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Preliminary Waste Form Characteristics Report Version 1.0

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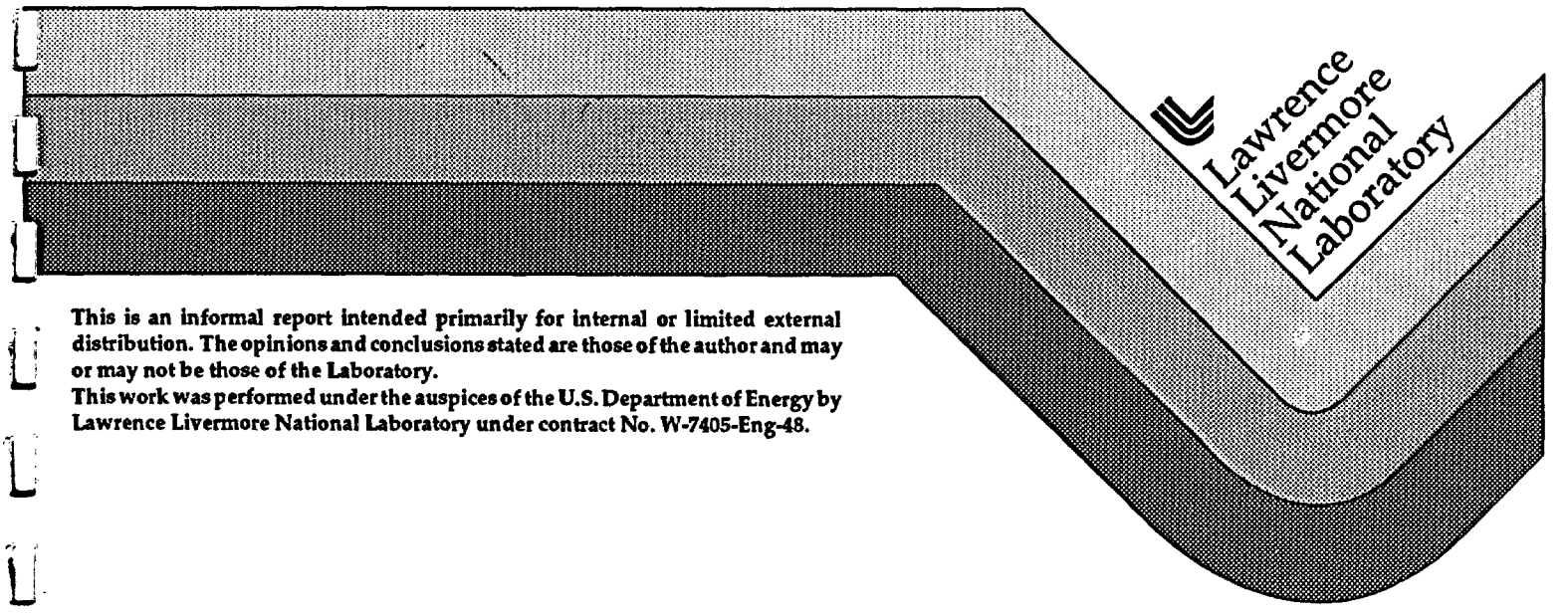

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Preliminary

Waste Form Characteristics Report

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Preface

Over the past several decades, sophisticated techniques have been developed to characterize the physical, thermal, chemical, mechanical, and radiological properties of nuclear radioactive waste form(s). (Here, "waste form" means the radioactive waste materials and any encapsulating or stabilizing matrix and is the definition provided by United States Nuclear Regulatory Commission in their regulation of Title 10 CFR 60.) Much of the early characterization was for design, operational efficiency, and safety of nuclear power plants. More recently, characterization activities have been directed at the design problem of safely emplacing radioactive waste form(s) in a suitable geological repository. The emplacement problem entails the team work of people from different technical disciplines; and the data exchange interface between the different technical personnel is of the utmost importance for an effective, efficient, and safe repository design. With this need in mind, we have assembled a preliminary data source of waste form characteristics. Most of the data was taken from the open literature. The remaining data were summarized, in a preliminary form, from early results of on-going waste form testing and model development activities. In assembling the data, we hoped to address waste form related informational needs for the wide variety of technical specialists that are part of a repository design team. We have been careful not to impose any limits or restrictions on waste form response before the repository design process because only an overall design analysis or performance assessment of the waste repository system can optimize the potential design trade-off options that satisfy requirements of a geological repository containing radioactive waste form(s).

Because this is the first version of our waste form characteristics report, we expect and will welcome comments and requests for other input from users, potential users, and others who are interested in waste form information. In this way, we hope to provide and satisfy the waste form informational needs of the different technical specialists performing the design tasks for a repository. We anticipate updating this report annually with new results from our testing and model development activities as well as responding to the additional informational needs requested by users. Some deficiencies in data form and data needs have been identified and will be addressed in future revisions.

The accumulation of data was greatly facilitated because of the cooperation, interest, and esprit de corps of the following individuals, all of which we graciously acknowledge and thank: Karl Notz, Robert Einziger, Charles Wilson, Walter Gray, Harry Smith, Steve Marschman, Andrew Luksic, George Mellinger, John Bates, Les Jardine, Son Nguyen, Homer Weed, Knud Pedersen, Gregory Gdowski, Richard Van Konynenburg, William Bourcier, Carol Bruton, Stan Prussin, Andrew Zolnay, David Stahl, Richard Morissette, and Diane Harrison-Giesler. In addition, we extend a special thanks to William O'Connell for his helpful and meaningful review; Robert Day for his relentless pursuance of numerous corrections and resolution of review comments; and finally, to Sue Garber, for the fantastic job, performed with a smile, of putting the pieces together (again and again).

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YUCCA MOUNTAIN PROJECT
PRELIMINARY
WASTE FORM CHARACTERIZATION REPORT
VERSION 1.0

1. Introduction

1.1 Overview

This report focuses on radioactive waste form characteristics that will be used to design a waste package and an engineered barrier system (EBS) for a suitable repository as part of the Yucca Mountain Project. The term waste form refers to irradiated reactor fuel, other high-level waste (HLW) in various physical forms, and other radioactive materials (other than HLW) which are received for emplacement in a geologic repository. Any encapsulating or stabilizing matrix is also referred to as a waste form.

This report is divided into three chapters..The first chapter outlines report organization, the second chapter covers properties and data which a design team would use to analyze the mechanical handling, thermal, structural, chemical, and nuclear responses of existing and future waste forms, and the third chapter provides a description of preliminary models which are useful for planning experimental testing and performance assessment activities.

The information in Chapter 1 includes a brief discussion of:

- Design goals.
- Regulatory requirements.
- Interpretation of design goals and regulatory requirements.

Disposal costs for irradiated fuel and other radioactive nuclear wastes are an integral part of energy and national security costs. To proceed with safe disposal, complete characterization information about these wastes must be uniformly and readily available to a variety of different design sub-teams so that their products will interface and integrate consistently into a total disposal system.

In order to quantify preliminary design decisions, an accumulation of waste form characteristic data is required. Chapter 2 contains waste form characteristics along with available analyses for preliminary design.

Chapter 3 expands on the physical, material, structural, chemical, and radiological analytical responses of the waste forms. This subject includes other topics, such as phases, chemical combinations, transport modes, and concentrations. At the present time models for many of the analytical responses of waste forms are still being developed.

1.2 Technical Objectives

When Congress passed the Nuclear Waste Policy Act (NWPA) of 1982 (Public Law 97-425), it began the process to establish a national repository for the permanent disposal of spent fuel and high-level waste. This Act gave the U.S. Department of Energy (DOE) the responsibility for siting, constructing, and operating a repository.

It gave the U.S. Environmental Protection Agency the responsibility for developing standards to protect the environment from offsite releases of radioactive material from a repository (40CFR191). The U.S. Nuclear Regulatory Commission (NRC) was responsible for announcing the technical requirements necessary to license all phases of repository operation (10CFR60). In 1987, Congress amended the Act to focus site characterization efforts on a site at Yucca Mountain in Nevada.

The technical objective of the waste package program is to develop a waste package and an associated EBS, and to demonstrate in an NRC licensing proceeding that the package and system meet all the regulations. NRC rule 10CFR60.113 mandates two specific performance objectives for the waste package and EBS after the repository closes and divides the period after closure into two time periods, referred to as "containment" and "controlled-release." The containment requirement applies primarily to the waste packages, and the controlled-release requirement applies primarily to the EBS:

> Containment [10CFR60.113 (a) (1) (ii) (A)]
>
> . . . the engineered barrier system shall be designed, assuming anticipated processes and events, so that: Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission taking into account the factors specified in 60.113(b) provided, that such period shall be not less than 300 years nor more than 1,000 years after the permanent closure of the repository.
>
> Controlled Release [10CFR60.113 (a) (1) (ii) (B)]
>
> . . . the engineered barrier system shall be designed, assuming anticipated processes and events, so that: . . . The release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the Commission; provided that this requirement does not apply to any radionuclide which is released at a rate of less than 0.1% of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay.

The requirements relating to postclosure performance of the total repository system [10CFR60.112] place additional requirements on the design and performance of the waste package and EBS as follows:

> The geologic setting and the engineered barrier system and the shafts, boreholes and their seals shall be designed to assure that releases of radioactive materials to the accessible environment following permanent

closure conform to such generally applicable standards for radioactivity as may have been established by the Environmental Protection Agency with respect to both anticipated processes and events and unanticipated processes and events.

A fourth major objective is to perform a "comparative evaluation of alternatives to the major design features that are important to waste isolation, with particular attention to the alternatives that would provide longer radionuclide containment and isolation" [10CFR60.21 (c) (1) (ii) (D)].

A number of other requirements apply to the waste package and EBS before the permanent closure of the repository. These requirements include radiological protection [10CFR 60.111 (a)], retrievability [10CFR60-111 (b)], and geologic repository operations area design criteria [10CFR60.131].

Finally, 10CFR60.135 sets forth specific design criteria for the waste package and its components that must be met. These criteria include constraints on the general performance of the package, its chemical reactivity, and provisions for its handling and labeling, as well as design criteria for the waste forms.

1.3 Quality Objectives

All information for the final design, design analysis, testing, and performance assessment of the waste package and EBS that will form a basis of the license application will be acquired or developed under an NQA-1 quality assurance program based on the criteria of Appendix B of 10CFR50. All participants in the project have developed or adopted quality assurance program plans (QAPPs) that reflect all requirements of the Project Office Quality Assurance Plan, which incorporates the provisions of the Office of Civilian Radioactive Waste Management (OCRWM) Quality Assurance Requirements (QARs). For waste package and EBS work, a system of quality procedures (QPs) are used to implement the QAPP. A software quality assurance plan (SQAP), which specifically addresses implementing the QAPP requirements to computer software, supplements the QAPP and QPs.

QPs establish methods to control scientific investigations, testing activities, design activities, and performance assessments that are described in the technical planning sections of the *OCRWM Yucca Mountain Project Waste Package Plan*. For example, the QPs describe how scientific investigations and design analyses are planned, controlled, and documented. They also describe which documents are quality assurance records and how these records are created, maintained, and stored. They also cover how documents are reviewed and how the document content is verified.

In the case of the present documentation of preliminary waste form characteristics, the data and analytic response models provided were taken primarily from the open literature. These data and models are considered as best available and

their quality and suitability cannot be assured for waste disposal design purposes except as noted in this report.

1.4 Types of Waste Forms

Waste forms in this report are divided into three categories: spent fuel, glass, and other waste forms. This division is selected because each category may have different design constraints and may require distinct solutions for the EBS.

1.5 Spent Fuel Waste Forms

In this report, spent fuel is understood to mean elements from the entire inventory of existing and future fuel assemblies from nuclear reactors. When spent fuel properties were compiled, it was assumed that issues central to waste package design must be considered, beginning when fuel is discharged from the reactor and ending after the "controlled release" time period.

Spent reactor fuels originate mainly from civilian nuclear power reactors. The vast majority of these fuels are from light water reactors (LWRs), which are either boiling water reactors (BWRs) or pressurized water reactors (PWRs).

Many varied issues and operations are involved in waste package design, and the importance of different spent fuel properties depends greatly on the particular aspect of design under consideration. Some properties, such as physical dimensions and masses, are well defined. The distribution of such properties must be considered in any arbitrary design intended to transport, handle, and to maintain containment of intact fuel assemblies or a combination of consolidated fuel pins and segregated assembly hardware. Given the relative range of assembly and fuel designs that are accommodated in existing power reactors, we must extrapolate properties with reasonable confidence from the present inventory to the total expected inventory that will be contained in the repository.

Most, if not all, of the information that must be available for the containment and EBS designs for the spent reactor fuels must be available for the other categories of waste forms. Therefore, we will discuss the information once and will present the available data for each waste form.

1.6 Physical Inventory

The spent fuel properties needed are included schematically in Figure 1 and are given in detail in Table 1. We assume that the entire inventory is comprised solely of light water reactor (LWR), boiling water reactor (BWR) and pressurized water reactor (PWR), fuel assemblies. We also assume that the characteristics of the assemblies are available at their arrival at the repository. After arrival, it is likely that the assemblies will be placed into temporary storage and, possibly, subjected to tests to verify external physical dimensions, to determine their contribution to reactivity of an array, their thermal power, etc. If a design accommodates intact fuel

assemblies, they can then be transferred directly to a container. If a design accommodates consolidated fuel pins and associated assembly hardware, additional handling is needed, and additional measurements may be required. For this purpose, the term container design is broadly interpreted to include the actual design of the container, its cover, means for handling individual spent fuel assemblies, and, if necessary, means for removal of fuel pins from the assemblies.

Properties of assemblies in the present pool of spent fuel are generally contained in the *Integrated Data Base for 1990*, DOE/RW-0006, Rev. 6, and in the Characteristics Data Base of ORNL. These include masses and physical dimensions, materials of construction, physical characteristics of fuel pins, etc. Assembly drawings of varying degree of complexity are also given in the latter. The data sets appear to be adequate for the majority of PWR fuel assemblies but are very sparse for BWR assemblies. Many of the details that are relevant to handling and disassembly of fuel manufactured by the General Electric Company are presently unavailable. (Characteristics of these are only inferred by reference to those of other manufacturers.) Some additional effort may be needed to ensure sufficiency and proprietary aspects of information to meet the needs of handling and the preliminary stages of design of containers and other facilities.

Table 1. Spent fuel physical characteristics.

	Number and physical dimensions	Handling hardware	Assembly drawings	Special properties	As-fabricated fuel characteristics	As-irradiated fuel characteristics
Fuel assemblies	Total width, length, and mass	Design of end plates for locating assembly and handling	Dimensions of end plates, spacers, and other hardware; fastener characteristics	Failed pins, control rods, etc.		
Fuel pins	Total and active lengths, location of active length, O.D., total masses	Design of end plugs for locating pin and handling		Failed pins, control rods, etc.	Total fuel mass, enrichment, pressurization, dimensions of fuel pellets	Estimates of changes in pressurization and physical dimensions from as-fabricated conditions

For waste forms such as glass, the physical inventory can be described simply in terms of container dimensions and mass, and the total amount of each waste form. Some forms, e.g., spent fuel hardware, may be difficult to characterize. However, these forms will probably be consolidated before being placed in the repository, and it may be necessary to do research to determine the compacted density of these materials. For these forms, the most important factors are the waste form mass density and the physical description.

1.7 Radionuclides

The inventory of radionuclides in any waste form is important for several reasons. It determines the amount of heat generated per unit mass of the waste form, and it determines the background radiation created by the waste form, in terms of intensities, energies, and kinds of radiation.

The potential release of radionuclides from a spent fuel element into the immediate surroundings of the element depends on a large number of factors, many of which are interrelated. The irradiation history of the fuel, measured by the burnup, together with the initial, as-built inventory of elements in the fuel, determine what the inventory will be when the fuel is discharged from the reactor and any time thereafter. The same factors also can be useful in estimating the amount of physical damage that has been found in the fuel pellets and in the fuel cladding. The operating temperatures and the chemistry of the reactor coolant water are also dependent factors in this estimating process.

A spent fuel assembly may have failed fuel pins. Presently, a detailed characterization of failed pins has not been completed. In any case, the dissolution and transport of radionuclides out of the fuel pin requires a breach in the fuel cladding.

For spent fuels, the mixture and amount of radionuclides within a pellet enclosed inside the cladding depends primarily on assembly burnup, which translates into the number of fission events per cubic centimeter of the pellet. At the pellet dimensional scale, there is some slight variation in the number of fission events in the radial direction across a pellet. This is termed the pellet rim effect, but it is not presently considered a significant variation for spent fuels with burnups less than 60 giga-watt days per metric tons of uranium (Gwd/MTU). At the assembly dimensional scale, the burnup rate can vary spatially in both the radial and axial directions within the core volume of an operating nuclear reactor; thus, there are burnup variations across an assembly and axially along an assembly. For preliminary/conceptual design purposes, the variations within the set of fuel rods of an assembly are not considered significant. The concentrations of gaseous nuclides and the potentially volatile nuclides within a fuel pellet depend primarily on the fission gas released (FGR) during reaction operation. At present little detailed information is available on the FGR spent fuel attribute.

For the high-level waste (HLW) forms, which are a mixture of by-products from spent fuel reprocessing plants, the radionuclide content will be measured by nuclide and radioactivity per unit volume. For the most part, the HLW have three phases, liquid, sludge, and salt cake. A stream of these three phases will be mixed and incorporated into a glass waste form. It is during the production of the glass waste form that the radionuclide content will be measured and recorded in the data base.

Fortunately, the radionuclide inventory is readily calculable or measurable. Computer codes are available which calculate the fuel inventory, and the industry generally uses these codes for fuel management. We must assume that measurements will be used to determine radionuclide inventories in other waste forms.

1.8 Decay Heat and Criticality

The nuclide inventory, types of isotopes, and their amounts determine how the waste may be stored in the individual waste package and in the repository as a whole. The properties we must know are shown in Fig. 1 and are also listed in Table 2.

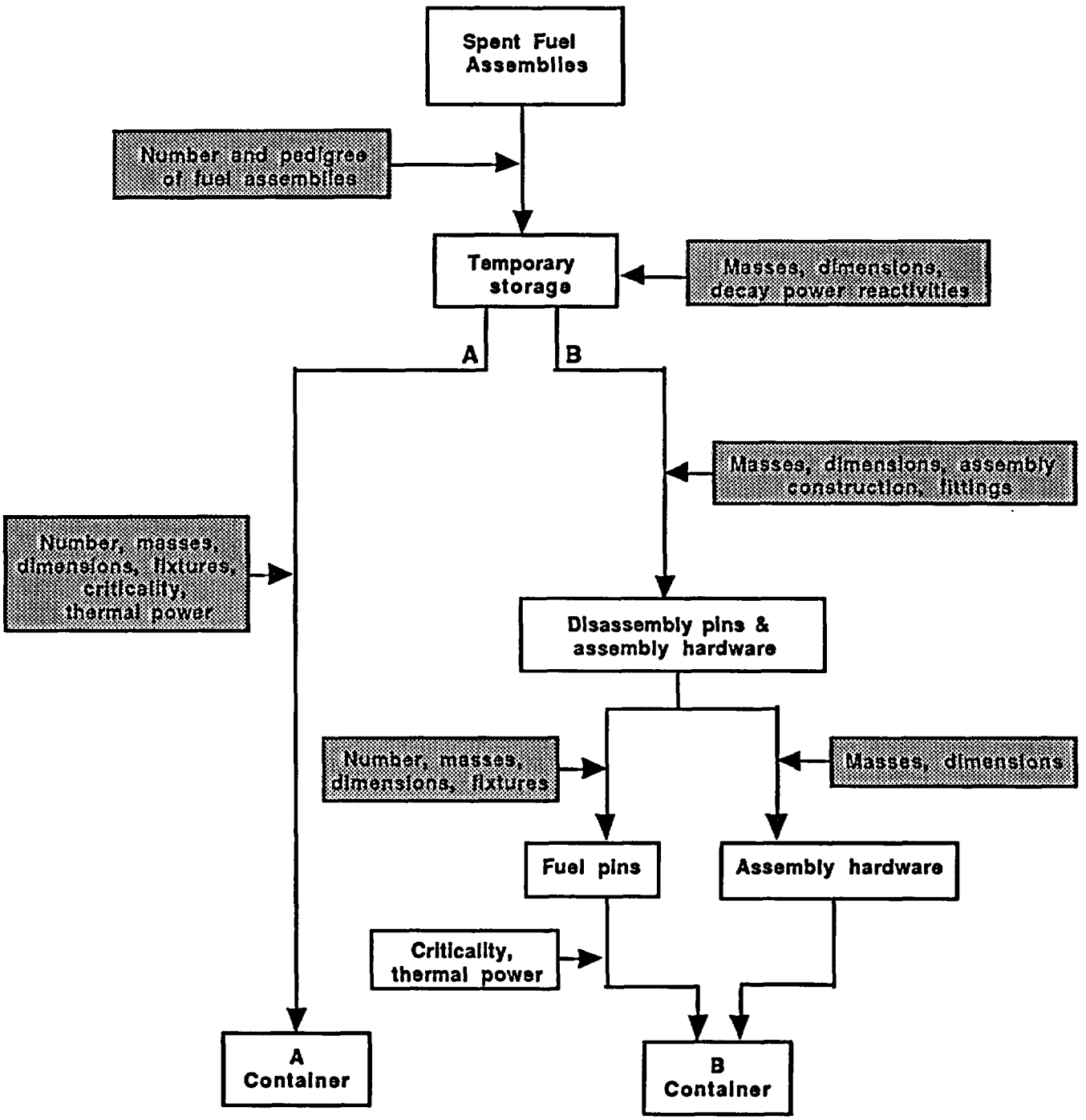

Figure 1. The effect of spent fuel properties on container design; mechanical, handling, thermal, criticality, and shielding.

Table 2. Spent fuel characteristics associated with thermal, criticality, and shielding subtask.

	History	Composition and thermal properties	Physical dimensions	Detailed geometry	As-fabricated fuel composition
Assembly hardware					
Thermal effects	Irradiation and post-irradiation storage and conditions	Construction materials and their thermal conductivities after irradiation	Total width, length, and mass	Distribution of materials in space, detailed geometry of fuel pin placement	
Criticality/ Shielding effects	Irradiation and post-irradiation storage and conditions; burnup	Construction materials and their neutron interaction characteristics	Total width, length, and mass	Distribution of materials in space, detailed geometry of fuel pin placement	
Fuel pins					
Thermal effects	Irradiation (burnup) and post-irradiation storage and conditions	Cladding material and irradiated fuel and their thermal conductivities	Total length, outer diameter, cladding thickness, fuel mass	Distribution of materials in space, including active dimensions of fuel pins	
Criticality/ Shielding effects	Irradiation (burnup) and post-irradiations storage and conditions	Construction materials and their neutron interaction characteristics	Total length, outer diameter, cladding thickness, fuel mass	Distribution of materials in space, including active dimensions of fuel pins	Enrichment and components added or deposited onto the fuel for reactivity control during reactor operation

It is assumed that guidelines or standards are established which define, for any EBS design, temperature limits and their spatial distributions, and the maximum effective neutron multiplication factor that can be achieved by any given container loading or geometry. It is also assumed that an approved methodology exists for determination of the isotopic and elemental composition of the spent fuel and the types and spectra of ionizing radiation emitted by fuel pins and assembly hardware as a function time and irradiation history. (The latter define the source terms for thermal and shielding calculations and the content of fissile and fertile material needed for criticality calculations.)

Reactivity and thermal properties of the assemblies and fuel pins are needed both for meeting temperature and reactivity limitations during temporary and long-term storage. These parameters essentially define the number and arrangement of fuel assemblies or consolidated fuel pins plus separated assembly hardware that can be accommodated in any single container and meet design requirements for the repository as a whole.

1.9 Radiation Field

The radiation field surrounding a given amount of any waste form is a determining factor in how the waste form and its container must be handled. The amount of shielding necessary during handling, storage, emplacement, and disposal is based on the radiation field and repository operational and performance requirements.

The important spent fuel properties here are all those that will ultimately impact the rates and quantities of radionuclides which can be released and transported from failed fuel pins to the container and beyond. These properties also cover external radioactive deposits which may be important during the handling of the spent fuel prior to loading into the waste containers. These properties are shown schematically in Figure 2 and listed in detail under subtopic Spent Fuel Characteristics Associated with Radioactivity Release and Radiation Field. We have assumed that an approved methodology exists to obtain the detailed elemental and isotopic composition of the fuel and cladding, as well as the spectrum and types of radiations emitted from fuel pins and assembly materials.

We have already discussed the importance of all of the properties of irradiated fuel and its cladding. However, external crud deposits on the assembly hardware and cladding are worthy of a few comments. Crud deposits typically contain radioactivity (e.g., ^{60}Co from neutron activation) with short half-lives compared to the time periods of interest in waste disposal. However, the length of these half-lives must be considered during handling. Their physical and chemical properties and the extent and content of radioactivities will be considered in design of the intermediate storage facilities, for assessing the extent of contamination possible throughout handling procedures, etc.

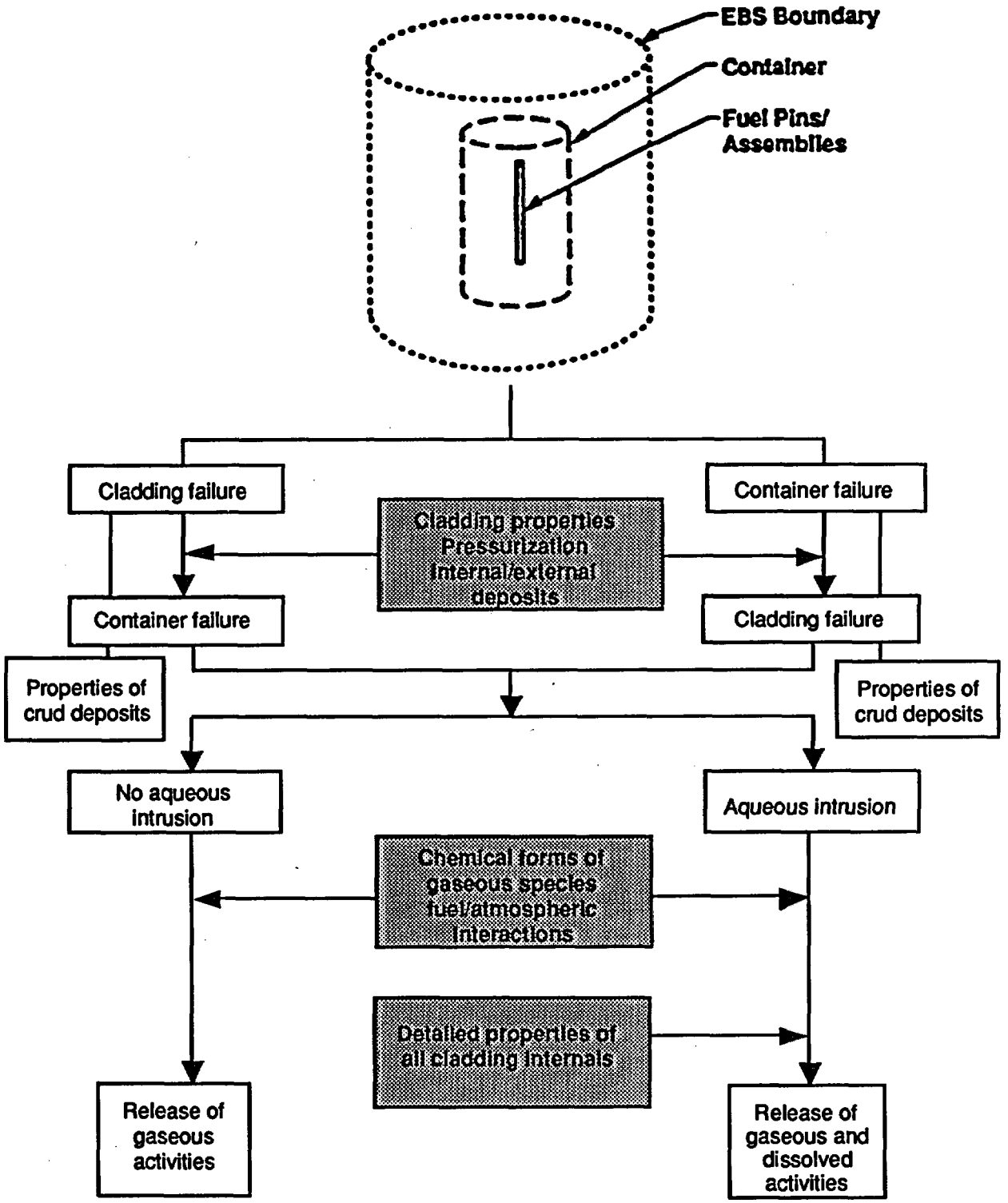

Figure 2. Radioactivity Release

The majority of spent fuel properties necessary for characterization of the radiation field are provided through the studies underway at the Materials Characterization Center (MCC) of the Pacific Northwest Laboratories (PNL). The principal information deficiencies are:

- Modern, high-burnup fuel from PWRs and BWRs that span the range of spent fuel properties likely to be found in the inventory of a repository are not yet included within the Approved Testing Materials (ATM) presently analyzed or on hand for measurement.

- Knowledge of the distribution of fission gas release to a reasonable degree of accuracy is not now available nor is it clear that a means has been provided to assess accurately the nature of this distribution for the larger fission gas releases. An activity is underway at MCC to elicit information from fuel vendors. However, it is not certain that this information will lead to sufficient confidence in the distribution function in the high burnup region to mitigate the need for further experimental measurements.

Spent Fuel Characteristics Associated with Radioactivity Release and Radiation Field

Fuel assemblies have the following properties:

- As-fabricated properties
  - Materials composition: materials of construction, masses, and chemical compositions, including major and minor constituents.
- As-irradiated properties
  - History: irradiation and post-irradiation storage.

Fuel pins have the following properties:

- As-fabricated properties
  - Physical dimensions and masses: total length, outer diameter, cladding thickness, as-fabricated fuel-pellet dimensions, plenum dimensions or total void volume of fuel pin, and fuel and cladding masses.
  - Actual compositions of fuel and cladding: includes cladding type and any liners added to enhance cladding performance during in-core service, fuel enrichment, and the composition and location of any components added or deposited onto the fuel for the purposes of reactivity control during in-core operation.
  - Fill gas composition and pressure.
  - Fuel morphology: average grain size, porosity.
- As-irradiated properties
  - History: irradiation (burnup) and post-irradiation storage.
    - — Identification of failed fuel pins as delivered.
    - — Cladding composition and morphology: including external crud deposits, thickness of external and internal oxide layers, hydride content, deposits of fission products and other fuel components.

— Composition and pressures of gases, including fission gases and helium.
— Fuel morphology and composition, including surface deposits; extent, properties, and composition of periphery or rim region; grain sizes and their radial distributions; fuel phases and their radial distributions; characteristics of grain boundaries, including nature and extent of materials segregated along grain boundaries; fuel-fragment size distributions; and estimates of total surface area per unit fuel mass.

1.10 Hardware

In discussing waste forms, the term hardware refers to the material contained in a fuel assembly, with the exception of the fuel pins. The amount of hardware and the specific components differ for PWRs and BWRs, and these differences extend even within a fuel class. Generally, the hardware will remain part of the fuel assembly, unless both are consolidated. When hardware is part of the assembly, the radioactivity of the hardware is small, compared with the activity of the fuel within the assembly. However, relatively short-lived hot spots of 60Co may be found in some assemblies. Thus, when the hardware is part of the assembly, the hardware reduces the specific emissions (number of emissions per unit mass or volume) of the fuel assembly.

If the fuel is disassembled or if large amounts of fuel assemblies are consolidated, we must know more specific details about the hardware: physical properties of the hardware, its radiological characteristics, and whether or not it is greater than Class C (GTCC).

1.11 Modeling

Models are necessary to predict future thermal, structural, chemical, and nuclear responses of waste forms placed in the expected environment of a suitable repository when it is not practical to experimentally measure such responses. Several methods are employed in modeling. Physical models can be built to a certain scale, or even built with components on a scale which is distorted from the rest of the model. Models can be based on knowledge of mathematical relationships governing the factors being studied. In some cases a computer model may combine some of the same procedures used on the physical models together with numerical processing. In other cases it may be possible to use an analog model, i.e., model the phenomenon being studied by using a different phenomenon which obeys the same mathematical laws.

In all cases we must know what laws or relationships govern the interactions of the physical variables and functions being studied. Thus, to do modeling we must first know what factors are involved and how they interact. We must plan and carry out experiments to gather sufficient data to calculate or deduce relationships. With

sufficient data, we can construct and run models. The models must be tested and validated, and only then can we use them to make predictions.

For spent fuel waste forms, we must model rates for cladding failure, oxidation, and dissolution of many materials. At this time we are planning or executing experiments. From the experiments that have been run, some relationships have been deduced, and the data has been presented in tabular, graphic, or empirical form.

1.12 Burnup Models

Models exist which predict radionuclide concentrations. These models can be used to calculate other properties which are directly dependent on the concentrations. However, more specific attributes, such as fission gas release, grain size, and pellet fragmentation are changes in fuel characteristics, which depend on the burnup and the thermal history of the fuel.

Although dependent on burnup and thermal history, fission gas release has been named as a criterion in selecting spent fuel ATMs. Investigations are underway to determine distributions of burnup and fission gas release in the present and future LWR spent fuel inventory.

No models exist to predict the total effect of exposure in a reactor core on fuel or other materials.

1.13 Glass Modeling Status

Models are being developed to predict the behavior of the glass waste forms in a Yucca Mountain repository during the period of regulatory concern. Information from these models will be used in performance assessments to calculate the release of radionuclides from breached glass waste containers over time. These assessments are required to demonstrate compliance with the containment and controlled release requirements of 10CFR60.113, and to find the fractional contribution of the glass waste form in the cumulative release limits of 40CFR191.13.

2. Design Data for Waste Form

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

- 2.1.1.1 Present Inventory
- 2.1.1.2 Projected Inventory
- 2.1.1.3 Radionuclide Activity vs. History
- 2.1.1.4 Decay Heat vs. Time
- 2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

- 2.1.2.1 Fuel Assemblies
- 2.1.2.2 PWR Fuel
- 2.1.2.3 BWR Fuel
- 2.1.2.4 Non-Zircaloy Clad Fuel
- 2.1.2.5 Hardware

2.1.3 Repository Response

- 2.1.3.1 Cladding Degradation
- 2.1.3.2 UO_2 Oxidation in Fuel
- 2.1.3.3 Gaseous Radionuclide Release from Cladding
- 2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel
- 2.1.3.5 Dissolution Radionuclide Release from UO_2
- 2.1.3.6 Soluble-Precipitated/Colloidal Species
- 2.1.3.7 Radionuclide Release from Hardware

2

Design Data for Waste Forms

The purpose of this chapter is to collect the data which are presently available from several sources on all the types of radioactive wastes which must be disposed of in accordance with 10 CFR 60.113.

The data are presented so that they are as much as possible directly usable as design criteria and design constraints for the containment and EBR design tasks.

The information as presented has been taken directly from the references so as to prevent introduction of errors. If further information on a given subject is necessary, it may be found in the appropriate reference.

By arranging the data in this manner, we have made it easier to update the document as new data become available.

2.1 Spent Fuel Waste Form

The Spent Fuel referred to in this section consists of irradiated fuel discharged from a Light Water Moderated nuclear reactor (LWR). All such spent fuels are assumed to be permanently discharged and eligible for repository disposal.

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  - 2.1.3.6 Soluble-Precipitated/Colloidal Species
  - 2.1.3.7 Radionuclide Release from Hardware

2.1 Spent Fuel Waste Form

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2.1.1.3 Radionuclide Activity vs. History

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2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

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2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

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2.1.3.2 UO_2 Oxidation in Fuel

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2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

2.1.1 Radionuclide Content

Knowledge of radionuclide content of the spent fuel is important to all aspects of the design of nuclear waste repositories as well as in the performance appraisal of the finished system design.

The radionuclide content is determined by the initial fuel composition, the fuel's irradiation history measured by the burnup, and the time the spent fuel has been stored out of the reactor core whether in wet or in dry storage.

The heat generated in the spent fuel, usually given as the linear heat generation rate, is a direct function of the radionuclide content. From the linear heat generation rate for the spent fuel assemblies we can calculate the total heat generation in a disposal container. Fuel assemblies or fuel elements can thus be selected for individual containers to give a desired heat distribution within the repository.

The radionuclide content also determines the radiation spectrum and the intensity which emanates from an assembly. This determines the radiation field which exists around any given container at any given time. From this we can determine the amount of shielding necessary during handling, transportation and interment.

There is a relationship between radionuclide content and fission gas release, in the sense that the amount of gas released is a function of both the burnup and of the centerline temperatures which existed in the fuel during its life in the reactor.

Fission gas release into the space between the cladding and the fuel is of importance to the designers because it may influence the failure rate of the cladding in the repository. A knowledge of the release makes possible the calculation of pressure which, combined with the temperature of the elements in the repository must be analyzed together with the properties of the materials as they are at a given time in the repository.

The fragment size distribution and grain size distributions in spent fuel as a function of burnup and other significant parameters influence fission gas release, and potentially, dissolution behavior. A more detailed knowledge of these parameters is needed.

The fission product inventory is also used to model the radionuclide transport which may take place through various modes out of the container and through the EBS in the event that the cladding and container should both fail.

Because the radionuclide inventory in the spent fuel decreases as a function of time, prediction of release rates becomes a very complicated function of a large number of variables.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 3.1. Historical quantities of spent fuel by Assembly Class (Reproduced from the LWR Quantities Data Base).

LWR QUANTITIES DATABASE
Historical Data
Discharged Assemblies by Assembly Class

ASSEMBLY CLASS	FUEL ASSEMBLIES	FUEL RODS	DEFECTIVE ASSEMBLIES	AVERAGE BURNUP (MWd/MT)	TOTAL WEIGHT (MT)	AVERAGE INITIAL ENRICH.
B&W 15 X 15	3,564	740K	67	28,004	1654.8	2.815
CE 14 X 14	3,329	551K	6	29591	1271.5	2.865
CE 16 X 16	1,231	238K	23	24884	512.5	2.554
CE 16 X 16 SYSTEM 80	188	41K	0	17699	78.8	2.137
GE BWR/2,3	14,809	827K	1478	21493	2762.1	2.384
GE BWR/4-6	20,470	1,194K	949	21233	3795.0	2.307
WE 14 X 14	2,949	520K	80	32309	1146.1	3.150
WE 15 X 15	5,557	1,133K	132	30127	2507.2	2.926
WE 17 X 17	5,873	1,552K	100	27835	2670.1	2.833
Big Rock Point	315	29K	52	19339	41.6	3.490
Dresden-1	891	32K	159	16227	90.8	2.166
Ft. Calhoun	426	73K	0	30549	154.0	2.912
Haddam Neck	734	150K	43	31320	303.2	3.819
Humboldt Bay	390	15K	1	14936	28.9	2.351
Indian Point	160	28K	0	16715	30.6	4.111
Lacrosse	333	33K	104	14708	38.0	3.727
Palisades	597	126K	21	22720	239.3	2.640
St. Lucie-2	236	84K	0	23626	88.9	2.347
San Onofre-1	468	53K	7	29029	171.4	3.792
Yankee Rowe	417	102K	0	28285	100.6	3.949
— GRAND TOTALS	62,749	7,521K	3222	25950	17606.6	2.718

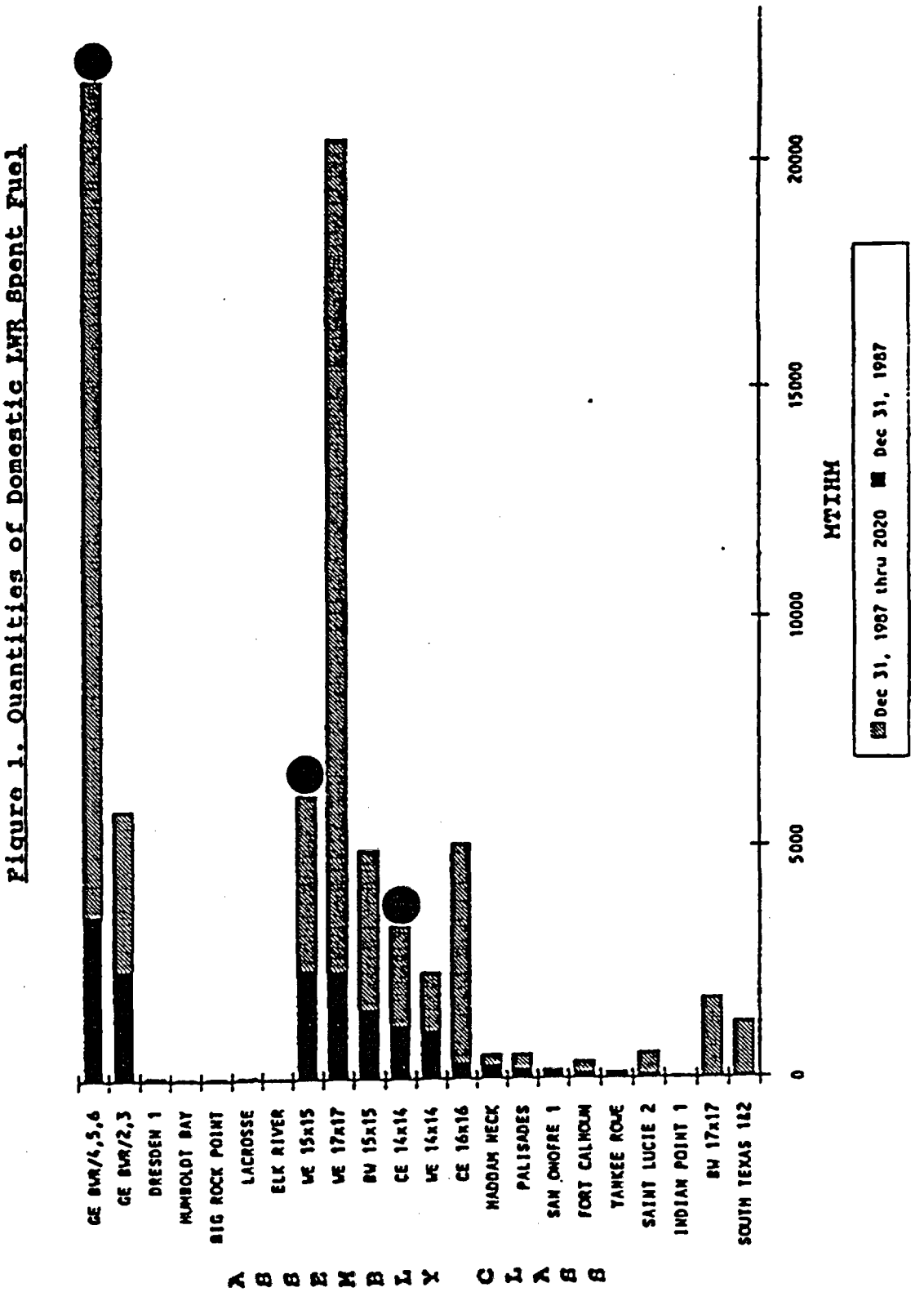
K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Table 2.2. Quantities of Domestic LWR Spent Fuel[a]

BWR Assembly Class	Historical Quantities as of Dec. 31, 1988 (MTIHM)
GE BWR/4,5,6	3795
GE BWR/2,3	2762
Dresden 1	91
Humboldt Bay	29
Big Rock Point	42
Lacrosse	38
Elk River (reprocessed)	5
PWR Assembly Class	**Historical Quantities as of Dec. 31, 1988 (MTIHM)**
WE 15x15	2507
WE 17x17	2670
BW 15x15	1655
CE 14x14	1272
WE 14x14	1146
CE 16x16	512
CE 16x16 System 80	79
South Texas	0
Haddam Neck	303
Palisades	239
San Onofre 1	171
Fort Calhoun	154
Yankee Rowe	101
Saint Lucie 2	89
Indian Point 1	31
BW 17x17	0

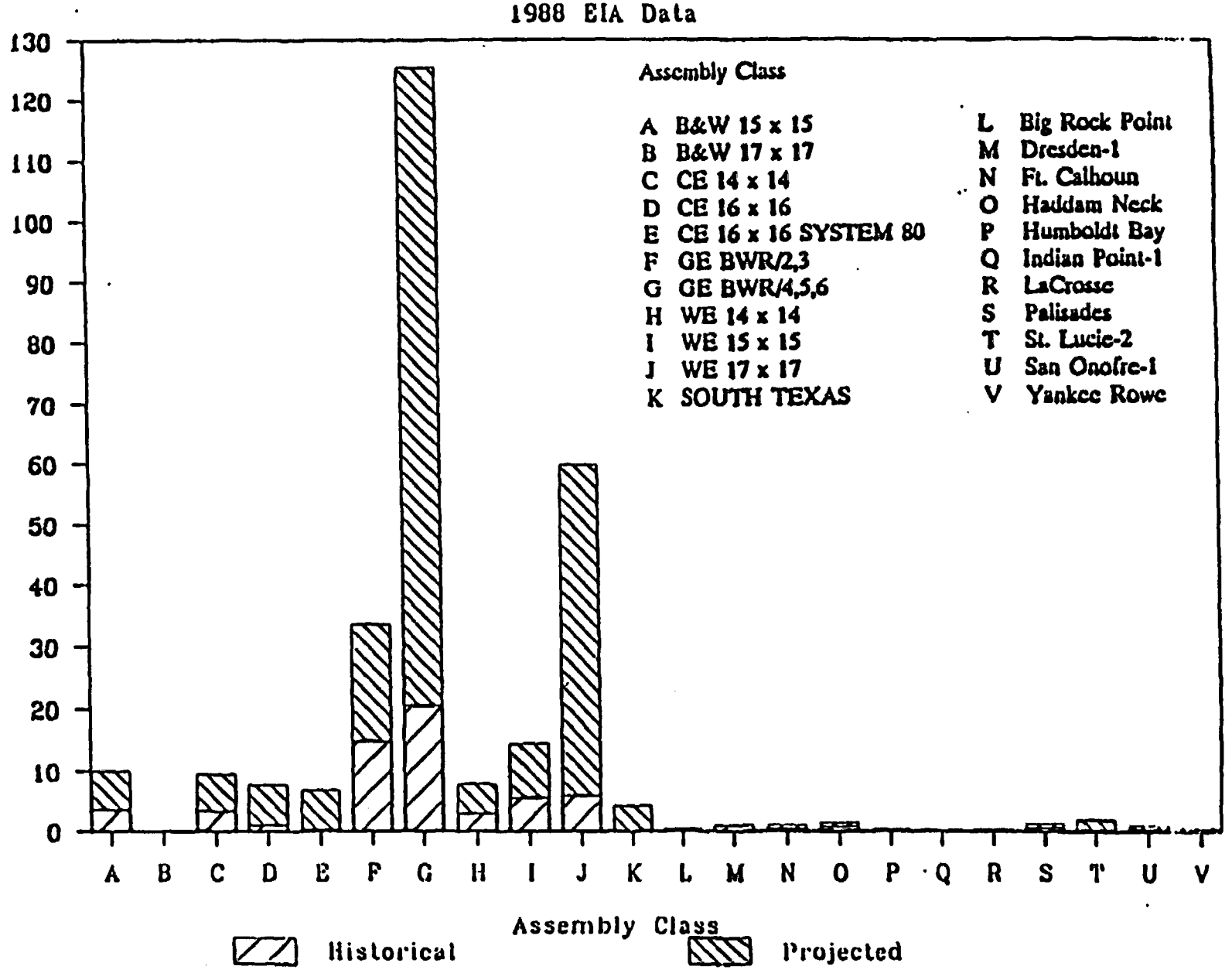
[a] R. S. Moore, D. A. Williamson, and K. J. Notz, A Classification Scheme for LWR Fuel Assemblies, ORNL/TM-10901, Oak Ridge National Laboratory, November 1988.

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.



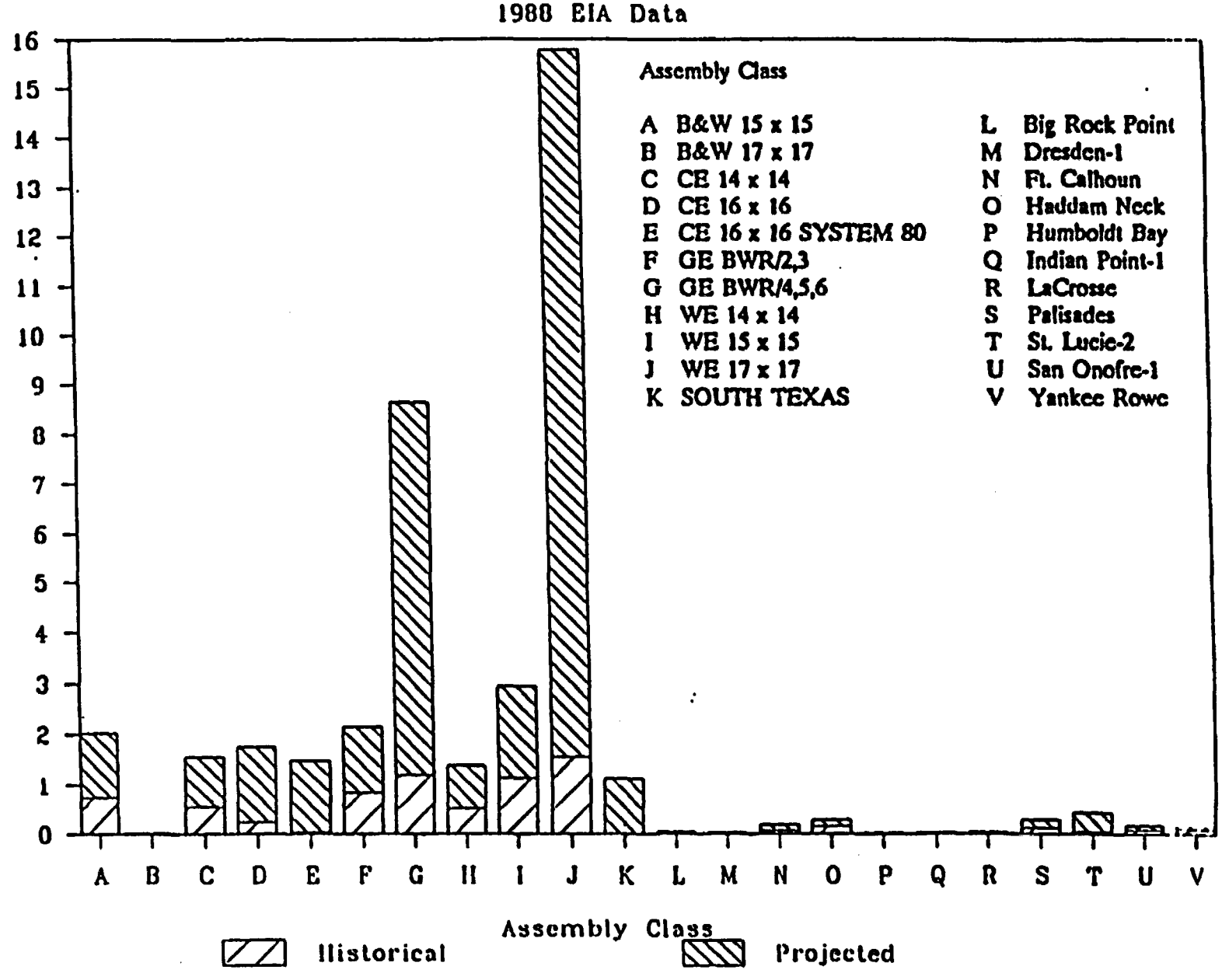
K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Figure 3.1 Assemblies by Assembly Class



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Figure 3.2 Fuel Rods by Assembly Class



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Table 3.4 Spent Fuel Distribution by Discharge Year, based on 1988 EIA Data

Discharge Year	BWR Assemblies	BWR Metric Tons	PWR Assemblies	PWR Metric Tons
		HISTORICAL		
1968	5	0.6	0	0.0
1969	96	9.7	0	0.0
1970	29	5.6	99	39.0
1971	408	64.0	64	26.5
1972	771	141.5	331	117.9
1973	577	95.2	165	67.1
1974	1314	244.6	574	207.3
1975	1170	215.0	797	321.8
1976	1584	298.6	920	396.6
1977	2045	382.7	1087	457.7
1978	2239	383.2	1661	696.7
1979	2131	399.9	1658	719.4
1980	3330	619.8	1469	624.0
1981	2467	458.7	1610	686.3
1982	1951	357.2	1519	652.8
1983	2698	491.3	1763	764.5
1984	2623	462.2	1953	848.2
1985	2674	485.2	2045	867.1
1986	2583	464.0	2365	1030.2
1987	3506	632.2	2715	1162.1
1988	3008	545.2	2746	1165.4

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Table 3.5 Historical Spent Fuel Distribution by Discharge Burnup, based on 1988 EIA Data

HISTORICAL DATA

Burnup (GWd/MTIHM)	BWR		PWR	
	Assemblies	Metric Tons	Assemblies	Metric Tons
0	30	5.6	0	0.0
1	46	8.5	0	0.0
2	178	34.3	0	0.0
3	432	79.7	1	0.4
4	899	164.0	7	2.6
5	70	5.9	40	7.2
6	189	35.5	9	4.1
7	334	62.4	0	0.0
8	182	30.7	29	12.1
9	680	124.2	109	43.6
10	1348	254.9	5	2.3
11	426	77.8	133	54.9
12	760	136.7	191	75.8
13	493	81.8	234	86.6
14	804	144.8	305	129.8
15	421	73.6	315	137.6
16	1090	196.7	512	220.8
17	1569	276.6	958	430.2
18	857	133.0	429	183.0
19	1970	366.4	568	246.0
20	1372	248.5	342	151.9
21	1413	258.4	378	168.5
22	1726	309.1	232	96.6
23	2390	444.8	345	139.7
24	2637	488.5	526	237.6
25	3215	592.4	922	377.1
26	2131	389.5	1013	446.9
27	1832	334.2	1367	573.3
28	3181	576.2	1088	460.6
29	1569	284.5	1308	550.5
30	2340	424.2	1395	588.4
31	475	85.8	1957	840.9
32	89	16.3	1460	626.0
33	31	5.6	1993	859.6
34	10	1.7	1956	840.6
35	4	0.7	1200	502.6

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Table 3.5 (cont.) Historical Spent Fuel Distribution by Discharge Burnup based on 1988 EIA Data

HISTORICAL DATA

Burnup (GWd/MTIHM)	BWR		PWR	
	Assemblies	Metric Tons	Assemblies	Metric Tons
36	0	0.0	1229	515.9
37	4	0.7	969	417.8
38	2	0.4	698	289.0
39	3	0.6	612	249.8
40	0	0.0	245	100.7
41	2	0.4	148	56.5
42	1	0.2	124	47.1
43	4	0.7	142	55.9
44	0	0.0	4	1.6
45	0	0.0	18	8.1
46	0	0.0	4	1.7
47	0	0.0	0	0.0
48	0	0.0	6	2.5
49	0	0.0	0	0.0
50	0	0.0	0	0.0
51	0	0.0	2	0.8
52	0	0.0	2	0.8
53	0	0.0	2	0.8
54	0	0.0	0	0.0
55	0	0.0	4	1.8
56	0	0.0	4	1.5
57	0	0.0	1	0.4

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Figure 3.3 BWR Discharges by Year

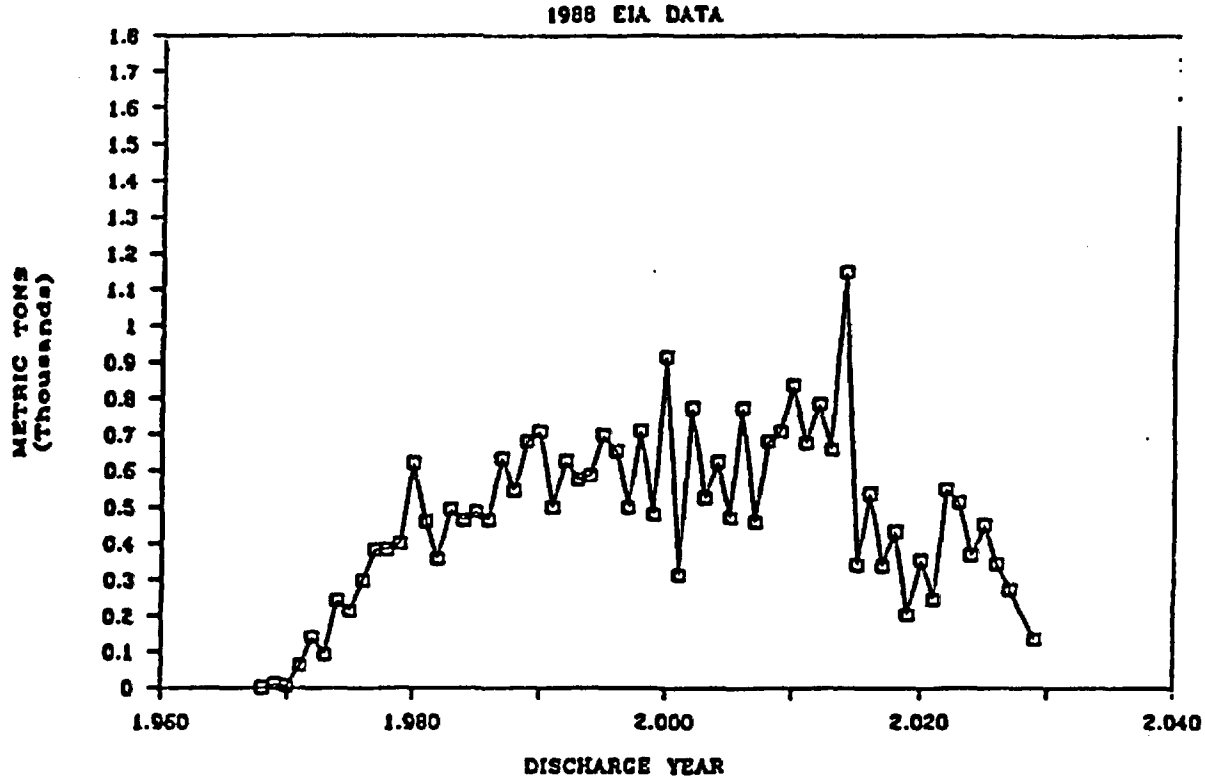
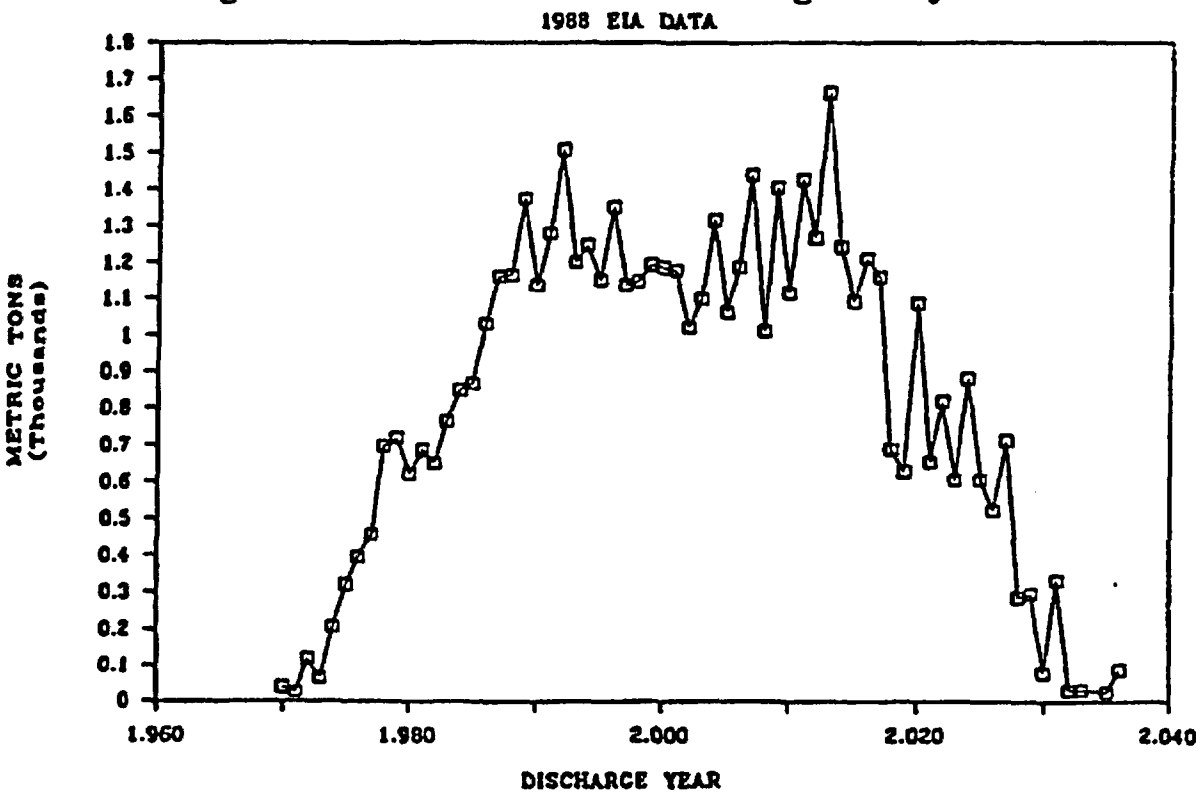


Figure 3.4 PWR Discharges by Year



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Figure 3.5 BWR Discharges by Burnup

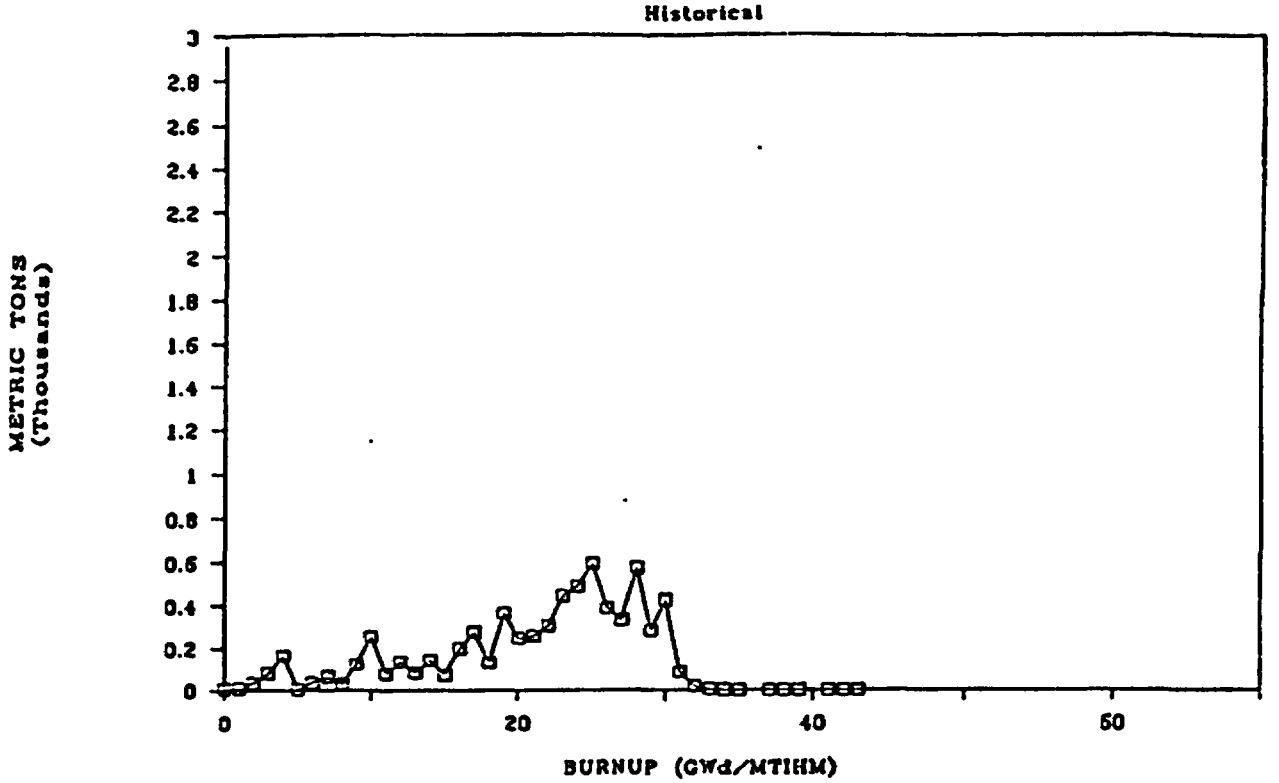
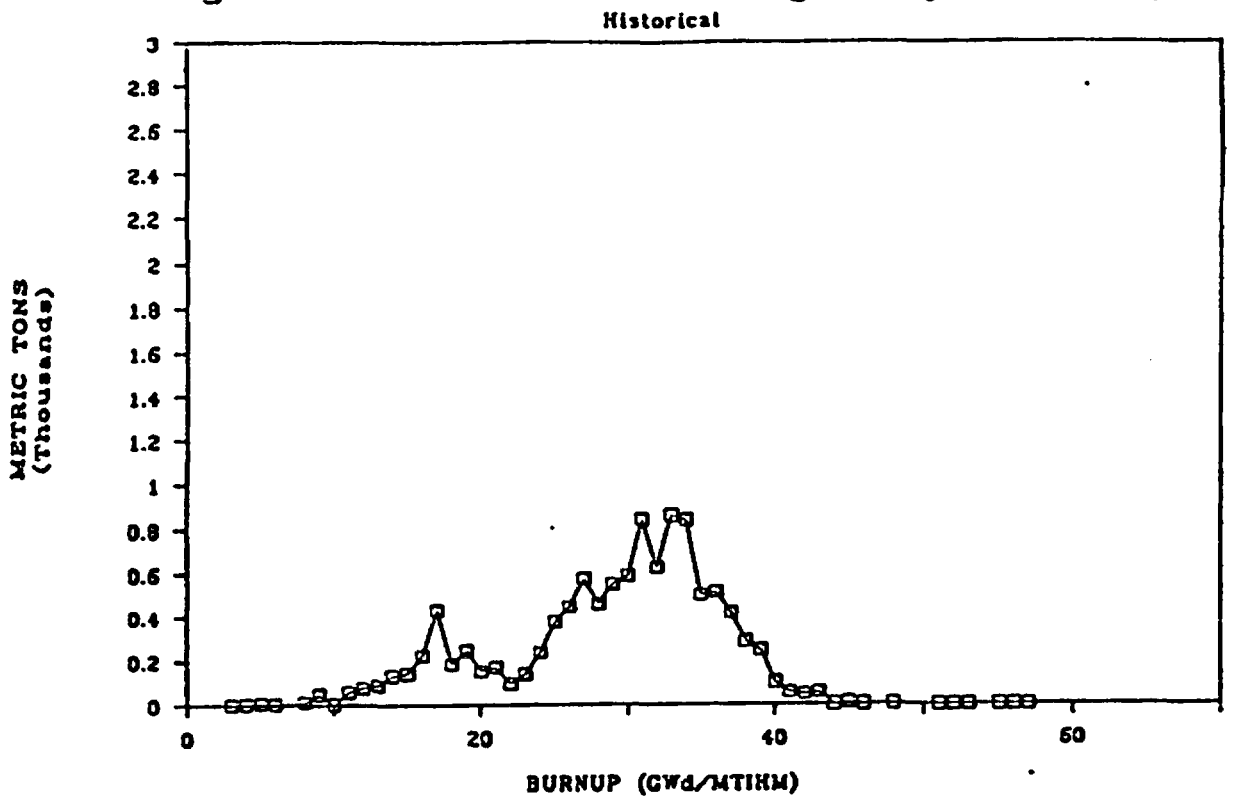


Figure 3.7 PWR Discharges by Burnup



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Table 4.1 Summary of the Quantities of LWR Spent Fuel.

		Historical or Projected					
		Through 1988			Projected		
		Reactor Type			Reactor Type		
		BWR	PWR	Summary	BWR	PWR	Summary
No. of Assemblies	Total	37124	25540	62664	123822	101952	225774
Mass, Mtihm	Total	6741	10850	17590	22100	44217	66317

Table 4.2 Summary of the Quantities of LWR Spent Fuel.

		Summary		
		Reactor Type		
		BWR	PWR	Summary
No. of Assemblies	Total	160946	127492	288438
Mass, Mtihm	Total	28841	55067	83908

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Table 4.3 Summary LWR Spent Fuel Burnup, Enrichment, and Age.

		Historical or Projected					
		Through 1988			Projected		
		Reactor Type			Reactor Type		
		BWR	PWR	Summary	BWR	PWR	Summary
Burnup, MWd/Mt	Minimum	0	3000	0	3000	5000	3000
	MEAN	21213	28908	25959	32904	41987	38960
	Maximum	43000	57000	57000	47000	67000	67000
	Standard Deviation	19979	20181	22679	30273	28528	32518
Enrichment	Minimum	0.0	0.7	0.0	0.7	1.3	0.7
	MEAN	2.3	2.9	2.7	3.2	3.9	3.7
	Maximum	3.9	4.9	4.9	3.9	5.2	5.2
	Standard Deviation	1.4	1.5	1.7	1.5	1.5	1.9
Discharge Date	Minimum	1968	1970	1968	1989	1989	1989
	MEAN	1981	1982	1982	2007	2007	2007
	Maximum	1988	1988	1988	2029	2036	2036

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Table 4.4 Summary LWR Spent Fuel Burnup, Enrichment, and Age.

