-frame computer files. This process is normally complete by August each year for the previous calendar year, after which the data are made available (on tape) to the Characteristics Database (CDB) System and other qualified users of these data. The CDB LWR Quantities Database extracts those data pertinent to systems integration activities and makes them available in a user-friendly, menu-driven database that runs on IBM-compatible microcomputers. Information contained in the LWR Quantities Database encompasses:

- o Number of assemblies,
- o Metric Tons of Heavy Metal,
- o Burnup,
- o Enrichment,
- o Fuel assembly class,
- o Discharge year,
- o Reactor type,
- o Fuel assembly type (historical data only),
- o Storage location (historical data only), and
- o Defective assemblies (historical data only).

This information is available in summary or detailed reports. All of the tables in this section were generated directly from the LWR Quantities Database and are representative of the output capabilities of this database.

2.3.2 Data Sources

The CDB System obtains quantitative information on historical and projected spent fuel discharges from the EIA via the RW-859 database. The EIA administers this database and the Nuclear Fuel Data Form RW-859, which is used to collect data from owners of commercial nuclear power plants and owners and caretakers of spent nuclear fuel.

The EIA uses Form RW-859 to collect data on historical assemblies discharged from domestic commercial nuclear reactors, spent fuel projected to be discharged, and spent fuel storage pool inventories and capacities. For each group of assemblies permanently discharged during the most recent refueling interval, the utilities submit the following data:

- o Discharge cycle,
- o End of cycle date,
- o Average initial loading weight for each group of assemblies, in metric tons of uranium (MTU),
- o Nominal initial enrichment assay for each assembly discharged, in weight percent,
- o Average discharge burnup for each group of assemblies, in gigawatt-days/metric ton of uranium,
- o Assembly type and rod array configuration for each group of assemblies,
- o Current storage location of each assembly, and
- o Number of assemblies in each group.

For these purposes, a group is defined as a collection of assemblies with similar characteristics (assembly type, enrichment, burnup, cycles of irradiation, etc.). The utilities also provide spent fuel discharge projections for their next five fueling cycles. Spent fuel discharge data are projected at the group level, where a group is characterized by the number of assemblies in it, the nominal initial enrichment, and the group average discharge burnup. They also report the projected cycle shutdown dates and the cycle burnup in terms of effective full-power days.

The EIA also produces the official DOE spent fuel projections, using three scenarios describing the future operation of nuclear power plants in the U.S. The three scenarios are: 1) No New Orders Case; 2) Lower Reference Case; and 3) Upper Reference Case. These scenarios reflect different assumptions about schedules for construction of nuclear power plants, cancellations, operating license renewal, and new orders for reactors. In addition to these three scenarios, the sensitivity of both the No New Orders Case and the Lower Reference Case to alternative assumptions are measured. Alternative assumptions include: 1) higher capacity factors; 2) lower capacity factors; 3) higher tails assay; 4) lower tails assay; 5) no increase in fuel burnup; 6) higher than projected increases in fuel burnup; and 7) with license renewal. Disaggregated (reactor-specific) spent fuel forecasts for the base No New Orders Case are published annually by the EIA (EIA 1991a). These disaggregated forecasts include:

- o Discharge cycle,
- o End of cycle data,
- o Number of assemblies,
- o Average initial loading weight for each group of assemblies, in metric tons of uranium,
- o Nominal initial enrichment assay for each assembly, in weight percent, and
- o Average discharge burnup for each group of assemblies, in megawatt-days per metric ton of uranium.

The utility-furnished historical data, along with the projections of spent fuel discharges from EIA's disaggregated forecasts, constitute the official RW-859 database (EIA 1991b). EIA distributes this database on magnetic tapes to ORNL for inclusion in the LWR Quantities Database portion of the CDB System.

2.3.3. Overview of LWR Spent Fuel Discharges

As of the end of 1990, 77,256 assemblies, representing 21,868 metric tons of heavy metal, had been permanently discharged from commercial LWRs in the United States. This figure does not include 2208 discharged assemblies from Fort St. Vrain (a High-Temperature Gas-Cooled reactor), 142 temporarily

discharged assemblies at various reactor sites, or approximately 1863 LWR assemblies that were reprocessed at the Nuclear Fuel Services Reprocessing Plant at West Valley, NY. Of the total that were permanently discharged, 58.6 percent are BWR assemblies and 41.4 percent are PWR assemblies. By weight, 62.4 percent are PWR assemblies and 37.6 percent are BWR assemblies. The average burnup is 26.6 GWd/MTIHM for all assemblies. It is 21.3 Gwd/MTIHM for the BWR assemblies and 29.8 Gwd/MTIHM for the PWR assemblies. The average enrichment is 2.74 percent for all assemblies. The enrichment averages 2.35 percent for BWR assemblies and 2.98 percent for PWR assemblies.

Table 2.3.1 presents additional detailed information on historically discharged assemblies, broken down by assembly class. By both number of assemblies and by weight, 93 percent of assemblies have been discharged from the multiple-reactor assembly classes. Of the historically discharged assemblies, 56.1 percent come from the GE BWR/2,3 and GE BWR/4-6 classes (20.4 and 35.7 percent, respectively). By weight, these two classes (13.3 and 23.3 percent), along with the WE 15 X 15 (13.0 percent) and WE 17 X 17 (18.2 percent) classes, account for 67.8 percent of the total. Since reactors from the two GE assembly classes completely dominate the BWR discharges, the burnup and enrichment averages for the classes are almost identical to the BWR totals, 21.4 Gwd/MTIHM and 2.34 percent. For the WE 15 X 15 and WE 17 X 17, the average burnups are 30.8 and 29.6 Gwd/MTIHM, respectively. The average enrichments are 2.98 and 2.95 percent, respectively. The slightly higher average burnup and enrichment in the WE 15 X 15 class fuel is attributable to the fact that the 10 reactors in this class have been in operation longer than the reactors in the WE 17 X 17 class. Thus, the impact of the lower enrichments and burnups typical of first and second cycle PWR fuel in the averages is decreased.

Total projected discharges of LWR spent fuel from existing reactors amount to about three times the amount currently in storage. This amounts to 217,362 assemblies and over 64,181 metric tons of heavy metal. Of this, BWR assemblies represent 54.1 percent of the assemblies, but only 32.6 percent of the weight. PWR assemblies represent 45.9 percent of the assemblies, but 67.4 percent of the weight. The average burnup of all the projected discharges is 41.5 Gwd/MTIHM. (34.8 for BWRs and 44.8 for PWRs). The average enrichment is 3.30 percent for BWRs and 4.14 percent for PWRs (3.87 percent overall).

Table 2.3.2 presents more detailed information on projected discharges, broken down by Assembly Class. Here, the multiple-reactor assembly classes account for almost 98 percent of the total, by both number of assemblies and by weight. The GE BWR/4-6 and WE 17 X 17 assembly classes become even more dominant in the future, accounting for 44.9 and 22.9 percent of the assemblies, respectively, and 27.3 and 34.3 percent of the weight. Projected burnups are 35.1 Gwd/MTIHM for the GE BWR/4-6 class and 45.4 Gwd/MTIHM for the WE 17 X 17 class. The average enrichments also rise, reaching 3.30 percent for the GE BWR/4-6 class and 4.21 percent for the WE 17 X 17 class.

Table 2.3.3 totals the historical and projected discharges, also by assembly class.

2.3.4 Assembly Class and Assembly Type Specific Quantities

Since the initial version of the Characteristics Database was issued in 1987, the EIA has worked diligently to improve the descriptions of fuel assembly types in the instructions to the utilities and in improving the response of the utilities. The CDB staff have assisted the EIA in this regard by supplying information on fuel assembly types (especially with regard to GE-built fuels), by developing the assembly class scheme for fuel assembly classification, and by reviewing utility RW-859 submittals for consistency with reload licensing documents submitted to the Nuclear Regulatory Commission. The net result is drastically improved fuel assembly type descriptions in the RW-859 database. In the original LWR Quantities Database, more than 50 percent of the fuel assembly types were unknown or insufficiently identified. The 1990 RW-859 database contains no unidentified fuel assembly types, and much of these data have been independently verified.

Tables 2.3.4 through 2.3.7 give examples of the power of the LWR Quantities Database to provide assembly class and assembly type specific data. Since the assembly type information is currently supplied by the utilities only at the end of the assembly lifetime and because utilities often modify reload contracts to implement new fuel designs, assembly type information is only supplied for historical data.

Tables 2.3.4 and 2.3.5 show the characteristics of the discharges for each of the assembly types discharged from GE BWR/4-6 and WE 17 X 17 class reactors. These reports, especially in the case of the GE BWR/4-6 class, show the introduction, acceptance and widespread use, and decline in use of fuel designs (through fuel assembly types).

While the more advanced fuel designs have not yet shown increasing burnups, much of this can be attributed to the use of older fuel designs in the startup of new reactors. This is the result of long, unanticipated delays in the construction and licensing of newer reactors. For example, the LaSalle County-2 reactor began operation in 1984 with GE BWR/4-6 8 X 8 GE-5 fuel. This fuel was last used as a scheduled reload in 1979 at the Brunswick-2 reactor. The effects of this can be seen in the 1987 discharge of this fuel assembly type -- 264 assemblies with an average enrichment of 1.546% and a burnup of 12,636 MWd/MTIHM. Of the 1467 GE BWR/4-6 8 X 8 GE-5 fuel assemblies discharged over the past four years, 1071 (73 percent) are from the initial cores of the LaSalle-1 and LaSalle-2 reactors. These trends will continue for the next several years as reactors currently coming online

continue to discharge their initial core assemblies. A second factor in the absence of obvious trends to higher burnups comes from the increasing time between the assembly's initial insertion in a reactor and its discharge. As cycle times of 18 to 24 months become more and more commonplace, incore assembly lifetimes of 6 to 8 years will not be uncommon.

Tables 2.3.6 and 2.3.7 present projected discharge data for the GE BWR/4-6 and WE 17 X 17 assembly classes. The data in these tables are broken down by discharge year and burnup bin. In these tables, the trend towards higher enrichments and burnups is clearly evident. Current BWR discharges, although somewhat spread out, peak at burnups between 25 and 30 Gwd/MTIHM. As early as 1992, this peak shifts to the 30 to 35 Gwd/MTIHM range, and by 1998, the peak is 35 to 40 Gwd/MTIHM. The peak discharges of current PWR assemblies have already moved from the 30 to 35 Gwd/MTIHM range to 35 to 40 Gwd/MTIHM. By 1994, the peak moves to 40 to 45 Gwd/MTIHM. Importantly, most of the assemblies upon which these particular burnup projections are based are already being irradiated.

2.3.5 Reactor-Specific Spent Fuel Discharges

The LWR Quantities Database also tracks fuel assemblies on a reactor-specific and utility-specific basis. Tables 2.3.8 and 2.3.9 provide information on the historical and projected discharges from a specific reactor, Monticello. The historical data are broken down by storage location and discharge year, showing the fuel that Monticello has shipped to the storage facility at Morris. The projected data are simply broken down by discharge year and show the scheduled shutdown of the Monticello reactor in 2010.

2.3.6 References for Section 2.3

EIA 1991a. U.S. Department of Energy, Energy Information Administration, *World Nuclear Fuel Cycle Requirements 1991*, DOE/EIA-0436(91), Washington, D.C., October 1991.

EIA 1991b. U.S. Department of Energy, Energy Information Administration, Nuclear Fuel Data Form RW-859, Washington, D.C. (data as of December 31, 1990).

Table 2.3.1. Historical quantities of spent fuel by Assembly Class (Reproduced from the LWR Quantities Database).

LWR QUANTITIES DATABASE
Historical Data through December 31, 1990
Discharged Assemblies by Assembly Class

ASSEMBLY CLASS	NUMBER OF ASSEMBLIES	DEFECTIVE ASSEMBLIES*	AVERAGE BURNUP (Mwd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
B&W 15 X 15	4311	67	28416	2001.5	2.868
B&W 17 X 17	4		29507	1.8	2.838
CE 14 X 14	3793	15	30351	1452.5	2.916
CE 16 X 16	1432	25	28363	596.1	2.843
CE SYSTEM 80	496		22091	203.6	2.424
GE BWR/2,3	15721	1478	21996	2921.8	2.398
GE BWR/4-6	27595	972	21094	5099.6	2.309
WE 14 X 14	3308	83	33023	1276.2	3.171
WE 15 X 15	6275	132	30841	2836.1	2.976
WE 17 X 17	8791	86	29595	3974.2	2.951
SOUTH TEXAS	120		13825	65.0	1.814
Big Rock Point	359	52	19968	47.4	3.483
Dresden 1	892	159	16241	90.9	2.166
Ft. Calhoun	477		31245	172.0	2.981
Humboldt Bay	390	1	14936	28.9	2.350
Haddam Neck	786	46	31303	324.5	3.826
Indian Point	160		16715	30.6	4.110
Lacrosse	333	104	14708	38.0	3.729
Palisades	656	24	23994	262.4	2.687
St. Lucie 2	396		30082	149.9	2.817
San Onofre 1	508	7	29106	186.2	3.812
Yankee Rowe	453		28569	108.9	3.966
--- GRAND TOTALS	77256	3251	26576	21868.1	2.744

*As reported by the Utilities

Table 2.3.2. Projected quantities of spent fuel by Assembly Class (Reproduced from the LWR Quantities Database).

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Discharged Assemblies by Assembly Class through 2040

ASSEMBLY CLASS	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (Mwd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
B&W 15 X 15	6048	42816	2807.5	3.882
B&W 17 X 17	3127	45617	1427.6	4.097
CE 14 X 14	5990	46781	2264.7	4.279
CE 16 X 16	6709	44685	2738.8	4.318
CE SYSTEM 80	7638	45740	3221.4	4.117
GE BWR/2,3	19598	33487	3404.0	3.251
GE BWR/4-6	97616	35077	17503.7	3.304
WE 14 X 14	4477	43812	1609.8	3.939
WE 15 X 15	8609	43179	3917.5	3.907
WE 17 X 17	49808	45448	21979.7	4.211
SOUTH TEXAS	2951	43401	1596.0	4.160
Big Rock Point BWR	276	25024	36.2	3.452
Ft. Calhoun PWR	661	41859	235.3	3.996
Haddam Neck PWR	714	37766	260.5	3.834
Palisades PWR	862	34197	344.6	3.524
St. Lucie 2 PWR	1482	49788	576.4	4.342
San Onofre 1 PWR	528	28348	195.5	4.119
Yankee Rowe PWR	268	30588	61.8	3.880
--- GRAND TOTALS	217362	41526	64180.9	3.865

Table 2.3.3. Total quantities (Historical and Projected) of spent fuel by Assembly Class (Reproduced from the LWR Quantities Database).

LWR QUANTITIES DATABASE
Total Discharge: Historical and Projected Data
Discharged Assemblies by Assembly Class

ASSEMBLY CLASS	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (Mwd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
B&W 15 X 15	10359	36823	4808.9	3.460
B&W 17 X 17	3131	45596	1429.4	4.095
CE 14 X 14	9783	40361	3717.1	3.746
CE 16 X 16	8141	41768	3334.9	4.054
CE SYSTEM 80	8134	44334	3425.0	4.016
GE BWR/2,3	35319	28179	6325.8	2.857
GE BWR/4-6	125211	31923	22603.2	3.080
WE 14 X 14	7785	39041	2886.0	3.599
WE 15 X 15	14884	37998	6753.6	3.516
WE 17 X 17	58599	43021	25953.9	4.018
SOUTH TEXAS	3071	42243	1661.0	4.068
Big Rock Point BWR	635	22159	83.6	3.469
Dresden 1 PWR	892	16241	90.9	2.166
Ft. Calhoun PWR	1138	37376	407.3	3.567
Humboldt Bay PWR	390	14936	28.9	2.350
Haddam Neck PWR	1500	34181	585.0	3.830
Indian Point PWR	160	16715	30.6	4.110
Lacrosse PWR	333	14708	38.0	3.729
Palisades PWR	1518	29786	607.1	3.162
St. Lucie 2 PWR	1878	45721	726.3	4.028
San Onofre 1 PWR	1036	28718	381.7	3.969
Yankee Rowe PWR	721	29300	170.7	3.935
–GRAND TOTALS	294618	37727	86049.0	3.580

BWR 141,165
PWR 133,453

Table 2.3.4. Historically discharged assemblies for the GE BWR/4-6 Assembly Class, broken down by Assembly Type and Discharge Year (reproduced from the LWR Quantities Database).

LWR QUANTITIES DATABASE
Historical Data through December 31, 1990
Data Broken Down By: Assembly Type, Discharge Year
Discharged Assemblies by Assembly Class: GE BWR/4-6

DISCHARGE YEAR	NUMBER OF ASSEMBLIES	DEFECTIVE ASSEMBLIES*	AVERAGE BURNUP (Mwd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
Assembly Type: GE BWR/4-6 7 X 7 GE-2					
1973	50	50	3741	9.5	2.500
1974	328	328	9199	63.3	2.500
1976	285	5	9718	55.8	1.098
1977	304	1	9848	59.5	1.107
1978	137	1	9620	26.8	1.098
1979	38		10747	7.4	1.097
Assembly Type Totals	1142	385	9373	222.4	1.559
Assembly Type: GE BWR/4-6 7 X 7 GE-3a					
1975	2	2	4653	0.4	2.120
1976	104	19	7947	19.5	1.389
1977	397	21	14032	74.4	1.993
1978	575	24	20874	107.8	2.410
1979	462	10	18787	86.6	2.312
1980	1233	22	23312	231.2	2.434
1981	490	8	24304	91.9	2.450
1982	337	4	23143	63.1	2.294
1983	12		22942	2.2	2.334
1984	64		24425	12.0	2.334
1985	76		21396	14.2	2.334
Assembly Type Totals	3752	110	21057	703.3	2.325
Assembly Type: GE BWR/4-6 7 X 7 GE-3b					
1976	191	3	10570	36.4	2.056
1977	1		19646	0.2	2.507
1978	312	3	20538	59.1	2.150
1979	435	14	25645	82.6	2.414
1980	112		26645	21.3	2.507
1981	97		25146	18.4	2.504
1982	36		26674	6.8	2.506
Assembly Type Totals	1184	20	21943	224.9	2.306
Assembly Type: GE BWR/4-6 8 X 8 ANF					
1989	220		26708	38.8	2.741
1990	512		29675	90.4	2.863
Assembly Type Totals	732		28784	129.1	2.826

* As reported by the utilities

Table 2.3.4. (continued)

LWR QUANTITIES DATABASE
Historical Data through December 31, 1990
Data Broken Down By: Assembly Type, Discharge Year
Discharged Assemblies by Assembly Class: GE BWR/4-6

DISCHARGE YEAR	NUMBER OF ASSEMBLIES	DEFECTIVE ASSEMBLIES*	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
Assembly Type: GE BWR/4-6 8 X 8 GE-4a					
1977	112		18924	20.6	2.190
1978	92	8	18801	16.9	2.238
1979	158	56	20073	29.0	2.392
1980	267	73	24241	49.1	2.670
1981	392	19	24825	72.2	2.650
1982	271	8	26187	49.9	2.735
1983	300	2	27497	55.3	2.736
1984	333	10	28652	61.2	2.763
1985	19		24842	3.5	2.730
Assembly Type Totals	1944	176	24993	357.6	2.631
Assembly Type: GE BWR/4-6 8 X 8 GE-4b					
1978	3	2	13892	0.6	2.192
1979	137	4	17153	25.6	2.177
1980	621		19261	116.0	2.164
1981	479		22602	89.5	2.422
1982	262	2	22328	48.9	2.194
1983	91		29507	17.0	2.685
1984	146	1	25220	27.3	2.739
1985	48		16416	9.0	2.114
Assembly Type Totals	1787	9	21368	333.7	2.311
Assembly Type: GE BWR/4-6 8 X 8 GE-5					
1980	78	2	2848	14.3	0.762
1981	33	12	20093	6.0	2.655
1982	220	44	22088	40.0	2.487
1983	950	18	27626	173.9	2.706
1984	772	27	27108	141.3	2.678
1985	630	1	19046	115.5	2.035
1986	48		27421	8.8	2.263
1987	264		12636	48.3	1.546
1988	644		23423	117.8	2.256
1989	343		27862	62.9	2.548
1990	216		23405	39.5	2.123
Assembly Type Totals	4198	104	23645	768.3	2.362

* As reported by the utilities

Table 2.3.4. (continued)

LWR QUANTITIES DATABASE
Historical Data through December 31, 1990
Data Broken Down By: Assembly Type, Discharge Year
Discharged Assemblies by Assembly Class: GE BWR/4-6

DISCHARGE YEAR	NUMBER OF ASSEMBLIES	DEFECTIVE ASSEMBLIES*	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
Assembly Type: GE BWR/4-6 8 X 8 GE-6					
1981	26	2	23200	4.8	2.883
1982	1		9565	0.2	2.861
1983	198	4	26833	36.2	2.806
1984	556	39	18516	101.7	2.818
1985	1437	96	23946	262.7	2.626
1986	1231		14792	225.9	1.768
1987	2583		18708	473.0	2.362
1988	1263	3	24977	231.3	2.462
1989	1329		20432	243.9	2.025
1990	929		20432	170.9	2.037
Assembly Type Totals	9553	144	20394	1750.6	2.297
Assembly Type: GE BWR/4-6 8 X 8 GE-7					
1984	1	1	5508	0.2	2.657
1986	8		16829	1.5	2.659
1987	225		8910	41.7	1.287
1988	322	1	15369	59.4	1.936
1989	1417	17	18086	261.7	2.227
1990	1329	5	24649	245.2	2.474
Assembly Type Totals	3302	24	19826	609.6	2.235
Assembly Type: GE BWR/4-6 9 X 9 ANF Prepress.					
1989	1		24000	0.2	3.319
Assembly Type Totals	1		24000	0.2	3.319
— TOTALS	27595	972	21094	5099.6	2.309

* As reported by the utilities

Table 2.3.5. Historically discharged assemblies for the WE 17 X 17 Assembly Class, broken down by Assembly Type and Discharge Year (reproduced from the LWR Quantities Database).

LWR QUANTITIES DATABASE
Historical Data through December 31, 1990
Data Broken Down By: Assembly Type, Discharge Year
Discharged Assemblies by Assembly Class: WE 17 X 17

DISCHARGE YEAR	NUMBER OF ASSEMBLIES	DEFECTIVE ASSEMBLIES*	AVERAGE BURNUP (Mwd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
Assembly Type: WE 17 X 17 ANF					
1986	60		37000	24.1	3.642
1988	71		38573	28.5	3.642
1990	85		38914	34.2	3.767
Assembly Type Totals	216		38271	86.8	3.691
Assembly Type: WE 17 X 17 B&W					
1990	1		28000	0.5	2.900
Assembly Type Totals	1		28000	0.5	2.900
Assembly Type: WE 17 X 17 WE LOPAR					
1976	2		16128	0.9	1.860
1978	1	1	15329	0.5	2.116
1979	243	3	17005	111.7	2.187
1980	233		27007	107.1	2.580
1981	209	2	29092	96.1	2.834
1982	436	17	25882	200.2	2.740
1983	415	16	24440	190.5	2.622
1984	795	26	27949	364.7	2.850
1985	465	1	28986	213.5	2.904
1986	812	6	28259	373.2	2.907
1987	850	2	30518	391.9	3.047
1988	653	6	30072	301.3	2.975
1989	838		34162	387.4	3.287
1990	892	5	33566	412.8	3.278
Assembly Type Totals	6844	85	29344	3151.8	2.952

* As reported by the utilities.

