DOE/RW-0184
VOLUME 1 of 6
UC-70, -71, and -85

Office of Civilian Radioactive Waste Management

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

DECEMBER 1987

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

PREFACE

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

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

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

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

ORDER FORM FOR PC DATA BASES

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ACKNOWLEDGEMENTS

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

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

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1. SUMMARY

1.1 INTRODUCTION

1.1.1 Objectives

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

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

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

1.1.2 Report and Data Base Structure

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

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

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

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

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

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

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

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

1.1.3 Methodology

1.1.3.1 Data Sources

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

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

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*Acronyms are defined on pp. ix and x.

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

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

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

1.1.3.2 Data Processing

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

1.1.3.3 Radiological Characteristics

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

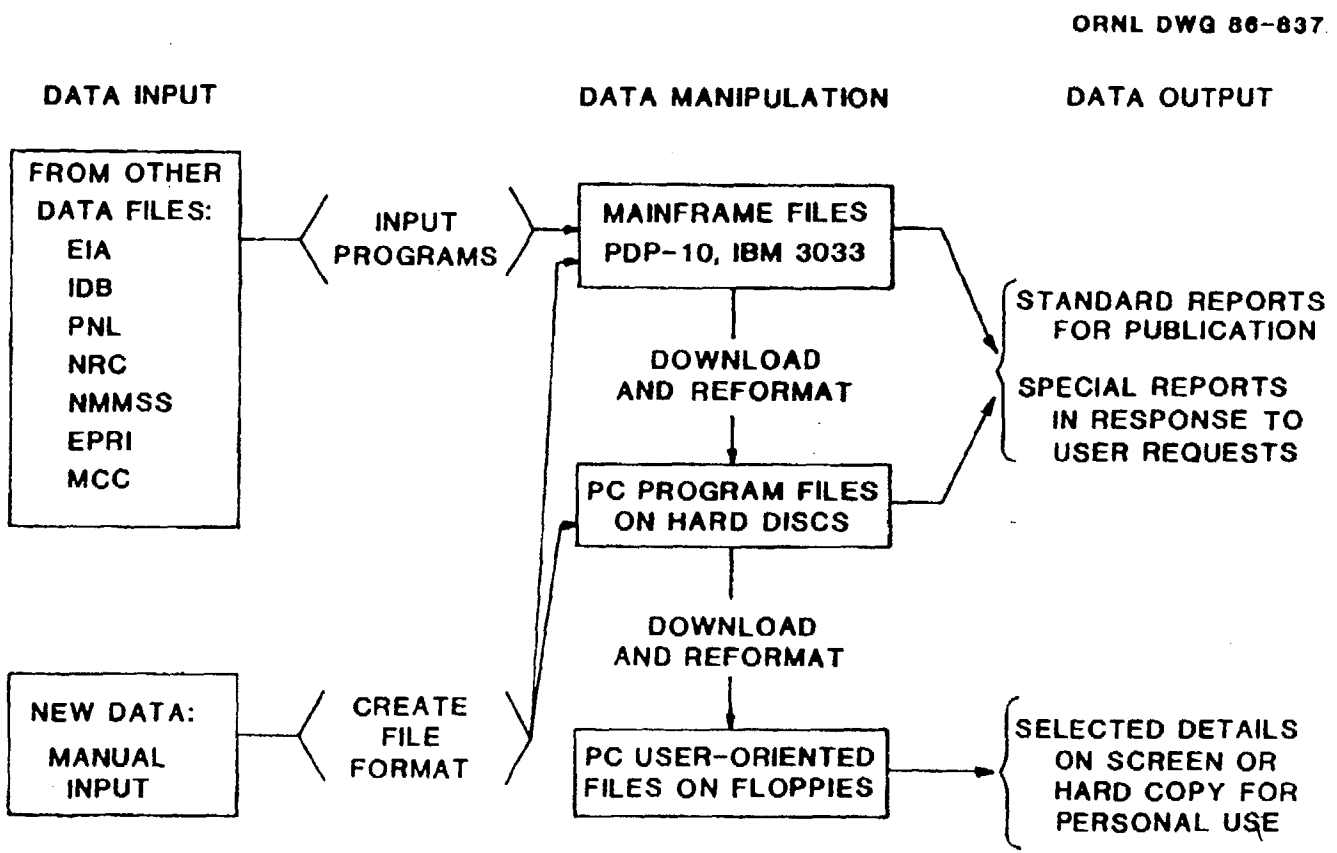

Fig. 1.1. Data processing in the Characterization Data Base.

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

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

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

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

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

1.1.4 Menu-Driven PC Data Bases

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

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

INFO

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

1.1.5 QA Plan/Accuracy and Reliability of Data

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

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

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

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

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

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

1.2 LWR SPENT FUEL (see Sect. 2)

1.2.1 Scope

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

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

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

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

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

1.2.2 Assemblies

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

1.2.3 Spent Fuel Inventory

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

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

Table 1.1.

CURRENTLY LISTED ASSEMBLIES

PRESSURIZED WATER REACTOR ASSEMBLIES

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

Table 1.2.

CURRENTLY LISTED ASSEMBLIES

BOILING WATER REACTOR ASSEMBLIES

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

Table 1.3. Technical data for each fuel assembly model

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

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

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

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

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

Fuel performance
- Enrichment (range)
- Maximum burnup

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

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

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

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

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

1.2.4 Defective Fuel

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

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

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

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

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

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

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

1.2.5 Special Fuel Forms

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

1.2.6 Spent Fuel Disassembly (SFD) Hardware

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

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

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

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

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

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

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

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

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

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

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

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

1.2.7 Nonfuel Assembly (NFA) Hardware

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

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

1.3 HIGH-LEVEL WASTE (see Sect. 3)

1.3.1 Scope

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

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

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

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

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

Table 1.4. Summary data for high-level waste

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

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

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

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

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

[e]Projected for the year 2020.

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

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

1.3.2 West Valley Demonstration Project

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

1.3.3 Defense Waste Processing Facility

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

1.3.4 Hanford Operations

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

1.3.5 Idaho National Engineering Laboratory

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

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

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

1.4.1 Scope

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

1.4.2 Fort St. Vrain Reactor

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

1.4.3 Peach Bottom I Reactor

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

Table 1.5. Summary of non-LWR spent fuels

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

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

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

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

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

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

[f]Not determined yet.

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

[h]Includes some FFTF and TRIGA fuels.

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

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

1.4.4 Research and Test Reactor Fuel

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

1.4.5 Miscellaneous Fuels

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

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

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

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

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

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

1.5 MISCELLANEOUS WASTES (see Sect. 5)

1.5.1 Scope

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

1.5.2 OCRWM-Generated Wastes

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

1.5.3 Commercial TRU Waste

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

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

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

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

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

1.5.4 Reactor Decommissioning

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

1.5.5 Radioisotope Capsules

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

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

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

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

1.5.6 Routine Reactor Operations

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

1.5.7 Summary of Miscellaneous Wastes

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

Table 1.8. Projected volumes of miscellaneous wastes[a]

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

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

[b]Depends on startup date for these facilities.

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

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

[e]Assumes 65 have been decommissioned.