		Summary		
		Reactor Type		
		BWR	PWR	Summary
Burnup, MWd/Mt	Minimum	0	3000	0
	MEAN	30172	39410	36234
	Maximum	47000	67000	67000
	Standard Deviation	32110	31340	34685
Enrichment	Minimum	0.0	0.7	0.0
	MEAN	3.0	3.7	3.5
	Maximum	3.9	5.2	5.2
	Standard Deviation	1.9	2.0	2.3
Discharge Date	Minimum	1968	1970	1968
	MEAN	2001	2002	2002
	Maximum	2029	2036	2036

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

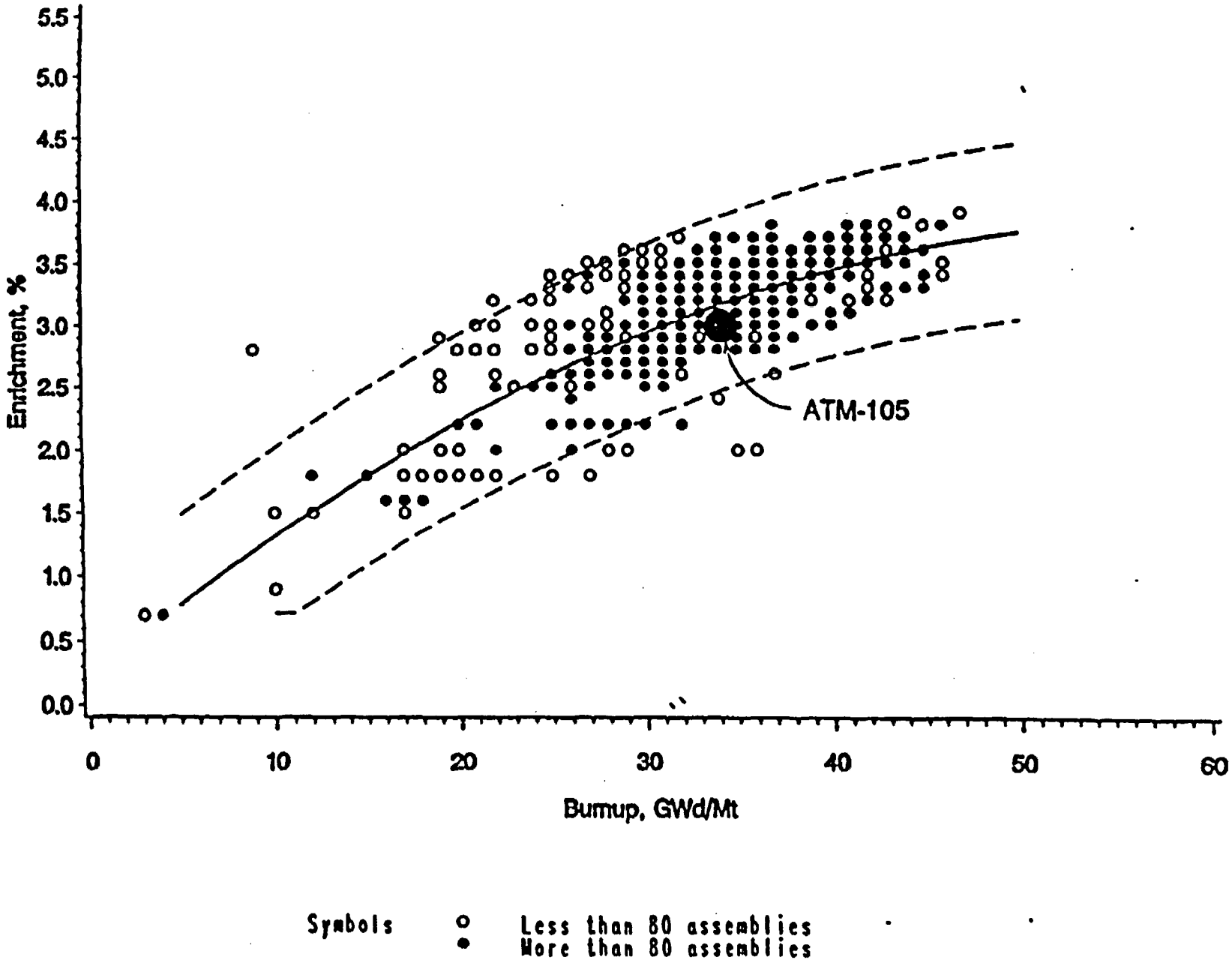
Table 3.3. Total quantities (Historical and Projected) of spent fuel by Assembly Class (reproduced from the LWR Quantities Data Base).

LWR QUANTITIES DATABASE
Totals - Historical and Projected Data
Discharged Assemblies by Assembly Class

ASSEMBLY CLASS	FUEL ASSEMBLIES	FUEL RODS	AVERAGE BURNUP (MWd/MT)	TOTAL WEIGHT (MT)	AVERAGE INITIAL ENRICH.
B&W 15 X 15	9,892	2,031K	36230	4586.4	3.446
CE 14 X 14	9,391	1,545K	39547	3554.5	3.726
CE 16 X 16	7,898	1,758K	41118	3234.1	3.926
CE 16 X 16 SYSTEM 80	6,715	1,477K	43186	2795.7	3.951
GE BWR/2,3	33,403	2,147K	27220	5998.2	2.838
GE BWR/4-6	125,409	8,645K	31091	22621.3	3.039
WE 14 X 14	7,818	1,392K	37910	2899.3	3.538
WE 15 X 15	14,451	2,947K	36967	6559.3	3.482
WE 17 X 17	59,759	15,778K	40767	26472.7	3.807
South Texas	4,258	1,124K	34904	2303.7	3.264
Big Rock Point	604	63K	20611	79.3	3.464
Dresden-1	891	32K	16227	90.8	2.166
Ft. Calhoun	1,094	191K	37237	391.6	3.549
Haddam Neck	1,407	287K	33892	548.7	3.833
Humboldt Bay	390	15K	14936	28.9	2.351
Indian Point	160	28K	16715	30.6	4.111
Lacrosse	333	33K	14708	38.0	3.727
Palisades	1,285	275K	32638	513.6	3.368
St. Lucie-2	1,911	459K	44725	741.1	4.260
San Onofre-1	964	142K	30434	354.4	3.988
Yankee Rowe	678	162K	29684	160.9	3.950
— GRAND TOTALS	288,523	40,531K	36234	83924.2	3.477

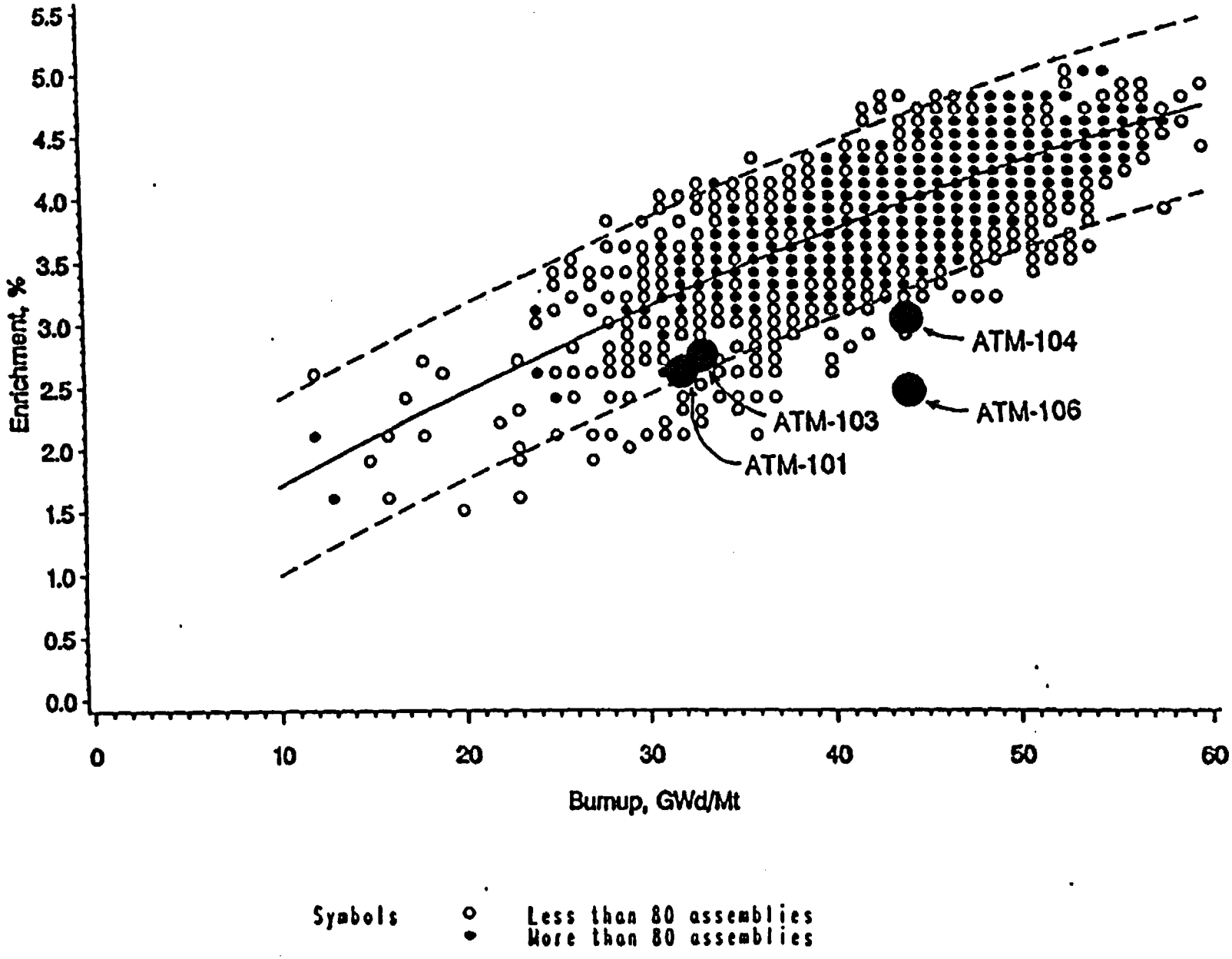
K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Figure 4.5 Enrichment as a Function of Burnup for BWRs



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

Figure 4.6 Enrichment as a Function of Burnup for PWRs



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) Sept. 1990.

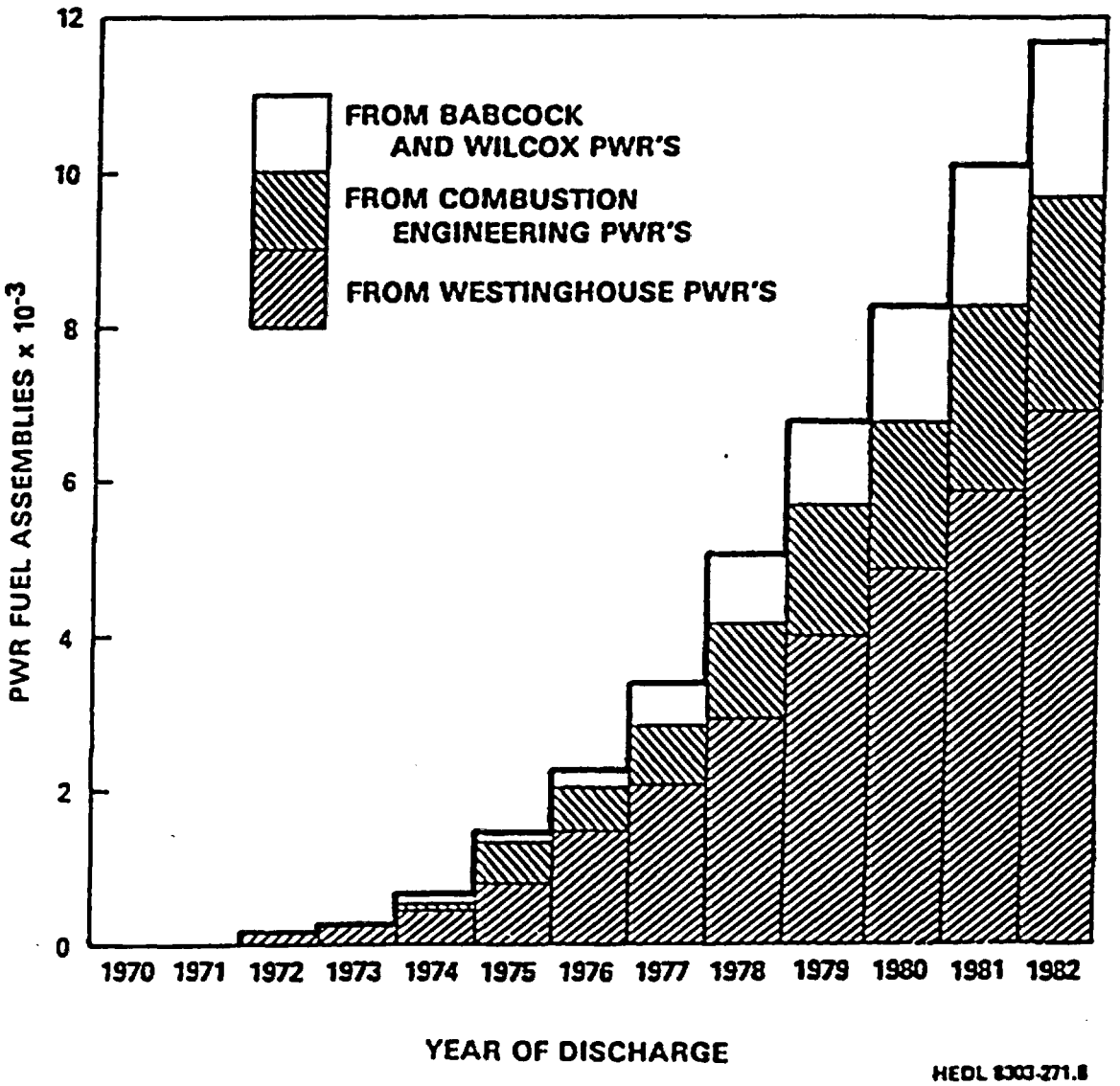


FIGURE 4. Cumulative Increase in All PWR Spent Fuel Assemblies with Time.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories.* HEDL-TME 83-28, Oct. 1983.

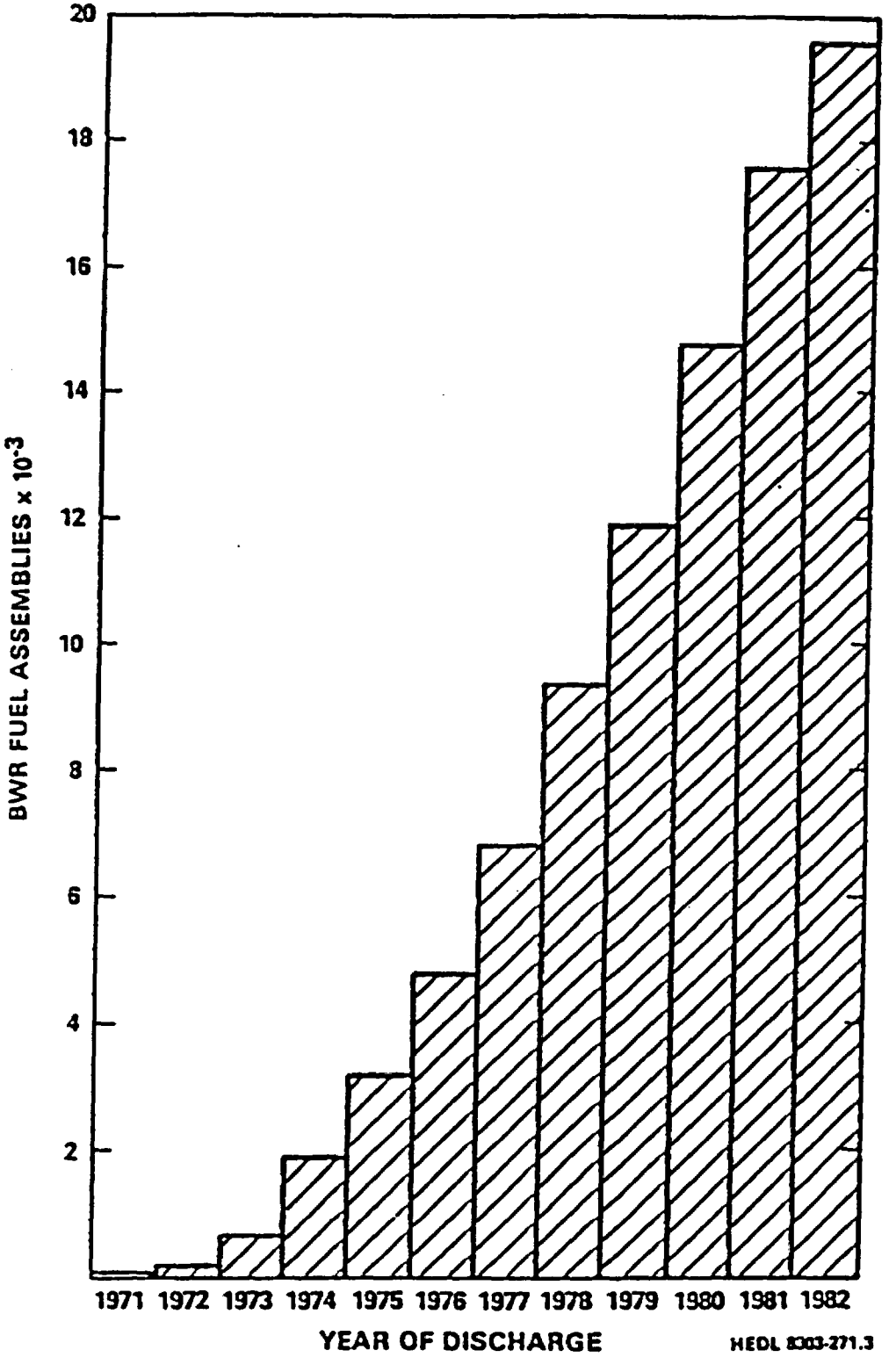


FIGURE 5. Cumulative Increase in All BWR Spent Fuel Assemblies with Time.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, Oct. 1983.

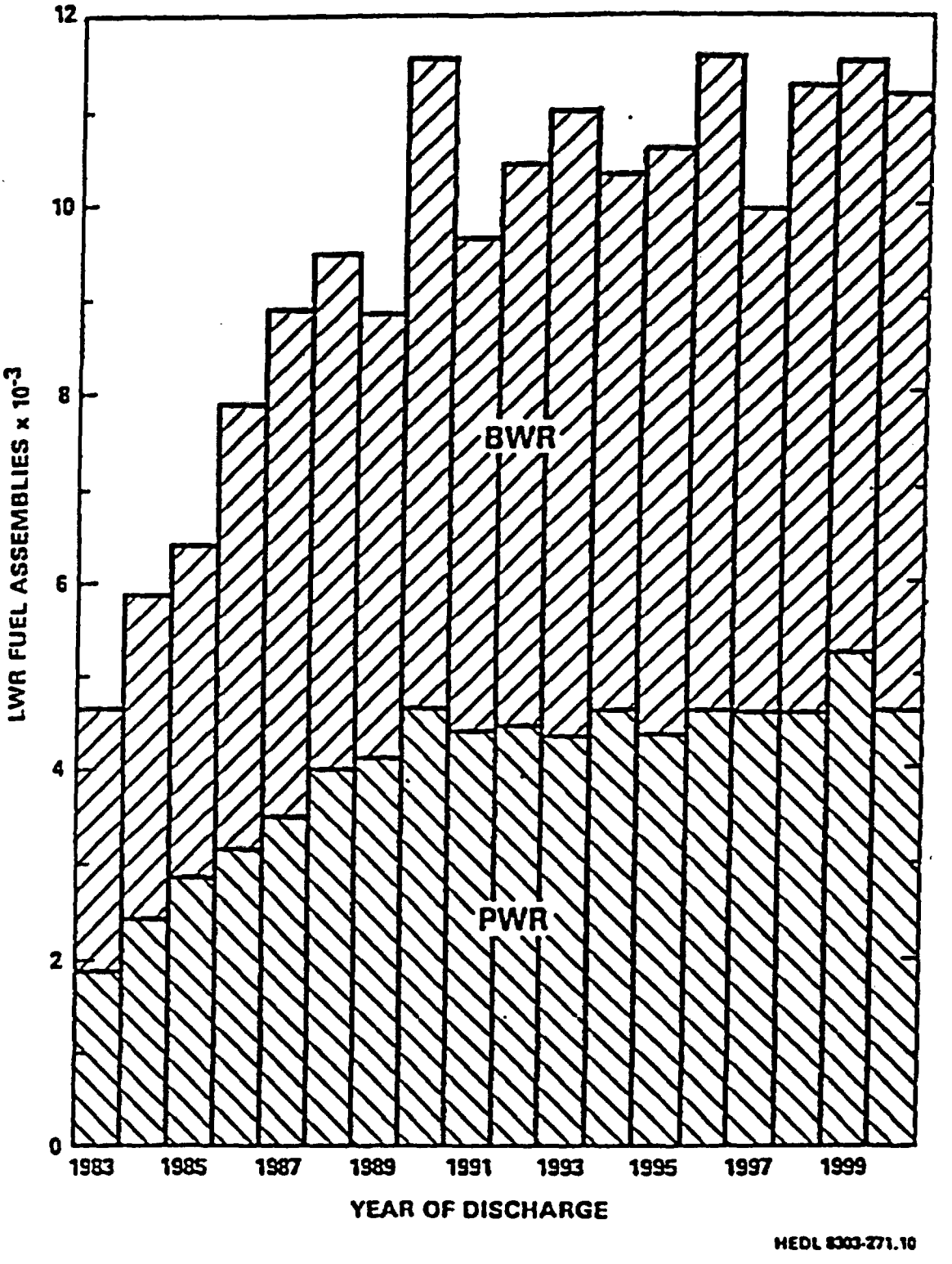


FIGURE 6. Projected Annual Discharges of Spent LWR Fuel Assemblies.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, Oct. 1983.

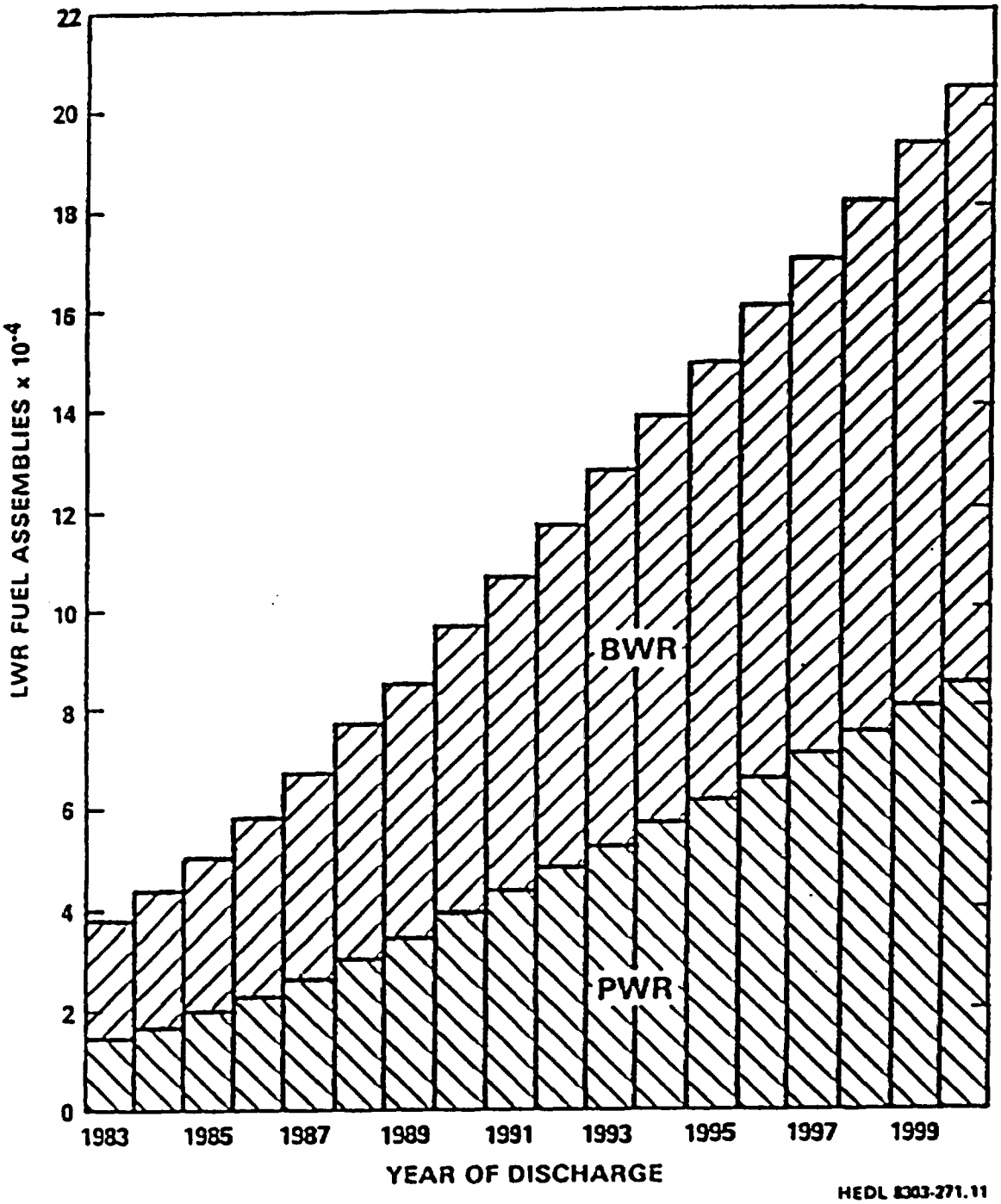


FIGURE 7. Cumulative Discharges of Spent LWR Fuel Assemblies.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, Oct. 1983.

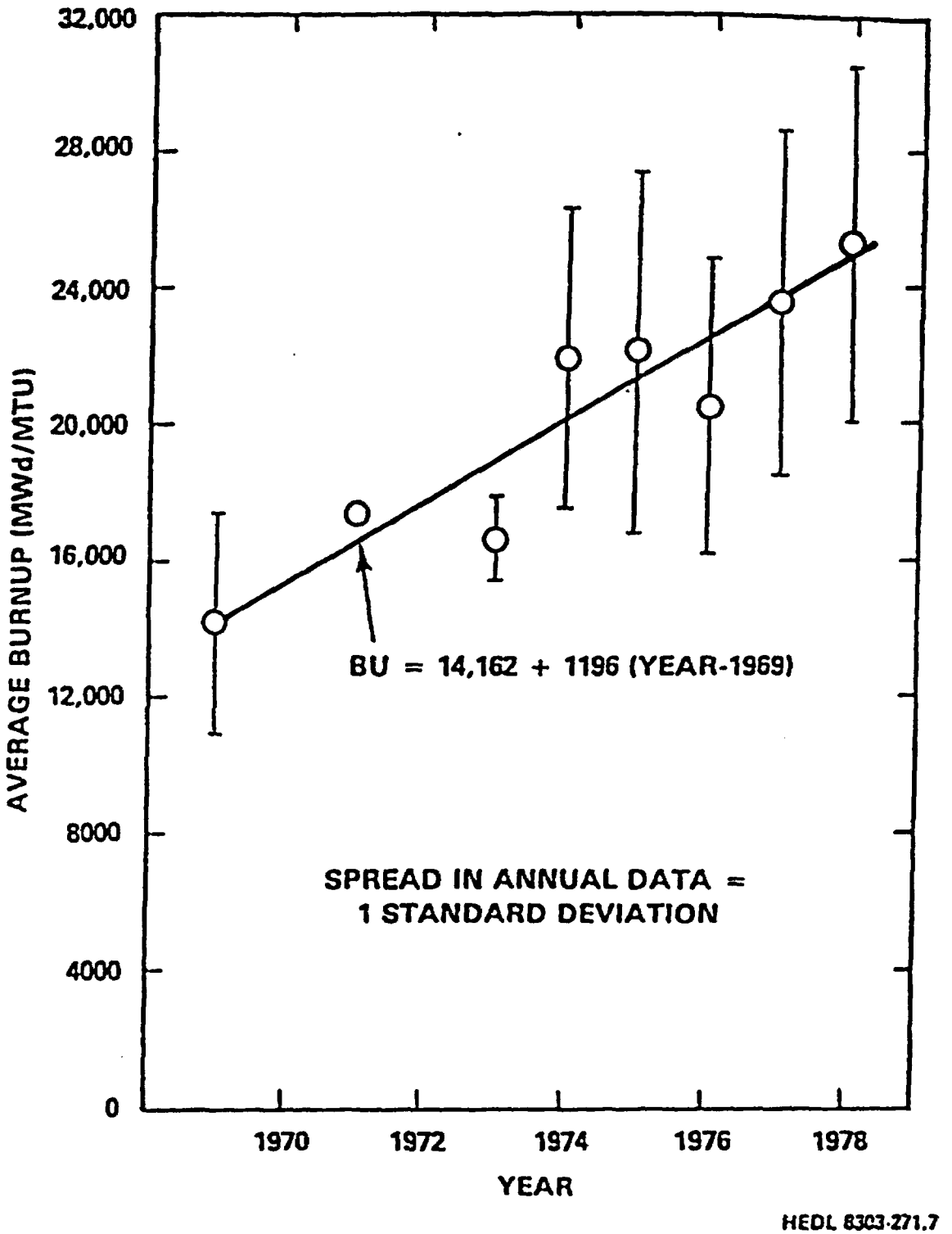


FIGURE 12. Temporal Variation in the Burnup of Fuel Discharged from LWRs (Adapted from Reference 26).

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, Oct. 1983.

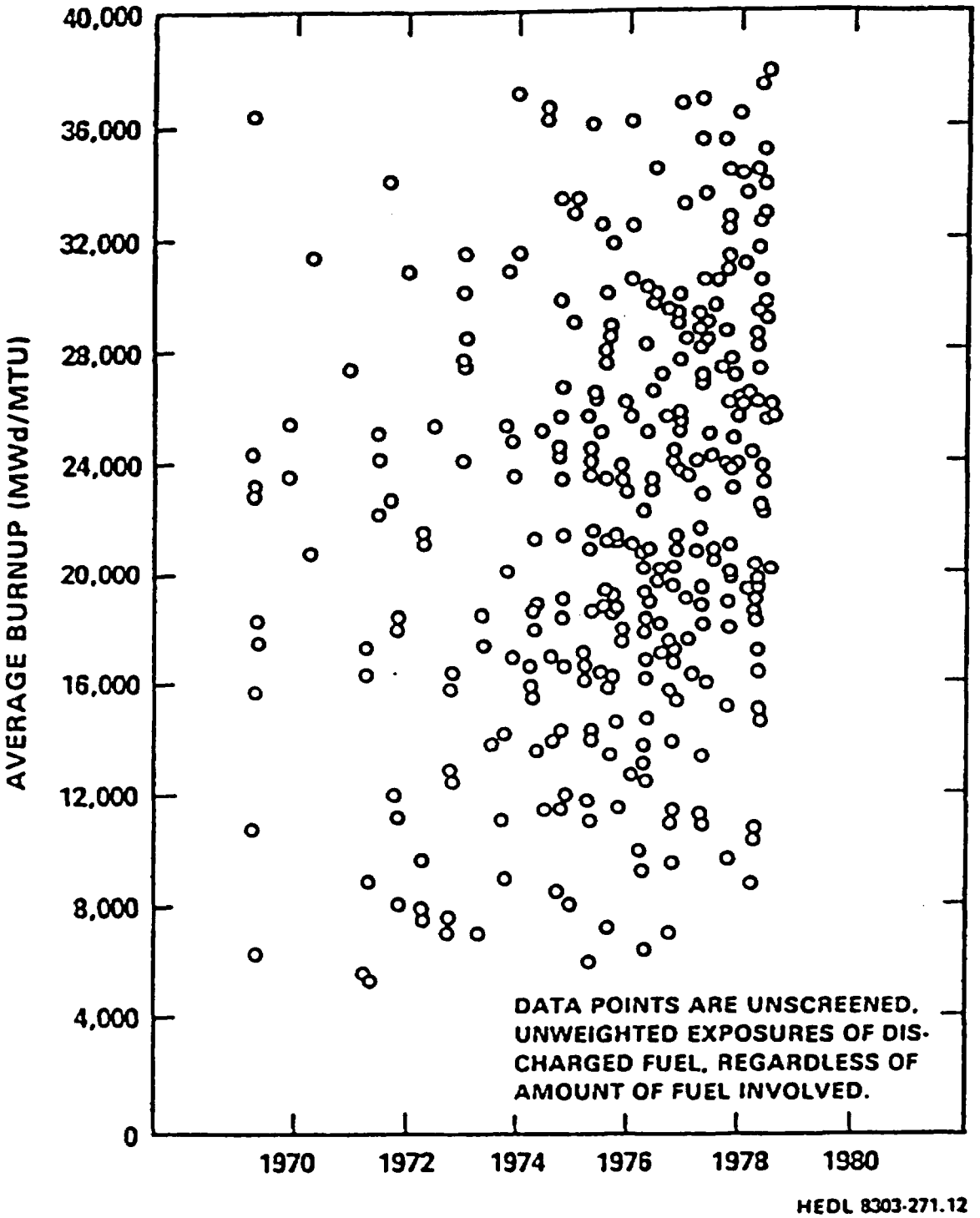


FIGURE 13. Average Burnup of Fuel Discharged from LWRs (Adapted from Reference 26).

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, Oct. 1983.

Table 4.22a Summary of burnup distribution percentiles

Reactor Type	Age Group	Minimum 0%	1%	5%	10%	25%	Median 50%	Mode	75%	90%	95%	99%	Maximum 100%
BWR	Historical	0	2000	7000	10000	16000	21000	23000	26000	29000	30000	35000	43000
PWR	Historical	3000	9000	15000	18000	25000	30000	31000	35000	38000	40000	48000	57000

Table 4.22b Summary of enrichment distribution percentiles

Reactor Type	Age Group	Minimum 0%	1%	5%	10%	25%	Median 50%	Mode	75%	90%	95%	99%	Maximum 100%
BWR	Historical	0.0	0.7	1.8	2.1	2.2	2.5	2.5	2.8	3.3	3.6	3.9	3.9
PWR	Historical	0.7	1.9	2.0	2.1	2.6	3.1	3.2	3.3	3.6	3.8	4.0	4.9

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

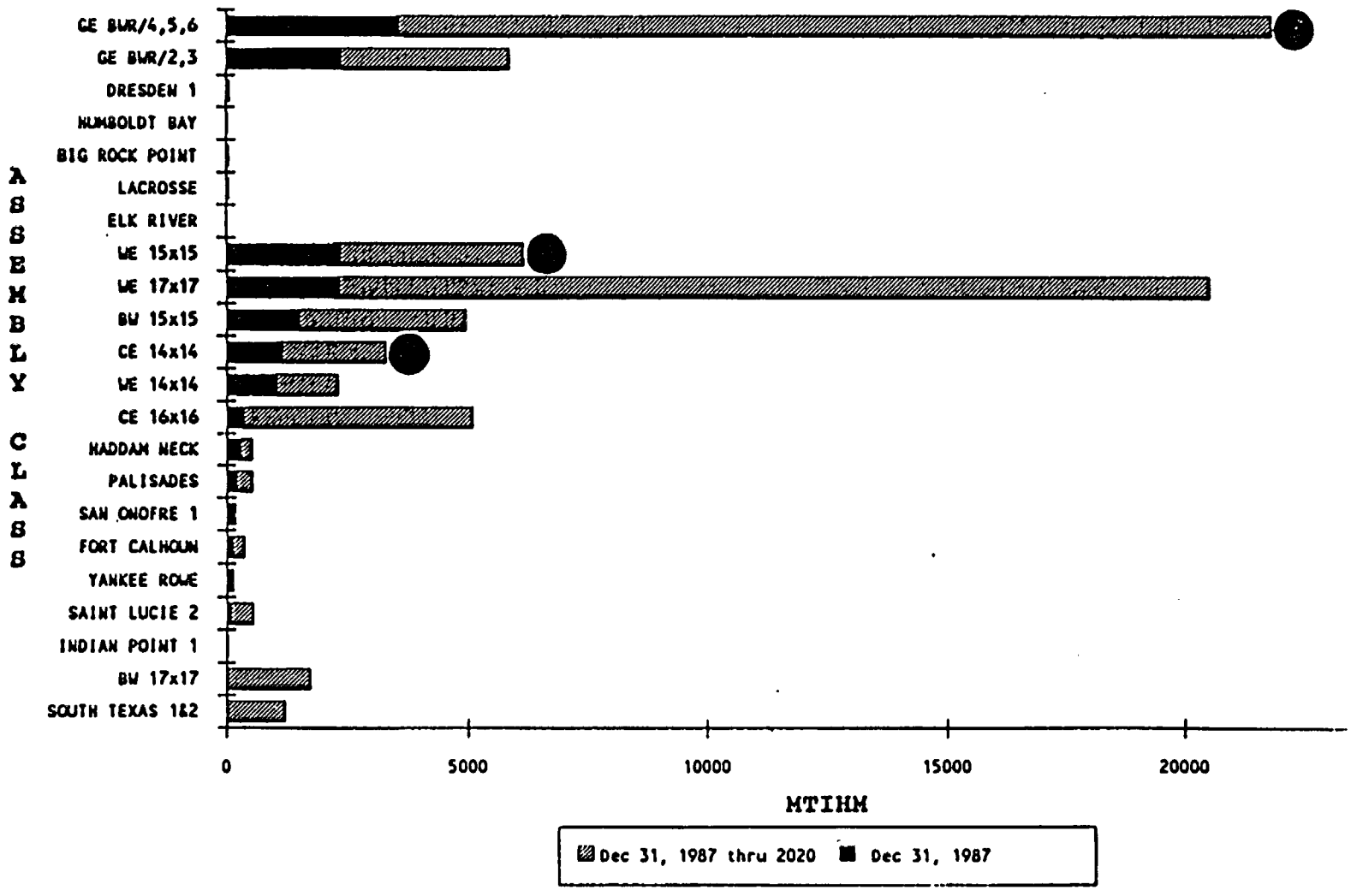
Table 2.2. Quantities of Domestic LWR Spent Fuel[a]

BWR Assembly Class	Projected Quantities from 1989 to 2036 (MTIHM)
GE BWR/4,5,6	18826
GE BWR/2,3	3236
Dresden 1	0
Humboldt Bay	0
Big Rock Point	38
Lacrosse	0
Elk River (reprocessed)	0
PWR Assembly Class	**Projected Quantities from 1989 to 2036 (MTIHM)**
WE 15x15	4052
WE 17x17	23803
BW 15x15	2932
CE 14x14	2283
WE 14x14	1753
CE 16x16	2722
CE 16x16 System 80	2716
South Texas	2304
Haddam Neck	246
Palisades	274
San Onofre 1	183
Fort Calhoun	238
Yankee Rowe	60
Saint Lucie 2	652
Indian Point 1	0
BW 17x17	0

[a] R. S. Moore, D. A. Williamson, and K. J. Notz, A Classification Scheme for LWR Fuel Assemblies, ORNL/TM-10901, Oak Ridge National Laboratory, November 1988.

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Figure 1. Quantities of Domestic LWR Spent Fuel



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.22a Summary of burnup distribution percentiles

Reactor Type	Age Group	Mininum 0%	1%	5%	10%	25%	Median 50%	Mode	75%	90%	95%	99%	Maximum 100%
BWR	Projected	3000	10000	16000	23000	29000	34000	33000	37000	40000	42000	45000	47000
BWR	Projected	5000	16000	27000	32000	38000	44000	46000	49000	52000	55000	60000	67000

Table 4.22b Summary of enrichment distribution percentiles

Reactor Type	Age Group	Minimum 0%	1%	5%	10%	25%	Median 50%	Mode	75%	90%	95%	99%	Maximum 100%
BWR	Projected	0.7	1.8	2.6	2.8	3.0	3.2	3.3	3.4	3.5	3.6	3.8	3.9
BWR	Projected	1.3	2.1	3.0	3.3	3.6	4.0	4.0	4.2	4.5	4.7	4.9	5.2

Table 3.2. Projected quantities of spent fuel by Assembly Class (reproduced from the LWR Quantities Data Base).

LWR QUANTITIES DATABASE
Projected Data: No New Orders Case with Extended Burnup
Projected Assemblies by Assembly Class through 2036

ASSEMBLY CLASS	FUEL ASSEMBLIES	FUEL RODS	AVERAGE BURNUP (MWd/MT)	TOTAL WEIGHT (MT)	AVERAGE INITIAL ENRICH.
B&W 15 X 15	6,328	1,291K	40874	2931.6	3.802
CE 14 X 14	6,062	994K	45092	2283.0	4.205
CE 16 X 16	6,667	1,520K	44175	2721.6	4.184
CE 16 X 16 SYSTEM 80	6,527	1,436K	43925	2716.9	4.004
GE BWR/2,3	18,594	1,320K	32108	3236.1	3.226
GE BWR/4-6	104,939	7,451K	33078	18826.3	3.187
WE 14 X 14	4,869	872K	41571	1753.2	3.791
WE 15 X 15	8,894	1,814K	41199	4052.1	3.826
WE 17 X 17	53,886	14,226K	42218	23802.6	3.916
South Texas	4,258	1,124K	34904	2303.7	3.264
Big Rock Point	289	34K	22015	37.7	3.435
Ft. Calhoun	668	118K	41572	237.6	3.962
Haddam Neck	673	137K	37068	245.5	3.851
Palisades	688	149K	41290	274.3	4.004
St. Lucie-2	1,675	375K	47601	652.2	4.521
San Onofre-1	496	89K	31755	183.0	4.171
Yankee Rowe	261	60K	32018	60.3	3.953
— GRAND TOTALS	225,774	33,010K	38964	66317.6	3.678

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 3.3. Total quantities (Historical and Projected) of spent fuel by Assembly Class (reproduced from the LWR Quantities Data Base).

LWR QUANTITIES DATABASE
Totals - Historical and Projected Data
Discharged Assemblies by Assembly Class

ASSEMBLY CLASS	FUEL ASSEMBLIES	FUEL RODS	AVERAGE BURNUP (MWd/MT)	TOTAL WEIGHT (MT)	AVERAGE INITIAL ENRICH.
B&W 15 X 15	9,892	2,031K	36230	4586.4	3.446
CE 14 X 14	9,391	1,545K	39547	3554.5	3.726
CE 16 X 16	7,898	1,758K	41118	3234.1	3.926
CE 16 X 16 SYSTEM 80	6,715	1,477K	43186	2795.7	3.951
GE BWR/2,3	33,403	2,147K	27220	5998.2	2.838
GE BWR/4-6	125,409	8,645K	31091	22621.3	3.039
WE 14 X 14	7,818	1,392K	37910	2899.3	3.538
WE 15 X 15	14,451	2,947K	36967	6559.3	3.482
WE 17 X 17	59,759	15,778K	40767	26472.7	3.807
South Texas	4,258	1,124K	34904	2303.7	3.264
Big Rock Point	604	63K	20611	79.3	3.464
Dresden-1	891	32K	16227	90.8	2.166
Ft. Calhoun	1,094	191K	37237	391.6	3.549
Haddam Neck	1,407	287K	33892	548.7	3.833
Humboldt Bay	390	15K	14936	28.9	2.351
Indian Point	160	28K	16715	30.6	4.111
Lacrosse	333	33K	14708	38.0	3.727
Palisades	1,285	275K	32638	513.6	3.368
St. Lucie-2	1,911	459K	44725	741.1	4.260
San Onofre-1	964	142K	30434	354.4	3.988
Yankee Rowe	678	162K	29684	160.9	3.950
— GRAND TOTALS	288,523	40,531K	36234	83924.2	3.477

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 3.4 Spent Fuel Distribution by Discharge Year, based on 1988 EIA Data

Discharge Year	BWR Assemblies	BWR Metric Tons	PWR Assemblies	PWR Metric Tons
		PROJECTED		
1989	3810	680.9	3160	1372.0
1990	3972	707.2	2707	1136.6
1991	2804	498.1	2942	1279.4
1992	3496	625.1	3526	1507.2
1993	3240	575.6	2774	1201.3
1994	3296	586.0	2903	1247.3
1995	3928	696.4	2669	1151.8
1996	3658	653.2	3138	1350.4
1997	2802	498.8	2656	1138.6
1998	3954	709.3	2663	1148.8
1999	2702	478.1	2775	1194.5
2000	5116	913.5	2789	1185.8

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 3.4 (cont.) Spent Fuel Distribution by Discharge Year, based on 1988 EIA Data

Discharge Year	BWR Assemblies	BWR Metric Tons	PWR Assemblies	PWR Metric Tons
		PROJECTED		
2001	1762	312.3	2694	1179.1
2002	4338	772.2	2381	1022.3
2003	2939	523.8	2549	1100.3
2004	3479	622.1	3025	1316.0
2005	2626	468.6	2467	1063.5
2006	4311	772.0	2738	1188.8
2007	2561	455.6	3373	1439.1
2008	3807	678.3	2342	1011.9
2009	3984	707.2	3267	1404.2
2010	4695	834.9	2573	1116.1
2011	3814	675.3	3320	1425.4
2012	4379	783.4	2917	1267.1
2013	3686	658.1	3814	1662.2
2014	6381	1149.8	2855	1244.1
2015	1886	340.4	2509	1092.7
2016	2966	534.9	2784	1211.9
2017	1878	337.6	2639	1161.3
2018	2392	429.9	1560	691.1
2019	1125	202.2	1445	631.5
2020	1979	352.2	2452	1087.2
2021	1363	246.0	1475	654.8
2022	3086	545.7	1889	819.3
2023	2857	511.3	1406	607.6
2024	2065	366.3	2025	882.2
2025	2489	449.5	1380	607.5
2026	1884	341.4	1190	525.1
2027	1548	271.9	1597	712.8
2028	0	0.0	622	287.7
2029	764	135.4	625	296.9
2030	0	0.0	177	81.3
2031	0	0.0	772	333.8
2032	0	0.0	68	31.4
2033	0	0.0	68	31.4
2034	0	0.0	0	0.0
2035	0	0.0	59	27.1
2036	0	0.0	193	89.1

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 3.6 Projected Spent Fuel Distribution by Discharge Burnup, based on 1988 EIA Data

PROJECTED DATA

Burnup (GWd/MTIHM)	BWR		PWR	
	Assemblies	Metric Tons	Assemblies	Metric Tons
0	0	0.0	0	0.0
1	0	0.0	0	0.0
2	0	0.0	0	0.0
3	76	14.1	0	0.0
4	276	48.7	0	0.0
5	4	0.5	20	7.6
6	16	2.1	24	9.2
7	0	0.0	0	0.0
8	112	19.1	0	0.0
9	317	56.1	128	64.8
10	1020	181.1	0	0.0
11	446	79.5	63	29.1
12	872	156.6	392	164.0
13	1659	302.0	480	202.2
14	1623	289.4	211	89.6
15	1428	254.4	178	77.8
16	756	131.3	521	220.5
17	812	143.1	225	98.0
18	184	32.9	512	211.1
19	348	61.3	449	195.5
20	631	112.9	377	175.6
21	590	104.1	439	201.3
22	1524	268.6	567	256.9
23	511	88.5	458	203.7
24	594	104.7	492	210.5
25	1749	310.8	637	277.0
26	2378	431.2	519	225.9
27	3658	655.3	434	190.9
28	2233	402.7	471	199.1
29	4389	788.4	794	360.3
30	5443	976.1	456	214.6
31	5442	977.4	1126	508.3
32	7084	1273.6	1537	671.4
33	8321	1497.1	1889	856.6
34	7643	1371.4	2545	1199.8
35	7300	1314.6	1892	848.9

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

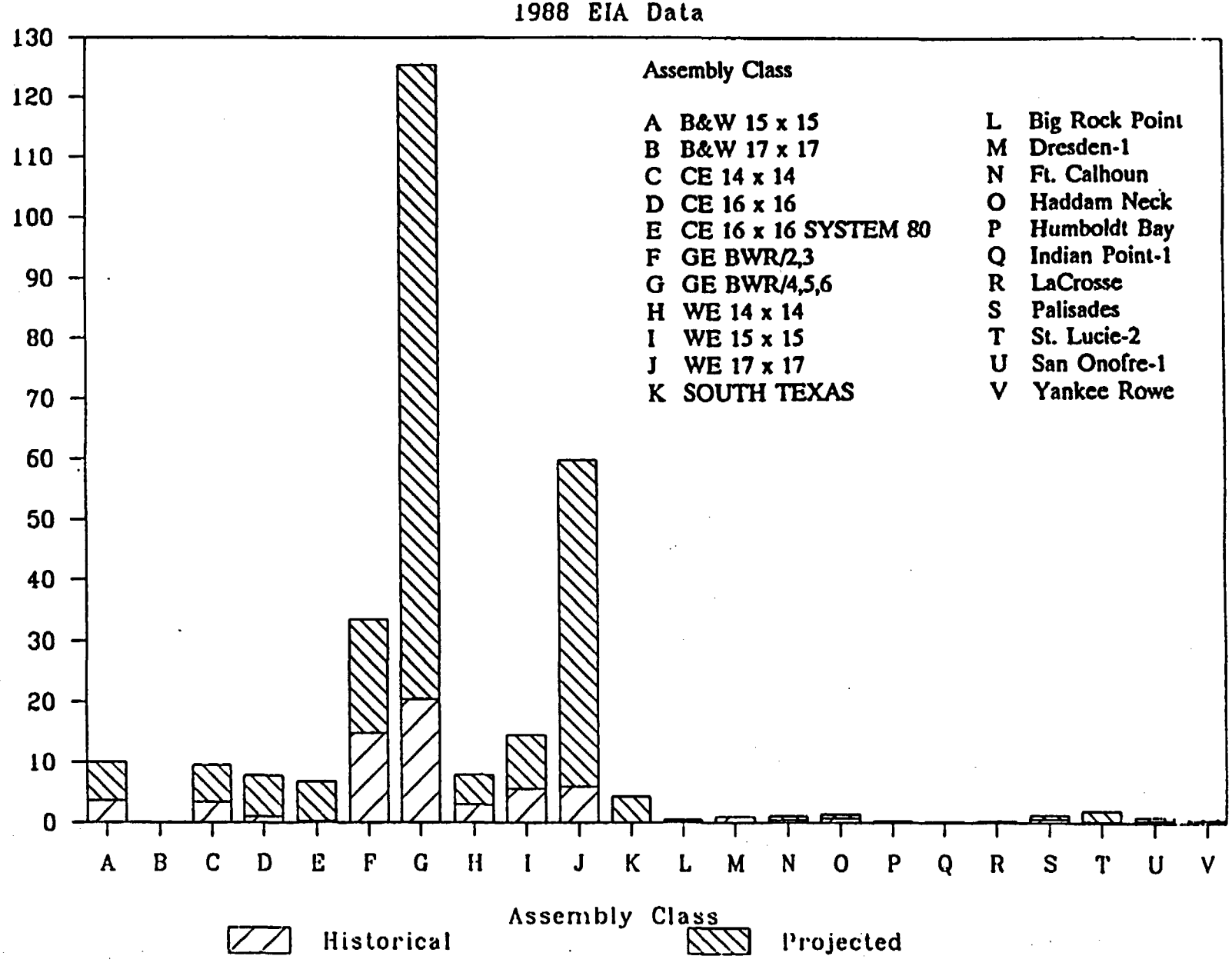
Table 3.6 (cont.) Projected Spent Fuel Distribution by Discharge Burnup, based on 1988 EIA Data

PROJECTED DATA

Burnup (GWd/MTIHM)	BWR		PWR	
	Assemblies	Metric Tons	Assemblies	Metric Tons
36	9028	1615.0	3249	1437.5
37	8714	1548.7	2560	1131.9
38	8035	1429.8	3712	1647.9
39	8510	1508.5	3221	1356.9
40	8707	1524.5	6448	2728.7
41	4199	738.5	4222	1818.2
42	2137	379.9	5029	2157.6
43	2326	413.5	4792	2076.0
44	1535	275.8	4744	2069.9
45	956	174.4	5719	2433.0
46	173	31.0	6911	2901.4
47	63	11.1	5825	2539.2
48	0	0.0	4710	2080.0
49	0	0.0	4848	2126.7
50	0	0.0	5029	2187.4
51	0	0.0	3114	1316.1
52	0	0.0	2765	1177.5
53	0	0.0	2401	1008.4
54	0	0.0	1204	487.8
55	0	0.0	1406	605.9
56	0	0.0	897	390.9
57	0	0.0	557	242.5
58	0	0.0	332	149.0
59	0	0.0	129	55.0
60	0	0.0	46	20.5
61	0	0.0	8	2.8
62	0	0.0	61	24.2
63	0	0.0	74	30.0
64	0	0.0	70	29.2
65	0	0.0	4	1.8
66	0	0.0	0	0.0
67	0	0.0	29	13.5

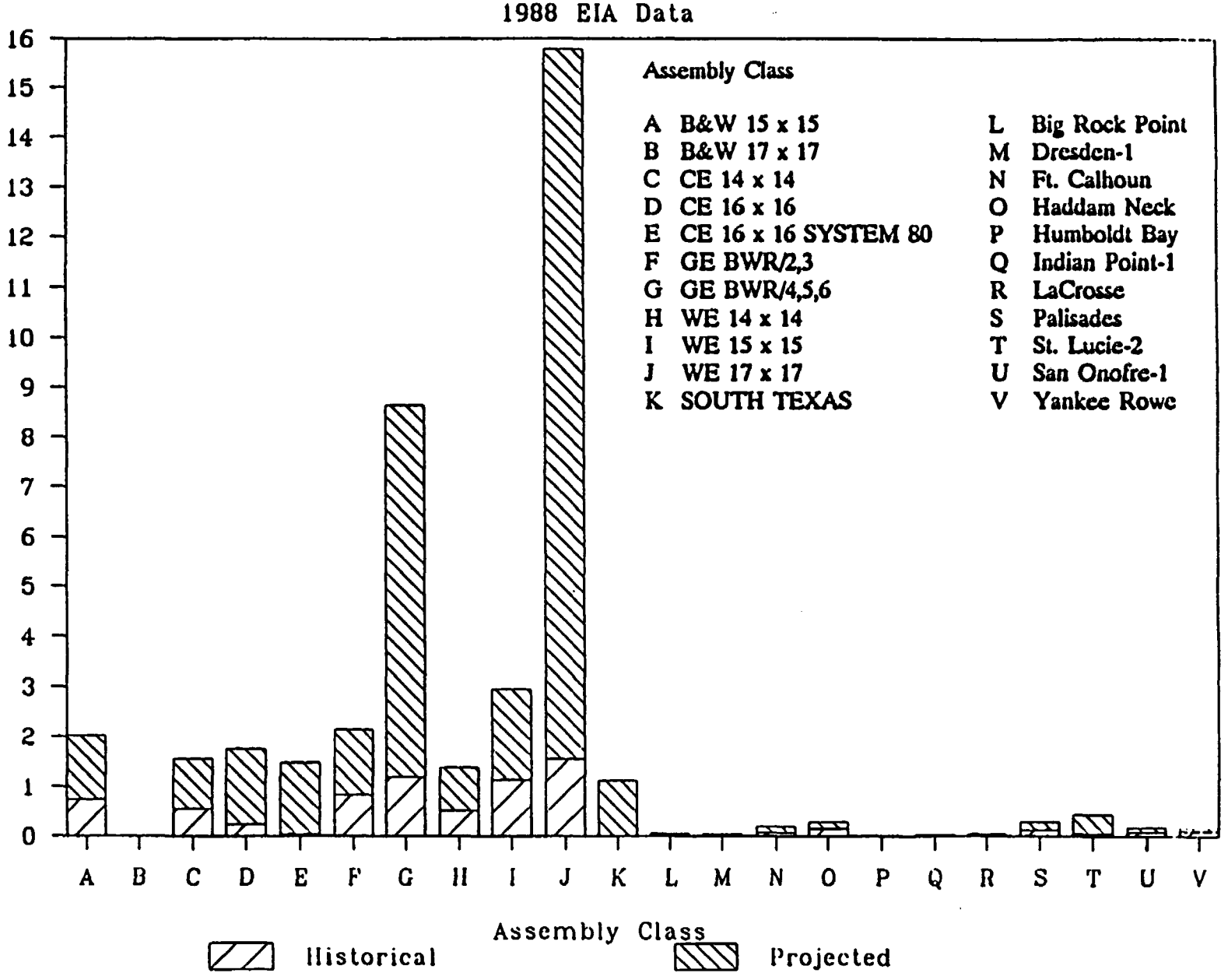
K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Figure 3.1 Assemblies by Assembly Class



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Figure 3.2 Fuel Rods by Assembly Class



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Figure 3.3 BWR Discharges by Year

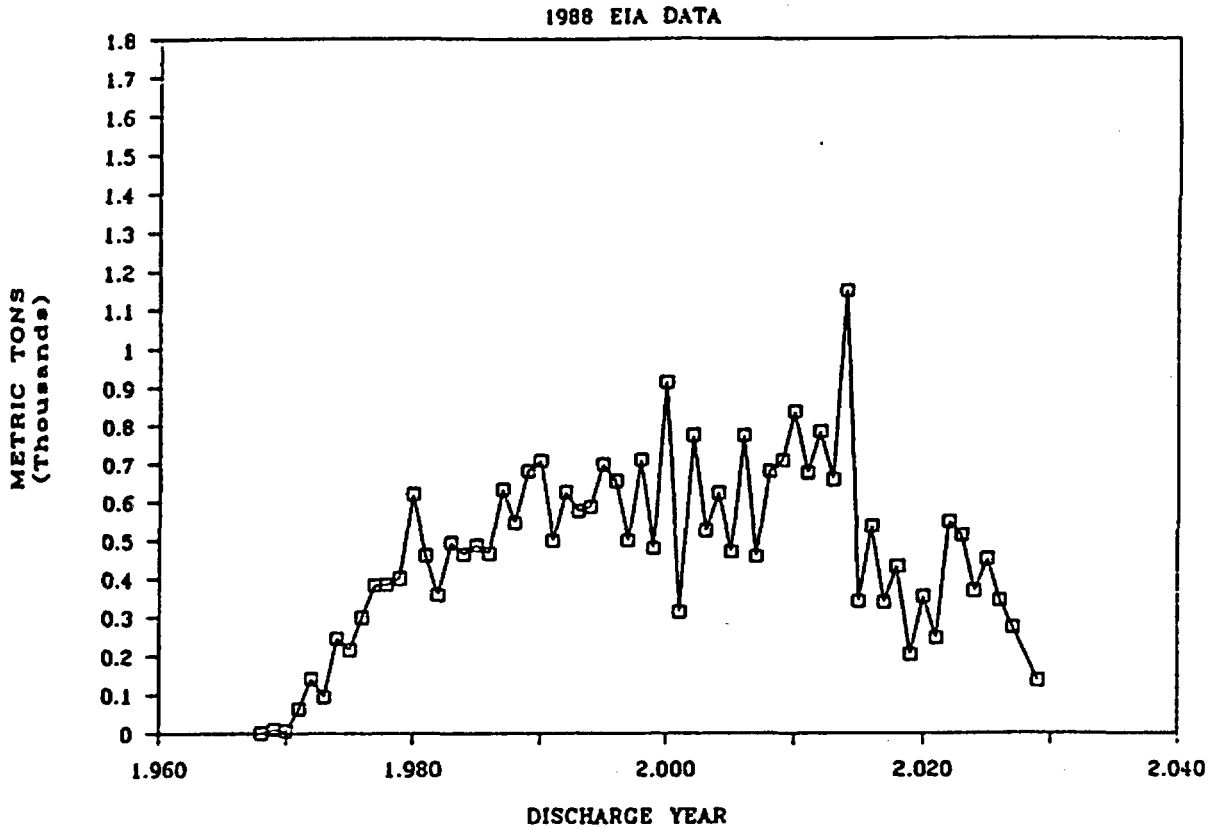
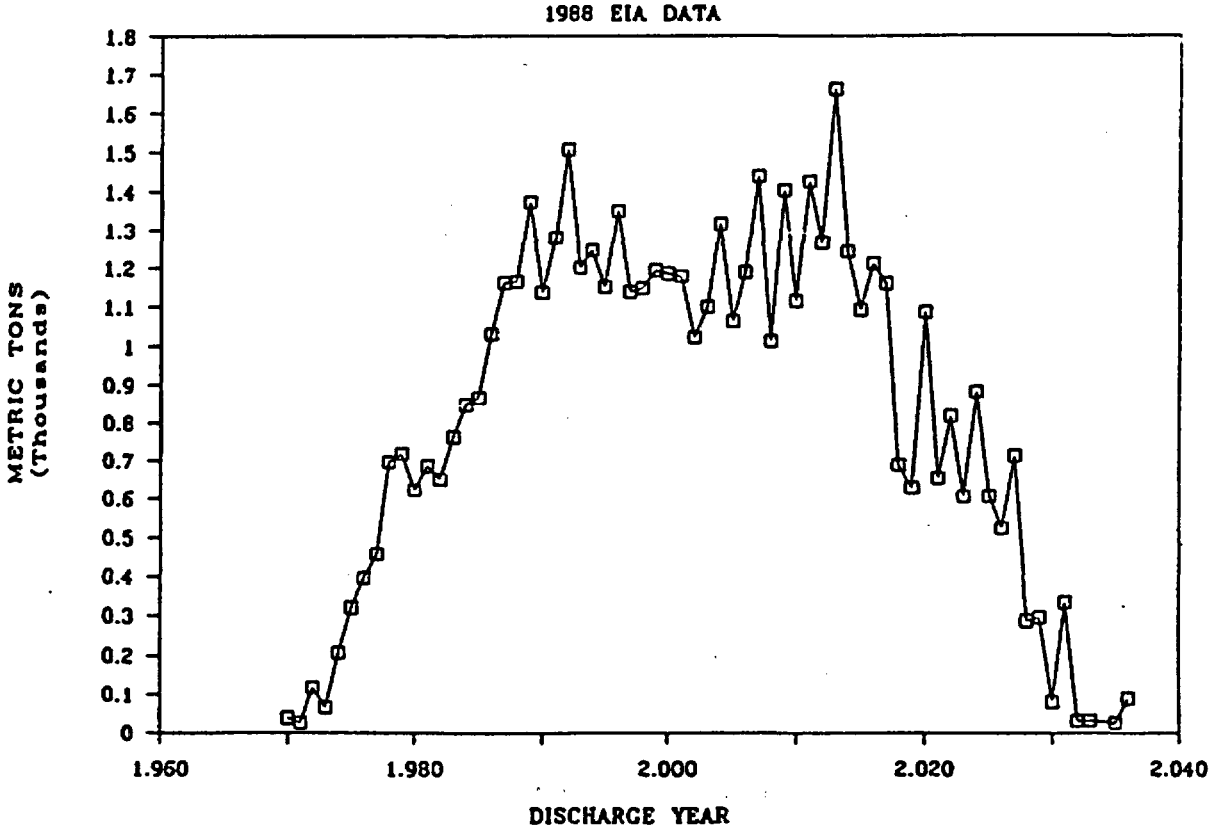


Figure 3.4 PWR Discharges by Year



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Figure 3.6 BWR Discharges by Burnup

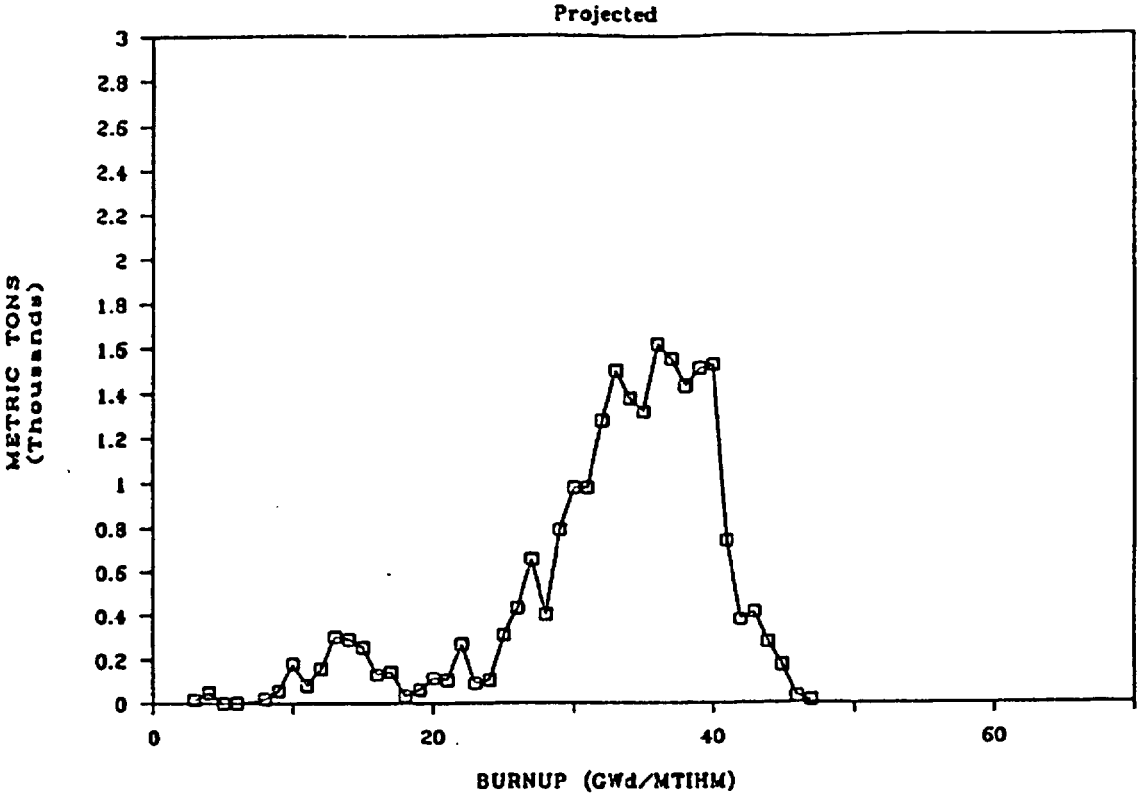
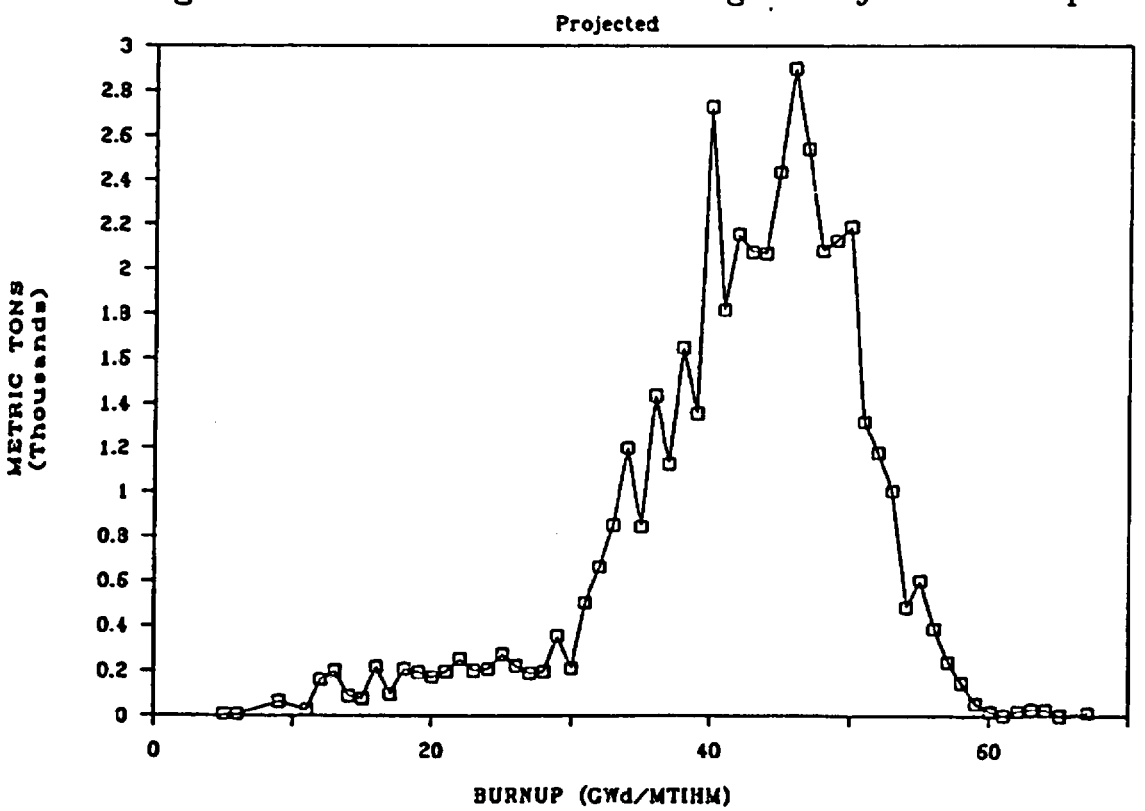


Figure 3.8 PWR Discharges by Burnup



K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.1 Summary of the Quantities of LWR Spent Fuel.

		Historical or Projected					
		Through 1988			Projected		
		Reactor Type			Reactor Type		
		BWR	PWR	Summary	BWR	PWR	Summary
No. of Assemblies	Total	37124	25540	62664	123822	101952	225774
Mass, Mtihm	Total	6741	10850	17590	22100	44217	66317

Table 4.2 Summary of the Quantities of LWR Spent Fuel.

		Summary		
		Reactor Type		
		BWR	PWR	Summary
No. of Assemblies	Total	160946	127492	288438
Mass, Mtihm	Total	28841	55067	83908

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.3 Summary LWR Spent Fuel Burnup, Enrichment, and Age.

		Historical or Projected					
		Through 1988			Projected		
		Reactor Type			Reactor Type		
		BWR	PWR	Summary	BWR	PWR	Summary
Burnup, MWd/Mt	Minimum	0	3000	0	3000	5000	3000
	MEAN	21213	28908	25959	32904	41987	38960
	Maximum	43000	57000	57000	47000	67000	67000
	Standard Deviation	19979	20181	22679	30273	28528	32518
Enrichment	Minimum	0.0	0.7	0.0	0.7	1.3	0.7
	MEAN	2.3	2.9	2.7	3.2	3.9	3.7
	Maximum	3.9	4.9	4.9	3.9	5.2	5.2
	Standard Deviation	1.4	1.5	1.7	1.5	1.5	1.9
Discharge Date	Minimum	1968	1970	1968	1989	1989	1989
	MEAN	1981	1982	1982	2007	2007	2007
	Maximum	1988	1988	1988	2029	2036	2036

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.4 Summary LWR Spent Fuel Burnup, Enrichment, and Age.