Table 2.3.5. (continued)

LWR QUANTITIES DATABASE
Historical Data through December 31, 1990
Data Broken Down By: Assembly Type, Discharge Year
Discharged Assemblies by Assembly Class: WE 17 X 17

DISCHARGE YEAR	NUMBER OF ASSEMBLIES	DEFECTIVE ASSEMBLIES*	AVERAGE BURNUP (Mwd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
Assembly Type: WE 17 X 17 WE OFA					
1984	5		36095	2.1	3.020
1986	67	1	14311	28.4	1.680
1987	278		21951	117.9	2.338
1988	271		33975	114.9	3.103
1989	395		27543	168.4	2.705
1990	623		32712	264.7	3.053
Assembly Type Totals	1639	1	29108	696.4	2.800
Assembly Type: WE 17 X 17 WE VANTAGE 5					
1988	4		46000	1.7	3.423
1990	87		39000	37.0	3.888
Assembly Type Totals	91		39308	38.7	3.868
---Grand Totals	8791	86	29595	3974.2	2.951

* As reported by the utilities.

Table 2.3.6. Projected quantities of spent fuel from GE BWR/4-6 Assembly Class, broken down by discharge year and burnup bin (Reproduced from the LWR Quantities Database)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: GE BWR/4-6

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
1991	0- 5000	4	3000	0.7	0.711
1991	15001-20000	132	17310	23.5	1.650
1991	20001-25000	572	23007	102.1	2.300
1991	25001-30000	1100	28359	200.8	2.695
1991	30001-35000	879	33251	157.2	3.082
1991	35001-40000	4	36000	0.7	2.990
1992	0- 5000	52	4077	9.2	0.940
1992	10001-15000	96	13327	17.0	1.416
1992	15001-20000	224	18438	39.5	1.728
1992	20001-25000	584	22416	103.7	2.208
1992	25001-30000	610	28392	111.2	2.599
1992	30001-35000	1421	32791	252.5	2.992
1992	35001-40000	113	36381	20.1	3.201
1993	20001-25000	176	24122	31.8	1.868
1993	25001-30000	788	28792	143.4	2.672
1993	30001-35000	1597	32959	286.7	3.060
1993	35001-40000	96	36000	16.6	3.310
1994	15001-20000	180	17600	31.9	1.950
1994	25001-30000	768	28610	138.8	2.806
1994	30001-35000	1449	32816	258.5	3.064
1994	35001-40000	575	37059	102.5	3.167
1995	25001-30000	264	29143	48.0	2.729
1995	30001-35000	2001	33192	360.5	3.050
1995	35001-40000	375	36128	67.2	3.324
1996	25001-30000	656	29140	118.9	2.834
1996	30001-35000	1114	32882	198.6	3.044
1996	35001-40000	1268	36340	224.8	3.213
1996	40001-45000	4	42000	0.7	3.140
1997	15001-20000	36	16000	6.4	3.200
1997	25001-30000	79	28206	14.0	2.845
1997	30001-35000	1717	33525	309.7	3.091
1997	35001-40000	1322	37011	233.9	3.267
1998	25001-30000	208	27629	37.7	3.033
1998	30001-35000	676	33538	121.7	3.041
1998	35001-40000	1666	36476	301.7	3.300
1998	40001-45000	144	41004	25.6	3.620

Table 2.3.6. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: GE BWR/4-6

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
1999	15001-20000	179	19155	32.4	2.510
1999	25001-30000	55	28562	9.7	3.011
1999	30001-35000	1312	33702	235.1	3.113
1999	35001-40000	1990	36638	354.4	3.209
2000	25001-30000	219	28024	39.4	3.062
2000	30001-35000	478	33399	86.5	3.228
2000	35001-40000	1835	36977	328.8	3.272
2001	25001-30000	198	29950	35.9	2.611
2001	30001-35000	385	33067	70.2	3.117
2001	35001-40000	1616	37293	289.7	3.282
2001	40001-45000	136	42066	24.3	3.687
2002	25001-30000	147	27931	26.7	3.083
2002	30001-35000	238	32440	43.2	3.264
2002	35001-40000	2532	37691	455.3	3.346
2002	40001-45000	557	40607	99.0	3.361
2003	25001-30000	92	28539	16.5	3.101
2003	30001-35000	91	33589	16.3	3.372
2003	35001-40000	1667	38031	298.1	3.385
2003	40001-45000	514	41087	91.6	3.356
2003	45001-50000	129	45039	22.9	3.874
2004	25001-30000	149	28315	26.9	3.077
2004	30001-35000	264	33177	48.2	3.185
2004	35001-40000	1721	38103	309.4	3.302
2004	40001-45000	223	40789	40.0	3.575
2005	25001-30000	73	28720	13.1	3.099
2005	30001-35000	310	33272	56.2	3.238
2005	35001-40000	1993	37709	357.5	3.328
2005	40001-45000	1221	41297	215.2	3.509
2006	25001-30000	145	28271	26.3	3.104
2006	30001-35000	110	34122	20.0	3.544
2006	35001-40000	1439	38228	262.3	3.333
2006	40001-45000	283	42626	51.5	3.678
2007	25001-30000	73	28938	13.2	3.114
2007	30001-35000	185	32627	33.3	3.264
2007	35001-40000	1618	37384	289.3	3.289
2007	40001-45000	1058	40925	186.5	3.510

Table 2.3.6. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: GE BWR/4-6

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2008	25001-30000	147	28547	26.7	3.092
2008	30001-35000	194	33642	35.3	3.285
2008	35001-40000	2097	37967	378.9	3.353
2008	40001-45000	913	40985	162.7	3.573
2008	45001-50000	128	45465	22.7	3.901
2009	25001-30000	73	29065	13.1	3.122
2009	30001-35000	111	32973	20.0	3.193
2009	35001-40000	1303	38147	233.4	3.297
2009	40001-45000	29	41310	5.5	3.359
2010	25001-30000	141	29221	25.5	3.164
2010	30001-35000	202	32951	36.4	3.281
2010	35001-40000	2020	38101	365.5	3.352
2010	40001-45000	1427	41145	252.7	3.512
2010	45001-50000	127	45800	22.6	3.922
2011	25001-30000	142	29416	25.7	3.205
2011	30001-35000	44	34214	8.0	3.203
2011	35001-40000	1081	38256	195.2	3.399
2011	40001-45000	315	42117	56.7	3.404
2012	15001-20000	132	16149	23.5	3.187
2012	30001-35000	272	30958	49.0	3.280
2012	35001-40000	497	38619	89.7	3.449
2012	40001-45000	1810	41647	324.5	3.561
2013	10001-15000	228	13349	41.4	2.971
2013	20001-25000	80	22262	14.5	3.048
2013	25001-30000	148	26255	26.9	3.082
2013	30001-35000	232	31593	42.1	3.170
2013	35001-40000	759	37903	138.0	3.403
2013	40001-45000	749	42575	134.4	3.494
2013	45001-50000	129	45128	22.8	3.596
2014	5001-10000	28	9457	5.1	3.217
2014	10001-15000	716	13694	128.7	3.285
2014	15001-20000	472	16632	84.0	3.363
2014	20001-25000	144	23879	25.6	3.319
2014	25001-30000	840	28250	150.8	3.445
2014	30001-35000	324	32708	57.8	3.427
2014	35001-40000	881	38378	157.3	3.486
2014	40001-45000	985	42159	176.9	3.626
2014	45001-50000	200	47572	35.2	3.903

Table 2.3.6. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: GE BWR/4-6

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2015	10001-15000	188	14900	35.2	3.529
2015	15001-20000	176	16850	31.2	3.942
2015	25001-30000	256	29811	47.2	3.783
2015	30001-35000	151	31837	26.8	3.901
2015	35001-40000	111	37691	20.0	3.454
2015	40001-45000	1691	42880	302.4	3.746
2015	45001-50000	20	45543	3.6	3.675
2016	5001-10000	16	9635	2.9	3.085
2016	10001-15000	400	14660	73.6	3.237
2016	15001-20000	16	19064	2.9	3.191
2016	25001-30000	416	28217	76.6	3.385
2016	30001-35000	67	33015	12.2	3.291
2016	35001-40000	544	38708	98.8	3.430
2016	40001-45000	1357	43210	243.8	3.677
2016	45001-50000	190	45180	33.5	3.915
2017	30001-35000	46	34166	8.1	3.364
2017	35001-40000	337	38795	61.3	3.411
2017	40001-45000	918	42736	164.4	3.590
2018	15001-20000	196	15558	36.4	3.422
2018	30001-35000	196	31284	36.4	3.605
2018	35001-40000	213	38821	38.1	3.539
2018	40001-45000	1064	43255	192.4	3.626
2018	45001-50000	536	45218	94.2	3.843
2019	30001-35000	46	33844	8.2	3.343
2019	35001-40000	233	38041	42.7	3.349
2019	40001-45000	1053	42820	186.2	3.663
2020	30001-35000	47	33104	8.4	3.298
2020	35001-40000	314	37812	56.5	3.378
2020	40001-45000	1434	42271	256.6	3.594
2020	45001-50000	203	45164	36.1	3.642
2021	30001-35000	3	31524	0.6	2.045
2021	35001-40000	304	38274	53.6	3.343
2021	40001-45000	818	42233	144.6	3.603
2021	45001-50000	145	45017	25.7	3.632
2022	10001-15000	471	13200	83.7	3.274
2022	20001-25000	116	21183	20.0	3.555
2022	25001-30000	354	28041	63.5	3.333
2022	30001-35000	164	31913	28.6	3.525
2022	35001-40000	444	37941	79.7	3.368
2022	40001-45000	1496	42142	267.2	3.591

Table 2.3.6. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: GE BWR/4-6

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2023	30001-35000	82	32669	14.6	3.246
2023	35001-40000	821	38459	148.8	3.468
2023	40001-45000	677	43486	120.9	3.742
2024	10001-15000	423	12745	76.0	2.945
2024	15001-20000	656	16928	114.8	3.263
2024	20001-25000	144	24913	25.4	2.831
2024	25001-30000	486	27659	86.3	3.378
2024	30001-35000	7	32928	1.3	2.857
2024	35001-40000	564	38382	98.5	3.255
2024	40001-45000	1111	41455	196.5	3.506
2025	10001-15000	452	13275	81.9	3.330
2025	15001-20000	292	17125	53.1	3.226
2025	20001-25000	124	22132	22.0	3.286
2025	25001-30000	553	27089	100.0	3.325
2025	30001-35000	511	33895	92.5	3.246
2025	35001-40000	1046	37165	189.2	3.324
2025	40001-45000	136	41344	24.8	3.427
2026	10001-15000	368	13219	66.5	3.430
2026	15001-20000	88	15654	16.3	3.386
2026	20001-25000	252	23731	45.0	3.451
2026	25001-30000	204	27475	37.9	3.386
2026	30001-35000	263	32853	47.2	3.393
2026	35001-40000	450	37803	81.8	3.417
2026	40001-45000	305	41464	53.0	3.375
2027	10001-15000	172	11160	31.8	3.649
2027	15001-20000	276	17827	44.8	3.802
2027	20001-25000	172	23296	31.8	3.737
2027	30001-35000	287	33351	48.8	3.898
2027	35001-40000	229	36991	42.3	3.808
2027	40001-45000	323	42790	55.3	3.759
2027	45001-50000	40	48560	7.3	3.940
2028	30001-35000	119	34508	21.0	3.285
2028	35001-40000	164	39737	29.1	3.299
2029	10001-15000	332	14537	58.8	3.210
2029	20001-25000	216	24280	38.3	3.210
2029	30001-35000	216	32858	38.3	3.210
	Grand Total	97616	35077	17503.7	3.304

Table 2.3.7. Projected quantities of spent fuel from WE 17 X 17 Assembly Class, broken down by discharge year and burnup bin (Reproduced from the LWR Quantities Database)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
1991	10001-15000	1	14000	0.5	2.100
1991	20001-25000	12	24000	5.1	3.250
1991	25001-30000	120	29528	52.1	2.796
1991	30001-35000	498	33802	221.8	3.344
1991	35001-40000	632	36742	283.1	3.335
1991	40001-45000	255	42472	113.6	3.706
1991	45001-50000	20	46643	8.6	3.600
1992	10001-15000	116	13000	50.3	1.610
1992	15001-20000	1	17000	0.4	1.486
1992	20001-25000	13	23748	5.5	3.153
1992	25001-30000	8	26000	3.7	2.800
1992	30001-35000	311	32933	138.1	3.303
1992	35001-40000	491	37676	216.9	3.580
1992	40001-45000	321	42491	145.2	3.737
1992	45001-50000	137	46458	59.4	3.734
1992	50001-55000	1	51000	0.5	3.600
1993	15001-20000	9	20000	3.6	1.610
1993	20001-25000	107	22841	45.2	2.414
1993	25001-30000	65	26000	30.0	2.411
1993	30001-35000	220	33690	96.7	3.633
1993	35001-40000	640	37797	282.2	3.594
1993	40001-45000	415	42909	181.3	3.828
1993	45001-50000	113	47695	48.9	3.938
1994	20001-25000	5	25000	2.3	2.100
1994	25001-30000	131	28456	56.5	2.784
1994	30001-35000	173	34301	76.9	3.533
1994	35001-40000	283	37855	128.3	3.668
1994	40001-45000	401	42949	180.7	3.869
1994	45001-50000	115	47397	51.2	3.846
1994	50001-55000	28	51000	13.0	3.923
1995	15001-20000	68	15952	27.4	1.635
1995	20001-25000	32	23000	13.6	2.100
1995	30001-35000	202	34257	88.6	3.480
1995	35001-40000	383	38123	165.6	3.783
1995	40001-45000	642	42755	290.4	3.951
1995	45001-50000	81	47078	36.8	4.037
1995	50001-55000	61	52713	25.3	4.250

Table 2.3.7. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
1996	15001-20000	1	19000	0.5	2.100
1996	20001-25000	3	24295	1.2	3.100
1996	25001-30000	125	28053	54.1	2.519
1996	30001-35000	118	33604	49.0	3.521
1996	35001-40000	515	38098	226.7	3.726
1996	40001-45000	493	43061	221.6	3.968
1996	45001-50000	135	47853	61.7	3.778
1996	50001-55000	105	51823	47.1	4.145
1997	20001-25000	8	24733	3.5	2.520
1997	25001-30000	64	27031	29.5	2.600
1997	30001-35000	182	34599	77.3	3.572
1997	35001-40000	599	37354	258.9	3.723
1997	40001-45000	279	42807	126.4	3.947
1997	45001-50000	136	47462	59.8	4.042
1997	50001-55000	91	52462	38.8	4.437
1998	15001-20000	1	19075	0.5	2.105
1998	25001-30000	63	28111	29.0	2.607
1998	30001-35000	154	34284	64.1	3.545
1998	35001-40000	252	38535	112.4	3.702
1998	40001-45000	540	42264	241.2	4.006
1998	45001-50000	290	47543	132.2	4.202
1998	50001-55000	99	51543	43.5	4.263
1999	30001-35000	113	34151	48.3	3.645
1999	35001-40000	679	37415	292.7	3.716
1999	40001-45000	424	42177	190.2	4.031
1999	45001-50000	262	46836	119.0	4.213
1999	50001-55000	197	52095	85.9	4.137
2000	30001-35000	40	34495	16.7	3.797
2000	35001-40000	348	37846	150.0	3.555
2000	40001-45000	374	41632	166.0	3.966
2000	45001-50000	286	47704	127.9	4.195
2000	50001-55000	124	51613	55.0	4.234
2001	30001-35000	72	34220	30.3	3.716
2001	35001-40000	442	37919	188.9	3.772
2001	40001-45000	535	41822	237.7	3.984
2001	45001-50000	428	47294	196.6	4.252
2001	50001-55000	83	52857	37.8	4.288
2001	55001-up	80	55306	33.0	4.374

Table 2.3.7. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2002	30001-35000	37	34541	15.5	3.649
2002	35001-40000	338	38330	146.3	3.845
2002	40001-45000	349	41830	157.1	4.022
2002	45001-50000	309	47114	137.2	4.167
2002	50001-55000	122	52142	54.9	4.274
2002	55001-up	39	56086	15.9	4.321
2003	30001-35000	37	34778	15.3	3.842
2003	35001-40000	188	36883	79.4	3.688
2003	40001-45000	544	42000	240.9	4.019
2003	45001-50000	547	47622	249.2	4.236
2003	50001-55000	199	53117	89.3	4.404
2003	55001-up	15	56044	6.5	4.343
2004	30001-35000	3	33719	1.1	3.526
2004	35001-40000	266	38696	112.3	3.877
2004	40001-45000	487	42586	215.9	3.977
2004	45001-50000	259	47532	115.0	4.280
2004	50001-55000	100	52275	44.1	4.430
2004	55001-up	108	57004	47.5	4.490
2005	35001-40000	173	38192	71.8	3.781
2005	40001-45000	430	42746	191.1	4.042
2005	45001-50000	305	47209	134.8	4.198
2005	50001-55000	399	51463	182.8	4.470
2005	55001-up	94	56888	40.9	4.434
2006	35001-40000	170	39038	72.5	3.899
2006	40001-45000	482	43001	212.6	4.056
2006	45001-50000	380	47029	164.3	4.382
2006	50001-55000	305	52331	136.9	4.517
2006	55001-up	208	59880	90.6	4.677
2007	40001-45000	246	42404	105.2	4.022
2007	45001-50000	269	47116	122.6	4.342
2007	50001-55000	248	53215	111.8	4.523
2007	55001-up	269	56363	123.2	4.769
2008	35001-40000	35	38999	14.7	3.925
2008	40001-45000	268	42810	113.5	4.124
2008	45001-50000	510	47419	222.9	4.353
2008	50001-55000	266	52194	118.7	4.502
2008	55001-up	275	59858	120.4	4.768

Table 2.3.7. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2009	35001-40000	2	37818	1.0	3.785
2009	40001-45000	202	43637	86.4	4.047
2009	45001-50000	371	48012	164.4	4.459
2009	50001-55000	189	52592	83.7	4.609
2009	55001-up	410	58675	185.7	4.865
2010	35001-40000	3	38674	1.3	3.476
2010	40001-45000	248	43377	105.6	4.188
2010	45001-50000	392	47849	169.4	4.340
2010	50001-55000	221	53448	98.3	4.583
2010	55001-up	268	60433	121.2	4.850
2011	35001-40000	2	37971	1.0	3.794
2011	40001-45000	112	43480	46.9	4.161
2011	45001-50000	510	48112	226.8	4.412
2011	50001-55000	237	52185	105.3	4.639
2011	55001-up	528	59475	237.5	4.884
2012	40001-45000	166	43268	70.7	4.233
2012	45001-50000	313	48081	137.6	4.397
2012	50001-55000	158	53374	71.9	4.586
2012	55001-up	255	59972	116.0	4.890
2013	35001-40000	2	37578	1.0	3.770
2013	40001-45000	172	43453	71.7	4.169
2013	45001-50000	474	47922	210.3	4.393
2013	50001-55000	243	52232	106.0	4.613
2013	55001-up	366	60063	161.6	4.866
2014	35001-40000	3	38969	1.3	3.492
2014	40001-45000	215	43677	90.7	4.138
2014	45001-50000	273	47949	119.8	4.283
2014	50001-55000	323	52303	145.0	4.640
2014	55001-up	485	58353	220.2	4.855
2015	15001-20000	48	16222	22.3	3.751
2015	30001-35000	48	32443	22.3	3.978
2015	35001-40000	2	39326	0.9	3.880
2015	40001-45000	140	43392	58.7	4.159
2015	45001-50000	495	47798	217.1	4.412
2015	50001-55000	157	52456	70.5	4.468
2015	55001-up	302	61367	133.1	4.928

Table 2.3.7. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2016	20001-25000	60	24048	27.7	4.234
2016	25001-30000	88	25526	40.7	4.058
2016	35001-40000	2	39294	0.9	3.878
2016	40001-45000	190	43864	82.6	4.105
2016	45001-50000	304	47868	132.4	4.489
2016	50001-55000	354	52139	158.6	4.574
2016	55001-up	534	59163	240.7	4.921
2017	25001-30000	65	28802	30.1	4.407
2017	40001-45000	122	44040	51.5	4.185
2017	45001-50000	376	47186	164.1	4.435
2017	50001-55000	205	52007	91.6	4.710
2017	55001-up	506	59759	227.5	4.905
2018	20001-25000	49	21656	20.7	4.494
2018	25001-30000	95	28942	41.2	4.499
2018	35001-40000	2	39408	0.9	3.885
2018	40001-45000	109	43407	45.2	4.310
2018	45001-50000	201	47914	86.5	4.357
2018	50001-55000	246	52428	109.2	4.666
2018	55001-up	451	59771	199.9	4.991
2019	40001-45000	139	43910	58.1	4.162
2019	45001-50000	287	47460	124.1	4.378
2019	50001-55000	407	52279	178.6	4.649
2019	55001-up	404	60007	180.4	4.948
2020	20001-25000	48	22019	22.1	4.705
2020	25001-30000	96	27859	44.2	4.516
2020	40001-45000	104	43718	43.8	4.148
2020	45001-50000	305	47294	130.3	4.387
2020	50001-55000	285	52504	126.0	4.733
2020	55001-up	417	61129	187.9	5.008
2021	15001-20000	41	16164	17.3	4.015
2021	20001-25000	79	22898	35.0	4.320
2021	25001-30000	140	27445	64.6	4.598
2021	30001-35000	86	31707	38.2	4.357
2021	35001-40000	2	39911	0.9	3.917
2021	40001-45000	101	44119	42.1	4.055
2021	45001-50000	289	47995	126.3	4.503
2021	50001-55000	301	52698	135.6	4.842
2021	55001-up	461	60303	208.8	4.994