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

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

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

2. LWR SPENT FUEL

2.1 INTRODUCTION

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

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

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

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

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

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

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

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

2.2 ASSEMBLY DESCRIPTIONS

2.2.1 Overview

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

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

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

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

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

2.2.2 Fuel Assemblies of Pressurized-water Reactors

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

2.2.2.1 Babcock & Wilcox 15 x 15 Array Design

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

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

Babcock & Wilcox 15 x 15 Mark B

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

Babcock & Wilcox 15 x 15 Mark BZ

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

2.2.2.2 Babcock & Wilcox 17 x 17 Mark C Array Design

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

2.2.2.3 Combustion Engineering 14 x 14 Array Design

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

Combustion Engineering 14 x 14 Standard

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

Combustion Engineering 14 x 14 Ft. Calhoun

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

Westinghouse 14 x 14 "Model C"

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

Advanced Nuclear Fuels 14 x 14 CE

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

Advanced Nucelar Fuels 14 x 14 Ft. Calhoun

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

2.2.2.4 Combustion Engineering 15 x 15 Array Design

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

Combustion Engineering 15 x 15

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

Advanced Nuclear Fuels 15 x 15 CE

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

2.2.2.5 Combustion Engineering 16 x 16 Array Design

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

Combustion Engineering 16 x 16 St. Lucie 2

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

Combustion Engineering 16 x 16 Ark. Nucl. 2

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

Combustion Engineering 16 x 16 San Onofre

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

Combustion Engineering 16 x 16 System 80

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

2.2.2.6 Westinghouse 14 x 14 Array Design

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

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

Westinghouse 14 x 14 Standard/SC

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

Westinghouse 14 x 14 Standard/ZCA

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

Westinghouse 14 x 14 Standard/ZCB

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

Westinghouse 14 x 14 OFA

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

Babcock & Wilcox 14 x 14 Ginna

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

Advanced Nuclear Fuel 14 x 14 WE

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

Advanced Nuclear Fuel 14 x 14 Top Rod

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

2.2.2.7 Westinghouse 15 x 15 Array Design

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

Westinghouse 15 x 15 Standard/SC

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

Westinghouse 15 x 15 Standard/ZC

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

Westinghouse 15 x 15 OFA

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

Advanced Nuclear Fuel 15 x 15 WE

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

Babcock & Wilcox 15 x 15 St. Steel

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

2.2.2.8 Westinghouse 15 x 16 Array Design

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

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

Westinghouse 15 x 16 Yankee-Rowe

No information is available at this time.

Combustion Engineering 15 x 16 Yankee-Rowe

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

Advanced Nuclear Fuels 15 x 16 Yankee-Rowe

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

2.2.2.9 Westinghouse 17 x 17 Array Design

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

Westinghouse 17 x 17 Standard

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

Westinghouse 17 x 17 OFA

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

Westinghouse 17 x 17 Vantage 5

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

Westinghouse 17 x 17 XLR

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

Advanced Nuclear Fuel 17 x 17 WE

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

Babcock & Wilcox 17 x 17 Mark BW

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

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

2.2.3 Fuel Assemblies of Boiling-Water Reactors

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

2.2.3.1 Advanced Nuclear Fuels 9 x 9 Array Design

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

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

Advanced Nuclear Fuels 9 x 9 JP-3

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

Advanced Nuclear Fuels 9 x 9 JP-4,5

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

2.2.3.3 Allis-Chalmers 10 x 10 Array Design

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

2.2.3.4 General Electric 6 x 6 Dresden Array Design

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

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

General Electric 6 x 6

No information is available at this time.

Advanced Nuclear Fuels 6 x 6

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

United Nuclear 6 x 6

No information is available at this time.

2.2.3.4 General Electric 6 x 6 Humboldt Bay Array Design

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

General Electric 6 x 6

No information is available at this time.

Advanced Nuclear Fuels 6 x 6

No information is available at this time.

2.2.3.5 General Electric 7 x 7 Array Design

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

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

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

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

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

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

General Electric 7 x 7 BWR/4,5

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

Advanced Nuclear Fuels 7 x 7

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

2.2.3.6 General Electric 8 x 8 Array Design

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

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

General Electric 8 x 8 BWR/2,3

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

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

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

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

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

Advanced Nuclear Fuels 8 x 8 JP-3

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

Advanced Nuclear Fuels 8 x 8 JP-4,5

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

2.2.3.7 General Electric 9 x 9 Array Design

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

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

General Electric 9 x 9

No further information is available at this time.

Advanced Nuclear Fuels 9 x 9 Big Rock Point

No further information is available at this time.

2.2.3.8 General Electric 11 x 11 Array Design

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

2.2.3.9 Westinghouse 8 x 8 Array Design

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

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

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

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

2.2.4 Future Fuel Assembly Designs

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

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

2.2.5 References for Section 2.2

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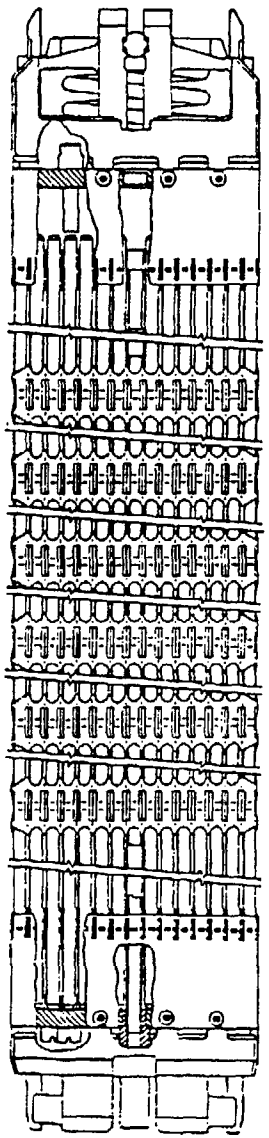
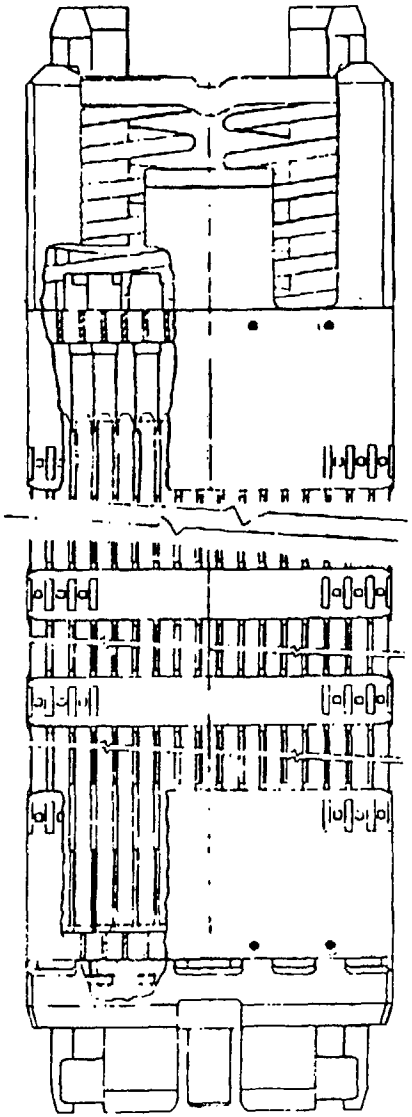
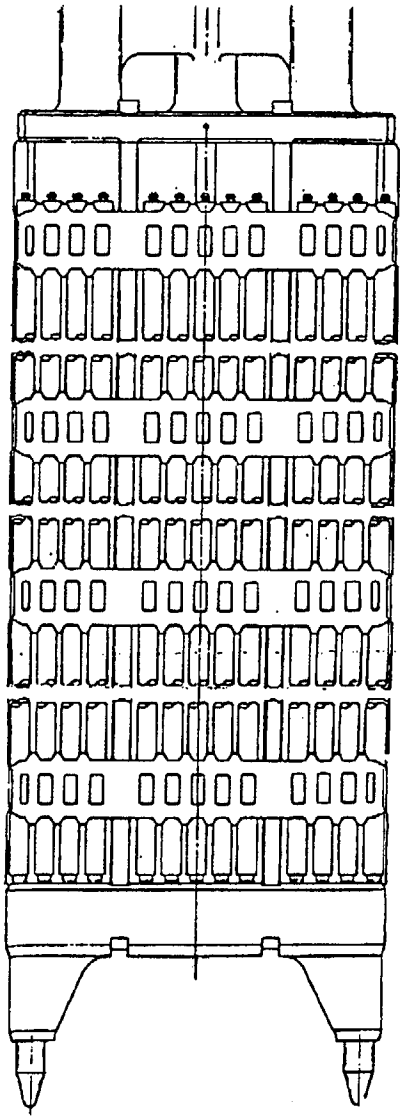