		Summary		
		Reactor Type		
		BWR	PWR	Summary
Burnup, MWd/Mt	Minimum	0	3000	0
	MEAN	30172	39410	36234
	Maximum	47000	67000	67000
	Standard Deviation	32110	31340	34685
Enrichment	Minimum	0.0	0.7	0.0
	MEAN	3.0	3.7	3.5
	Maximum	3.9	5.2	5.2
	Standard Deviation	1.9	2.0	2.3
Discharge Date	Minimum	1968	1970	1968
	MEAN	2001	2002	2002
	Maximum	2029	2036	2036

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

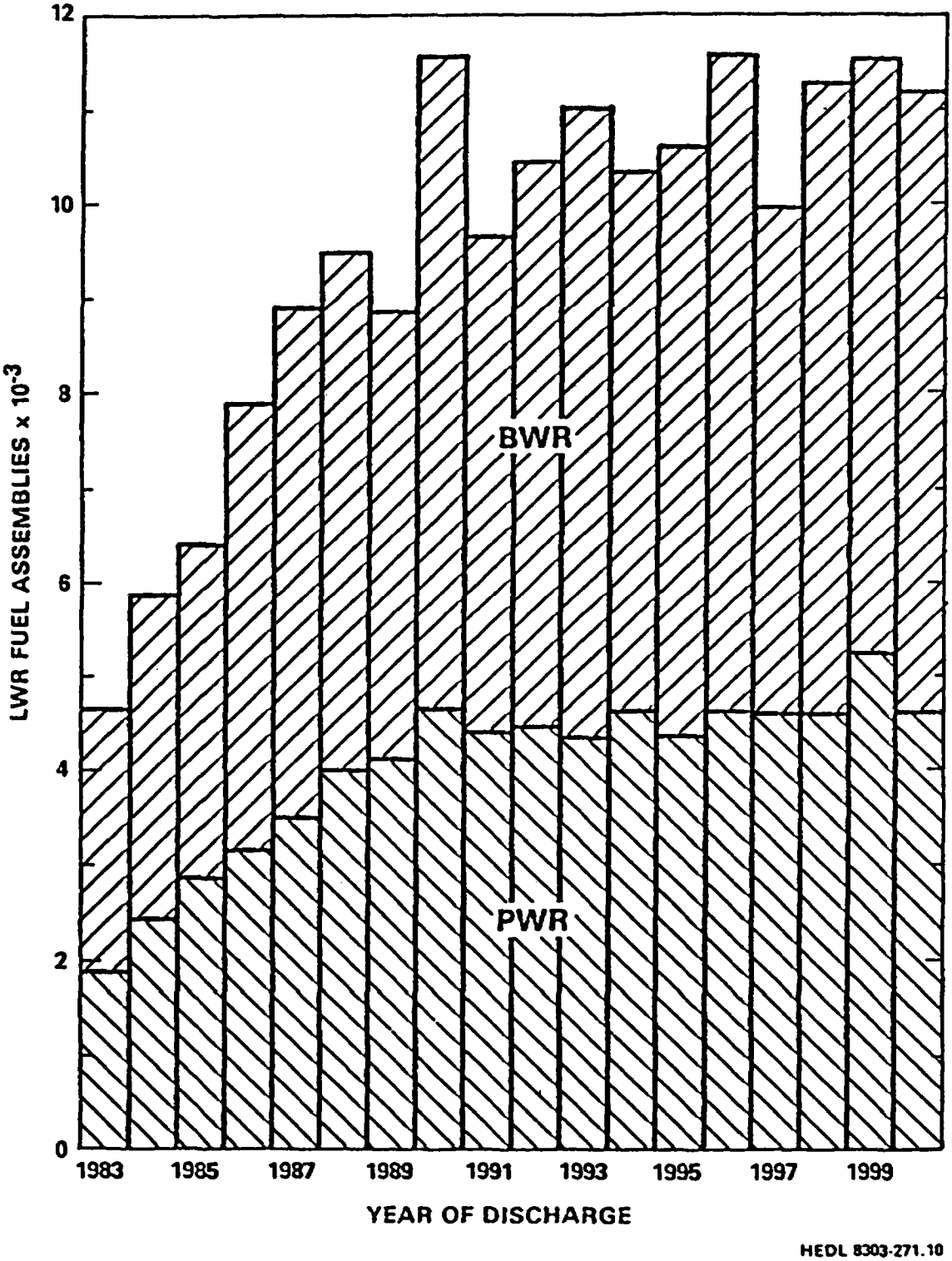


FIGURE 6. Projected Annual Discharges of Spent LWR Fuel Assemblies.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, October, 1983.

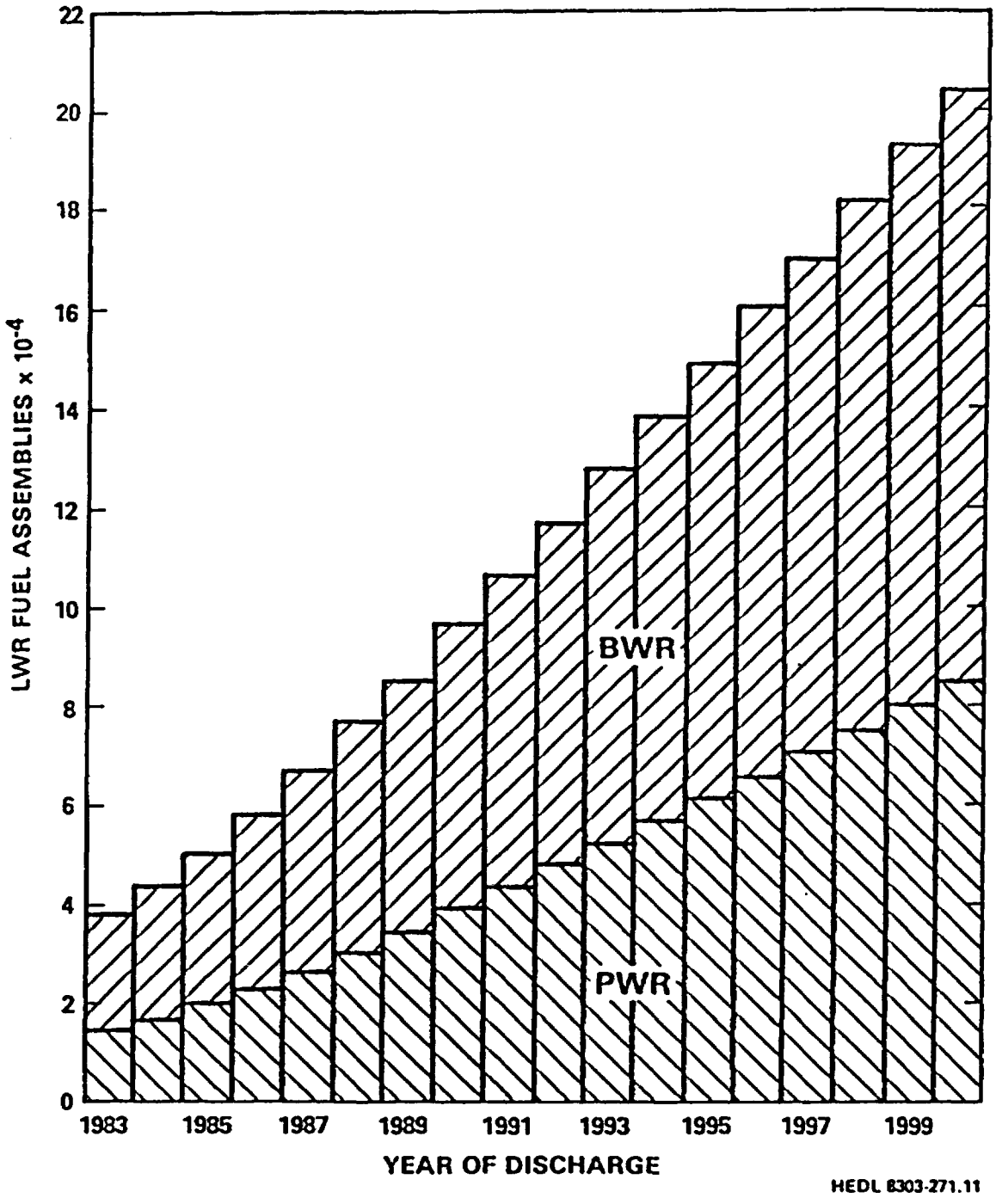


FIGURE 7. Cumulative Discharges of Spent LWR Fuel Assemblies.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, October, 1983.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 4.6 Summary of Radioactivity (curies/MTIHM) of BWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.10 Summary of Radioactivity (curies/MTIHM) of PWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.14 Radioactivity (in curies/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.14 (cont.) Radioactivity (in curies/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.17 Radioactivity (curies/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.17 (cont.) Radioactivity (curies/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

TABLE 6

ACTIVITY OF SELECTED RADIONUCLIDES IN A PWR FUEL ASSEMBLY IRRADIATED TO AN AVERAGE BURNUP OF 33,000 MWd/MTU*

Radionuclide	Activity (curies)						
	Discharge	1 yr	10 yr	100 yr	300 yr	1000 yr	10,000 yr
Americium-241	5.015E 01	1.397E-02	7.740E-02	1.731E-03	1.269E-03	4.139E-02	4.734E-03
Americium-243	7.621E 00	7.631E 00	7.625E 00	7.563E 00	7.427E 00	6.971E 00	3.084E 00
Carbon-14	6.853E-01	6.852E-01	6.844E-01	6.770E-01	6.608E-01	6.072E-01	2.044E-01
Cesium-135	1.711E-01	1.714E-01	1.714E-01	1.714E-01	1.714E-01	1.714E-01	1.709E-01
Cesium-137	4.786E 04	4.677E 04	3.801E 04	4.785E 03	4.783E 01	4.777E-06	0.0
Neptunium-237	1.403E-01	1.436E-01	1.450E-01	1.914E-01	2.883E-01	4.613E-01	5.435E-01
Plutonium-238	9.832E 02	1.054E 03	1.001E 03	4.970E 02	1.052E 02	4.867E-01	6.144E-20
Plutonium-239	1.400E 02	1.424E 02	1.424E 02	1.421E 02	1.413E 02	1.387E 02	1.084E 02
Plutonium-240	2.358E 02	2.358E 02	2.361E 02	2.352E 02	2.305E 02	2.145E 02	8.525E 01
Plutonium-242	8.294E-01	8.295E-01	8.295E-01	8.294E-01	8.291E-01	8.281E-01	8.147E-01
Radium-226	5.867E-09	1.104E-08	1.457E-07	1.145E-05	1.142E-04	1.336E-03	5.733E-02
Strontium-90	3.493E 04	3.408E 04	2.729E 04	2.964E 03	2.138E 01	6.780E-07	0.0
Technicium-99	6.095E 00	6.124E 00	6.124E 00	6.122E 00	6.118E 00	6.104E 00	5.927E 00
Tin-126	3.577E-01	3.577E-01	3.577E-01	3.575E-01	3.570E-01	3.553E-01	3.338E-01

*The fuel assembly initially contained 0.461 MTU, enriched to 3.2% in ^{235}U. (Adapted from Reference 3.)

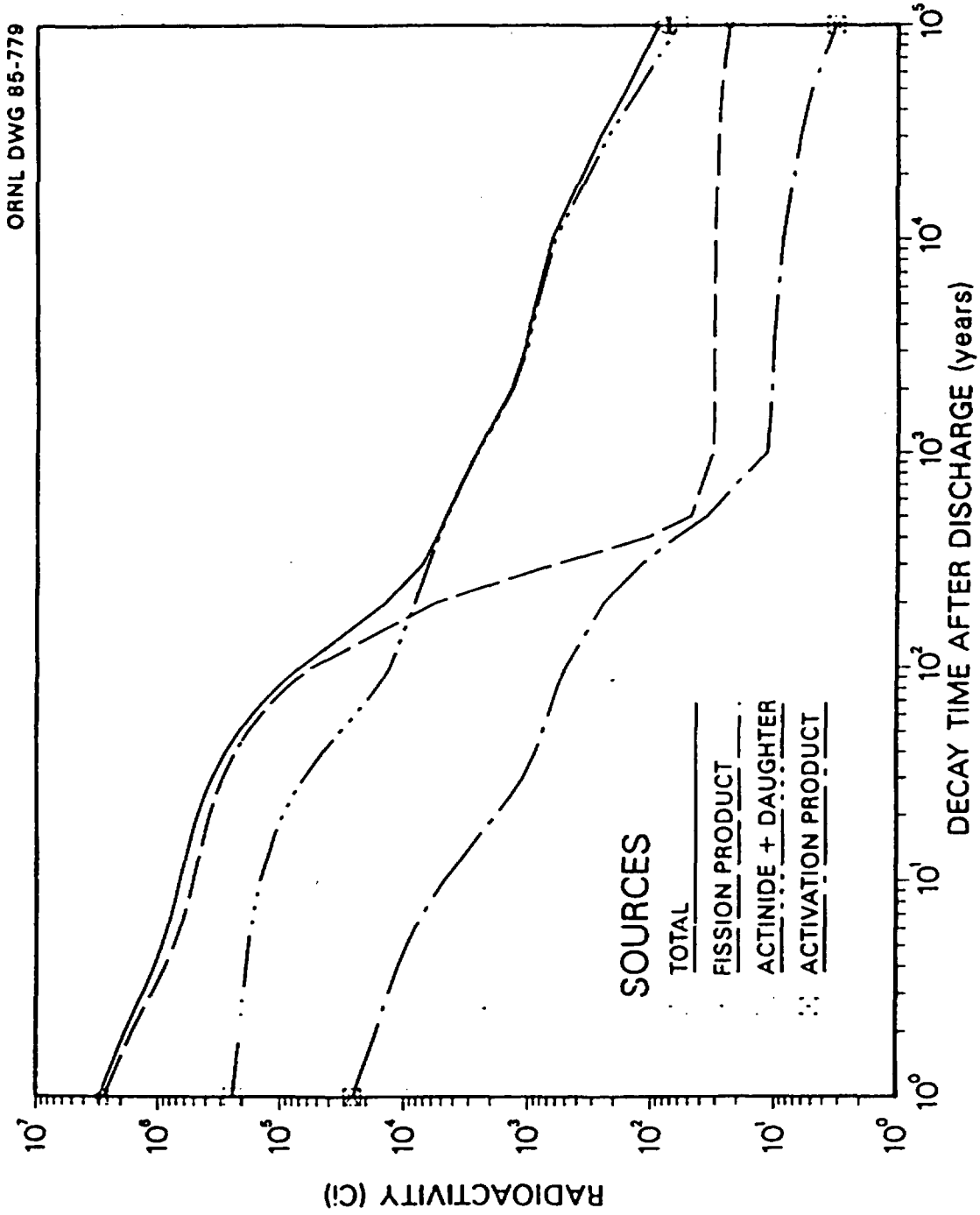


Fig. 3.1. Radioactivity produced by 1 metric ton of initial heavy metal: PWR; 60,000 MWd.

Initial enrichment ~4.1–4.15% U-235 for PWR 60,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

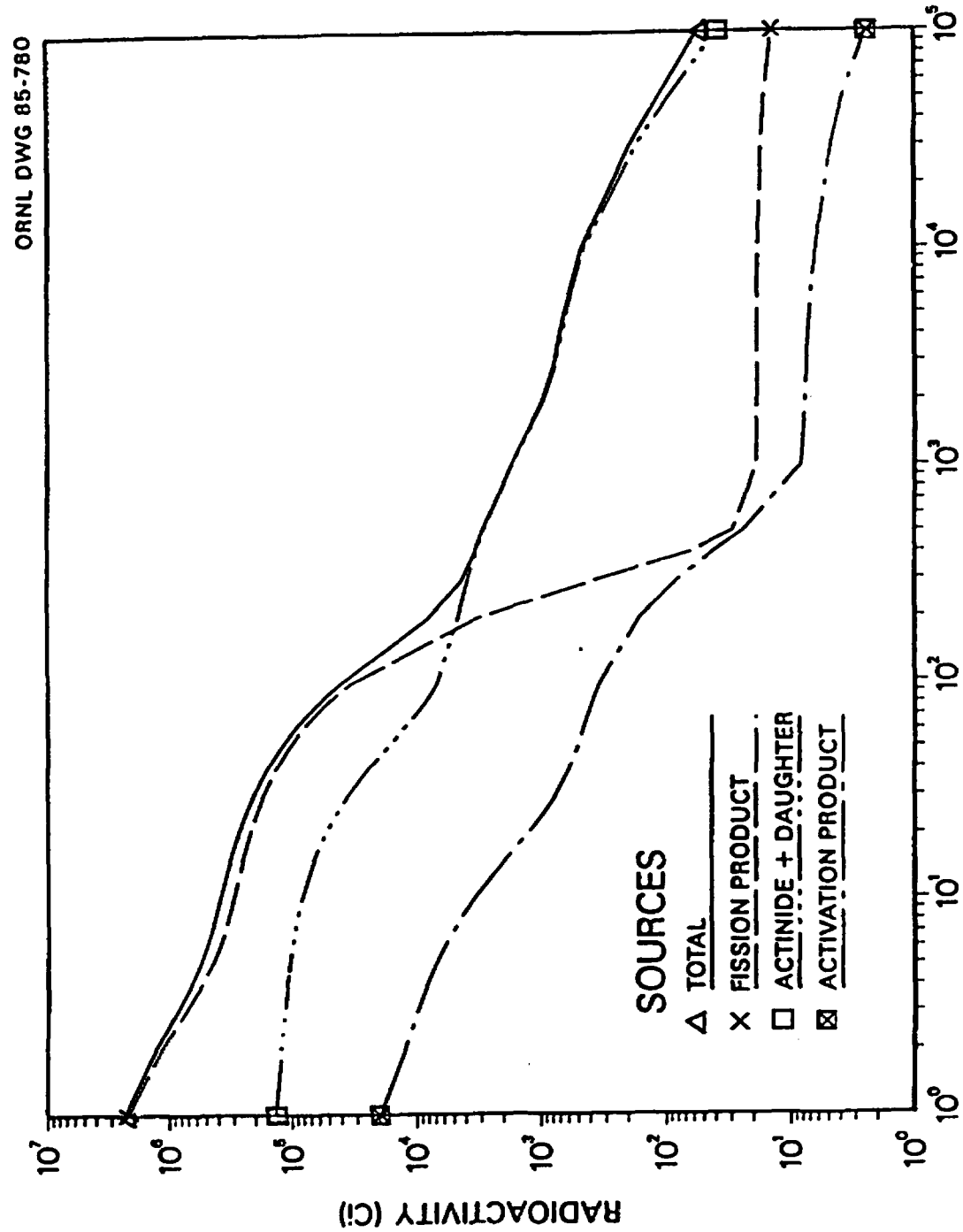


Fig. 3.2. Radioactivity produced by 1 metric ton of initial heavy metal: PWR; 33,000 MWd.

Initial enrichment ~3.2% U-235 for PWR 33,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

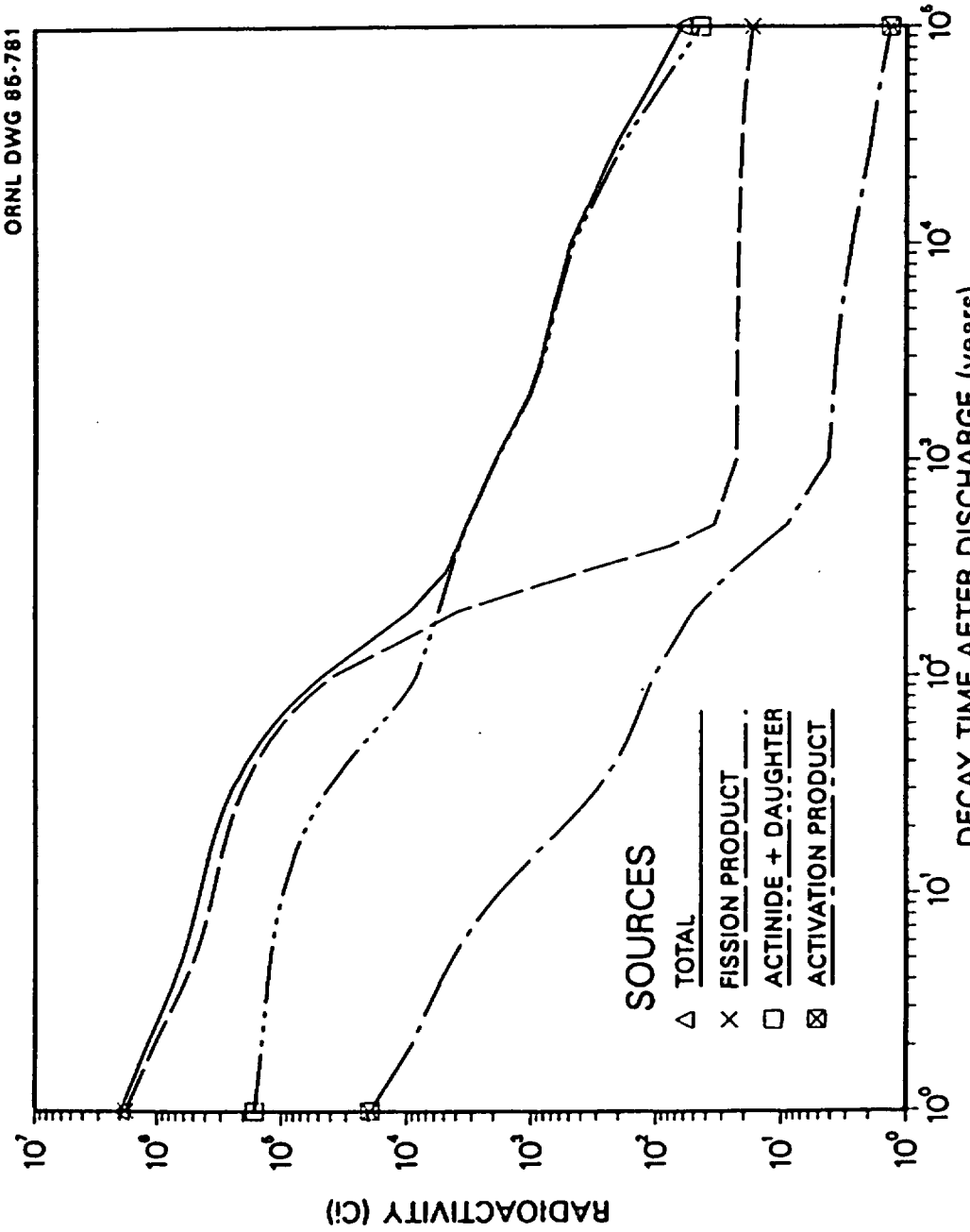


Fig. 3.3. Radioactivity produced by 1 metric ton of initial heavy metal: BWR; 40,000 MWd.

Initial enrichment ~3.5% U-235 for BWR 40,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

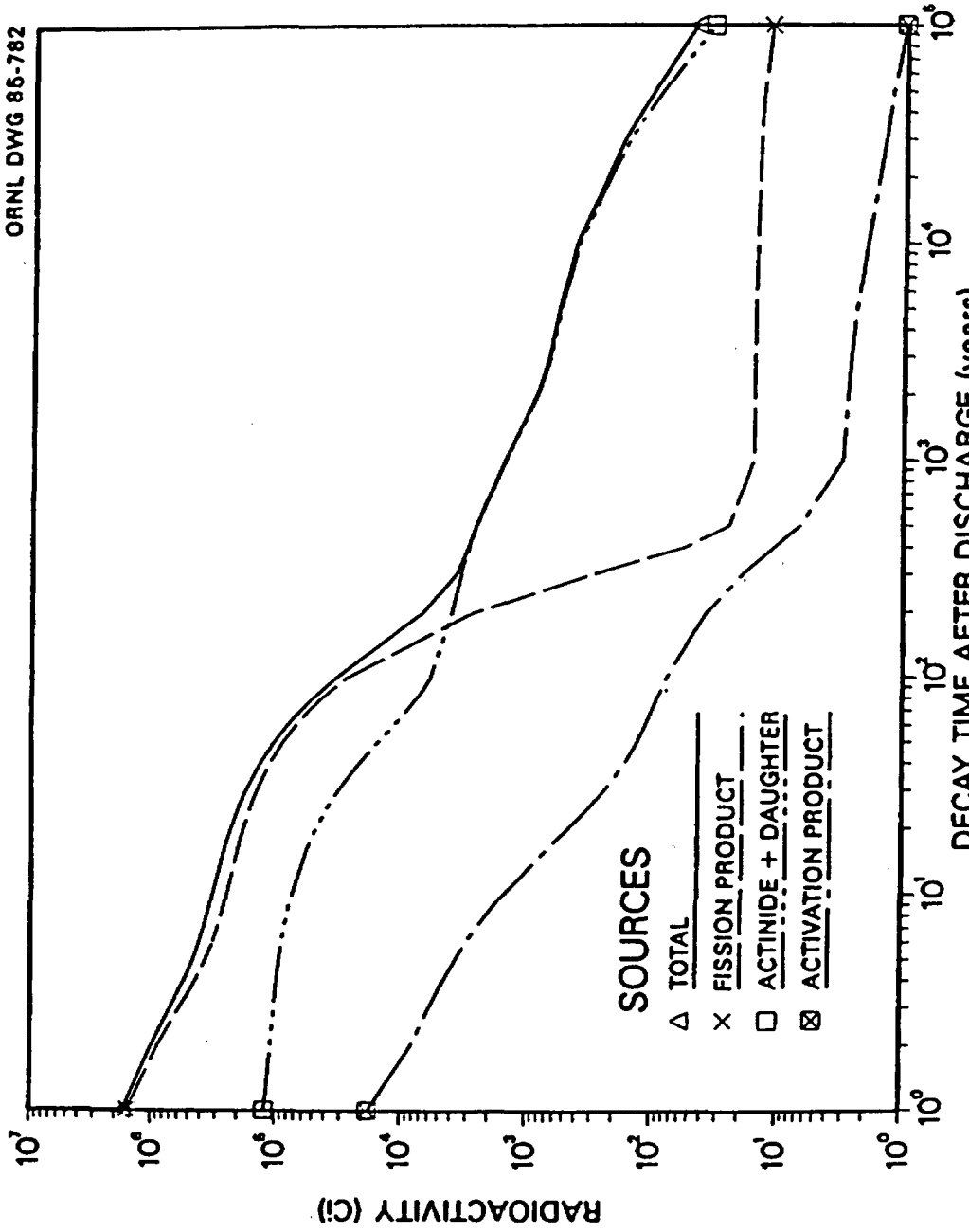


Fig. 3.4. Radioactivity produced by 1 metric ton of initial heavy metal: BWR; 27,500 MWd.

Initial enrichment ~2.75% U-235 for BWR 27,500 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

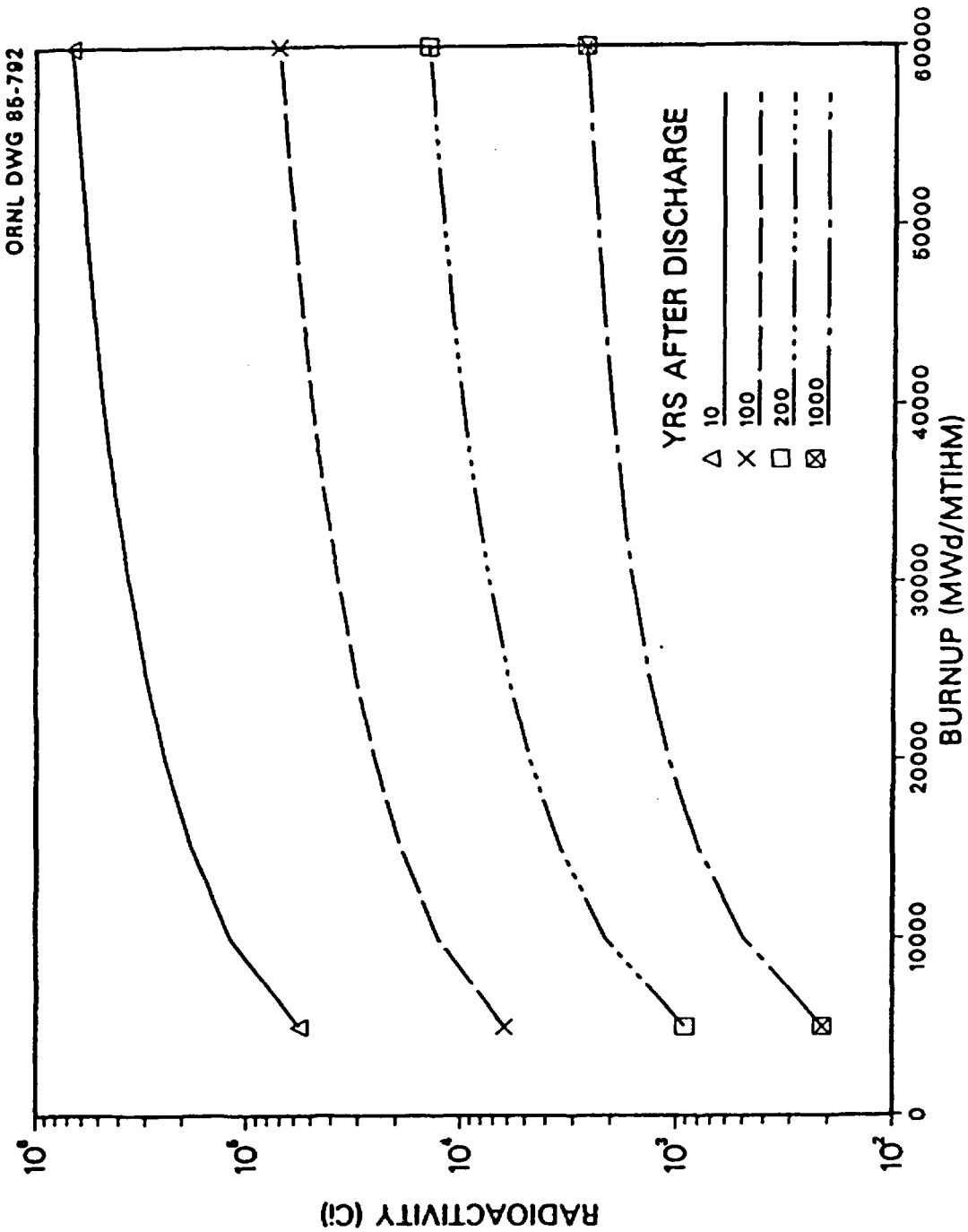


Fig. 3.14. Radioactivity produced by 1 metric ton of initial heavy metal for a PWR.

Initial enrichment ~4.15% U-235 for PWR 60,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

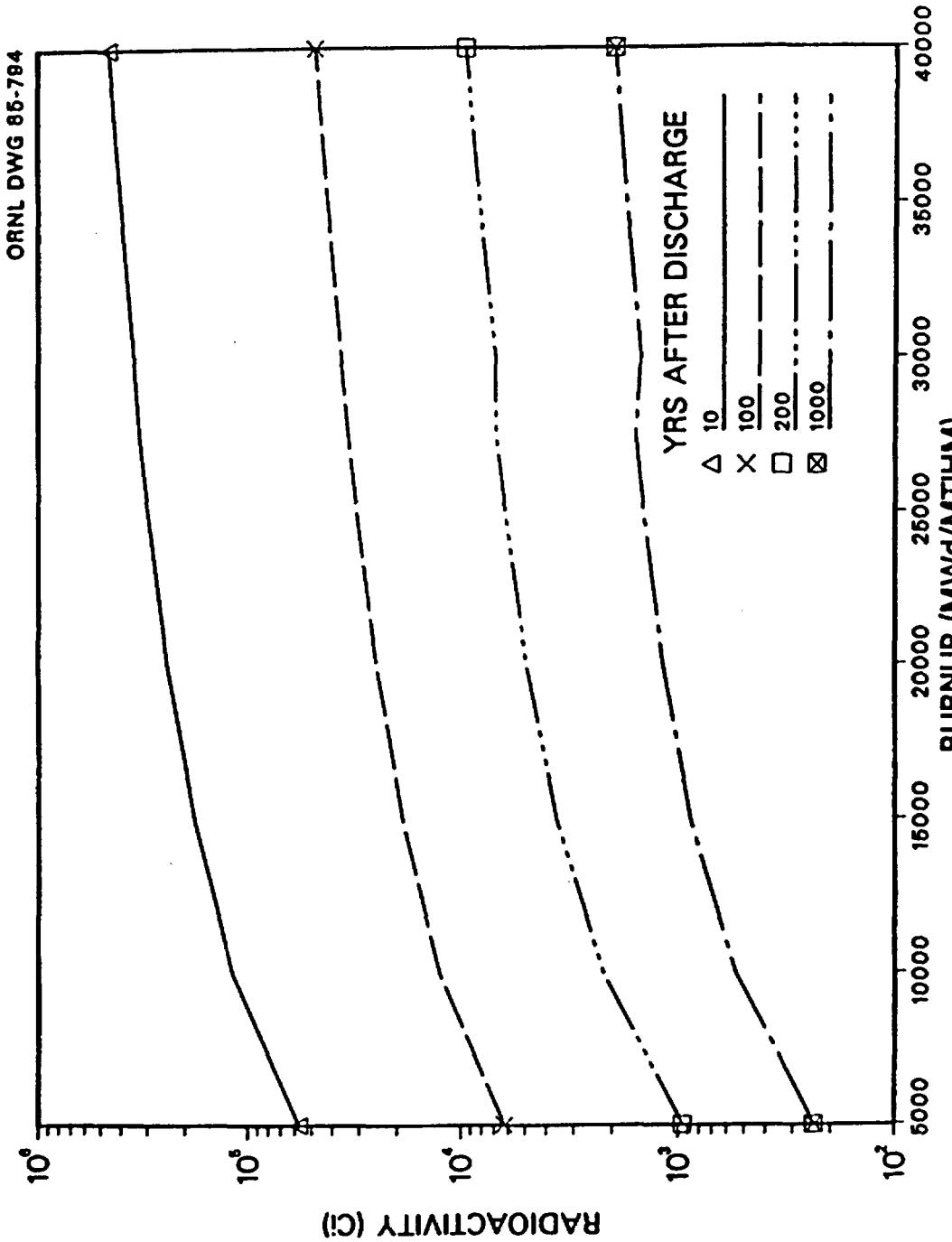


Fig. 3.16. Radioactivity produced by 1 metric ton of initial heavy metal for a BWR.

Initial enrichment ~3.5% U-235 for BWR 40,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.9. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 60,000-MWd/MTIHM PWR

(Includes all structural material)

	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 Kn-222. Others contributing 0.64 Ci each include: Pb-209, Bi-213, At-217, Fr-221, Ra-225, Ac-225, and Th-229.

Table 3.10. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 33,000-MWd/MTIHM PWR (Includes all structural material)

	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 3.11. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 40,000-MWd/MTIHM BWR (Includes all structural material)

Isotope[a]	Time since discharge (years)					
	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 3.12. Variation of radioactivity (Ci/MTIHM) for significant actinides as a function of time since discharge from a 27,500-MWd/MTIHM BWR (Includes all structural material)

	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.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 4.7 Summary of Thermal Output (watts/MTIHM) of BWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.11 Summary of Thermal Output (watts/MTIHM) of PWR spent fuel as a function of Burnup, Initial Enrichment, and Decay Time.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.15 Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.15 (cont.) Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.18 Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.18 (cont.) Decay heat (watts/MTIHM) by radionuclide (contributing ≥1% of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

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

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 4.20

Decay Heat Distribution Parameters for Greater Than 5-Year-Old Fuel in 1998.

		Reactor Type		
		BWR	PWR	Aggregate
Decay Heat, Watts	Minimum	11	47	11
	MEAN	804	1148	1022
	Maximum	1792	2586	2586
	Standard Deviation	1132	1335	1357

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

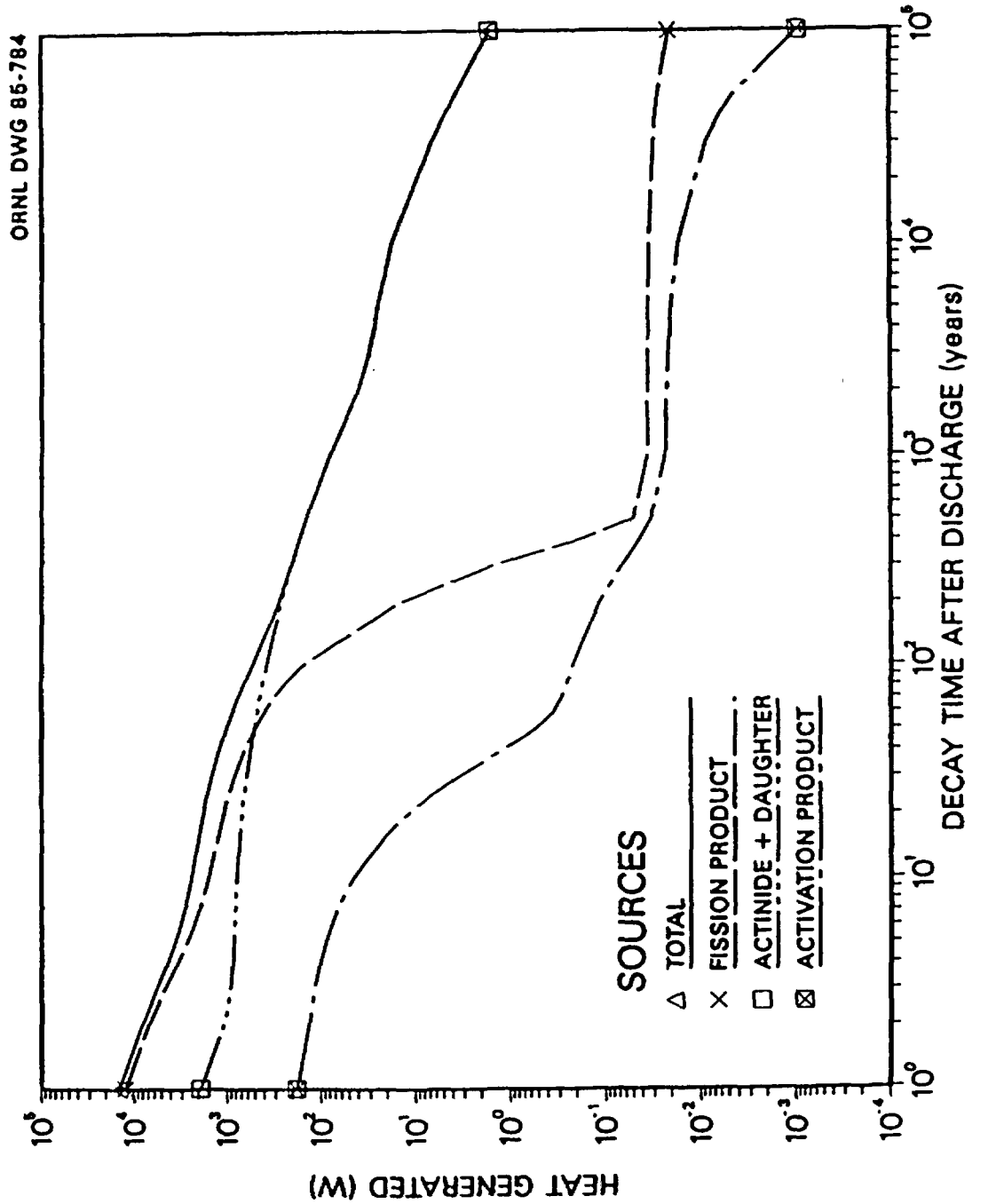


Fig. 3.6. Heat generated by 1 metric ton of initial heavy metal; PWR; 60,000 MWd.

Initial enrichment ~4.15% U-235 for PWR 60,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

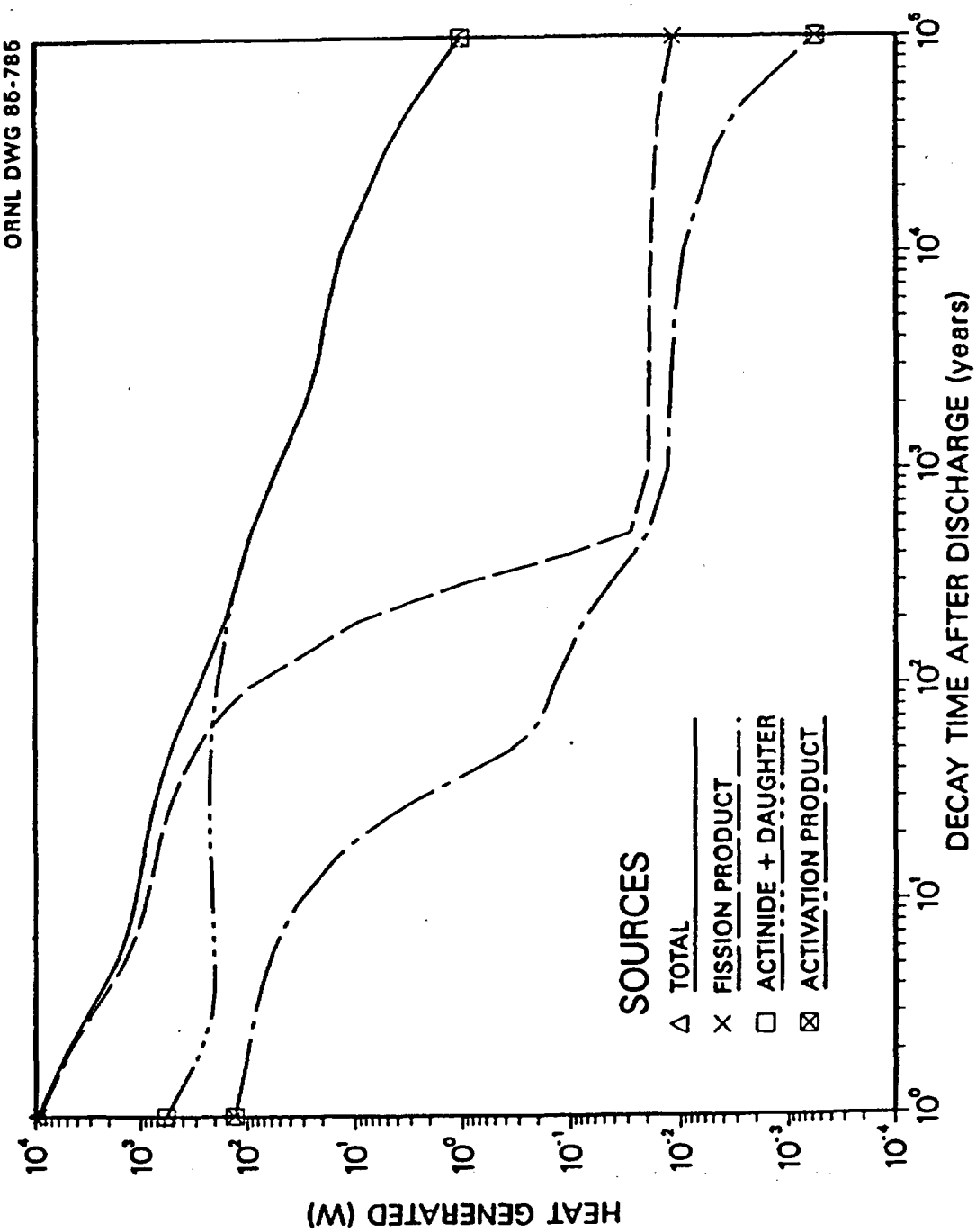


Fig. 3.7. Heat generated by 1 metric ton of initial heavy metal: PWR; 33,000 MWd.

Initial enrichment ~3.2% U-235 for PWR 33,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

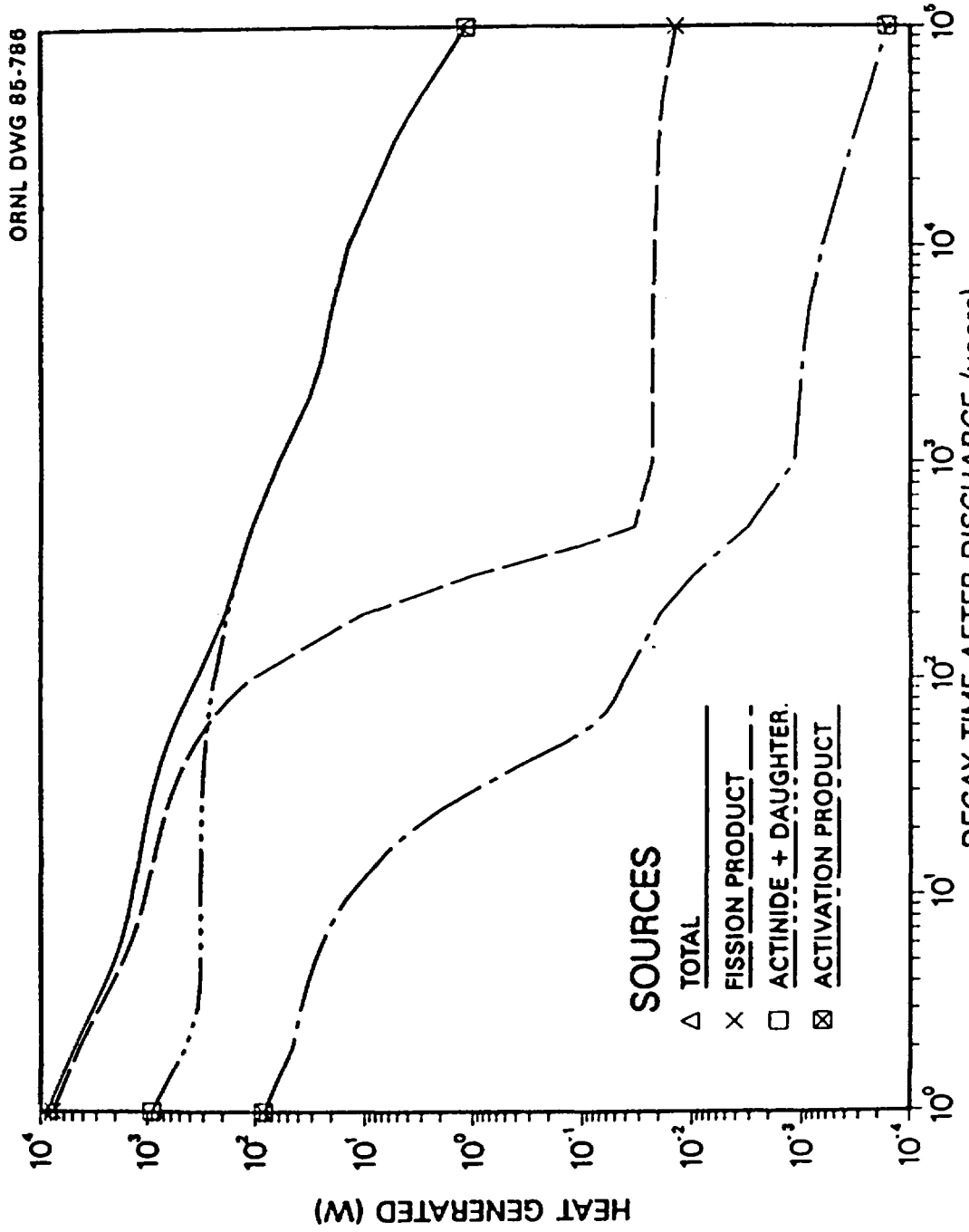


Fig. 3.8. Heat generated by 1 metric ton of initial heavy metal: BWR; 40,000 MWd.

Initial enrichment ~3.5% U-235 for BWR 40,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

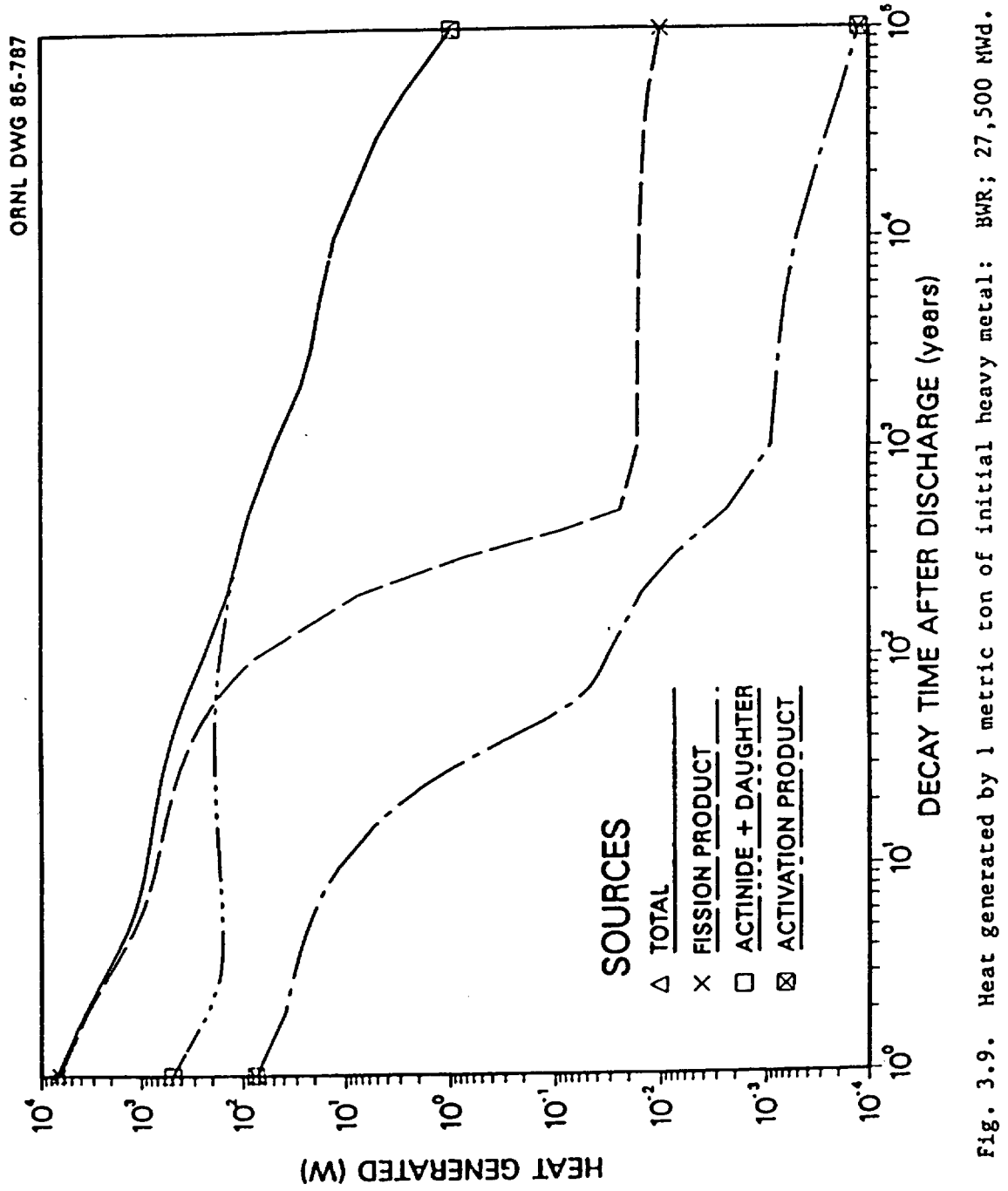


Fig. 3.9. Heat generated by 1 metric ton of initial heavy metal: BWR; 27,500 MWd.

Initial enrichment ~2.75% U-235 for BWR 27,500 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

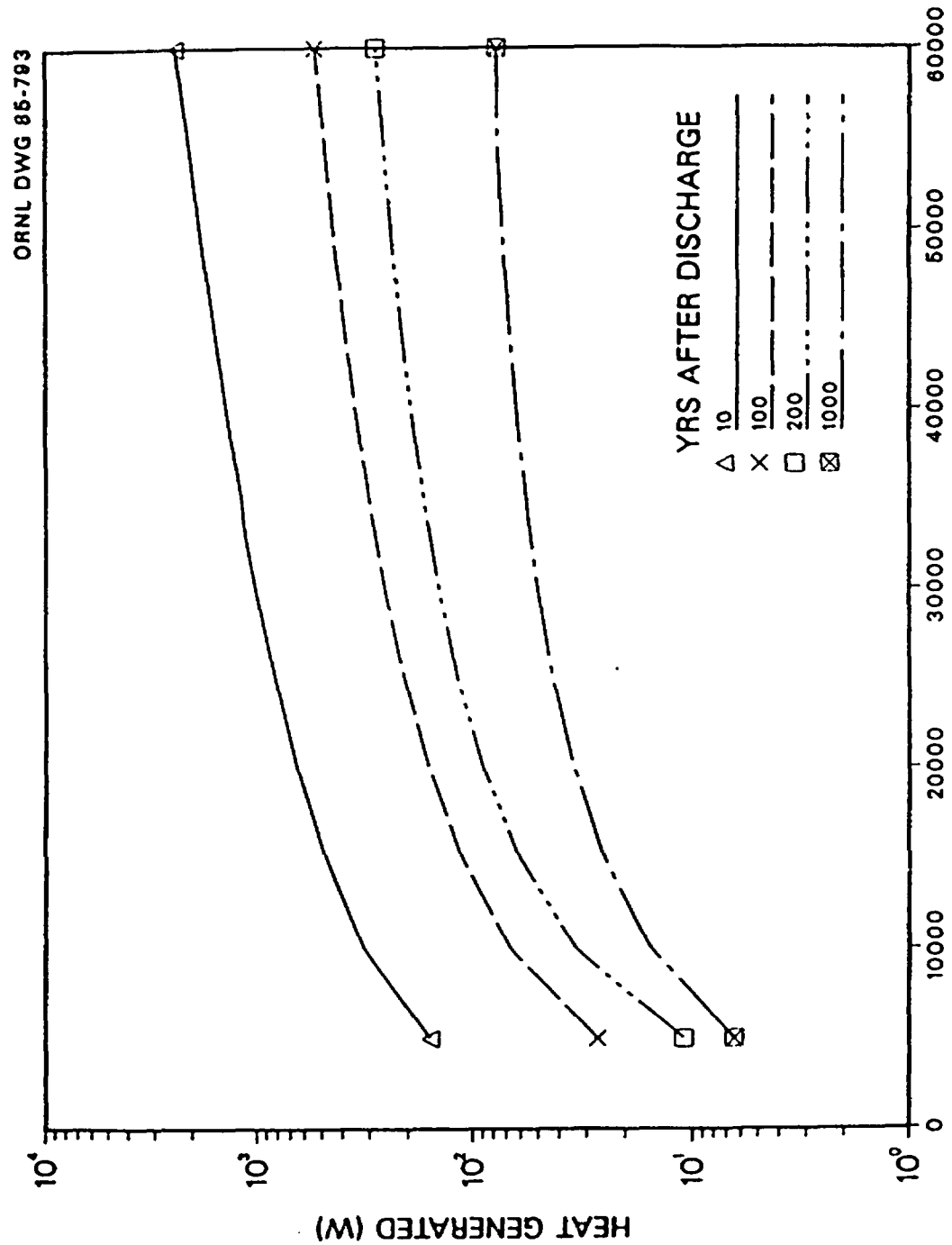


Fig. 3.15. Heat generated by 1 metric ton of initial heavy metal for a PWR.

Initial enrichment ~4.15% U-235 for PWR 60,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

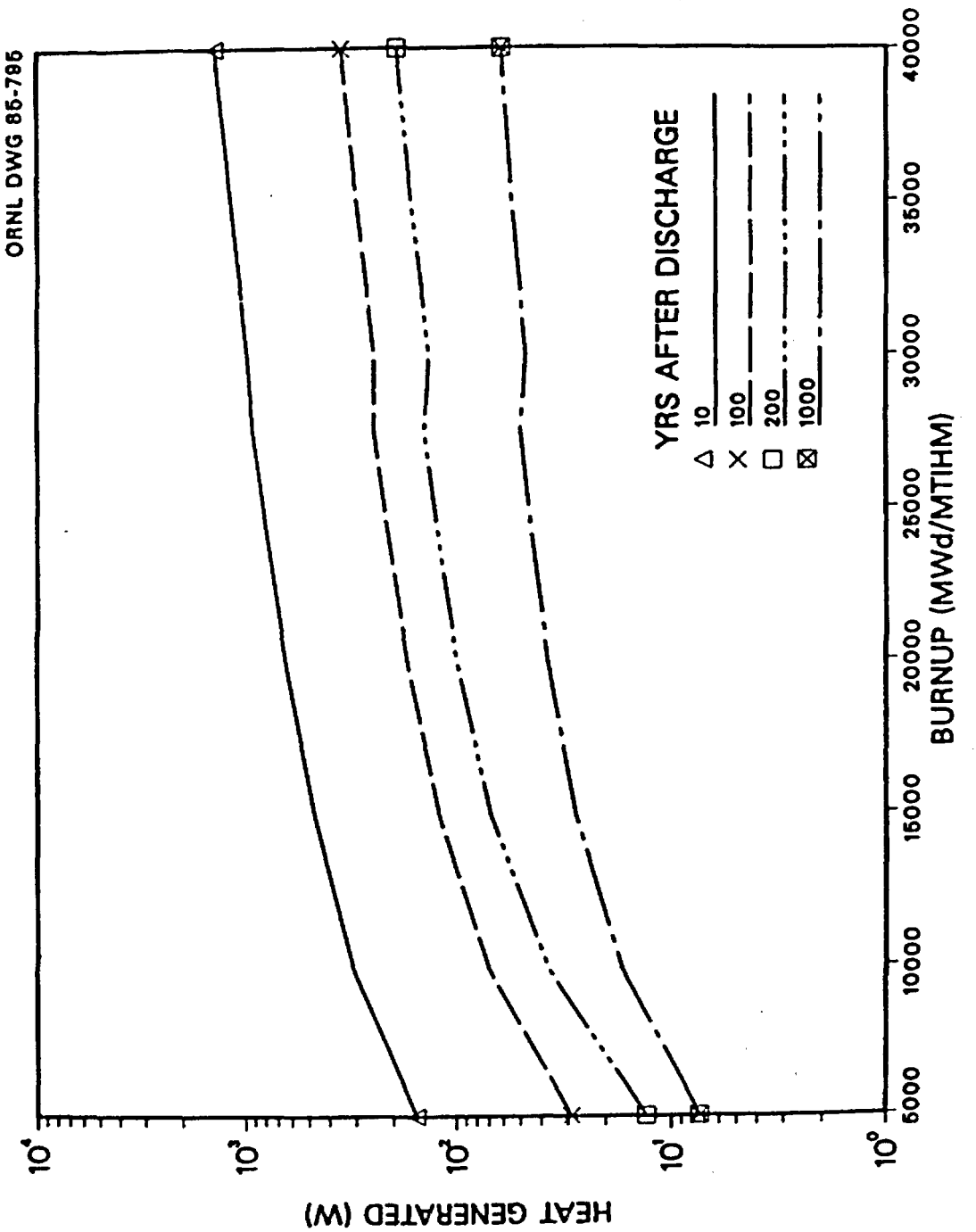


Fig. 3.17. Heat generated by 1 metric ton of initial heavy metal for a BWR.

Initial enrichment ~3.5% U-235 for BWR 40,000 MWd/MT data

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.13. 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)

Isotope[a]	Time since discharge (years)					
	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.14. 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)

Isotope[a]	Time since discharge (years)					
	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.15. 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)

	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.16. 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)

Isotope[a]	Time since discharge (years)					
	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

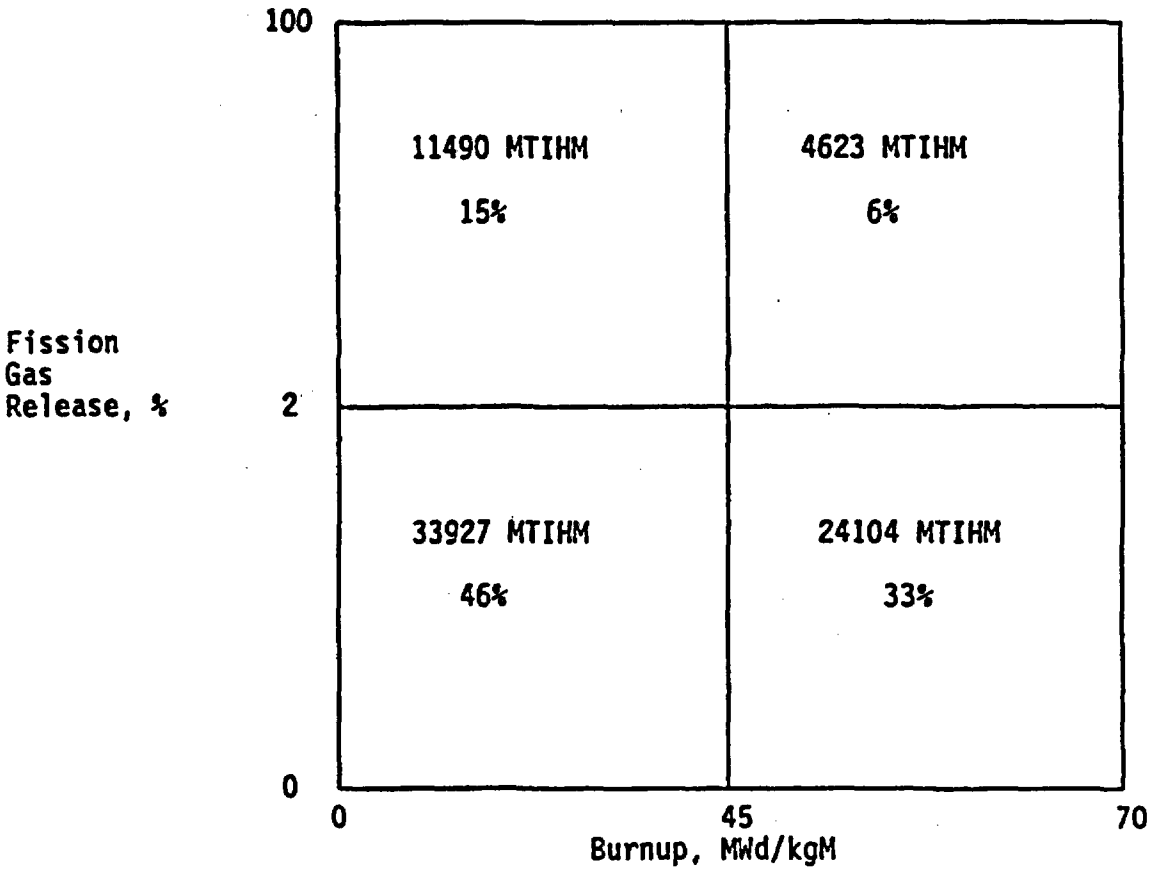


FIGURE 2.1. Burnup and Fission Gas Release Distribution of Spent Fuel Inventory Projected Through 2020.