Table 2.3.7. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2022	15001-20000	21	18473	9.7	4.471
2022	20001-25000	47	23224	21.8	4.201
2022	35001-40000	21	36947	9.7	4.743
2022	40001-45000	253	44043	108.3	4.163
2022	45001-50000	148	47903	66.0	4.490
2022	50001-55000	297	52390	129.7	4.637
2022	55001-up	264	59652	117.4	4.873
2023	15001-20000	41	17825	17.3	3.768
2023	20001-25000	35	22281	14.8	4.029
2023	35001-40000	43	35849	18.3	3.969
2023	40001-45000	248	43446	103.9	4.130
2023	45001-50000	186	47869	81.3	4.355
2023	50001-55000	142	52827	62.9	4.387
2023	55001-up	220	57286	99.6	4.819
2024	15001-20000	25	18626	10.6	4.023
2024	20001-25000	24	20792	10.2	4.471
2024	25001-30000	55	25320	23.4	4.742
2024	30001-35000	52	30568	22.2	4.632
2024	35001-40000	27	37422	11.6	4.250
2024	40001-45000	157	43785	66.9	4.099
2024	45001-50000	217	48031	93.8	4.508
2024	50001-55000	196	52150	84.4	4.824
2024	55001-up	275	60128	118.1	4.837
2025	15001-20000	180	18520	79.5	4.105
2025	20001-25000	1	24315	0.5	4.323
2025	25001-30000	107	27444	45.3	4.065
2025	30001-35000	110	32897	46.3	4.917
2025	35001-40000	109	36088	49.6	4.417
2025	40001-45000	88	43680	36.4	3.876
2025	45001-50000	254	47270	110.0	4.294
2025	50001-55000	152	52570	67.0	4.508
2025	55001-up	391	58847	176.4	5.011
2026	15001-20000	41	17178	17.3	4.023
2026	20001-25000	35	22318	14.8	4.093
2026	25001-30000	84	29904	38.8	4.477
2026	30001-35000	41	34357	17.3	4.259
2026	35001-40000	2	39149	0.9	3.869
2026	40001-45000	35	44637	14.8	4.400
2026	45001-50000	158	46693	67.6	4.280
2026	50001-55000	157	51713	70.9	4.704
2026	55001-up	66	60934	30.0	4.920

Table 2.3.7. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2027	15001-20000	65	15387	27.5	3.883
2027	20001-25000	252	21826	112.5	4.092
2027	25001-30000	55	26289	25.5	4.343
2027	30001-35000	118	30931	52.1	4.325
2027	35001-40000	2	39212	0.9	3.873
2027	40001-45000	128	42671	56.5	3.844
2027	45001-50000	298	46450	127.0	4.287
2027	50001-55000	153	53637	71.4	4.599
2027	55001-up	170	58509	77.8	4.798
2028	15001-20000	25	17994	10.6	4.316
2028	25001-30000	59	26658	25.0	4.283
2028	35001-40000	69	37897	28.3	3.835
2028	40001-45000	48	42871	22.1	4.255
2028	45001-50000	98	47435	41.7	3.995
2028	50001-55000	158	52756	69.9	4.543
2028	55001-up	13	59003	6.2	4.383
2029	20001-25000	80	22918	37.1	4.308
2029	35001-40000	43	38520	17.2	3.322
2029	40001-45000	35	41442	16.3	4.382
2029	45001-50000	105	47263	48.1	4.215
2029	50001-55000	46	52532	21.2	4.279
2029	55001-up	41	60938	19.2	4.535
2030	10001-15000	64	14426	25.8	3.603
2030	15001-20000	63	16035	29.1	4.242
2030	30001-35000	1	31515	0.5	4.460
2030	35001-40000	129	37291	55.8	4.245
2030	45001-50000	53	48673	22.1	4.059
2030	50001-55000	160	51529	70.1	4.440
2030	55001-up	15	59634	6.7	4.436
2031	35001-40000	2	38932	1.0	3.855
2031	45001-50000	70	48023	29.6	4.108
2031	50001-55000	17	50949	8.0	4.064
2031	55001-up	13	55801	6.1	4.167
2032	20001-25000	36	22528	16.6	4.176
2032	25001-30000	44	25317	20.3	3.948
2032	35001-40000	2	38684	1.0	3.839
2032	40001-45000	33	43596	15.2	4.453
2032	45001-50000	101	47918	43.7	4.126
2032	50001-55000	33	51862	15.0	4.378
2032	55001-up	46	56610	21.1	4.534

Table 2.3.7. (continued)

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Data Broken Down By: Discharge Year and Burnup Bin
Projected Assemblies for Assembly Class: WE 17 X 17

DISCHARGE YEAR	BURNUP BIN	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICHMENT
2033	10001-15000	64	14643	25.8	3.560
2033	40001-45000	88	41075	37.2	3.986
2033	45001-50000	19	45791	8.9	3.739
2033	50001-55000	80	50813	33.0	4.064
2035	15001-20000	36	17820	16.6	3.879
2035	20001-25000	44	20027	20.3	3.614
2035	30001-35000	33	34486	15.2	3.879
2035	35001-40000	47	37987	21.7	3.631
2035	40001-45000	33	44313	15.2	3.879
	Grand Total	49808	45448	21979.7	4.211

Table 2.3.8. Historical quantities of spent fuel from Monticello, broken down by storage pool and discharge year (Reproduced from the LWR Quantities Database).

LWR QUANTITIES DATABASE
Historical Data through December 31, 1990
Data Broken Down By: Storage Pool, Discharge Year
Discharged Assemblies by Reactor: Monticello

DISCHARGE YEAR	NUMBER OF ASSEMBLIES	DEFECTIVE ASSEMBLIES*	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
		Pool: MONTICELLO			
1973	2	2	8198	0.4	2.250
1974	11	11	12307	2.1	2.250
1975	35	33	16492	6.8	2.250
1980	1		32651	0.2	2.620
1981	4		37050	0.7	2.620
1982	8		35681	1.4	2.620
1984	1		26240	0.2	2.620
1986	120		28333	21.6	2.621
1987	136		28222	24.4	2.569
1989	128		33719	22.7	2.769
--Pool Totals	446	46	28515	80.5	2.604
		Pool: MORRIS (GE)			
1973	11	11	8042	2.1	2.250
1974	111	75	12709	21.5	2.250
1975	314	93	16855	60.8	2.250
1977	20		23839	3.7	2.300
1978	8		18490	1.5	2.190
1980	147		23551	27.1	2.345
1981	100		24464	18.4	2.476
1982	160		25138	29.5	2.498
1984	187		27302	33.6	2.564
--Pool Totals	1058	179	21081	198.2	2.375
--- TOTALS	1504	225	23228	278.7	2.441

* As reported by the utilities

Table 2.3.9. Projected quantities of spent fuel from Monticello, broken down by discharge year (Reproduced from the LWR Quantities Database).

LWR QUANTITIES DATABASE
Projected Data
Data Broken Down By: Discharge Year
Discharged Assemblies by Reactor: Monticello

DISCHARGE YEAR	NUMBER OF ASSEMBLIES	AVERAGE BURNUP (MWd/MTIHM)	TOTAL WEIGHT (MTIHM)	AVERAGE INITIAL ENRICH.
1991	136	34824	23.4	2.858
1993	136	34529	23.4	3.043
1995	136	37618	23.4	3.163
1997	136	37559	23.4	3.173
1999	136	38588	23.4	3.244
2001	132	39040	22.8	3.273
2002	126	41073	21.7	3.400
2004	126	41032	21.7	3.398
2006	125	41194	21.6	3.408
2008	125	41519	21.4	3.429
2010	121	41825	20.7	3.448
2011	484	26797	83.2	3.427
— TOTALS	1919	35837	330.1	3.296

2.4 RADIOLOGICAL PROPERTIES OF SPENT FUEL

2.4.1 Overview

The design, licensing, and operation of the waste management system for interim storage and long-term disposal of LWR spent fuel requires detailed knowledge of the radiological properties of the fuel as a function of time. For example, radionuclide inventories are needed for performance assessments, radioactivities are needed for safety analyses, and decay heat generation rates are needed for storage cask and repository design. These data must be available for PWR and BWR fuels having a variety of burnups and enrichments and covering a wide range of cooling times. This section presents a summary of such radiological data for intact LWR spent fuel assemblies calculated using the ORIGEN2 code (Croff 1980.) The fuel assembly hardware is included in these tables, using a generic BWR assembly and a generic PWR assembly; non-fuel assembly (NFA) hardware was included in the neutronic modeling of the core in the preparation of the ORIGEN2 cross-section libraries, but is not included with spent fuel since it is not an integral component of fuel assemblies. The radiological properties of spent fuel disassembly hardware are discussed in Section 2.7. The ORIGEN2 code provides "core-average" results; i.e. the average composition of the entire core. Local variations such as end-effects and burnable absorbers were factored into the core during the calculation of effective cross-sections (Ludwig 1989). The complete, detailed collection of spent fuel data is available from the LWR Radiological Database. Graphical presentation of these data is quite helpful in noting trends; a representative collection of such graphs is being prepared and will be published (Moore 1992).

The radiological properties in the database include the composition (grams/MTIHM), radioactivity (curies/MTIHM), thermal power (watts/MTIHM), and neutron production rate (neutrons/sec/MTIHM), all by nuclide, and the photon production rate (photons/sec/MTIHM) by energy group. Totals for each quality are also included. These properties were calculated for 36 basic combinations of reactor type, burnup, and initial enrichment, each at 23 cooling times. The burnup ranged from 7,500 to 50,000 MWd/MTIHM for BWRs and from 10,000 to 60,000 MWD/MTIHM for PWRs, with corresponding enrichments. Cooling times range from one year to one million years. Table 2.4.1 lists the 36 reactor, burnup, and enrichment combinations and the reactor cycle parameters used in the ORIGEN2 calculations.

Tables 2.4.2 to 2.4.9 present total radioactivity, thermal power, neutron production rate, and photon production rate for all 36 of the basic combinations of reactor, burnup, and initial enrichment at six cooling times. Tables 2.4.10 to 2.4.15 list the radionuclides that contribute one percent or more to the radioactivity, thermal power, and neutron production rate for a 40,000 MWD/MTIHM PWR and a 30,000 MWD/MTIHM BWR at six cooling times, and gives the contribution per MTIHM for each of these radionuclides. These specific cases are representative of the conditions anticipated for spent fuel inventories. For more detailed information or data at other conditions, the reader should query the LWR Radiological Database. This database has been redesigned to include an interpolation feature, described in Appendix 1C, that will permit the user to obtain radiological characteristics at any intermediate combination of burnup, enrichment, and cooling time within the range of the basic combinations.

A derived quantity of interest for thermal analysis is the "integral heat"; i.e. the integral of instantaneous thermal output over a specified period of time. This is available to users via the LWR Radiological Database as a menu option, with freedom of choice regarding both initial and ending time. An improved calculational procedure is used, base on an analytical expression for the heat output, rather than the trapezoidal approximation used previously.

2.4.2 Methodology

The ORIGEN2 computer code was used to calculate the radiological properties of spent fuel. (See Appendix 1A for an overview of ORIGEN2 and Appendix 1B for a listing of decay library data.) An ORIGEN2 calculation requires the selection of the appropriate cross-section set, fuel design, and reactor cycle. One-group cross-section sets have been calculated for specific reactors, fuels, and burnups using sophisticated reactor-physics codes and are available for a number of combinations of these. The LWR radiological calculations for this work used recently updated BWR and PWR standard-burnup (27,500 and 33,000 MWd/MTIHM respectively) and extended-burnup (50,000 and 60,000 MWd/MTIHM respectively) cross sections (Ludwig 1989). The WE 17 X 17 WE LOPAR and the GE BWR/4-6 8 X 8 GE-5 fuel assembly types were used as the reference PWR and BWR designs. Although the physical design and materials of construction will vary for assemblies from vendor to vendor or by fuel class, these differences have very little influence on the radiological properties of intact spent fuel. (This is obviously not the case for individual hardware components however, which are dealt with specifically in the LWR Assemblies Database.)

A recent study of the important parameters in ORIGEN2 calculations (Welch 1990) has shown the significance of enrichment. Therefore, improved coverage of enrichment has been incorporated in this revision. Calculations were carried out at three enrichments - a low value, a mid-range value, and a high value - for each burnup (see Table 2.4.1). The mid-range enrichment was determined from a regression on EIA historical and projected data and the enrichments used encompass most fuels of interest.

The cycle characteristics listed in Table 2.4.1, number of cycles, cycle length, downtime, and power level, reflect recently published EPRI data on reactor operations (Koppe 1989). The power levels given in Table 2.4.1 are

for the core average. In actual operation, individual assemblies will deviate from these average values at various times during their overall lifetime. The ORIGEN2 code provides average values for equilibrium fuel which has completed its planned exposure.

The major effect of these improvements was to "smooth" the discontinuity present in the level of actinides reported by the original LWR Radiological Database and in derivative properties such as neutron source strength and thermal output. The smoothing was very effective for the PWR cases, but the BWR cases still have a small "shoulder," due to the greater complexity in modeling BWRs. This aspect deserves further attention in the future.

2.4.3 References for Section 2.4

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

Koppe 1989. R. H. Koppe and E. A. J. Olson, *The Influence of Fuel-Cycle Duration on Nuclear Unit Performance - an Update*, EPRI NP-6333, April 1989.

Ludwig 1989. S. B. Ludwig and J. P. Renier, *Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code*, ORNL/TM-11018, December 1989.

Moore 1992. R. S. Moore and K. J. Notz, *Spent Fuel Decay Properties: Graphical Presentation of ORIGEN2 Data from CDB Data Files,* ONRL/TM-12149, in preparation.

Welch 1990. T. D. Welch, K. J. Notz, and R. J. Andermann, Jr., "Important Paramenters in ORIGEN2 Calculations of Spent Fuel Compositions" in *Proceedings of the International Topical Meeting on Nuclear and Hazardous Waste Management SPECTRUM '90,* Knoxville, TN, October 1990.

Table 2.4.1 Fuel parameters and reactor cycle characteristics for LWR Spent Fuel Radiological ORIGEN2 calculations.[a]

Burnup, MWd/MTIHM	Initial Enrichment, % low	mid	high	No. of cycles	Cycle length, days	Downtown between cycles, days[b]	Power level, MW/MT
BWR standard-burnup cross sections							
7,500	0.72	1.05	1.75	1	300	80	25
15,000	1.09	1.79	2.49	2	300	80	25
22,500	1.72	2.42	3.12	3	300	80	25
30,000	2.23	2.93	3.63	4	300	80	25
BWR high-burnup cross sections							
40,000	2.74	3.44	4.14	4	400	170	25
50,000	3.04	3.74	4.44	4	500	170	25
PWR standard-burnup cross sections							
10,000	0.99	1.69	2.39	1	300	80	33.3
20,000	1.74	2.44	3.14	2	300	80	33.3
30,000	2.41	3.11	3.81	3	300	80	33.3
PWR high-burnup cross sections							
40,000	3.02	3.72	4.42	3	400	120	33.3
50,000	3.56	4.26	4.96	3	500	120	33.3
60,000	4.03	4.73	5.43	3	600	120	33.3

[a]The burnup, initial-enrichment combinations shown here are the 36 basic combinations used in the LWR Radiological Data Base. Burnup is the product of cycle length, number of cycles, and power level.

[b]Based on utility data (Koppe 1989).

Table 2.4.2. Summary of Radioactivity (curies/MTIHM) of BWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 7,500 MWd/MTIHM			
0.72	1.055E+06	1.316E+05	1.202E+04	1.066E+03	2.767E+02	2.624E+01
1.05	9.523E+05	1.137E+05	1.076E+04	8.601E+02	2.531E+02	2.489E+01
1.75	8.879E+05	1.004E+05	1.005E+04	6.233E+02	2.192E+02	2.449E+01
			BURNUP = 15,000 MWd/MTIHM			
1.09	1.445E+06	2.153E+05	2.038E+04	1.467E+03	3.405E+02	3.532E+01
1.79	1.366E+06	2.040E+05	1.999E+04	1.228E+03	3.155E+02	3.477E+01
2.49	1.310E+06	1.939E+05	1.958E+04	1.014E+03	2.911E+02	3.467E+01
			BURNUP = 22,500 MWd/MTIHM			
1.72	1.696E+06	2.911E+05	2.881E+04	1.675E+03	3.933E+02	4.434E+01
2.42	1.631E+06	2.849E+05	2.880E+04	1.482E+03	3.694E+02	4.393E+01
3.12	1.575E+06	2.774E+05	2.858E+04	1.291E+03	3.466E+02	4.371E+01
			BURNUP = 30,000 MWd/MTIHM			
2.23	1.890E+06	3.628E+05	3.706E+04	1.859E+03	4.525E+02	5.361E+01
2.93	1.828E+06	3.595E+05	3.727E+04	1.692E+03	4.265E+02	5.309E+01
3.63	1.773E+06	3.543E+05	3.721E+04	1.519E+03	4.022E+02	5.266E+01
			BURNUP = 40,000 MWd/MTIHM			
2.74	2.396E+06	5.132E+05	5.357E+04	2.530E+03	6.400E+02	7.804E+01
3.44	2.338E+06	5.121E+05	5.410E+04	2.382E+03	6.079E+02	7.771E+01
4.14	2.280E+06	5.092E+05	5.432E+04	2.224E+03	5.780E+02	7.694E+01
			BURNUP = 50,000 MWd/MTIHM			
3.04	2.383E+06	5.601E+05	5.953E+04	2.677E+03	7.018E+02	8.697E+01
3.74	2.332E+06	5.593E+05	6.013E+04	2.531E+03	6.675E+02	8.675E+01
4.44	2.279E+06	5.571E+05	6.041E+04	2.375E+03	6.349E+02	8.591E+01

Table 2.4.3. Summary of Thermal Output (watts/MTIHM) of BWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 7,500 MWd/MTIHM			
0.72	4.081E+03	2.887E+02	1.258E+02	3.404E+01	8.278E+00	5.958E-01
1.05	3.656E+03	2.602E+02	1.002E+02	2.737E+01	7.587E+00	5.676E-01
1.75	3.385E+03	2.475E+02	7.416E+01	1.969E+01	6.561E+00	5.527E-01
			BURNUP = 15,000 MWd/MTIHM			
1.09	5.818E+03	5.277E+02	1.954E+02	4.669E+01	1.002E+01	7.455E-01
1.79	5.359E+03	5.055E+02	1.670E+02	3.904E+01	9.321E+00	7.299E-01
2.49	5.046E+03	4.921E+02	1.420E+02	3.212E+01	8.602E+00	7.231E-01
			BURNUP = 22,500 MWd/MTIHM			
1.72	7.089E+03	8.003E+02	2.483E+02	5.301E+01	1.142E+01	8.879E-01
2.42	6.606E+03	7.670E+02	2.250E+02	4.696E+01	1.077E+01	8.733E-01
3.12	6.223E+03	7.442E+02	2.012E+02	4.082E+01	1.012E+01	8.640E-01
			BURNUP = 30,000 MWd/MTIHM			
2.23	8.221E+03	1.105E+03	3.021E+02	5.843E+01	1.299E+01	1.038E+00
2.93	7.694E+03	1.050E+03	2.812E+02	5.331E+01	1.231E+01	1.019E+00
3.63	7.245E+03	1.011E+03	2.583E+02	4.786E+01	1.164E+01	1.003E+00
			BURNUP = 40,000 MWd/MTIHM			
2.74	1.136E+04	1.908E+03	4.644E+02	7.880E+01	1.814E+01	1.512E+00
3.44	1.075E+04	1.784E+03	4.469E+02	7.443E+01	1.733E+01	1.495E+00
4.14	1.018E+04	1.687E+03	4.244E+02	6.963E+01	1.654E+01	1.469E+00
			BURNUP = 50,000 MWd/MTIHM			
3.04	1.173E+04	2.227E+03	5.197E+02	8.301E+01	1.984E+01	1.675E+00
3.74	1.113E+04	2.080E+03	5.028E+02	7.874E+01	1.896E+01	1.660E+00
4.44	1.055E+04	1.959E+03	4.802E+02	7.408E+01	1.811E+01	1.632E+00

Table 2.4.4. Summary of Total Neutrons (neutrons/second/MTIHM) from BWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 7,500 MWd/MTIHM			
0.72	3.419E+07	1.757E+07	4.573E+06	1.991E+06	7.595E+05	2.254E+05
1.05	1.573E+07	7.779E+06	3.322E+06	1.582E+06	5.990E+05	1.419E+05
1.75	5.884E+06	2.955E+06	2.154E+06	1.132E+06	4.378E+05	7.795E+04
			BURNUP = 15,000 MWd/MTIHM			
1.09	1.768E+08	9.519E+07	9.390E+06	3.166E+06	1.291E+06	4.996E+05
1.79	7.920E+07	3.820E+07	6.186E+06	2.461E+06	9.715E+05	3.156E+05
2.49	3.836E+07	1.665E+07	4.416E+06	1.970E+06	7.627E+05	2.029E+05
			BURNUP = 22,500 MWd/MTIHM			
1.72	3.838E+08	2.212E+08	1.547E+07	4.390E+06	1.810E+06	7.044E+05
2.42	2.015E+08	1.067E+08	1.017E+07	3.372E+06	1.379E+06	4.987E+05
3.12	1.087E+08	5.233E+07	7.201E+06	2.737E+06	1.097E+06	3.497E+05
			BURNUP = 30,000 MWd/MTIHM			
2.23	7.175E+08	4.391E+08	2.558E+07	6.552E+06	2.600E+06	9.139E+05
2.93	4.089E+08	2.358E+08	1.656E+07	4.688E+06	1.918E+06	6.885E+05
3.63	2.351E+08	1.260E+08	1.140E+07	3.671E+06	1.504E+06	5.110E+05
			BURNUP = 40,000 MWd/MTIHM			
2.74	2.304E+09	1.527E+09	8.018E+07	2.196E+07	7.254E+06	1.460E+06
3.44	1.547E+09	1.006E+09	5.442E+07	1.398E+07	4.873E+06	1.178E+06
4.14	1.024E+09	6.486E+08	3.750E+07	9.436E+06	3.465E+06	9.536E+05
			BURNUP = 50,000 MWd/MTIHM			
3.04	3.007E+09	2.002E+09	1.073E+08	3.196E+07	1.018E+07	1.688E+06
3.74	2.089E+09	1.369E+09	7.393E+07	2.021E+07	6.730E+06	1.343E+06
4.44	1.428E+09	9.141E+08	5.128E+07	1.327E+07	4.647E+06	1.088E+06

Table 2.4.5. Summary of Total Photons (photons/second/MTIHM) from BWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 7,500 MWd/MTIHM			
0.72	3.287E+16	1.866E+15	2.478E+14	1.599E+13	1.108E+12	2.232E+11
1.05	2.961E+16	1.748E+15	2.250E+14	1.196E+13	9.376E+11	2.045E+11
1.75	2.756E+16	1.757E+15	2.159E+14	7.623E+12	7.578E+11	2.027E+11
			BURNUP = 15,000 MWd/MTIHM			
1.09	4.564E+16	3.383E+15	4.272E+14	2.363E+13	1.735E+12	3.291E+11
1.79	4.276E+16	3.439E+15	4.265E+14	1.857E+13	1.394E+12	3.168E+11
2.49	4.070E+16	3.485E+15	4.246E+14	1.432E+13	1.173E+12	3.122E+11
			BURNUP = 22,500 MWd/MTIHM			
1.72	5.451E+16	5.082E+15	6.114E+14	2.728E+13	2.333E+12	4.283E+11
2.42	5.174E+16	5.159E+15	6.193E+14	2.304E+13	1.900E+12	4.189E+11
3.12	4.943E+16	5.210E+15	6.222E+14	1.907E+13	1.600E+12	4.118E+11
			BURNUP = 30,000 MWd/MTIHM			
2.23	6.198E+16	6.787E+15	7.891E+14	3.033E+13	3.058E+12	5.296E+11
2.93	5.914E+16	6.869E+15	8.023E+14	2.652E+13	2.506E+12	5.204E+11
3.63	5.658E+16	6.922E+15	8.098E+14	2.281E+13	2.103E+12	5.104E+11
			BURNUP = 40,000 MWd/MTIHM			
2.74	8.217E+16	1.001E+16	1.120E+15	4.218E+13	5.234E+12	8.098E+11
3.44	7.936E+16	1.012E+16	1.140E+15	3.862E+13	4.476E+12	8.059E+11
4.14	7.652E+16	1.020E+16	1.156E+15	3.500E+13	3.836E+12	7.932E+11
			BURNUP = 50,000 MWd/MTIHM			
3.04	8.224E+16	1.115E+16	1.237E+15	4.408E+13	5.943E+12	9.161E+11
3.74	7.976E+16	1.126E+16	1.260E+15	4.062E+13	5.144E+12	9.140E+11
4.44	7.712E+16	1.135E+16	1.276E+15	3.709E+13	4.447E+12	9.010E+11