15 X 15



17 X 17

Figure 2.2.1 Schematic drawings of Babcock & Wilcox Fuel Assembly Designs



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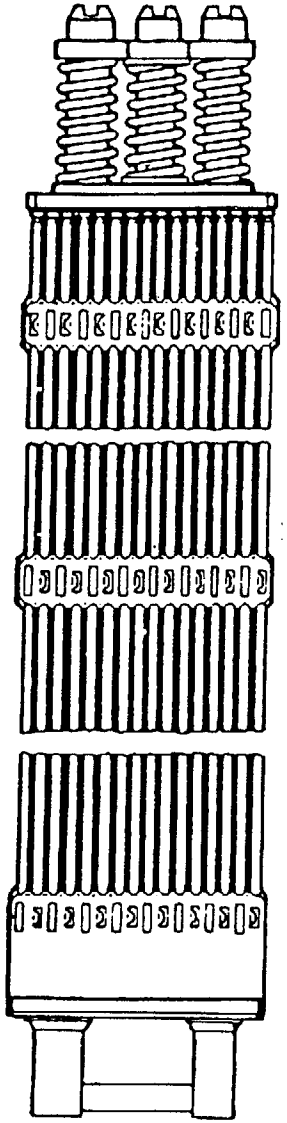
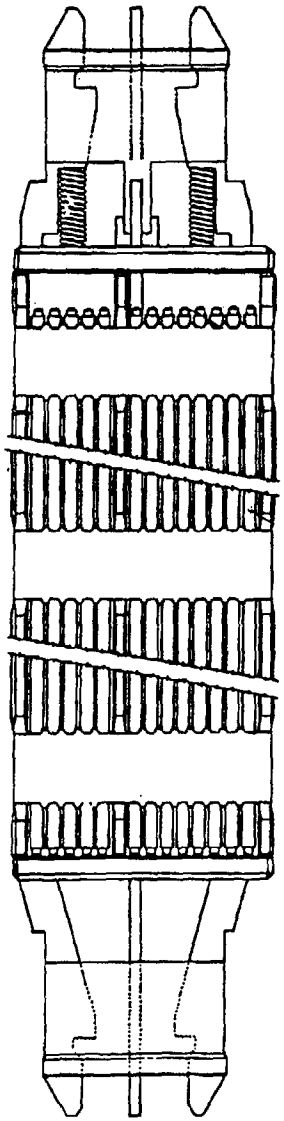
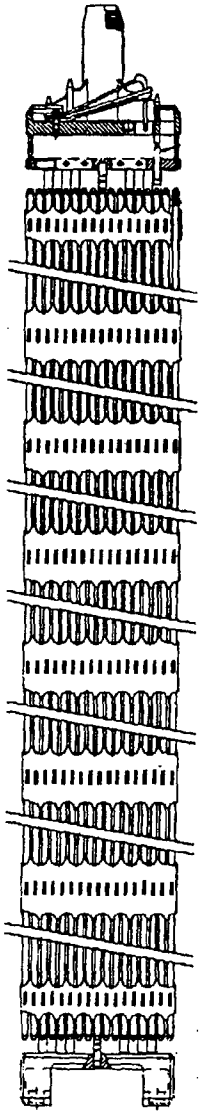


14 X 14, 16 X 16

Figure 2.2.2 Schematic drawings of Combustion Engineering Fuel Assembly Designs



15 X 16



14 X 14, 15 X 15, 17 X 17

Figure 2.2.3 Schematic drawings of Westinghouse Fuel Assembly Designs

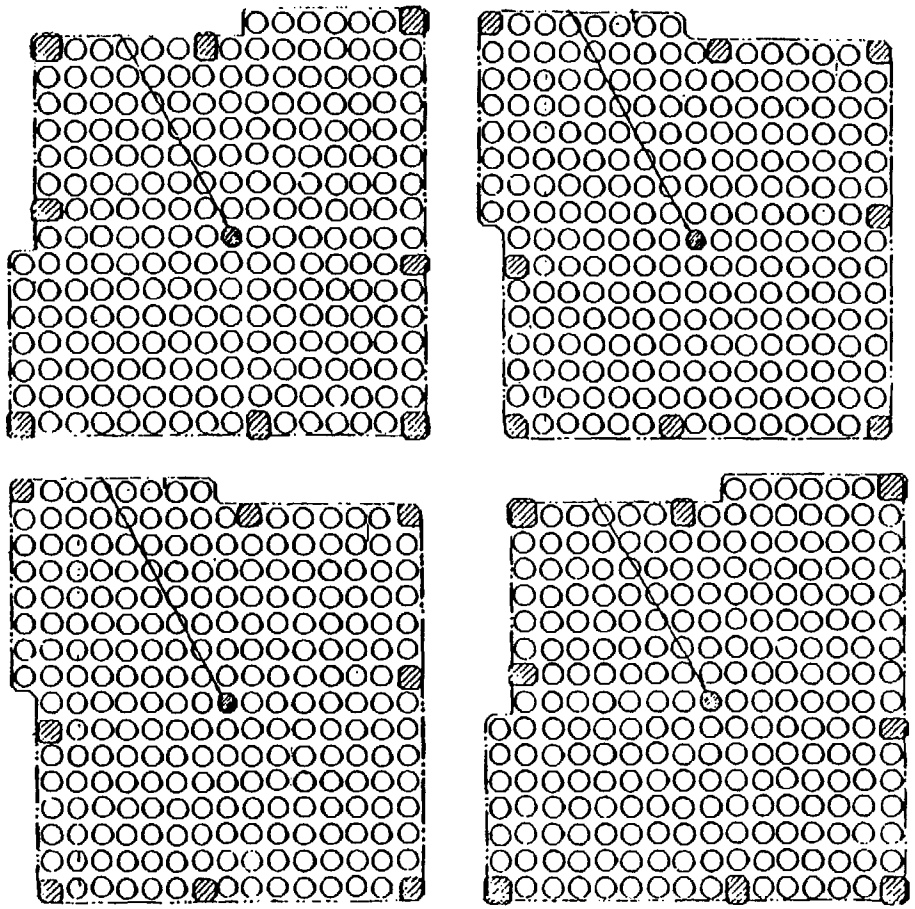
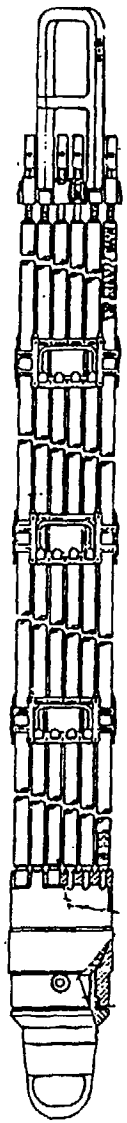
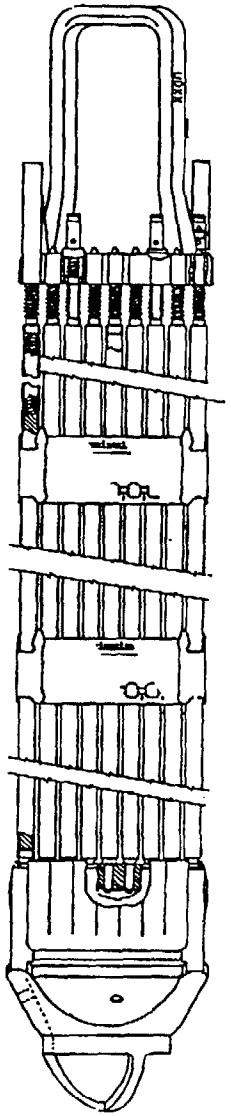