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

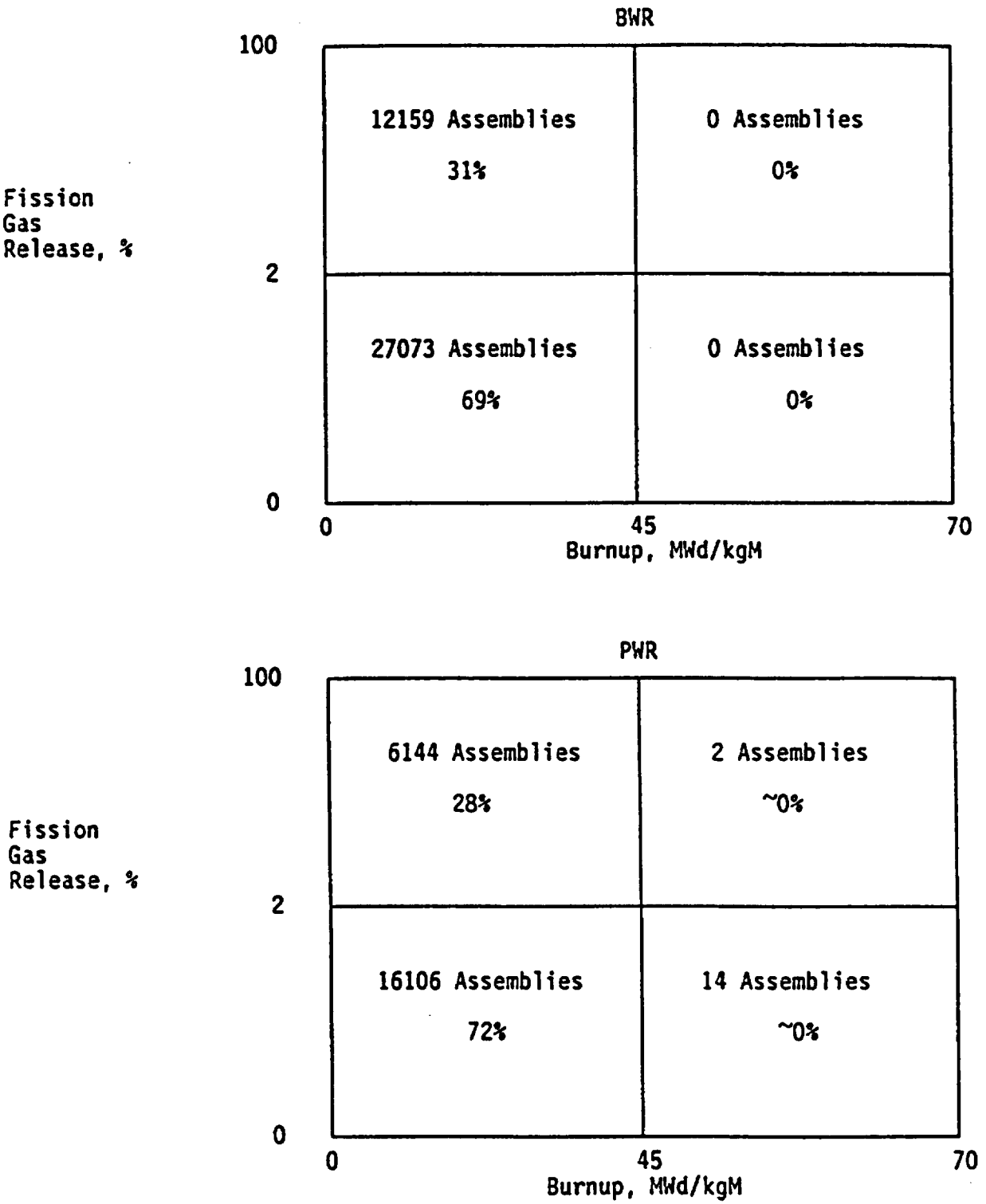


FIGURE 7.3. Distribution Summary for Spent Fuel Discharged Through 1987

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

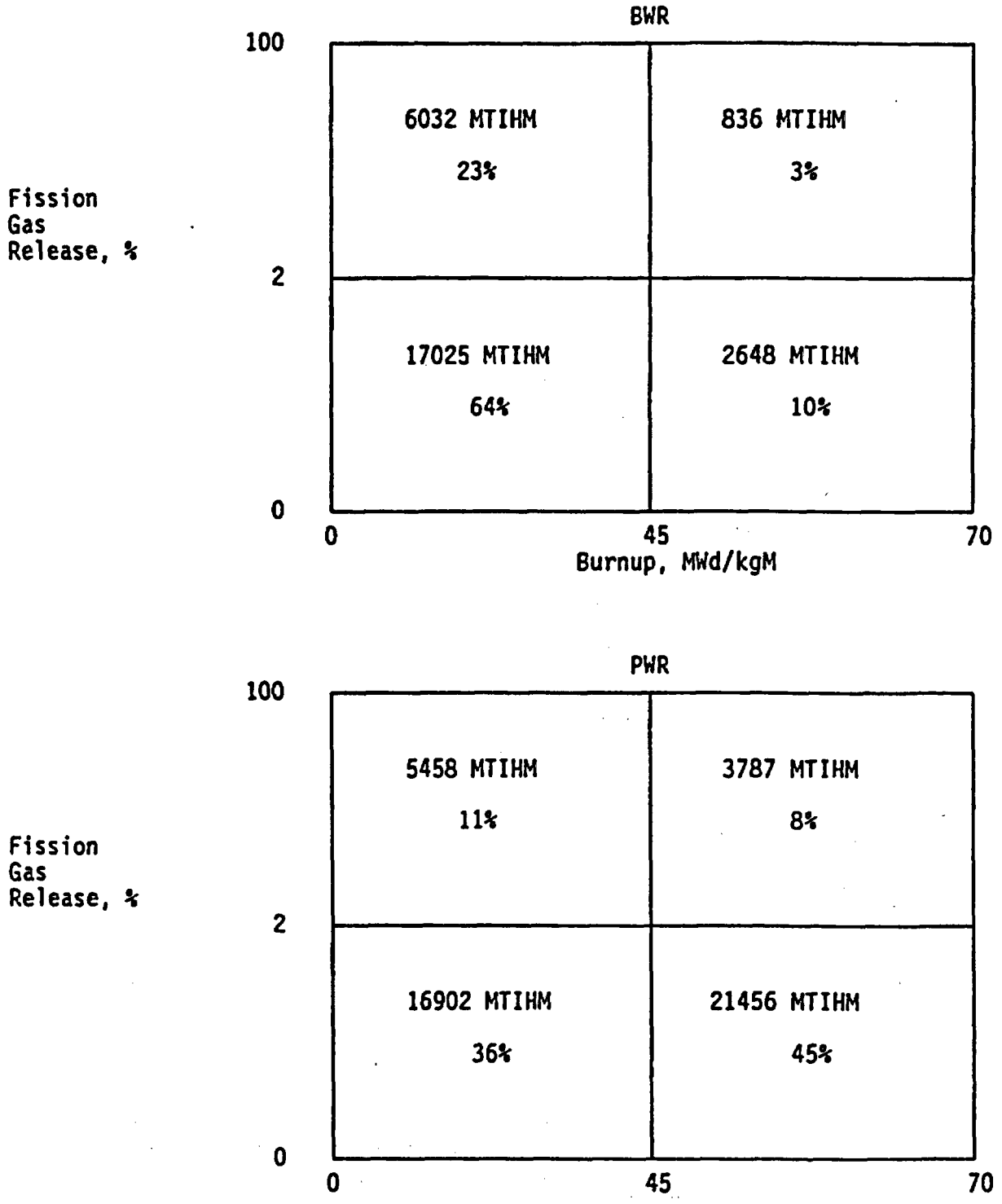


FIGURE 7.4. Distribution Summary for Projected Spent Fuel Inventory

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

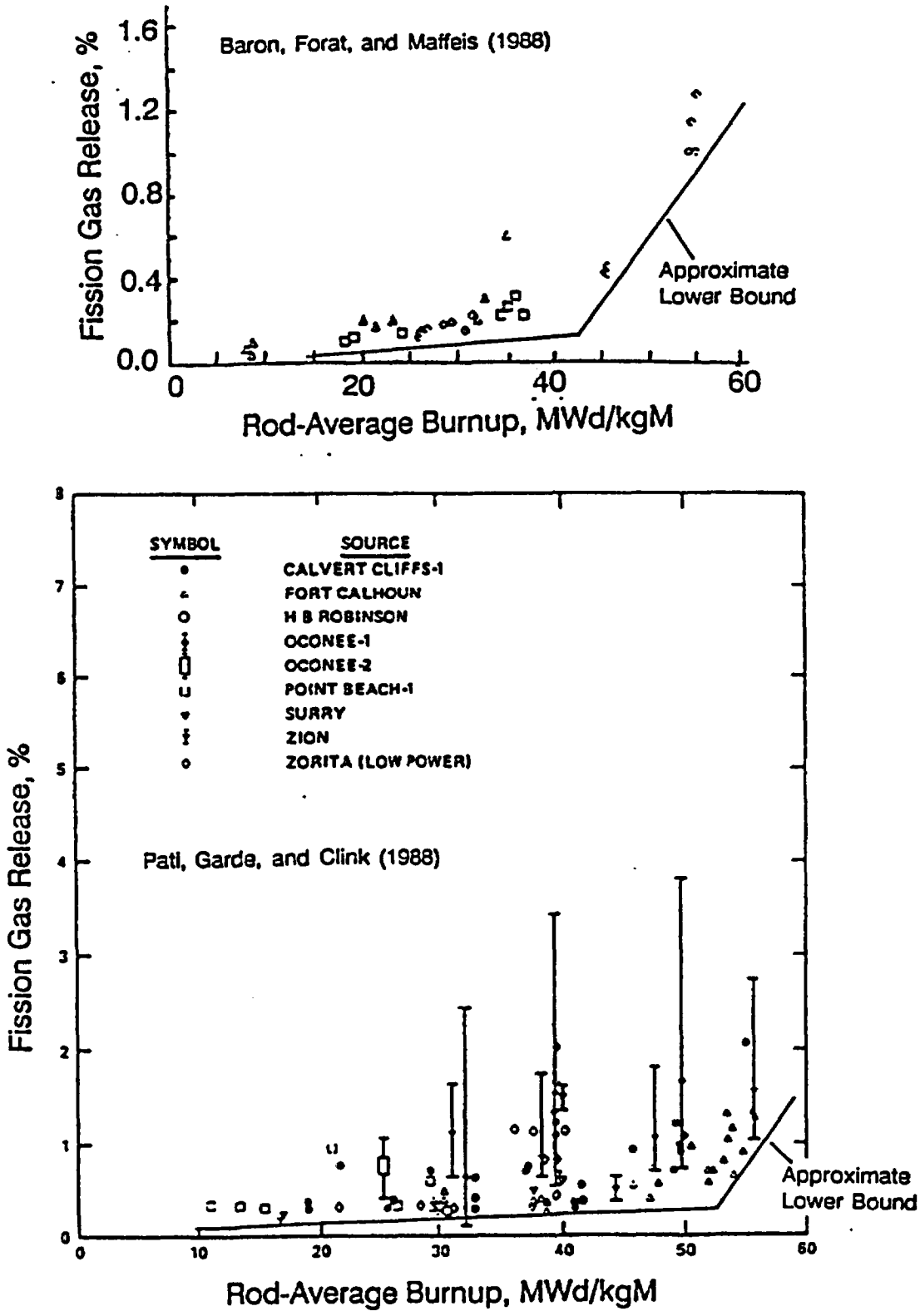


FIGURE 7.1. Example of Indicated Burnup Dependency of Fission Gas Release at Low Temperature

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

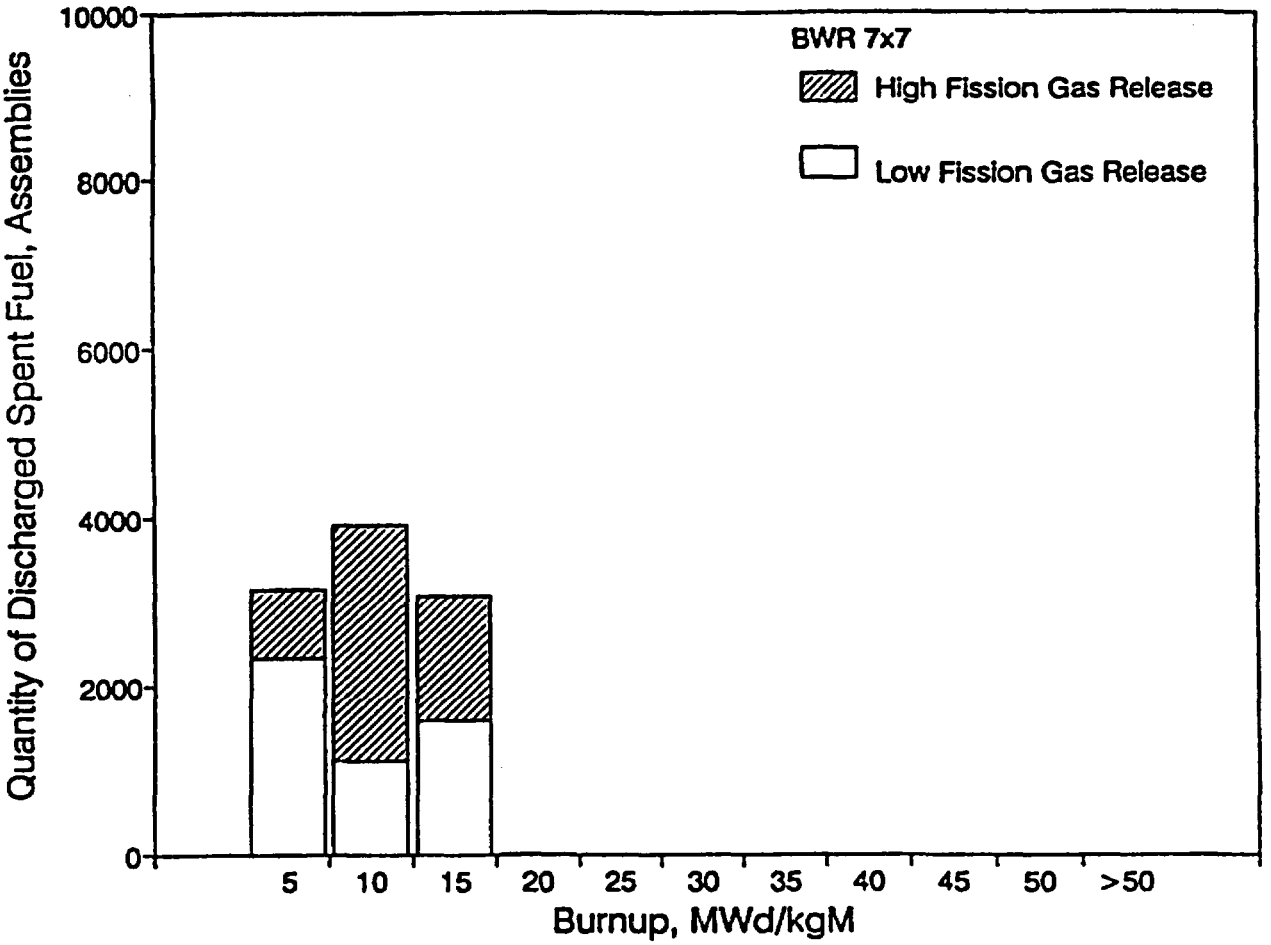


FIGURE 6.7. Fission Gas Release Distribution for 7x7 BWR Rod Group Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project,* Pacific Northwest Laboratory Report (Draft) 1991.

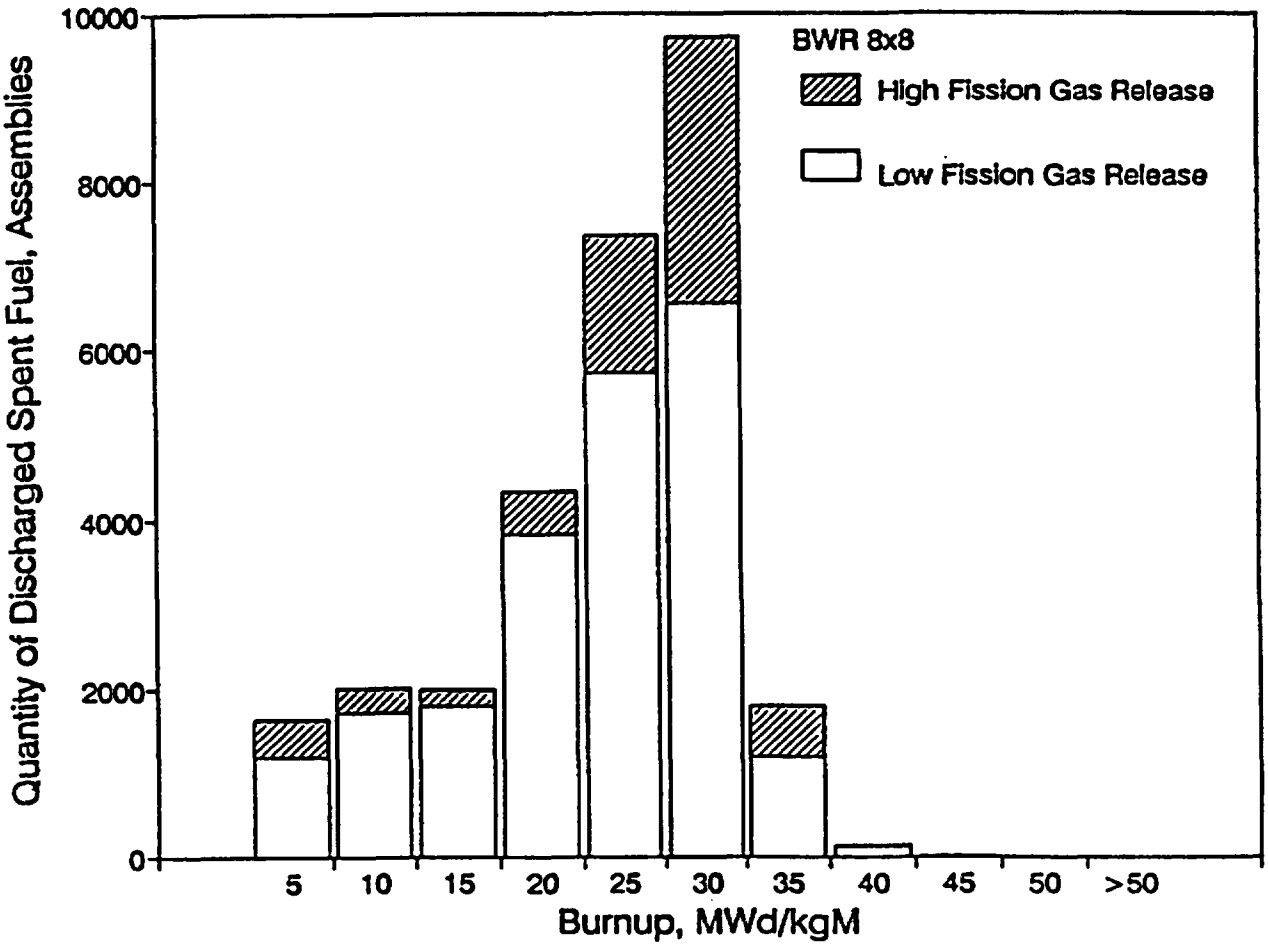


FIGURE 6.8. Fission Gas Release Distribution for 8x8 BWR Rod Group Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

No 9x9 Fuel Discharged Through 1987; No Data in the CDB.

FIGURE 6.9. Fission Gas Release Distribution for 9x9 BWR Rod Group Through 1987

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

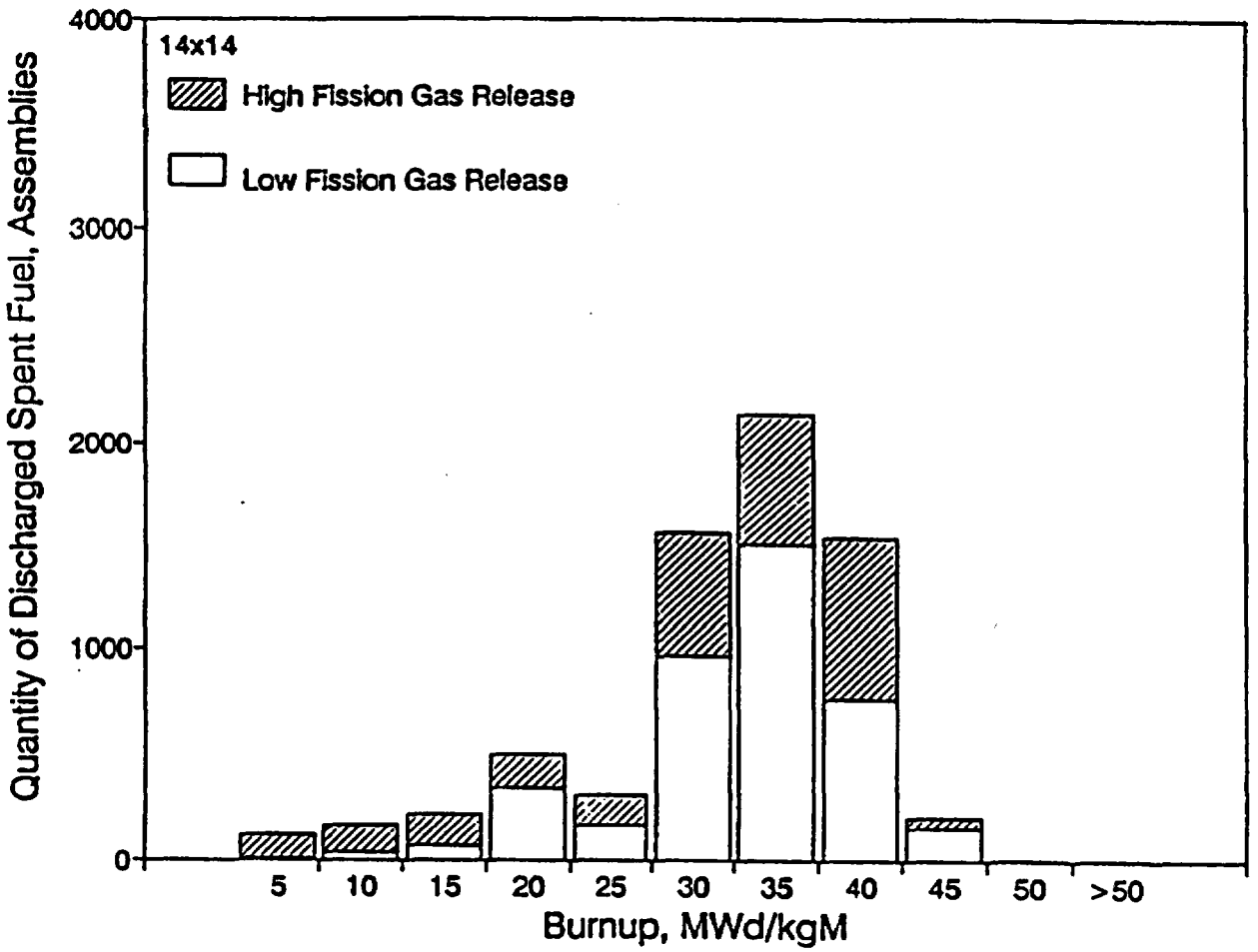


FIGURE 6.10. Fission Gas Release Distribution for 14x14 PWR Rod Group Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

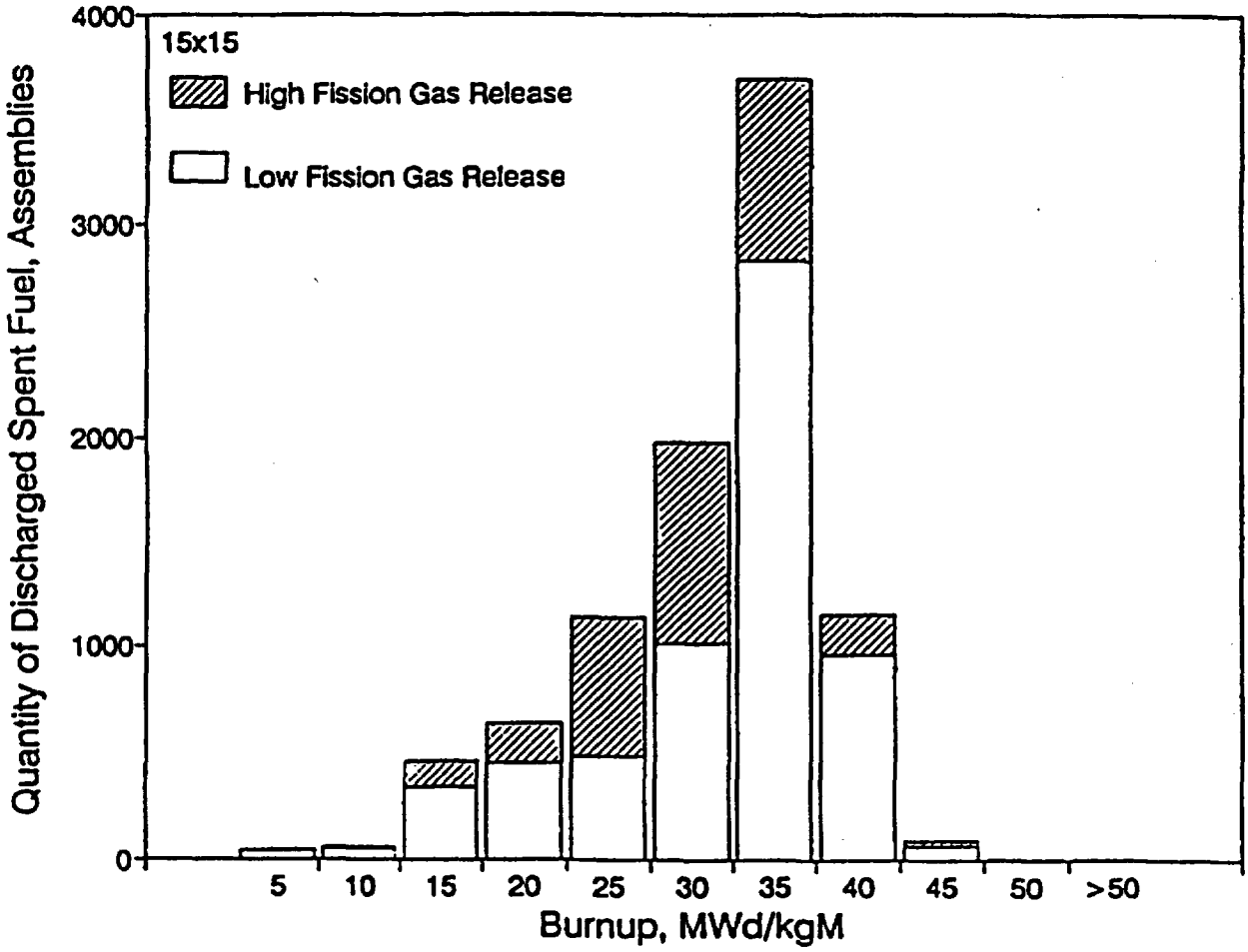


FIGURE 6.11. Fission Gas Release Distribution for 15x15 PWR Rod Group Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

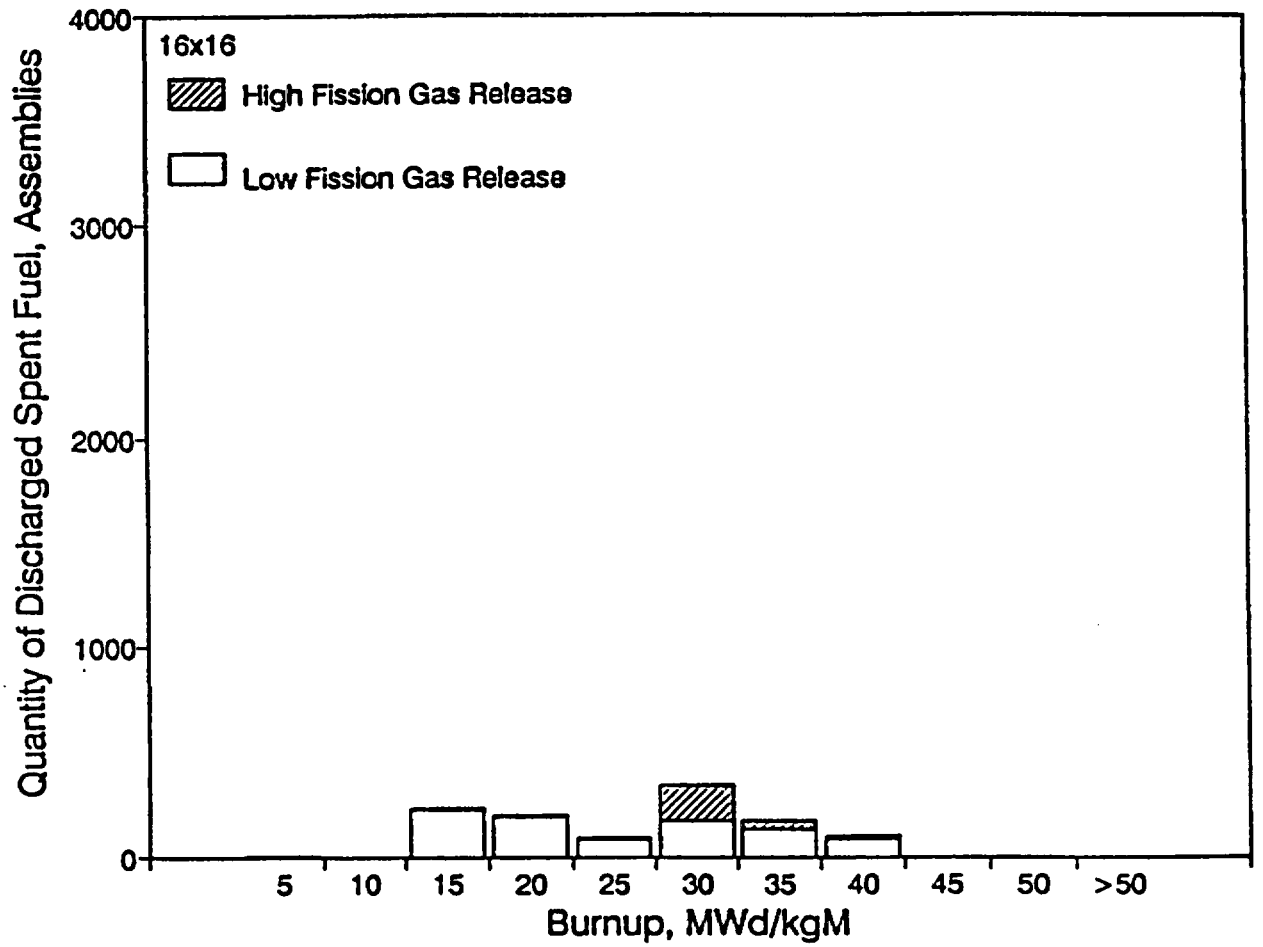


FIGURE 6.12. Fission Gas Release Distribution for 16x16 PWR Rod Group Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

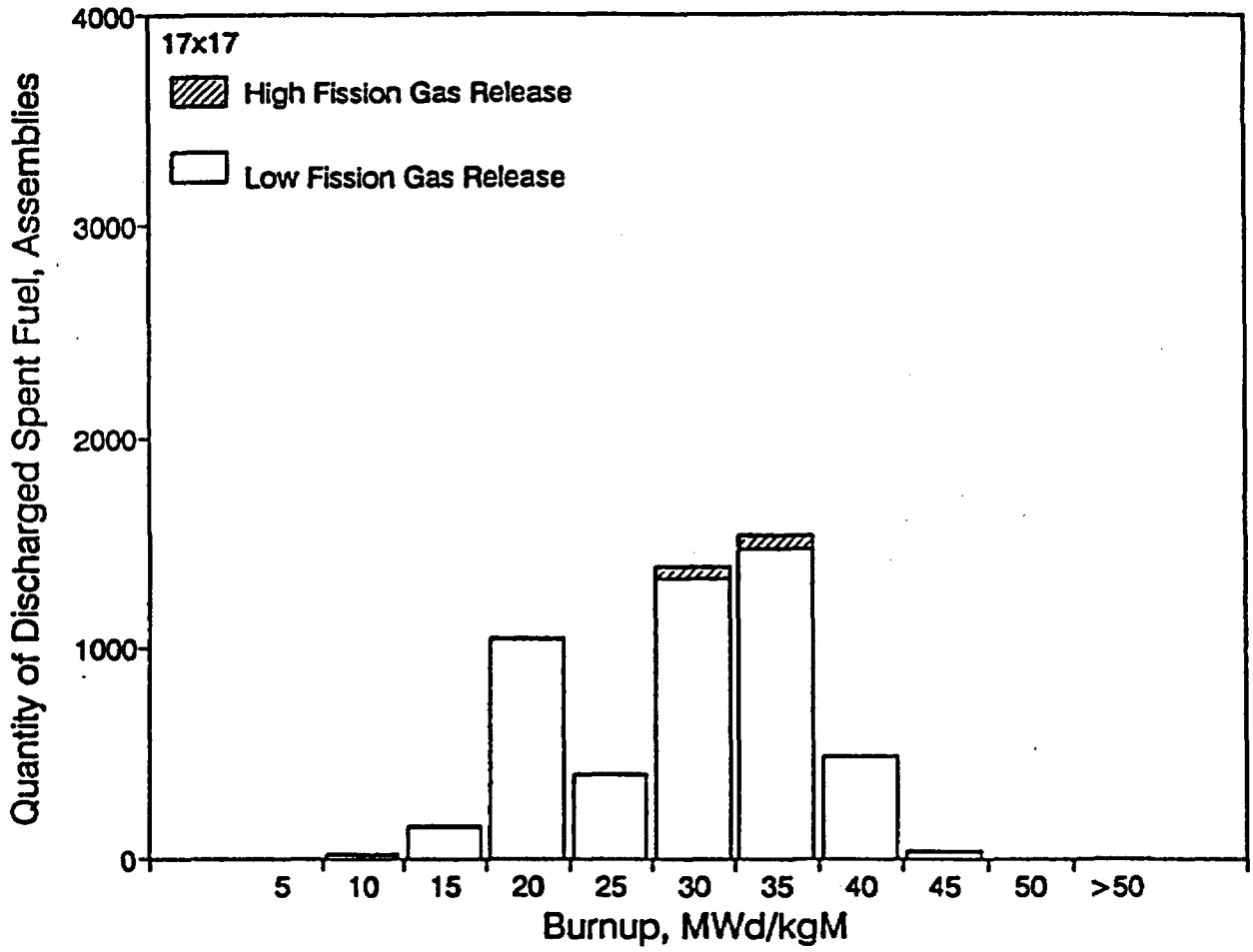


FIGURE 6.13. Fission Gas Release Distribution for 17x17 PWR Rod Group Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

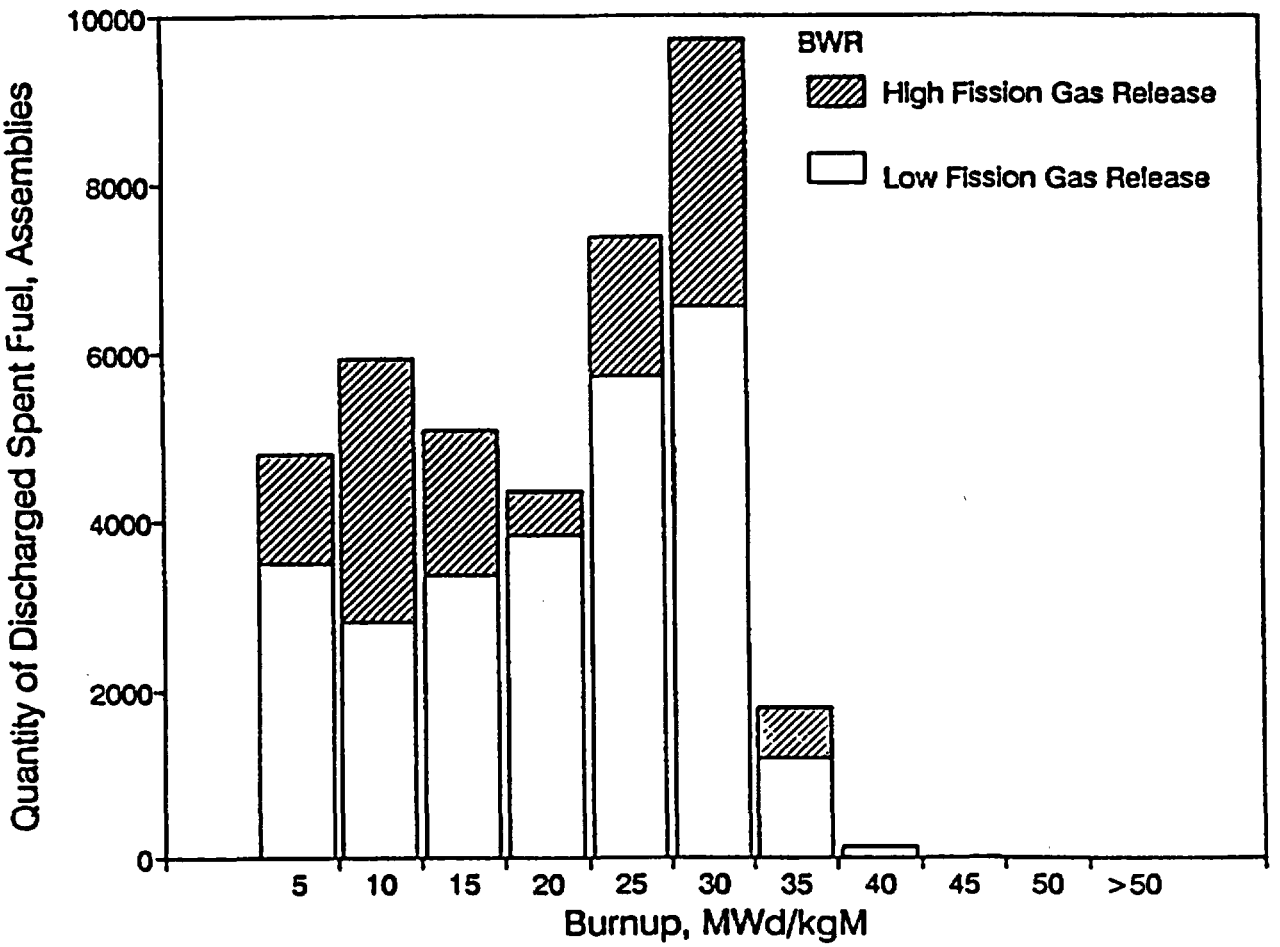


FIGURE 6.14. Fission Gas Release Distribution for BWR Spent Fuel Discharged Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

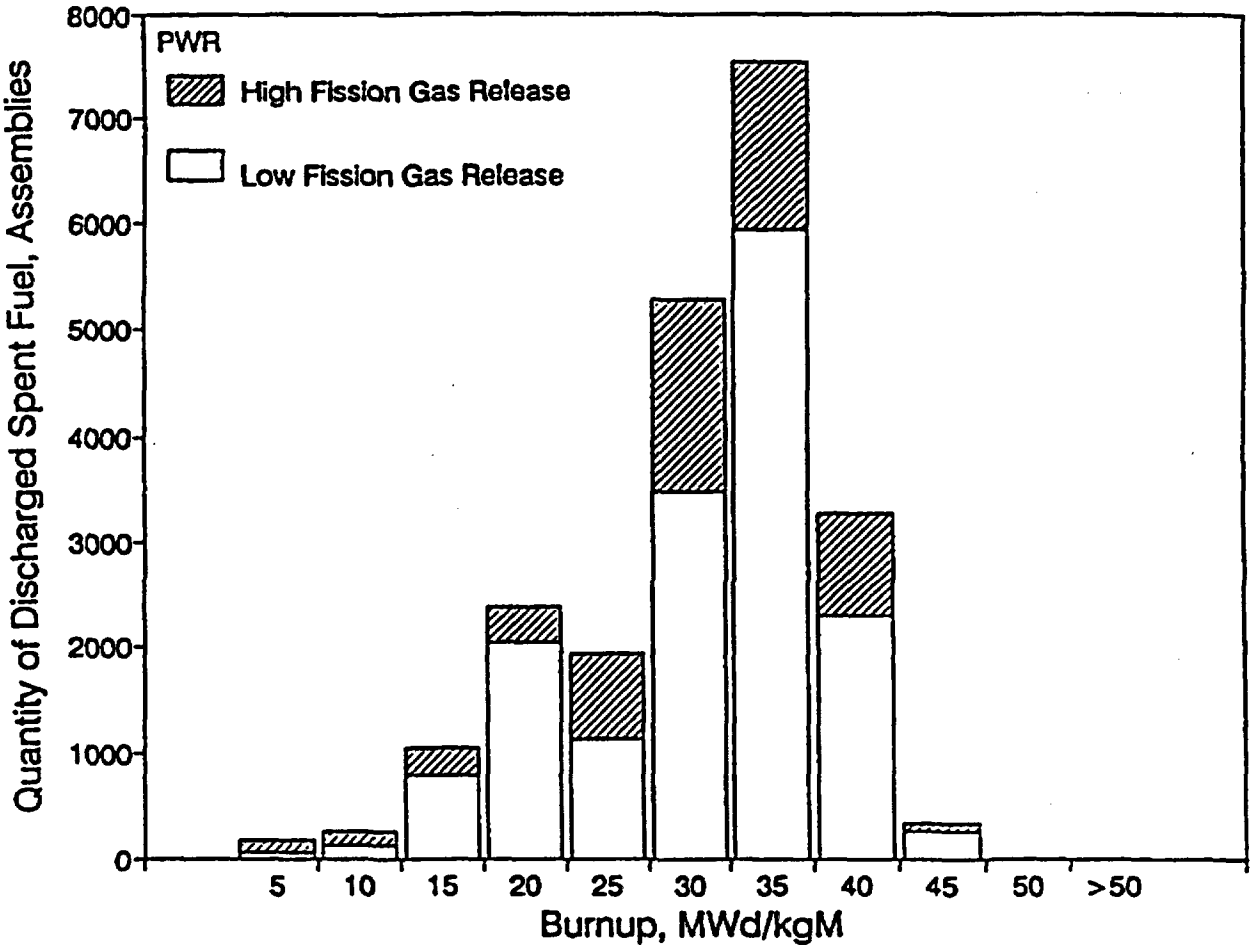


FIGURE 6.15. Fission Gas Release Distribution for PWR Spent Fuel Discharged Through 1987

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

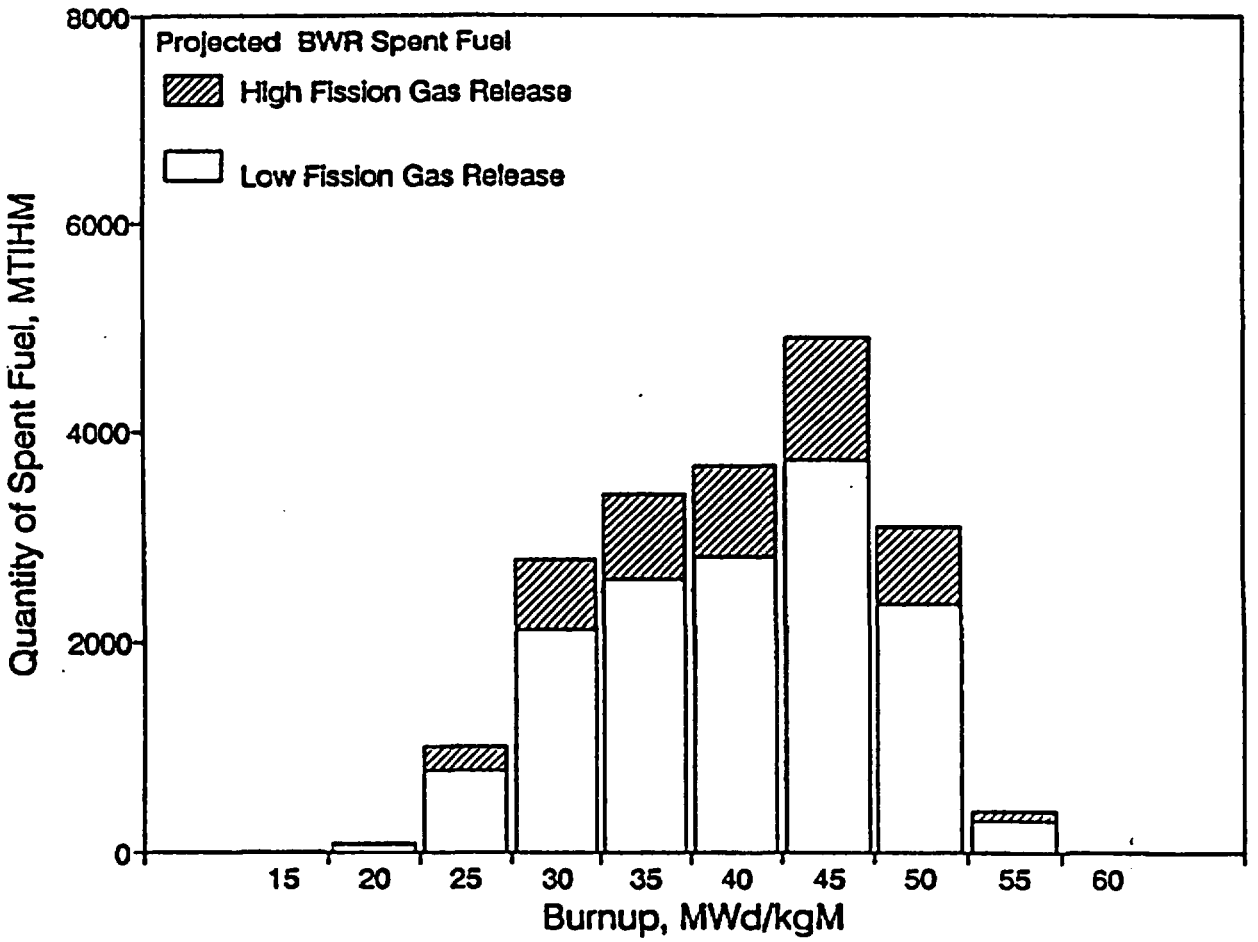


FIGURE 6.16. Projected Fission Gas Release Distribution for BWR Spent Fuel Discharged 1989 Through 2020

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project*, Pacific Northwest Laboratory Report (Draft) 1991.

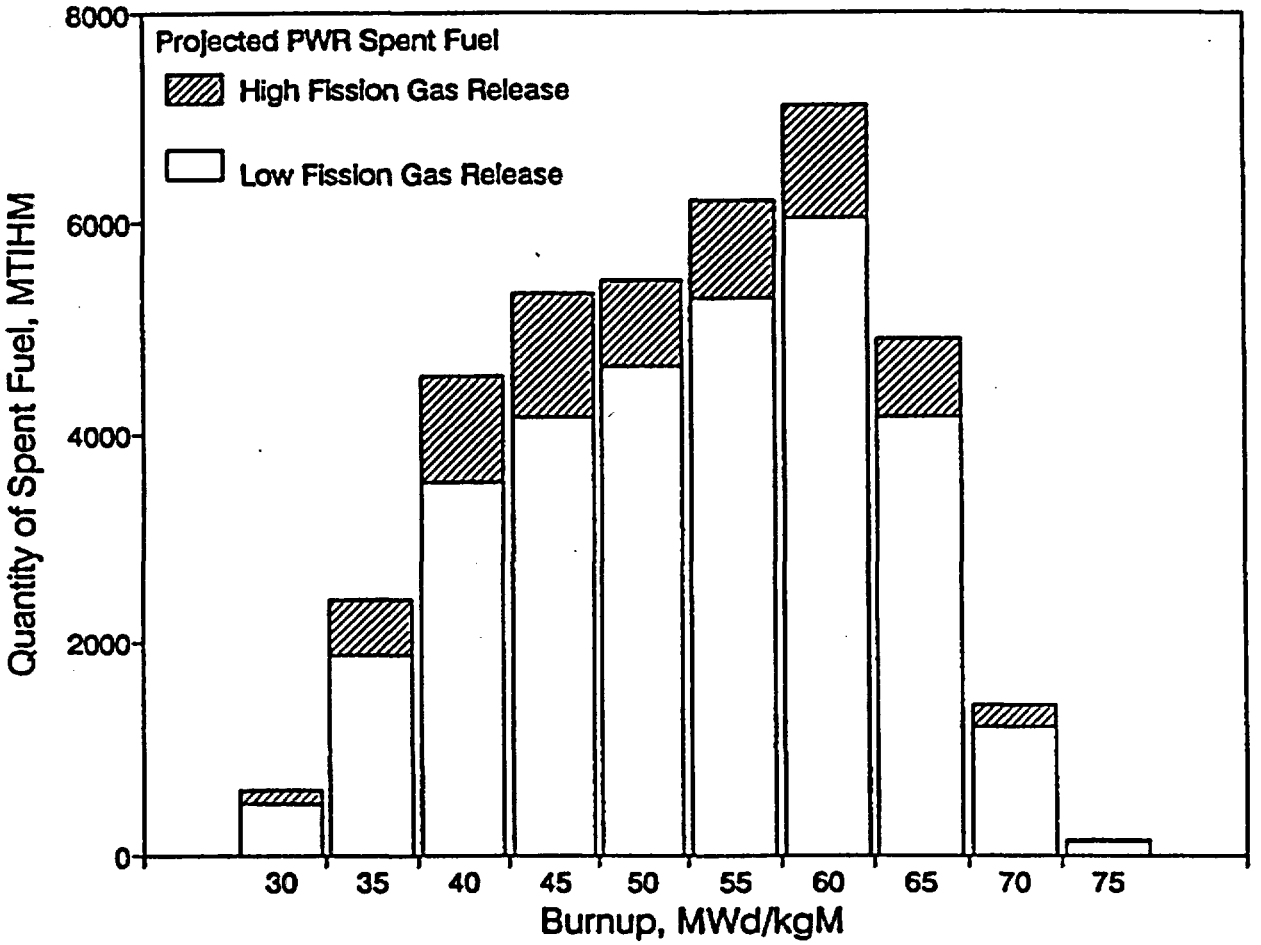


FIGURE 6.17. Projected Fission Gas Release Distribution for PWR Spent Fuel Discharged 1989 Through 2020

(High and Low Fission Gas Release are greater than 2% and less than 2%, respectively.)

M.E. Cunningham, et al., *The Impact of Burnup and Fission Gas Release Distributions of the U.S. LWR Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project,* Pacific Northwest Laboratory Report (Draft) 1991.

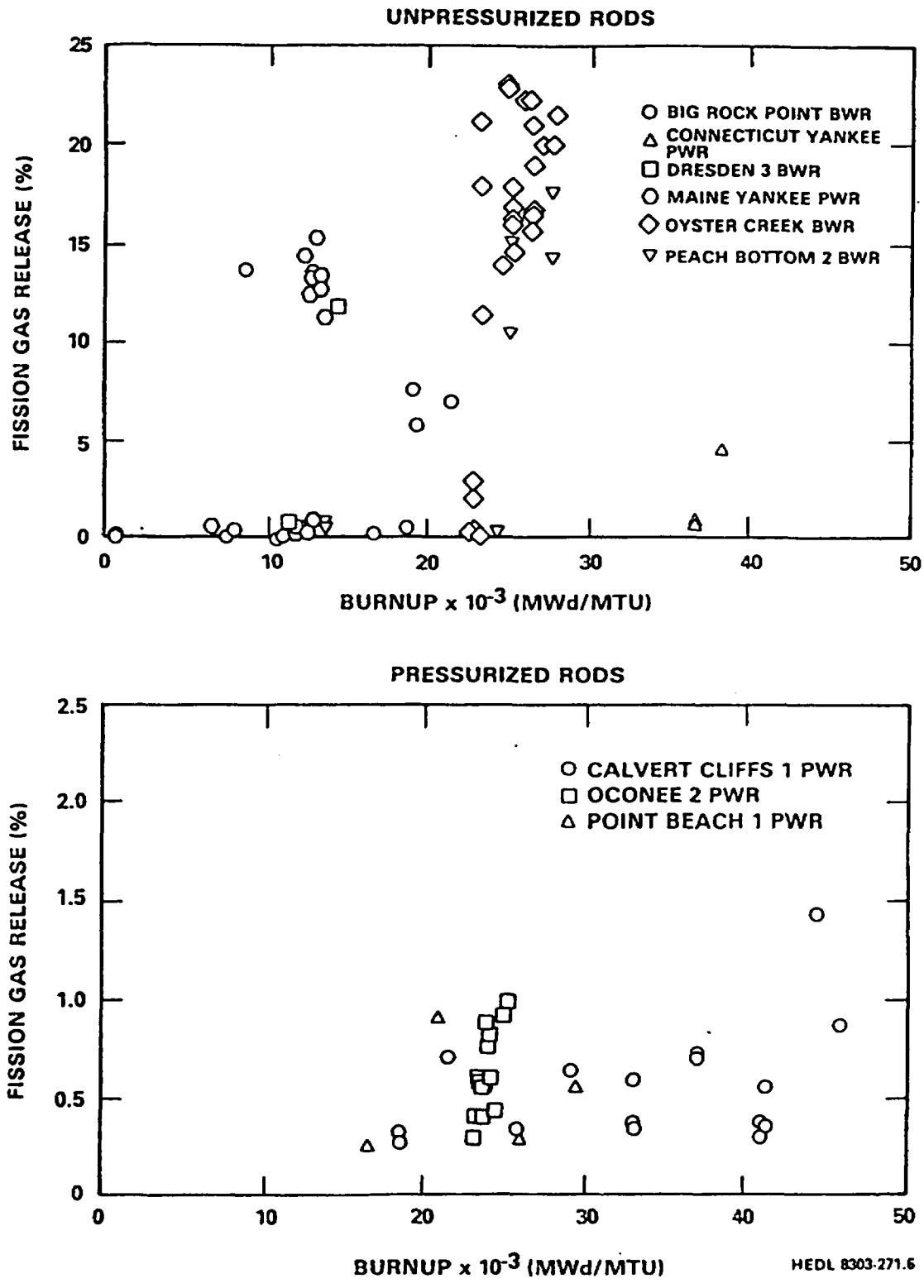


FIGURE 14. Comparison of Fission Gas Release from Unpressurized and Pressurized LWR Fuel Rods.(8)

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, October, 1983.

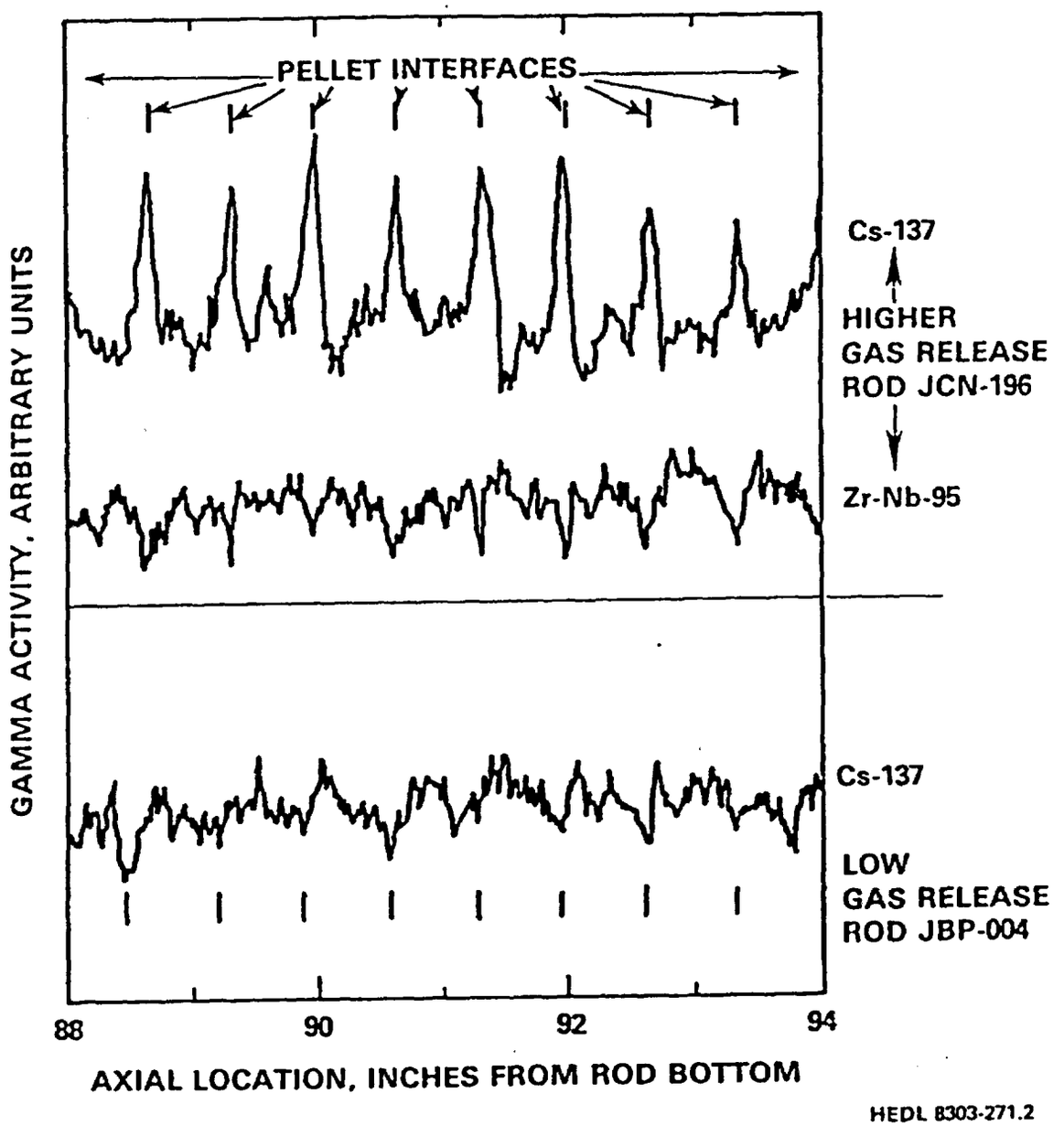


FIGURE 15. Portions of ^{137}Cs Gamma Scans from the Peak Power Position (88 in. to 94 in.) of High- and Low-Gas-Release Fuel Rods from the Maine Yankee PWR.(38)

(This graph is illustrative only and readers are advised to consult the reference for discussion of variability.)

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, October, 1983.

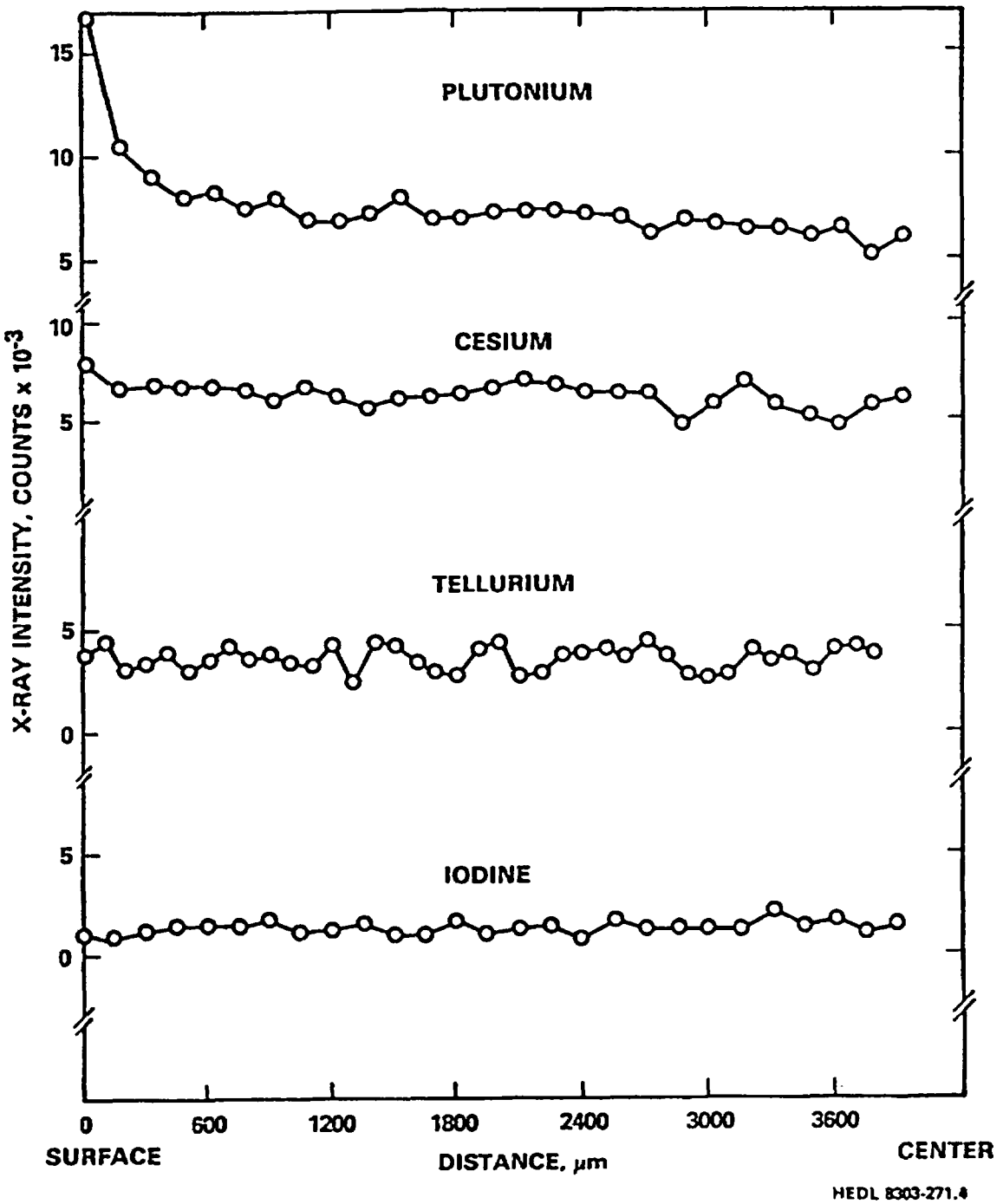


FIGURE 16. Microprobe-Measured X-Ray Intensities for Plutonium, Cesium, Tellurium, and Iodine (Adapted from Reference 40).

(This graph is illustrative only and readers are advised to consult the reference for discussion of variability across radius of a pellet.)

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, October, 1983.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

2.1.2 Structural Characteristics and Dimensions

Spent fuel elements may exist in several physical forms depending on the type of reactor from which they came and who was the manufacturer of the fuel. This has typically resulted in a division of fuel elements into classes.

Several types of spent fuel do not fall into the general classification; these include fuels for unusual, one-of-a-kind reactors, fuel assemblies which have been dismantled, etc.

In order to have the smallest number of standard designs of disposal containers, designers must know the dimensions, weights, shapes and material compositions, as well as the amounts of spent fuels which are intended for disposal in the repository.

It must also be known what special handling devices must be used in order to pick up and handle the many types of fuel and if they must be supported during handling, transportation, and after internment.

This section presents those properties which are most obviously necessary to the designers, although not all have been presented here. Some are not readily available and others, such as assembly drawings of fuel elements can be obtained from the complete report on characteristics of spent fuel, DOE/RW-0184. It should be noted that all "as manufactured dimensions" of Zircaloy will be altered due to stress-induced and irradiation growth-induced strain field during reactor operation. For the long fuel rods, the irradiation growth-induced strain and total length increase in the axial direction must be considered in dimensional tolerances of spent fuel rod and spent fuel assembly containers and handling techniques. A discussion of available models to predict irradiation growth induced strain can be found in an ASTM STP-824 publication (D.G. Franklin and R.B. Adamson, eds., Zirconium in the Nuclear Industry, Sixth Int. Symposium, Vancouver, B.C., pp. 343-382, 1984.)

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

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2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 5.1. Characteristics of CDB Assembly Classes

Assembly Class	Reactor Type	Assembly Length	Assembly Width	Array Size(s) Used
Multi-reactor classes				
GE BWR/2,3	BWR	171.2	5.44	7 x 7, 8 x 8, 9 x 9
GE BWR/4-6	BWR	176.2	5.44	7 x 7, 8 x 8, 9 x 9
B&W 15 X 15	PWR	165.7	8.54	15 x 15
B&W 17 X 17	PWR	165.7	8.54	17 x 17
CE 14 X 14	PWR	157.	8.1	14 x 14
CE 16 X 16	PWR	176.8	8.1	16 x 16
CE 16 X 16 System 80	PWR	178.3	8.1	16 x 16
WE 14 X 14	PWR	159.8	7.76	14 x 14
WE 15 X 15	PWR	159.8	8.44	15 x 15
WE 17 X 17	PWR	159.8	8.44	17 x 17
SOUTH TEXAS	PWR	199.	8.43	17 x 17
Single-reactor classes				
BIG ROCK POINT	BWR	84.0	6.52	12 x 12, 11 x 11, 9 x 9, 8 x 8, 7 x 7
DRESDEN 1	BWR	134.4	4.28	6 x 6, 7 x 7, 8 x 8
ELK RIVER	BWR	81.6	3.5	5 x 5
HUMBOLDT BAY	BWR	95.	4.67	6 x 6, 7 x 7
LACROSSE	BWR	102.5	5.62	10 x 10
FT. CALHOUN	PWR	146.	8.1	14 x 14
HADDAM NECK	PWR	137.1	8.42	15 x 15
INDIAN POINT	PWR	138.8	6.27	13 x 14 (14 x 14)
PALISADES	PWR	147.5	8.2	15 x 15
PATHFINDER	BWR			
ST. LUCIE 2	PWR	158.2	8.1	16 x 16
SAN ONOFRE 1	PWR	137.1	7.76	14 x 14
YANKEE ROWE	PWR	111.8	7.62	15 x 16 (16 x 16), 17 x 18 (18 x 18)

Dimensions are nominal before irradiation. All dimensions are in inches. Lengths are rounded to the next higher tenth of an inch. Lengths of some newer fuel assemblies use slightly (0.1 in.) longer fuel designs. Widths are rounded to the next higher hundredth of an inch. Fuel assembly widths for GE BWR/2,3 and GE BWR/4,5,6 Classes include 80 mil fuel channels. Assemblies with thicker channels (100 and 120 mil) have larger widths.

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

Table 5.2. Summary of Fuel Design Usage

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

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Table 5.2. (cont.) Summary of Fuel Design Usage

Name	Year Introduced	Where Used	Brief Description
GENERAL ELECTRIC FUEL DESIGNS			
	General Electric Built Reactors		
Early Fuels (GE-1)	First Use: 1959 Continuing at Big Rock Point	Dresden-1 Humboldt Bay Big Rock Point	Fuels for BWR/1 reactors
GE-2	First Use: 1969 Last Discharge: 1979	GE BWR/2,3 GE BWR/4,5,6	Original 7 x 7 Array
GE-3	First Use: 1972 Last Discharge: 1983	GE BWR/2,3 GE BWR/4,5,6	Improved 7 x 7 Array - thicker cladding, hydrogen getter chamfered pellets.
GE-4	First Use: 1974 Last Discharge: 1986	GE BWR/2,3 GE BWR/4,5,6	Original 8 x 8 Array - introduction of water rod.
GE-5	First Use: 1975 Last Discharge:	GE BWR/2,3 GE BWR/4,5,6	8 x 8 Retrofit fuel - two water rods, axial natural uranium blankets, longer active fuel rod length.
Prepressurized (GE-6 and GE-7)	First Use: 1977 Peach Bottom-2	GE BWR/2,3 GE BWR/4,5,6	Retrofit fuel with fuel rods prepressurized to 3 atm helium.
Barrier (GE-6 and GE-7)	First Use: 1979 Quad Cities-1	GE BWR/2,3 GE BWR/4,5,6	Pressurized Retrofit fuel with pure zirconium barrier on cladding interior.
GE-8	First Use: 1981 Brown's Ferry-3	GE BWR/2,3 GE BWR/4,5,6	Increased number of water rods (3-6), larger diameter fuel pellets, higher stack density, axial gadolinia distribution, improved upper tie plate, prepressurization increase to 5 atms in BWR/3-6 reactors.
GE-9	First Use: 1987 Hatch-1	GE BWR/2,3 GE BWR/4,5,6	Single large water rod, possibly ferrule-type spacers.
GE-10	First Use: 1988(?) Cooper Station (?)	GE BWR/4,5,6	
GE-11	First Use: 1990 WNP-2, Fitzpatrick	GE BWR/4,5,6	9 x 9 fuel rod array with 74 fuel rods and two large-diameter water rods.
BABCOCK & WILCOX FUEL DESIGNS			
	Babcock & Wilcox Reactors		
Mark B2	First Use: Oconee 2	B&W 15 x 15	
Mark B3	First Use: Oconee 2	B&W 15 x 15	Increased fuel pellet density and changed spacer from corrugated to spring type.
Mark B4	First Use: 1975 Oconee 1	B&W 15 x 15	Introduced fuel rod prepressurization; modified end fitting reduced fuel assembly pressure drop.

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Table 5.2. (cont.) Summary of Fuel Design Usage

Name	Year Introduced	Where Used	Brief Description
BABCOCK & WILCOX FUEL DESIGNS (cont.)			
Mark B5	First Use: 1982 Rancho Seco	B&W 15 x 15	Eliminated the use of retainers for burnable absorber holddown by using modified end fitting. Inconel 718 holddown spring.
Mark BxZ	First Use: 1979 Oconee 1	B&W 15 x 15	Mark B4 and B5 fuels with Zircaloy intermediate grid spacers.
Mark B6	First Use: 1988 Ark. Nuclear One-1	B&W 15 x 15	Zircaloy intermediate spacers grids; skirtless upper end grid, and removable upper end fitting.
Mark B7	First Use: 1988 Oconee 3	B&W 15 x 15	Mark B6 features plus slightly shorter lower end fitting, slightly longer fuel rod, and increased plenum volume.
Mark B8	First Use: 1989 Oconee 3	B&W 15 x 15	Debris fretting resistent fuel rod design, reduced prepressurization.
Mark C	First Use: 1976 Oconee 2	B&W 15 x 15	Four demonstration assemblies of fuel design intended for B&W 17 x 17 class reactors.
BABCOCK & WILCOX FUEL DESIGNS Other Reactor Vendors			
St. Steel	First Use: Haddam Neck	Haddam Neck	Stainless steel clad assemblies for use at WE-built reactors.
Westinghouse	First Use: 1974 Ginna	WE 14 x 14	Demonstration assemblies for use at WE-built reactors.
Mark BW	First Use: 1989 McGuire 1	WE 17 x 17	Lead test assemblies under irradiation; full core reload scheduled for 1991.
ADVANCED NUCLEAR FUELS, INC. PWR Fuels			
Westinghouse	First Use: 1974 Ginna	WE 14 x 14 WE 15 x 15 WE 17 x 17	Fuels designed for use at WE-built reactors.
Toprod	First Use: 1981 Prairie Island-1	WE 14 x 14	Fuels for use at WE 14 x 14 plants; fueled rods containing Gadolinia.
Part Length	First Use: 1986 (?) Robinson-2	WE 15 x 15	Fuel for use at WE-built reactors; bottom 42 in. of fuel rod contains stainless steel inserts.
Comb. Eng.	First Use: 1980 Fort Calhoun	CE 14 x 14 Fort Calhoun	Fuels designed for use at CE-built reactors.

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Table 5.2. (cont.) Summary of Fuel Design Usage

Name	Year Introduced	Where Used	Brief Description
ADVANCED NUCLEAR FUELS, INC. (cont.)			
Palisades	First Use: 1975 Palisades	Palisades	Fuel designed for use at Palisades reactor.
Yankee Rowe	First Use: 1975 Yankee Rowe	Yankee Rowe	Fuel designed for use at Yankee Rowe.
ADVANCED NUCLEAR FUELS, INC. **BWR Fuels**			
Early BWR Fuels	First Use: 1977 1974 1972	Dresden 1 Humboldt Bay Big Rock Point	Fuels designed for GE BWR/1 reactors.
7 x 7 Arrays	First Use: 1972 Oyster Creek	GE BWR/2,3	7 x 7 array designed for use at GE-built reactors.
8 x 8 Arrays	First Use: 1975 Oyster Creek	GE BWR/2,3 GE BWR/4,5,6	8 x 8 array designed for use at GE-built reactors.
9x9	First Use: 1983 Dresden-2	GE BWR/2,3 GE BWR/4,5,6	9 x 9 fuel array for GE-built reactors; 2 water rods.
9x9-5	First Use: Unknown	GE BWR/4,5,6	9 x 9 fuel array for GE-built reactors; 5 water rods.
9x9-IX	First Use: 1989 WNP-2	GE BWR/4,5,6	9 x 9 fuel array for GE-built reactors; 72 fuel rods; Zirconium barrier used on all rods except Gadolinia rods; 1.65" square water channel.
9x9-9X	First Use: 1989 WNP-2	GE BWR/4,5,6	9 x 9 fuel array for GE-built reactors; 72 fuel rods; 1.65" square water channel.
COMBUSTION ENGINEERING FUEL DESIGNS			
Standard	First Use:	CE 14 x 14 CE 16 x 16 CE 16 x 16 System 80 Fort Calhoun St. Lucie 2	
Palisades	First Use:	Palisades	
Yankee Rowe	First Use:	Yankee Rowe	

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Table 5.3. Listing of Assembly Types by Assembly Class.

Assembly Class / Assembly Type	Rodarray Code	Assemblies In Storage	Status	Comments
BABCOCK & WILCOX 15 X 15				
B&W 15 X 15 B&W Mark B	B1515B	567	Discharged	
B&W 15 X 15 B&W Mark B2	B1515B2	92		
B&W 15 X 15 B&W Mark B3	B1515B3	615	Discharged	
B&W 15 X 15 B&W Mark B4	B1515B4	2071	Incore	
B&W 15 X 15 B&W Mark B4Z	B1515B4Z	36	Incore	
B&W 15 X 15 B&W Mark B5	B1515B5	56	Incore	
B&W 15 X 15 B&W Mark B5Z	B1515B5Z	43	Incore	
B&W 15 X 15 B&W Mark B6	B1515B6	0	Incore	
B&W 15 X 15 B&W Mark B7	B1515B7	0	Incore	
B&W 15 X 15 B&W Mark B8	B1515B8	68	Incore	
B&W 15 X 15 B&W Mark BGd	B1515BG	4	Discharged	Lead Assembly
B&W 15 X 15 WE	B1515W	0	Projected	Lead Assembly in 1991.
BABCOCK & WILCOX 17 X 17				
B&W 17 X 17 B&W Mark C	B1717B	4	Discharged	Lead Assembly
COMBUSTION ENGINEERING 14 X 14				
CE 14 X 14 CE	C1414C	2810	Incore	
CE 14 X 14 ANF	C1414A	323	Incore	
CE 14 X 14 WE	C1414W	189	Incore	
COMBUSTION ENGINEERING 16 X 16				
CE 16 X 16 CE San Onofre	C1616C	1043	Incore	
CE 16 X 16 CE ANO2	C1616C		Incore	
COMBUSTION ENGINEERING 16 X 16 SYSTEM 80				
CE 16 X 16 CE System 80	C1616CS8	188	Incore	
GENERAL ELECTRIC BWR/2,3				
GE BWR/2,3 7 X 7 GE-2a	G2307G2A	1672	Discharged	
GE BWR/2,3 7 X 7 GE-2b	G2307G2B	5047	Discharged	
GE BWR/2,3 7 X 7 GE-3	G2307G3	394	Discharged	
GE BWR/2,3 7 X 7 ANF	G2307A	260	Discharged	
GE BWR/2,3 8 X 8 GE-4	G2308G4	3876	Discharged	
GE BWR/2,3 8 X 8 GE-5	G2308G5	792	Incore	
GE BWR/2,3 8 X 8 GE Pressurized	G2308GP	1836	Incore	
GE BWR/2,3 8 X 8 GE Barrier	G2308GB	248	Incore	
GE BWR/2,3 8 X 8 GE-8a	G2308G8A		Incore	
GE BWR/2,3 8 X 8 GE-8b	G2308G8B		Incore	
GE BWR/2,3 8 X 8 GE-9a	G2308G9A		Lead Assembly	
GE BWR/2,3 8 X 8 GE-9b	G2308G9B		Lead Assembly	
GE BWR/2,3 8 X 8 ANF	G2308A	68	Incore	
GE BWR/2,3 8 X 8 ANF Pressurized	G2308AP		Incore	
GE BWR/2,3 9 X 9 ANF	G2309A		Incore	
GE BWR/2,3 9 X 9 ANF 9-5	G2309A5		Unknown	
GE BWR/2,3 9 X 9 ANF IX	G2309AIX		Lead Assembly	
GE BWR/2,3 9 X 9 ANF 9X	G2309A9X		Lead Assembly	

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:. (cont). Listing of Assembly Types by Assembly Class.