Table 2.4.6. Summary of Radioactivity (curies/MTIHM) of PWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 10,000 MWd/MTIHM			
0.99	1.295E+06	1.542E+05	1.450E+04	1.133E+03	3.071E+02	2.952E+01
1.69	1.191E+06	1.360E+05	1.346E+04	8.422E+02	2.703E+02	2.854E+01
2.39	1.145E+06	1.266E+05	1.300E+04	6.459E+02	2.389E+02	2.852E+01
			BURNUP = 20,000 MWd/MTIHM			
1.74	1.828E+06	2.634E+05	2.597E+04	1.527E+03	3.862E+02	4.132E+01
2.44	1.756E+06	2.550E+05	2.578E+04	1.310E+03	3.600E+02	4.095E+01
3.14	1.701E+06	2.466E+05	2.547E+04	1.110E+03	3.341E+02	4.090E+01
			BURNUP = 30,000 MWd/MTIHM			
2.41	2.180E+06	3.629E+05	3.719E+04	1.799E+03	4.660E+02	5.298E+01
3.11	2.110E+06	3.589E+05	3.732E+04	1.619E+03	4.388E+02	5.254E+01
3.81	2.051E+06	3.533E+05	3.722E+04	1.436E+03	4.125E+02	5.225E+01
			BURNUP = 40,000 MWd/MTIHM			
3.02	2.501E+06	4.735E+05	4.926E+04	2.297E+03	5.977E+02	7.035E+01
3.72	2.435E+06	4.703E+05	4.949E+04	2.125E+03	5.677E+02	6.973E+01
4.42	2.374E+06	4.656E+05	4.948E+04	1.948E+03	5.392E+02	6.903E+01
			BURNUP = 50,000 MWd/MTIHM			
3.56	2.789E+06	5.668E+05	6.033E+04	2.559E+03	7.020E+02	8.391E+01
4.26	2.723E+06	5.652E+05	6.073E+04	2.395E+03	6.677E+02	8.330E+01
4.96	2.658E+06	5.620E+05	6.087E+04	2.223E+03	6.347E+02	8.239E+01
			BURNUP = 60,000 MWd/MTIHM			
4.03	3.045E+06	6.552E+05	7.114E+04	2.817E+03	8.217E+02	9.817E+01
4.73	2.981E+06	6.548E+05	7.174E+04	2.658E+03	7.829E+02	9.784E+01
5.43	2.916E+06	6.532E+05	7.204E+04	2.487E+03	7.448E+02	9.690E+01

Table 2.4.7. Summary of Thermal Output (watts/MTIHM) of PWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 10,000 MWd/MTIHM			
0.99	5.071E+03	3.593E+02	1.368E+02	3.605E+01	9.161E+00	6.615E-01
1.69	4.603E+03	3.359E+02	1.035E+02	2.666E+01	8.069E+00	6.355E-01
2.39	4.390E+03	3.268E+02	8.283E+01	2.030E+01	7.111E+00	6.278E-01
			BURNUP = 20,000 MWd/MTIHM			
1.74	7.430E+03	6.915E+02	2.143E+02	4.838E+01	1.132E+01	8.504E-01
2.44	6.974E+03	6.707E+02	1.894E+02	4.148E+01	1.058E+01	8.369E-01
3.14	6.634E+03	6.554E+02	1.661E+02	3.502E+01	9.817E+00	8.302E-01
			BURNUP = 30,000 MWd/MTIHM			
2.41	9.270E+03	1.068E+03	2.838E+02	5.656E+01	1.347E+01	1.036E+00
3.11	8.728E+03	1.028E+03	2.624E+02	5.095E+01	1.272E+01	1.018E+00
3.81	8.281E+03	9.992E+02	2.398E+02	4.517E+01	1.198E+01	1.005E+00
			BURNUP = 40,000 MWd/MTIHM			
3.02	1.117E+04	1.539E+03	3.938E+02	7.196E+01	1.717E+01	1.381E+00
3.72	1.058E+04	1.467E+03	3.720E+02	6.672E+01	1.637E+01	1.358E+00
4.42	1.006E+04	1.412E+03	3.479E+02	6.119E+01	1.559E+01	1.333E+00
			BURNUP = 50,000 MWd/MTIHM			
3.56	1.299E+04	2.032E+03	4.778E+02	7.961E+01	2.005E+01	1.622E+00
4.26	1.235E+04	1.926E+03	4.569E+02	7.469E+01	1.915E+01	1.597E+00
4.96	1.175E+04	1.843E+03	4.325E+02	6.941E+01	1.825E+01	1.566E+00
			BURNUP = 60,000 MWd/MTIHM			
4.03	1.479E+04	2.582E+03	5.664E+02	8.705E+01	2.342E+01	1.886E+00
4.73	1.411E+04	2.441E+03	5.476E+02	8.232E+01	2.239E+01	1.866E+00
5.43	1.346E+04	2.324E+03	5.233E+02	7.720E+01	2.135E+01	1.831E+00

Table 2.4.8. Summary of Total Neutrons (neutron/second/MTIHM) from PWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 10,000 MWd/MTIHM			
0.99	3.720E+07	1.981E+07	4.913E+06	2.183E+06	8.522E+05	2.522E+05
1.69	1.278E+07	6.492E+06	3.162E+06	1.580E+06	6.112E+05	1.324E+05
2.39	5.745E+06	2.998E+06	2.210E+06	1.193E+06	4.702E+05	8.119E+04
			BURNUP = 20,000 MWd/MTIHM			
1.74	1.820E+08	1.004E+08	9.930E+06	3.473E+06	1.442E+06	5.310E+05
2.44	8.933E+07	4.508E+07	6.842E+06	2.767E+06	1.120E+06	3.551E+05
3.14	4.614E+07	2.130E+07	5.019E+06	2.258E+06	8.976E+05	2.389E+05
			BURNUP = 30,000 MWd/MTIHM			
2.41	4.737E+08	2.836E+08	1.882E+07	5.411E+06	2.235E+06	8.064E+05
3.11	2.623E+08	1.473E+08	1.256E+07	4.114E+06	1.713E+06	5.926E+05
3.81	1.477E+08	7.690E+07	8.933E+06	3.329E+06	1.374E+06	4.305E+05
			BURNUP = 40,000 MWd/MTIHM			
3.02	1.070E+09	6.757E+08	3.823E+07	9.931E+06	3.702E+06	1.071E+06
3.72	6.775E+08	4.128E+08	2.616E+07	7.040E+06	2.744E+06	8.515E+05
4.42	4.281E+08	2.496E+08	1.858E+07	5.390E+06	2.153E+06	6.707E+05
			BURNUP = 50,000 MWd/MTIHM			
3.56	1.807E+09	1.186E+09	6.477E+07	1.802E+07	6.170E+06	1.327E+06
4.26	1.205E+09	7.736E+08	4.452E+07	1.198E+07	4.329E+06	1.073E+06
4.96	7.954E+08	4.960E+08	3.124E+07	8.476E+06	3.213E+06	8.655E+05
			BURNUP = 60,000 MWd/MTIHM			
4.03	2.837E+09	1.894E+09	1.055E+08	3.321E+07	1.065E+07	1.674E+06
4.73	1.975E+09	1.306E+09	7.364E+07	2.156E+07	7.221E+06	1.333E+06
5.43	1.353E+09	8.806E+08	5.171E+07	1.446E+07	5.089E+06	1.077E+06

Table 2.4.9. Summary of Total Photons (photons/second/MTIHM) from PWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

Initial Enrichment,	Years After Discharge					
%	1	10	100	1000	10k	100k
			BURNUP = 10,000 MWd/MTIHM			
0.99	4.083E+16	2.383E+15	3.024E+14	1.659E+13	1.260E+12	2.494E+11
1.69	3.732E+16	2.353E+15	2.880E+14	1.104E+13	1.004E+12	2.356E+11
2.39	3.572E+16	2.375E+15	2.836E+14	7.576E+12	8.421E+11	2.399E+11
			BURNUP = 20,000 MWd/MTIHM			
1.74	5.880E+16	4.611E+15	5.532E+14	2.357E+13	1.975E+12	3.804E+11
2.44	5.586E+16	4.672E+15	5.571E+14	1.910E+13	1.636E+12	3.712E+11
3.14	5.359E+16	4.712E+15	5.576E+14	1.520E+13	1.398E+12	3.674E+11
			BURNUP = 30,000 MWd/MTIHM			
2.41	7.196E+16	6.936E+15	7.971E+14	2.796E+13	2.839E+12	5.063E+11
3.11	6.873E+16	7.007E+15	8.088E+14	2.408E+13	2.356E+12	4.969E+11
3.81	6.595E+16	7.050E+15	8.153E+14	2.039E+13	2.003E+12	4.889E+11
			BURNUP = 40,000 MWd/MTIHM			
3.02	8.437E+16	9.261E+15	1.045E+15	3.644E+13	4.074E+12	7.022E+11
3.72	8.120E+16	9.337E+15	1.060E+15	3.261E+13	3.469E+12	6.909E+11
4.42	7.824E+16	9.386E+15	1.070E+15	2.885E+13	2.988E+12	6.772E+11
			BURNUP = 50,000 MWd/MTIHM			
3.56	9.680E+16	1.160E+16	1.275E+15	3.990E+13	5.198E+12	8.584E+11
4.26	9.350E+16	1.169E+16	1.293E+15	3.623E+13	4.485E+12	8.480E+11
4.96	9.021E+16	1.174E+16	1.308E+15	3.257E+13	3.888E+12	8.313E+11
			BURNUP = 60,000 MWd/MTIHM			
4.03	1.086E+17	1.388E+16	1.491E+15	4.266E+13	6.417E+12	1.024E+12
4.73	1.054E+17	1.400E+16	1.516E+15	3.915E+13	5.624E+12	1.018E+12
5.43	1.020E+17	1.408E+16	1.535E+15	3.560E+13	4.924E+12	1.002E+12

Table 2.4.10. Radioactivity (in curies/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

Nuclide	Enrichment 2.23%		2.93%		3.63%	
	Radio-activity	Percent of Total	Radio-activity	Percent of Total	Radio-activity	Percent of Total
Decay Time = 1 Year						
Sr 90	5.69E+04	3.01	6.28E+04	3.44	6.76E+04	3.81
Y 90	5.69E+04	3.01	6.28E+04	3.44	6.76E+04	3.81
Zr 95	2.13E+04	1.13	2.19E+04	1.20	2.25E+04	1.27
Nb 95	4.80E+04	2.54	4.93E+04	2.70	5.05E+04	2.85
Ru106	2.46E+05	13.01	2.09E+05	11.43	1.77E+05	9.97
Rh106	2.46E+05	13.01	2.09E+05	11.43	1.77E+05	9.97
Cs134	1.07E+05	5.66	9.71E+04	5.31	8.73E+04	4.92
Cs137	9.09E+04	4.81	9.09E+04	4.97	9.08E+04	5.12
Ba137m	8.60E+04	4.55	8.59E+04	4.70	8.59E+04	4.84
Ce144	3.07E+05	16.26	3.20E+05	17.53	3.32E+05	18.74
Pr144	3.07E+05	16.26	3.20E+05	17.53	3.32E+05	18.73
Pm147	7.86E+04	4.16	8.59E+04	4.70	9.39E+04	5.29
Pu241	1.38E+05	7.32	1.24E+05	6.76	1.07E+05	6.06
Decay Time = 10 Years						
Kr 85	3.90E+03	1.08	4.23E+03	1.18	4.49E+03	1.27
Sr 90	4.59E+04	12.66	5.07E+04	14.10	5.45E+04	15.39
Y 90	4.59E+04	12.66	5.07E+04	14.10	5.45E+04	15.39
Cs134	5.19E+03	1.43	4.71E+03	1.31	4.24E+03	1.20
Cs137	7.39E+04	20.36	7.38E+04	20.53	7.38E+04	20.82
Ba137m	6.99E+04	19.27	6.98E+04	19.43	6.98E+04	19.70
Pm147	7.29E+03	2.01	7.97E+03	2.22	8.70E+03	2.46
Eu154	4.34E+03	1.20	3.73E+03	1.04	3.19E+03	0.90
Pu241	8.97E+04	24.73	8.01E+04	22.29	6.96E+04	19.66
Decay Time = 100 Years						
Sr 90	5.39E+03	14.55	5.95E+03	15.97	6.40E+03	17.21
Y 90	5.39E+03	14.55	5.95E+03	15.98	6.41E+03	17.22
Cs137	9.23E+03	24.91	9.22E+03	24.75	9.22E+03	24.77
Ba137m	8.73E+03	23.57	8.73E+03	23.42	8.72E+03	23.44
Pu238	1.40E+03	3.77	1.22E+03	3.28	1.03E+03	2.75
Pu240	4.70E+02	1.27	4.38E+02	1.18	4.04E+02	1.09
Pu241	1.18E+03	3.18	1.05E+03	2.82	9.15E+02	2.46
Am241	4.38E+03	11.83	3.92E+03	10.52	3.41E+03	9.16

Table 2.4.10. (cont.) Radioactivity (in curies/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

Nuclide	Enrichment 2.23%		2.93%		3.63%	
	Radio-activity	Percent of Total	Radio-activity	Percent of Total	Radio-activity	Percent of Total
			Decay Time = 1000 Years			
Np239	2.75E+01	1.48	1.76E+01	1.04	1.10E+01	0.73
Pu239	2.98E+02	16.02	2.93E+02	17.30	2.86E+02	18.84
Pu240	4.27E+02	22.98	3.98E+02	23.54	3.67E+02	24.19
Am241	1.05E+03	56.24	9.34E+02	55.21	8.13E+02	53.50
Am243	2.75E+01	1.48	1.76E+01	1.04	1.10E+01	0.73
			Decay Time = 10,000 Years			
Tc 99	1.14E+01	2.52	1.16E+01	2.72	1.18E+01	2.93
Np239	1.18E+01	2.61	7.57E+00	1.77	4.74E+00	1.18
Pu239	2.34E+02	51.70	2.28E+02	53.57	2.23E+02	55.32
Pu240	1.64E+02	36.35	1.53E+02	35.97	1.41E+02	35.17
Am243	1.18E+01	2.61	7.57E+00	1.77	4.74E+00	1.18
			Decay Time = 100,000 Years			
Ni 59	8.12E-01	1.52	7.04E-01	1.33	6.12E-01	1.16
Zr 93	2.14E+00	3.99	2.13E+00	4.02	2.12E+00	4.03
Nb 93m	2.03E+00	3.79	2.03E+00	3.82	2.02E+00	3.83
Tc 99	8.52E+00	15.88	8.66E+00	16.31	8.78E+00	16.68
Pb210	9.53E-01	1.78	1.03E+00	1.94	1.12E+00	2.13
Pb214	9.54E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Bi210	9.54E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Bi214	9.54E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Po210	9.54E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Po214	9.53E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Po218	9.54E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Rn222	9.54E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Ra226	9.54E-01	1.78	1.03E+00	1.95	1.12E+00	2.13
Th230	9.45E-01	1.76	1.02E+00	1.93	1.11E+00	2.11
Pa233	1.26E+00	2.35	1.17E+00	2.21	1.06E+00	2.01
U234	1.45E+00	2.70	1.56E+00	2.94	1.69E+00	3.21
Np237	1.26E+00	2.35	1.17E+00	2.21	1.06E+00	2.01
Pu239	1.79E+01	33.40	1.73E+01	32.68	1.68E+01	31.92
Pu242	1.97E+00	3.67	1.47E+00	2.77	1.07E+00	2.03

Table 2.4.11. Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

Nuclide	Enrichment 2.23%		2.93%		3.63%	
	Decay heat	Percent of Total	Decay heat	Percent of Total	Decay heat	Percent of Total
			Decay Time = 1 Year			
Co 60	9.86E+01	1.20	8.58E+01	1.11	7.46E+01	1.03
Sr 90	6.60E+01	0.80	7.29E+01	0.95	7.84E+01	1.08
Y 90	3.16E+02	3.84	3.48E+02	4.52	3.75E+02	5.17
Zr 95	1.08E+02	1.31	1.11E+02	1.44	1.14E+02	1.57
Nb 95	2.30E+02	2.80	2.36E+02	3.07	2.42E+02	3.35
Rh106	2.36E+03	28.69	2.00E+03	26.04	1.70E+03	23.41
Cs134	1.09E+03	13.24	9.88E+02	12.85	8.88E+02	12.26
Cs137	1.01E+02	1.22	1.00E+02	1.31	1.00E+02	1.39
Ba137m	3.38E+02	4.11	3.37E+02	4.39	3.37E+02	4.65
Ce144	2.04E+02	2.48	2.13E+02	2.76	2.20E+02	3.04
Pr144	2.26E+03	27.48	2.35E+03	30.61	2.44E+03	33.69
Pu238	9.84E+01	1.20	8.61E+01	1.12	7.24E+01	1.00
Cm242	4.87E+02	5.93	3.81E+02	4.95	2.88E+02	3.97
Cm244	1.52E+02	1.85	8.09E+01	1.05	4.24E+01	0.58
			Decay Time = 10 Years			
Co 60	3.02E+01	2.73	2.63E+01	2.50	2.28E+01	2.26
Sr 90	5.33E+01	4.82	5.88E+01	5.60	6.33E+01	6.26
Y 90	2.55E+02	23.04	2.81E+02	26.77	3.02E+02	29.90
Cs134	5.28E+01	4.78	4.80E+01	4.57	4.31E+01	4.27
Cs137	8.17E+01	7.40	8.16E+01	7.77	8.16E+01	8.07
Ba137m	2.74E+02	24.84	2.74E+02	26.12	2.74E+02	27.11
Eu154	3.88E+01	3.51	3.34E+01	3.18	2.86E+01	2.82
Pu238	9.38E+01	8.49	8.19E+01	7.80	6.87E+01	6.80
Pu240	1.45E+01	1.31	1.36E+01	1.30	1.26E+01	1.25
Am241	6.71E+01	6.08	6.01E+01	5.72	5.23E+01	5.18
Cm244	1.08E+02	9.77	5.73E+01	5.46	3.00E+01	2.97
			Decay Time = 100 Years			
Sr 90	6.26E+00	2.07	6.91E+00	2.46	7.43E+00	2.88
Y 90	2.99E+01	9.90	3.30E+01	11.74	3.55E+01	13.75
Cs137	1.02E+01	3.38	1.02E+01	3.63	1.02E+01	3.95
Ba137m	3.43E+01	11.35	3.43E+01	12.19	3.42E+01	13.26
Pu238	4.63E+01	15.34	4.05E+01	14.39	3.40E+01	13.15
Pu239	9.40E+00	3.11	9.24E+00	3.29	9.04E+00	3.50
Pu240	1.46E+01	4.84	1.36E+01	4.85	1.26E+01	4.87
Am241	1.46E+02	48.21	1.30E+02	46.32	1.13E+02	43.85
Cm244	3.45E+00	1.14	1.83E+00	0.65	9.58E-01	0.37

Table 2.4.11. (cont.) Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

Nuclide	Enrichment 2.23%		2.93%		3.63%	
	Decay heat	Percent of Total	Decay heat	Percent of Total	Decay heat	Percent of Total
			Decay Time = 1000 Years			
Pu239	9.18E+00	15.71	9.02E+00	16.92	8.82E+00	18.43
Pu240	1.33E+01	22.77	1.24E+01	23.26	1.14E+01	23.91
Am241	3.47E+01	59.44	3.10E+01	58.21	2.70E+01	56.41
Am243	8.84E-01	1.51	5.66E-01	1.06	3.55E-01	0.74
			Decay Time = 10,000 Years			
Pu239	7.21E+00	55.50	7.04E+00	57.20	6.86E+00	58.91
Pu240	5.12E+00	39.43	4.78E+00	38.80	4.41E+00	37.84
Am243	3.80E-01	2.92	2.43E-01	1.98	1.52E-01	1.31
			Decay Time = 100,000 Years			
Bi214	1.22E-02	1.18	1.32E-02	1.30	1.44E-02	1.43
Po210	3.06E-02	2.94	3.31E-02	3.25	3.59E-02	3.58
Po213	2.04E-02	1.97	1.90E-02	1.86	1.71E-02	1.71
Po214	4.43E-02	4.26	4.80E-02	4.71	5.20E-02	5.18
Po218	3.46E-02	3.33	3.74E-02	3.67	4.06E-02	4.05
At217	1.76E-02	1.69	1.64E-02	1.61	1.48E-02	1.47
Rn222	3.16E-02	3.04	3.42E-02	3.36	3.71E-02	3.70
Fr221	1.59E-02	1.53	1.48E-02	1.45	1.34E-02	1.33
Ra226	2.75E-02	2.65	2.98E-02	2.93	3.23E-02	3.22
Ac225	1.44E-02	1.39	1.34E-02	1.31	1.21E-02	1.21
Th229	1.26E-02	1.21	1.17E-02	1.15	1.06E-02	1.06
Th230	2.68E-02	2.58	2.90E-02	2.84	3.14E-02	3.13
U233	1.31E-02	1.27	1.22E-02	1.20	1.10E-02	1.10
U234	4.16E-02	4.01	4.50E-02	4.42	4.87E-02	4.85
U236	8.59E-03	0.83	9.64E-03	0.95	1.05E-02	1.04
Np237	3.86E-02	3.71	3.58E-02	3.52	3.24E-02	3.23
Pu239	5.52E-01	53.15	5.35E-01	52.47	5.18E-01	51.65
Pu242	5.81E-02	5.60	4.34E-02	4.26	3.16E-02	3.15

Table 2.4.12. Neutron production rate (neutrons/sec/MTIHM) by radionuclide (contributing ≥ 1% of total) for BWR spent fuel at a discharge burnup of 30,000 MWd/MTIHM.