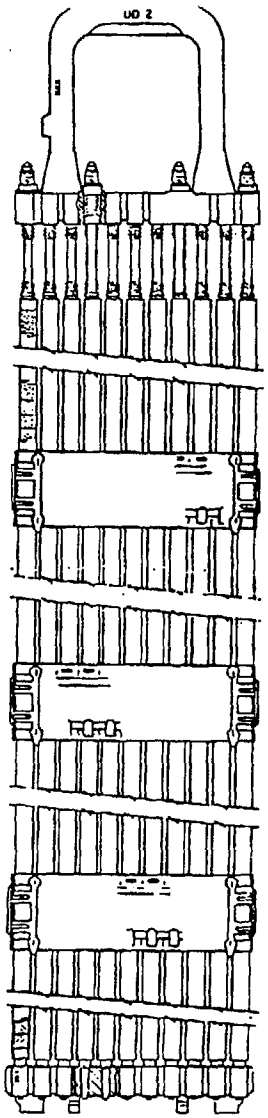
Figure 2.2.4 Schematic drawings of Westinghouse 15 X 16 Fuel Assemblies, Types A and B



6 X 6

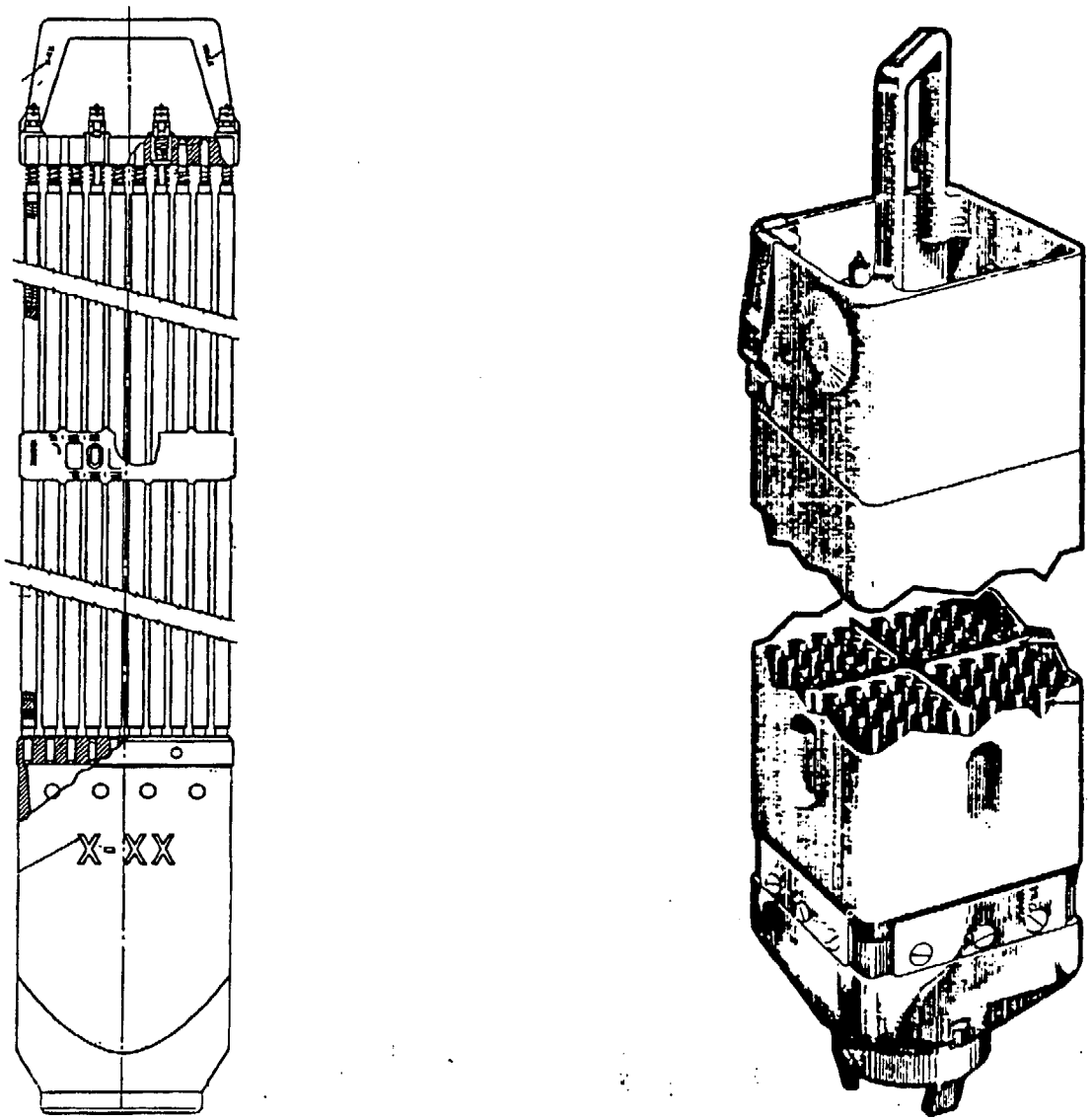


7 X 7, 8 X 8, ANF 9 X 9



11 X 11

Figure 2.2.5 Schematic drawings of General Electric Fuel Assembly Designs



10 X 10

8 X 8 QUAD+

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

Table 2.2.1 Example of Physical Description Report.