Class / Type	Rodarray Code	Assemblies In Storage	Status	Comments
L ELECTRIC BWR/4,5,6				
/4-6 7 X 7 GE-2	G4607G2	1142	Discharged	
/4-6 7 X 7 GE-3a	G4607G3A	3752	Discharged	
/4-6 7 X 7 GE-3b	G4607G3B	1184	Discharged	
/4-6 8 X 8 GE-4a	G4608G4A	1784	Discharged	
/4-6 8 X 8 GE-4b	G4608G4B	1787	Discharged	
/4-6 8 X 8 GE-5	G4608G5	3455	Incore	
/4-6 8 X 8 GE Pressurized	G4608GP	6591	Incore	
/4-6 8 X 8 GE Barrier	G4608GB	775	Incore	
/4-6 8 X 8 GE-8a	G4608G8A		Incore	
/4-6 8 X 8 GE-8b	G4608G8B		Incore	
/4-6 8 X 8 GE-9a	G4608G9A		Lead Assembly	
/4-6 8 X 8 GE-9b	G4608G9B		Lead Assembly	
/4-6 8 X 8 GE-10	G4608G10		Lead Assembly	
/4-6 9 X 9 GE-11	G4609G11		Future	
/4-6 8 X 8 ANF	G4608A		Incore	
/4-6 8 X 8 ANF Pressurized	G4608AP		Incore	
/4-6 9 X 9 ANF	G4609A		Incore	
R/4-6 9 X 9 ANF 9-5	G4609A5		Lead Assembly	
R/4-6 9 X 9 ANF IX	G4609AIX		Lead Assembly	
R/4-6 9 X 9 ANF 9X	G4609A9X		Lead Assembly	
R/4-6 8 X 8 WE	G4608W		Lead Assembly	
R/4-6 10 X 10 SVEA 96	•		Future	
GHOUSE 14 X 14				
X 14 WE Std	W1414W	581	Incore	
X 14 WE LOPAR	W1414WL	1376	Incore	
X 14 WE OFA	W1414WO	88	Incore	
X 14 WE Vantage 5	W1414WV5			
X 14 ANF	W1414A	559	Incore	
X 14 ANF Top Rod	W1414ATR	299	Incore	
X 14 B&W	W1414B	2	Discharged	
IGHOUSE 15 X 15				
X 15 WE Std	W1515W	1395	Incore	
X 15 WE LOPAR	W1515WL	3149	Incore	
X 15 WE OFA	W1515WO	266	Incore	
X 15 WE Part Length	W1515WPL		Incore	
X 15 WE Vantage 5	W1515WV5			
X 15 ANF Westinghouse	W1515A	743	Incore	
X 15 B&W Mark BW	W1515B		Future	
NGHOUSE 17 X 17				
X 17 WE LOPAR	W1717W	5106	Incore	
X 17 WE OFA	W1717WO	628	Incore	
X 17 WE Vantage 5	W1717WV5	4	Incore	
X 17 WE Vantage +	W1717WV+		Lead Assembly	
X 17 WE Vantage H	W1717WVH		Lead Assembly	
X 17 ANF Westinghouse	W1717A	139	Incore	
X 17 B&W Mark BW	W1717B		Lead Assembly	
I TEXAS				
H TEXAS 17 X 17 WE	WST17W		Incore	

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Table 5.3. (cont.) Listing of Assembly Types by Assembly Class.

Assembly Class / Assembly Type	Rodarray Code	Assemblies In Storage	Status	Comments
BIG ROCK POINT				
BIG ROCK POINT 12 X 12 GE	XBR12G		Reprocessed	
BIG ROCK POINT 11 X 11 GE	XBR11G	6	Discharged	
BIG ROCK POINT 9 X 9 GE	XBR09G	143	Discharged	
BIG ROCK POINT 7 X 7 GE	XBR07G	4	Discharged	Lead Assembly
BIG ROCK POINT 8 X 8 GE	XBR08G	2	Discharged	Lead Assembly
BIG ROCK POINT 9 X 9 ANF	XBR09A	4	Discharged	
BIG ROCK POINT 11 X 11 ANF	XBR11A	145	Incore	
BIG ROCK POINT 11 X 11 NFS	XBR11N	8	Discharged	
DRESDEN 1				
DRESDEN 1 6 X 6 GE Type I	XDR06G1	1	Reprocessed	
DRESDEN 1 7 x 7 GE Type II	XDR07G		Reprocessed	
DRESDEN 1 6 x 6 GE Type III-B	XDR06G3B	163	Discharged	
DRESDEN 1 6 x 6 GE Type III-F	XDR06G3F	96	Discharged	
DRESDEN 1 6 x 6 GE Type V	XDR06G5	106	Discharged	
DRESDEN 1 7 x 7 GE SA-1	XDR07GSA		Discharged	
DRESDEN 1 8 x 8 GE PF Fuels	XDR08G	1	Discharged	
DRESDEN 1 6 X 6 UNC	XDR06U	458	Discharged	
DRESDEN 1 6 X 6 ANF	XDR06A	66	Discharged	
FORT CALHOUN				
FT. CALHOUN 14 X 14 CE	XFC14C	290	Incore	
FT. CALHOUN 14 X 14 ANF	XFC14A	136	Incore	
FT. CALHOUN 14 x 14 WE	XFC14W		Future	
HUMBOLDT BAY				
HUMBOLDT BAY 7 X 7 GE Type I	XHB07G1		Reprocessed	
HUMBOLDT BAY 7 x 7 GE Type II	XHB07G2	88	Discharged	
HUMBOLDT BAY 6 X 6 GE	XHB06G	176	Discharged	
HUMBOLDT BAY 6 X 6 ANF	XHB06A	126	Discharged	
HADDAM NECK				
HADDAM NECK 15 X 15	XHN15W	309	Discharged	
HADDAM NECK 15 x 15 NUM Zir	XHN15MZ	2	Discharged	Lead Assembly
HADDAM NECK 15 x 15 NUM SS	XHN15MS	2	Discharged	Lead Assembly
HADDAM NECK 15 x 15 GGA Zir	XHN15IZ	2	Discharged	Lead Assembly
HADDAM NECK 15 x 15 GGA SS	XHN15IS	1	Discharged	Lead Assembly
HADDAM NECK 15 X 15 B&W SS	XHN15B	418	Incore	
HADDAM NECK 15 X 15 B&W Zir	XHN15BZ			Lead Assembly
INDIAN POINT 1				
INDIAN POINT 13 X 14 B&W	XIP14B	0	Reprocessed	
INDIAN POINT 13 X 14 WE	XIP14W	160	Discharged	

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Table 5.3. (cont.) Listing of Assembly Types by Assembly Class.

Assembly Class / Assembly Type	Rodarray Code	Assemblies In Storage	Status	Comments
LACROSSE				
LACROSSE 10 X 10 AC	XLC10L	155	Discharged	
LACROSSE 10 X 10 ANF	XLC10A	178	Discharged	
PALISADES				
PALISADES 15 X 15 CE	XPA15C	273	Discharged	
PALISADES 15 X 15 ANF	XPA15A	324	Incore	
ST.LUCIE 2				
ST. LUCIE 2 16 X 16 CE	XSL16C	236	Incore	
SAN ONOFRE 1				
SAN ONOFRE 1 14 X 14 WE	XSO14W	468	Incore	
YANKEE ROWE				
YANKEE ROWE 17 X 18 WE	XYR18W	76	Discharged	
YANKEE ROWE 15 X 16 UNC	XYR16U	73	Discharged	
YANKEE ROWE 15 X 16 ANF	XYR16A	228	Discharged	
YANKEE ROWE 15 X 16 CE	XYR16C	40	Incore	

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Table 3.2. Assumed fuel assembly structural material mass distribution

	PWR[a]			BWR[a]		
		Mass			Mass	
	Material	kg/MTHM	kg/assembly	Material	kg/MTHM	kg/assembly
Fuel Zone						
Cladding	Zircaloy-4	223.0	102.9	Zircaloy-2	279.5	51.2
Fuel channel[b]	--	--	--	Zircaloy-4	227.5	41.7
Grid spacers	Inconel 718	12.8	5.9	Zircaloy-4	10.6	1.9
Grid-spacer springs	Inconel 718			Inconel X-750	1.8	0.3
Grid-brazing material	Nicrobraze 50	2.6	1.2	--	--	--
Miscellaneous	SS 304[c]	9.9	4.6	--	--	--
Fuel-gas plenum zone						
Cladding	Zircaloy-4	12.0	5.5	Zircaloy-2	25.4	4.7
Fuel channel[b]	--	--	--	Zircaloy-4	20.7	3.8
Plenum spring	SS 302	4.2	1.9	SS 302	6.0	1.1
End fitting zone						
Top end fitting	SS 304	14.8	6.8	SS 304	10.9	2.0
Bottom end fitting	SS 304	12.4	5.7	SS 304	26.1	4.8
Expansion springs	--	--	--	Inconel X-750	2.1	0.4
Total		291.7	134.5		610.6	111.9

[a]Source: A. G. Croff, M. A. Bjerke, G. W. Morrison, and L. M. Petrie, Revised Uranium — Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code, ORNL/TM-6051, September 1978.

[b]Assumed to be discarded with fuel assembly, channels are often reused with fresh fuel.

[c]Distributed throughout the PWR core in sleeves and so forth.

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

Physical Description Report Page: 2

Combustion Enigneering SYSTEM80 12-Rod Full-Length Control Element

Composition:

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

Used at the Following Reactors:

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

Used with the Following Fuel Assembly Types:

Vendor	Array	Version
Combustion Engineering	16 x 16	System 80

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

Radiological Description Report Page 1

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

ISOTOPIC COMPOSITION

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

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

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

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

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

K.J. Notz, *Characteristics of Potential Repository Waste*, DOE/RW-0184-R1, V.1 (draft), July, 1990.

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

Radiological Description Report Page 2

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

PHOTON SPECTRA

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

METALLIC COMPOSITION
(Materials modeled to obtain this report)

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

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

K.J. Notz, *Characteristics of Potential Repository Waste*, DOE/RW-0184-R1, V.1 (draft), July, 1990.

TABLE 2-1

SUMMARY OF THE QUANTITIES OF NFA COMPONENTS PROJECTED TO BE AVAILABLE FOR DELIVERY TO THE FWMS -- FOR CASES WHERE COMPONENTS ARE DELIVERED AS AN INTEGRAL PART OF THE FUEL ASSEMBLY AND WHERE THEY ARE DELIVERED IN EITHER AN UNCOMPACTED OR COMPACTED FORM[a,b]

	Total Units	Can Dimensions (in)	Can Capacity	Total No. Cans or Units	Weight of Loaded Can or Unit (lb)
As Integral Part of Fuel Assys					
PWR Control Rod Assemblies	10,000		In Fuel Assy	10,000	149
PWR Burnable Poison Assys (West)	55,000		In Fuel Assy	55,000	156
PWR Burnable Poison Assys (B&W)	6,500		In Fuel Assy	6,500	57
PWR Neutron Source Assemblies	320		In Fuel Assy	320	51
PWR Thimble Plug Assemblies	2,900		In Fuel Assy	2,900	13
BWR Fuel Channels	110,000		In Fuel Assy	110,000	98
BWR Control Assemblies[c]	14,500		No Can	14,500	225
BWR Instrument Assemblies	5,000		In Fuel Assy	5,000	2
BWR Poison Curtains[c]	750	10.5x10.5x176	74	11	2,263
Compacted					
PWR Control Rod Assys - Rod Sets	10,000	9x9x160	15	667	2,437
- Spiders	10,000	9x9x160	20	500	482
PWR Burnable Poison Assys (West)					
- Rod Sets	55,000	9x9x160	15	3,667	2,527
- Spiders	55,000	9x9x160	40	1,375	682
PWR Burnable Poison Assys (B&W)					
- Rod Sets	6,500	9x9x160	19	342	1,253
- Spiders	6,500	9x9x160	26	250	530
PWR Neutron Source Assemblies					
- Rod Sets	320	9x9x160	15	21	967
- Spiders	320	9x9x160	53	6	746
PWR Thimble Plug Assys - Rod Sets	2,900	9x9x160	114	25	778
- Spiders	2,900	9x9x160	40	73	682
BWR Fuel Channels	110,000	6x6x168	7	15,714	909
BWR Control Assemblies[d]	14,500	No Can	No Can	14,500	225
BWR Instrument Assemblies	5,000	6x6x160	47	106	306
BWR Poison Curtains	750	10.5x10.5x176	74	11	2,263
Uncompacted					
PWR Control Rod Assemblies	10,000	9x9x162	1	10,000	475
PWR Burnable Poison Assys (West)	55,000	9x9x160	1	55,000	478
PWR Burnable Poison Assys (B&W)	6,500	9x9x160	1	6,500	379
PWR Neutron Source Assemblies	320	9x9x160	1	320	373
PWR Thimble Plug Assemblies	2,900	9x9x160	13	223	491
BWR Fuel Channels	110,000	No Can	No Can	110,000	98
BWR Control Assemblies	14,500	No Can	No Can	14,500	225
BWR Instrument Assemblies	5,000	6x6x160	47	106	306
BWR Poison Curtains	750	10.5x10.5x176	74	11	2,263

[a] Assumes all NFA components listed are classified as greater-than-Class C waste.

[b] Quantities are estimated to be those equivalent to the production of a nominal 70,000 MTU of SNF assemblies.

[c] Not integral.

[d] Uncompacted.

E.R. Johnson Associates, Inc. (compilers), *Acceptance of Non-Fuel Assembly Hardware by the Federal Waste Management System*, ORNL/SUB./86-SA094/8, JAI-328, March, 1990.

TABLE 3-1

SUMMARY COMPARISON OF ATTRIBUTES OF CONTROL ROD ASSEMBLIES IN PWRs[a]

	Fuel Array	Total Length (in)	Spider Length (in)	No. Rods	Total Weight (lb)	Poison	No. Assys in Core
Westinghouse							
	14x14	159	8	16	128	Ag-In-Cd	53 Full Length; 8 Part Length[b]
	14x14	157	8	16	128	Ag-In-Cd	53 Full Length; 8 Part Length[b]
	14x14	134	8	16	109	Ag-In-Cd	53 Full Length; 8 Part Length[b]
	15x15	157	8	20	165	Ag-In-Cd	53 Full Length; 8 Part Length[b]
	17x17	161	8	24	149	Ag-In-Cd	53 Full Length; 8 Part Length[b]
	17x17	161	8	24	180	Hf	53 Full Length; 8 Part Length[b]
	17x17	161	8	24	149	Ag-In-Cd	53 Full Length; 8 Part Length[b]
	17x17 (Hybrid)	161	8	24	93	B_4C/Ag-In-Cd	53 Full Length; 8 Part Length[b]
	17x17	161	8	24	100	Ag-In-Cd	
B&W							
	15x15	160	4	16	130	Ag-In-Cd	61
Combustion Engineering							
	16x16	253	8[c]	4	95	Inconel 625	13
	16x16	253	8[c]	12	192	B_4C	48
	16x16	181	8[c]	5	92	Inconel 625	8
	16x16	181	8[c]	5	72	B_4C	83
	16x16	181	8[c]	5	91	Inconel 625	8
	16x16	181	8[c]	5	71	B_4C	73
	16x16	163	8[c]	5	83	Inconel 625	8
	16x16	163	8[c]	5	66	B_4C	83
	14x14	161	8[c]	5	105	B_4C	12
	14x14	161	8[c]	5	82	B_4C	8
	14x14	161	8[c]	5	63	B_4C	12
	14x14	161	8[c]	5	77	B_4C	65
	14x14	152	8[c]	5	63	B_4C	4
	14x14	152	8[c]	5	67	B_4C	45
	15x15	151	8[c]	Cruciform	214	Ag-In-Cd	45

[a] Source: DOE/RW-0184, Vol. 5

[b] Salem FSAR

[c] Estimated (assumed)

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 5.9. Spent Fuel Disassembly Hardware for Major PWR Assembly Types. Listing by Assembly Class.

Class Name / Assembly Type Name	Hardware Weight	Hardware Composition
B&W 15 x 15 Assembly Class		
B&W 15 x 15 B&W Mark B	35.6 kg	26% Zirc, 25% Inc, 49% SS
B&W 15 x 15 B&W Mark BZ	35.6 kg	40% Zirc, 11% Inc, 49% SS
B&W 17 x 17 Assembly Class		
B&W 17 x 17 B&W Mark C	42.3 kg	28% Zirc, 25% Inc, 47% SS
CE 14 x 14 Assembly Class		
CE 14 x 14 CE	29.8 kg	58% Zirc, 8% Inc, 34% SS
CE 14 x 14 ANF	33.3 kg	45% Zirc, 15% Inc, 40% SS
CE 14 x 14 WE	34.1 kg	32% Zirc, 26% Inc, 42% SS
CE 16 x 16 Assembly Class		
CE 16 x 16 CE ANO2	40.1 kg	50% Zirc, 6% Inc, 44% SS
CE 16 x 16 CE SONGS	42.6 kg	45% Zirc, 14% Inc, 41% SS
CE 16 x 16 System 80 Assembly Class		
CE 16 x 16 CE System 80	44.0 kg	44% Zirc, 13% Inc, 43% SS
WE 14 x 14 Assembly Class		
WE 14 x 14 WE Std	32.0 kg	20% Inc, 80% SS
WE 14 x 14 WE LOPAR	31.8 kg	25% Zirc, 20% Inc, 55% SS
WE 14 x 14 WE OFA	32.1 kg	55% Zirc, 8% Inc, 37% SS
WE 14 x 14 ANF	28.4 kg	49% Zirc, 4% Inc, 47% SS
WE 14 x 14 ANF Toprod	24.6 kg	54% Zirc, 4% Inc, 42% SS
WE 15 x 15 Assembly Class		
WE 15 x 15 WE Std	35.8 kg	24% Inc, 76% SS
WE 15 x 15 WE LOPAR	35.6 kg	26% Zirc, 24% Inc, 50% SS
WE 15 x 15 WE OFA	32.6 kg	53% Zirc, 9% Inc, 38% SS
WE 15 x 15 ANF	27.3 kg	53% Zirc, 4% Inc, 43% SS
WE 17 x 17 Assembly Class		
WE 17 x 17 WE LOPAR	29.6 kg	32% Zirc, 22% Inc, 45% SS
WE 17 x 17 WE OFA	32.3 kg	51% Zirc, 8% Inc, 40% SS
WE 17 x 17 ANF	34.6 kg	59% Zirc, 7% Inc, 34% SS

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

TABLE 2

MECHANICAL DESIGN PARAMETERS OF PWR FUEL ASSEMBLIES*

Rod Array	Westinghouse 15 x 15	Westinghouse 17 x 17	Combustion Engineering 14 x 14	Combustion Engineering 16 x 16	Babcock & Wilcox 15 x 15	Babcock & Wilcox 17 x 17	Exxon 15 x 15
Fuel Assemblies							
Transverse Dimension (in., cm)	8.426(21.402)	8.426(21.402)	7.98(20.27)	7.98(20.27)	8:536(21.681)	8.436(21.427)	
Assembly Weight (lb, kg)	1420(644)	1450(658)	1280(581)	1446(656)			
Overall Assembly Length (in., cm)	161.3(409.7)	161.3(409.7)	156.7(398)	176.8(448.9)		165.6(420.6)	
Fuel Rods							
Number per Assembly	204	264	176	236	208	264	204
Rod Pitch (in., cm)	0.563(1.430)	0.496(1.260)	0.580(1.473)	0.506(1.285)	0.568(1.443)	0.501(1.273)	0.563(1.430)
Length (in., cm)	149.7(380.2)	151.6(385.1)	145.9(370.6)	161.0(408.9)	153.7(390.4)	152.1(386.4)	152.0(386.1)
Fueled Length (in., cm)	144.0(365.8)	143.7(365.0)	136.7(347.2)	150.0(381.0	141.8(360.2)	143.0(363.2)	144.0(365.8)
OD (in., cm)	0.422(1.072)	0.374(0.950)	0.440(1.118)	0.382(0.970)	0.430(1.092)	0.379(0.963)	0.424(1.077)
Diametral Gap (in., cm)	0.0075(0.0190)	0.0065(0.0165)	0.0085(0.0216)	0.007(0.0178)	0.0084(0.0213)	0.0078(0.0198)	0.0075(0.0190)
Cladding Thickness (in., cm)	0.0243(0.0617)	0.0225(0.0572)	0.026(0.0660)	0.025(0.0635)	0.0265(0.0673)	0.0240(0.0610)	0.030(0.0762)
Cladding Material	Zircaloy-4*	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Fuel Pellets							
Density (% TD)	95	95	94.75	95	95	95	94
Diameter (in., cm)	0.3659(0.9294)	0.3225(0.8192)	0.3795(0.9639)	0.325(0.8255)	0.3686(0.9362)	0.3232(0.8209)	0.3565(0.9055)
Length (in., cm)	0.600(1.524)	0.530(1.346)	0.650(1.651)	0.390(0.991)	0.600(1.524)	0.375(0.952)	0.273(0.693)
Guide Tubes							
Number	20	24	4	4	16	24	20
Upper OD (in., cm)	0.544(1.382)	0.480(1.219)	1.115(2.832)	1.115(2.832)		0.465(1.181)	
Wall Thickness (in., cm)	0.017(0.043)	0.016(0.041)	0.036(0.091)	0.036(0.091)		0.017(0.043)	
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	
Instrument Tubes							
Number	1	1	1	1	1	1	1
OD (in., cm)	0.544(1.382)	0.480(1.219)	1.115(2.832)	0.417(1.059)		0.420(1.067)	
Wall Thickness (in., cm)	0.017(0.043)	0.016(0.041)	0.036(0.091)	0.027(0.069)		0.01512(0.0384)	
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	
Tie Plate							
Material	304 SS	304 SS	304 SS	304 SS	304 SS	304 SS	
Spacers							
Numbers	7	8	9	12	8	8	
Material	Inconel 718*	Inconel 718	Zircaloy-4	Zircaloy-4	Inconel 718	Inconel 718	
Springs	Inconel 718	Inconel 718	Zircaloy-4	Zircaloy-4	Inconel 718	Inconel 718	
Plenum Springs							
Working Length	6.80(17.27)	6.70(17.02)	8.60(21.86)	6.48(16.46)			
Material	Inconel 718	Inconel 718	Inconel 718	Inconel 718			

*Updated from Reference 3.

*Zircaloy is a registered trademark of Westinghouse Electric Corp., Pittsburgh, PA

Table 2.2. Mechanical design parameters for Westinghouse PWR fuel assemblies[a]

Design component	Rod array							
	17 × 17			15 × 15		14 × 14		16 × 16[b]
	Standard	OFA	VANTAGE	Standard	OFA	Standard	OFA	Standard
Assembly								
Transverse dimension, in.	8.426	8.426	8.426	8.426	8.426	7.763	7.763	7.763
Assembly weight, lb	1467	1365	1365	1440	1443	1274	1135	1310
Uranium/assembly, lb	1017.23	932.61	932.61	1011.86	1011.86	887.77	786.61	905.32
UO_2/assembly, lb	1154.00	1058.00	1058.00	1147.90	1147.90	1007.14	892.37	1027.04
Overall length, in.	159.8	159.8	160.1	159.765	159.765	159.71	159.71	159.8
Rod replacement capabilities	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Disassembly capabilities	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Fuel rods								
Date of commercial operation	1975	1984	1987	1967	1983	1969	1984	1981
Number per assembly	264	264	264	204	204	179	179	235
Rod pitch, in.	0.496	0.496	0.496	0.563	0.563	0.556	0.556	0.485
Length, in.	151.635	151.635	152.3	151.83	151.83	151.83	151.83	151.64
Fuel length, in.	144	144	144	144	144	144	144	144
OD, in.	0.374	0.36	0.36	0.422	0.422	0.422	0.400	0.374
Diametral gap, in.	0.0065	0.0062	0.0065	0.0075	0.0075	0.0075	0.0070	0.0065
Clad thickness, in.	0.0225	0.0225	0.0225	0.0243	0.0243	0.0243	0.0243	0.0225
Clad material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Fuel pellets								
Type	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Density, % TD	95	95	95	95	95	95	95	95
Diameter, in.	0.3225	0.3088	0.3088	0.3659	0.3695	0.3659	0.3444	0.3225
Length, in.	0.53	0.51	0.51	0.60	0.60	0.60	0.565	0.53
Total weight/rod, lb	4.37	4.01	4.01	5.52	5.52	5.63	4.99	4.37
Spacer pellets	None	None	None	None	None	None	None	None
Plenum spring								
Working length, in.	6.90	6.90	7.405	7.136	7.136	7.136	7.158	6.90
Material	SS	SS	SS	SS	SS	SS	SS	SS
Miscellaneous								
Prepressurization, atm	Variable	Variable	Variable	Variable	Variable	Variable	Variable	Variable
Gas used	Helium	Helium	Helium	Helium	Helium	Helium	Helium	Helium
Spacer grids								
Top and bottom grids								
Number/assembly	2	2	2	2	2	2	2	2
Material	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718	Inconel 718
Intermediate grids								
Number/assembly	6	6	6	5	5	5	5	6
Material	Inconel 718	Zr-4	Zr-4	Inconel 718	Zr-4	Inconel 718	Zr-4	Inconel 718
Intermediate flow mixer	None	None		None	None	None	None	None
Number/assembly			3					
Material			Zr-4					
Guide tubes								
Number/assembly	24	24	24	20	20	16	16	20
OD, in.	0.474	0.474	0.474	0.546	0.532	0.539	0.527	0.471
Wall thickness, in.	0.016	0.016	0.016	0.017	0.017	0.017	0.017	0.016
Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Instrument tube								
Number/assembly	1	1	1	1	1	1	1	1
OD, in.	0.48	0.476	0.476	0.546	0.533	0.422	0.4019	0.473
Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Top and bottom nozzles material	SS	SS	SS	SS	SS	SS	SS	SS
Approximate no. of assemblies shipped by Westinghouse	6000	800	0	5200	400	4000	30	400

[a]Source: L. Iyengar, Westinghouse Electric Corporation, letter to J. W. Roddy, Oak Ridge National Laboratory, December 17, 1984.
[b]All of these assemblies have been exported.

Table 2.3. Mechanical design parameters for Combustion Engineering PWR fuel assemblies[a]

Design component	Rod array	
	14 × 14R	16 × 16
Fuel assemblies		
Width dimension, in.	8.12	8.23
Assembly weight, lb (typical)	1204	1435
Overall length, in. (typical)	157	177
Rod replacement capabilities	Yes	Yes
Disassembly capabilities	Yes	Yes
Fuel rods		
Date of introduction (first criticality)	11/3/72	12/6/78
Number per assembly (unshimmed)	176	236
Rod pitch, in.	0.580	0.5063
Rod length, in. (typical)	146	161
Active fuel length, in.	136.7	150
OD, in.	0.440	0.382
Diametral gap, in.	0.0075	0.0070
Clad thickness, in.	0.028	0.025
Clad material (composition)	Zircaloy-4	Zircaloy-4
Total weight/rod, lb	6.7	5.7
Fuel pellets		
Density, % theoretical	95	95
Diameter, in.	0.3765	0.325
Length, in.	0.450	0.390
Total weight/rod, lb	5.4	4.5
Guide tubes[b]		
Number	5	5
OD, in	1.115	0.980
Wall thickness, in.	0.040	0.040
Tie plate		
Material	304 SS	304 SS
Total weight/assembly, lb	NA[c]	NA
Spacers		
Number (top and bottom)	2	2
Material (composition)	Al_2O_3	Al_2O_3
Total weight/rod, lb	0.004	0.005
Plenum springs		
Working length, in.	8.6	10.0
Material (composition)	SS	SS
Total weight/rod, lb	0.05	0.07
Miscellaneous		
Prepressurized to atm (typical)	Variable	Variable
Gas used	100% He	100% He

[a]Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.

[b]Guide tubes may be used to guide the control rod assembly or to contain instrumentation which is located in the center guide tube.

[c]Not available.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 2.4. Number of Combustion Engineering PWR fuel assemblies active and discharged[a]

Reactor	Core assemblies per cycle	Total active and discharged		
		14 × 14	16 × 16	15 × 15
Arkansas Nuclear One-2	177	-	345	-
Calvert Cliffs 1	217	693	-	-
Calvert Cliffs 2	217	609	-	-
Fort Calhoun	133	289	-	-
Maine Yankee	217	650	-	-
Millstone 2	217	361	-	-
Palisades	204	-	-	272
St. Lucie-1	217	497	-	-
St. Lucie-2	217	-	297	-
SONGS-2	217	-	217	-
SONGS-3	217	-	217	-

[a]Source: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 1985.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 2.5. Mechanical design parameters for Babcock and Wilcox PWR fuel assemblies[a]

Design component	Rod array		
	15 × 15	17 × 17	15 × 15 SS
Assembly			
Transverse dimension, in.	8.536	8.536	8.466
Assembly weight, lb	1515	1506	NA[b]
Overall length, in.	165-5/8	165-23/32	137.066 + .565 spring protrusion
Rod replacement capabilities	None	None	Grippable top end
Disassembly capabilities	None	None	Locking cups on upper nuts
Fuel rods			
Date of introduction	1971	1976	1976
Number per assembly	208	264	204
Rod pitch, in.	0.568	0.502	0.563
Length, in.	153.68	152.688	126.68
Fueled length, in.	141.8	143.0	120.5
OD, in.	0.430	0.379	0.422
Diametral gap, in.	0.0084	0.0078	0.0065
Clad thickness, in.	0.0265	0.0240	0.0165
Clad material	Zircaloy-4	Zircaloy-4	304 SS
Total weight/rod, lb	7.0	4.9	5.9
Fuel pellets			
Density, % TD	95	95	95
Diameter, in.	0.3686	0.3232	0.3825
Length, in.	0.600	0.375	0.458
Total weight/rod, lb	5.58	Unavailable	Unavailable
Guide tubes			
Number	16	24	20
OD, in.	0.530	0.564	0.543, upper 106.8 in. 0.479, lower 20.95 in.
Wall thickness, in.	0.016	0.0175	0.012
Weight/assembly with end plugs, lb	16.5	24	17
Material	Zircaloy-4	Zircaloy-4	304 SS
Instrument tubes			
Number	1	1	1
OD, in.	0.493	0.420	0.422
Material (composition)	Zircaloy-4	Zircaloy-4	304 SS
Total weight/assembly, lb	0.7	0.7	0.78
Tie plate			
Material	NA	NA	NA
Spacers			
Number	3	3	--
Material (composition)	Zircaloy-4	Zircaloy-4	--
Total weight/rod, lb	.028	Unavailable	--
Plenum springs			
Working length, in.	7.435	5.9735	5.01
Material (composition)	302 SS	302 SS	302 SS
Total weight/rod, lb	Unavailable	Unavailable	Unavailable
Miscellaneous			
Prepressurized to, psig	465	435	40
Gas used	Helium	Helium	Helium

[a]Source: K. O. Stein, Nuclear Power Division, Babcock and Wilcox, letter to J. W. Roddy, Oak Ridge National Laboratory, January 25, 1985.
[b]Not available.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 2.6. Control and burnable poison rods in PWRs used by Babcock and Wilcox[a]

Design component	Rod array		
	Standard	Long life	
	15 × 15		17 × 17
Control rod assembly			
Clad material	304 SS	UNS N06625[b]	304 SS
Clad length, in.	145.5	147.5	148-7/8
Clad OD, in.	0.440	0.441	0.377
Clad ID, in.	0.398	0.396	0.310
Pellet material	Ag-In-Cd	Ag-In-Cd	B_4C
Pellet OD, in.	0.392	0.386	0.285
Prepressure	1 atm He	465 psig He	1 atm He
Plenum volume, $in.^3$	0.4214	--	0.7075
Assembly weight, lb	130	130	65
Pellet stack length, in.	134	139	139
Burnable poison rod assembly			
Clad material	Zircaloy-4		Zircaloy-4
Clad length, in.	147-1/4		148
Clad OD, in.	0.430		0.371
Clad ID, in.	0.360		0.309
Pellet material	Al_2O_3–B_4C		Al_2O_3–B_4C
Pellet OD, in.	0.340		0.293
Prepressure	1 atm He		1 atm He
Plenum volume, $in.^3$	0.840		0.8774
Assembly weight, lb	57		60
Pellet stack length, in.	126		126

[a]Source: K. O. Stein, Utility Power Generation Division, Babcock and Wilcox, letter to J. W. Roddy, Oak Ridge National Laboratory, January 25, 1985.

[b]NiCrMoCb alloy.

Table 2.7. Number of PWR fuel assemblies shipped by Babcock and Wilcox[a]

Reactor	Rod array		
	15 × 15	17 × 17	15 × 15 SS
Oconee 1	646	—	—
Oconee 2	533	2 MkC 2 MkCR	—
Oconee 3	521	—	—
ANO-1 Unit 1	493	—	—
Rancho Seco	432	—	—
Davis Besse	317	—	—
Crystal River	437	—	—
TMI-1	385	—	—
Conn Yankee	—	—	368
TVA Bellefonte I	—	205	—
TVA Bellefonte II	—	205	—

[a]Source: K. O. Stein, Nuclear Power Division, Babcock and Wilcox, letter to J. W. Roddy, Oak Ridge National Laboratory, January 25, 1985.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 2.8. Mechanical design parameters for Exxon Nuclear PWR fuel assemblies[a]

Design component	Rod array			
	14 × 14	15 × 15	17 × 17	14 × 14[b]
Assembly				
Transverse dimension, in.	7.763	8.426	8.426	8.105
Assembly weight, lb	NA[c]	1425	NA	1280
Overall length, in.	162	162	162	157
Rod replacement capability	Yes	Yes	Yes	Yes
Disassembly capability	Yes	Yes	Yes	Yes
Fuel rods				
Number per assembly	179	204	264	176
Rod pitch, in.	0.556	0.563	0.496	0.580
Length, in.	152	152	152	147
Fueled length, in.	144	144	144	137
OD, in.	0.417/0.424	0.424	0.360/0.376	0.440
Diametral gap, in.				
Clad thickness, in.	0.0295/0.030	0.030	0.025/0.024	0.031
Clad material	Zr-4	Zr-4	Zr-4	Zr-4
Total weight/rod, lb	NA	NA	NA	NA
Fuel pellets				
Type	UO_2	UO_2	UO_2	UO_2
Density, % TD	94	94	94	94
Diameter, in.	0.3505/0.3565	0.3565	0.303/0.321	0.370
Length, in.	NA	NA	NA	NA
Total weight/rod, lb	NA	NA	NA	NA
Spacers				
Number	7	7	7	7
Material	Zr-4/Inconel-718	Zr-4/Inconel-718	Zr-4/Inconel-718	Zr-4/Inconel-718
Total weight/rod, lb	2–3	2–3	2–3	2–3
Plenum springs				
Working length, in.	NA	NA	NA	NA
Material	Inconel-718	Inconel-718	Inconel-718	Inconel-718
Miscellaneous				
Prepressurization, atm	>20	>20	>20	>20
Gas used	Helium	Helium	Helium	Helium
Guide tubes				
Number	16	20	24	5
OD, in.	0.541	0.544	0.480	1.115
Wall thickness, in.	0.017	0.0165	0.016	0.040
Material	Zr-4	Zr-4	Zr-4	Zr-4
Instrument tubes				
Number	1	1	1	NA
OD, in.	NA	NA	NA	NA
Material	Zr-4	Zr-4	Zr-4	Zr-4
Tie plate				
Material	SS 304L, Inconel springs	SS 304L, Inconel springs	SS 304L, Inconel springs	SS 304L, Inconel springs
Total weight/assembly, lb	25	25	25	25

[a]Source: C. J. Busselman, Exxon Nuclear Company, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, March 28, 1985.
[b]Produced only for Combustion Engineering.
[c]Not available.

Note: Exxon Nuclear Company, Inc. has become Advanced Fuels Corp. (Siemens).

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 5.10. Spent Fuel Disassembly Hardware for Major BWR Assembly Types. Listing by Assembly Class.

Class Name / Assembly Type Name	Hardware Weight[a]	Hardware Materials	Comments
GE BWR/2,3 Assembly Class			
GE BWR/2,3 7 x 7 GE-2a	8.4 kg	20% Zirc, 4% Inc, 76% SS	b
GE BWR/2,3 7 x 7 GE-2b	8.44 kg	20% Zirc, 4% Inc, 76% SS	
GE BWR/2,3 7 x 7 GE-3	8.3 kg	20% Zirc, 4% Inc, 76% SS	
GE BWR/2,3 8 x 8 GE-4	9.9 kg	28% Zirc, 4% Inc, 68% SS	c
GE BWR/2,3 8 x 8 GE-5	10.9 kg	35% Zirc, 3% Inc, 62% SS	b,c
GE BWR/2,3 8 x 8 GE Prepressurized	10.9 kg	35% Zirc, 3% Inc, 62% SS	b,c
GE BWR/2,3 8 x 8 GE Barrier	10.9 kg	35% Zirc, 3% Inc, 62% SS	b,c
GE BWR/2,3 8 x 8 GE-8	12.7 kg	44% Zirc, 3% Inc, 53% SS	
GE BWR/2,3 7 x 7 ANF	13.6 kg	41% Zirc, 5% Inc, 55% SS	
GE BWR/2,3 8 x 8 ANF	8.1 kg	28% Zirc, 5% Inc, 67% SS	
GE BWR/2,3 9 x 9 ANF	9.3 kg	27% Zirc, 6% Inc, 67% SS	
GE BWR/4,5,6 Assembly Class			
GE BWR/4,5,6 7 x 7 GE-2	8.4 kg	20% Zirc, 4% Inc, 76% SS	
GE BWR/4,5,6 7 x 7 GE-3a	8.4 kg	20% Zirc, 4% Inc, 76% SS	
GE BWR/4,5,6 7 x 7 GE-3b	8.4 kg	20% Zirc, 4% Inc, 76% SS	
GE BWR/4,5,6 8 x 8 GE-4a	10.0 kg	28% Zirc, 4% Inc, 68% SS	c
GE BWR/4,5,6 8 x 8 GE-4b	10.0 kg	28% Zirc, 4% Inc, 68% SS	b,c
GE BWR/4,5,6 8 x 8 GE-5	11.0 kg	35% Zirc, 3% Inc, 62% SS	c
GE BWR/4,5,6 8 x 8 GE Prepressurized	11.0 kg	35% Zirc, 3% Inc, 62% SS	b,c
GE BWR/4,5,6 8 x 8 GE Barrier	11.0 kg	35% Zirc, 3% Inc, 62% SS	b,c
GE BWR/4,5,6 8 x 8 GE-8	12.9 kg	44% Zirc, 3% Inc, 53% SS	b,c,d
GE BWR/4,5,6 8 x 8 ANF	9.0 kg	35% Zirc, 4% Inc, 61% SS	
GE BWR/4,5,6 9 x 9 ANF	9.3 kg	28% Zirc, 6% Inc, 66% SS	

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

[b] Estimated on the basis on similar assemblies.

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

[d] Four water rods assumed.

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

TABLE 1

SUMMARY OF GENERAL ELECTRIC BWR FUEL DESIGNS(a)

Fuel Rod Array	7 x 7		7 x 7R	8 x 8	8 x 8R
Introduction Date	1966	1968	1972	1973	1977
Fuel Rod OD (cm)	1.430	1.448	1.430	1.252	1.227
Fuel Rod ID (cm)	1.268		1.242	1.080	1.064
Nominal Cladding Thickness (mil)	32	35.5	37	34	32
Nominal Diametral Gap (mil)	11	12	12	9	9
Pellet Type	long, sharp corners		short, chamfered		
Hydrogen Getter	No		Yes	Yes	Yes
Peak Linear Power (W/cm)	607		607	440	440
Prepressurized to 3 atm	No		No	No	Yes(b)
Cumulative Fuel Assemblies(c)	10,289		5824	10,731	1898
Assemblies Sipped at Least Once(d)	10,289		5793	5698	7
Estimated Rod Failure Rate (%)	0.98		0.04	0.03	0

(a) Adapted from Reference 6.
(b) Starting with Fall 1979 deliveries.
(c) Fabricated and put into operation as of Spring 1979.
(d) See Section III.A.3.b for an explanation of sipping.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, October, 1983.

TABLE 3

MECHANICAL DESIGN PARAMETERS OF BWR FUEL ASSEMBLIES*

Rod Array	General Electric 7 x 7	General Electric 8 x 8	Exxon 8 x 8
Fuel Assemblies			
Transverse Dimension (in., cm)	5.518(14.016)	5.518(14.016)	
Assembly Weight (lb, kg)	600(272.16)	600(272.16)	
Overall Assembly Length (in., cm)	171.2(434.8)	171.2-178.5 (434.8-452.6)	
Fuel Rods			
Number per Assembly	49	63	60
Rod Pitch (in., cm)	0.738(1.874)	0.640(1.626)	0.842(2.139)
Length (in., cm)	161.1(409.2)	161.1(409.2)	156.9(398.5)
Fueled Length (in., cm)	144(365.8)	146(370.8)	144(365.8)
OD (in., cm)	0.563(1.430)	0.493(1.252)	0.5015(1.274)
Diametral Gap (in., cm)	0.012(0.0305)	0.009(0.0229)	0.010(0.0254)
Cladding Thickness (in., cm)	0.032(0.0813)	0.034(0.0864)	0.036(0.0914)
Cladding Material	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel Pellets			
Density	95	95	95
Diameter (in., cm)	0.487(1.237)	0.416(1.057)	0.4195(1.066)
Length (in., cm)	0.500(1.270)	0.420(1.067)	0.320(0.813)
Tie Plate			
Material	304 SS	304 SS	
Spacers			
Number	7	7	
Material	Zircaloy-4	Zircaloy-4	
Springs	Inconel	Inconel	
Plenum Springs			
Working Length (in., cm)	10.6(26.9)	10.6-16.0 (26.9-30.6)	
Material	Inconel	Inconel	
Compression Springs			
Working Length (in., cm)	0.94(2.39)	0.84(2.13)	
Material	Inconel	Inconel	

*Updated from Reference 3.

R.E. Woodley, *The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories*, HEDL-TME 83-28, October, 1983.

Table 2.9. General Electric BWR product lines and characteristics[a]

Product line class	Year of introduction	Plants and characteristics
BWR/1	1955	Dresden-1, Big Rock Point, Humboldt Bay, KRB — Initial commercial BWRs — First internal steam separation
BWR/2	1963	Oyster Creek — The first turnkey plant — Elimination of dual cycle
BWR/3	1965	Dresden-2 — The first jet pump application — Improved emergency core cooling system (ECCS)
BWR/4	1966	Browns Ferry — Increased power density 20%
BWR/5	1969	Zimmer — Improved safeguards — Valve flow control
BWR/6	1972	BWR/6 — 8 × 8 fuel bundle — Added fuel bundles, increased output — Improved recirculation system performance — Improved ECCS performance — Reduced fuel duty

[a]Source: E. D. Fuller, J. R. Finney, and H. E. Streeter, EWR/6 Nuclear System from General Electric — A Performance Description, NEDO-10569A, April 1972.

Table 2.10. Summary of General Electric BWR reactor fuel designs[a]

Design component	Rod array				
	7 × 7		7 × 7R	8 × 8	8 × 8R
Introduction date	1966	1968	1972	1973	1977
Fuel rod OD, in.	0.563	0.570	0.563	0.493	0.483
Fuel rod ID, in.	0.499		0.0489	0.425	0.419
Nominal cladding thickness, mil	32	35.5	37	34	32
Nominal diametral gap, mil	11	12	12	9	9
Pellet type	Long, sharp corners		Short, chamfered		
Hydrogen getter	No		Yes	Yes	Yes
Peak liner power, W/cm	607		607	440	440
Prepressurized to 3 atm	No		No	No	Yes

[a]Source: R. E. Woodley, The Characteristics of Spent LWR Fuel Relevant to Its Storage in Geologic Formations, HEDL-TME 83-28, October 1983.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 2.11. Mechanical design parameters of BWR fuel assemblies[a]

Design component	Rod array		
	BWR/1–5 (General Electric)		QUAD+ (Westinghouse)
	7 × 7	8 × 8	8 × 8
Fuel assemblies			
Transverse dimension, in.	5.518	5.518	5.50
Assembly weight, lb	600	600	600
Overall assembly length, in.	171.2	171.2–178.5	175.5
Fuel rods			
Number per assembly	49	62–63	64
Rod pitch, in.	0.738	0.640	0.609
Length, in.	161.1	161.1	160.6
Fueled length, in.	144–146	144–146	150
OD, in.	0.563–0.570	0.483–0.493	0.458
Diametral gap, in.	0.011–0.012	0.009	0.083
Cladding thickness, in.	0.032–0.037	0.032–0.034	0.029
Cladding material	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel pellets			
Density, % TD	95	95	95
Diameter, in.	0.487	0.416	0.3913
Length, in.	0.500	0.420	0.470
Tie plate			
Material	304 SS	304 SS	304 SS
Spacers			
Number	7	7	
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Springs	Inconel	Inconel	Zircaloy-4
Plenum springs			
Working length, in.	10.6	10.6-16.0	9.56
Material	Inconel	Inconel	302 SS
Compression springs			
Working length, in.	0.94	0.84	0.84
Material	Inconel	Inconel	Inconel

[a]Source: R. E. Woodley, The Characteristics of Spent LWR Fuel Relevant to Its Storage in Geologic Formations, HEDL-TME 83-28, October 1983 and E. M. Greene, Spent Fuel Data for Waste Storage Programs, HEDL-TME 79-20, September 1980.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 2.12. Mechanical design parameters for Exxon Nuclear BWR fuel assemblies[a]

Design component	Replacement array		
	BWR/1–5	BWR/2–6	New
	7 × 7	8 × 8	9 × 9
Fuel assemblies			
Transverse dimension, in.	5.25	5.25	5.25
Assembly weight, lb	590	580	570
Overall length, in.	174	174	174
Rod replacement capability	Yes	Yes	Yes
Disassembly capability	Yes	Yes	Yes
Fuel rods			
Date of introduction	1971	1974	1981
Number per assembly	49	63	79
Rod pitch, in.	0.73	0.64	0.57
Length, in.	145	145	145
OD, in.	0.59	0.48	0.42
Diametral gap, in.		None at end of life	
Clad thickness, in.	0.03	0.03	0.03
Clad material	Zr-2	Zr-2	Zr-2
Total weight/rod, lb	12.0	9.0	7.0
Fuel pellets			
Type	UO_2	UO_2	UO_2
Density, % TD	94	94	94
Diameter, in.	0.49	0.40	0.36
Length, in.	NA[b]	NA	NA
Total weight/rod, lb	NA	7.0	5.7
Plenum springs			
Working length, in.	10	10	13
Material	Inconel	Inconel	Inconel
Total weight/rod, lb	0.09	0.09	0.09
Compression springs			
Working length, in.	0.8	0.9	1.3
Material	Inconel	Inconel	Inconel
Total weight/rod, lb	0.007	0.007	0.007
Tie plate			
Material	CF-3 (304L)	CF-3 (304L)	CF-3 (304L)
Weight, lb	12	12	12

[a]Source: G. J. Busselman, Exxon Nuclear Company, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, March 28, 1985.
[b]Not available.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 2.13. Mechanical design parameters for Allis-Chalmers BWR fuel assemblies[a]

Design component	Rod array (10 × 10)
Fuel assemblies	
Transverse dimension, in.	NA[b]
Assembly weight, lb	NA
Overall assembly length, in.	NA
Fuel rods	
Number per assembly	100
Rod pitch, in.	0.565
Length, in.	NA
Fueled length, in.	83
OD, in.	0.396
Diametral gap, in.	0.006
Cladding thickness, in.	0.020
Cladding material	348 H SS
Fuel pellets	
Density, % TD	95
Diameter, in.	0.350
Length, in.	0.350–1.050
Tie plate	
Material	304 SS
Spacers	
Number	NA
Material	NA
Springs	NA
Plenum springs	
Working length, in.	NA
Material	NA
Compression springs	
Working length, in.	NA
Material	NA

[a]Source: Allis-Chalmers, Initial Testing of the La Crosse Boiling Water Reactor, ACNP-67533, December 1967.

[b]Not available.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 5.4 Non-Zircaloy Clad Fuels from Commercial LWRs

Assembly Class	Historical Discharges	Number Reprocessed	Historical (In Storage)	Projected	Total
		STAINLESS STEEL CLAD			
Big Rock Point (reprocessed)	66	66	0	0	0
Dresden-1 (reprocessed)	110	109	1 (<1 MT)	0	1 (<1 MT)
Haddam Neck (SS 304)	734	0	734 (303 MT)	673 (246 MT)	1407 (549 MT)
Humboldt Bay (reprocessed)	189	189	0	0	0
Indian Point-1 (SS 304)	280	120	160 (31 MT)	0	160 (31 MT)
San Onofre- (SS 304)	468	0	468 (171 MT)	496 (183 MT)	964 (354 MT)
Yankee Rowe (SS 304)	438	362	76 (21 MT)	0	76 (21 MT)
BWR TOTALS	365	364	1 (< 1 MT)	0	1 (< 1 MT)
PWR TOTALS	1920	482	1438 (526 MT)	1169 (429 MT)	2607 (955 MT)
GRAND TOTAL	2285	846	1439 (526 MT)	1169 (429 MT)	2608 (955 MT)
		ZIRLO CLAD			
WE 15 x 15 North Anna	0	0	0	0	2 + Future Use

Note: Historical Discharges and Number of Assemblies Reprocessed are estimates, based on continuing investigation into fuels reprocessed at West Valley.

Stainless cladding that was not reprocessed is SS 304.

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

TABLE 3-2

CHARACTERISTICS OF REFERENCE PWR CONTROL ROD ASSEMBLIES

Overall Length (in)	161
Length of Spider (in)	8
Length of Control Rods (in)	153
Diameter of Control Rods (in)	0.385
Overall Weight (lb)	149
Weight of Spider (lb)	8
Weight of Control Rods (lb)	141
No. of Control Rods	24
Weight/Control Rod (lb)	5.9
No. CRAs in Core	61
Lifetime (yrs)	15 (2 sets for life of reactor)

TABLE 3-4

CHARACTERISTICS OF REFERENCE PWR BURNABLE POISON ASSEMBLIES

	Westinghouse	B&W
Overall Length (in)	156	160
Length of Spider (in)	4	6
Length of Rods (in)	152	154
Diameter of Rods (in)	0.385	0.430
Overall Weight (lb)	156	57
Weight of Spider (lb)	9	8
Weights of Rods (lb)	147	49
No. of Rods	24	16
Weight/Rod (lb)	6.1	3.1
No. BPA + APSA in Core	96	76
Lifetime (yr)	3 (10 sets for life of reactor)	3 (10 sets for life of reactor)

E.R. Johnson Associates, Inc. (compilers), *Acceptance of Non-Fuel Assembly Hardware by the Federal Waste Management System*, ORNL/SUB./86-SA094/8, JAI-328, March, 1990.

TABLE 3-3

SUMMARY COMPARISON OF ATTRIBUTES OF BURNABLE POISON ASSEMBLIES IN PWRs[a]

	Fuel Array	Total Length (in)	Spider Length (in)	Total Weight (lb)	Poison	No. Poison Rods	No. Orifice Rods
Westinghouse							
4-Rod Neutron Poison	14x14	156	6	17	Borosilicate Glass	4	12
12-Rod Neutron Poison	14x14	156	6	32	Borosilicate Glass	12	4
BPA 16-Rod Neutron Poison	14x14	156	6	40	Borosilicate Glass	16	0
WABA 4-Rod Neutron Poison	14x14	154	4	16	B_4C	4	12
WABA 12-Rod Neutron Poison	14x14	154	4	28	B_4C	12	4
WABA 16-Rod Neutron Poison	14x14	154	4	34	B_4C	16	0
WABA 8-Rod Neutron Poison	15x15	154	4	23	B_4C	8	12
WABA 20-Rod Neutron Poison	15x15	154	4	41	B_4C	20	0
WABA 4-Rod Neutron Poison	15x15	147	4	17	B_4C	4	16
WABA 12-Rod Neutron Poison	15x15	147	4	28	B_4C	12	8
BPA 4-Rod Neutron Poison	15x15	156	4	20	Borosilicate Glass	4	16
BPA 10-Rod Neutron Poison	15x15	156	4	34	Borosilicate Glass	10	10
BPA 20-Rod Neutron Poison	15x15	156	4	54	Borosilicate Glass	20	0
WABA 4-Rod Neutron Poison	17x17	154	4	17	B_4C	4	20
WABA 16-Rod Neutron Poison	17x17	154	4	37	B_4C	16	8
WABA 24-Rod Neutron Poison	17x17	154	4	51	B_4C	24	0
BPA 4-Rod Neutron Poison	17x17	156	4	19	Borosilicate Glass	4	20
BPA 10-Rod Neutron Poison	17x17	156	4	28	Borosilicate Glass	10	14
BPA 16-Rod Neutron Poison	17x17	156	4	38	Borosilicate Glass	16	8
BPA 24-Rod Neutron Poison	17x17	157	5	50	Borosilicate Glass	24	0
B&W							
Ax Pwr Shaping Assembly	15x15	160	5	71	Inconel 600	16	0
Ax Pwr Shaping Assembly	15x15	160	4	57	Ag-In-Cd	16	0
Burnable Poison Assembly	15x15	154	6	57	B_4C; Al_2O_3	16	0
Combustion Engineering							
None - Burnable Poison In Fuel							

[a] Source: DOE/RW-0184, Vol. 5

TABLE 3-5

SUMMARY COMPARISON OF ATTRIBUTES OF NEUTRON SOURCE ASSEMBLIES IN PWRs[a]

Type	Fuel Array	Total Length (in)	Spider Length (in)[b]	Total Weight (lb)	Source	Burnable Poison	No. Burnable Poison Rods	No. Orifice Rods	No. Source Rods	
									Primary	Secondary
Westinghouse										
Version 1	14x14	125	3	48	Cf/Sb-Be	Borosil Glass	12	0	1	3
Version 2	14x14	158	3	48	Cf/Sb-Be	Borosil Glass	12	0	1	3
Version 3	14x14	158	3	48	Pu-Be/Sb-Be	Borosil Glass	12	0	1	3
116-Inch Source	14x14	116	3	22	Sb-Be	N/A	0	12	0	4
137-Inch Source	14x14	138	3	24	Sb-Be	N/A	0	12	0	4
143-Inch Source	14x14	143	3	23	Sb-Be	N/A	0	12	0	4
157-Inch Source	14x14	157	3	28	Sb-Be	N/A	0	12	0	4
Version 1	15x15	158	3	52	Pu-Be/Sb-Be	Borosil Glass	12	4	1	3
Version 2	15x15	158	3	52	Po-Be/Sb-Be	Borosil Glass	12	4	1	3
Version 3	15x15	156	3	34	Cf/Ag-In-Cd	Borosil Glass	12	7	1	0
116-Inch Source	15x15	116	3	28	Sb-Be	N/A	0	16	0	4
143-Inch Source	15x15	143	3	25	Sb-Be	N/A	0	16	0	4
157-Inch Source	15x15	156	3	19	Sb-Be	N/A	0	16	0	4
Version 1	17x17	156	3	39	Cf/Sb-Be	Borosil Glass	12	8	1	3
Version 2	17x17	156	3	34	Cf	Borosil Glass	12	11	1	0
Version 3	17x17	156	3	51	Cf	Borosil Glass	23	0	1	0
On-Spider	17x17	157	3	24	Sb-Be	N/A	0	20	0	4
12 BP Rod	17x17	156	3	39	Sb-Be	Borosil Glass	12	8	0	4
20 BP Rod	17x17	156	3	51	Sb-Be	Borosil Glass	20	0	0	4
B&W										
Primary	15x15	146	4	1	Am-Be	N/A	0	0	1	0
Regenerative	15x15	145	4	46	Sb-Be	N/A	0	0	0	8
Combustion Engineering										
Standard	16x16	99	3	8	Pu-Be/Sb-Be	N/A	0	0	1	1
128-Inch Core	14x14	100	3	11	Pu-Be/Sb-Be	N/A	0	0	1	1
137-Inch Core	14x14	106	3	11	Pu-Be/Sb-Be	N/A	0	0	1	1
Sustaining	15x15	117	3	5	Sb-Be	N/A	0	0	0	1
Startup Source	15x15	117	3	5	Pu-Be/Sb-Be	N/A	0	0	1	1

[a] Source: DOE/RW-0184, Vol. 5

[b] Estimated

TABLE 3-6

CHARACTERISTICS OF REFERENCE PWR NEUTRON SOURCE ASSEMBLIES

Overall Length (in)	157
Length of Spider (in)	3
Length of Rods (in)	154
Diameter of Rods (in)	0.385
Overall Weight (lb)	51
Weight of Spider (lb)	8
Weight of Rods (lb)	43
No. of Rods	24
Weight/Rod (lb)	1.8
No. NSAs in Core	4
Lifetime (yrs)	(1 set for life of reactor)

TABLE 3-8

CHARACTERISTICS OF REFERENCE PWR THIMBLE PLUG ASSEMBLIES

Overall Length (in)	12
Length of Spider (in)	4
Length of Rods (in)	8
Diameter of Rods (in)	0.424
Overall Weight (lb)	13
Weight of Spider (lb)	9
Weight of Rods (lb)	4
No. of Rods	24
Weight/Rod (lb)	0.16
No. TPAs in Core	36
Lifetime (yrs)	Plant Life

E.R. Johnson Associates, Inc. (compilers), *Acceptance of Non-Fuel Assembly Hardware by the Federal Waste Management System*, ORNL/SUB./86-SA094/8, JAI-328, March, 1990.

TABLE 3-7

SUMMARY COMPARISON OF ATTRIBUTES OF THIMBLE PLUG ASSEMBLIES IN PWRs[a]

	Fuel Array	Total Length (in)	Spider Length (in)[b]	Total Weight (lb)	No. Orifice Rods
Westinghouse					
Water Displacement	14x14	156	4	21	16
Standard	14x14	12	4	10	16
Standard	15x15	10	4	11	20
Standard	17x17	12	4	13	24
B&W					
Standard	15x15	16	5	16	16
Combustion Engineering					
None Described					

[a] Source: DOE/RW-0184, Vol. 5

[b] Estimated

TABLE 3-9

CHARACTERISTICS OF REFERENCE BWR FUEL CHANNEL

Overall Length (in)	167
Wall Thickness (in)	0.120
Inside Width (in)	5.3
Overall Weight (lb)	98

TABLE 3-10

CHARACTERISTICS OF REFERENCE BWR CONTROL ASSEMBLY

Overall Length (in)	174
Length of Control Blades (in)	144
Length of Handle (in)	6
Length of Base (in)	24
Thickness of Handle & Blades (in)	0.26
Width of Handle & Blades (in)	9.81
Diameter of Base (in)	9.265 (w/o bearings) 10.182 (at bearing locations)
No. of Assemblies in Core	185
Lifetime (yrs)	3-25 (Assumes 2 sets for life of reactor)

TABLE 3-11

SUMMARY OF QUANTITIES OF NFA COMPONENTS ASSOCIATED WITH 70,000 MTU SNF

Component	Items
PWR - Control Rod Assemblies	10,000
PWR - Burnable Poison Assemblies	61,500
PWR - Neutron Source Assemblies	320
PWR - Thimble Plug Assemblies	2,900
BWR - Fuel Channels	110,000
BWR - Control Assemblies	14,500
BWR - Neutron Sources	Negligible
BWR - Instrumentation Assemblies	5,000
BWR - Poison Curtains	750

E.R. Johnson Associates, Inc. (compilers), *Acceptance of Non-Fuel Assembly Hardware by the Federal Waste Management System*, ORNL/SUB./86-SA094/8, JAI-328, March, 1990.

TABLE 4-1

SUMMARY OF THE QUANTITIES OF NFA COMPONENTS PROJECTED TO BE AVAILABLE FOR DELIVERY TO THE FWMS -- FOR CASES WHERE COMPONENTS ARE DELIVERED AS AN INTEGRAL PART OF THE FUEL ASSEMBLY AND WHERE THEY ARE DELIVERED IN EITHER AN UNCOMPACTED OR COMPACTED FORM[a,b]

	Total Units	Can Dimensions (in)	Can Capacity	Total No. Cans or Units	Weight of Loaded Can or Unit (lb)
As Integral Part of Fuel Assys					
PWR Control Rod Assemblies	10,000		In Fuel Assy	10,000	149
PWR Burnable Poison Assys (West)	55,000		In Fuel Assy	55,000	156
PWR Burnable Poison Assys (B&W)	6,500		In Fuel Assy	6,500	57
PWR Neutron Source Assemblies	320		In Fuel Assy	320	51
PWR Thimble Plug Assemblies	2,900		In Fuel Assy	2,900	13
BWR Fuel Channels	110,000		In Fuel Assy	110,000	98
BWR Control Assemblies[c]	14,500		No Can	14,500	225
BWR Instrument Assemblies	5,000		In Fuel Assy	5,000	2
BWR Poison Curtains[c]	750	10.5x10.5x176	74	11	2,263
Compacted					
PWR Control Rod Assys - Rod Sets	10,000	9x9x160	15	667	2,437
- Spiders	10,000	9x9x160	20	500	482
PWR Burnable Poison Assys (West)					
- Rod Sets	55,000	9x9x160	15	3,667	2,527
- Spiders	55,000	9x9x160	40	1,375	682
PWR Burnable Poison Assys (B&W)					
- Rod Sets	6,500	9x9x160	19	342	1,253
- Spiders	6,500	9x9x160	26	250	530
PWR Neutron Source Assemblies					
- Rod Sets	320	9x9x160	15	21	967
- Spiders	320	9x9x160	53	6	746
PWR Thimble Plug Assys - Rod Sets	2,900	9x9x160	114	25	778
- Spiders	2,900	9x9x160	40	73	682
BWR Fuel Channels	110,000	6x6x168	7	15,714	909
BWR Control Assemblies[d]	14,500	No Can	No Can	14,500	225
BWR Instrument Assemblies	5,000	6x6x160	47	106	306
BWR Poison Curtains	750	10.5x10.5x176	74	11	2,263
Uncompacted					
PWR Control Rod Assemblies	10,000	9x9x162	1	10,000	475
PWR Burnable Poison Assys (West)	55,000	9x9x160	1	55,000	478
PWR Burnable Poison Assys (B&W)	6,500	9x9x160	1	6,500	379
PWR Neutron Source Assemblies	320	9x9x160	1	320	373
PWR Thimble Plug Assemblies	2,900	9x9x160	13	223	491
BWR Fuel Channels	110,000	No Can	No Can	110,000	98
BWR Control Assemblies	14,500	No Can	No Can	14,500	225
BWR Instrument Assemblies	5,000	6x6x160	47	106	306
BWR Poison Curtains	750	10.5x10.5x176	74	11	2,263

[a] Assumes all NFA components listed are classified as greater-than-Class C waste.

[b] Quantities are estimated to be those equivalent to the production of a nominal 70,000 MTU of SNF assemblies.

[c] Not integral.

[d] Uncompacted.

E.R. Johnson Associates, Inc. (compilers), *Acceptance of Non-Fuel Assembly Hardware by the Federal Waste Management System*, ORNL/SUB./86-SA094/8, JAI-328, March, 1990.

TABLE 5-1

ESTIMATED NUMBER OF FROM-REACTOR SHIPMENTS REQUIRED FOR NFA HARDWARE

(Shipped Integral to Fuel Assembly and Separately in Both Canned and Uncompacted & Compacted Form)

	Total Amount (Assys/Units)	Can Dimensions (in)	Can Capacity	Total No. Cans or Uncanned Items	Weight Loaded Can or Unit (lb)	Cask Capacity (Cans/Items) Truck	Rail	Total Additional[a] Shipments Truck	Rail	Added Weight Shipped (CWT)[b] Truck	Rail
Integral With Fuel Assemblies											
PWR Control Rod Assemblies	10,000	Integral w/FA	Integral w/FA	10,000	149	Integral w/FA		Integral w/FA		6,705	8,195
PWR Burnable Poison Assys (West)	55,000	Integral w/FA	Integral w/FA	55,000	156	Integral w/FA		Integral w/FA		38,610	47,190
PWR Burnable Poison Assys (B&W	6,500	Integral w/FA	Integral w/FA	6,500	57	Integral w/FA		Integral w/FA		1,667	2,038
PWR Neutron Source Assemblies	320	Integral w/FA	Integral w/FA	320	51	Integral w/FA		Integral w/FA		73	90
PWR Thimble Plug assemblies	2,900	Integral w/FA	Integral w/FA	2,900	13	Integral w/FA		Integral w/FA		170	207
BWR Fuel Channels	110,000	Integral w/FA	Integral w/FA	110,000	98	Integral w/FA		Integral w/FA		48,510	59,290
BWR Control Assemblies[c]	14,500	No Can	No Can	14,500	225	1	12	6,525	665	14,681	17,944
BWR Instrument Assemblies	5,000	Integral w/FA	Integral w/FA	5,000	2	Integral w/FA		Integral w/FA		45	55
BWR Poison Curtains[c]	750	10.5x10.5x176	74	11	2,263	1	12	5	1	112	137
					Total			6,530	666	110,573	135,146
Compacted											
PWR Control Rod Assys - Rod Sets	10,000	9x9x160	15	667	2,437	4	21	75	17	7,315	8,940
- Spiders	10,000	9x9x160	20	500	482	4	21	56	13	1,084	1,326
PWR Burnable Poison Assys (West)											
- Rod Sets	55,000	9x9x160	15	3,667	2,527	4	21	413	96	41,699	50,966
- Spiders	55,000	9x9x160	40	1,375	682	4	21	155	36	4,220	5,158
PWR Burnable Poison Assys (B&W)											
- Rod Sets	6,500	9x9x160	19	342	1,253	4	21	38	9	1,928	2,357
- Spiders	6,500	9x9x160	26	250	530	4	21	28	7	596	729
PWR Neutron Source Assemblies											
- Rod Sets	320	9x9x160	15	21	967	4	21	3	1	93	113
- Spiders	320	9x9x160	53	6	746	4	21	1	1	20	25
PWR Thimble Plug Assys - Rod Sets	2,900	9x9x160	114	25	778	4	21	3	1	88	107
- Spiders	2,900	9x9x160	40	73	682	4	21	8	2	224	274
BWR Fuel Channels	110,000	6x6x168	7	15,714	909	9	48	786	180	64,278	78,562
BWR Control Assemblies[d]	14,500	No Can	No Can	14,500	225	1	12	6,525	665	14,681	17,944
BWR Instrument Assemblies	5,000	6x6x160	47	106	306	9	48	5	1	146	178
BWR Poison Curtains[d]	750	10.5x10.5x176	74	11	2,263	1	12	5	1	112	137
					Total			8,101	1,030	136,484	166,816
Uncompacted											
PWR Control Rod Assemblies	10,000	9x9x162	1	10,000	475	4	21	1,125	262	21,375	26,125
PWR Burnable Poison Assys (West)	55,000	9x9x160	1	55,000	478	4	21	6,188	1,440	118,305	144,595
PWR Burnable Poison Assys (B&W)	6,500	9x9x160	1	6,500	379	4	21	731	170	11,086	13,549
PWR Neutron Source Assemblies	320	9x9x160	1	320	373	4	21	36	9	537	656
PWR Thimble Plug Assemblies	2,900	9x9x160	13	223	491	4	21	25	6	493	602
BWR Fuel Channels	110,000	No Can	No Can	110,000	98	9	48	5,500	1,260	48,510	59,290
BWR Control Assemblies	14,500	No Can	No Can	14,500	225	1	12	6,525	665	14,681	17,944
BWR Instrument Assemblies	5,000	6x6x160	47	106	306	9	48	5	1	146	178
BWR Poison Curtains	750	10.5x10.5x176	74	11	2,263	1	12	5	1	112	137
					Total			20,140	3,814	215,245	263,076

[a] Number of shipments additional to the shipment of SNF assemblies over a period of 25 years. Assumes 45% of material is shipped by truck and 55% by rail.

[b] Empty can weights used are as follows: 9x9x162 in -- 326 lb; 9x9x160 in -- 322 lb; 6x6x160 in -- 212 lb; 6x6x168 in -- 223 lb; 10.5x10.5x176 in -- 413 lb

[c] Cannot be shipped integral with SNF assemblies.

[d] Assumes that it is not compacted at reactor site.

[e] No further compaction possible.

TABLE 6-8

TOTAL NUMBER OF CANS AND CAN SIZES FOR REPOSITORY DISPOSAL

(For Both MRS and No-MRS Options)

Description	Storage and/or Disposal of Reactor-Canned Uncompacted NFA Hardware[a]		Storage and/or Disposal of Reactor-Canned & Compacted NFA Hardware[b]		Storage and/or Disposal of Intact NFA Hardware Received Integral W/SNF Assemblies[c]		Storage and/or Disposal of FWMS-Compacted NFA Hardware Received Integral W/SNF Assemblies[d]		Storage and/or Disposal of FWMS-Compacted NFA Hardware Received In Canned Uncompacted Form[d]	
	Can Dimensions (in)	Total No. Cans	Can Dimensions (in)	Total No. Cans	Can Dimensions (in)	Total No. Cans	Can Dimensions (in)	Total No. Cans	Can Dimensions (in)	Total No. Cans
All PWR Hardware	9x9x160-162	72,043[e]	9x9x160	6,926[f]	-	-	9x9x160	6,926	9x9x160	6,926
BWR Fuel Channels	9x9x168	8,462	6x6x168	15,714[f]	-	-	9x9x168	8,462	9x9x168	8,462
BWR Control Assys	10.5x10.5x176	14,500	10.5x10.5x176	3,038	10.5x10.5x176	14,500	10.5x10.5x176	3,038	10.5x10.5x176	3,038
BWR Instrument Rods	9x9x160	89	6x6x160	106[f]	-	-	9x9x160	89	9x9x160	89
BWR Poison Curtains	10.5x10.5x176	11[f]	10.5x10.5x176	11[f]	10.5x10.5x176	11[f]	10.5x10.5x176	11[f]	10.5x10.5x176	11[f]
Total		95,105		25,795		14,511		18,526		18,526

[a] Assumes BWR fuel channels, and instrument rods are compacted and canned in FWMS facilities; and BWR control assemblies are canned there.

[b] Assumes BWR control assemblies are compacted and canned in FWMS facilities.

[c] Assumes BWR control assemblies are canned in FWMS facilities.

[d] Assumes all NFA hardware is compacted and canned in FWMS facilities.

[e] Received in canned form.

[f] Received in compacted and canned form.