Nuclide	Enrichment 2.23%		2.93%		3.63%	
	Neutron production rate	Percent of Total	Neutron production rate	Percent of Total	Neutron production rate	Percent of Total
			Decay Time = 1 Year			
Pu238	3.29E+06	0.46	2.87E+06	0.70	2.42E+06	1.03
Cm242	1.04E+08	14.48	8.12E+07	19.84	6.14E+07	26.09
Cm244	6.04E+08	84.13	3.20E+08	78.34	1.68E+08	71.39
			Decay Time = 10 Years			
Pu238	3.13E+06	0.71	2.73E+06	1.16	2.29E+06	1.82
Pu240	2.21E+06	0.50	2.08E+06	0.88	1.93E+06	1.53
Am241	1.94E+06	0.44	1.74E+06	0.74	1.51E+06	1.20
Cm244	4.28E+08	97.39	2.27E+08	96.31	1.19E+08	94.39
			Decay Time = 100 Years			
Pu238	1.55E+06	6.05	1.35E+06	8.16	1.13E+06	9.95
Pu239	2.22E+05	0.87	2.19E+05	1.32	2.14E+05	1.88
Pu240	2.23E+06	8.72	2.08E+06	12.57	1.92E+06	16.84
Pu242	1.04E+06	4.06	7.76E+05	4.69	5.66E+05	4.96
Am241	4.21E+06	16.45	3.76E+06	22.72	3.27E+06	28.69
Cm244	1.36E+07	53.35	7.24E+06	43.75	3.80E+06	33.29
Cm246	2.52E+06	9.87	9.99E+05	6.03	3.93E+05	3.44
			Decay Time = 1,000 Years			
Pu239	2.17E+05	3.31	2.13E+05	4.55	2.09E+05	5.68
Pu240	2.03E+06	30.98	1.89E+06	40.37	1.75E+06	47.57
Pu242	1.04E+06	15.84	7.75E+05	16.53	5.65E+05	15.40
Am241	1.00E+06	15.31	8.96E+05	19.12	7.80E+05	21.24
Cm246	2.21E+06	33.77	8.75E+05	18.67	3.44E+05	9.37
			Decay Time = 10,000 Years			
Pu239	1.70E+05	6.56	1.66E+05	8.68	1.62E+05	10.78
Pu240	7.82E+05	30.06	7.29E+05	38.01	6.72E+05	44.71
Pu242	1.02E+06	39.29	7.63E+05	39.79	5.56E+05	37.01
Cm246	5.92E+05	22.76	2.34E+05	12.21	9.20E+04	6.12
			Decay Time = 100,000 Years			
U238	1.22E+04	1.33	1.21E+04	1.76	1.21E+04	2.36
Pu239	1.30E+04	1.43	1.26E+04	1.84	1.22E+04	2.40
Pu242	8.69E+05	95.12	6.49E+05	94.33	4.73E+05	92.66

Table 2.4.13. Radioactivity (curies/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

Nuclide	Enrichment 3.02%		3.72%		4.42%	
	Radio-activity	Percent of Total	Radio-activity	Percent of Total	Radio-activity	Percent of Total
			Decay Time = 1 Year			
Sr 90	7.76E+04	3.10	8.39E+04	3.44	8.91E+04	3.75
Y 90	7.77E+04	3.11	8.39E+04	3.44	8.92E+04	3.76
Zr 95	2.74E+04	1.10	2.82E+04	1.16	2.89E+04	1.22
Nb 95	6.18E+04	2.47	6.34E+04	2.60	6.50E+04	2.74
Ru106	3.21E+05	12.84	2.81E+05	11.55	2.45E+05	10.33
Rh106	3.21E+05	12.84	2.81E+05	11.55	2.45E+05	10.33
Cs134	1.65E+05	6.59	1.53E+05	6.28	1.41E+05	5.93
Cs137	1.21E+05	4.84	1.21E+05	4.97	1.21E+05	5.10
Ba137m	1.15E+05	4.58	1.14E+05	4.70	1.14E+05	4.82
Ce144	4.16E+05	16.61	4.30E+05	17.65	4.43E+05	18.65
Pr144	4.16E+05	16.62	4.30E+05	17.65	4.43E+05	18.65
Pn147	9.10E+04	3.64	9.84E+04	4.04	1.07E+05	4.49
Pu241	1.62E+05	6.48	1.48E+05	6.07	1.32E+05	5.57
			Decay Time = 10 Years			
Kr 85	5.31E+03	1.12	5.66E+03	1.20	5.95E+03	1.28
Sr 90	6.27E+04	13.23	6.77E+04	14.39	7.19E+04	15.45
Y 90	6.27E+04	13.24	6.77E+04	14.40	7.20E+04	15.45
Cs134	8.00E+03	1.69	7.42E+03	1.58	6.83E+03	1.47
Cs137	9.83E+04	20.77	9.82E+04	20.89	9.82E+04	21.10
Ba137m	9.30E+04	19.65	9.30E+04	19.77	9.29E+04	19.96
Pm147	8.44E+03	1.78	9.12E+03	1.94	9.90E+03	2.13
Eu154	6.36E+03	1.34	5.70E+03	1.21	5.08E+03	1.09
Pu241	1.05E+05	22.18	9.57E+04	20.36	8.57E+04	18.40
Cm244	4.75E+03	1.00	2.88E+03	0.61	1.72E+03	0.37
			Decay Time = 100 Years			
Sr 90	7.36E+03	14.93	7.95E+03	16.06	8.45E+03	17.07
Y 90	7.36E+03	14.94	7.95E+03	16.06	8.45E+03	17.07
Cs137	1.23E+04	24.96	1.23E+04	24.83	1.23E+04	24.82
Ba137m	1.16E+04	23.61	1.16E+04	23.49	1.16E+04	23.47
Pu238	2.27E+03	4.62	2.07E+03	4.19	1.83E+03	3.70
Pu240	6.22E+02	1.26	5.84E+02	1.18	5.44E+02	1.10
Pu241	1.38E+03	2.80	1.26E+03	2.54	1.13E+03	2.27
Am241	5.13E+03	10.41	4.68E+03	9.45	4.19E+03	8.47

Table 2.4.13 (cont.) Radioactivity (curies/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

Nuclide	Enrichment 3.02%		3.72%		4.42%	
	Radio-activity	Percent of Total	Radio-activity	Percent of Total	Radio-activity	Percent of Total
			Decay Time = 1000 Years			
Np239	3.57E+01	1.55	2.52E+01	1.19	1.75E+01	0.90
Pu239	3.93E+02	17.13	3.87E+02	18.22	3.80E+02	19.49
Pu240	5.66E+02	24.64	5.31E+02	24.99	4.95E+02	25.41
Am241	1.22E+03	53.21	1.11E+03	52.46	9.98E+02	51.24
Am243	3.57E+01	1.55	2.52E+01	1.19	1.75E+01	0.90
			Decay Time = 10,000 Years			
Tc 99	1.48E+01	2.48	1.50E+01	2.65	1.53E+01	2.83
Np239	1.53E+01	2.56	1.08E+01	1.91	7.50E+00	1.39
Pu239	3.09E+02	51.70	3.03E+02	53.29	2.96E+02	54.82
Pu240	2.18E+02	36.47	2.05E+02	36.03	1.91E+02	35.34
Am243	1.53E+01	2.56	1.08E+01	1.91	7.50E+00	1.39
			Decay Time = 100,000 Years			
Ni 59	1.31E+00	1.86	1.18E+00	1.69	1.06E+00	1.53
Zr 93	2.36E+00	3.35	2.41E+00	3.45	2.45E+00	3.54
Nb 93m	2.24E+00	3.18	2.29E+00	3.28	2.32E+00	3.37
Tc 99	1.10E+01	15.69	1.12E+01	16.09	1.14E+01	16.49
Pb210	1.38E+00	1.97	1.44E+00	2.07	1.50E+00	2.18
Pb214	1.38E+00	1.97	1.44E+00	2.07	1.50E+00	2.18
Bi210	1.38E+00	1.97	1.44E+00	2.07	1.50E+00	2.18
Bi214	1.38E+00	1.97	1.44E+00	2.07	1.50E+00	2.18
Po210	1.38E+00	1.97	1.44E+00	2.07	1.50E+00	2.18
Po214	1.38E+00	1.97	1.44E+00	2.07	1.50E+00	2.18
Po218	1.38E+00	1.97	1.45E+00	2.07	1.50E+00	2.18
Rn222	1.38E+00	1.97	1.45E+00	2.07	1.50E+00	2.18
Ra226	1.38E+00	1.97	1.45E+00	2.07	1.50E+00	2.18
Th230	1.37E+00	1.95	1.43E+00	2.05	1.49E+00	2.16
Pa233	1.60E+00	2.27	1.52E+00	2.18	1.41E+00	2.05
U234	2.08E+00	2.96	2.17E+00	3.11	2.26E+00	3.27
Np237	1.60E+00	2.27	1.52E+00	2.18	1.41E+00	2.05
Pu239	2.36E+01	33.59	2.30E+01	32.99	2.24E+01	32.40
Pu242	2.27E+00	3.23	1.81E+00	2.59	1.41E+00	2.04

Table 2.4.14. Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

Nuclide	Enrichment 3.02%		3.72%		4.42%	
	Decay heat	Percent of Total	Decay heat	Percent of Total	Decay heat	Percent of Total
			Decay Time = 1 Year			
Co 60	1.44E+02	1.29	1.29E+02	1.22	1.16E+02	1.15
Sr 90	9.01E+01	0.81	9.73E+01	0.92	1.03E+02	1.03
Y 90	4.30E+02	3.85	4.65E+02	4.39	4.94E+02	4.91
Zr 95	1.39E+02	1.24	1.43E+02	1.35	1.46E+02	1.45
Nb 95	2.96E+02	2.65	3.04E+02	2.87	3.12E+02	3.10
Rh106	3.08E+03	27.57	2.70E+03	25.49	2.35E+03	23.38
Cs134	1.68E+03	15.03	1.56E+03	14.72	1.43E+03	14.24
Cs137	1.34E+02	1.20	1.34E+02	1.26	1.34E+02	1.33
Ba137m	4.50E+02	4.03	4.49E+02	4.25	4.49E+02	4.47
Ce144	2.76E+02	2.47	2.85E+02	2.70	2.94E+02	2.92
Pr144	3.05E+03	27.35	3.16E+03	29.86	3.25E+03	32.35
Eu154	1.17E+02	1.05	1.05E+02	1.00	9.39E+01	0.93
Pu238	1.61E+02	1.45	1.47E+02	1.39	1.30E+02	1.29
Cm242	5.82E+02	5.21	4.78E+02	4.52	3.84E+02	3.81
Cm244	2.35E+02	2.10	1.42E+02	1.34	8.50E+01	0.85
			Decay Time = 10 Years			
Co 60	4.41E+01	2.86	3.96E+01	2.70	3.54E+01	2.51
Sr 90	7.27E+01	4.73	7.86E+01	5.35	8.35E+01	5.91
Y 90	3.48E+02	22.58	3.75E+02	25.58	3.99E+02	28.24
Cs134	8.14E+01	5.29	7.56E+01	5.15	6.95E+01	4.92
Cs137	1.09E+02	7.07	1.09E+02	7.41	1.09E+02	7.70
Ba137m	3.65E+02	23.74	3.65E+02	24.89	3.65E+02	25.84
Eu154	5.69E+01	3.69	5.10E+01	3.48	4.54E+01	3.22
Pu238	1.53E+02	9.93	1.39E+02	9.49	1.23E+02	8.71
Pu240	1.92E+01	1.25	1.81E+01	1.24	1.70E+01	1.20
Am241	7.83E+01	5.09	7.15E+01	4.88	6.41E+01	4.54
Cm244	1.66E+02	10.80	1.01E+02	6.87	6.02E+01	4.27
			Decay Time = 100 Years			
Sr 90	8.54E+00	2.17	9.22E+00	2.48	9.80E+00	2.82
Y 90	4.08E+01	10.36	4.40E+01	11.84	4.68E+01	13.46
Cs137	1.36E+01	3.45	1.36E+01	3.65	1.36E+01	3.90
Ba137m	4.57E+01	11.59	4.56E+01	12.27	4.56E+01	13.11
Pu238	7.54E+01	19.14	6.87E+01	18.46	6.06E+01	17.42
Pu239	1.24E+01	3.15	1.22E+01	3.29	1.20E+01	3.45
Pu240	1.94E+01	4.92	1.82E+01	4.89	1.70E+01	4.87
Am241	1.70E+02	43.24	1.55E+02	41.76	1.39E+02	40.00
Cm244	5.30E+00	1.35	3.22E+00	0.86	1.92E+00	0.55

Table 2.4.14. (cont.) Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

Nuclide	Enrichment 3.02%		3.72%		4.42%	
	Decay heat	Percent of Total	Decay heat	Percent of Total	Decay heat	Percent of Total
			Decay Time = 1000 Years			
Pu239	1.21E+01	16.85	1.19E+01	17.88	1.17E+01	19.12
Pu240	1.76E+01	24.49	1.65E+01	24.79	1.54E+01	25.18
Am241	4.06E+01	56.42	3.70E+01	55.51	3.32E+01	54.18
Am243	1.15E+00	1.59	8.10E-01	1.21	5.61E-01	0.92
			Decay Time = 10,000 Years			
Pu239	9.52E+00	55.46	9.32E+00	56.95	9.11E+00	58.43
Pu240	6.79E+00	39.53	6.37E+00	38.91	5.93E+00	38.06
Am243	4.93E-01	2.87	3.48E-01	2.12	2.41E-01	1.55
			Decay Time = 100,000 Years			
Bi214	1.77E-02	1.28	1.85E-02	1.36	1.93E-02	1.45
Po210	4.43E-02	3.21	4.63E-02	3.41	4.82E-02	3.62
Po213	2.58E-02	1.87	2.46E-02	1.81	2.29E-02	1.72
Po214	6.42E-02	4.65	6.71E-02	4.94	6.98E-02	5.24
Po218	5.01E-02	3.63	5.24E-02	3.86	5.45E-02	4.09
At217	2.23E-02	1.61	2.12E-02	1.56	1.97E-02	1.48
Rn222	4.59E-02	3.32	4.79E-02	3.53	4.99E-02	3.74
Fr221	2.01E-02	1.46	1.92E-02	1.41	1.78E-02	1.34
Ra226	4.00E-02	2.89	4.17E-02	3.07	4.34E-02	3.26
Ac225	1.82E-02	1.32	1.73E-02	1.28	1.62E-02	1.21
Th229	1.60E-02	1.16	1.52E-02	1.12	1.41E-02	1.06
Th230	3.88E-02	2.81	4.05E-02	2.98	4.22E-02	3.17
U233	1.66E-02	1.20	1.58E-02	1.16	1.47E-02	1.11
U234	5.99E-02	4.34	6.25E-02	4.60	6.50E-02	4.88
U236	1.16E-02	0.84	1.27E-02	0.93	1.36E-02	1.02
Np237	4.88E-02	3.53	4.64E-02	3.42	4.32E-02	3.24
Pu239	7.28E-01	52.73	7.09E-01	52.20	6.89E-01	51.71
Pu242	6.72E-02	4.86	5.34E-02	3.93	4.16E-02	3.12

Table 2.4.15. Neutron production rate (neutrons/sec/MTIHM) by radionuclide (contributing ≥ 1% of total) for PWR spent fuel at a discharge burnup of 40,000 MWd/MTIHM.

Nuclide	Enrichment 3.02%		3.72%		4.42%	
	Neutron production rate	Percent of Total	Neutron production rate	Percent of Total	Neutron production rate	Percent of Total
			Decay Time = 1 Year			
Pu238	5.39E+06	0.50	4.92E+06	0.73	4.35E+06	1.02
Cm242	1.24E+08	11.61	1.02E+08	15.05	8.18E+07	19.10
Cm244	9.29E+08	86.83	5.63E+08	83.16	3.37E+08	78.65
			Decay Time = 10 Years			
Pu238	5.10E+06	0.76	4.65E+06	1.13	4.10E+06	1.64
Pu240	2.92E+06	0.43	2.77E+06	0.67	2.59E+06	1.04
Cm244	6.58E+08	97.42	3.99E+08	96.71	2.39E+08	95.59
			Decay Time = 100 Years			
Pu238	2.52E+06	6.58	2.29E+06	8.76	2.02E+06	10.90
Pu239	2.94E+05	0.77	2.89E+05	1.10	2.84E+05	1.53
Pu240	2.96E+06	7.74	2.78E+06	10.61	2.59E+06	13.93
Pu242	1.20E+06	3.14	9.54E+05	3.65	7.45E+05	4.01
Am241	4.92E+06	12.87	4.49E+06	17.15	4.02E+06	21.64
Cm244	2.10E+07	54.93	1.27E+07	48.69	7.61E+06	40.99
Cm246	5.14E+06	13.45	2.47E+06	9.42	1.17E+06	6.28
			Decay Time = 1,000 Years			
Pu239	2.87E+05	2.89	2.82E+05	4.01	2.77E+05	5.13
Pu240	2.69E+06	27.08	2.52E+06	35.85	2.35E+06	43.63
Pu242	1.20E+06	12.08	9.53E+05	13.54	7.44E+05	13.80
Am241	1.17E+06	11.81	1.07E+06	15.19	9.58E+05	17.77
Cm246	4.51E+06	45.38	2.16E+06	30.69	1.02E+06	18.97
			Decay Time = 10,000 Years			
Pu239	2.25E+05	6.08	2.20E+05	8.04	2.15E+05	10.01
Pu240	1.04E+06	27.98	9.72E+05	35.43	9.06E+05	42.07
Pu242	1.18E+06	31.90	9.38E+05	34.20	7.32E+05	34.01
Cm246	1.21E+06	32.58	5.78E+05	21.06	2.74E+05	12.71
			Decay Time = 100,000 Years			
U238	1.20E+04	1.12	1.19E+04	1.40	1.19E+04	1.77
Pu239	1.72E+04	1.61	1.68E+04	1.97	1.63E+04	2.43
Pu242	1.01E+06	93.85	7.99E+05	93.81	6.23E+05	92.89
Cm248	2.04E+04	1.90	7.45E+03	0.88	2.71E+03	0.40

2.5 DEFECTIVE LWR FUELS

2.5.1 Introduction

The characterization of defective LWR fuel is necessary input to the design and performance assessment of an eventual deep geological repository. These data assist in the determination of the need for special handling of these fuels and to ascertain the impacts on long-term repository performance.

Data on defective assemblies along with other quantitative information (described in section 2.3) were obtained via the Energy Information Administration's Nuclear Fuel Data Form RW-859. The EIA aggregates the data reported by the utilities on this form into the RW-859 Database. The latest compilation includes spent fuel discharges as of December 31, 1990 (EIA 1991). Additional data has been obtained from previous compilations, the prior work of Lawson (Lawson 1988), and the continuing work of Bailey (Bailey 1990).

Defective fuel is usually reported on an assembly basis, as is done in this section. It should be noted that only one defective fuel pin is needed to report an entire assembly (with 49 to 264 pins) as defective. In general, and especially for the newer fuels, there is usually only one defective pin per defective assembly. Thus, the assembly defective percent is greater than the pin defective rate by large factors. Utility data are reported on an assembly basis.

Fuel design (as discussed in section 2.2) has had and will continue to have an important role in fuel performance. Differences in fuel design prove to be especially important in the analysis of defective fuels.

2.5.2 Defective Fuel Categories

According to Appendix E of the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-level Radioactive Waste (10 CFR 961), three classifications of defective fuel are identified (classed as "failed" in 10 CFR 961):

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

In 1984, the EIA began collecting data on defective fuel, grouping the fuel according to seven defect categories as follows:

Code 1: Visually Observed Failure or Damage
Code 2: Encapsulated or Other Remedial Action Taken
Code 3: Require 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)

In 1987, when the RW-859 data collection form was revised, DOE was in the process of assessing defective fuel types, and the EIA requested that the utilities leave the defective code blank until a code list is provided. Because the classifications and codes available are somewhat ambiguous, different interpretations are possible. Fuel meeting one definition of defective may not meet another. Additionally, it is not clear from the RW-859 database if some utilities responded that they have no defective fuel or simply did not supply defective fuel data. Reporting of defective fuel in the RW-859 form has been optional.

Lawson prepared guidelines that merged the 10 CFR 961 classes and the 7 early EIA defective codes into four categories: F-1, F-2, F-3, and F-1,2. These are based directly on the 10 CFR 961 categories. Category F-1 includes fuels in 10 CFR 961 class F-1 and/or with EIA defect codes 1, 3, 4, 5, or 6. Category F-2 includes fuels in 10 CFR 961 class F-2 and/or with EIA defect code 7. Category F-3 includes fuels in 10 CFR 961 Class F-3 and/or with EIA defect code 2. Category F-1,2 includes fuels in both 10 CFR 961 classes F-1 and F-2 and/or with both EIA defect code 7 and EIA defect code 1, 3, 4, 5, or 6. Table 2.5.1 presents the number of defective assemblies by assembly class using Lawson's defect categories. Where no category was reported the categories are listed as "unknown."

2.5.3 Defective BWR Fuels

The fuels used in domestic boiling water reactors (BWRs) represent seven assembly classes. By far the largest number of reactors (9 and 28, respectively) are GE BWR/2,3 and GE BWR/4-6 class reactors. The spent fuel from these classes respectively represent (by MTIHM) 34.5% and 60.1% of existing permanently discharged BWR assemblies and 16.7% and 83.1% of the projected discharges to 2037.

Of the eight BWR fuel designs which have been discharged to date, six were manufactured by General Electric (GE) and two were manufactured by Advanced Nuclear Fuels (ANF, formerly Exxon Nuclear). Table 2.5.2 presents the total number of discharged BWR assemblies and defective assemblies reported to the EIA by assembly class and fuel design. It is immediately obvious that improved fuel designs have drastically reduced fuel failures in BWR fuels. GE's Model 3 fuel (Improved 7 X 7) was specifically intended to reduce the unacceptably high failure rate in the original 7 X 7 fuel (Model 2). This was largely accomplished by the introduction of a hydrogen getter in the plenum region of each fuel rod and stricter tolerances on the amount of entrained water vapor in the fuel rod. These efforts dramatically reduced hydriding. GE Model 4 fuel introduced the 8 X 8 array and a single water rod for additional internal moderation. This fuel assembly had 14 additional fuel rods per bundle and reduced the average linear heat generation rate from about 5.9 to about 4.6 kw/ft for a BWR/2,3 reactor and from about 7.4 to about 5.8 for a BWR/4-6. GE Model 5 (8 X 8 Retrofit) fuel added a second water rod and natural uranium axial blankets. Fuel rods for Prepressurized Fuel contained 3 atmospheres of helium. Barrier Fuel introduced a pure zirconium barrier on the interior of the fuel rod cladding. This zirconium "barrier" inhibits crack formation and

reduces failures. None of the BWR/2,3 or BWR/4-6 class fuel supplied by ANF has been reported as failed. Newer fuel designs (GE Model 8 and 9 fuel, and ANF 9 X 9 fuels) are currently in use in many reactors and failure rates are not available for them.

2.5.4 Defective PWR Fuels

The fuels used in domestic pressurized water reactors (PWRs) represent 16 assembly classes. Seven of these 16 assembly classes represent 69 of the 78 domestic PWRs. The spent fuel from these seven classes represents 90% of the current PWR fuel inventory (by weight), and 90.1% of the projected discharges to 2037. The remaining nine assembly classes are either reactor-specific or none of the reactors in the class are currently under active construction.