PHYSICAL DESCRIPTION REPORT PAGE: 1

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

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

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

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

Comments:

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

PHYSICAL DESCRIPTION REPORT PAGE: 2

Babcock & Wilcox 15 X 15 Mark BZ PWR

FUEL ASSEMBLY HARDWARE PARTS AND MATERIALS

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

Drawing Numbers Associated With Assembly:

02-32958F
11-55248F

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

PHYSICAL DESCRIPTION REPORT PAGE: 3

Babcock & Wilcox 15 X 15 Mark BZ PWR

FUEL ROD DESCRIPTION TABLE

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

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

PHYSICAL DESCRIPTION REPORT PAGE: 4

Babcock & Wilcox 15 X 15 Mark BZ PWR

FUEL ROD DESCRIPTION TABLE continued

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

Comments:

2.3 QUANTITIES OF INTACT SPENT FUEL

2.3.1 Overview

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

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

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

Installed GW(e)

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

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

Cumulative spent fuel discharged (MTIHM)[a]

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

[a]1986 EIA data for CY-1985.

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

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

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

2.3.2 Assembly Type/Reactor Specific Inventories and Projections

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

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

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

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

2.3.3 Reactor-Specific Projections

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

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

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

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

2.3.4 References for Section 2.3

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

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

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

Figure 2.3.1 Overview of LWR Quantities Data Base Historical Data.

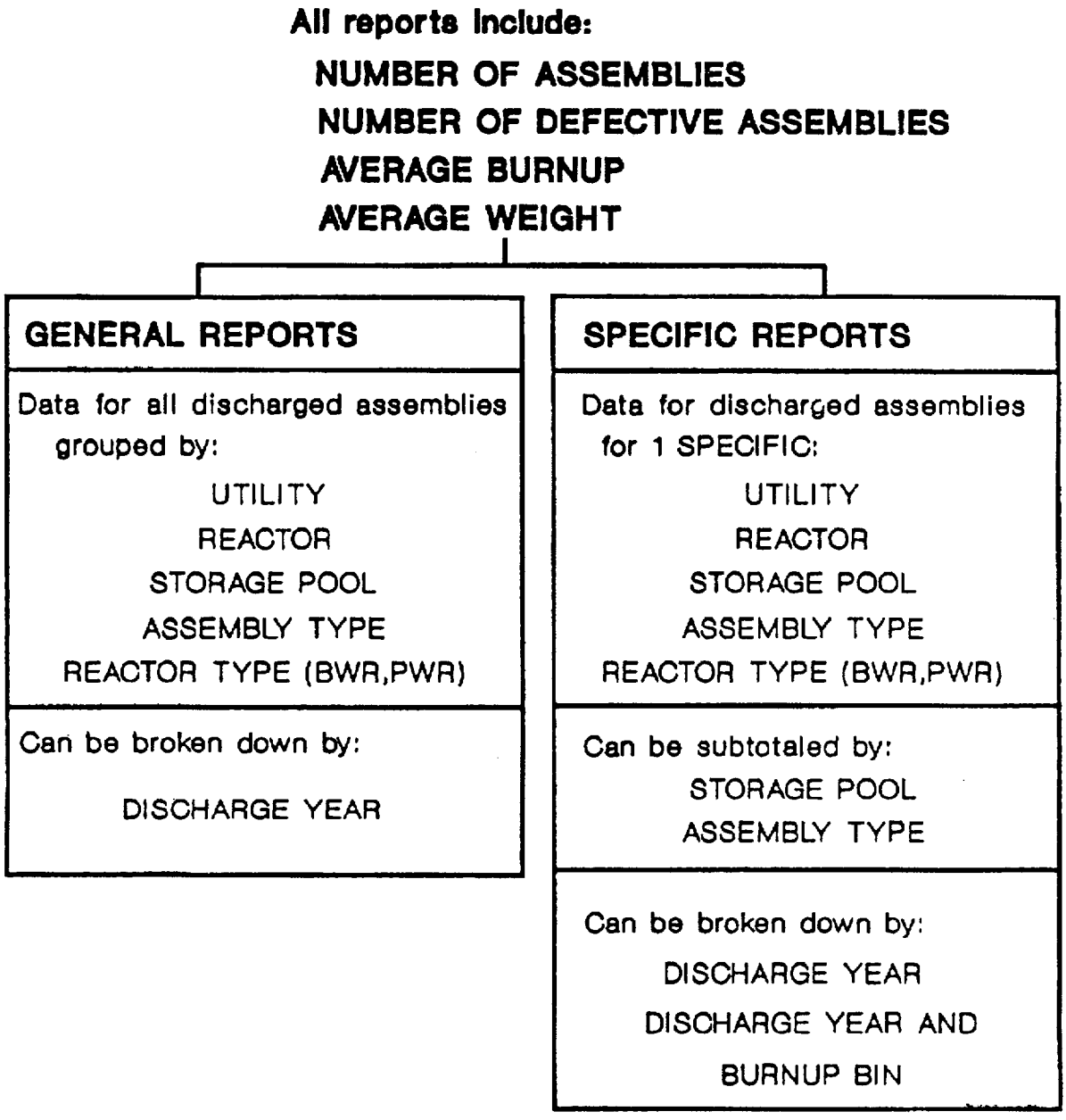

Figure 2.3.2 Overview of LWR Quantities Data Base Projected Data.