TABLE 7-1

ESTIMATED NUMBER OF FROM-MRS SHIPMENTS REQUIRED FOR NFA HARDWARE

(For MRS Cases Only)

	Can Dimensions (in.)	No. Cans	Weight of Loaded Can (lb.)	Cask Capacity[e] (No. Cans)	Total Additional Shipments	Added Weight Shipped (CWT)
Storage and/or Disposal of Reactor-Canned Uncompacted NFA Hardware[a]						
All PWR Hardware	9x9x160-162	72,043	468 Av.	28	2,573	337,323
BWR Fuel Channels	9x9x168	8,462	1,611	28	302	136,323
BWR Control Assemblies	10.5x10.5x176	14,500	638	17	853	92,510
BWR Instrument Assemblies	9x9x160	89	502	28	3	447
BWR Poison Curtains	10.5x10.5x176	11	2,263	17	1	249
	Total	95,105			3,732	566,852
Storage and/or Disposal of Reactor-Canned and Compacted NFA Hardware[b]						
All PWR Hardware	9x9x160	6,926	1,837 Av.	28	247	127,262
BWR Fuel Channels	6x6x168	15,714	909	61	258	142,840
BWR Control Assemblies	10.5x10.5x176	3,038	1,487	17	179	45,175
BWR Instrument Assemblies	6x6x160	106	306	61	2	324
BWR Poison Curtains	10.5x10.5x176	11	2,263	17	1	249
	Total	25,795			687	315,850
Storage and/or Disposal of Intact NFA Hardware Received Integral With SNF Assemblies[c]						
BWR Control Assemblies	10.5x10.5x176	14,500	638	17	853	92,510
BWR Poison Curtains	10.5x10.5x176	11	2,263	17	1	249
	Total	14,511			854	92,759
Storage and/or Disposal of MRS-Compacted NFA Hardware[d]						
All PWR Hardware	9x9x160	6,926	1,837 Av.	28	247	127,262
BWR Fuel Channels	9x9x168	8,462	1,611	28	302	136,323
BWR Control Assemblies	10.5x10.5x176	3,038	1,487	17	179	45,175
BWR Instrument Assemblies	9x9x160	89	502	28	3	447
BWR Poison Curtains	10.5x10.5x176	11	2,263	28	1	249
	Total	18,526			732	309,456

[a] Assumes BWR fuel channels, and instrument assemblies are compacted and canned in FWMS facilities; and BWR control assemblies are canned there.

[b] Assumes BWR control assemblies are compacted and canned in FWMS facilities.

[c] Assumes BWR control assemblies are canned in FWMS facilities.

[d] Assumes all NFA hardware is compacted and canned in FWMS facilities.

[e] Capacity of cask described in Section 7.2 for NFA hardware of various forms.

Table 3.3. Assumed elemental compositions (g/ton of metal) of LWR fuel-assembly structural materials[a]

Element	Atomic number	Zircaloy-2	Zircaloy-4	Inconel-718	Inconel X-750	Stainless steel 302	Stainless steel 304	Nicrobraze 50
H	1	13	13	0	0	0	0	0
B	5	0.33	0.33	0	0	0	0	50
C	6	120	120	400	399	1,500	800	100
N	7	80	80	1,300	1,300	1,300	1,300	66
O	8	950	950	0	0	0	0	43
Al	13	24	24	5,992	7,982	0	0	100
Si	14	0	0	1,997	2,993	10,000	10,000	511
P	15	0	0	0	0	450	450	103,244
S	16	35	35	70	70	300	300	100
Ti	22	20	20	7,990	24,943	0	0	100
V	23	20	20	0	0	0	0	0
Cr	24	1,000	1,250	189,753	149,660	180,000	190,000	149,709
Mn[b]	25	20	20	1,997	6,984	20,000	20,000	100
Fe	26	1,500	2,250	179,766	67,846	697,740	688,440	471
Co[b]	27	10	10	4,694	6,485	800	800	381
Ni	28	500	20	519,625	721,861	89,200	89,200	744,438
Cu	29	20	20	999	499	0	0	0
Zr[b]	40	979,630	979,110	0	0	0	0	100
Nb	41	0	0	55,458	8,980	0	0	0
Mo	42	0	0	29,961	0	0	0	0
Cd	48	0.25	0.25	0	0	0	0	0
Sn	50	16,000	16,000	0	0	0	0	0
Hf	72	78	78	0	0	0	0	0
W	74	20	20	0	0	0	0	100
U	92	0.2	0.2	0	0	0	0	0
Density, g/cm^3	--	6.56	6.56	8.19	8.30	8.02	8.02	--

[a]Source: A. G. Croff, M. A. Bjerke, G. W. Morrison, and L. M. Petrie, Revised Uranium - Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code, ORNL/TM-6051, September 1978.

[b]Value used in ORIGEN should be less than this (actual) value if the materials are not in the active fuel zone.

Table 2.1. Fuel assembly materials[a]

Design component	Subcomponent	Alloy or material
Fuel pellets		Uranium dioxide
Fuel rods		Zircaloy-2 (BWR) Zircaloy-4 (PWR)
		304 SS, 348H
Fuel spacers	Grid	304 SS
		Inconel 718
		Zircaloy-4
	Springs	Inconel 718, 625
		Zircaloy-4
Upper tie plates	Bail/tie plate	304 SS
	Bolts/nuts	304 SS
		Inconel 600
	Springs	Inconel 718, X750
Lower tie plates	Tie plate/nozzle	304 SS, CF-8
Tie rods		Zircaloy-4 304 SS

[a]Source: E. M. Greene, Spent Fuel Data for Waste Storage Programs, HEDL-TME 79-20, September 1980.

K.J. Notz, T.D. Welch, R.S. Moore, and W.J. Reich, *Preliminary Waste Form Characteristics*, ORNL-TM-11681 (draft) September, 1990.

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

Physical Description Report Page: 1

Combustion Enigneering SYSTEM80 12-Rod Full-Length Control Element

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

Dimensions:

Total Length:	253	inches
Total Weight:	192.2	pounds

Cladding:

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

Poison:

Primary Material:	Boron Carbide (CE)	
Poison Length:	148	inches
Pellet Diameter:	0.737	inches

Plenum Spring Material: St. Steel 302

Spider Material: St. Steel 304

Number of Control Rods: 12

Life Expectancy: 4000 EFPD

K.J. Notz, *Characteristics of Potential Repository Waste,* DOE/RW-0184-R1, V.1 (draft), July, 1990.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

TABLE A1

RATES AND OXIDATION DEPTHS OCCURRING IN THE LOW-TEMPERATURE CORROSION OF ZIRCALOY CLADDING

Temperature (°C)	Corrosion Rate ($mg/dm^2 \cdot day$) Eq. 1	Eq. 2	Oxidation Depth (μm)* Eq. 1	Eq. 2
250	4.45×10^{-3}	1.75×10^{-3}	0.071	0.028
300	3.60×10^{-2}	2.35×10^{-2}	0.577	0.377
350	2.08×10^{-1}	2.09×10^{-1}	3.33	3.35
400	9.25×10^{-1}	1.34	14.8	21.5

*Under isothermal conditions for one year.

References

A1. E. Hillner, "Corrosion of Zirconium-Base Alloys - An Overview," Zirconium in the Nuclear Industry, ASTM STP 633, American Society for Testing and Materials, Philadelphia, PA, 1977.

A2. A. B. Johnson Jr, E. R. Gilbert and R. J. Guenther, Behavior of Spent Nuclear Fuel and Storage System Components in Dry Interim Storage, PNL-4189, Pacific Northwest Laboratory, Richland, WA, August 1982.

A3. D. G. Boase and T. T. Vandergraaf, "The Canadian Spent Fuel Storage Canister: Some Material Aspects," Nucl. Tech. 32, p. 60, 1977.

Eq 1. Weight gain rate ($mg/dm^2 \cdot day$) = $1.12 \times 10^8 \exp(-12{,}529/T)$ (Ref. A1)

Eq 2. Weight gain rate ($mg/dm^2 \cdot day$) = $1.53 \times 10^{10} \exp(-15{,}590/T)$ (Ref. A3)

T = temperature in °K

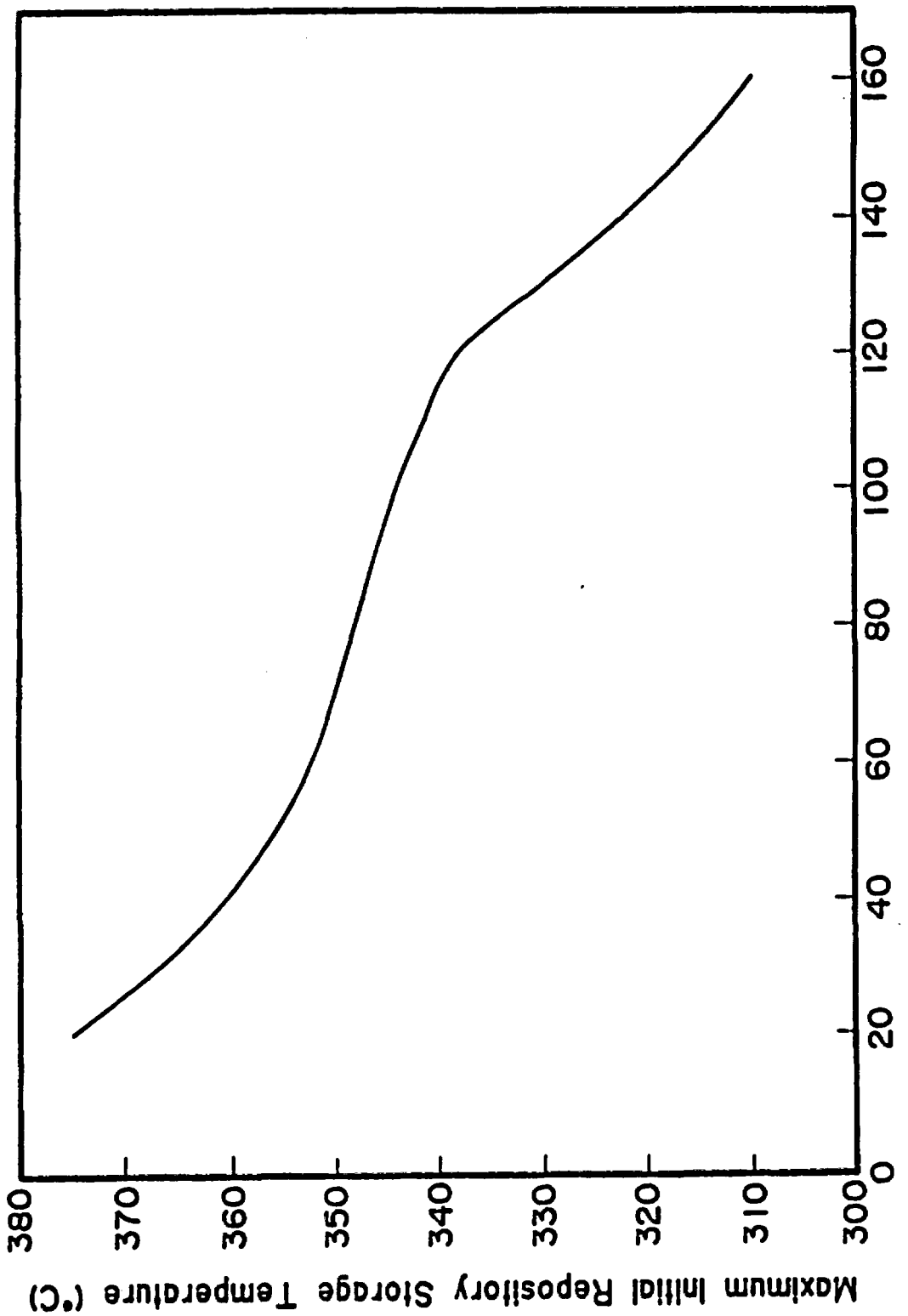
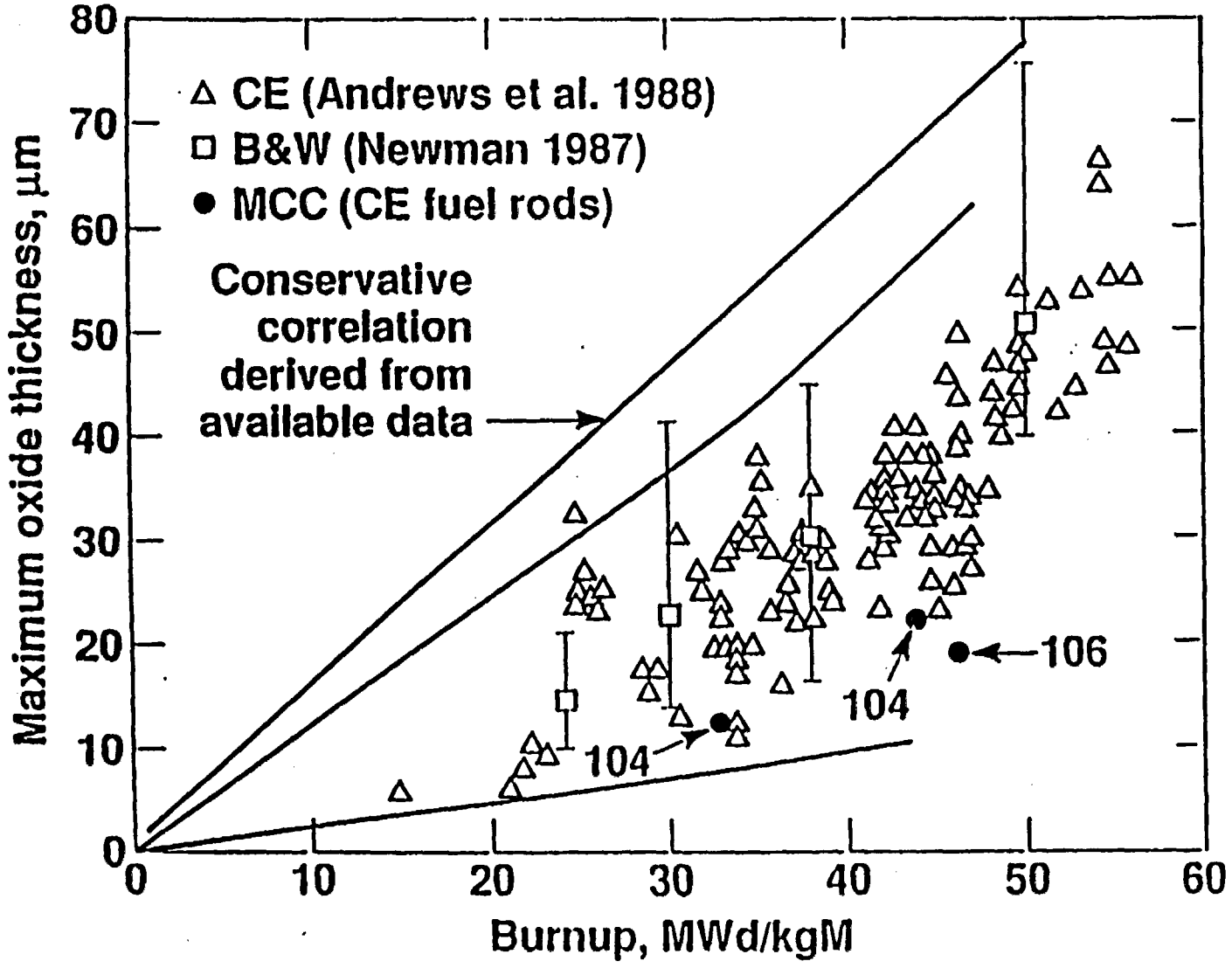


FIGURE 2. PREDICTED MAXIMUM INITIAL REPOSITORY STORAGE TEMPERATURE FROM A 1000 YEAR DEFORMATION AND FRACTURE MAP ANALYSIS. (No hydride failure mechanisms included.)

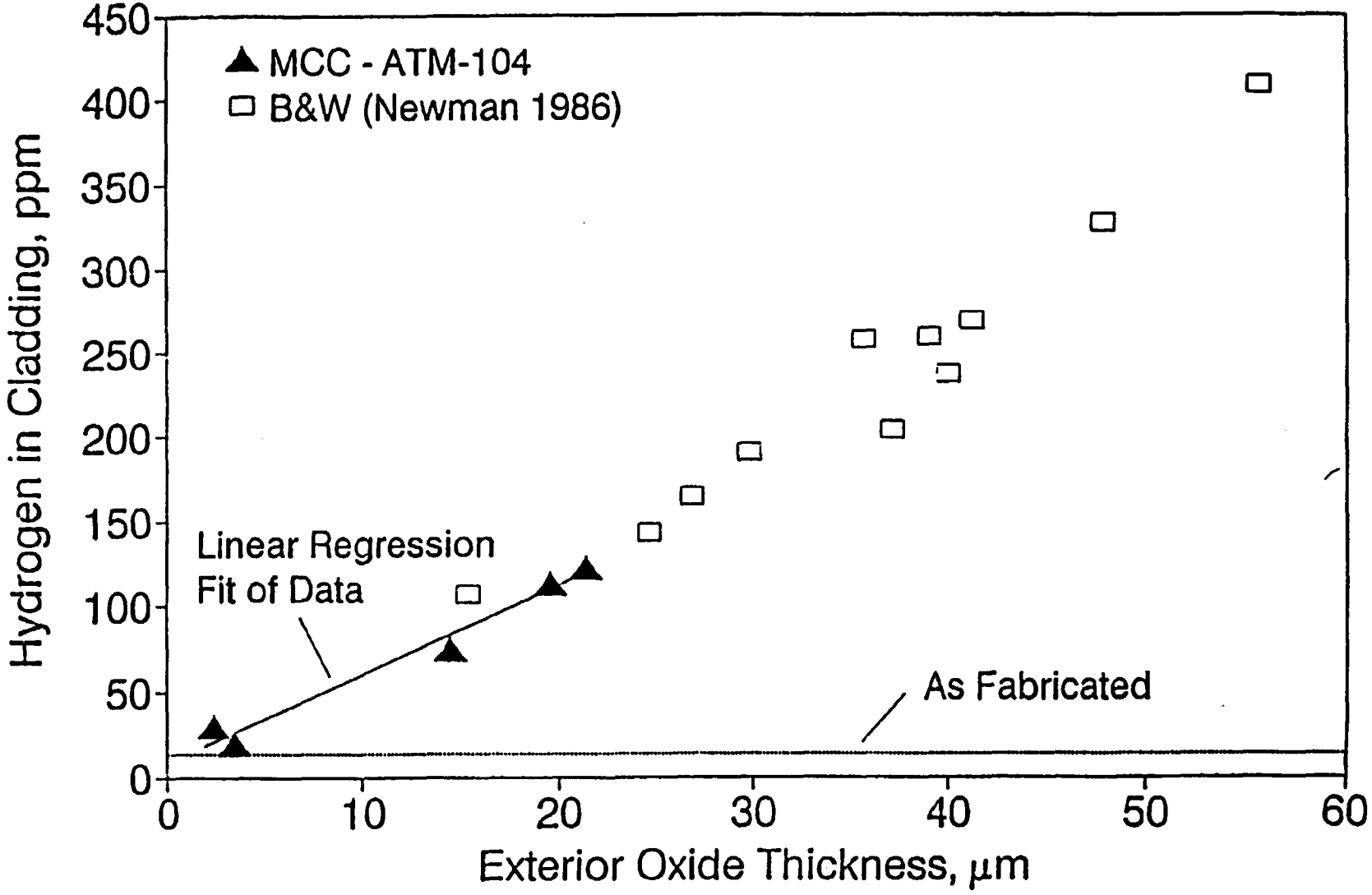
L. Santanam, H. Shaw, B.A. Chin, "Modeling of Zircaloy Cladding Degradation Under Repository Conditions," LLNL Report UCRL-100211, July 1989.

METHOD FOR CORRELATING MAXIMUM OXIDE THICKNESS WITH BURNUP



R.B. Stout, *Spent Fuel Characteristics Overview*, presented to the Nuclear Waste Technical Review Board, August, 1990.

Hydrogen Content in Cladding with a Range of Oxide Thicknesses



Experimental and model development information for hydride precipitation failure mechanisms of Zircaloy cladding during decreasing temperature period over repository time scales is not available. Experimental results by R.E. Einziger and R. Kohli [Low Temperature Rupture Behavior of Zircaloy Clad Pressurized Water Reactor Spent Fuel Under Dry Storage Conditions, Nuclear Tech. Vol. 67, pp. 107-123, October, 1939] demonstrated the potential for reorientation of hydride precipitates which is a potential failure mechanism for Zircaloy cladding.

Experimental and model development information for fluoride corrosion failure mechanisms of Zircaloy cladding during the potential aqueous contact period over repository time scales is not available. A scoping investigation for the fluoride corrosion influences was reported by N.H. Uziemblo and H. D. Smith [An Investigation of the Influence of Fluoride on Corrosion of Zircaloy-4: Initial Report, PNL-6859, August, 1989].

For any cladding that has a defect hole and that is exposed to an oxygen gas environment, there is a potential for extensive cladding failure if UO_2 oxidizes to the higher oxide phases of U_3O_8 and/or UO_3. This failure of cladding (either Zircaloy or stainless steel) would occur due to the large volume increase (~30%) of UO_2 spent fuel phase transition to U_3O_8 or UO_3 phase. The rate of oxidation to the higher oxidation states is expected to be rapid (relative to repository time scales) at oxygen exposure temperatures above 250°C (R.E. Einziger, Effects of an Oxidizing Atmosphere in a Spent Fuel Packaging Facility, draft copy reviewed for FOCUS '91 conference, Battelle, PNL, Richland, WA, August, 1991). Thus, any defected rods would be considered as extensively failed (an open crack the length of the rod) if exposed to an oxygen atmosphere above 250°C. Between 250° and 200°C this phase transition may still be possible. However, below 200°C there is some preliminary evidence that UO_2 spent fuel oxidation may be stabilized at a lattice structure similar to U_4O_9, which has a slightly smaller volume than the initial UO_2 spent fuel lattice structure. Thus, rod failure due to the oxidation phase transition of UO_2 spent fuel to U_4O_9 at low temperatures (below 200°C) would not be expected to extensively fail any defected cladding having small pin holes.

In addition to the above mechanism of cladding degradation, Zircaloy will also oxidize very rapidly at temperatures above 350°C when exposed to oxygen in the air. While there is considerable scatter in the literature data, the following post-transition expression is provided (D.G. Farwick and R.A. Moen, Properties of Light Water Reactor Spent Fuel Cladding, HEDL-TME 78-70, August, 1979).

$$\Delta W(mg/cm^2) = 8.5 \times 10^6 \exp(-31000/RT)\, t$$

ΔW . . . weight gain, R . . . gas constant (1.98 cal/mole °K)
T . . . temperature °K, t . . . time hours

At 625°K, assuming that only pure ZrO_2 forms at an ideal density of 4.2 gm/cm³, the rate of oxide film growth on Zircaloy is approximately 3.0×10^{-4} cm/yr (1.1×10^{-4} inch/yr).

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

2.1.3.2. UO_2 Oxidation in Fuel

Spent fuel from power reactors contains mixtures, alloy subsets, and compounds of elements; but the aggregate atomic densities in spent fuel are dominated by uranium and oxygen atoms. With the exception of some UO_2 fuels with burnable poisons (primarily gadolinia in BWR rods), the other elements with significant atomic densities in spent fuel evolve during reactor operation from neutron reactions and fission plus fission decay events. Due to nuclear decay processes, the intrinsic chemical composition and activity of spent fuel will continue to evolve after it is removed from reactors as its radioactivity decays over time. During the time interval when the radioactivity levels are significant, which is the time interval relevant for design and for performance assessment of a geological repository, it is important to develop an understanding and to develop models that describe potential chemical responses in spent fuel and its potential degradational impacts on repository design and performance. One such potential impact is the oxidation response of spent fuel. The oxidation of spent fuel results in an initial phase change of the UO_2 lattice to a U_4O_9 lattice, and the next phase change is probably to U_3O_8 although it has not been observed yet at low temperatures (<200°C). The U_4O_9 lattice is non-stoichiometric with an oxygen to uranium weight ratio (O/U) at ~2.4. Preliminary indications are that the UO_2 has a O/U of ~2.4 at the time just before it transforms into the U_4O_9 phase.[1,2] Also, in the oxygen weight gain versus time response, a plateau appears as the O/U approaches ~2.4. Part of this plateau response is due to geometrical effects of a U_4O_9 phase change front propagating into UO_2 grain volumes. The experimental data clearly show a front of U_4O_9 lattice structure propagating into grains of the UO_2 lattice structure. However, the plateau time response may be indicative of a metastable phase change delay kinetics or a diffusional related delay time until the oxygen density can attain a critical value to satisfy the stoichiometry and energy conditions for phase changes. The next phase observed after the non-stoichiometric U_4O_9 phase was U_3O_8 at temperatures above 250°C.[3] The phase change kinetics and the oxidation rate response for the U_3O_8 phase at temperature below 250°C are not presently well understood. The importance of the temperature-time oxidation rate and phase kinetics of spent fuel impacts the potential aqueous release rate because of

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1 L. E. Thomas and R. E. Einziger, *Mater. Charact.* 28, 149 (1992).

2 R. E. Einziger, L. E. Thomas, H. C. Buchanan, and R. B. Stout, *J. Nucl. Mater.*, 190, 53 (1992).

3 R. E. Einziger and R. V. Strain, *Nucl. Technol.*, 75, 82 (1986).

possible different intrinsic dissolution rates for the various oxidation phases and because the phase transformation can significantly increase the initial surface area of fragments by cracking open grain boundaries (the transformation from UO_2 to U_4O_9 is a slight density increase) and by microcracking and flaking grain volumes (the transformations from U_4O_9 to U_3O_8 is significant (~33%) density decrease).

The existing data and preliminary modeling of low temperature oxidation rate and phase change response will be described in Section 3.2.

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2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

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2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

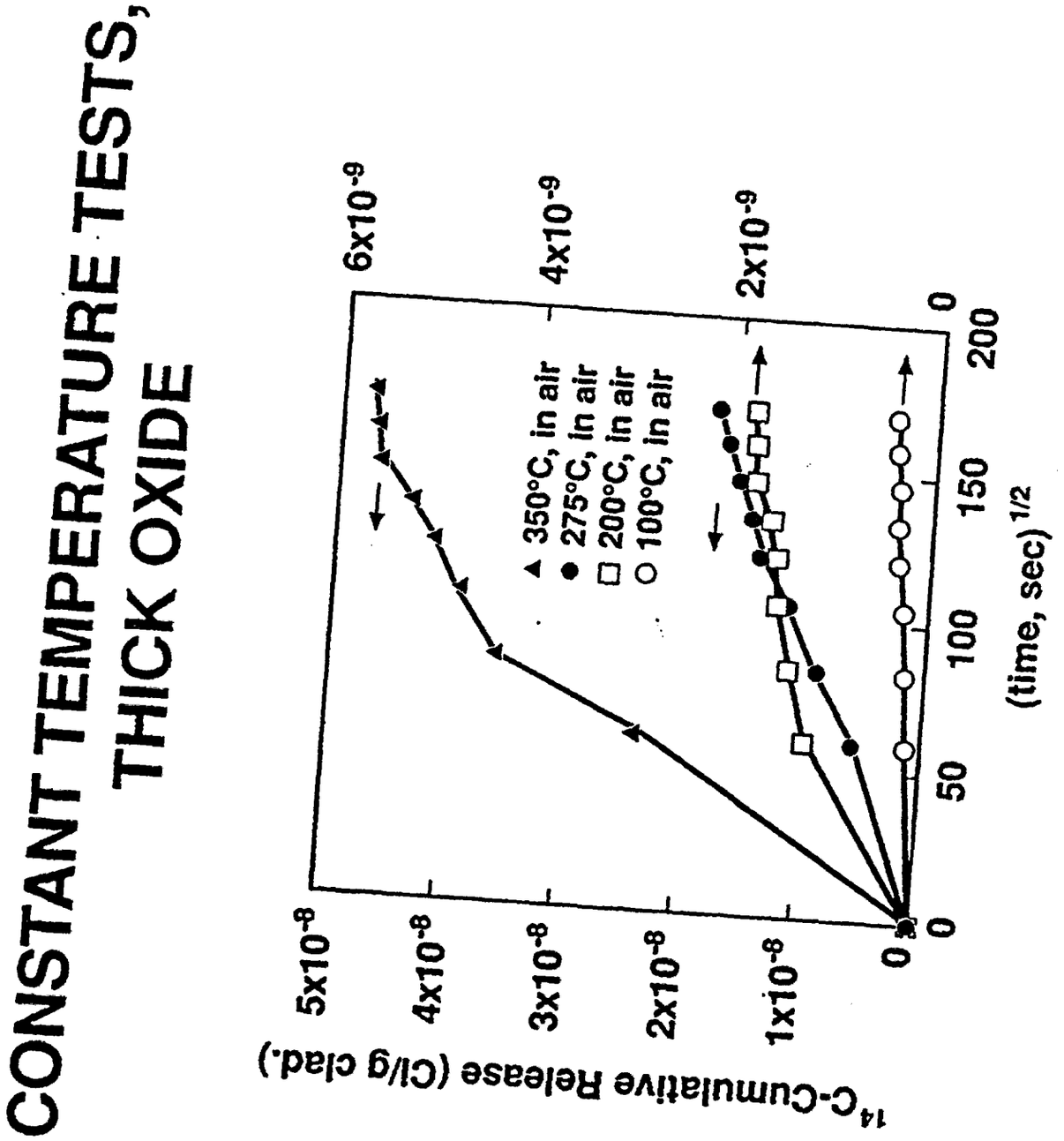
2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

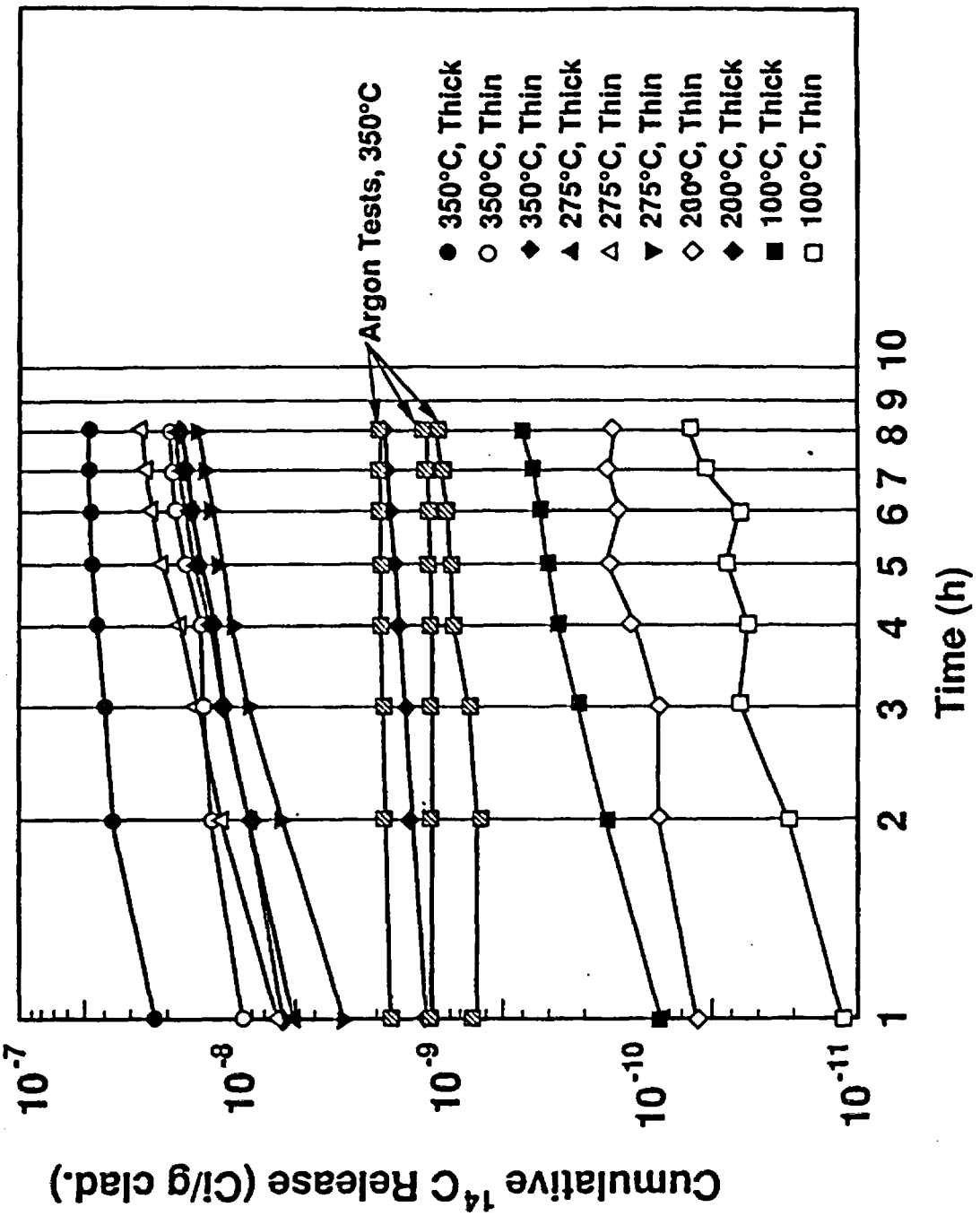
2.1.3.7 Radionuclide Release from Hardware



PWR fuel with oxide thickness ~12 to 20 μm and burnup ~27 to 30 MWd/kg HM.

H.D. Smith, *Spent Fuel Cladding Degradation*, presented to the Nuclear Waste Technical Review Board, August, 1990.

OBSERVED ^{14}C RELEASE FROM ZIRCALOY-4 SPENT FUEL CLADDING



PWR fuel with oxide thickness ~12 to 20 μm and burnup ~27 to 30 MWd/kg HM.

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2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Gaseous Radionuclide Release from UO_2 Fuel

We have an equation that describes the fission gas release curves presented at the "Status and Future Directions of Spent Fuel ATM Acquisition and Characterization" by (9) C. E. Beyer of the MCC at PNL in March 28-29, 1989. The equation is good for burn-up $\geq$ 20 MWd/kgM and for fission gas releases $\leq$ 60%.

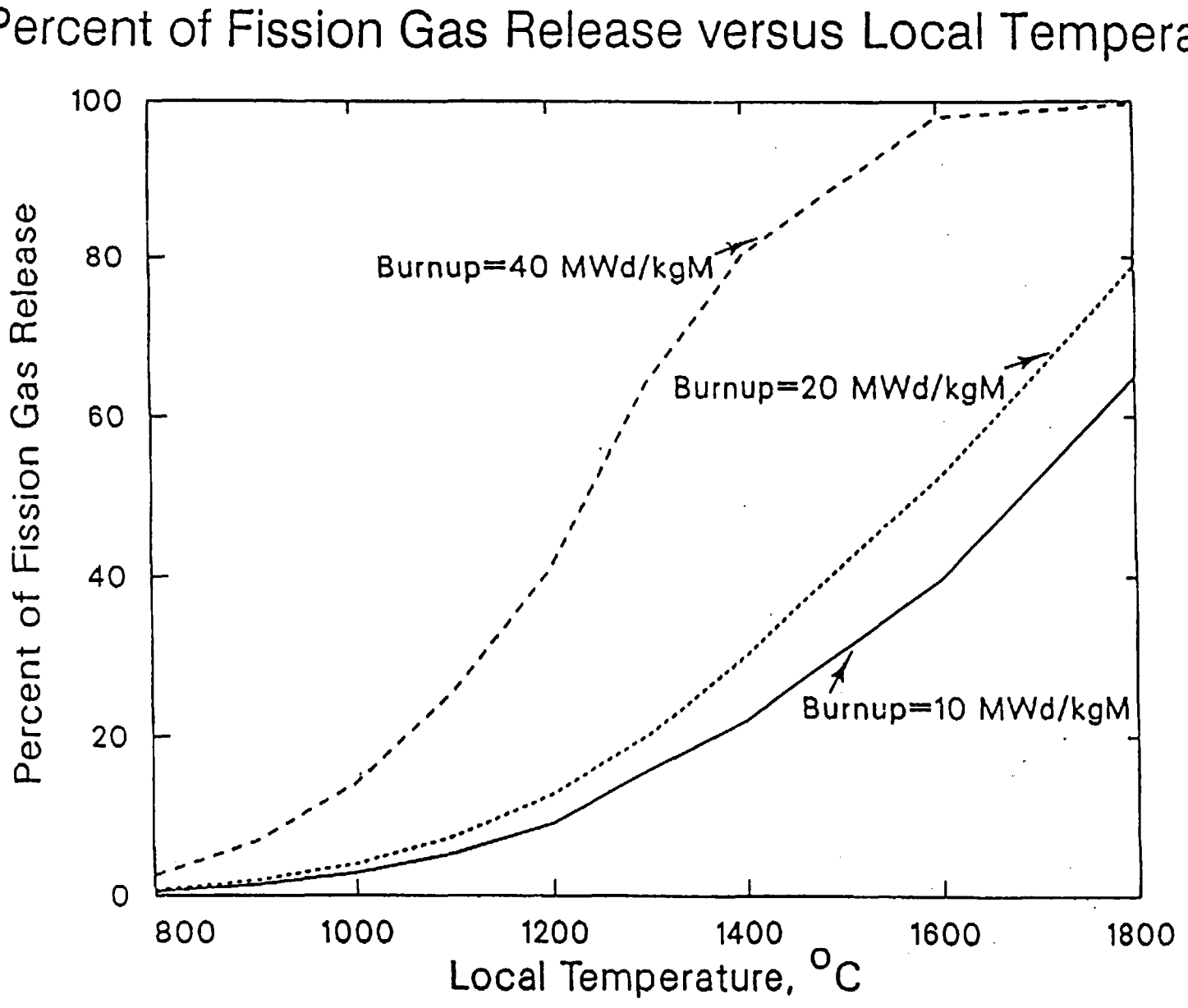
This plot is shown in one of the attached figures. The added curves in the second figure are those calculated for burn-ups of 30, 50 and 60 MWd/kgM, using the equation

$$\log_{10}[\text{fractional release}] = \frac{13}{8}\log_{10}[\text{Burn-up (MWd/kgM)}] - \frac{4420}{T(K)}$$

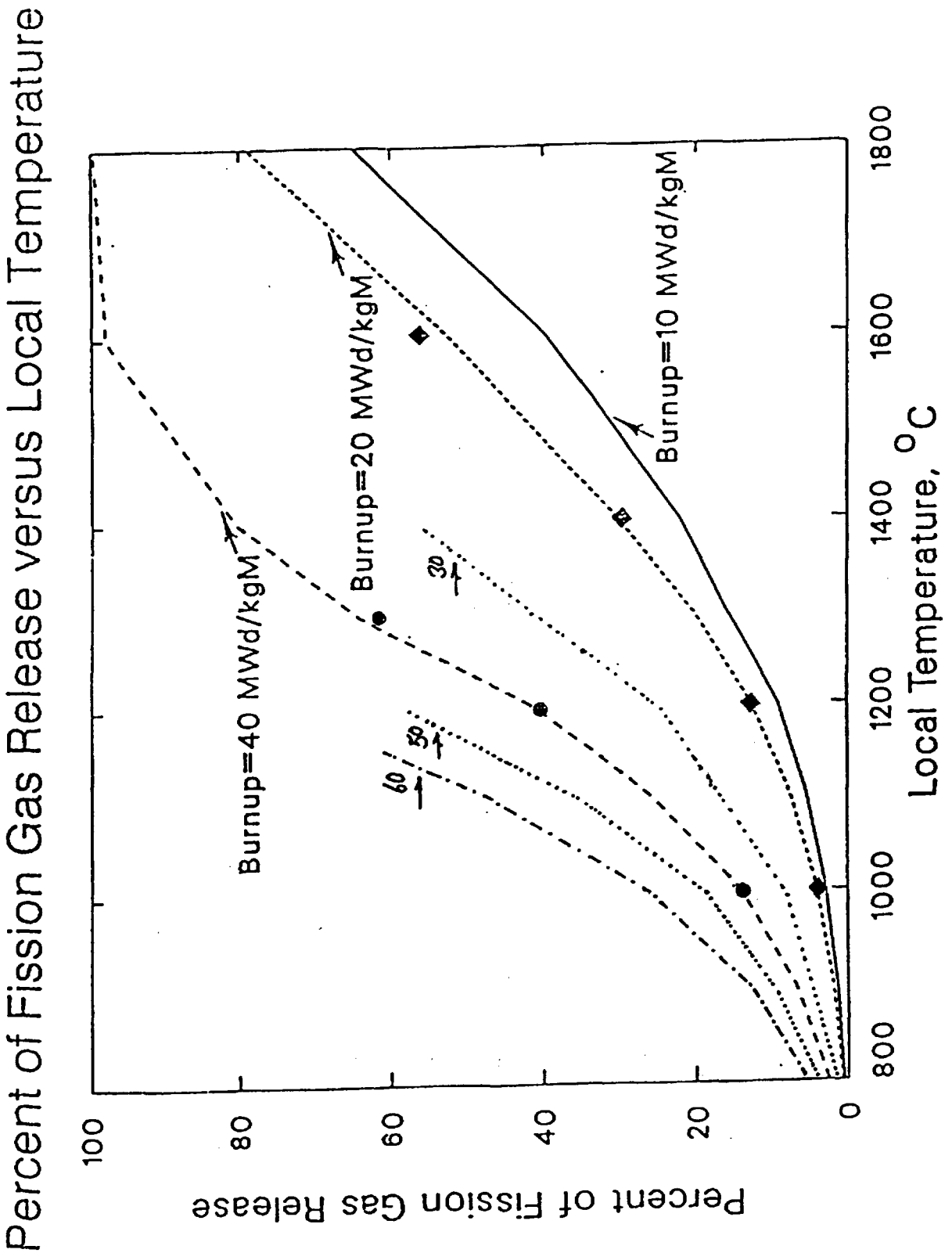
This expression overestimates the release above 60% according to the MCC curves. The points superimposed on the 20 and 40 MWd/kgM curves are calculated according to the equation and show that the fit to the MCC curves is quite good.

Herman R. Leider
Physical Chemistry Section
Chemistry & Materials Science Dept.

Attachments

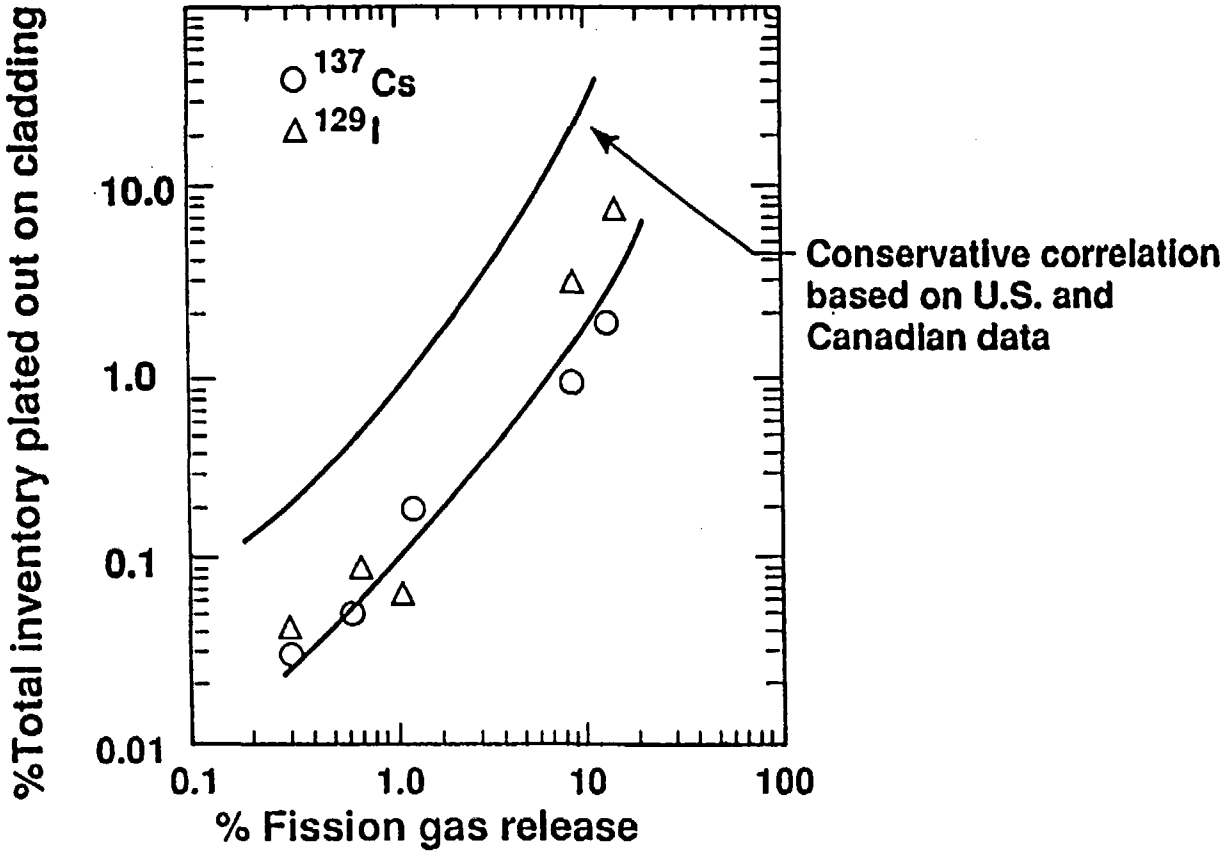


C.E. Beyer, in *Status and Future Direction of Spent Fuel ATM Acquisition and Characterization*, meeting in Richland, Washington, March, 1989.



C.E. Beyer, in *Status and Future Direction of Spent Fuel ATM Acquisition and Characterization*, meeting in Richland, Washington, March, 1989.

METHOD OF CORRELATING GAP AND GRAIN BOUNDARY INVENTORY WITH ROD-AVERAGE FISSION GAS RELEASE



R.B. Stout, *Spent Fuel Characteristics Overview*, presented to the Nuclear Waste Technical Review Board, August, 1990.

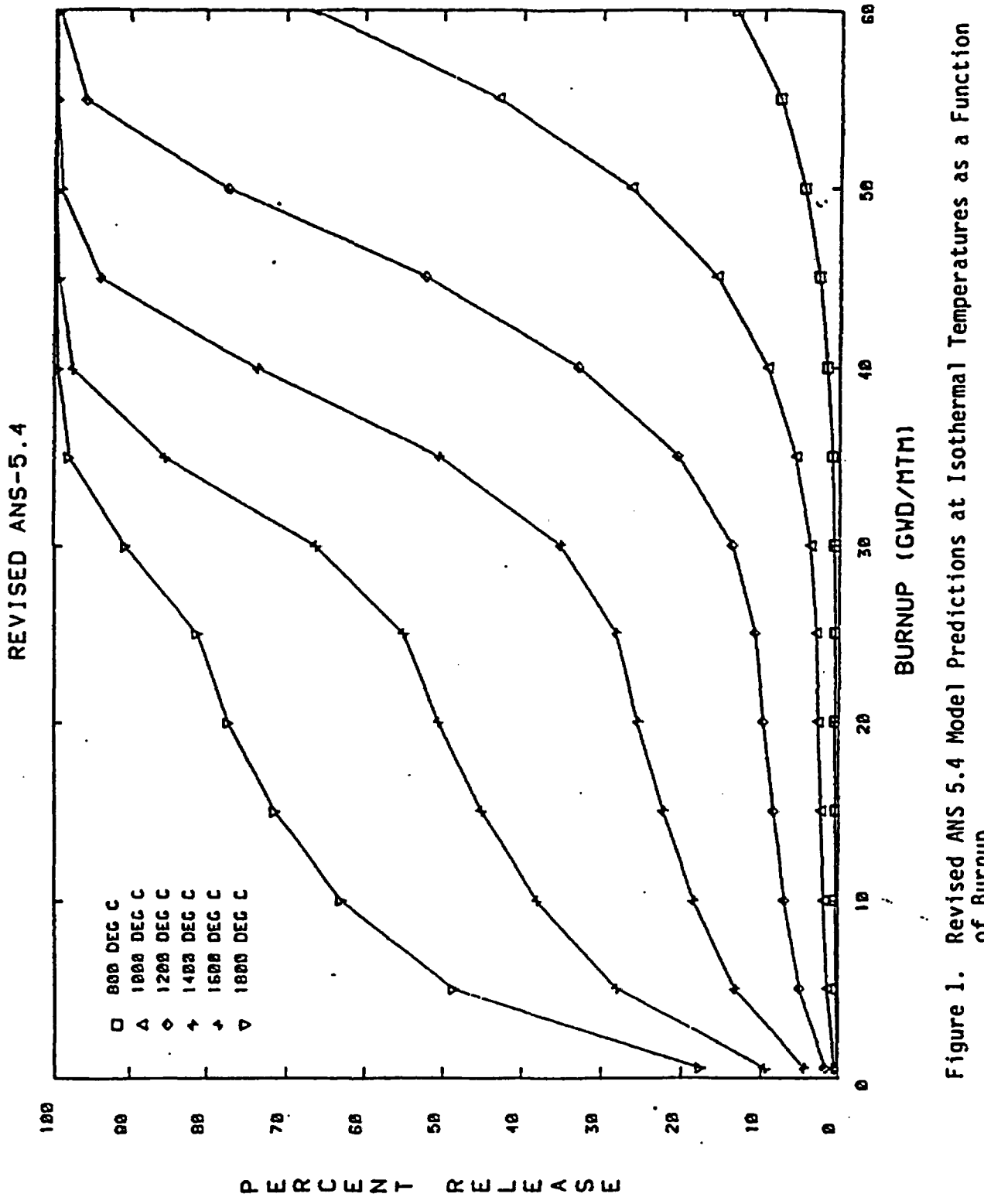


Figure 1. Revised ANS 5.4 Model Predictions at Isothermal Temperatures as a Function of Burnup.

C.E. Beyer, in *Status and Future Direction of Spent Fuel ATM Acquisition and Characterization,* meeting in Richland, Washington, March, 1989.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

STUDIES ON SPENT FUEL DISSOLUTION BEHAVIOR* UNDER YUCCA MOUNTAIN REPOSITORY CONDITIONS

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ABSTRACT

Nuclide concentrations measured in laboratory tests with PWR spent fuel specimens in Nevada Test Site J-13 well water are compared to equilibrium concentrations calculated using the EQ3/6 geochemical modeling code. Actinide concentrations in the laboratory tests reach steady-state values lower than those required to meet Nuclear Regulatory Commission (NRC) release limits. Differences between measured and calculated actinide concentrations are discussed in terms of the effects of temperature (25°C to 90°C), sample filtration, oxygen fugacity, secondary phase precipitation, and the thermodynamic data in use. The concentrations of fission product radionuclides in the laboratory tests tend to increase continuously with time, in contrast to the behavior of the actinides.

1.0 INTRODUCTION

The Yucca Mountain Project of the U. S. Department of Energy is studying the potential dissolution and radionuclide release behavior of spent fuel in a candidate repository site at Yucca Mountain, Nevada. The repository horizon under study lies in the unsaturated zone 200 to 400 meters above the water table. With the exception of C-14, which may migrate in a vapor phase,[1] and possibly I-129, the majority of long-lived radionuclides present in spent nuclear fuel will be transported from a failed waste package in the repository via dissolution or suspension in water in the absence of a major geological event such as volcanism.

*This material also is important in understanding Section 3.4.

[1] Published in Ceramic Transactions, V-9, pp. 423-442. Nuclear Waste Mgt. III, G. B. Mellinger, ed. Westerville, Ohio, 1990.

Spent fuel will not be contacted by liquid water infiltrating the rock until several hundred years after disposal when the repository has cooled to below the 95°C boiling temperature of water at the repository elevation. The potential dissolution behavior of spent fuel during the repository post-thermal period is being studied using geochemical models and laboratory tests with actual spent fuel specimens.* Selected initial results from these studies are discussed in the present paper.

2.0 LABORATORY TESTS

Three spent fuel dissolution test series have been conducted in laboratory hot cells using spent fuel specimens of various configurations. Results from the Series 2 and Series 3 tests with bare fuel particules are discussed in the present paper. The Series 2 tests used unsealed fused silica test vellels and were run for five cycles in air at ambient hot cell temperature (25°C). The Series 3 tests used sealed stainless steel vessels and were run for three cycles at 25°C and 85°C. Each test cycle was started in fresh Nevada Test Site J-13 well water and was about six months in duration. Periodic solution samples were taken during each test cycle and the sample volume was replenished with fresh J-13 water. Five bare fuel specimens tested in these two tests series are identified in Table 1 and the test configurations are shown in Figure 1. Additional information on the laboratory tests is provided in references 3 and 4.

2.1 Actinide Results

Actinide concentrations (U, Np, Pu, Am and Cm) measured in solution samples rapidly reached maximum levels during the first test cycle and then generally dropped to lower steady-state levels in later test cycles. The concentrations of uranium and the activities of Pu-239+240 and Am-241 measured in 0.4 μm filtered solution samples are plotted in Figure 2. The initial concentration peaks are attributed to dissolution of more readily soluble UO_{2+x} oxidized phases present initially of the fuel particle surfaces, and to kinetic factors limiting the nucleation and growth of secondary phases that may ultimately control actinide concentrations at lower levels.

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Table 1. Bare Fuel Test Identification

Identification	Description	Starting Fuel Wt. (g)
HBR-2-25	Series 2, H.B. Robinson Fuel, 25°C	83.10
TP-2-25	Series 2, Turkey Point Fuel, 25°C	27.21
HBR-3-25	Series 3, H.B. Robinson Fuel, 25°C	80.70
HBR-3-85	Series 3, H.B. Robinson Fuel, 85°C	85.55
TP-3-85	Series 3, Turkey Point Fuel, 85°C	86.17

Uranium (U) concentrations at 25°C were lower in the Series 3 tests than in the Series 2 tests, and with the exception of the Cycle 1 data, U concentrations in the 85°C Series 3 tests were lower than those in the 25°C tests. The very low U concentrations measured during Cycle 1 of the HBR-3-85 test were attributed to a vessel corrosion anomaly. In the later cycles of the Series 2 tests, U concentrations tended to stabilize at steady-state levels of about 1 to 2 µg/ml. In Cycles 2 and 3 of the Series 3 tests, U concentrations stabilized at about 0.3 µg/ml at 25 °C and about 0.15 µg/ml at 85°C. Precipitated crystals of the calcium-uranium-silicates, uranophane (Figure 3) and haiweeite, and possibly the uranium-silicate soddyite, were found on filters used to filter cycle termination rinse solutions from both 85°C tests. Phase identifications were based on examinations by X-ray diffraction and microanalysis in the SEM.[4] Secondary phases controlling actinide concentrations other than U were not found.

The 0.4 µm filtered Pu-239+240 solution activities measured in Cycles 2 through 5 of the TP-2-25 test generally ranged from about 100 to 200 pCi/ml (Figure 2). Activities as low as about 20 pCi/ml were measured in the HBR-2-25 test. During Cycles 2 and 3 of the HBR-3-25 test, activities varied from about 60 to 100 pCi/ml. A value of 100 pCi/ml, which corresponds to a Pu concentration of about 4.4 x 10^{-9} M (M = molarity), would appear to be a reasonable estimate of steady-state Pu-239+240 activities in 0.4 µm filtered solutions in the 25°C. Significantly lower activities on the order of 1 pCi/ml were measured in the 85°C tests. The lower activities at 85°C may result from enhanced nucleation and growth of secondary phases at the higher temperature that limit pU concentration.

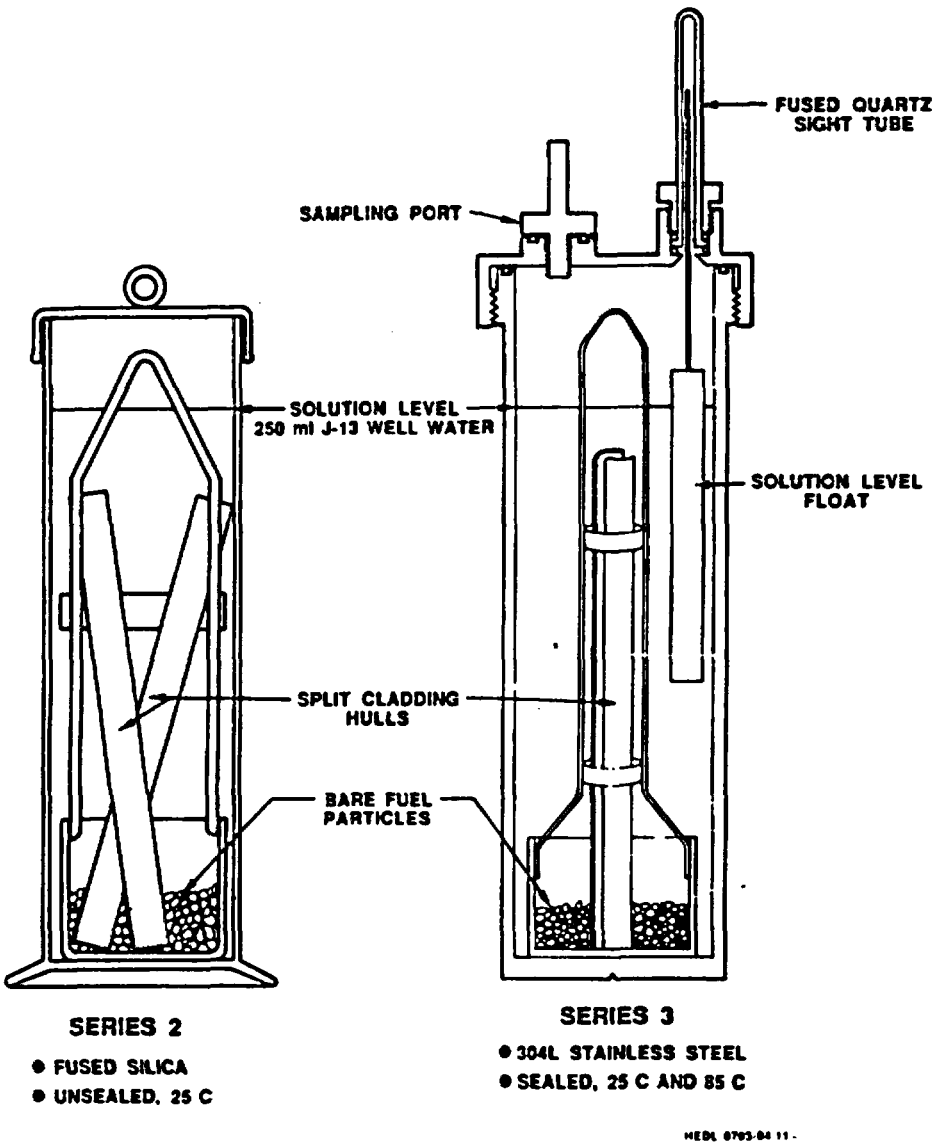

Figure 1. Test Configurations for the Series 2 and Series 3 Bare Fuel Dissolution Tests.

Selected solution samples were centrifuged through membrane filters that provide an estimated filtration size of approximately 2 nm.* Filtering to 2 nm caused Pu-239+240 activities to decrease by about 20 to 40%. No significant differences between 0.4 μm filtered sample data are considered the most significant relative to radionuclide release because larger particles probably would not be transported by water, whereas colloidal particles greater than 2 nm may remain in stable suspension and be transported by water movement.

Table 2. J-13 Well Water Analysis[2]

Component	Concentration (μg/ml)	Component	Concentration (μg/ml)
Li	0.042	Si	27.0
Na	43.9	F	2.2
K	5.11	Cl	6.9
Ca	12.5	NO_3	9.6
Mg	1.92	SO_4	18.7
Sr	0.035	HCO_3	125.3
Al	0.012		
Fe	0.006	pH	7.6

Steady-state Am-241 activities on the order of 100 pCi/ml, corresponding to Am concentrations of about 1.5×10^{-10} M, were measured in 0.4 μm filtered samples during cycles 2 and 3 of the TP-2-25 and HBR-3-25 tests. The 100 pCi/ml value would appear to be a conservative estimate for Am-241 activity at steady-state and 25°C considering that activities on the order of 10 pCi/ml were measured during Cycles 2, 4 and 5 of the HBR-2-25 test. Much lower 0.4 μm filtered Am-241 activities of about 0.3 pCi/ml were measured during Cycles 2 and 3 of the two 85°C tests. The effects of both 0.4 μm and 2 nm filtration were in general greater for Am-241 than for Pu-239+240. Association of Am with an apparent suspended phase is suggested by unfiltered data from the 85°C tests plotted as dashed lines in Figure 2, and by a relatively large fraction of 0.4 μm filtered Am-241 activity removed by 2 nm filtration (not shown). Cm-244 activity measured in most samples was similar to that measured for Am-241 in each of the tests. However, Cm-244 alpha decays with an 18-year half-life to Pu-240 and will not be present during the repository post-thermal period.

Measured Np-237 activities in most samples were generally not much greater than the detection limit of 0.1 pCi/ml and were below detection limits in several samples. Measured Np-237 activities showed very little dependence on temperature, vessel type or sample filtration. Following initially higher values at the beginning of Cycle1, Np-237 activities generally ranged from 0.1 to 0.5 pCi/ml.

* Amicon Corporation Model CF-25 centrifuge membrane cone filter

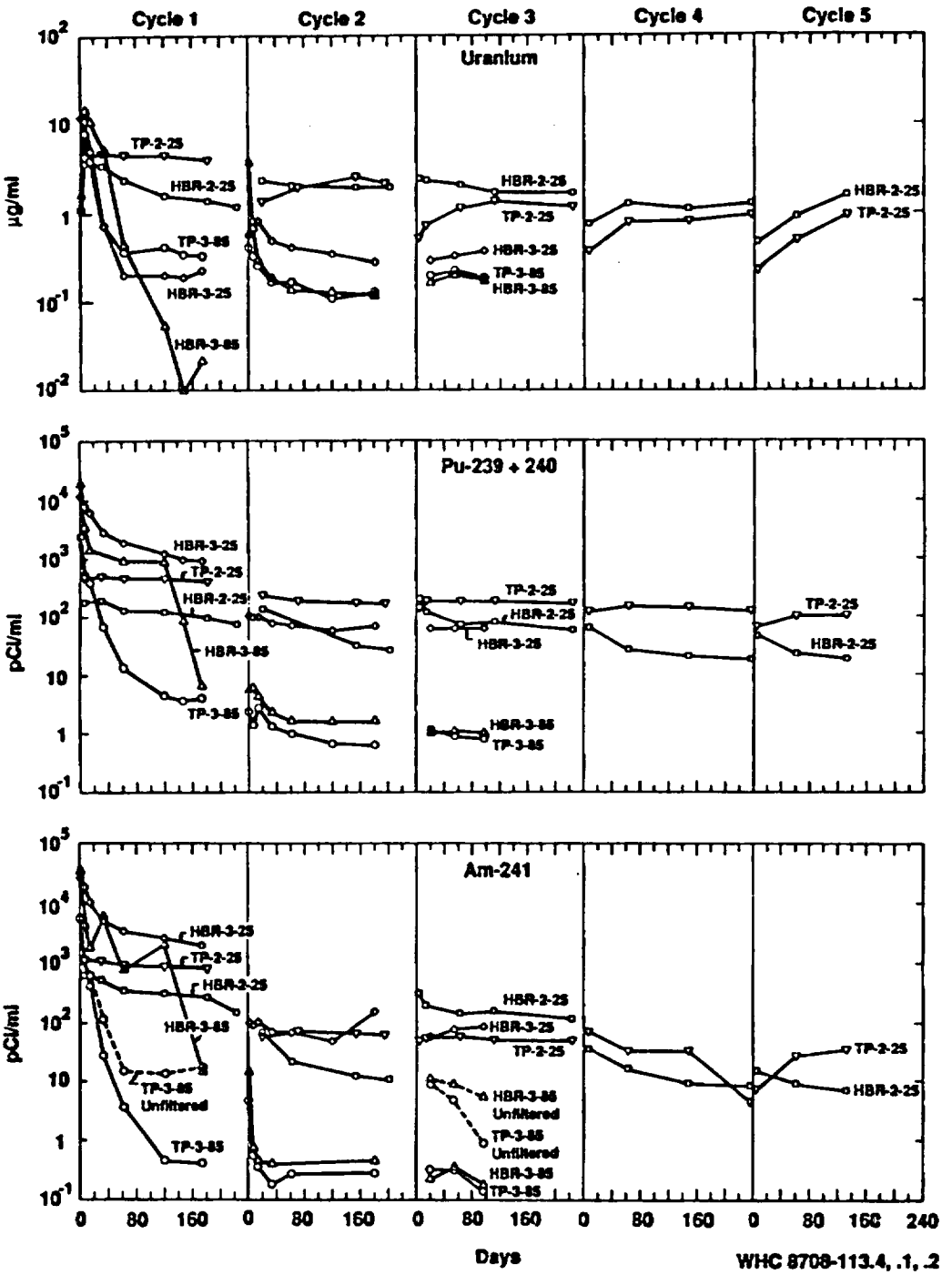

Figure 2. Uranium Concentrations (top), Pu-239+240 Activities (center) and Am-241 Activities (bottom) Measured in 0.4 µm Filtered Solution Samples.

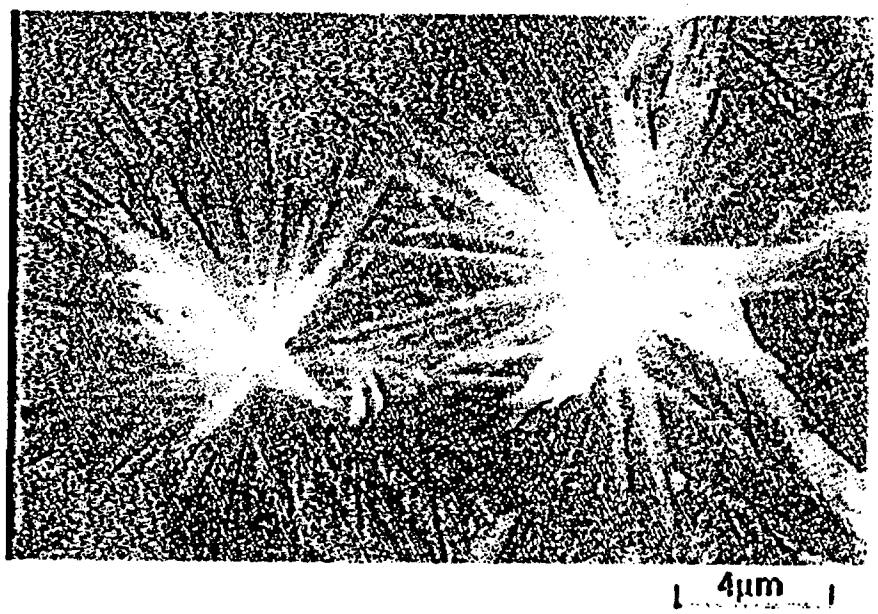

Figure 3. Acicular crystals of Uranophane formed on spent fuel grains in the 85°C Series 3 tests

2.2 Fission Product Results

Specimen inventory fractions of the fission product radionuclides Cs-137, Sr-90, Tc-99, and I-129 measured in solution are plotted in Figure 4 for the HBR-2-25 and HBR-3-85 tests. Each data point represents the fraction of the ORIGEN-2 calculated speciment inventory in solution on the sample data plus the inventory fraction calculated to have been removed in previous samples from the test cycle. During Cycle 1 of the HBR-3-85 test, Tc-99 fell to below detectable levels as a result of the corrosion anomaly that occurred in this test. Cycle 1 Cs-137 gap inventory release was about 0.7% from the HBR fuel and is therefore off-scale in Figure 4. Sr-90 was not measured during Cycle 1 of the Series 2 tests, and appeared to be limited by association with an unknown precipitated phase in the 85°C tests.

The inventory fractions of Cs-137, Sr-90, Tc-99 and I-129 in solution increased continuously with time, with the exception of the anomalous precipitation of Tc-99 in Cycle 1 of the HBR-3-85 test and the limit on Sr-90 activity in solution at 85°C. The continuous release rates of the fission products in units of inventory fraction per year are given in Figure 4 for the final cycle of the two tests. Because the actual quantity of fuel matrix dissolution and precipitation of actinides was not measured, it is not known to what degree the continuous fission product release resulted form preferential leaching of grain boundaries where fission products were thought to concentrate during irradiation. Whether as a result of increased matrix dissolution or increased grain boundary leaching, the soluble fission product release rate is greater in the later test cycles at the higher temperature.

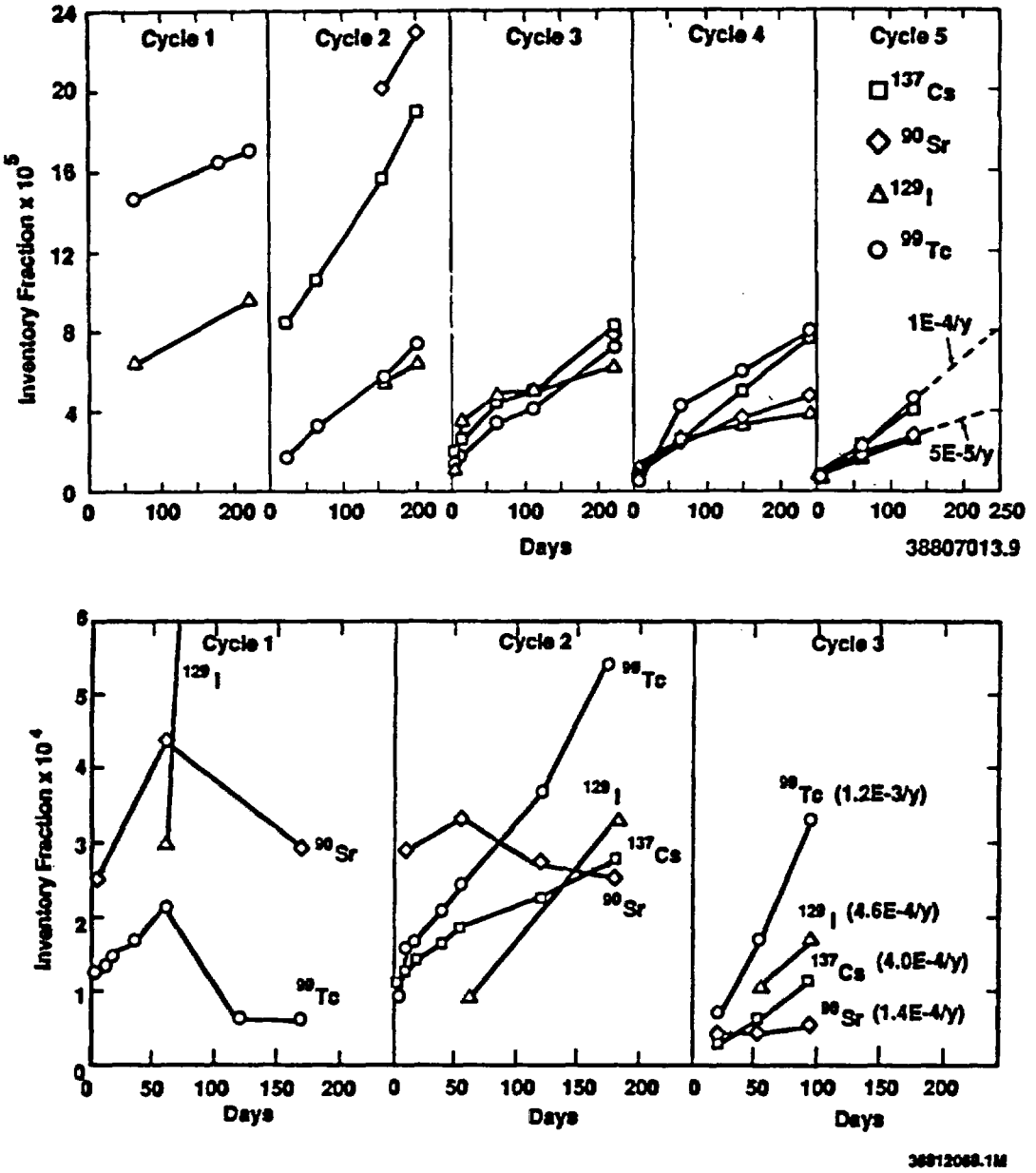

Figure 4. Inventory Fractions of Cs-137, Sr-90, Tc-99 and I-129 Measured in Solution in the HBR-2-25 Test (top) and in the HBR-3-85 Test (bottom). Approximate annual fractional release rates are listed for each nuclide during the last cycle plotted.