2.5.4.1 Fuels for B&W Reactors

To date, only B&W has supplied fuels to B&W 15 X 15 class reactors. Two major fuel designs have been identified - assemblies with inconel grid spacers and assemblies with zircaloy grid spacers. Sixty-seven defective assemblies have been reported of the 4,311 assemblies discharged (1.6 percent).

2.5.4.2 Fuels for CE Reactors

CE, ANF, and WE have supplied fuels to CE 14 X 14 class reactors. Of these, none of the 2,976 CE-supplied assemblies and none of the 471 ANF-supplied assemblies have been described as defective. Fifteen (4.3 percent) of the 350 WE-supplied assemblies, used at Millstone 2, have been listed as defective. To date, only CE has supplied fuels to CE 16 X 16 class reactors. Of the 1432 assemblies discharged, 25 (1.8 percent) have been described as defective. From CE SYSTEM 80 reactors none of the 496 assemblies distinguished to date are labeled as defective.

2.5.4.3 Fuels for WE Reactors

The two major suppliers of fuels to WE 14 X 14, WE 15 X 15, and WE 17 X 17 class reactors have been Westinghouse and ANF. Table 2.5.3 presents the total number of discharged assemblies and defective assemblies for WE-built reactors by assembly class and fuel design. The initial WE fuel design (used only in WE 14 X 14 and WE 15 X 15 class reactors) had Zircaloy-clad fuel rods and stainless steel guide tubes. A revised fuel design replaced the stainless steel guide tubes with Zircaloy and was designated LOPAR, for Low Parasitic fuel. This fuel design was used in reloads at WE 14 X 14 and WE 15 X 15 class reactors and served as initial core loadings at many WE 17 X 17 class reactors. The Westinghouse Optimized Fuel Assembly (OFA) design reduced the fuel rod diameter (and thus the uranium per assembly) and replaced the intermediate Inconel spacer grids with Zircaloy spacer grids. Vantage 5 fuel assemblies have product features which include integral fuel burnable absorbers, intermediate flow mixer grids, axial blankets, increased discharge burnup, and a reconstitutable top nozzle. The percentage of defects has continued to decrease with the introduction of new fuel designs.

ANF fuels have been used in several reactors (to date, primarily WE 14 X 14 and WE 15 X 15 classes). The TOPROD fuel design has slightly longer fuel rods with a smaller diameter. The TOPROD fuel also utilizes Gadolinia as a burnable absorber in several of the fuel rods.

2.5.5 Conclusions

Improved fuel designs have dramatically reduced the number of defective fuel assemblies. Data supplied to the EIA on recent fuel designs indicate virtually no failures. While the use of failure rates based on such a statistically small sample is not conclusive, it seems clear that the major failure mechanisms have been identified and addressed by design improvements.

It remains to be seen if this improved performance will continue as average burnups increase and reactors reach the latter portion of their lives. Thus far, increased burnup does not seem to have caused any noticeable increase in defective fuel. Mechanisms for dealing with some effects of reactor aging--debris filtering bottom nozzles, for instance--are already in use in many reactors.

Detailed characterization of failed fuel continues to be needed for input to the design of the Civilian Radioactive Waste Management System and to repository performance assessment. In order to provide this, clearer and less ambiguous definitions and classifications of defective fuel must be developed and uniformly applied.

2.5.6 References for Section 2.5

EIA 1991. U.S. Department of Energy, Energy Information Administration, Nuclear Fuel Data Form RW-859, Washington, D.C. (data as of December 31, 1990).

Lawson 1988. C.G. Lawson, D.L. Phung, and K.J. Notz, "Characteristics of Defective LWR Spent Fuel", in *Proceedings of the International Topical Meeting on Nuclear and Hazardous Waste Management Spectrum '88*, Pasco, WA, September 1988.

Bailey 1990. Bailey, W.J., and S. Wu, *Fuel Performance Annual Report for 1988*, NUREG/CR-3950, Vol. 6, U.S. Nuclear Regulatory Commission, March 1990.

Table 2.5.1 Number of Defective Assemblies By Assembly Class[a]

Assembly Class	Assemblies Discharged	Defect Category[b] F-1	F-2	F-3	F-1,2	Unknown
GE BWR/2,3	15721	0	1478	0	0	26
GE BWR/4-6	27595	160	616	0	170	0
Big Rock	359	10	41	0	1	0
Dresden 1	892	0	159	0	0	0
Humboldt Bay	390	0	0	0	1	0
LaCrosse	333	50	54	0	0	0
Subtotal BWR Fuels	45290	220	2348	0	172	26
BW 15 X 15	4311	3	34	27	3	0
CE 14 X 14	3793	0	15	0	0	0
CE 16 X 16	1432	4	9	0	12	0
CE SYSTEM 80	496	0	0	0	0	0
WE 14 X 14	3308	14	10	0	58	1
WE 15 X 15	6275	29	91	0	9	3
WE 17 X 17	8791	21	34	0	22	9
SOUTH TEXAS	120	0	0	0	0	0
Fort Calhoun	477	0	0	0	0	0
Haddam Neck	786	0	43	0	0	3
Indian Point 1	160	0	0	0	0	0
Palisades	656	1	20	0	0	3
St. Lucie 2	396	0	0	0	0	0
San Onofre 1	508	5	0	0	2	0
Yankee Rowe	453	0	0	0	0	0
Subtotal PWR Fuels	31842	77	256	27	106	19
TOTAL ALL FUELS	77132	297	2604	27	278	45

[a] Data as reported by the utilities (EIA 1991)
[b] Per Lawson 1988; assignment of EIA codes to 10 CFR 961 categories:

	F-1	F-2	F-3
EIA CODE 1	X		
EIA CODE 2			X
EIA CODE 3	X		
EIA CODE 4	X		
EIA CODE 5	X		
EIA CODE 6	X		
EIA CODE 7		X	

Table 2.5.2. Defective BWR Fuels by Assembly Class and Fuel Design.

Fuel Design	GE BWR/2,3 (9 reactors) Discharged Assemblies	GE BWR/2,3 (9 reactors) Defective Assemblies	GE BWR/2,3 (9 reactors) Percent Defective	GE BWR/4-6 (28 reactors) Discharged Assemblies	GE BWR/4-6 (28 reactors) Defective Assemblies	GE BWR/4-6 (28 reactors) Percent Defective
GE-2 (7 X 7 Fuel)	6719	1140	17.0%	1142	385	33.7%
GE-3 (Improved 7 X 7 Fuel)	394	7	1.78%	4936	130	2.63%
GE-4 (Original 8 X 8 Fuel)	3716	1	0.03%	3731	185	4.96%
GE-5 (8 X 8 Retrofit Fuel)	792	1	0.13%	4198	104	2.48%
GE Prepressurized Fuel	2096	0	0.00%	9553	144	1.51%
GE Barrier Fuel	760	0	0.00%	3302	24	0.73%
ANF 7 X 7 Fuel	260	0	0.00%	NA		
ANF 8 X 8 Fuel	980	0	0.00%	732	0	0.00%

NA - Not Applicable

Table 2.5.3. Defective PWR Fuels by Assembly Class and Fuel Design.

Fuel Design	WE 14 X 14 (6 reactors) Discharged Assemblies	WE 14 X 14 (6 reactors) Defective Assemblies	WE 14 X 14 (6 reactors) Percent Defective	WE 15 X 15 (10 reactors) Discharged Assemblies	WE 15 X 15 (10 reactors) Defective Assemblies	WE 15 X 15 (10 reactors) Percent Defective	WE 17 X 17 (33 reactors) Discharged Assemblies	WE 17 X 17 (33 reactors) Defective Assemblies	WE 17 X 17 (33 reactors) Percent Defective
WE Standard	603	1	0.2%	1513	103	6.8%	NA		
WE LOPAR	1415	78	5.5%	3245	13	0.4%	6844	85	1.2%
WE OFA	390	3	0.8%	726	4	0.6%	1639	1	0.1%
WE VANTAGE 5	NYD			NYD			91	0	0.0%
ANF for WE	608	0	0.0%	791	12	1.5%	216	0	0.0%
ANF TOPROD	290	1	0.3%	NA			NA		

NA - Not Applicable
NYD - Not Yet Discharged

2.6 SPECIAL LWR FUEL FORMS

Most (but not all) light-water reactor (LWR) fuel assemblies are currently being stored intact and have relatively standard dimensions, as reported elsewhere in this report. This section provides information on fuels that are different from most LWR fuel assemblies because they have been disassembled or are highly degraded in some fashion, because their design parameters are quite different in some way, or because their fuel rods have been consolidated. This section also includes information on LWR burnable absorbers, reprocessed fuel, and LWR crud. Fuel rods and assemblies that have been cut apart and disassembled for testing, evaluation, and research, and have thus largely lost their LWR physical characteristics, are covered in Section 4.5, Miscellaneous Fuels.

2.6.1 Degraded Fuel from TMI-2

Degraded spent fuel and core debris from the Three Mile Island-2 reactor accident have been shipped to DOE's Idaho National Engineering Laboratory. Shipments were completed in May 1990. The core loading of the 926 MW(e) PWR TMI-2 reactor at the time of the accident consisted of 177 fuel assemblies containing 82,023 kg of uranium with an enrichment of about 2.51%. The fuel and debris, including Zircaloy cladding and various structural materials, were shipped in special containers of three types: a fuel canister, a knockout canister, and a filter canister. These were specially designed for different modes of loading, depending on the physical state and water content of the degraded fuel. The three types of container have the same external dimensions: 14 in. diameter and 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 (Childress 1986). Figure 2.6.1 shows the type of canister used for most of the dry debris and spent fuel. The total number of canisters used was 342, and the total mass of fuel and debris shipped was about 135 MT. The total packaged volume, based on the external dimensions of the canisters, was about 130 m^3.

Assuming that each shipment canister will require an overpack canister before being placed in a repository, the number of overpack canisters required would be 342. No attempt has been made to design such canisters, but it appears likely that canisters with an outside diameter of approximately 20 in. and an overall length of approximately 15 ft would be adequate.

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 × 17 assemblies for the South Texas plants are more than 3 feet longer than standard 17 × 17 fuel assemblies (so-called "Texas longs"). Early fuel assemblies for both PWRs and BWRs were shorter than current fuel assemblies. The fuel assemblies at the Dresden-1 and Humboldt Bay were 6 × 6 arrays; at the Big Rock Point reactor, the original fuel assembly was a 12 × 12 array but has been replaced with both 9 × 9 and 11 × 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; thus 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

Consolidation of LWR fuel assemblies is being considered in order to decrease the occupied volume of the assemblies (by about 50%). This could provide a number of potential benefits, e.g., increased pool storage, increased transport cask capacity, increased MRS storage cask capacity, and increased repository packaging efficiency. The two major counter-balancing factors are conducting the consolidation operation itself, and dealing with the spent fuel disassembly (SFD) hardware. Some of the hardware, perhaps much of it, will be GTCC and therefore still require repository disposal.

Two basic approaches have been tested: wet consolidation and dry consolidation. The former was done in storage pools at utilities or at the West Valley reprocessing plant site. Dry consolidation was tested in hot cells at INEL under a DOE program. A total of 105 assemblies have been consolidated in five test programs, representing four assembly classes and five model types, as follows:

Fuel vendor	Class	Assembly type	Model reference
B&W	15 × 15	Mark B	Duke 1983
CE	14 × 14	Standard	Cope 1985
WE	14 × 14	Std-ZCA	Failey 1987
WE	14 × 14	Std-ZCB	Cline 1988
WE	15 × 15	Std-ZC	Feldman 1987

A summary of these tests is given in Table 2.6.1. These 105 assemblies are listed in the LWR Quantities Data Base

as either "consolidated" (the 57 that were consolidated wet by the utilities) or "encapsulated" (the 48 that were consolidated dry by DOE at INEL and placed in cans).

2.6.4 Shippingport Reactor LWR Fuel

2.6.4.1 Introduction

The Shippingport Atomic Power Station, located at Shippingport, Pennsylvania, was the first large-scale central station nuclear power plant in the United States. It began operation in 1957 as a uranium-fueled pressurized LWR with a design gross electrical output capacity of 68 MW(e), and continued operating in that mode until 1964, with three refuelings during that period. The reactor core in place during that time is referred to as PWR Core 1. In 1964, the reactor was shut down for installation of a new redesigned core of greatly increased output power, referred to as PWR Core 2. This core had a design gross electrical output of 150 MW(e) and operated from 1965 to 1974 with one intermediate refueling. Both Core 1 and Core 2 were of the seed-and-blanket type, using highly enriched uranium (90.4 to 93.2% ^{235}U) in the seed fuel elements and natural uranium in the blanket fuel elements. The intermediate refuelings of each core were of the seed portion only.

The chronology of the PWR Core 1 and Core 2 operations and refuelings is summarized in Table 2.6.2. Physical parameters of the reactor and its two cores are given in Table 2.6.3. Figure 2.6.2 shows a cross-sectional view of the reactor vessel with Core 1 in place. The configurations of the seed and blanket portions of Core 1 and Core 2 are shown schematically in Figs. 2.6.3 and 2.6.4, respectively.

Shippingport operations in the PWR mode were terminated in 1974, when the reactor was shut down to prepare for installation of a light-water breeder reactor (LWBR) core based on the ^{233}U–thorium fuel cycle. The LWBR started up in 1977 and continued operation until 1982, when the reactor was permanently shut down for decommissioning. Physical dismantling began in 1985 and was completed in 1990. In this section, only the PWR fuel is discussed; the LWBR fuel is discussed in Sect. 4.4.

2.6.4.2 PWR Core 1 Description

The seed portion of Core 1 consisted of 32 seed fuel assemblies made of flat plates of uranium-zirconium alloy containing 90.4% enriched uranium, covered with a Zircaloy-2 cladding. The total quantity of ^{235}U in Seed 1 was 75 kg. Seeds 2, 3, and 4 each contained 90 kg of ^{235}U; this change was made in order to obtain greater fuel lifetime. Each seed fuel assembly consisted of four square subassemblies in a square array, separated so that a cruciform control rod could be inserted. Because of this configuration, a seed fuel assembly is sometimes referred to as a seed fuel cluster; these terms are synonymous. With end fittings included, a seed fuel assembly had a length of about 110 in. and a cross-section about 5.5 in. square. Active fuel length was 70.75 in.

The blanket portion of Core 1 consisted of 113 fuel assemblies containing a total of 14,560 kg of natural uranium dioxide pellets, which corresponds to about 12,830 kg of natural uranium. The fuel pellets were encased in Zircaloy-2 tubes about 10 in. long and 0.411 in. outside diameter. These fuel rods were welded into a square array of holes in top and bottom end plates to form a fuel bundle. Each bundle contained 120 fuel rods, and had overall dimensions of about 5.19 in. square by 10.25 in. long. Seven such bundles were stacked vertically in a Zircaloy-2 box, together with spacers, springs, and other fittings, to form a fuel assembly about 5.5 in. square × 105 in. long, of which 70.75 in. was active fuel length (Shippingport 1958, Bettis 1969).

2.6.4.3 PWR Core 2 Description

The Core 2 design incorporated a number of significant departures from Core 1. The seed and blanket fuel assembly cross-sectional dimensions were increased from about 5.5 in. square to about 7.4 in. square, and the active fuel length was increased from about 70 in. to about 92 in. The number of fuel assemblies in the core was decreased to 20 in the seed portion and 77 in the blanket portion. The total quantity of ^{235}U in the seed portion of the core was increased to 336 kg for Seed 1 and subsequently to 390 kg for Seed 2. Plate-type fuel assemblies were used in both the seed and blanket of Core 2, whereas Core 1 used plate-type assemblies for the seed and rod-pellet assemblies for the blanket. In Core 2, the seed fuel was made of highly enriched ^{235}U (90.4% enrichment for Seed 1 and 93.2% enrichment for Seed 2) diluted with ZrO_2, in contrast to the 90.4% enriched uranium–Zircaloy alloy used in Core 1.

Some design similarities remained. In both Core 1 and Core 2, each seed fuel assembly consisted of a cluster of four square sections separated so as to permit the insertion of a cruciform control rod (Bettis 1968).

2.6.4.4 Disposition of Core 1 Fuel

The uranium loadings of Core 1 seed fuel at beginning and end of exposure are summarized in Table 2.6.4. After removal from the reactor, the depleted seed fuel assemblies of Core 1 were shipped to the Expended Core Facility (ECF) of the Naval Reactors Facility at Idaho Falls. Shipment dates were as follows:

Seed 1	September 1961
Seed 2	March 1962
Seed 3	June 1963
Seed 4	April 1965 to January 1967

At ECF, selected samples of each seed were examined to determine fuel performance during operation at Shippingport. Following this, all but one of the seed fuel assemblies were shipped to the Idaho Chemical Processing Plant for recovery of ^{235}U. One seed fuel assembly is still at ECF (Connors 1989).

The Core 1 blanket fuel assemblies were shipped to ECF, where several test specimens were removed for examination. The remaining blanket fuel assemblies were shipped to Hanford, where they were processed in the Redox plant for reclamation. The burnup of the blanket fuel was 11,100 MWd/MTIHM (Dressler 1968).

The test specimens of the blanket were retained at ECF. These specimens are individual blanket bundles, fuel rods, and fuel rod sections consisting typically of single rows of fuel rods separated from fuel bundles. The fuel content of these test specimens retained in the ECF inventory is approximately 227 kg of natural uranium (Connors 1989).

2.6.4.5 Disposition of Core 2 Fuel

Following reactor shutdown in 1974, the Seed 2 and blanket fuel assemblies of Core 2 were removed. Seed 1 had already been removed at the time of the intermediate seed refueling in May-June 1969. The uranium loadings of the individual Seed 1 and Seed 2 fuel assemblies at beginning and end of life are given in Tables 2.6.5 and 2.6.6 (Connors 1988).

The blanket of PWR Core 2 produced 50.9% of the total core power output averaged over Seed 1 lifetime and 50.2% over Seed 2 lifetime. The depletion of the PWR Core 2 blanket was 8,250 MWd/MTIHM for Seed 1 and 6,023 MWd/MTIHM for Seed 2 (Bettis 1973, 1983.)

The Core 2 Seed 1 and Seed 2 assemblies were shipped to the ECF at the Naval Reactors Facility in Idaho. Thirty-nine of the seed assemblies are still in storage at the ECF. The fortieth seed assembly (from Seed 2) was disassembled and sectioned for examination (Connors 1988).

Seventy-three of the blanket assemblies were shipped to Hanford, Washington for reclamation; the remaining four of the total core complement of 77 are in storage at ECF for ultimate shipment to Hanford (Connors 1988).

2.6.4.6 Fuel Inventories Reported by Sites

The most recent submittal from INEL to the IDB (Berreth 1990) shows the following Shippingport fuel in storage at ICPP (as zirconium-clad UO_2 pellets):

Description	No. of containers	Total ^{235}U (kg)	Total U (kg)
Shippingport PWR Core 2	28	305.802	392.026

No other Shippingport fuel is reported by INEL. However, INEL does not report on items that are in storage at ECF. Since the ECF does not publish a report on its inventories, there is, at present, no confirmation of the items stated to be at ECF in Connors 1988 and 1989. A previous IDB report (IDB 1989) gave the following information on Shippingport PWR fuel in storage at Savannah River and PNL:

Location	Description	Total U (kg)	^{235}U (kg)	Pu (kg)	Total heavy metal (kg)
SRS	Shippingport, UO_2	16.429	0.023	0.108	16.537
PNL	Shippingport	3.9	0.1	0.1	4.0

Shippingport PWR fuel reportedly shipped to Hanford is not listed as being in inventory in its submittals to the IDB (Turner 1989).

2.6.5 Burnable Absorbers

Burnable absorbers, also known as neutron absorbers or burnable poisons, are used in both BWR and PWR reactors to control power peaking early in the fuel cycle and to assist in power shaping and fuel burnup optimization. Absorbers have been placed both external to the fuel assembly and internal. The internal absorbers may be either as fueled or unfueled rods. An external absorber is defined as any absorber that is not integrally attached to the assembly, i.e., it is non-fuel assembly hardware. Thus, the absorber may be physically inside the assembly, as in the case of PWR burnable absorbers located in control-rod positions, but still be defined as an external absorber since it is not an integral part of the fuel assembly and can be removed during a fuel reload. Internal absorbers are defined as any absorber that is an integral part of the fuel assembly and therefore will be in-core for the entire fuel assembly lifetime and will remain with the assembly after discharge.

The chemical form of absorbers may vary. PWR reactors typically use Al_2O_3-B_4C pellets in Zircaloy-4 tubes or borosilicate glass rods in stainless steel tubes. More recently, UO_2-Gd_2O_3 pellets have been used in PWR reactors. The Gd_2O_3 forms a solid solution with the UO_2. These tubes are located either in a fuel-rod position (internal) or a control-rod guide thimble (external) depending upon the fuel design. The latest PWR burnable poison, introduced by Westinghouse, is a zirconium boride coating on the fuel pellets. BWR reactors typically use UO_2-Gd_2O_3 in a fuel-rod (internal) position. The gadolinia

content reduces the thermal conductivity of the pellet and thus lower fuel enrichments are used. More detailed descriptions follow.

2.6.5.1 BWR Burnable Absorbers

Early uses of burnable absorbers in BWR reactors began with "poison curtains" which where actually borated stainless steel plates hung between the fuel channels. This external burnable absorber provided for excess reactivity control during initial reactor startup. It appears that the first application of an internal (integral with the fuel assembly) burnable absorber was begun by General Electric during the second reload fuel for the Dresden-1 reactor which consisted of 36 fuel rods of GE 6 × 6 type III-B uranium oxide fuel doped with 0.15% erbium oxide. The third Dresden-1 reload consisted of 35 fuel rods of type III-F fuel and a single, nonfueled burnable absorber rod made from 95.3% Al_2O_3-4.7% Gd_2O_3. The fourth reload fuel, type V, again used 36 fueled rods with some rods containing UO_2-Gd_2O_3 pellets.

Data on recent uses of burnable absorbers in BWR reactors is very limited. This kind of information is treated as proprietary by GE. In the *General Electric Standard Application for Reactor Fuel*, NEDO-24011-A-8, it is stated that the gadolinia concentrations for all GE BWR fuel is considered proprietary. Gadolinia distributions within the assembly are also considered proprietary for all but the Original 8 × 8 fuel design. For this latter design, it is known that for a fuel enrichment of 2.19%, 3 fueled absorber rods are used. Enrichments of 2.19%–2.62% have 4 absorber rods. Higher enrichments use 5 absorber rods per assembly. Most of the information on GE fuel has come from searches of the federal docket.

Available data on 7 × 7, Improved 7 × 7, and Original 8 × 8 fuel designs suggest that the gadolinia tends to be utilized in the highest enrichment assemblies. The gadolinia-containing rods themselves often have a lower enrichment because the gadolinia lowers the thermal conductivity. Most of the numerical data shows the gadolinia concentration to be between 2 and 4% with 3 to 5 fueled absorber rods per assembly. The gadolinia concentration and the number of fuel rods with burnable absorbers tends to increase with increasing fuel enrichment. GE prepressurized fuel is thought to have up to 8 absorber rods per assembly with up to 171 g of Gd_2O_3 per rod (approximately 4%).