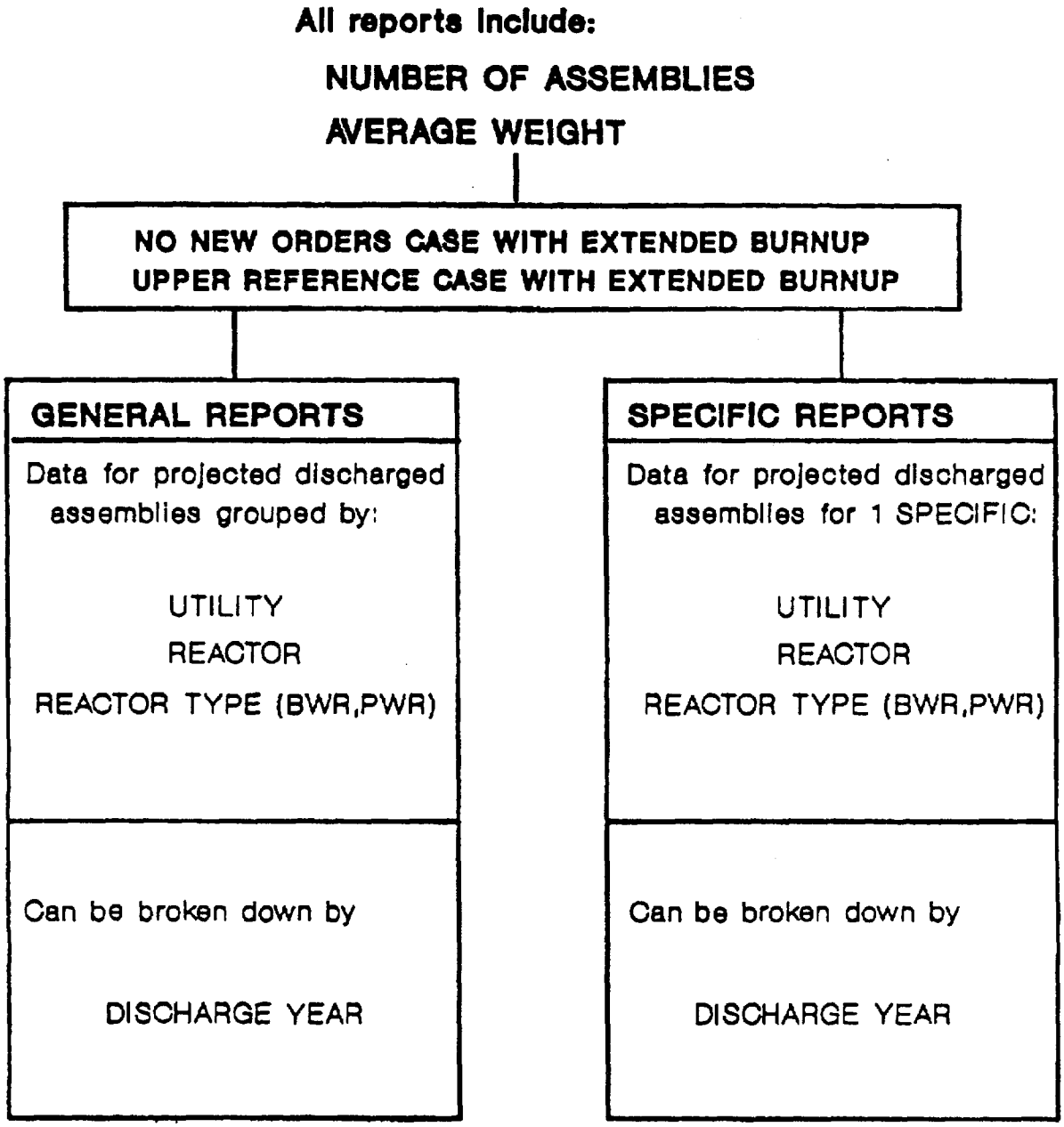

Table 2.3.1 Projected installed nuclear electric power capacity for alternative DOE/EIA scenarios[a]

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

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

[b]Actual data.

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

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

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

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

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

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

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

[c]Actual data.

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

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

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

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

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

Table 2.3.4 Sample Report from LWR Quantities Data Base

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

PAGE 8

LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

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

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LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Utility

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

Table 2.3.5 Sample Report from LWR Quantities Data Base*

PAGE 1

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

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

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

Table 2.3.6 Sample Report from LWR Quantities Data Base

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

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

Table 2.3.7 Sample Report from LWR Quantities Data Base

PAGE 1

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

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

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

PAGE 2

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

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

2.4 RADIOLOGICAL PROPERTIES OF INTACT SPENT FUEL

2.4.1 Overview

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

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

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

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

2.4.2 Major Contributors to Radioactivity

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

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

2.4.3 Major Contributors to Thermal Power

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

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

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

2.4.4 Neutron Sources

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

2.4.5 Photon Production

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

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

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

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

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

2.4.6 References for Section 2.4

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

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

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

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

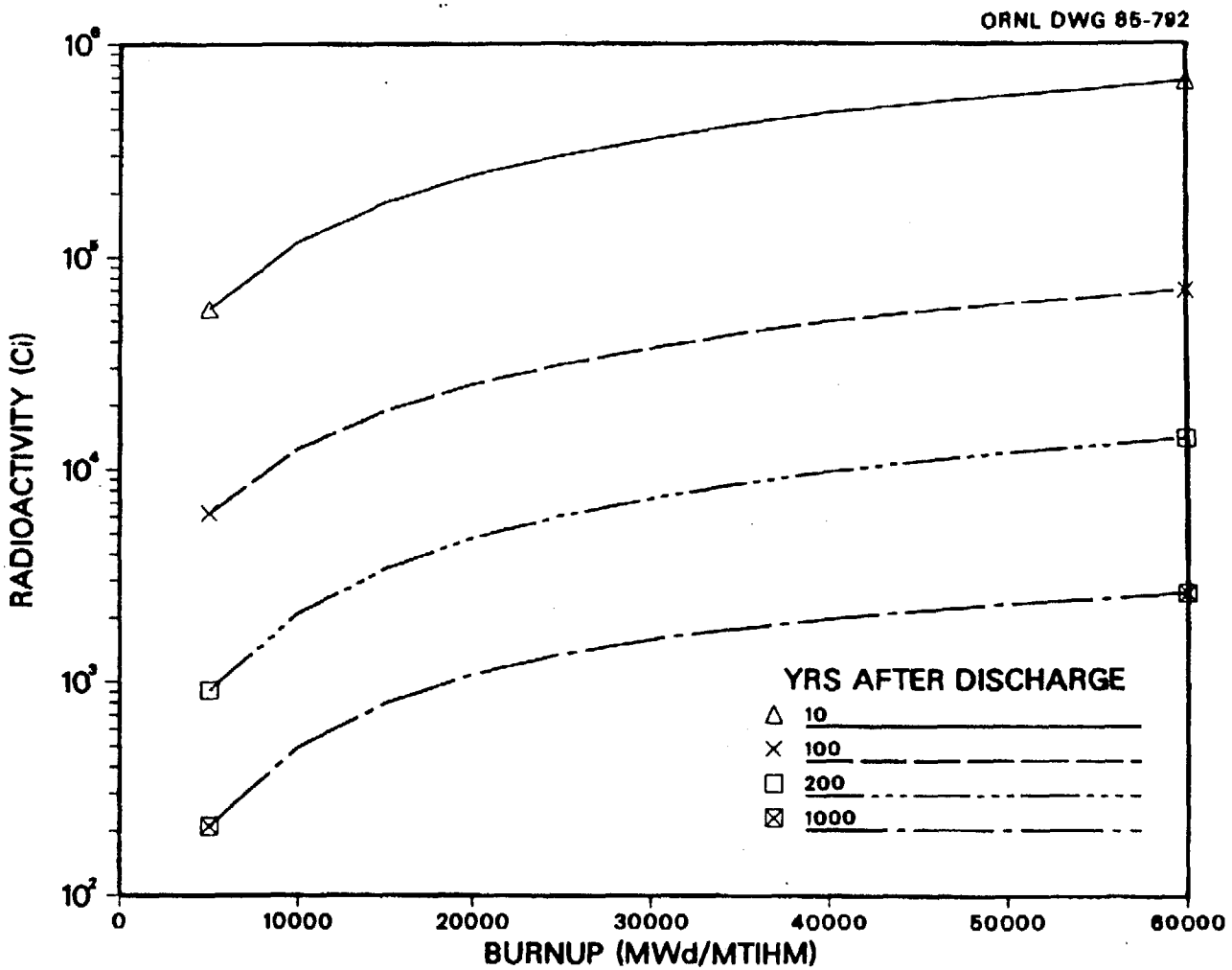

Figure 2.4.1. Radioactivity produced by 1 metric ton of initial heavy metal for a PWR (Source: Roddy 1986)