3.0 GEOCHEMICAL MODELING

3.1 Actinide Concentrations in Solution

Spent fuel dissolution in J-13 well water was simulated using the geochemical modeling code EQ3/6[5] to determine whether steady-state actinide concentrations measured in the tests could be related to the precipitation of actinide-bearing solids. Version 3245 of the EQ3/6 code and version 3270R13 of the supporting thermodynamic database were used to simulate spent fuel dissolution at 25°C and 90°C assuming atmospheric CO_2 gas fugacity and two different O_2 gas fugacities of $10^{-0.7}$ (atmospheric) and 10^{-12} bars (see later discussion). The simulation process is described in more detail elsewhere.[6] The computer silulations yield: 1) the sequence of solids that precipitate and sequester elements released during spent fuel dissolution, and 2) the corresponding elemental concentrations in solution. Approximate steady-state actinide concentrations measured at 25°C and 85°C in the Series 3 laboratory tests were compared in Table 3 to concentrations of actinides in equilibrium with the listed solids as calculated in the EQ3/6 simulations. Comparisons of simulation results with experimental results are being used to determine the adequacy of the thermodynamic database and to identify additional aqueous species and minerals for which data are needed.

Table 3. Comparison of Measured and Predicted Actinide Concentrations (log M)
(New runs have not been completed) May 22, 1993 RBS

	Measured[(a)]		EQ3/6[(b)]				
			25°C		90°C		
Actinide	25°C	85°C	-0.7	-12.0	-0.7	-12.0	Phase
U	-5.9	-6.2	-7.2/-7.0*	-7.1/-6.9	-8.8/-7.6	-8.5/-7.5	H
			-7.0/-6.9	-6.9/-6.8	-7.6	-7.5	H + S
			-6.9/-4.3	-6.8/-4.2	-7.6/-6.0	-7.5/-5.9	S
			-4.3	-4.2	-6.0	-5.9	S + Sch
			-4.2	-4.1	-6.0/-5.8	-5.8/-5.6	Sch
Np	-8.9	-9.1	-6.2	-9.0	-5.2	-8.0	NpO_2
Pu	-8.4	-10.4	-12.4	-13.8	-11.9	-14.6	PuO_2
			-4.3	-5.7	-4.2	-6.9	$Pu(OH)_4$
Am	-9.8	-12.3	-8.3	-8.3	--	--	$Am(OH)CO_3$
			--	--	-8.4	-8.4	$Am(OH)_3$
Cm	-11.3	-14.3	Cm not in thermodynamic database				

(a) Series 3 tests, 0.4 μm filtered.
(b) At oxygen fugacities log f_{O2} = -0.7 (atmospheric) and log f_{O2} = -12.0 with solubility control by precipitated secondary phases as listed. H = haiweeite; S = soddyite; Sch = schoepite. All phases are in crystalline state except $Pu(OH)_4$ which is amorphous.
* -7.2/-7.0 refers to a range in concentration from -7.2 to -7.0.

Uranium (U) concentrations in the simulations vary as a function of the secondary U-bearing precipitates. The following sequence of mineral assemblages are predicted to precipitate and sequester U as increasing amounts of spent fuel dissolve: haiweeite, haiweeite plus soddyite, soddyite, soddyite plus schoepite, and schoepite. The relative compositions of these phases and of U-bearing phases that were observed in residues from the 85°C laboratory tests are shown in Figure 5. Unique, and steadily increasing, concentrations of U in solution are related to each mineral assemblage. The concentration of U varies not only as the precipitates vary, but also during the precipitation of a single mineral, such as soddyite, because of changes in the pH and overall chemical characteristics of the fluid. As previously discussed, uranophane, haiweeite, and possibly soddyite were found in the 85°C Series 3 tests. Unfortunately, reliable thermodynamic data for uranophane were not available, which complicates comparison of the laboratory test results to the calculated solubility limits. Haiweeite, a Ca-U-silicate like uranophane, is predicated to precipitate at U concentrations that are lower than the measured steady-state values. In the absence of data for uranophane, the experimental concentrations of U would appear to be consistent with the precipitation of soddyite at both 25°C and 90°C in the simulations.

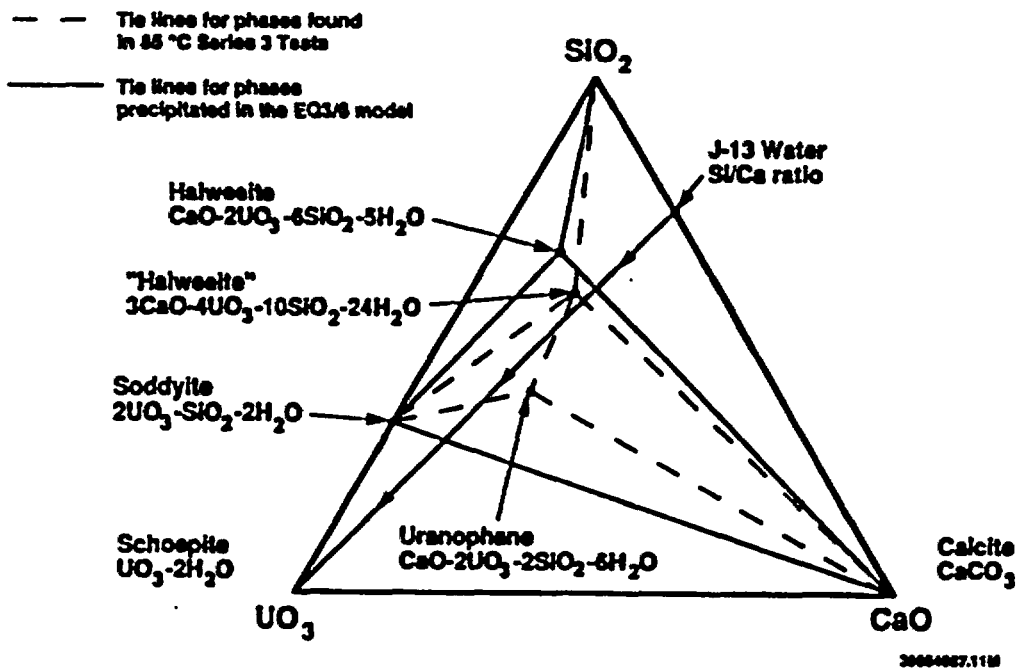

Figure 5. Relative Compositions (mole %) of U-bearing Phases Indicated as Controlling U Concentration in the EQ3/6 Simulation and for which Indications were Observed in the 85°C Series 3 Tests.

Neptunium concentration is controlled by equilibrium with NpO_2 in the simulations. However, the predicted concentration of Np is highly dependent on solution Eh and pH.[7] The O_2 fugacity in the simulations was reduced from $10^{-0.7}$ bars to 10^{-12} bars in order to produce good agreement between the measured and predicted concentrations of Np at 25°C. An O_2 fugacity of 10^{-12} bars may correspond to conditions at the fuel surface in an otherwise oxygenated system (i.e., contains an air cap) that is poorly buffered. Eh was not measured during the laboratory tests, and redox equilibrium may not have been established among the various species and phases within the sealed stainless steel vessels. An oxygen fugacity of 10^{-12} bars over-estimates Np concentration at 90°C, however, because the experimental data do not reflect predicted increases in Np concentration with temperature. The thermodyanamic data for Np and other actinides must, consequently, be critically evaluated at elevated temperature.

Significant differences exist between measured and predicted Pu and Am concentrations in Table 3. Measured Am concentrations may have been lower than those predicted because of Am removal from solution by phases such as lanthanide precipitates that were not accounted for in the EQ3/6 simulations. Another possible mechanism controlling Am concentration not accounted for in the simulation may have been sorption. Although $Am(OH)CO_3$ is predicted to control Am concentration at 25°C and $Am(OH)_3$ precipitates at 90°C, the Am concentration in equilibrium with both phases is about the same.

Predicted Pu concentrations in equilibrium with crystalline PuO_2 at both temperatures and oxygen fugacities are much lower than those measured. Pu concentrations measured at 25°C are similar to those reported by Rai and Ryan,[8] who measured the solubility of PuO_2 and hydrous $PuO_2 \cdot xH_2O$ in water for periods of up to 1300 days at 25°C. At a pH of 8, which was the extrapolated lower limit of their data and the approximate pH in the Series 2 and 3 tests, they reported that Pu concentrations ranged from about $10^{-7.4}$ M, where amorphous $PuO_2 \cdot xH_2O$ was thought to control concentration, down to about 10^{-9} M where aging of the amorphous material produced a more (but incompletely) crystalline PuO_2 that was thought to control concentration. Concentrations of Pu in equilibrium with amorphous $Pu(OH)_4$, calculated in recognition of the fact that an amorphous or less crystalline phase is more likely to precipitate than crystalline PuO_2, are listed in Table 3. Measured Pu concentrations would be expected to fall between the equilibrium concentrations for PuO_2 and $Pu(OH)_4$, becoming closer to PuO_2 with aging. Equilibrium with amorphous $Pu(OH)_4$ and crystalline PuO_2 at O_2 fugacities of $10^{-0.7}$ and 10^{-12} bars yields predicted Pu concentrations that bracket measured results at both 25°C and 85°C.

3.2 Sources of Discrepancy Between Measured and Predicted Results

Discrepancies between measured and predicted concentrations are to be expected considering database limitations and uncertainty in the interpretation of measured apparent steady-state actinide concentrations. Care must be taken in interpreting the 90°C simulation results because insufficient data exist to accurately calculate the temperature-dependence of the thermodynamic properties of many radionuclide-bearing solids and solution species. The 3270 thermodynamic database is constantly updated through inclusion of new and revised thermodynamic data and the selection of a consistent set of aqueous complexes for each chemical element. Puigdomenech and Bruno[9] have constructed a thermodynamic database for U minerals and aqueous species that they showed to be in reasonable agreement with available experimental solubility data in systems in which U is complexed by OH^- and $CO_3^{=}$. The 3270 database contains many of the same aqueous species and minerals, but Puigdomenech and Bruno have included recent data for aqueous uranyl hydroxides from Lemire[10] which are not yet in the EQ3/6 database. Future plans include a critical evaluation of simulations of spent fuel dissolution made using the Puigdomenech and Bruno U database, and comparison with simulations made using the latest version of the EQ3/6 database. Inclusion of standard Nuclear Energy Agency (NEA) data for U minerals and species will also help to standardize future databases.

Until the U database is better established, calculated U concentrations must be recognized as preliminary and speculative. Simulation results can be used as a vehicle for identifying geochemical trends and studying the interactions between solid precipitation and elemental concentrations in solution. Seemingly small changes in the thermodynamic database can have potentially large impacts on predictions. For example, U concentrations calculated to be in equilibrium with schoepite using version 3270 of the EQ3/6 database are radically lower than those predicted in 1987[6] using an older database. The species $(UO_2)_3(OH)_7^-$ and $(UO_2)_2(OH)_3CO_3^-$ were omitted from version 3270 of the EQ3/6 database because their validity was questioned. $UO_2(CO_3)_2^{-2}$ and $UO_2(CO_3)_3^{-4}$ were left as the only dominant U species in solution throughout the EQ3/6 simulations. U concentrations accordingly remain lower during U mineral precipitation. Future work must address the sensitivity of the results to variations in thermodynamic data and the choice of a self-consistent set of aqueous species for elements of interest.

Comparisons between experimental results and predictions in Table 3 are predicated on the assumption that the listed solid phases precipitate from solution and control the solution composition. Except for some U-bearing minerals, no minerals containing radionuclides have been identified in the laboratory tests. Detection and characterization of actinide-bearing secondary phases may be difficult because of the extremely small masses of these actinides involved. Precipitates limiting actinide

concentrations in the laboratory tests may also be amorphous, colloidal, or in some other less-than-perfect crystalline state. For instance, Rai and Ryan[8] observed that early Pu precipitates tend to be hydrated oxides which undergo aging to more crystalline solids. The concentrations of the affected actinides would, therefore, gradually decrease as aging progresses.

The chemistry of trivalent Am and Cm can be expected to be almost identical to that of the light lanthanide fission product elements which are present in much greater concentrations in spent fuel than are Am and Cm. Am and Cm may, therefore, be present in dilute solid solution with secondary phases formed by the lanthanides, which would result in lower measured solution concentrations than predicted for Am based on equilibration with $Am(OH)CO_3$ or $Am(OH)_3$. Pu and Np, and possibly Am and Cm, may also have been incorporated at low concentrations in solid solution with the U-bearing precipitates or other secondary phases. Efforts are planned to separate crystals of uranophane from test residues and to perform radiochemical analyses of these crystals to check for incorporation of other radionuclides. Sorption of actinides on colloids or other surfaces such as the fuel or test hardware may also control solution concentrations, but the impact or sorption was not considered in the simulations. Other factors, such as local variations in redox potential, may also contribute to differences between measured and predicted solubilities.

As it is not currently reasonable to expect a geochemical model to predict accurately the effects of all potential concentration-controlling processes over thousands of years, we hope to use modeling predictions to establish upper limits, or conservative estimates, of radionuclide concentrations over time. Lower limits to radionuclide concentrations imposed by solid precipitation are also of interest, however, as a baseline for further calculations, and because radionuclide concentrations may be expected to approach the lower limits over extended time periods. Accordingly, we assume in this paper that the actinide concentrations are controlled by the most stable and insoluble precipitates for which data are available. The consequences of precipitation of progressively less stable precipitates will be explored in future calculations, and upper limits of radionuclide concentrations controlled by solid precipitation will be estimated. In the case of Pu, for example, we have begun to explore the upper limits to Pu concentration as controlled by the precipitation of amorphous $Pu(OH)_4$. Comparison of modeling results with experimental results helps to identify phenomena which may revise our estimates of concentration limits. Processes such as sorption and aging of solids to forms of increasing crystallinity tend to lower element concentrations in solution, and increase the conservative nature of our estimates. However, consideration of colloid formation and colloid migration with the fluid phase may lead to an increase in our estimates of mobile concentrations over those made considering precipitation phenomena alone.

4.0 RADIONUCLIDE RELEASES

Annual actinide releases per failed waste package were calculated assuming that water flowing at a rate of 20 1/yr per waste package transports the actinides at the approximate concentrations measured at steady-state in Cycles 2 and 3 of the HBR-3-25 test. Each waste package was assumed to contain 3140 kg of fuel with an average burnup of approximately 33,000 MWd/MTM. The logarithms of the waste package 1000-year inventory fractions transported annually for each actinide under such conditions is given in Table 4. These releases are at least three orders of magnitude lower than the Nuclear Regulatory Commission (NRC) requirement in 10 CRF 60.113[11] that annual radionuclide releases during the post-containment period shall not exceed one part in 100,000 of the 1000-year inventories. The calculated annual release results would appear to be particularly encouraging for Pu and Am because isotopes of these two actinide elements account for about 98% of the total activity present in spent fuel at 1000 years. These values may be conservative in that they are based on the higher steady-state Pu and Am concentrations measured at 25°C and assume a conservative (high) estimate of the water flux through the repository. The calculated releases do, however, assume maintenance of steady values for actinide concentrations over time, whereas the geochemical simulations suggest that actinide concentrations, and U concentrations in particular, may vary with time. Confidence in such release predictions will be greatly increased when the chemical mechanisms of solubility control are identified and successfully modeled.

Table 4. Annual Actinide Releases as a Fraction of the 1000-Year Inventories Based on HBR-3-25 Test Data

Actinide	Concentration Log(M)	Log (Release)*
U	-5.9	-8.6
Np	-8.9	-8.8
Pu	-8.4	-9.0
Am	-9.8	-9.1

*Assumes water flow rate of 20 1/yr per waste package transporting actinides at the indicated concentrations. Each waste package is assumed to contain 3140 kg of 33,000 MWd/MTM burnup PWR fuel.

Measured activities of the more soluble fission product radionuclides Cs-137, Sr-90, Tc-99 and I-129 continuously increase in solution at rates generally corresponding to annual release rates in the range of 10^{-4} to 10^{-3} of specimen inventory per year (Figure 4). These release rates imply a problem in meeting the NRC 10^{-5} annual fractional release limit for the more soluble radionuclides if the waste form alone is expecting to carry the burden of compliance in the unanticipated case of large quantities of water contacting the waste. However, there are two factors that make these release rates uncertain. First, the degree to which these radionuclides are preferentially released from grain boundaries where they may be concentrated during irradiation has not yet been determined. Preferential release could be expected to provide a lesser contribution over time as exposed grain boundary inventories are depleted and release rates approach the congruent fuel matrix dissolution rate. A second factor is the extent to which the fuel may be degraded over time by exposure to the repository environment. Degradation of the fuel as a result of oxidation to higher oxygen stoichiometries such as U_3O_8, or as a result of preferential grain boundary dissolution, may cause increases in surface area and increased rates of nuclide dissolution from grain boundaries and from the fuel matrix over time.

Flow-through tests in which uranium minerals do not precipitate are being developed to measure the degree to which soluble nuclides are preferentially released during the initial phases of fuel dissolution. Dissolution tests using spent fuel specimens that have been degraded by slow, low-temperature oxidation are also planned. Results from these tests should provide a better understanding of potential long-term releases of the soluble and volatile radionuclides. Additional characterization of potential release of C-14 is important because it is soluble as bicarbonate and could also be released in the vapor phase as CO_2.

5.0 CONCLUSIONS

Laboratory testing and geochemical simulation of the dissolution of spent fuel under conditions selected for relevance to the proposed Yucca Mountain repository have resulted in the following conclusions.

1. Radionuclides of interest in spent fuel appear to fall into three categories of potential release mechanisms: 1) radionuclides whose release appears to be controlled by concentration-limiting mechanisms, 2) more highly soluble radionuclides, and 3) radionuclides that are released in the vapor phase (principally C-14).

2. The principal radionuclides whose releases appear to be controlled by concentration-limiting mechanisms are the actinides U, Np, Pu, Am and Cm.

Steady-state concentrations measured for these actinide elements are at least three orders of magnitude lower than those required to meet NRC release limits based on conservative estimates of water fluxes through the repository. This result is of particular significance because isotopes of Pu and Am account for about 98% of the activity in spent fuel at 1000 years. However, results from geochemical modeling suggest that steady-state concentrations may vary significantly with time because of changes in solution composition and the identity of precipitating phases.

3. Good agreement between measured and predicted concentrations was obtained for Np based on equilibration with NpO_2 at 25°C when the oxygen fugacity in the simulation was set at 10^{-12} bars. A broad range of solubilities that bracketed the measured values were predicted for Pu depending upon the assumed oxygen fugacity and solubility-controlling phase. Measured Am concentrations were less than predicted based on data for equilibration with $Am(OH)CO_3$ and $Am(OH)_3$.

4. Dissolution rates for soluble radionuclides (Cs-137, Sr-90, Tc-99 and I-129) exceeding 10^{-5} of specimen inventory per year were measured during the laboratory tests. The implications of these data relative to long-term release of soluble radionuclides from a failed waste package are uncertain. The degree to which these radionuclides were preferentially released from grain boundaries where they may have concentrated during irradiation was not determined. Preferential release could be expected to provide a lesser contribution over time as exposed grain boundary inventories are depleted. However, physical degradation of the fuel over time from exposure to the oxidizing repository environment may result in accelerated release of soluble nuclides.

5. Additional work is required to identify solid phases that control actinide concentrations, and to acquire reliable thermodynamic data on these phases for use in geochemical modeling. In this regard, identification of any stable suspended phases that can be transported by water movement is also important. In addition, we must better understand the potential release of soluble and volatile radionuclides, which may initially depend on preferential release from gap and grain boundary inventories, but may ultimately depend on the rate of fuel degradation by oxidation or other processes in the post-containment repository environment.

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Lawrence Livermore National Laboratory

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January 22, 1991

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QA

SEPDB Administrator
Sandia National Laboratory
Organization 6310
P.O. Box 5800
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Subject: Submission of Data to the SEPDB

Attached are a Technical Data Information Form (TDIF) and associated data for inclusion in the SEPDB. These data are taken from two reports:

1) C.N. Wilson, "Results from Cycles 1 and 2 of NNWSI Series 2 Dissolution Tests," HEDL-TME85-22, May 1987.

2) C.N. Wilson, "Results from the NNWSI Series 3 Spent Fuel Dissolution Tests," PNL-7170, June 1990.

The pertinent solubility data taken after "steady-state" was reached are given in Table 1. In cases where several values from different samples with different geometries and different burnup histories were shown, the most conservative upper value is indicated. Since we don't know the cause of the scatter, it is prudent to assume the worst case, pending a better understanding of the spread in the steady-state solubilities. Where filtered and unfiltered values were available, the filtered data were used because solubility is the information desired.

Table 2 indicates the specific source for each data value.

For slow flow of water over the spent fuel, the solubility can be used to determine the mass of each radionuclide dissolved as a function of time. Given solubilities, C, a flow rate of water contacting the spent fuel, Φ, and a time, t, over which dissolution occurs, the total amount of any nuclide, i, dissolved and transported, M_i, is given by

$$M_i = C_i \Phi t$$

Please contact Mike Revelli of my staff at FTS 532-1982 for further information.

L.J. Jardine
LLNL Technical Project Officer
for the Yucca Mountain Project

LJJ/JB:jw

Attachments

c: C. Newbury, YMPO

Table 1. Solubility Data, C_i

Species	Upper Limit Steady-State Concentration (μg/ml)	
	25°C	85°C
U	≤ 5	≤ 0.5
$^{239+240}Pu$	≤ 5×10^{-3}	≤ 6×10^{-5}
^{241}Am	≤ 3×10^{-4}	≤ 1.5×10^{-7}
^{244}Cm	≤ 1.2×10^{-5}	≤ 2.4×10^{-9}
^{237}Np	≤ 4×10^{-4}	≤ 1.4×10^{-3}

Only data for the solubility limited species are listed in the above table.

Table 2. Solubility Data Sources

Species	References	
	25°C	85°C
U	Ref. 1, Fig. 5	Ref. 2, Fig. 3.1
$^{239+240}Pu$	Ref. 1, Fig. 6	Ref. 2, Fig. 3.12
^{241}Am	Ref. 1, Fig. 7	Ref. 2, Fig. 3.15
^{244}Cm	Ref. 1, Fig. 8	Ref. 2, Fig. 3.18
^{237}Np	Ref. 2, Fig. 3.20	Ref. 2, Fig. 3.20

Conversion factors from pCi to µg taken from Ref. 2, Table A.1.

The following describes data and an analysis procedure to obtain the release rate time response for a fully wetted mass of spent fuel dissolving without solubility limitations in water. The description is from an LLNL report UCRL-ID-107289 published in December, 1991.

Waste package analysts and designers have to understand the long term dissolution of waste form in groundwater to safely dispose of high level nuclear waste in an underground repository. The dissolution and transport processes in groundwater flow are generally considered to be the main route by which radionuclides could be released to the biosphere from a geological repository.

Many researchers have investigated the dissolution of UO_2, spent fuel and uraninite (a naturally occurring UO_2 mineral) in aqueous solutions, under either reducing or oxidizing conditions, and as a function of various other environmental variables. Experimental data on the dissolution rates of UO_2, spent fuel and uraninite have been reviewed by Arnell and Langmuir,[1] Parks and Pohl,[2] Bruno et al,[3] and most recently by Grambow.[4]

Important variables considered in the many investigations were pH, temperature, oxygen fugacity, carbonate/bicarbonate concentrations and other reacting media. The dissolution data are very scattered, and vary as much as six orders of magnitude.[4] The dependence of the dissolution rates of UO_2, spent fuel and uraninite on these variables is not clear because of uncertainties regarding redox chemistry of uranium in solutions and in solid phases, secondary-phase formation, and surface area measurement. In addition, the previous studies were conducted under experimental conditions which were either inadequately controlled or which simulated complex repositorial conditions. The results of such studies are difficult to interpret. Several of these researchers have developed equations to correlate dissolution rates as a function of relevant variables. [5-8] However, none of the rate laws is universal, and inconsistencies or incompatibilities among the proposed laws are common.

Data indicate that UO_2 is easily oxidized to U_4O_9 and U_3O_7 in an air[9,10] and can be further oxidized to either U_4O_8 [9,10,11] or schoepite, $UO_3 \cdot 2H_2O$.[12] The UO_2 surface oxidation may lead to higher leach rates because of possibly higher dissolution rates of U_3O_7, U_4O_8 or schoepite relative to that of UO_2 [4] because of the increase of surface area of the fuels due to surface cracking.

Discussion

We are estimating a source term for liberation of radionuclides from spent fuel dissolving under conditions of temperature and water composition related to those anticipated for a potential repository at Yucca Mountain. This is done in the same spirit as estimates that have been made for repositories in Germany[13] and Sweden.[14] It is implicit in the following treatment that fission products are dissolved congruently with the UO_2 fuel matrix, except for those volatile species that have partially vaporized and that fraction that has migrated to near-surface grain boundaries and are possibly dissolved independent of the matrix dissolution. Most fission products and higher actinides are distributed throughout the UO_2 matrix, however.

Recent measurements on UO_2 [15] and spent fuel (SF)[16] under comparable conditions have provided dissolution rates for UO_2 between 25°C and 85°C in waters of various composition and for SF in deionized water (DIW) at 25°C. These experiments were done in contact with air. The results are shown in Figures 1 and 2. The rate of dissolution of SF in DIW at 25°C is 1.2–1.7 x 10^{-12} g cm^{-2} sec^{-1}. This is similar to the rate for UO_2 in DIW at 25°C at ~5 x 10^{-12} g cm^{-2} sec^{-1}. Given the great variability in other reported values[4] this is reasonable agreement. In fact, the observed dissolution rate for SF at 25°C is about the same as that of UO_2 in (DIW + Ca + Si), a simulation of ground water.[14]

A model for dissolution is used in which the dissolution front propagates linearly in time, much like a recently published model for the advance of the oxidation front during oxidation of UO_2 and spent fuel.[16-19] This implies that the particle geometry is retained. We can describe the change in characteristic dimension of a SF particle (a sort of "radius"), X as follows:

$$X(t) = X_o - \left(\frac{Q}{\rho}\right)t \ , \qquad (1)$$

where $X(t)$ = the characteristic dimension as a function of time
X_o = the original dimension (half of the actual size)
t = time
Q = dissolution rate per unit area
ρ = density

The time for complete dissolution of a particle of original size X_o is then

$$t_\infty = \frac{X_o \rho}{Q} \qquad (2)$$

This dissolution time is proportional to size, of course, and for an ensemble of particles of different sizes, t_∞ for the ensemble is that for the largest particle.

Some data are available on the size distribution of spent fuel fragments. These data are given for two different fuels but the distributions are quite similar. The aggregate of these two sets of data can be adequately described by the simplified distribution shown in Table I.

Table I

Approximate Size (cm) ($2X_o$)	Weight (Volume) Fraction
0.15	.02
0.25	.14
0.35	.29
0.50	.38
0.70	.17

Using the relationship of equation (1), we can calculate the time to dissolve a given weight (volume) fraction of an amount of SF as a function of time. For generality, we treat time as the dimensionless quantity t/t_∞, with t_∞ defined above. This is shown in Figure 3 for the size distribution given in Table I*, and also for a single size with $X_o = 0.35$ cm. Here V_o and V(t) are the original volume of a particle and its volume at arbitrary time, respectively, The volume is proportional to the characteristic dimension

$$V_o = kX_o^3 \text{ and } V(t) = kX^3(t)$$

where k is a constant depending on shape. Since geometry is retained, as noted above,

* Each size was calculated separately and the time responses were added together.

$$\frac{V(t)}{V_o} = \left(\frac{X(t)}{X_o}\right)^3 = 1 - 3\left(\frac{Q}{X_o\rho}\right)t + 3\left(\frac{Q}{X_o\rho}\right)^2 t^2 - \left(\frac{Q}{X_o\rho}\right)^3 t^3, \quad (3)$$

and the dissolution rate is $-\frac{d\left(\frac{V(t)}{V_o}\right)}{dt}$

Initially, i.e., $t \rightarrow 0$

$$\text{Rate } (t=0) = 3\left(\frac{Q}{X_o\rho}\right)$$

and the extrapolated time for total dissolution is

$$t^*_\infty = \frac{X_o\rho}{3Q}$$

In Figure 4 we show that the rate of dissolution relative to the initial rate varies with time for both the system with X_o = 0.35 cm and for the distribution of Table I.

The measured dissolution rate for UO_2[15] and spent fuel[16] allow us to calculate actual times for dissolution. As is evident from Figure 3, the overall dissolution rate is greatest at early time and approaches zero as t_∞ is approached; therefore, as a conservative approximation, we have also calculated the total dissolution time extrapolated from the initial rate, t^*_∞. These times calculated for the size distribution in Table I are given in Table II. The actual dissolution rates are derived from the bottom curve in Figure 1. We chose this curve as most representative of the expected ground water. The rate equation used is

$$Q(t)\ (g\ cm^{-2}\ sec^{-1}) = 6.43 \times 10^{-9} \exp -\left(\frac{4740}{RT(K)}\right) \quad (R \text{ is in cal/mole K}) \quad (5)$$

Table II

Temperature (°C)	Dissolution Time (years)	
	t^*_∞	t_∞
25	8.0×10^3	5.5×10^4
85	2.2×10^3	1.5×10^4

Conclusions

These times are calculated for the case of bare fuel immersed in unlimited quantities of flowing water at flow rates sufficient to prevent any species from forming a saturated solution. Nonetheless, this estimate provides a "core" value on which to apply "credits" corresponding to features of realistic repository performance such as frequency of cladding and container failure, actual amounts of ground water and various transport rates, etc. Of course, this "core" estimate is based on only one particular dissolution rate, as is discussed above. Future measurements of dissolution rate may change this value considerably. The estimates presented here ignore the possibility that grain boundary dissolution behaves differently than bulk SF dissolution.

Dissolution tests are now under way that are designed to define the mechanism of the dissolution process of UO_2 and SF in terms of oxidizing potential, temperature, pH and other water composition variables generally appropriate to a potential repository at Yucca Mountain. When these tests are completed, considerably more realistic estimates will be possible. These tests will also clarify the contribution of radionuclides from grain boundaries to the total dissolution rate.[16]

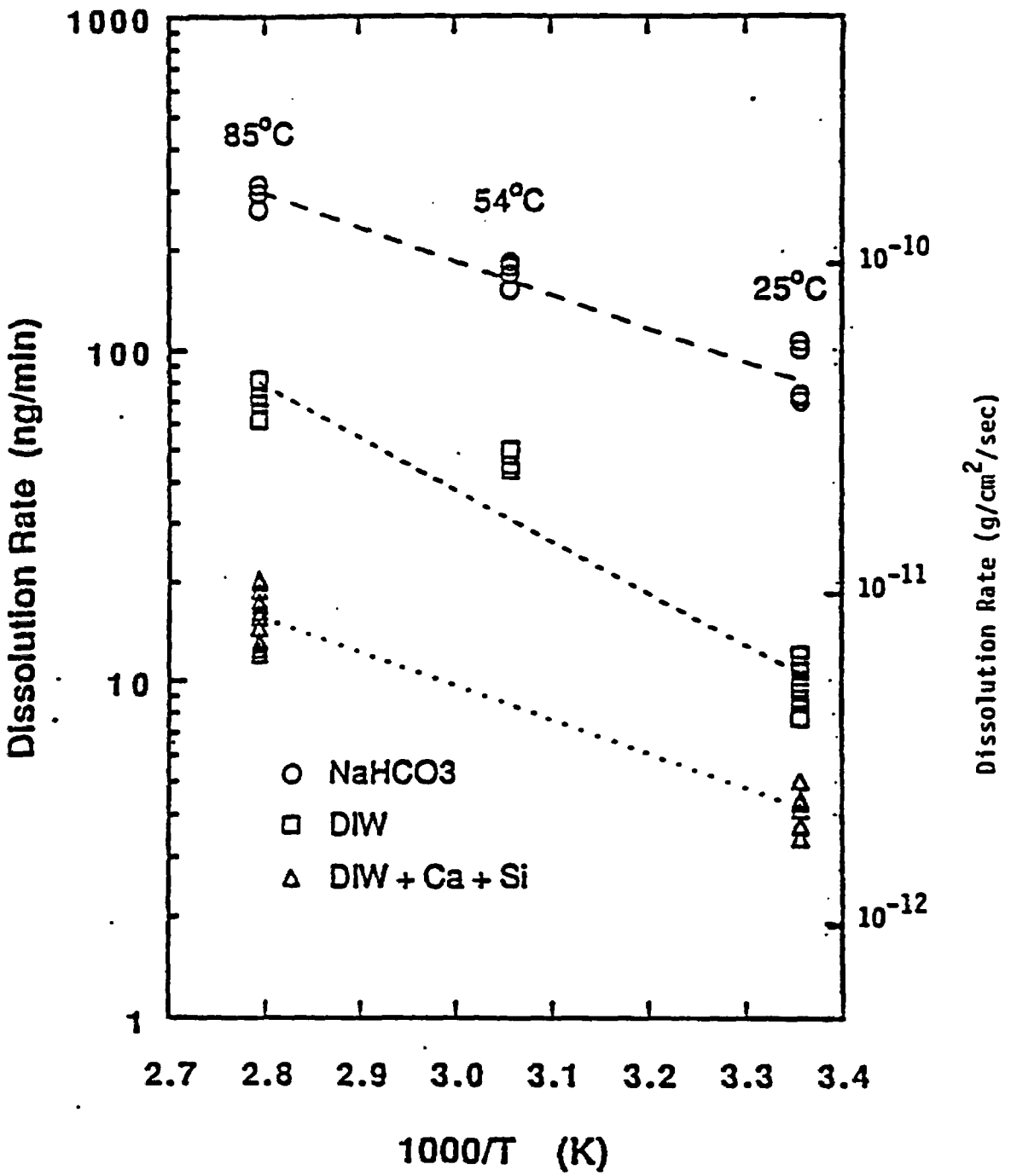

Figure 1. Arrhenius plots of the dissolution rate of UO_2 in waters of various composition.[15]

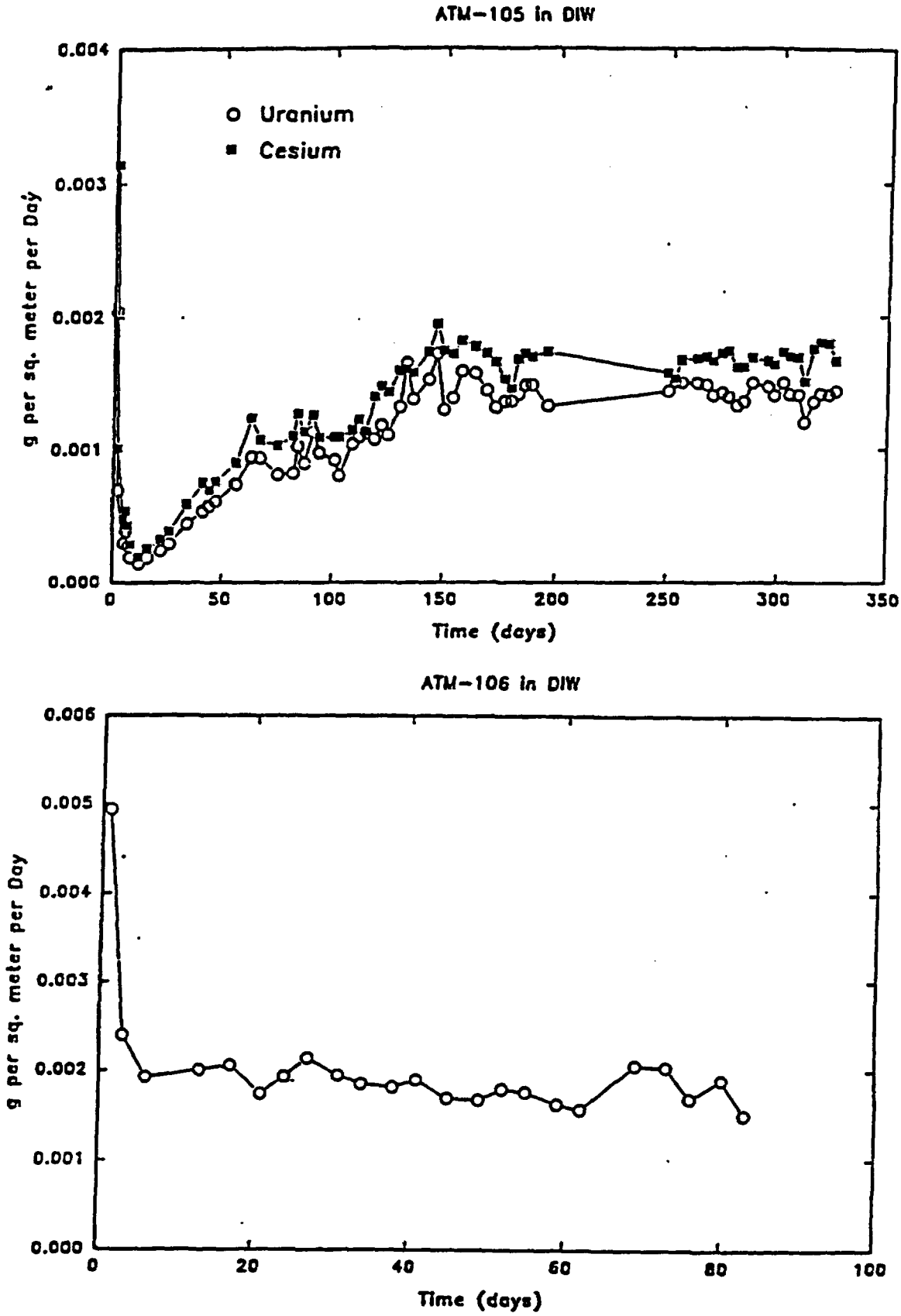

Figure 2. The approach to steady-state of the dissolution rate of two spent fuel samples.[1] Experiments were done at 25°C using deionized water (DIW).

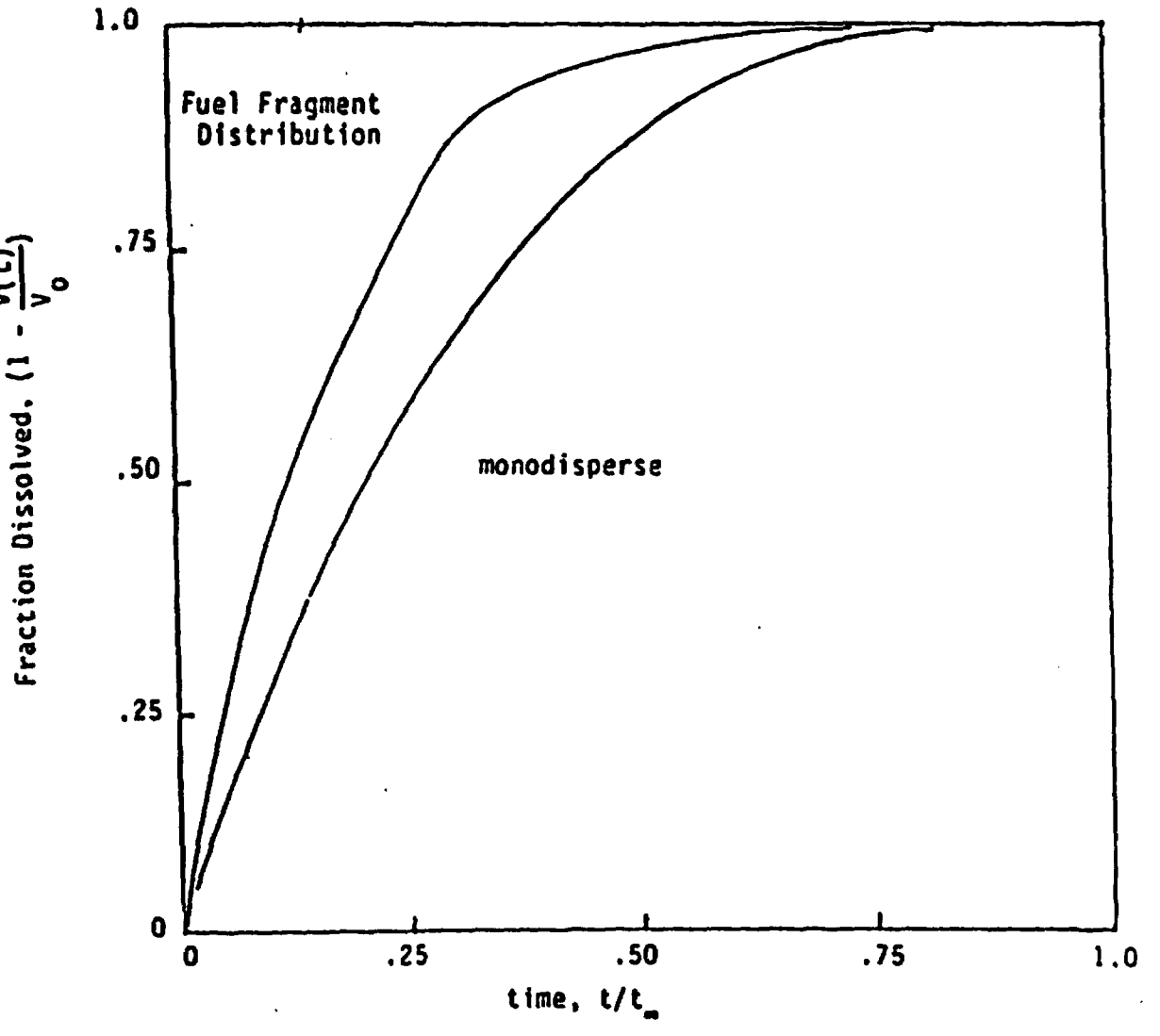

Figure 3. Calculation of the fractional dissolution in terms of dimensionless time, according to equation (3). Monodisperse refers to a single particle size.

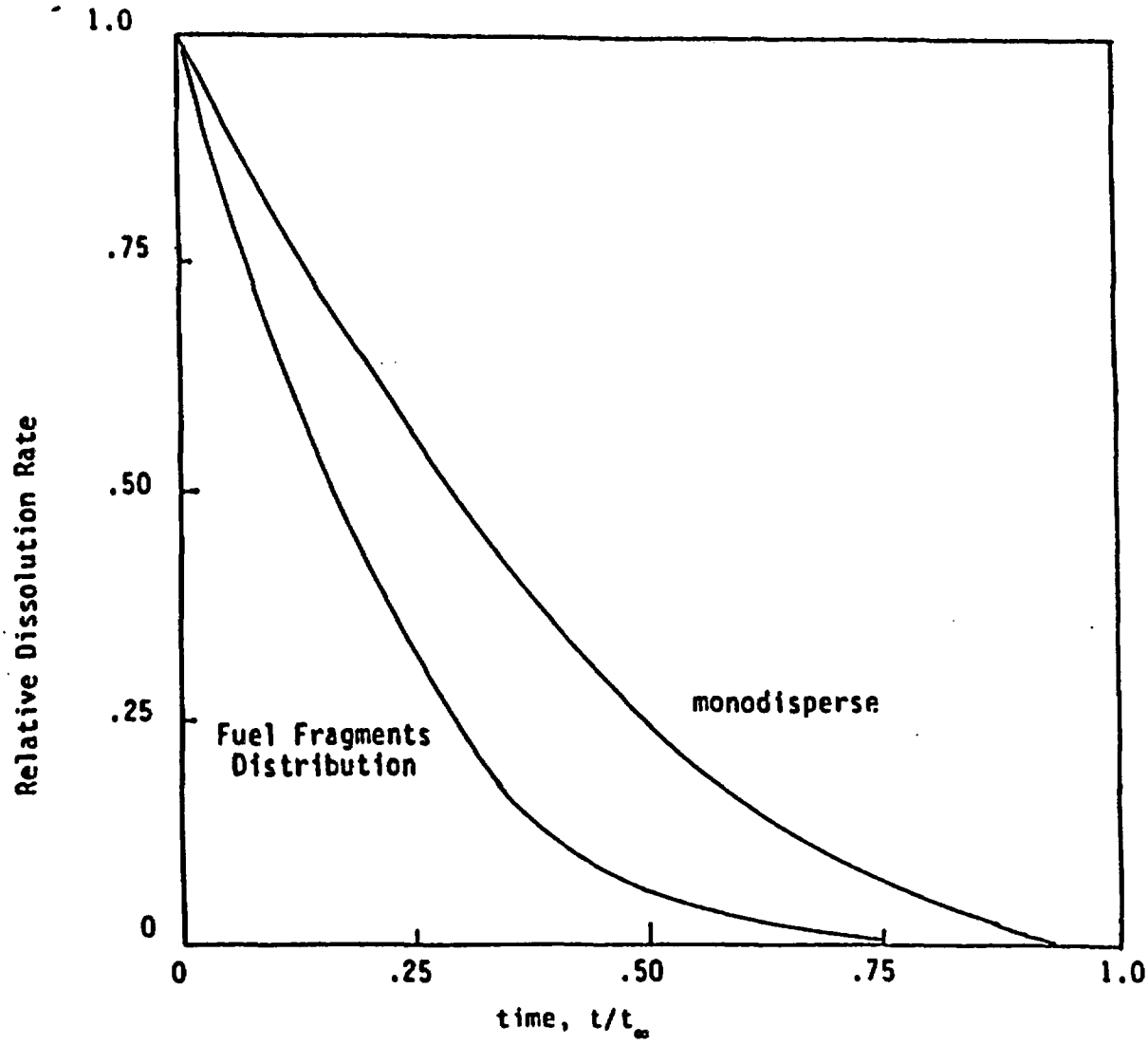

Figure 4. Evolution of the normalized dissolution rate with time as the particle size decreases, according to equation (3).

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2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

TABLE II
Phases Identified on Reacted UO_2 Surface

Phase	Formula	Appearance
Schoepite	$UO_3 \cdot 2H_2O$	Dark yellow crystals
Dehydrated Schoepite	$UO_3 \cdot 0.8H_2O$	Yellow crystals with reflective face
Compreignacite	$K_2U_6O_{19} \cdot 11H_2O$	Yellow crystals
Uranophane	$Ca(UO_2)_2(SiO_3)_2(OH)_2 \cdot 5H_2O$	Fine white needles
Boltwoodite	$K(H_3O)UO_2(SiO_4) \cdot nH_2O$	Yellow crystals
Sklodowskite	$Mg(UO_2)_2(SiO_3OH)_2 \cdot 5H_2O$	Fine needles
Becquerelite	$CaU_6O_{19} \cdot 10H_2O$	Dark yellow crystals
Fluoropolymer	Not determined	White feathers

J.K. Bates, *Identification of Secondary Phases Formed During Unsaturated Reaction of UO_2 with EJ-13 Water*, Materials Research Society Symposium proceedings 176, 499 (1990).

Table 2.

U		
Solid	Concentration	
	mg/l	molality (moles/kg)
Haiweeite $Ca(UO_2)_2Si_6O_{15}{\cdot}5H_2O$	0.1641E-3	0.6893E-9
Soddyite $(UO_2)_2SiO_4{\cdot}2H_2O$	0.015	0.6096E-7
Sklodowskite $Mg(H_3O)_2(UO_2)_2-(SiO_4)_2{\cdot}4H_2O$	11.05	0.4642E-4
$CaUO_4$	12.59	0.5289E-4
Schoepite $UO_3{\cdot}2H_2O$	38.90	0.1634E-3
$UO_2(OH)_2$(beta)	56.73	0.2383E-3
Uranophane $Ca(UO_2)_2(SiO_3)_2(OH)_2$	142.48	0.5986E-03

Table 3.

Np		
Solid	Concentration	
	mg/l	molality (moles/kg)
NpO_2	0.59	0.2468E-5
$NpO_2(OH)$(am)	129.39	0.5459E-3
$NaNpO_2CO_3{\cdot}3.5H_2O$	139.99	0.5906E-3
am = amorphous		

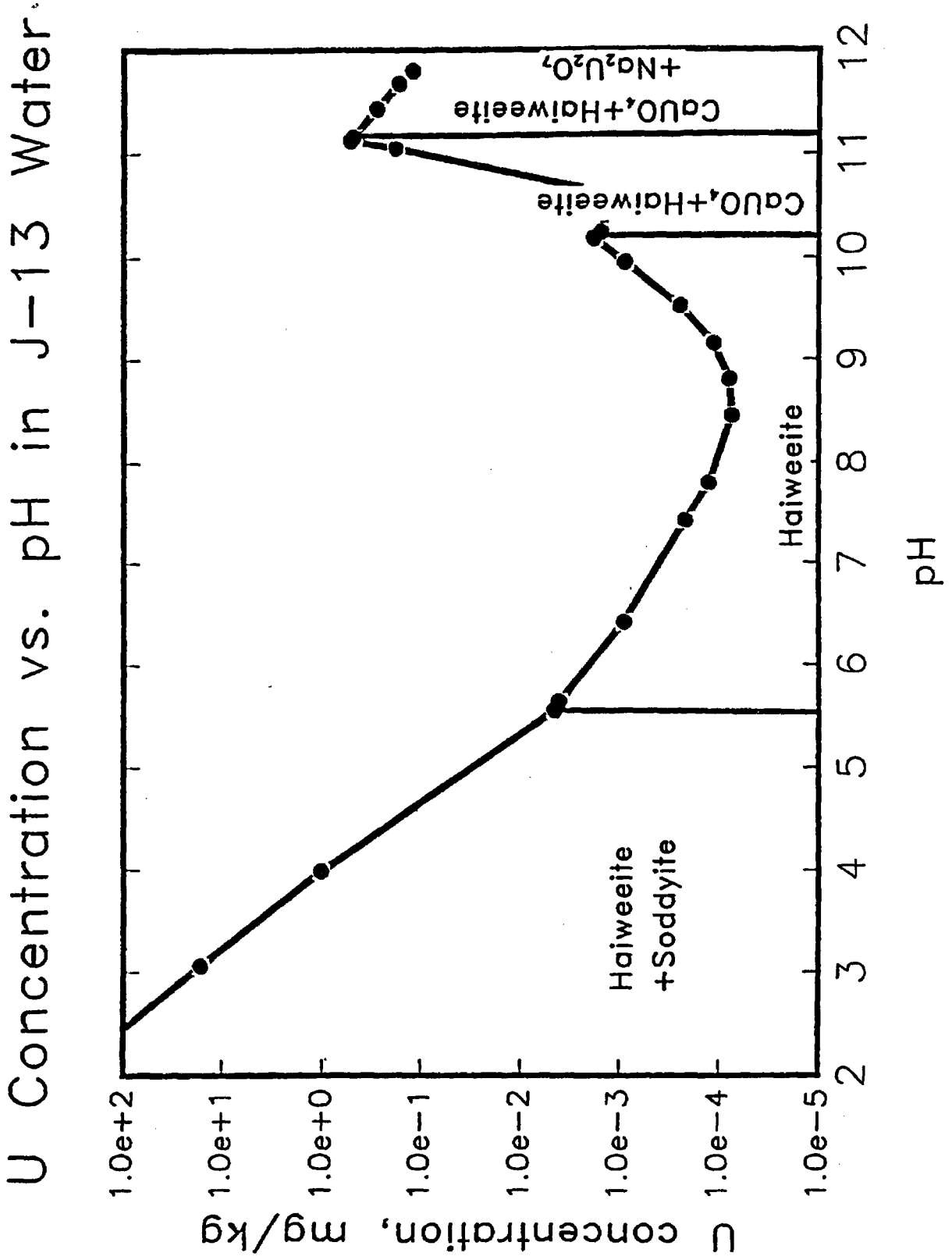
Table 4.

Pu		
Solid	Concentration	
	mg/l	molality (moles/kg)
PuO_2	0.39E-6	0.1612E-11
$PuO_2(OH)_2$	0.015	0.6204E-7
$Pu(OH)_4$	27.97	0.1146E-3

Table 5.

Am		
Solid	Concentration	
	mg/l	molality (moles/kg)
$AmOHCO_3$	0.0041	0.1696E-7
$Am(OH)_3$	8.42	0.3464E-4
$Am(OH)_3$(am)	158.66	0.6529E-3

C.J. Bruton, *Solubility Controls on Radionuclide Concentrations in Solution: Preliminary Results for U, Np, Pu, and Am*, LLNL draft report, November, 1990.



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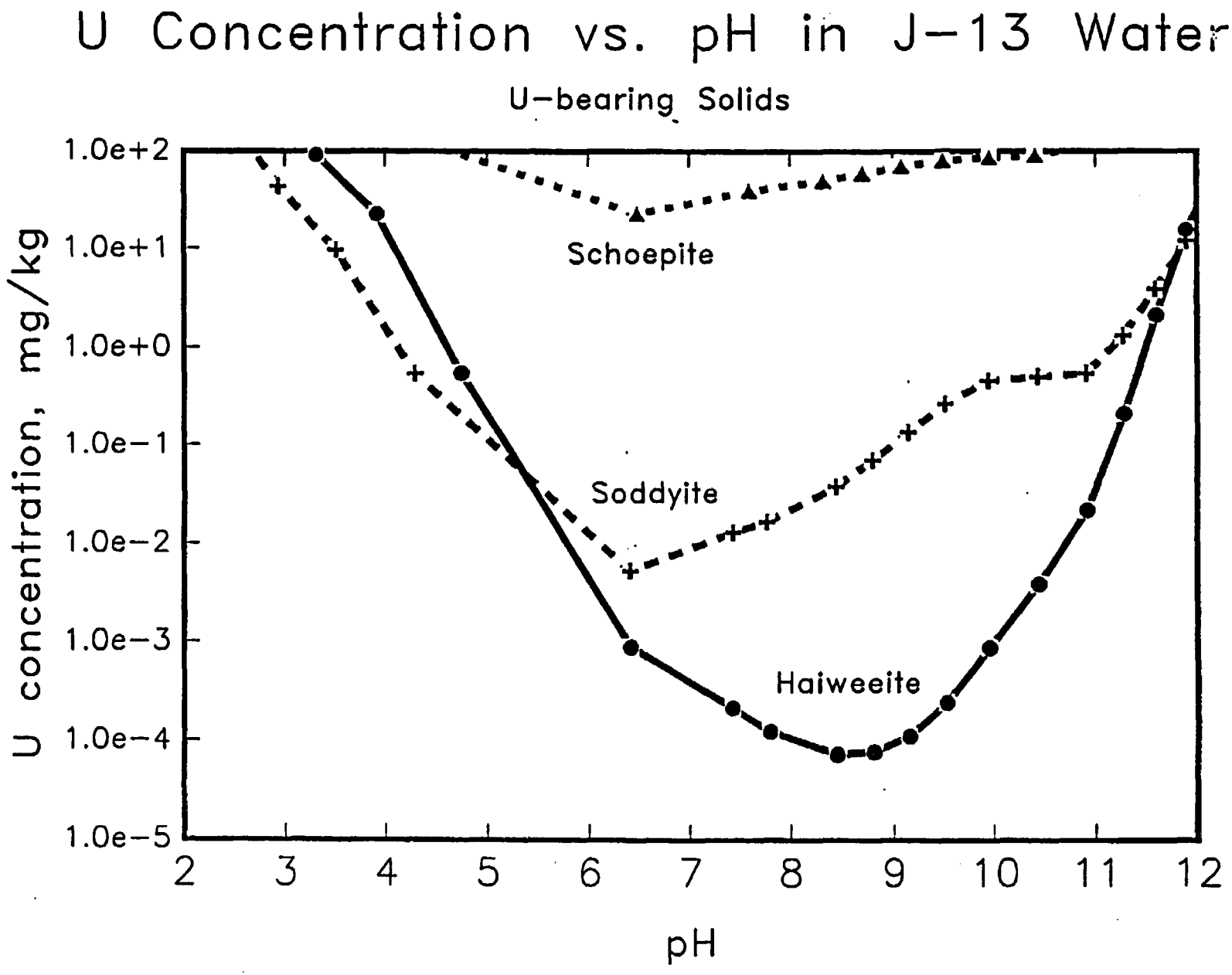

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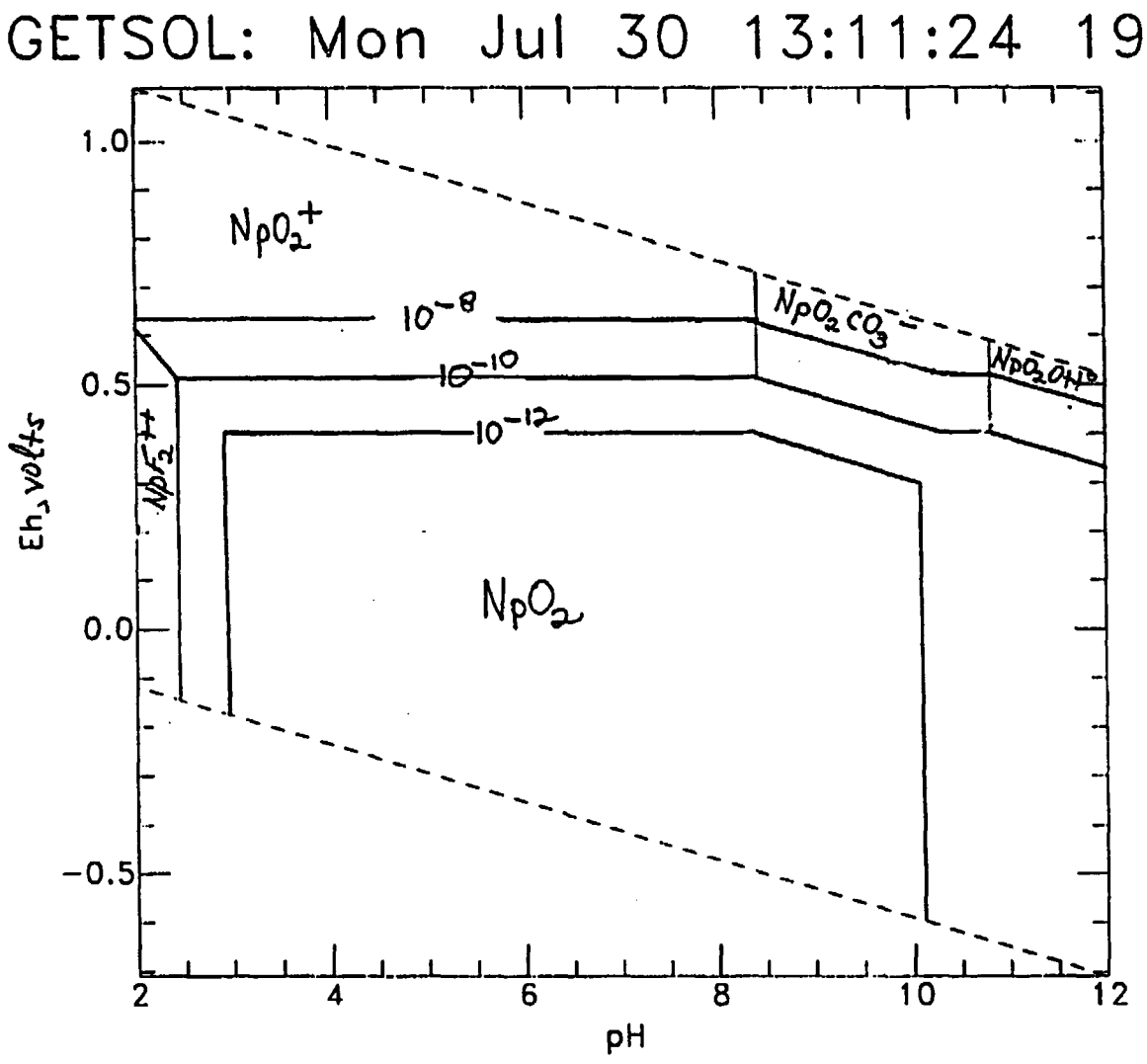

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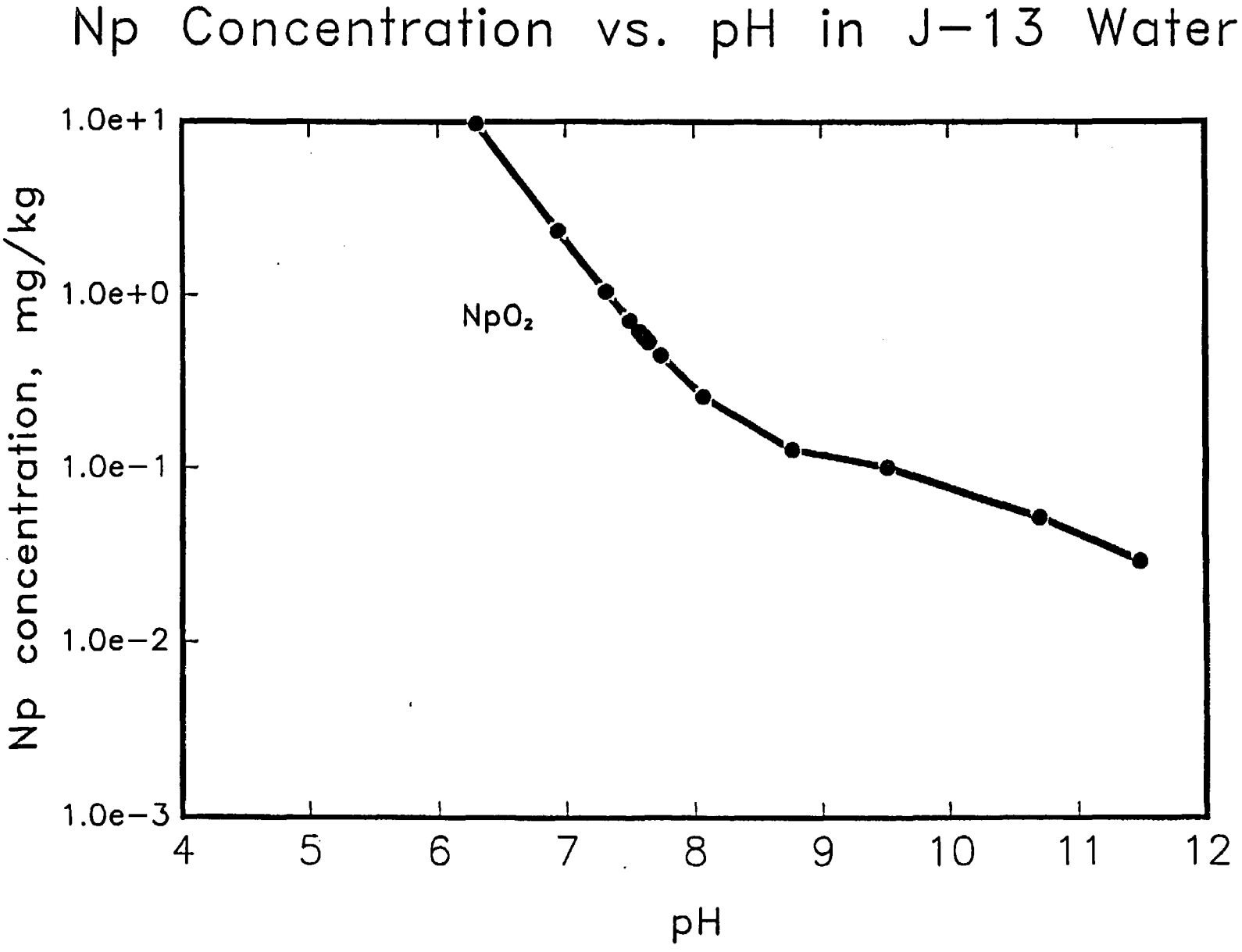
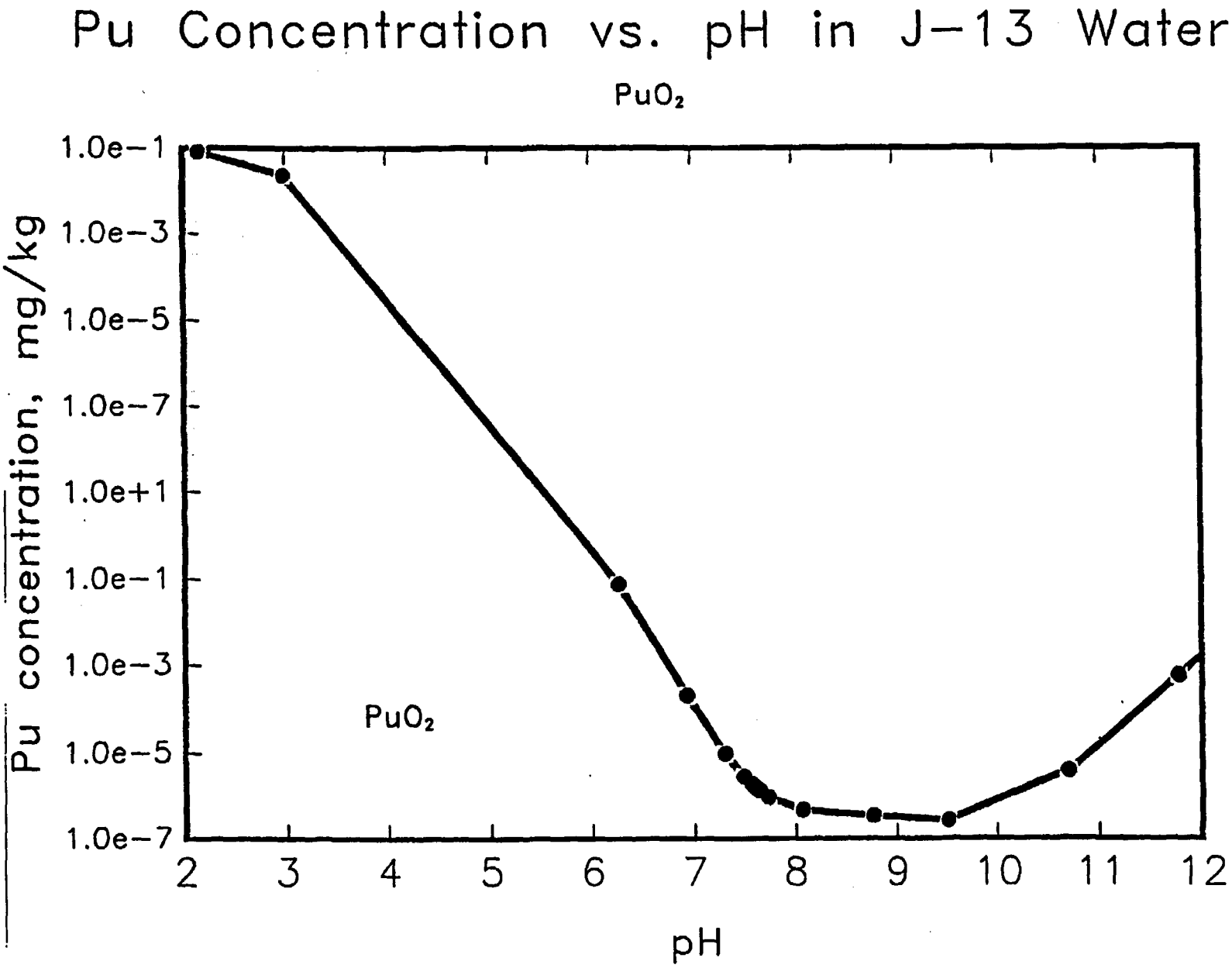


C.J. Bruton, *Solubility Controls on Radionuclide Concentrations in Solution: Preliminary Results for U, Np, Pu, and Am*, LLNL draft report, November, 1990.