GE experience with optimizing fuel performance has resulted in more complex BWR fuel management concepts including radial and axial enrichment variations within assemblies and fuel rods. It would be reasonable to assume that the gadolinia concentrations have also become more complex. Though only limited data exists, it does appear that the latest GE fuel designs (such as GE-8 and GE-9 extended burnup fuel) utilize axial variations and possibly radial variations in gadolinia concentrations.

Data from Advanced Nuclear Fuels show that gadolinia concentrations in their BWR fuels varies from 1.8 to 5% with the number of fueled burnable absorber rods varying from 1 to 10 per assembly. The ANF 8 × 8 XN-3 fuel design has 2 water rods with 5 or 6 fueled absorber rods containing 2 wt % Gd_2O_3. The ANF 9 × 9 fuel design with 2 water rods has 7 to 10 fueled absorber rods containing 4 to 5 wt % Gd_2O_3. The ANF 9 × 9 IX and 9X fuel designs have 9 water rods with 6 fueled absorber rods (5 rods with 1.8% Gd_2O_3 and 1 rod with 4.5% Gd_2O_3). The 9 × 9 designs contain 2% of Gd_2O_3 per fuel rod. ANF uses only fueled burnable absorbers for BWR fuel.

2.6.5.2 PWR Burnable Absorbers

PWR reactors have used fueled and unfueled integral burnable absorbers and external burnable absorbers, referred to as burnable poison assemblies (BPA). Fueled absorbers (internal) are used by ANF and Westinghouse (WE), unfueled absorbers (internal) are used by ANF and Combustion Engineering (CE), and external absorbers (not fueled) are used by Babcock & Wilcox (B&W) and WE.

Fueled burnable absorbers of the type used in BWRs are becoming increasingly popular for PWR reactors. ANF uses approximately 80 g/rod of 4–10 wt % UO_2-Gd_2O_3 in their designs. The use of gadolinia, which ANF says has become a standard feature on their fuels, probably stems from their BWR experience. Typically, 2 to 12 absorber rods/assembly are used. The Gd_2O_3 reduces the thermal conductivity and thus lower enrichments are used in these rods. ANF uses these types of absorbers on most of their reload fuel for CE and WE fuel designs.

Fueled burnable absorbers are used by Westinghouse on its Vantage 5 fuel design (production started in 1984). This fuel introduced what WE calls an integral fuel burnable absorber (IFBA) which integrates the burnable absorber material directly into the fuel rod in the form of a thin zirconium diboride coating on the pellet surface. Detailed data on the IFBA (material, thickness, etc.) appears to be proprietary. It is known that this type of absorber is used in approximately 80% of current WE fuel production. It is believed that approximately 120 rods/assembly are coated (45% of the rods/assembly) and that the amount of ZrB_2 is around 3 g/rod (0.2% ZrB_2 by weight).

Internal burnable absorbers are used by CE in all of their fuel. This is dictated by the CE core design which uses only 5 large guide tubes per assembly for control rods, therefore the use of external burnable absorbers is not feasible. Unfueled rods containing 0.8–1.0 kg/rod of Al_2O_3-B_4C are used with the number of absorber rods varying up to 16 per assembly. ANF uses unfueled burnable absorbers for the CE 14 × 14 assembly class. This is the only fuel ANF manufactured that contains unfueled rods. ANF uses Al_2O_3-B_4C as the absorber with 652 g B_4C/rod and a maximum of 4 rods/assembly. In

some cases, this is supplemented with gadolinia in fueled absorber rods.

External burnable absorbers are not fueled and are described in Chapter 2.8 under "Non-Fuel Assembly Hardware."

2.6.6 Commercially Reprocessed Fuel

The Nuclear Fuel Services (NFS) Reprocessing Plant at West Valley reprocessed spent fuel as a commercial venture from 1965 to 1972. Fuels from 9 different reactors were handled and processed during 28 campaigns. Table 2.6.7 lists these campaigns, their dates, and the amount of fuel reprocessed. Available records from the NFS docket and related reports are in terms of MT of heavy metal and do not directly address the number and type of fuel assemblies reprocessed at West Valley. For the campaigns involving commercial reactor fuel, this information is of interest for several reasons.

First, early reactors and the fuel designs used by them changed drastically over a short period of time as operating experience at the reactors was gained. For example, the initial fuel at Big Rock Point was a stainless-steel-clad 12 × 12 array. Along with this fuel, 28 developmental assemblies (with stainless steel, Inconel, and Zircaloy clads) were irradiated early in the Big Rock Point lifetime. Cycle 2 reload fuel, or 'B' fuel, was a Zircaloy-2 clad 11 × 11 array. Cycle 3 reload fuel, or 'C' fuel, was also a Zircaloy-2 11 × 11 fuel, but the UO_2 was in powder, not pellet form. Most of these assemblies were likely reprocessed during the two Big Rock Point campaigns and will not need to be considered as intact spent fuel by the CWMS. However, if a few of these assemblies were not reprocessed (as the evidence appears to suggest), then the CWMS must consider the handling, transport, storage, and eventual emplacement of these assemblies.

Second, the numbers and type of assemblies reprocessed at West Valley are of interest from the standpoint of reinforcing the accountability records for these reactors. Although it is expected that assembly-by-assembly accountability would have been documented and retained, this is not completely certain. A detailed, reactor-by-reactor, cycle-by-cycle analysis of early reactor fuel usage, including spent fuel shipments to West Valley, receipt records, and reprocessing records is nearly completed (Moore 1992). The number of assemblies in each batch is an estimate based on the preliminary results of this analysis. The above reference will clarify as many uncertainties as possible.

Fuel from the Elk River reactor was reprocessed in part in Italy and the balance is stored at Savannah River (see Chapter 4, Table 4.5.9). This was mixed UO_2-ThO_2 fuel.

2.6.7 LWR Crud

Neutron-activated corrosion product deposits, commonly referred to as "crud," are found on LWR fuel assembly surfaces. These crud deposits are formed when corrosion products from out-of-core reactor components are transported by the primary coolant system and deposited on LWR fuel assembly surfaces. Subsequent neutron activation causes the crud deposits to become radioactive. A tightly bound cladding oxide layer (such as zirconium oxide) is not considered crud.

Crud buildup on fuel rods can result in cladding degradation, fuel rod overheating due to reduced heat transfer, and radioactive contamination of primary coolant loop components due to spallation and movement of crud deposits. After the spent fuel is discharged, the crud deposits can spall and peel from the assemblies during fuel handling and transportation and become a source of dispersible contamination.

Methods to reduce crud buildup have been developed. Improvements in the control of coolant water chemistry, coolant processing, and use of more corrosion-resistant materials in the primary coolant loop have led to reduced crud deposits. Crud characteristics are different for PWR and BWR reactors and also vary for different reactors, different assembly types, from cycle to cycle, and radially and axially within a single assembly.

2.6.7.1 BWR Reactors

Crud deposits in a BWR reactor typically exist as one or two layers of iron-based corrosion products. BWR fuel assemblies tend to have a larger deposit of crud than do PWR fuel assemblies. The crud is typically composed of a flocculent, outer layer that is loosely bound to a dense, tenacious inner layer. This inner layer is tightly bound to the cladding oxide layer. Some BWR crud has been found that only consists of the loosely bound layer. The total thickness of BWR crud deposits ranges from 10 to 100 μm (Jardine 1986). BWR crud is primarily Fe_2O_3 with an Fe content of about 87 wt % of the total metal content of the crud. Some deposits of magnetite, Fe_3O_4, have been observed. Small quantities of other oxides, such as nickel, zinc, copper, and magnesium, are also common. In early BWR fuel assemblies (pre-1971), a higher copper content was observed due to the corrosion of copper alloy heat exchangers used in the feedwater system. The main source of crud in recent BWR reactor designs is the carbon steel used in the primary feedwater system, which corrodes more rapidly than stainless steel. A typical composition of BWR crud is shown in Table 2.6.8 (Hazelton, 1987).

The type of feedwater treatment system used has a significant effect on crud buildup. BWR reactors using a deep bed polishing system will produce 4 to 10 times more crud buildup than reactors using a powdered resin demineralizer system (Jardine 1986).

Composed principally of iron oxide, BWR crud usually appears to be red, reddish brown, or orange, though other variations in color have been observed. The heaviest crud deposits occur in the subcooled zone of the reactor (lower 1/4 to 1/3 of the core). Heavy deposits have also been noted in the peripheral, low-flux regions of the core. Close examination of the deposits reveals that crud consists of individual agglomerated particles ranging in size from 0.1 to 10 μm. The smaller particles in this range can become airborne during dry handling.

Estimates have been made of the total quantity of crud in a typical BWR reactor. The total iron content in a typical BWR reactor core (in the form of Fe_2O_3) was estimated as 35 to 160 kg at the end of two cycles (Hazelton, 1987).

The primary source of radiation from the crud is ^{60}Co. Other radioactive sources exist in crud, but ^{60}Co has the longest half-life and will thus become more predominant with fuel age. It also has the strongest gamma emission. The degree of ^{60}Co activity is dependent upon the type of feedwater treatment system used. BWR reactors which use a deep bed polishing system typically have a ^{60}Co activity of 180 μCi/cm^2 whereas a reactor with a powdered resin demineralizer system will have a lower activity, about 110 μCi/cm^2. Table 2.6.9 shows the radioisotopes that are commonly found in both PWR and BWR crud deposits along with their half-lives.

2.6.7.2 PWR Reactors

Crud deposits in a PWR reactor usually consist of a single layer of dense, tenacious ferrous oxides. The most common form of PWR crud is a partially substituted nickel ferrite (Hazelton, 1987) of the form $Ni_xFe_{3-x}O_4$, where $0 \leq \times \leq 1$. Other deposits including Fe_3O_4, NiO, SiO_2, and CrO_3 have been found. The nickel compounds characteristic of PWR crud are due to the use of high nickel alloys (such as Inconel and Incoloy) in the primary coolant loop. The total thickness of PWR crud ranges from 0.1 to 10 μm for modern reactors (Jardine 1986). Earlier reactors which did not exercise as much control over coolant chemistry had crud thicknesses up to 85 μm. The typical composition of PWR crud is shown in Table 2.6.10 (Hazelton, 1987).

Nickel ferrite and magnetite, which are the major constituents of PWR crud, cause it to appear black or shades of dark grey. Color variations, such as reddish brown or tan deposits, have been observed due to other constituents. The heaviest deposits occur at the top of the core where a crud thickness of 5 μm is common. Typical midcore thicknesses are 0.3 to 0.5 μm. The heaviest known deposit was recorded for an older Westinghouse fuel where crud in the upper core was up to 101 μm thick (Hazelton, 1987). Particle sizes for PWR crud is in the same range as for BWRs, 0.1 to 10 μm.

Estimates have been made of the total quantity of crud in a typical PWR reactor. The total in-core metal content after the first cycle ranges from 15 to 24 kg iron and 6 to 8 kg nickel. The heaviest deposits for PWR reactors seem to occur during the first fuel cycle (Hazelton, 1987).

The primary radioactive sources in PWR crud are ^{60}Co, ^{58}Co, and ^{54}Mn (Pick, 1987). Radioactivity measurements of PWR crud, which vary greatly, have shown the ^{60}Co source to range from 0.1 to 140 μCi/cm^2. The major source of the cobalt is the high cobalt alloy Stellite used in the PWR primary coolant loop.

2.6.8 References for Section 2.6

Berreth 1990. Letter from J. R. Berreth, INEL, to M. J. Bonkoski, DOE Idaho Operations Office, "FY-90 Integrated Data Base Information," dated March 28, 1990, with revisions by D. Knecht, June 15, 1990.

Bettis 1968. AEC Research and Development report WAPD-296, *PWR Core 2 Reactor Design Description Report*, March 1968.

Bettis 1969. AEC Research and Development Report WAPD-320, *PWR Core 1 Component Examination Program Summary Report*, October 1969.

Bettis 1973. AEC Research and Development report WAPD-332, *Shippingport Operations from Initial PWR Core 2 Power Operation to Power Operation After First Refueling*, June 1973.

Bettis 1983. AEC Research and Development report WAPS-335, *PWR Core 2 Depletion (April 1964 to February 1974)*, June 1983.

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

Cline 1988. J. E. Cline (SAIC) and L. McCarten (Northern States Power Co.), "Direct-Assay Characterization of Irradiated Hardware, Application to Fuel Skeletons," *Nuclear Plant Journal*, pp. 60-62, July-August 1988.

Connors 1988. Personal communication, D. R. Connors (Westinghouse) to R. Salmon (ORNL), September 12, 1988.

Connors 1989. Personal communication, D. R. Connors (Westinghouse) to R. Salmon (ORNL), January 9, 1989.

Cope 1985. A. W. Cope and R. L. Moscardini (Combustion Engineering), *Fuel Consolidation Demonstration, Program Overview*, EPRI Report No. NP-4327, November 1985.

Crawford 1967. J. A. Crawford, *Residual Content of Shippingport Core 1 Seed Fuel: Calculation vs Experiment*, *Transactions of the American Nuclear Society*, Vol. 10, pp. 302-303 (1967).

Dressler 1968. R. T. Dressler, R. D. O'Hara et. al., *Shippingport Operations During PWR Core 1 Depletion* (Dec. 1957 to Feb. 1964), WAPD-294, December 1968.

Duke 1983. Duke/Westinghouse, "Spent Fuel Consolidation Demonstration," September 30, 1983, internal company report.

Failey 1987. M. P. Failey and R. Tayloe (Battelle/Columbus), *Spent Nuclear Fuel Consolidation and Characterization; Activity Determination and Classification of Assembly Skeletons*, Report No. ESEERCO EP85-4, December 1987.

Feldman 1987. E. M. Feldman and C. G. Cox (EG&G/ID), *Dry Rod Consolidation Technology Test Plan for Technology Data Collection*, Document No. DRCT-TDC-3 Rev. 1, February 1987.

Hazelton 1987. R. F. Hazelton, *Characteristics of Fuel Crud and its impact on Storage, Handling, and Shipment of Spent Fuel*, PNL-6273, Pacific Northwest Laboratory, September 1987.

IDB 1989. *Integrated Data Base for 1989*, DOE/RW-0006, Rev. 5, November 1989.

Jardine 1986. Leslie J. Jardine, "Preliminary Characterization of Radioactive Crud for Design of Dry Spent-Fuel Handling Facilities," *Proceedings of the ANS 1986 Winter Meeting*, Washington, DC, November 16-20, 1986, Vol. 53.

Metzger 1989. Letter, S. W. Metzger (EG&G Idaho) to S. N. Storch (ORNL), March 21, 1989.

Moore 1989. R. S. Moore, "The Use of Neutron Absorbers in LWRs," Automated Sciences Group, Oak Ridge, Tennessee, presentation made to the Characteristics Data Base Program at Oak Ridge National Laboratory, December 8, 1989.

Moore 1992. R. S. Moore, *LWR Fuel Assemblies Reprocessed at West Valley*, in preparation.

Pick 1987. M. E. Pick, "Characterization of the Radioactive Deposits on PWR Primary Circuit Surfaces," *Proceedings of the 1987 International Decommissioning Symposium*, CONF-871018-Vol. 2, Oct. 4-8, 1987.

Shippingport 1958. *The Shippingport Pressurized Water Reactor*, Addison-Wesley Publishing Company, Inc., Reading, Massachusetts, 1958.

Turner 1989. Letter from D. A. Turner, Westinghouse Hanford, to G. J. Bracken, DOE/RL, May 17, 1989.

Table 2.6.1. Summary of assemblies used for consolidation tests

Utility	Reactor	Consolidation		Assembly class	Descriptions		
		By	When		No. used	Age (years)	Burnup (MWd/MT)
		Wet consolidation tests					
Duke Power Co.	Oconee 2	West.	1982	B&W 15x15	4	5	26,000
Northeast Utilities	Millstone 2	C-E	1986	CE 14x14	12		
Rochester G&E	Ginna	US Tool and Die	1986	WE 14x14	5	14.7	
Northern States	Prairie	West. Island	1987	WE 14x14	36	3-10	39,000
		Dry consolidation tests					
VEPCO and	Surrey and	DOE-	1987	WE 15x15	36	6-12	25-35,000
Florida P&L	Turkey Point	INEL	1987	WE 15x15	12	10-12	26-28,000

Table 2.6.2 Chronology of Shippingport PWR Core 1 and Core 2 operations[a]

Core	Seed	Blanket	Start of operating period	End of operating period
Core 1	Seed 1	1	Dec. 1957	Nov. 1959
	Seed 2	1	Apr. 1960	Aug. 1961
	Seed 3	1	Oct. 1961	Nov. 1962
	Seed 4	1	Jan. 1963	Feb. 1964
Core 2	Seed 1	2	Apr. 1965	Mar. 1969
	Seed 2	2	July 1969	Feb. 1974

[a]Core 1 used one blanket throughout its four operating periods, and Core 2 used one blanket throughout its two operating periods. This table shows the shutdowns for refueling purposes. There were some operational shutdowns for other reasons. Sources of data: Bettis 1968, 1969, 1973, 1983.

Table 2.6.3 Shippingport PWR design characteristics[a]

Characteristics	Core 1	Core 2
General		
Reactor power, MW thermal	225	505
Electrical output (gross), MW(e)	68	150
Coolant pressure, psig	2,000	2,000
Reactor vessel design		
Outside diameter, ft	10.5	b
Height, ft	31	b
Wall thickness, in.	8.375	b
Material	c	b
Core design		
Type	Seed and blanket	Seed and blanket
Diameter, ft (approx.)	6.8	6.8
Height, ft (approx.)	6.0	8.0
Seed fuel		
Type of assembly	Plate	Plate
Number of assemblies	32	20
Total U per assembly, kg (seed 1 subsequent seeds)	2.59/3.11	18.6/20.9
U-235 per assembly, kg (seed 1 subsequent seeds)	2.34/2.81	16.8/19.5
Total U, kg (seed 1 subsequent seeds)	83/99.6	371.7/418.5
Total U-235, kg (seed 1 subsequent seeds)	75/90	336/390
Initial enrichment, % U-235	90.4	90.4/93.2[d]
Cladding material	Zircaloy	Zircaloy
Blanket fuel		
Type of assembly	Rod bundle	Plate
Number of assemblies	113	77
Total U per assembly, kg	113.5	222.1
U-235 per assembly, kg	0.80	1.58
Total U, kg	12,830	17,100
Total U-235, kg	91.3	121.7
Initial enrichment, % U-235	0.71	0.71
Cladding material	Zircaloy	Zircaloy

[a]Sources of data: Shippingport 1958, Bettis 1968, 1969, 1973, Connors 1988. Bettis 1968 is the principal source for Core 2.

[b]Core 1 and Core 2 used the same reactor vessel except for minor modifications necessitated by changes in instrumentation.

[c]The vessel was made of ASTM-A302 manganese-moly steel with 0.25 in. of types 304L, 308, and 309 stainless steel cladding. Design pressure was 2,500 psig.

[d]Seed 1 and Seed 2 of Core 2 had enrichments of 90.4% and 93.2%, respectively.

Table 2.6.4. Depletion of Shippingport PWR Core 1 seed fuel[a]

Seed	Initial loading (kg U-235)	Depletion (kg U-235)	Final loading (kg-U-235)
Seed 1	75	33.68	41.32
Seed 2	90	45.12	44.88
Seed 3	90	42.82	47.18
Seed 4	90	39.39	50.61
Total	345	161.01	183.99

[a]Initial enrichment of all PWR Core 1 seed fuel was 90.4%.
Source: Connors 1989.

Table 2.6.5. Shippingport PWR Core 2 Seed 1 uranium loading[a]

Cluster	As built, kg		End of life, kg		% U-235 Depletion
	U^{total}	U-235	U^{total}	U-235	
G-2A-G01-01	18.57	16.79	11.30	7.70	54.1
-02	18.57	16.79	11.30	7.70	54.1
-03	18.57	16.79	11.30	7.70	54.1
-04	18.59	16.81	11.10	7.45	55.7
-05	18.59	16.81	11.31	7.71	54.1
-07	18.57	16.80	11.30	7.70	54.2
-08	18.58	16.80	12.79	9.42	43.9
-09	18.57	16.80	11.30	7.70	54.2
-10	18.57	16.79	12.79	9.42	55.7
-11	18.56	16.79	11.08	7.44	55.7
-12	18.58	16.80	11.30	7.71	54.1
-14	18.57	16.79	11.09	7.44	55.7
-15	18.53	16.76	11.07	7.43	55.7
-16	18.54	16.76	12.77	9.40	43.9
-17	18.55	16.78	11.08	7.44	55.7
-18	18.57	16.79	11.08	7.44	55.7
-19	18.56	16.78	12.78	9.41	43.9
-20	18.56	16.79	11.08	7.44	55.7
G-2A-G01-21	18.55	16.78	11.29	7.69	54.2
G-2A-G01-06					
Subassembly	4.65	4.21	2.78	1.87	55.6
Subassembly	4.64	4.20	2.77	1.86	55.7
Section	4.54	4.10	2.71	1.82	55.6
Section	3.33	3.02	1.99	1.34	55.6
Section	0.92	0.84	0.55	0.37	55.6
Plate	0.10	0.09	0.06	0.04	56.0
Plate	0.10	0.09	0.06	0.04	56.0
Plate	0.28	0.26	0.17	0.11	55.6
Total	389.81	335.81	230.20	158.79	

[a]Initial enrichment of PWR Core 2 Seed 1 was 90.4%. Source: Connors 1988.

Table 2.6.6. Shippingport PWR Core 2 Seed 2 uranium loading[a]

Cluster	As built, kg		End of life, kg		% U-235 Depletion
	U^{total}	U-235	U^{total}	U-235	
G-2B-C01-04	20.96	19.53	17.58	14.2	27.2
-C01-11	20.95	19.51	17.58	14.2	27.2
-C01-16	20.94	19.51	17.58	14.2	27.2
-C02-02	20.94	19.51	14.05	11.3	41.8
-C02-05	20.96	19.52	14.05	11.3	41.8
-C02-06	20.93	19.50	14.05	11.3	41.7
-C02-07	20.95	19.52	14.74	11.9	39.0
-C02-08	20.93	19.50	14.74	11.9	38.9
-C02-09	20.94	19.50	14.74	11.3	41.7
-C02-10	20.94	19.50	14.74	11.9	38.9
-C02-13	20.94	19.51	14.74	11.9	39.0
-C02-14	20.96	19.52	14.05	11.3	41.8
-C02-18	20.94	19.51	14.05	11.3	41.8
-C02-19	20.94	19.50	14.74	11.9	38.9
-C02-20	20.94	19.51	14.74	11.9	39.0
-E01-17	20.95	19.51	17.58	14.2	27.2
-E02-01	20.96	19.52	14.05	11.3	41.8
-E02-03	20.98	19.54	14.05	11.3	41.9
-E02-12	20.95	19.52	14.74	11.9	39.0
-E02-15	20.94	19.51	14.74	11.9	39.0
Total	418.94	390.25	301.33	243.0	

[a]Initial enrichment of Core 2 Seed 2 was 93.2%. Source: Connors 1988.