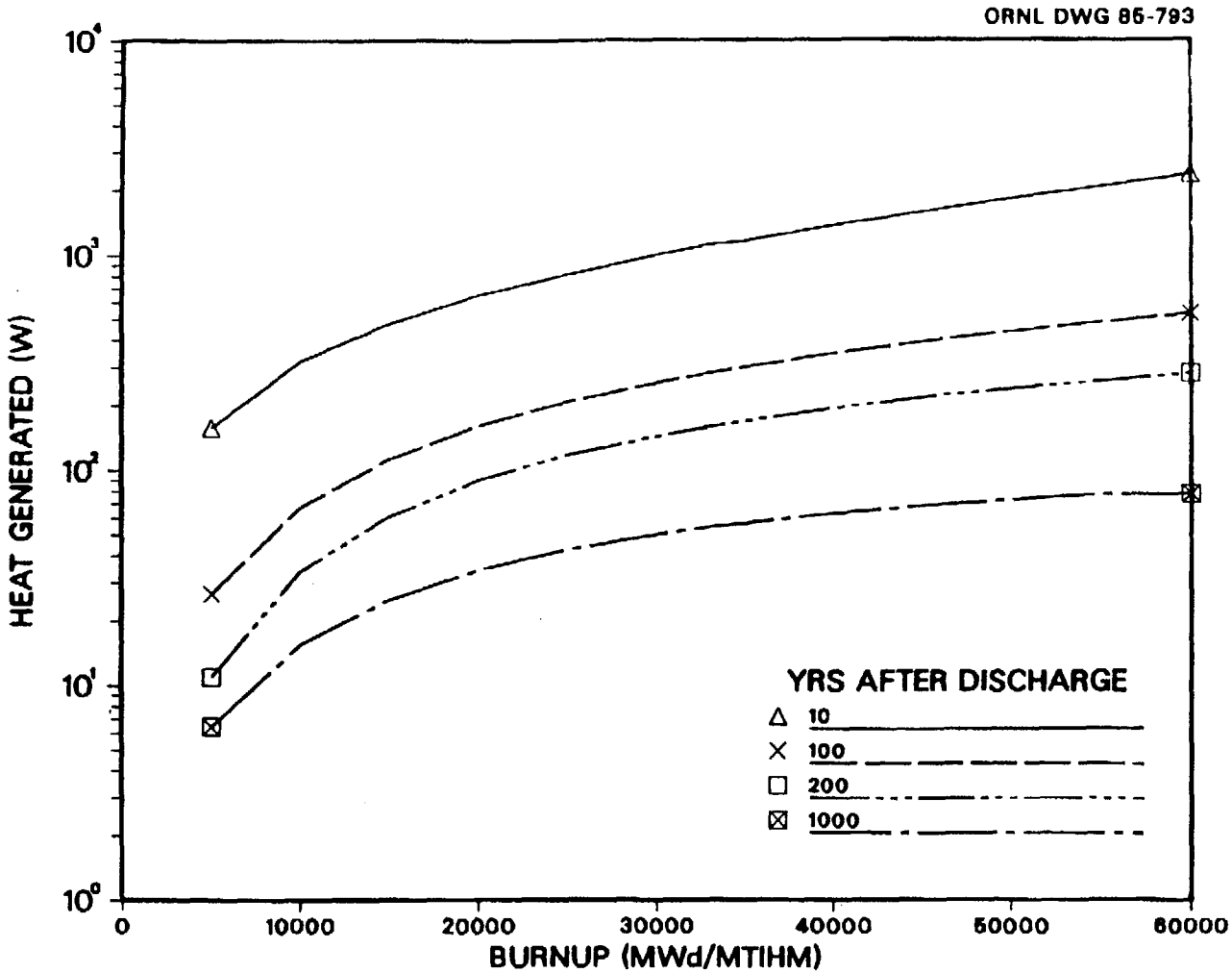

Figure 2.4.2. Heat generated by 1 metric ton of initial heavy metal for a PWR (Source: Roddy 1986)

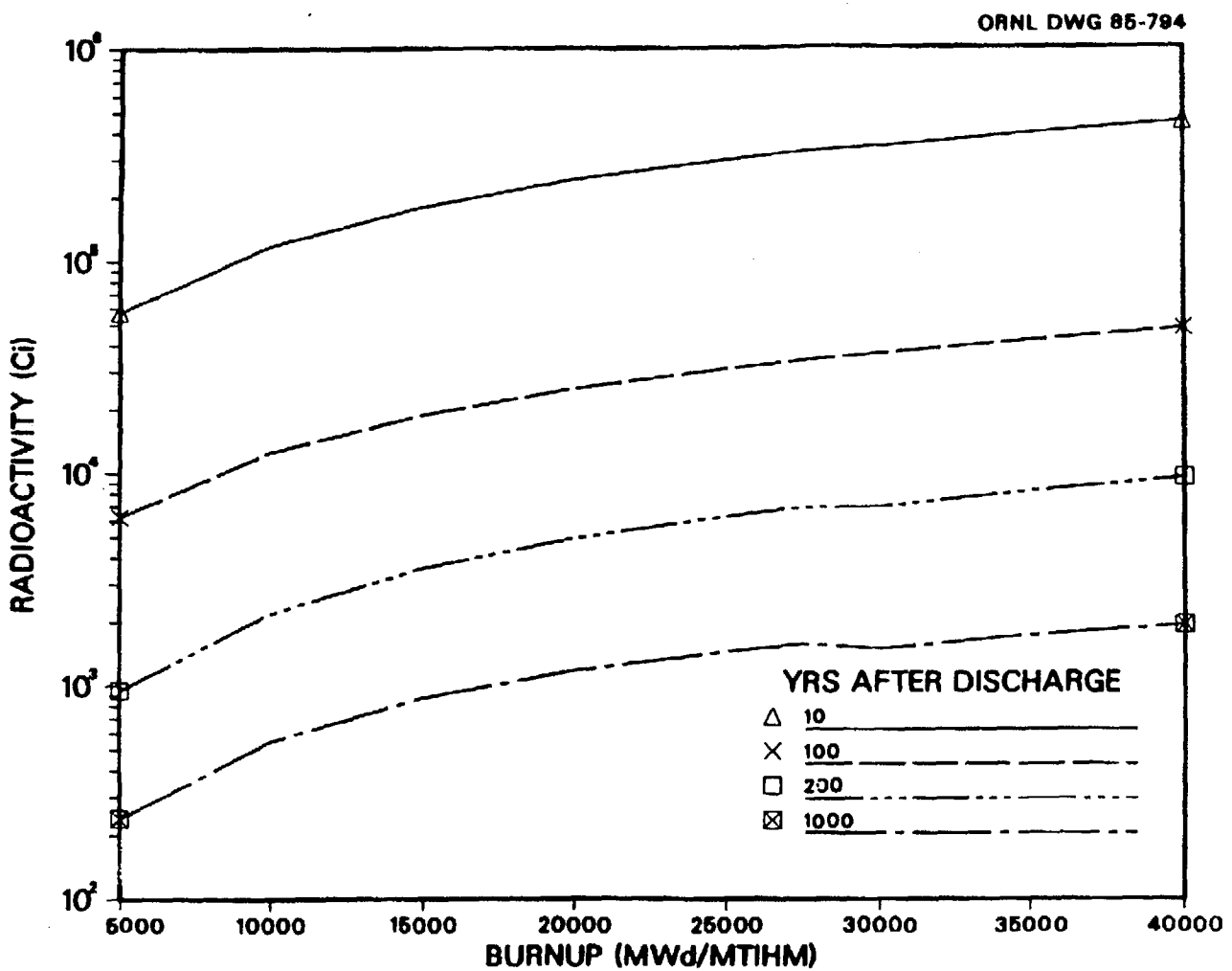

Figure 2.4.3. Radioactivity produced by 1 metric ton of initial heavy metal for a BWR (Source: Roddy 1986)