Table 2.6.7. Summary of fuel reprocessed at the Nuclear Fuel Services' reprocessing plant at West Valley, New York

Lot No.	Reactor	Commencement date	Number of assemblies[a]	Uranium (MT)	Plutonium (kg)	Avg. burnup (MWd/MTIHM)
2	NPR	04/22/66		19.7	1.7	75
1	NPR	05/20/66		28.8	2.3	75
3	NPR	07/15/66		46.7	50.9	1,287
4	Dresden-1	11/12/66	484	50.0	191.0	8,500
5	Yankee Rowe	06/07/67	186	49.8	185.1	11,200
6	NPR	09/02/67		26.6	52.6	2,700
7	NPR	12/02/67		26.1	47.4	2,700
8	NPR	01/06/68		42.4	75.4	2,700
9	NPR	05/05/68		38.8	79.1	2,850
10	NPR	06/29/68		55.3	115.7	2,870
11	Indian Point-1	11/15/68	124	1.1[b]	–	–
12	NPR	02/13/69		48.9	102.5	2,850
13	Yankee Rowe	05/14/69	74	19.6	176.0	20,500
14	NPR	08/16/69		30.3	–	c
15	Dresden-1	10/01/69	203	21.5	104.6	10,900
16	Indian Point-1	11/23/69	80	15.6	107.6	15,794
17	Yankee Rowe	06/02/70	36	9.3	95.6	24,381
18	Pathfinder	08/14/70	143	9.6	7.1	2,231[d]
19	Big Rock Point	11/25/70	142	18.4	72.8	9,212
20	Indian Point-1	01/11/71	40	7.6	68.1	23,455
21	NPR	02/25/71		15.8	25.4	2,868
22	Bonus Superheater	04/15/71		1.7	0.9	1,552
22	Bonus Boiler	04/18/71		2.4	4.0	3,230
23	Humboldt Bay	05/02/71	270	20.8	87.2	10,466
24	Yankee Rowe	07/16/71	36	9.5	95.7	23,653
25	CVNPA-Parr	10/04/71		3.5	11.6	9,783
26	Big Rock Point	11/30/71	45	5.8	27.9	13,567
27	SEFOR	12/12/71		–	95.5	–

[a]Estimates, based on the total U content and the U per assembly.

[b]The initial core for Indian Point-1 used highly enriched uranium and thorium. Approximately 16 metric tonnes of heavy metal (uranium + thorium) were reprocessed.

[c]This batch of fuel from NPR had not been irradiated.

[d]Average burnup is for irradiated fuel only; 96 assemblies had been irradiated and 47 assemblies had not.

Table 2.6.8. Typical composition of BWR fuel crud

Element	Crud composition (wt % metal)
Iron	87
Copper	2.0
Zinc	4.4
Nickel	3.3
Manganese	2.2
Chromium	1.1
Cobalt	0.3

Table 2.6.9. Primary radioisotopes in PWR and BWR crud

Radioisotope	Half-life (days)
^{51}Cr	27.7
^{54}Mn	312
^{58}Co	70.8
^{59}Fe	44.6
^{60}Co	1924
^{65}Zn	244

Table 2.6.10. Typical composition of PWR fuel crud

Element	Crud composition (wt % metal)
Iron	70–80
Nickel	19–24
Chromium	0.8–2.5
Cobalt	0.11

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. The effect of SFD hardware on the operation of the CRWMS will be significant only if large quantities of fuel assemblies are consolidated, either at-reactor or at an interim storage facility. Data sources are listed in the bibliography at the end of this section. The primary incentive for consolidation is to reduce the volume occupied by the assembly fuel pins. However, the SFD hardware which was removed from the assembly must still be disposed of.

SFD hardware for PWR fuel assemblies includes guide tubes; instrument tubes; top and bottom nozzles; grid spacers; holddown 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 absorbers, and/or instrumentation are inserted. Recently, most guide and instrument tubes have been made of Zircaloy, although early assemblies used 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 also been used. Grid spacers, which historically have been made of a spring-like material such as Inconel, have recently been made of Zircaloy because of its low neutron absorption cross section. The grid spacers are attached 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 holddown springs are typically made of a nickel-based alloy and are used to hold the fuel assembly against the bottom core support plate.

For BWR fuel assemblies, SFD hardware includes the top and bottom tie plates, compression springs for individual fuel rods, grid spacers, and water rods (or water cross or channel). In a BWR assembly, structural support is provided by the grid spacers, the top and bottom tie plates, fueled tie rods, and the 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 not to damage the integrity of the fuel rod cladding. The water rod(s) is a Zircaloy tube which provides additional nonboiling water for neutron moderation. BWR assemblies typically use a separate compression spring for each individual fuel rod. These springs, which are 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. For example, structural support for the fuel assemblies used at the Palisades and Yankee-Rowe reactors is provided by solid bars of Zircaloy rather than guide tubes. The top end fittings of assemblies for CE 14 x 14 and CE 16 x 16 class reactors are not solid pieces of metal. The top end fittings to these assemblies are two flat plates separated by five large metal posts surrounded by Inconel holddown springs. New fuel designs for BWR reload fuel have water crosses, water channels, or large-diameter water rods.

Nonfueled integral 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 the consolidation, but would be included in the consolidated canister with the fuel rods. The extra handling operation involved with identifying, separating, and safely disposing of the nonfueled rods 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 will remain with the assembly, where the radioactivity of the hardware is generally small compared to the radioactivity of the fuel and fission products ("Hot spots" could be caused by ^{60}Co in the end fittings of some assemblies). If fuel assemblies are consolidated, it will be important to know how much SFD hardware there will be, what the radiological characteristics of it are, and whether or not it is GTCC. This section provides the basis for such information.

2.7.2 Quantitative Characterization

A general description of the major aspects of SFD hardware for the different fuel designs is included in Appendix 2A to this report. Specific pieces of SFD hardware are described in as much detail as possible in the LWR Assemblies Database. This description includes the name of the specific pieces, the number of pieces per assembly, the weights, and the construction materials. For each assembly type, these parts are listed on page 2 of the Physical Description Report.

Summary quantitative information on how much SFD hardware is associated with PWR fuel assemblies in given in Table 2.7.1 and for BWR assemblies in Table 2.7.2. The number of assemblies of each type is addressed by the LWR Quantities Database.

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.

Another nuclide of considerable interest in activated hardware is ^{60}Co, even though this is not a "Class C" nuclide. The interest in ^{60}Co is the result of the two high-energy gamma rays emitted by the decay, and the potential shielding problem caused by these gamma rays. This potential shielding problem is of concern, not only for SFD hardware, but also for hardware which is still a part of an intact assembly. In particular, the bottom end fitting and the top end fitting are both rather massive and made of stainless steel and therefore contain appreciable amounts of natural cobalt (the source of ^{60}Co after neutron activation). The bottom end fitting, being closer to the active core zone than the top end fitting, becomes more highly activated.

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 (mainly stainless steel 304). These materials were chosen for their 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 the upper limit of 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 necessarily affect the physical and chemical characteristics of the material. Thus, as long as the niobium concentrations are below the acceptable 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 upper limits and ASTM specifications. However, more stringent controls by the fuel vendors on the purchase specifications for these materials may well lead to significantly lower initial concentrations. Recent experimental work (Luksic 1992) show very low (< 5 ppm) levels of niobium in some unirradiated samples of Zircaloy. The values used for nitrogen, cobalt, nickel, and niobium in the ORIGEN2 calcultions are given below. Detailed compositions of these materials are given in Table 2.7.3.

Composition of Materials

Material	Nitrogen	Cobalt	Nickel	Niobium
Inconel-718	1300 ppm	4700 ppm	52.0%	5.55%
Inconel X-750	1300 ppm	6500 ppm	72.2%	0.9%
Nicrobraze 50	66 ppm	381 ppm	74.4%	---
St. Steel 302	1300 ppm	800 ppm	8.92%	100 ppm
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 computer code ORIGEN2 (Croff 1980) has been used to estimate the buildup of activation products from these isotopes that are present in SFD hardware. ORIGEN2 calculates the concentration of all nuclides 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.

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 and the recently updated BWR and PWR standard- and extended-burnup cross sections (Ludwig 1989) were used to calculate hardware activation for PWRs and BWRs. Activation of the materials used in SFD hardware has been calculated for a lower, near-standard burnup and a higher burnup for both BWRs and PWRs (the reactor conditions are defined in Table 2.4.1). For PWRs, concentrations of isotopes in materials exposed to lower burnup (30,000 MWd/MTIHM) were calculated using the standard PWR model; concentrations in materials exposed to higher burnup (60,000 MWd/MTIHM) were calculated using the PWR extended burnup model. An initial enrichment of 3.11% was used for the former and 4.73% for the latter. For BWRs, concentrations of isotopes in materials exposed to lower burnup (30,000 MWd/MTIHM) were calculated using the standard BWR model; concentrations in materials exposed to higher burnup (50,000 MWd/MTIHM) were calculated using the BWR extended burnup model. An initial enrichment of 2.93% was used for the former and 3.74% for the latter.

For a specific fuel assembly, ORIGEN2 results can be used to estimate the radionuclide inventory of materials irradiated in the fueled region of the reactor. Input parameters include the composition of the material and the irradiation history. The results of the calculation are directly applicable to materials irradiated in the core's fueled region. Outside the fueled region, the results are

not applicable due to changes in the absolute magnitude of the neutron flux and shifts in the neutron spectrum. For SFD hardware, much of the material of interest is located at the end fittings, outside of the fueled region.

To estimate the radionuclide inventory in these components, scaling factors are applied to ORIGEN2 calculations to compensate for the changed neutron flux outside the fueled region. The scaling factors used for these calculations are based upon the experiential work of Luksic (Luksic 1989). In this work, 38 samples were obtained from three spent fuel assemblies. Each sample was individually analyzed for both elemental composition and for radionuclide conten:. Based on the results of the analysis, scaling factors were developed that relate the activation rate in the reactor's fueled region to those outside. These factors, which have an uncertainty of $\pm$50%, are presented below; these are the nominal actual values. These factors are applied to ORIGEN2 incore results by multiplying the fueled-region radionuclide inventories, as calculated in ORIGEN2, by the appropriate scaling factor for the region in which the material is located.

ORIGEN2 Activation Scaling Factors
(relative to fueled region).

Region	PWR	BWR
Top End Fitting	0.1	0.1
Gas Plenum	0.2	0.2
Fueled Region	1.0	1.0
Bottom End Fitting	0.2	0.15

It should be noted that the results of Luksic's experimental work for the incore samples were compared directly to ORIGEN2 runs, and reasonable agreement was found. In performing this comparison, Luksic used the original ORIGEN2 cross section sets for BWRs and PWRs, not Ludwig's updated cross section sets. Several specific differences have been identified between the results from these two cross section sets, including a fairly significant increase (40 - 60%) in the amount of ^{60}Co predicted in spent fuel by the new cross section sets. Other differences include a decrease in the predicted amount of ^{14}C and increases in the predicted amount of ^{93}Zr. These differences do not affect the activation scaling factors, which were derived from the experimental data alone, but do highlight possible discrepancies between ORIGEN2 calculations and experimental predictions which should be addressed in future validation of ORIGEN2.

These global factors are only useful for average radionuclide inventory estimates, and should not be applied to small sections due to significant variations in the neutron flux from even a small change in position. Based on Luksic's sample analysis, it was found that the activation rate of small regions within the top end fitting of a PWR fuel assembly can vary by a large factor. For example, a sample taken from the lower portion of the top end fitting could have 10 times more radioactivity than a similar sample from the upper portion of the top end fitting. The bottom end fitting showed much less variation, probably because it is physically smaller. It is therefore important to note that these are average scaling factors applicable only to the general regions for which they were developed.

Some vendor-specific differences were noted. The scaling factor for the top end fitting of the Combustion Engineering assembly is one-half that for a Westinghouse, GE, or B&W assembly. This is a result of the longer gas plenum in the Combustion Engineering fuel rods which places the Combustion Engineering top end fitting further away from the fueled region and in a lower flux region than for the other end fittings. However, the flux has an order of magnitude reduction over the length of the end fittings for both assembly types. The General Electric bottom end fitting has a lower scaling factor than any of the pressurized water reactors. This is apparently due to the greater length of the end fitting and a greater reduction of flux over its length. These scaling factors are applied directly to the results of ORIGEN2 incore activation calculations.

2.7.4 Results of Radiological Characterization

ORIGEN2 runs have been made with five materials for each reactor type. The specific ORIGEN2 runs made are shown below:

ORIGEN2 Runs Made for SFD
Hardware Characterization

Reactor	Burnup	Materials
BWR	30/50 GWd	Inconel-718
		Inconel X-750
		Stainless Steel 302
		Stainless Steel 304
		Zircaloy-2
		Zircaloy-4
PWR	30/60 GWd	Inconel-718
		Inconel X-750
		Nicrobraze 50
		Stainless Steel 302
		Stainless Steel 304
		Zircaloy-4

The output of these ORIGEN2 runs consists of the concentrations of an isotope in one kilogram of that material. These concentrations are expressed in grams. The overall photon spectra from the irradiated material are also an output. Curies and watt values are calculated isotope by isotope. These results have been downloaded from the ORIGEN2 output to data files in the LWR Assemblies Database.

Two Radiological Description Reports are available from the LWR Assemblies Database. One 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 the 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 X-750, respectively. These reports are for materials exposed to 30,000 MWd 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 GE BWR/4-6 8 X 8 GE Prepressurized fuel assembly is given in Table 2.7.7.

These results generally show that ^{94}Nb is the major cause for escalation of SFD hardware into the GTCC range. This is sometimes true even where natural niobium is merely a trace impurity. Where it is an added alloy constituent, the hardware component will be far above the GTCC limit. The other significant contributor is ^{63}Ni, primarily for the Inconels, which are high in nickel content.

2.7.5 Bibliography for Section 2.7

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

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

Exxon 1986. Exxon Nuclear Company, Inc. *Fuel Assembly Characterization Data*, December 1986.

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

Ludwig 1989. S. B. Ludwig and J. P. Renier, *Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code*, ORNL/TM-11018, December 1989.

Luksic 1989. A. T. Luksic, *Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Disposal*, PNL-6906, Vol. 1, June 1989.

Luksic 1992. A. T. Luksic, Pacific Northwest Laboratories, personnel communication, 1992.

NRC 1990. Nuclear Regulatory Commission Bulletin No. 90-02, "Loss of Thermal Margin Caused by Channel Box Bow", March 20, 1990.

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

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

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

Class Name Assembly Type Name	SFD Hardware Weight
B&W 15 x 15 Assembly Class	
B&W 15 x 15 B&W Mark B	35.6 kg
B&W 15 x 15 B&W Mark BZ	35.6 kg
B&W 17 x 17 Assembly Class	
B&W 17 x 17 B&W Mark C	43.2 kg
CE 14 x 14 Assembly Class	
CE 14 x 14 CE	29.8 kg
CE 14 x 14 ANF	33.3 kg
CE 14 x 14 WE	34.1 kg
CE 16 x 16 Assembly Class	
CE 16 x 16 CE	40.1 - 42.6 kg
CE SYSTEM 80 Assembly Class	
CE SYSTEM 80 16 x 16 CE	44.0 kg
WE 14 X 14 Assembly Class	
WE 14 X 14 WE Standard	32.0 kg
WE 14 X 14 WE LOPAR	31.8 kg
WE 14 X 14 WE OFA	32.1 kg
WE 14 X 14 WE Vantage 5	NA
WE 14 X 14 B&W	NA
WE 14 X 14 ANF	28.4 kg
WE 14 X 14 ANF Toprod	24.6 kg
WE 15 X 15 Assembly Class	
WE 15 X 15 WE Standard	35.8 kg
WE 15 X 15 WE LOPAR	NA
WE 15 X 15 WE OFA	32.6 kg
WE 15 X 15 WE Vantage 5	NA
WE 15 X 15 ANF	27.3 kg
WE 17 X 17 Assembly Class	
WE 17 X 17 WE LOPAR	29.6 kg
WE 17 X 17 WE OFA	32.3 kg
WE 17 X 17 WE Vantage 5	NA
WE 17 X 17 WE Vantage 5H	NA
WE 17 X 17 WE Vantage +	NA
WE 17 X 17 ANF	34.6 kg
WE 17 X 17 B&W	NA
SOUTH TEXAS Assembly Class	
South Texas 17 x 17 WE	NA

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

Class Name Assembly Type Name	SFD Hardware Weight[a]	Comments
GE BWR/2,3 Assembly Class		
GE BWR/2,3 7 x 7 GE-2a	8.4 kg	b
GE BWR/2,3 7 x 7 GE-2b	8.44 kg	
GE BWR/2,3 7 x 7 GE-3	8.3 kg	
GE BWR/2,3 8 x 8 GE-4	9.9 kg	c
GE BWR/2,3 8 x 8 GE-5	10.9 kg	b,c
GE BWR/2,3 8 x 8 GE Prepressurized	10.9 kg	b,c
GE BWR/2,3 8 x 8 GE Barrier	10.9 kg	b,c
GE BWR/2,3 8 x 8 GE-8a	12.7 kg	b,c,d
GE BWR/2,3 8 x 8 GE-8b	12.7 kg	b,c,d
GE BWR/2,3 8 x 8 GE-9a	11.2 kg	b,c,e
GE BWR/2,3 8 x 8 GE-9b	11.2 kg	b,c,e
GE BWR/2,3 7 x 7 ANF	9.8 kg	
GE BWR/2,3 8 x 8 ANF	8.1 kg	
GE BWR/2,3 8 x 8 ANF Prepressurized	8.1 kg	
GE BWR/2,3 9 x 9 ANF	9.3 kg	
GE BWR/2,3 9 x 9 ANF Model 9-5	11.3 kg	b
GE BWR/2,3 9 x 9 ANF IX	11.3 kg	b,c
GE BWR/2,3 9 x 9 ANF 9X	11.3 kg	b,c
GE BWR/4-6 Assembly Class		
GE BWR/4-6 7 x 7 GE-2	8.4 kg	
GE BWR/4-6 7 x 7 GE-3a	8.4 kg	
GE BWR/4-6 7 x 7 GE-3b	8.4 kg	
GE BWR/4-6 8 x 8 GE-4a	10.0 kg	c
GE BWR/4-6 8 x 8 GE-4b	10.0 kg	b,c
GE BWR/4-6 8 x 8 GE-5	11.0 kg	c
GE BWR/4-6 8 x 8 GE Prepressurized	11.0 kg	b,c
GE BWR/4-6 8 x 8 GE Barrier	11.0 kg	b,c
GE BWR/4-6 8 x 8 GE-8a	12.9 kg	b,c,d
GE BWR/4-6 8 x 8 GE-8b	12.9 kg	b,c,d
GE BWR/4-6 8 x 8 GE-9a	11.3 kg	b,c,e
GE BWR/4-6 8 x 8 GE-9b	11.3 kg	b,c,e
GE BWR/4-6 8 x 8 ANF	9.0 kg	
GE BWR/4-6 8 x 8 ANF Prepressurized	9.0 kg	
GE BWR/4-6 9 x 9 ANF	9.3 kg	
GE BWR/4-6 9 x 9 ANF Model 9-5	11.3 kg	b
GE BWR/4-6 9 x 9 ANF IX	11.3 kg	b,c
GE BWR/4-6 9 x 9 ANF 9X	11.3 kg	b,c
GE BWR/4-6 8 x 8 WE QUAD+	NA	

[a] The weight of fuel channels is directly dependent of the thickness of the channel. 80, 100, and 120 mil fuel channels weigh approximately 30, 38, and 45 kg, respectively. Since the thickness of the channel is not assembly type specific, the weight of fuel channels is not included in the SFD hardware weights given.

[b] Estimated on the basis of similar assemblies.

[c] Estimated on the basis of calculated weights of water rods and water channels.

[d] Four water rods assumed.

[e] One large-diameter water rod assumed.

Table 2.7.3. Elemental Compositions of Materials Used for SFD Hardware ORIGEN2 Calculations (All concentrations in parts per million).

Element	Inconel-718	Inconel X-750	Nicrobraze 50	St. Steel 302	St. Steel 304	Zircaloy-2	Zircaloy-4
Hydrogen						13.0	13.0
Boron			50.0			0.33	0.33
Carbon	400.0	399.0	100.0	1,500.0	800.0	120.0	120.0
Nitrogen	1,300.0	1,300.0	66.0	1,300.0	1,300.0	80.0	80.0
Oxygen			430.0			950.0	950.0
Aluminum	5,992.0	7,982.0	100.0			24.0	24.0
Silicon	1,997.0	2,993.0	511.0	10,000.0	10,000.0		
Phosphorus			103,244.0	450.0	450.0		
Sulfur	70.0	70.0	100.0	300.0	300.0	35.0	35.0
Titanium	7,990.0	24,943.0	100.0			20.0	20.0
Vanadium						20.0	20.0
Chromium	189,753.0	149,660.0	149,709.0	180,000.0	190,000.0	1,000.0	1,250.0
Manganese	1,997.0	6,984.0	100.0	20,000.0	20,000.0	20.0	20.0
Iron	179,766.0	67,846.0	471.0	696,450.0	697,150.0	1,500.0	2,250.0
Cobalt	4,694.0	6,485.0	381.0	800.0	800.0	10.0	10.0
Nickel	519,623.0	721,859.0	744,438.0	89,200.0	89,200.0	500.0	20.0
Copper	999.0	499.0				20.0	20.0
Zirconium			100.0			979,589.0	979,069.0
Niobium	55,458.0	8,980.0		100.0	100.0	120.0	120.0
Molybdenum	29,961.0						
Cadmium						0.25	0.25
Tin						16,000.0	16,000.0
Hafnium						78.0	78.0
Tungsten			100.0			20.0	20.0
Uranium						0.2	0.2

Table 2.7.4. Sample Material Report for Zircaloy-2 from LWR Assemblies Database

Radiological Report (Curies)
for
Zircaloy-2
Boiling Water Reactor - In Core Zone
Standard Burnup (30000 MWd) - 15 Years After Discharge

Isotope	Curies
(Values are per kg of irradiated material)	
C 14	1.370E-03
FE 55	1.151E-02
CO 60	2.019E-01
NI 59	1.888E-04
NI 63	2.813E-02
SR 90	6.390E-06
Y 90	6.390E-06
ZR 93	1.196E-03
NB 93M	6.546E-04
NB 94	2.281E-04
TC 99	2.856E-08
SN119M	3.779E-06
SN121M	1.997E-03
SB125	1.545E-01
TE125M	3.770E-02
I129	9.397E-17
TOTAL	4.394E-01

Table 2.7.5. Sample Material Report for Stainless Steel 304 from LWR Assemblies Database

Radiological Report (Curies)
for
St.Steel 304
Boiling Water Reactor - In Core Zone
Standard Burnup (30000 MWd) - 15 Years After Discharge

Isotope	Curies
(Values are per kg of irradiated material)	
C 14	2.226E-02
MN 54	1.959E-04
FE 55	5.267E+00
CO 60	1.628E+01
NI 59	3.368E-02
NI 63	5.011E+00
NB 94	1.901E-04
TOTAL	2.661E+01