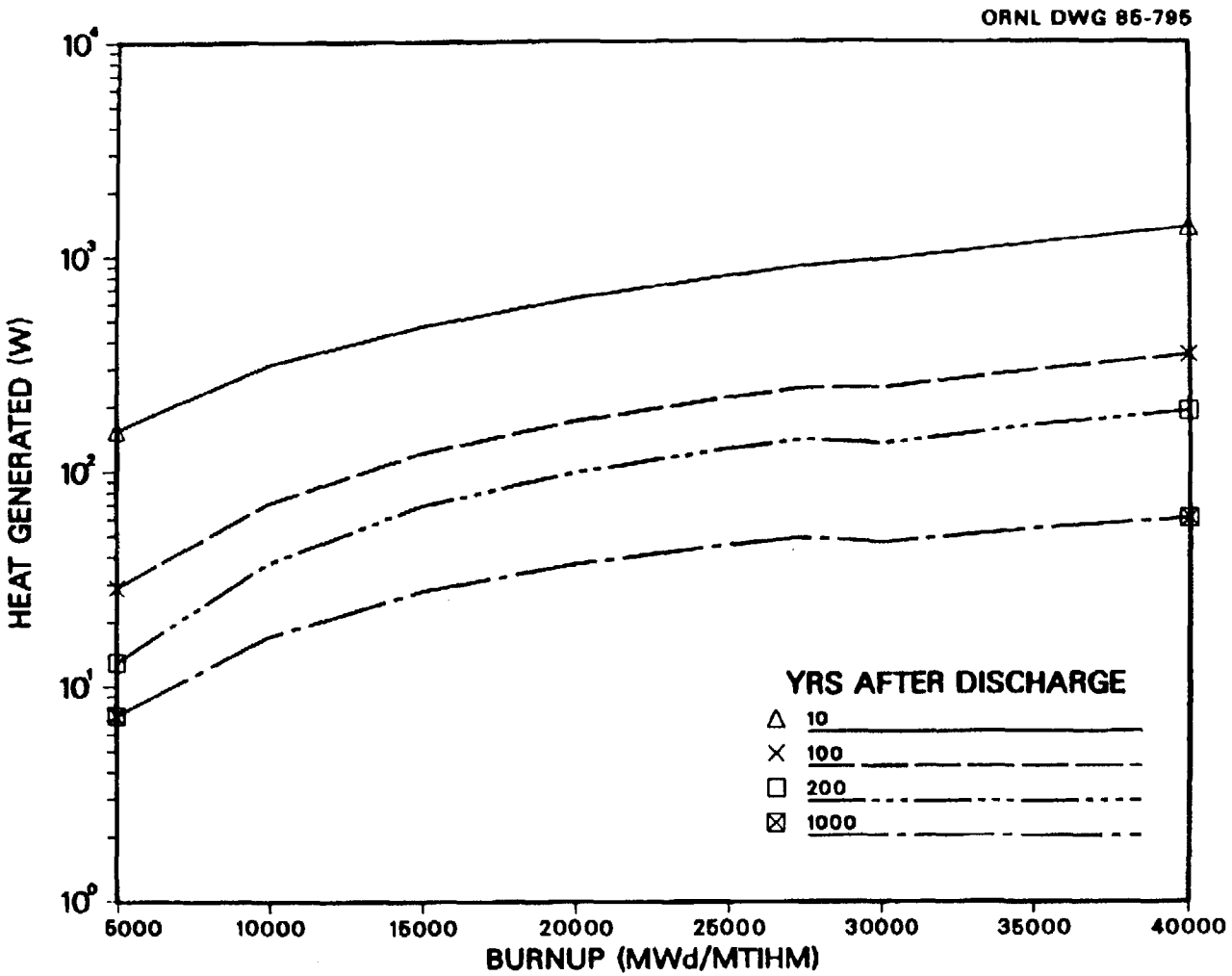

Figure 2.4.4. Heat generated by 1 metric ton of initial heavy metal for a BWR (Source: Roddy 1986)

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

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

[a]Nuclides contributing >0.1% are listed.

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

[c]Only activation products contribute to this nuclide.

[d]A.P. = Activation products.

[e]F.P. = Fission products.

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

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

[a]Nuclides contributing >0.1% are listed.

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

[c]Only activation products contribute to this nuclide.

[d]A.P. = Activation products.

[e]F.P. = Fission products.

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

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

[a]Nuclides contributing >0.1% are listed.

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

[c]Only activation products contribute to this nuclide.

[d]A.P. = Activation products.

[e]F.P. = Fission products.

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

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

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

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

[c] Only activation products contribute to this nuclide.

[d] A.P. = Activation products.

[e] F.P. = Fission products.

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

(Source: Roddy 1986)

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

[a]Nuclides contributing >0.1% are listed.

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

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

(Source: Roddy 1986)

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

[a]Nuclides contributing >0.1% are listed.

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

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

(Source: Roddy 1986)

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

[a]Nuclides contributing >0.1% are listed.

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

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

(Source: Roddy 1986)

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

[a]Nuclides contributing >0.1% are listed.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

[a]Nuclides contributing >0.1% are listed.

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

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

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

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

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

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

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

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

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

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

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

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

2.5 DEFECTIVE FUEL

2.5.1 Introduction

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

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

2.5.2 Description of Irradiated Fuel Defects

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

2.5.2.1 Defective Fuel Assemblies

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

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

2.5.2.2 Defective Fuel Rods

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

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

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

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

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

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

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

2.5.3 Methods for Detecting Defective Fuel

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

2.5.3.1 Indirect Fuel Monitoring Methods

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

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

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

2.5.3.2 Direct Fuel Inspection Methods

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

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

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

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

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

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

2.5.4 Data Sources

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

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

2.5.5 Approaches to Categorization of Defective SNF

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

2.5.5.1 Categorization in Terms of Fuel Performance

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

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

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

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

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

2.5.5.2 Categories Defined in 10CFR961

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

10CFR961 mentions three classes of failed fuel as follows:

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

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

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

2.5.5.3 Categories Defined in DOE/EIA Form RW-859

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

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

This system also has some shortcomings from the characterization viewpoint:

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

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

2.5.5.4 Reconciliation of the Categories

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

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

2.5.6 Statistical Information

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

2.5.6.1 Data from Nuclear Utilities

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

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

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

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

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

2.5.6.2 Data from Nuclear Fuel Vendors

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

o Advanced Nuclear Fuel Corporation (formerly Exxon Nuclear Company)

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

o Babcock & Wilcox (B&W) Fuels

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

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

o Combustion Engineering (CE)

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