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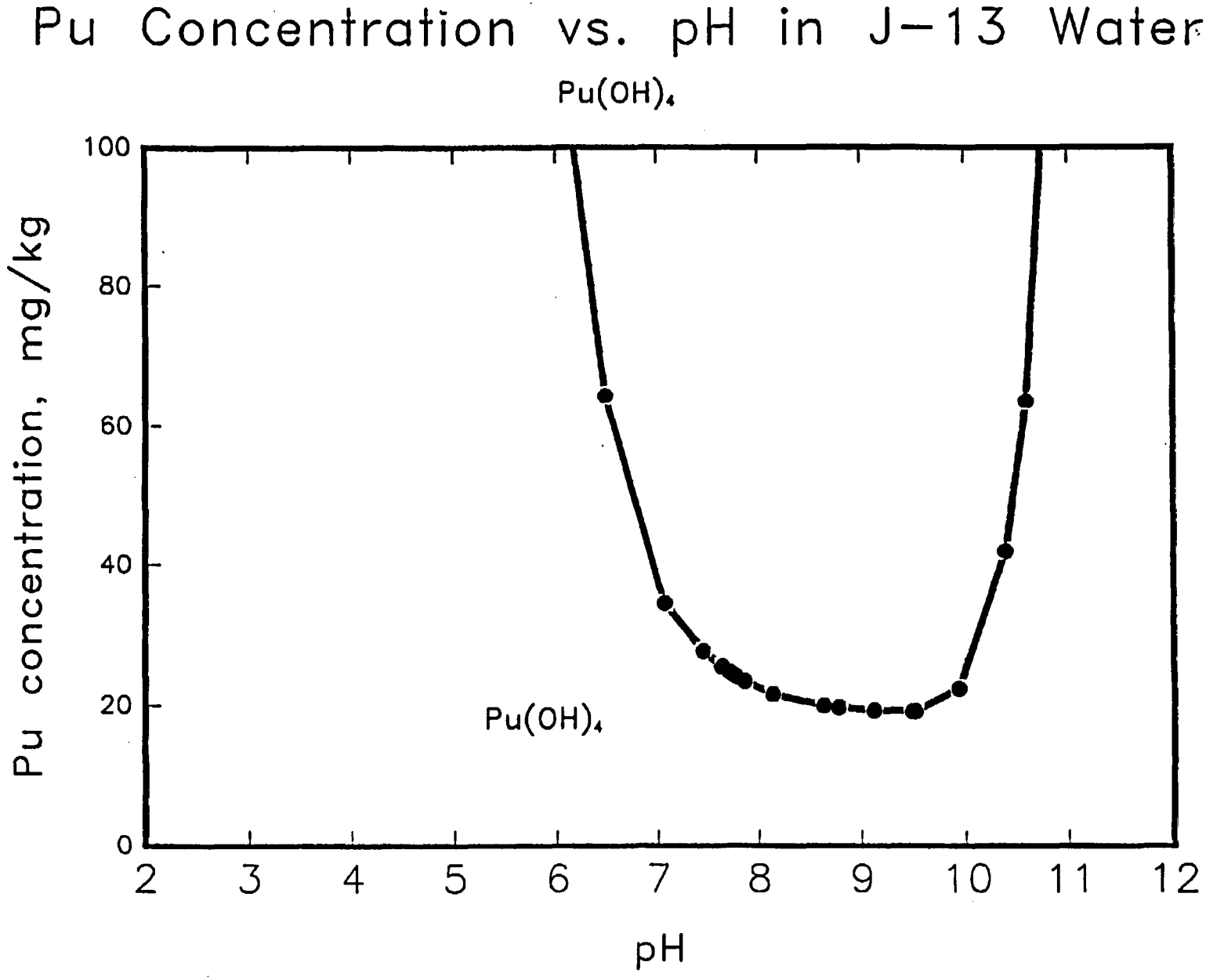

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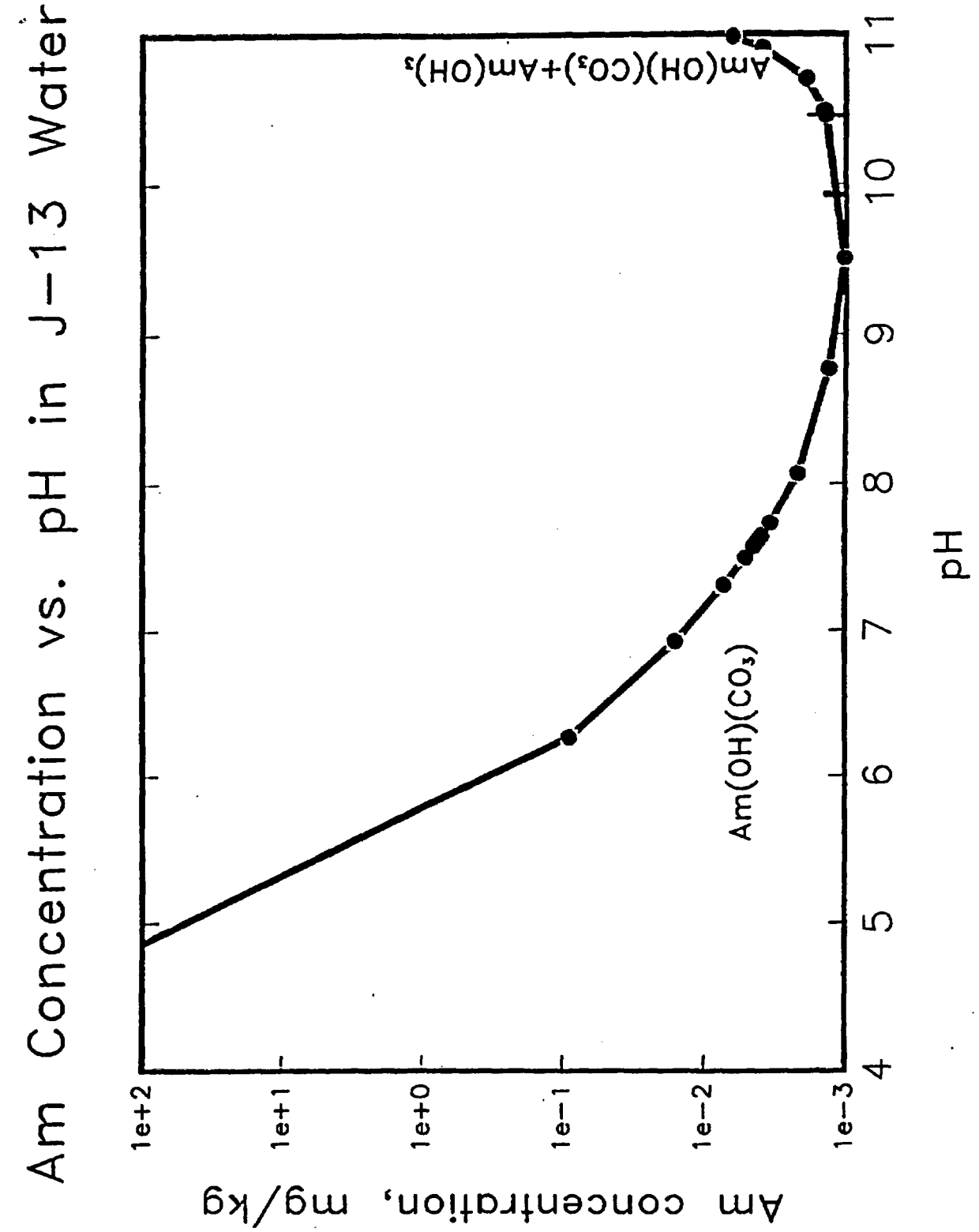

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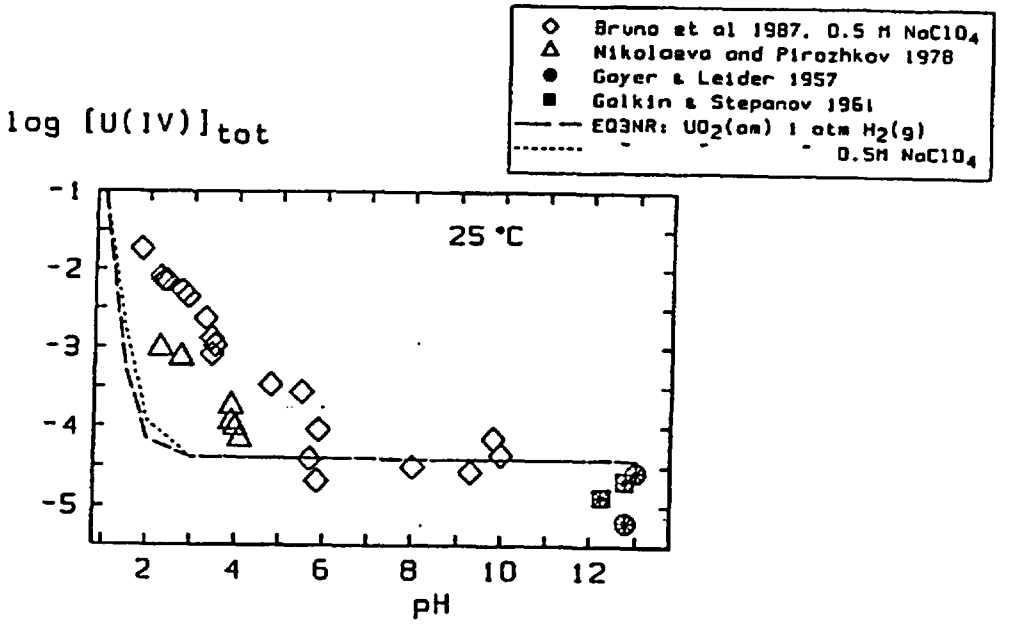


Figure 5.12 The solubility of UO_2(am) (both in diluted solutions and in 0.5 M $NaClO_4$) as a function of pH at 25°C.

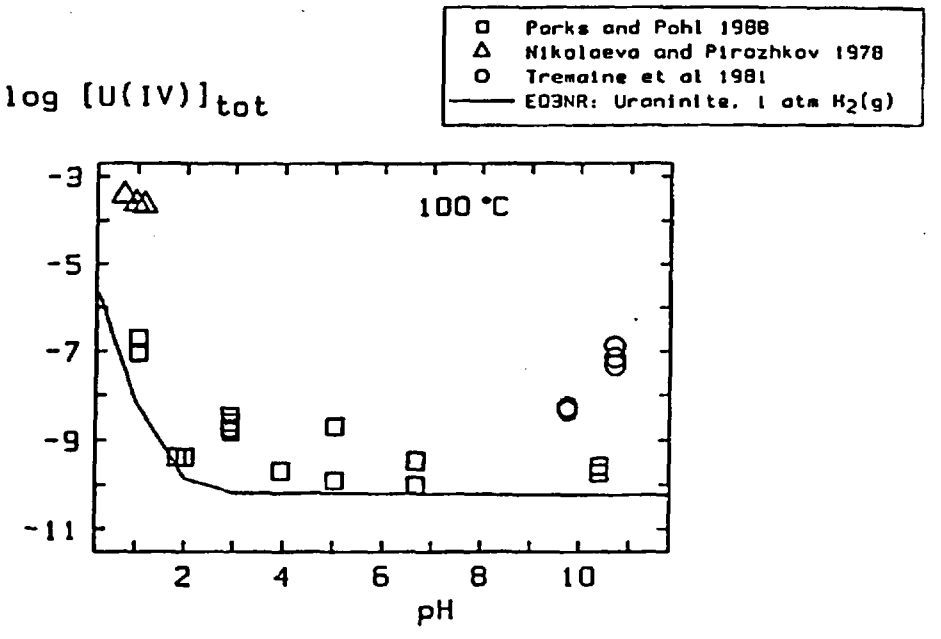


Figure 5.14 The solubility of crystalline UO_2(s) versus pH at 100°C.

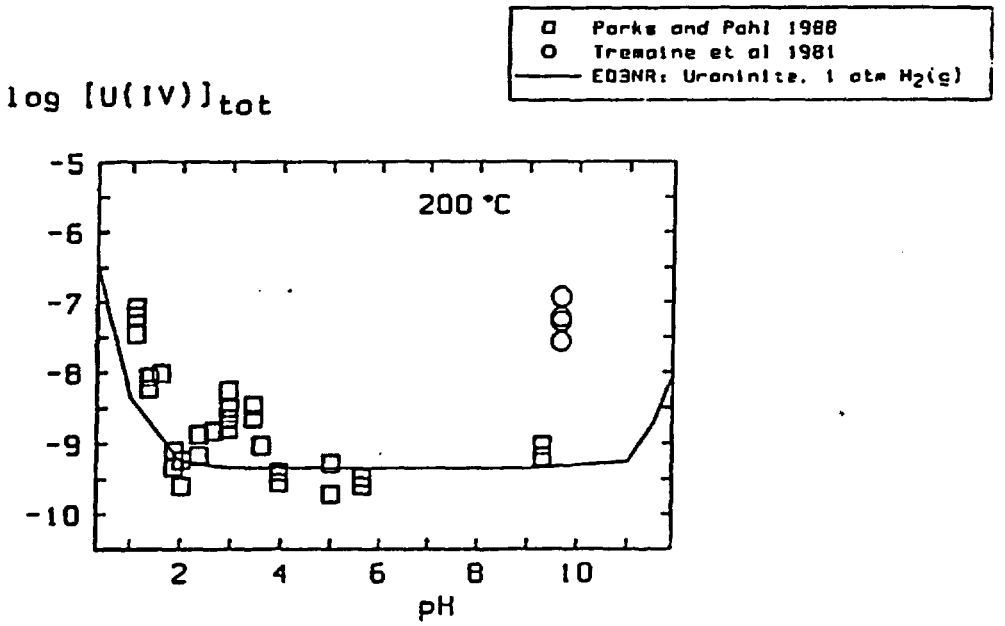


Figure 5.15 The solubility of crystalline UO_2(s) versus pH at 200°C.

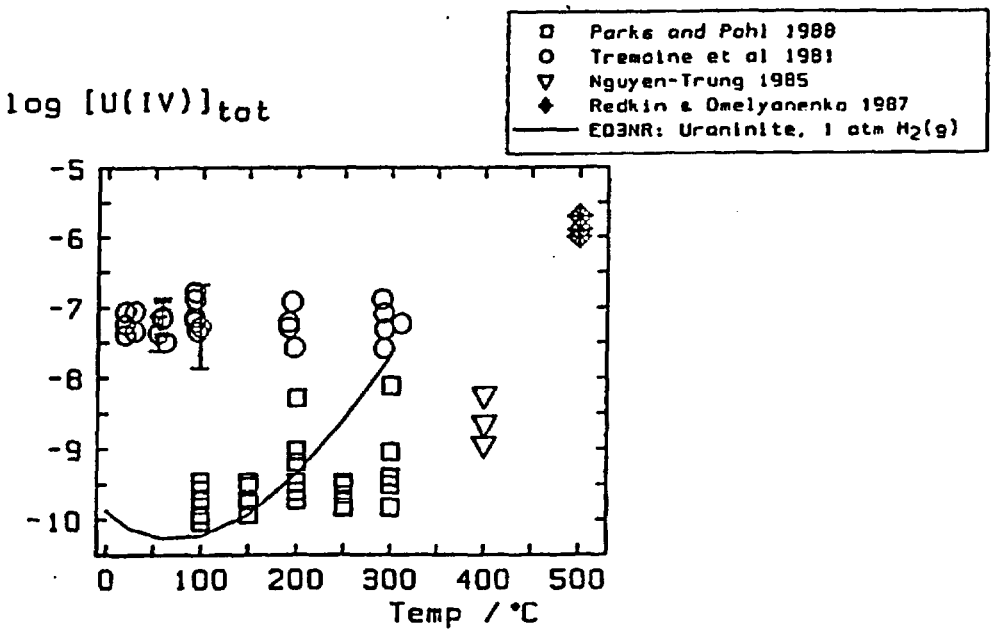


Figure 5.16 The calculated solubility of crystalline UO_2(s) in water at 1 atm H_2(g) versus T, compared with experimental literature values (for UO_2(c) solubility either in water or in diluted solutions of pH>5).

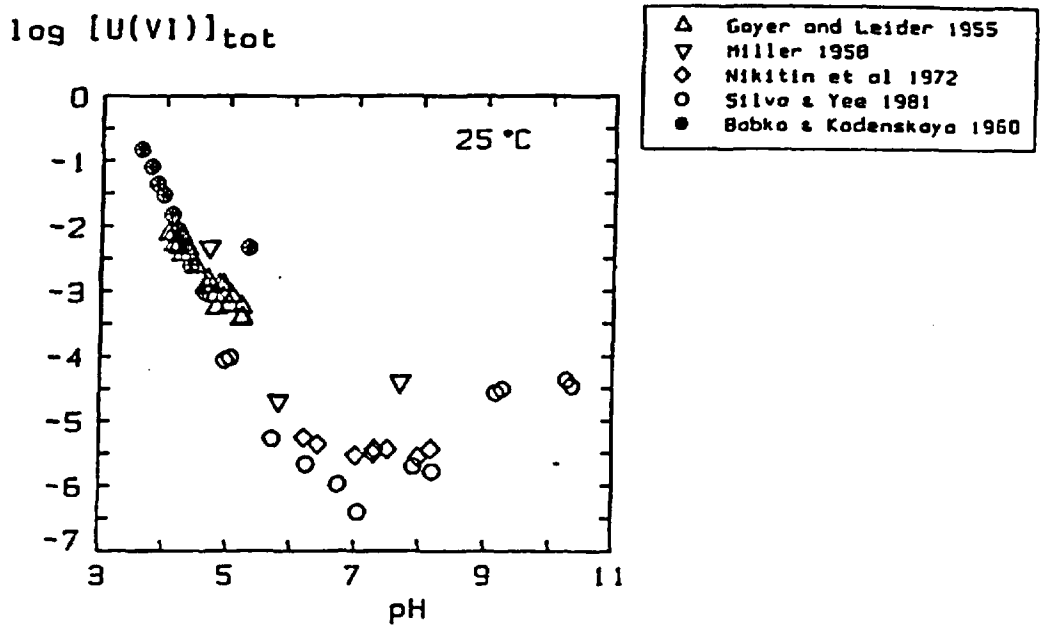


Figure 2.2 Some of the experimental literature data for the solubility of U(VI) hydroxide as a function of pH at 25°C.

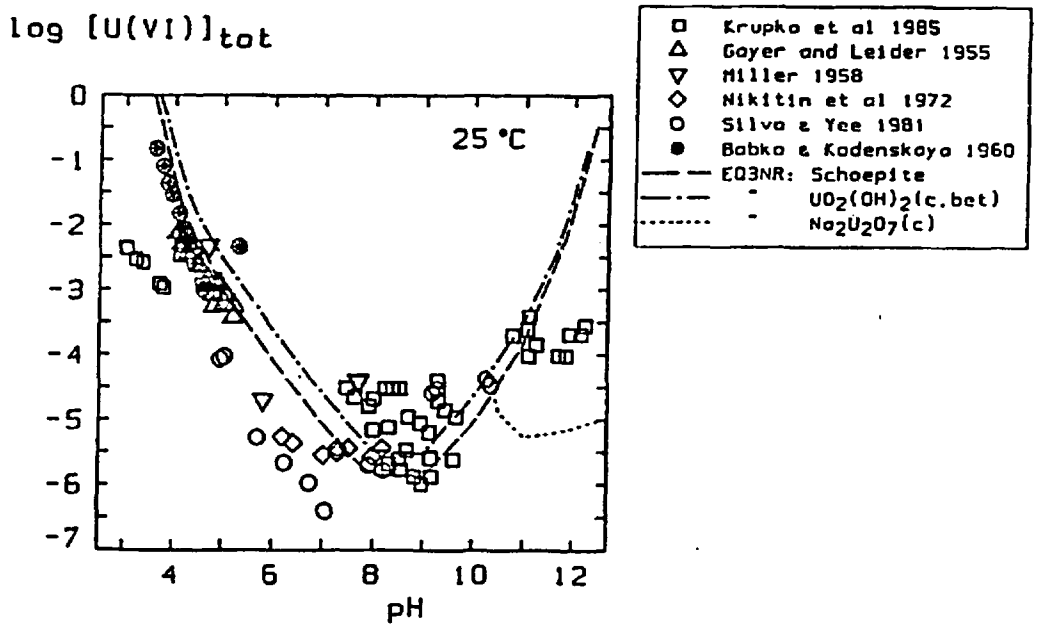


Figure 5.1 The solubility at 25°C of $UO_2(OH)_2$(c), schoepite and $Na_2U_2O_7$(c) as a function of pH.

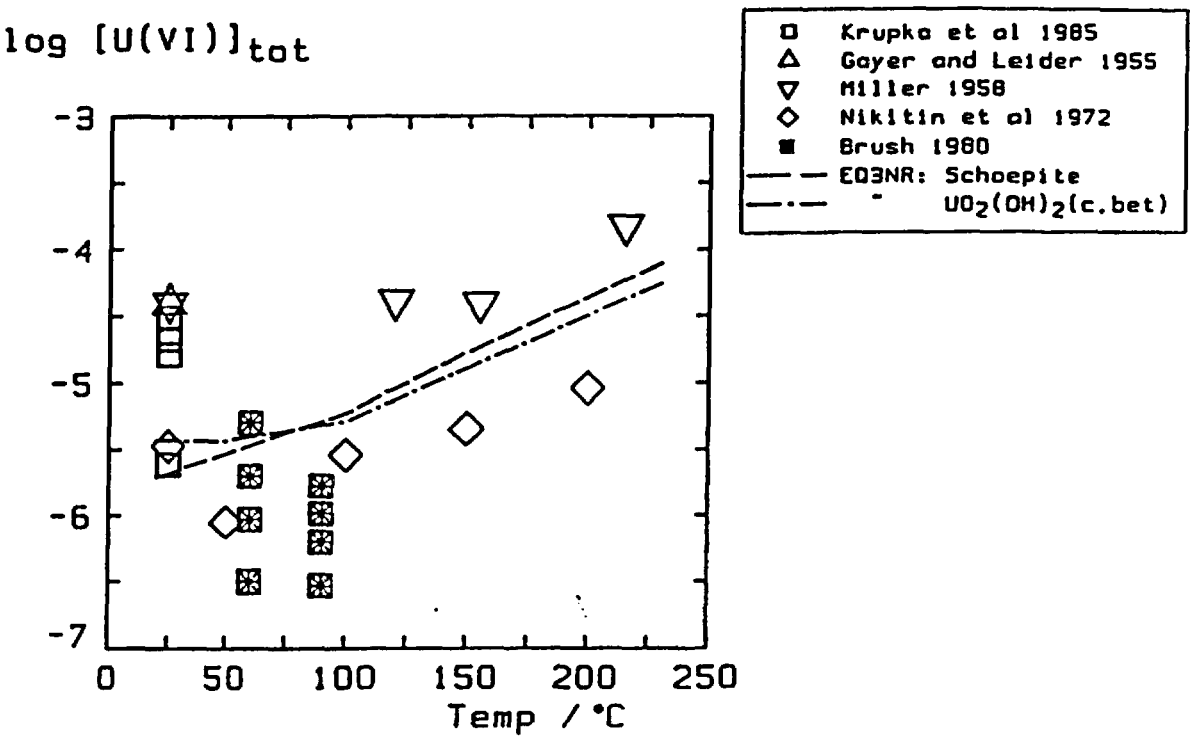


Figure 5.6 The calculated solubility of schoepite and $UO_2(OH)_2(c)$ in water as a function of T, compared with experimental values in the pH range 7 to 8.

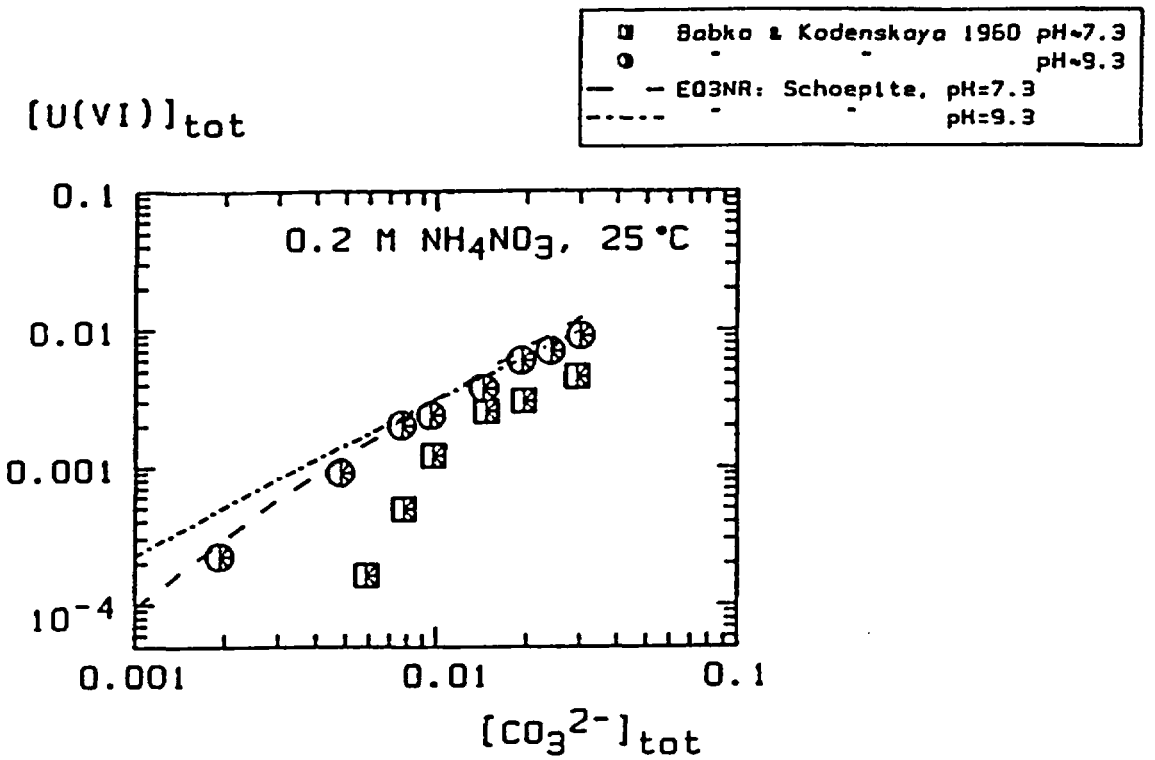


Figure 5.4 The solubility at 25°C of schoepite and as a function of carbonate concentration in 0.2 M NH_4NO_3.

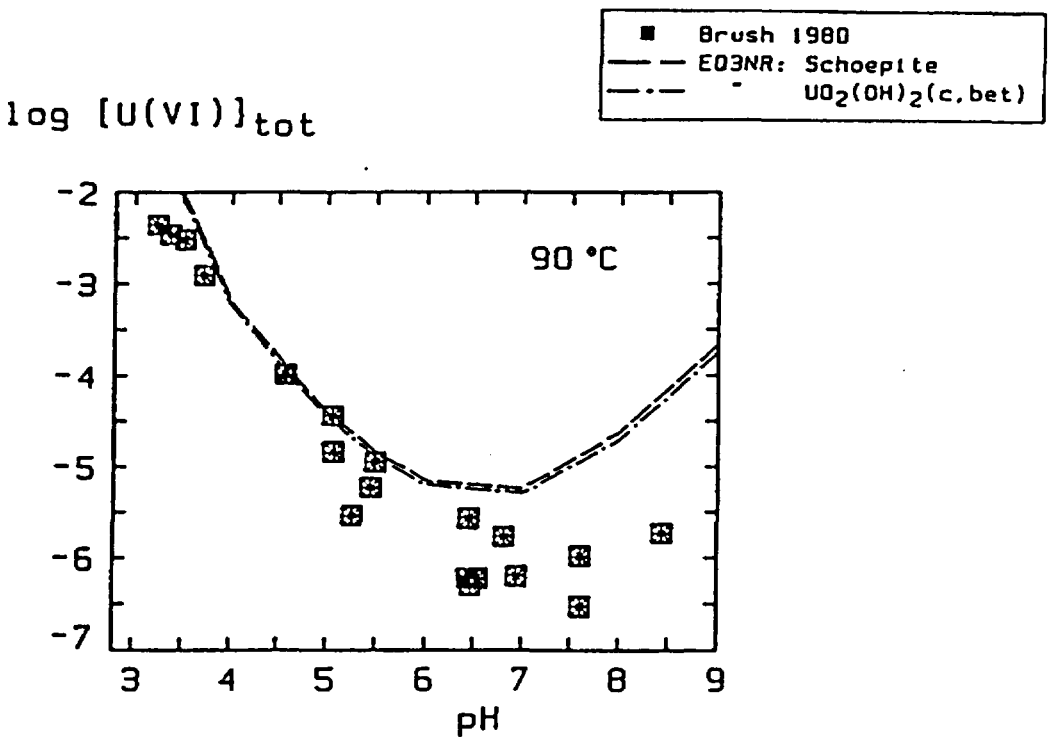


Figure 5.5 The solubility at 90°C of $UO_2(OH)_2(c)$ as a function of pH.

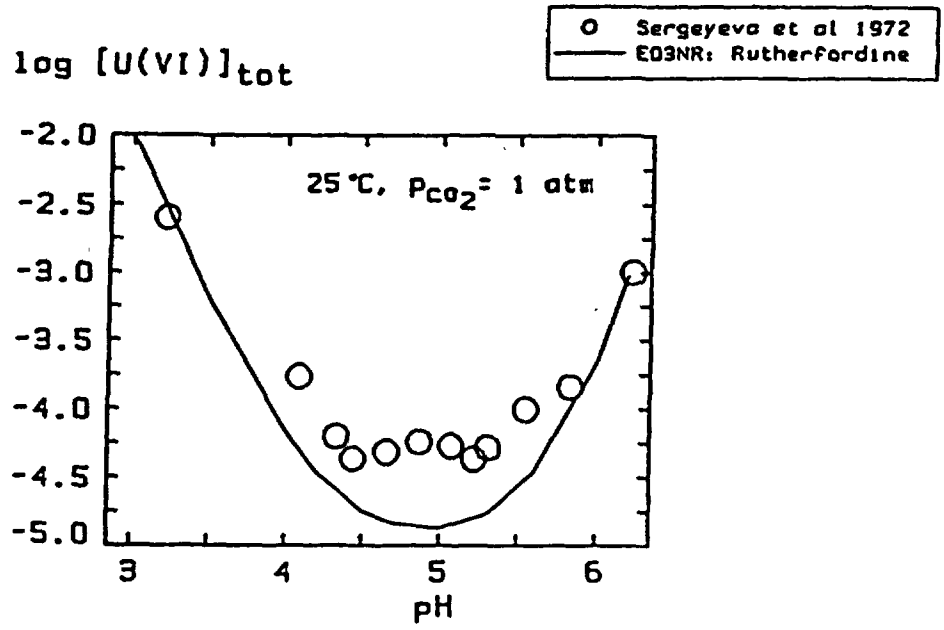


Figure 5.7 The solubility of rutherfordine ($UO_2CO_3(c)$) in dilute solutions as a function of pH at p_{CO2}=1 atm and 25°C.

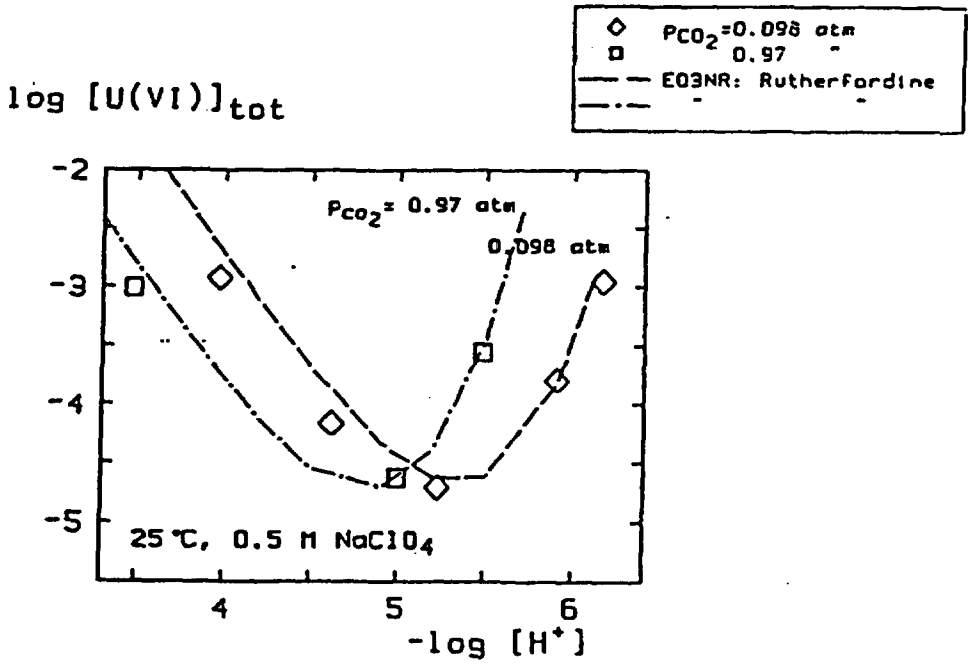


Figure 5.8 The solubility of rutherfordine ($UO_2CO_3(c)$) in 0.5 M $NaClO_4$ solutions as a function of pH at 25°C and p_{CO2}=0.97 and 0.098 atm.

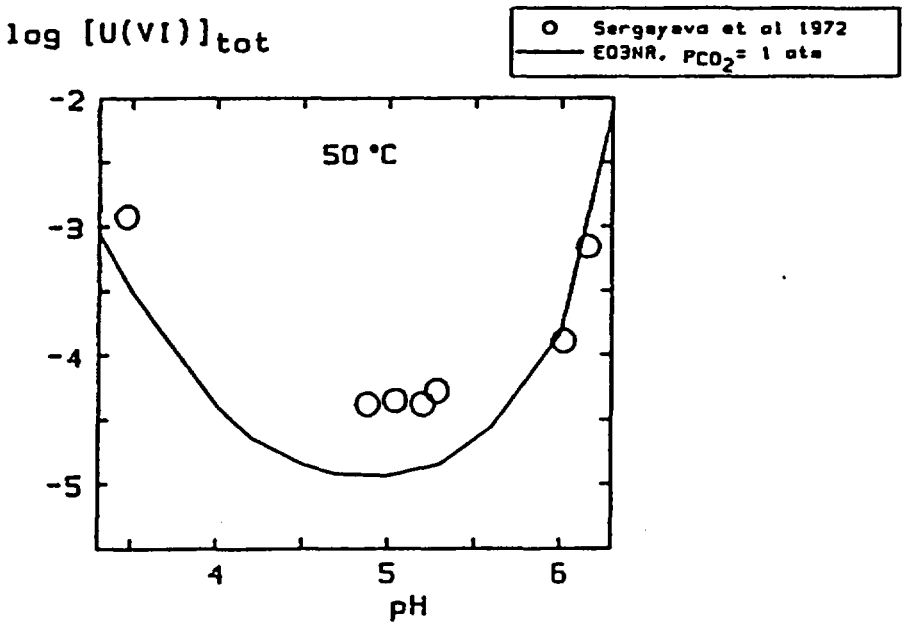


Figure 5.10 The solubility of rutherfordine ($UO_2CO_3(c)$) in dilute solutions as a function of pH at p_{CO2}=1 atm and 50°C.

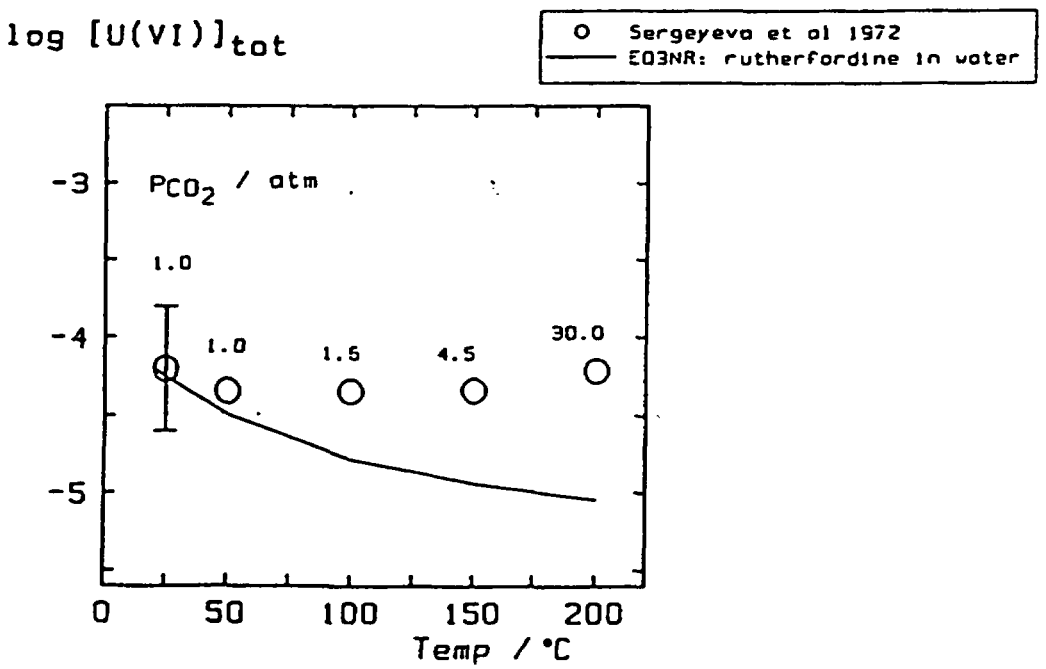
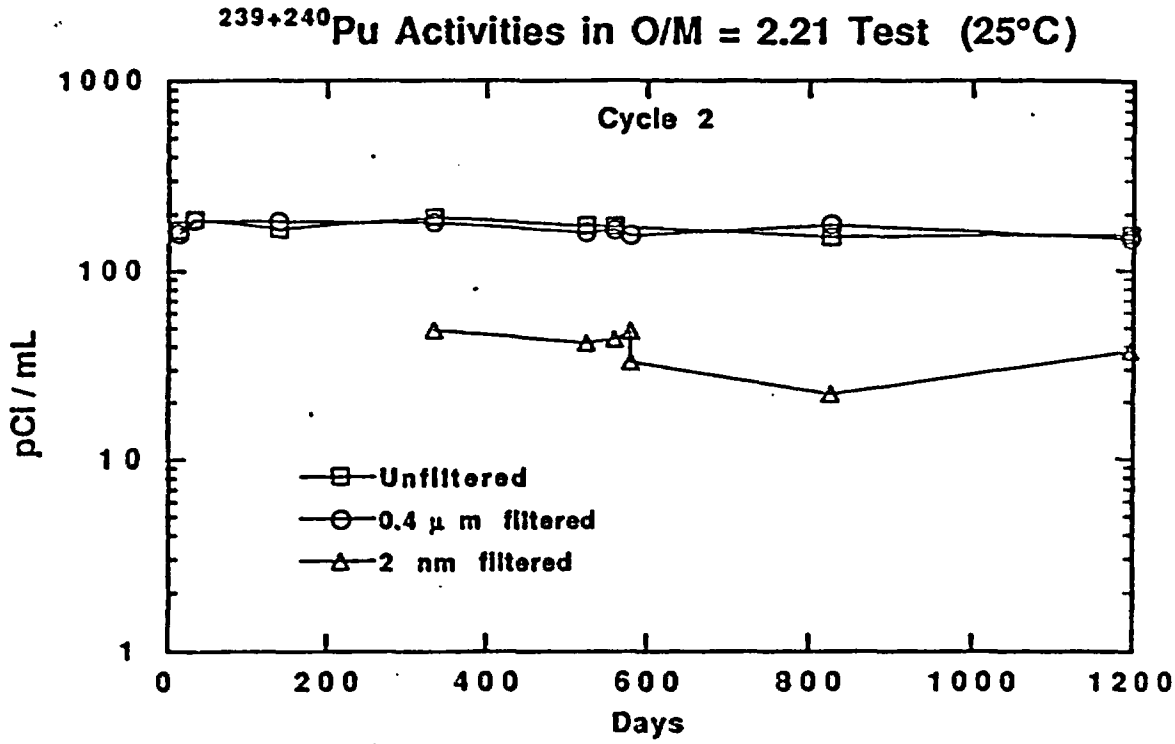


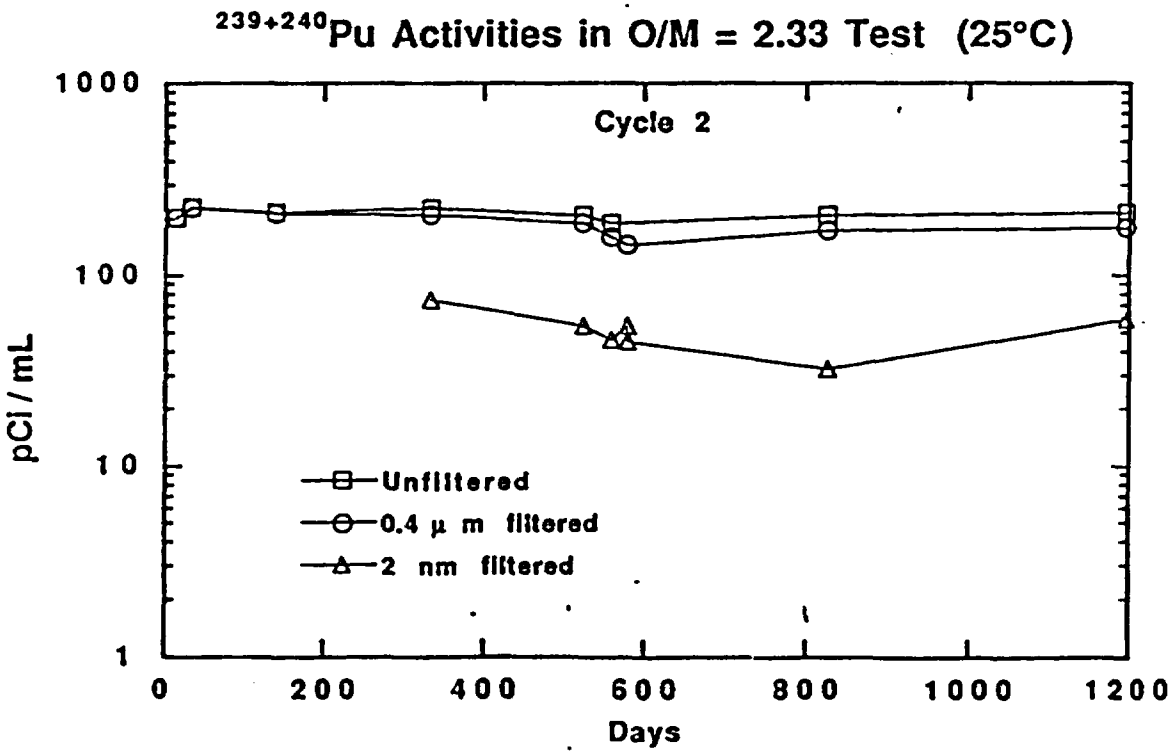
Figure 5.11 The calculated solubility of rutherfordine ($UO_2CO_3(c)$) in water as a function of temperature at the given values for the partial pressure of $CO_2(g)$ (p_{CO2}) compared with experimental results of Sergeyeva et al (1972).

Colloid Formation in Actinides

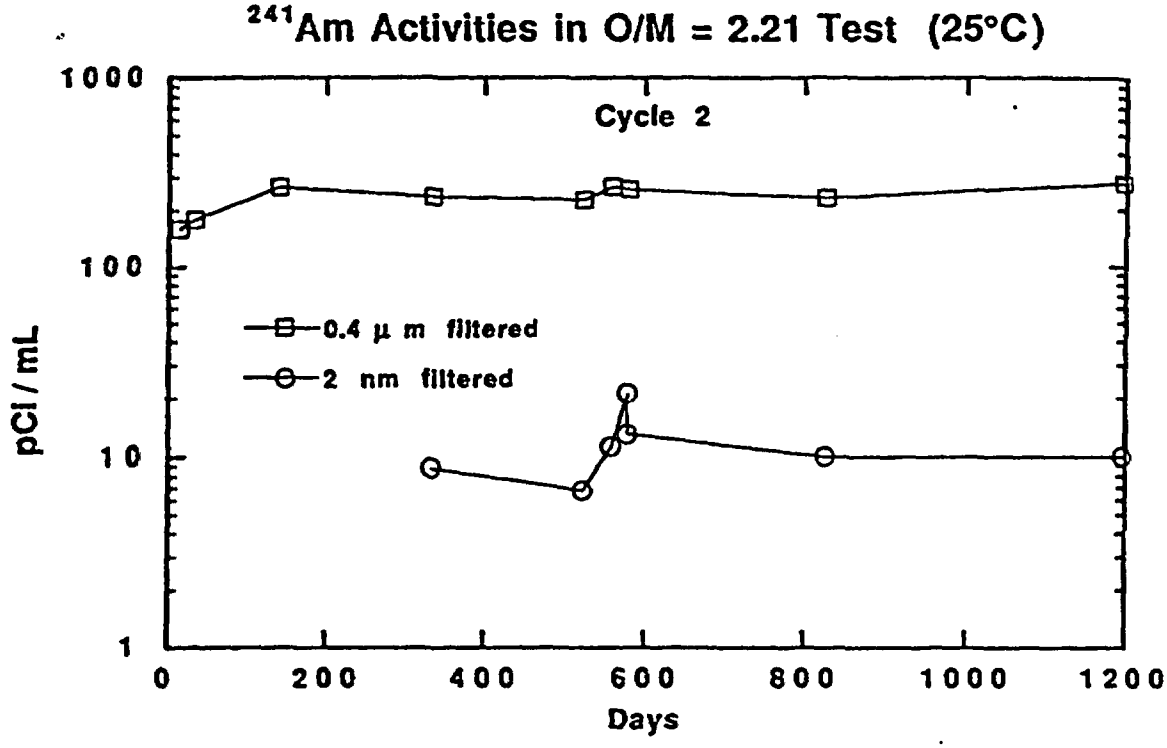
	Pu		Am		Cm		U		Np	
	25°C	85°C	25°C	85°C	25°C	85°C	25°C	85°C	25°C	85°C
Fraction in Solution	.25	.25	.05	.01	.05	.01	1	1	1	1
Fraction as Colloid	.75	.75	.95	.99	.95	.99	0	0	0	0

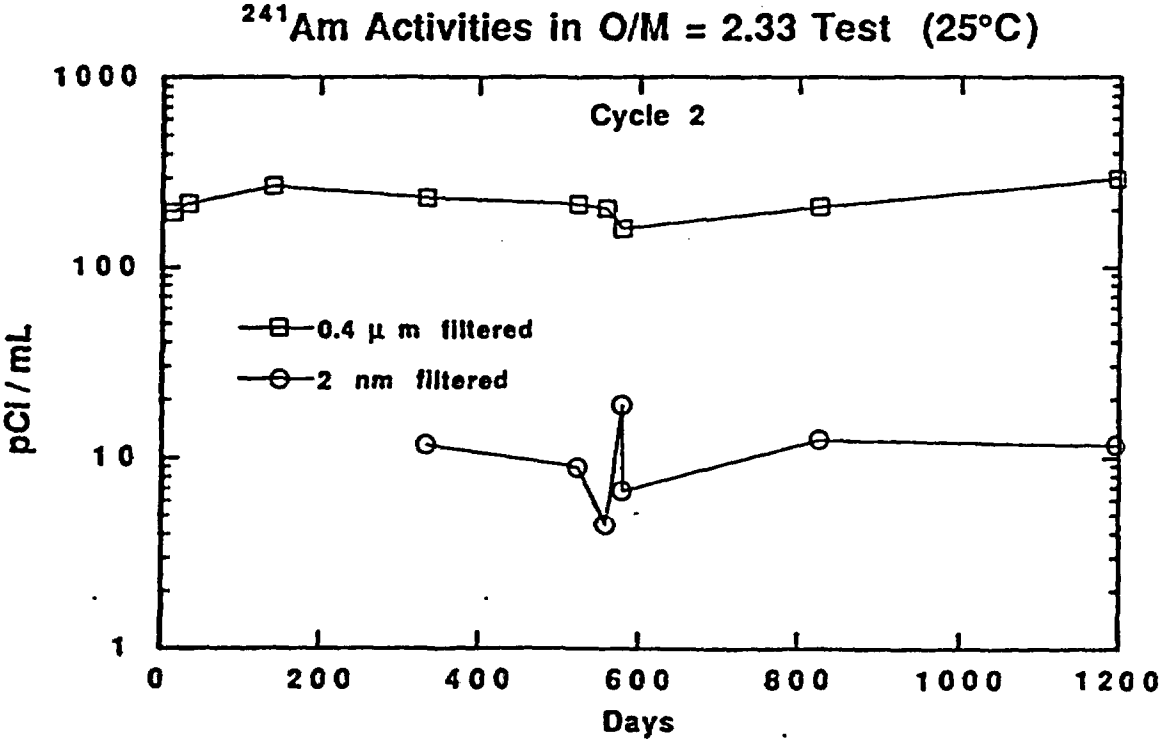
C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.



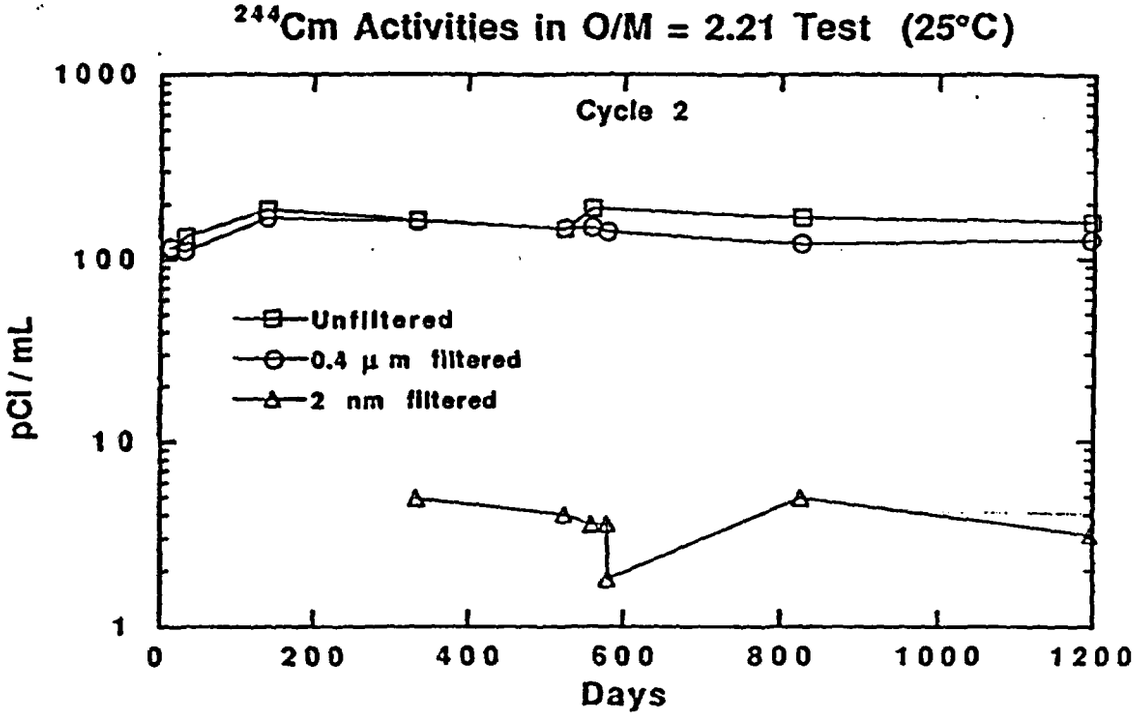


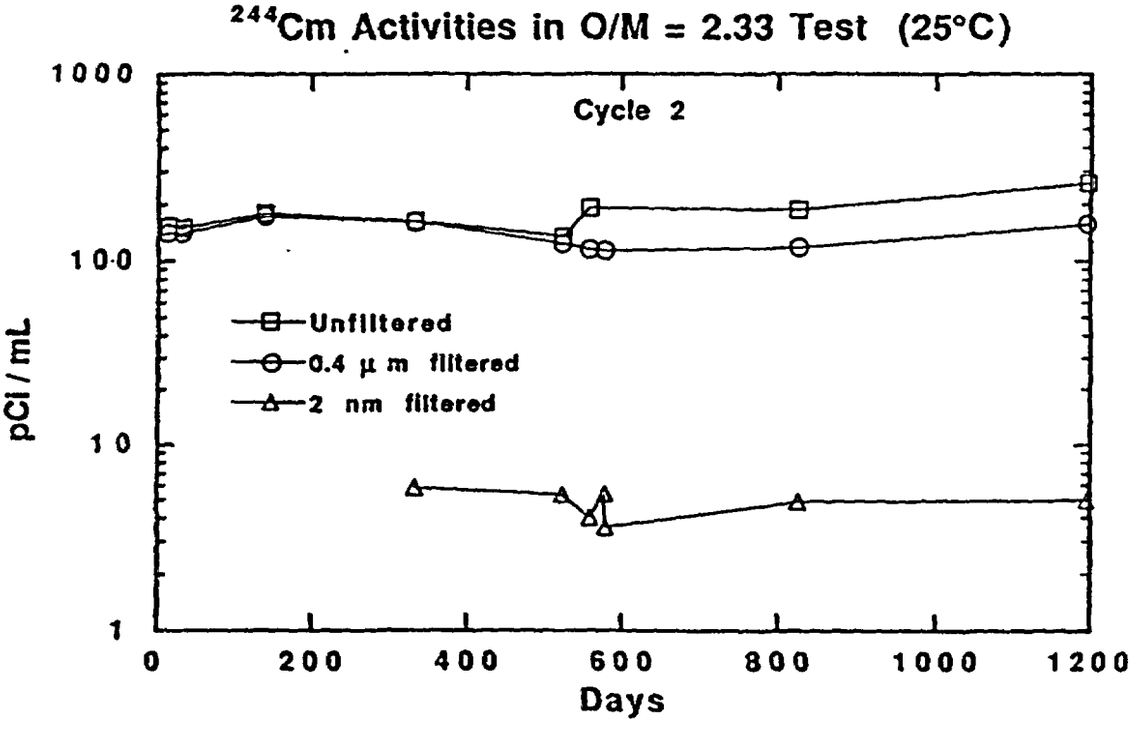
C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.



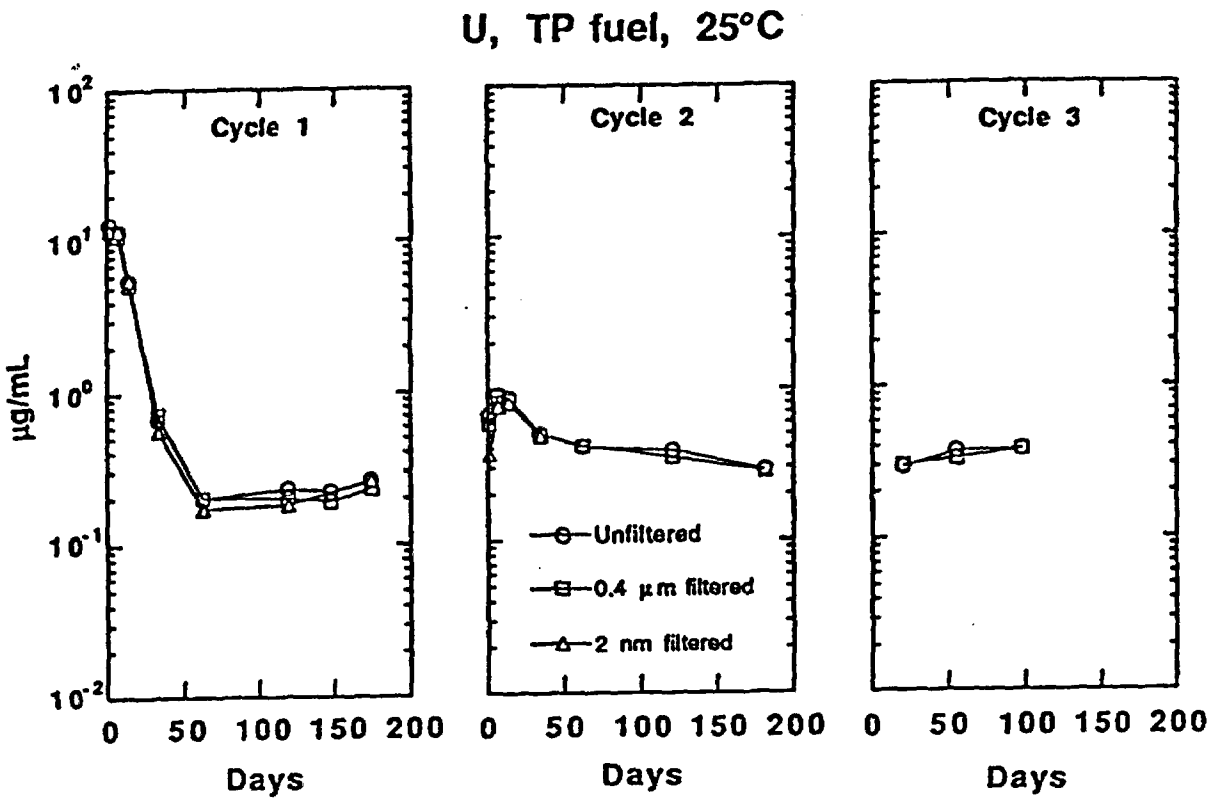


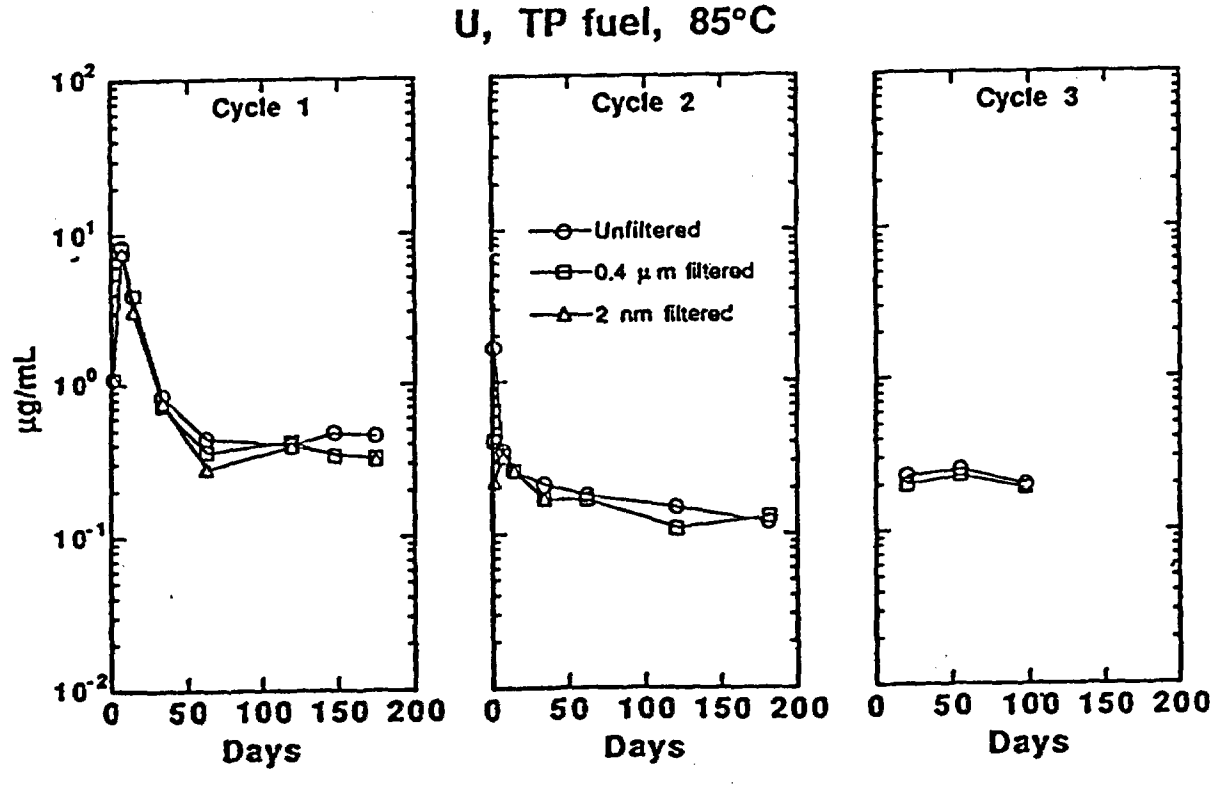
C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.



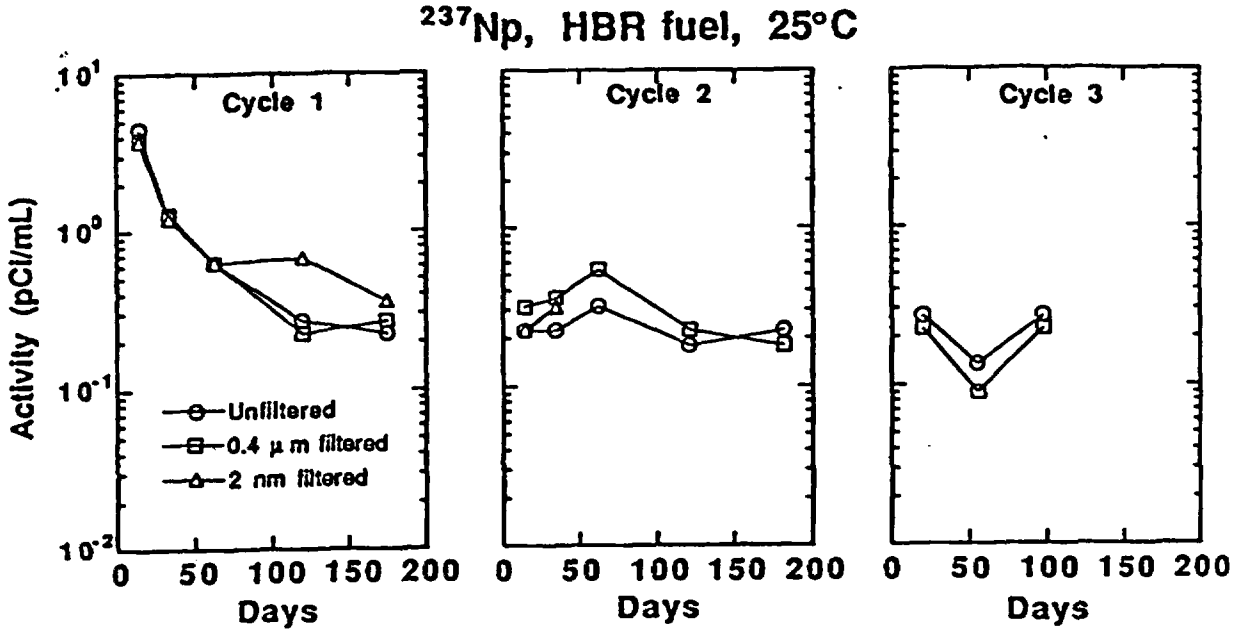


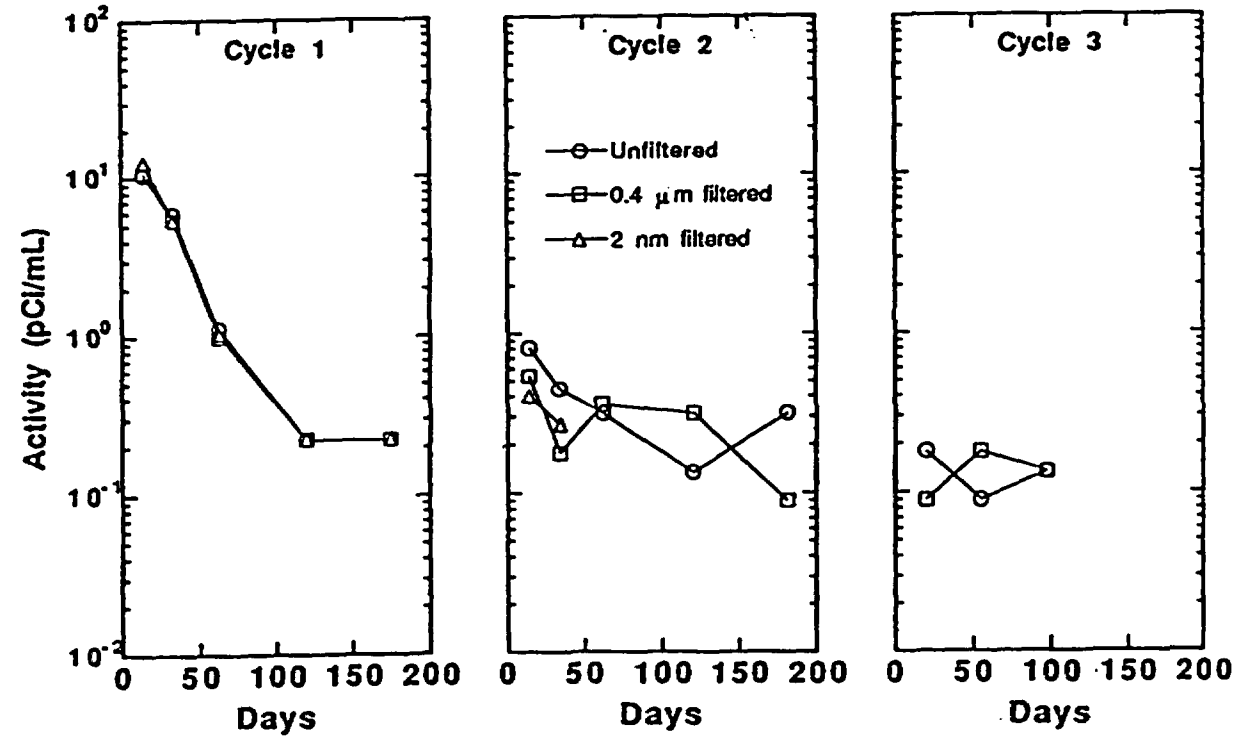
C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.



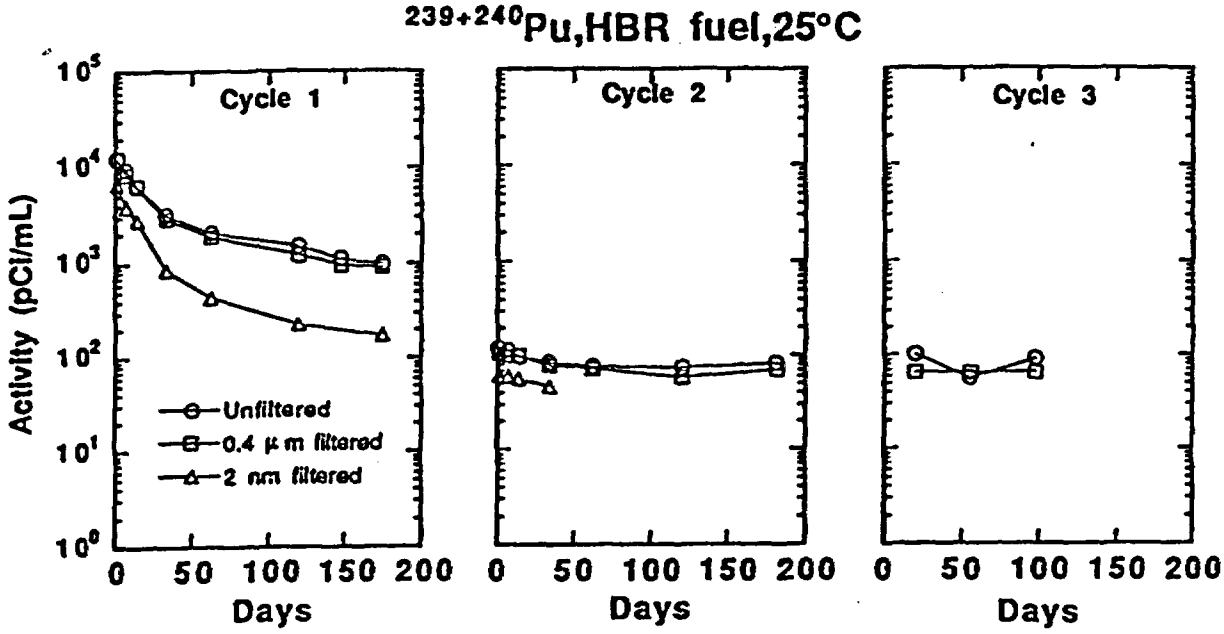


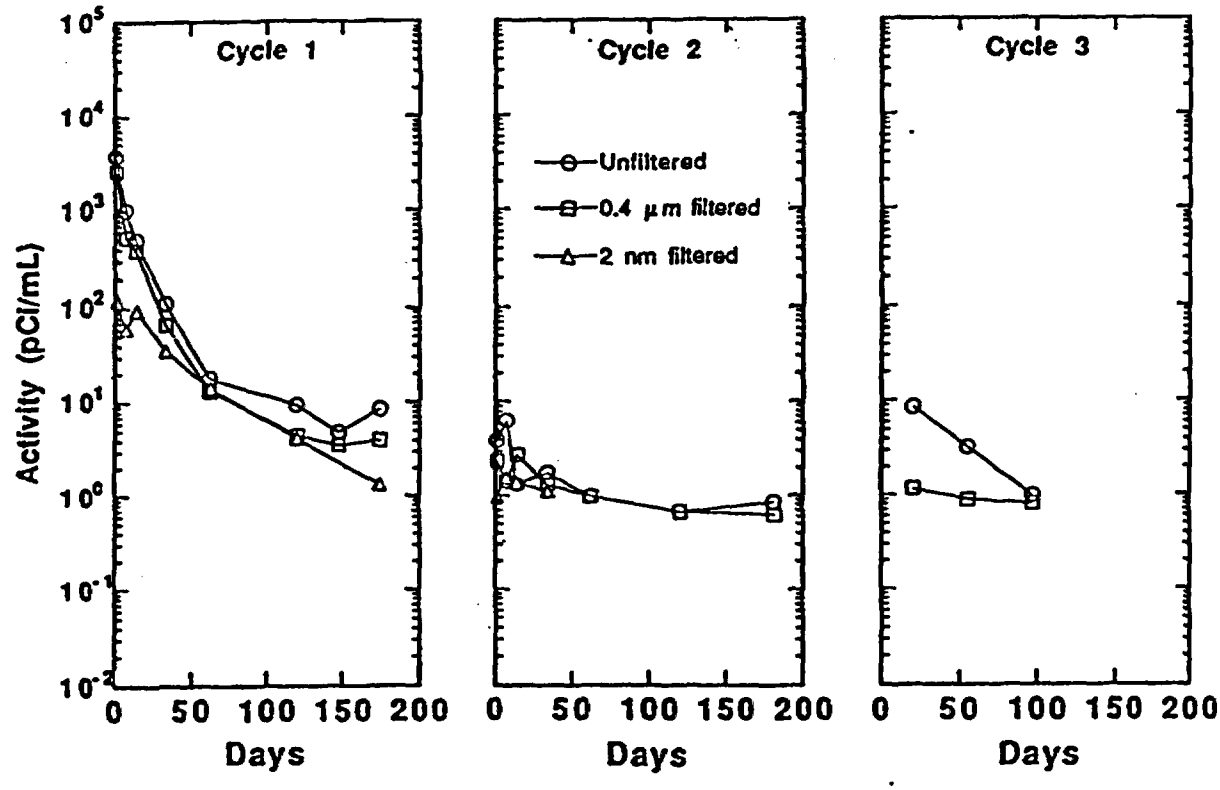
C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.



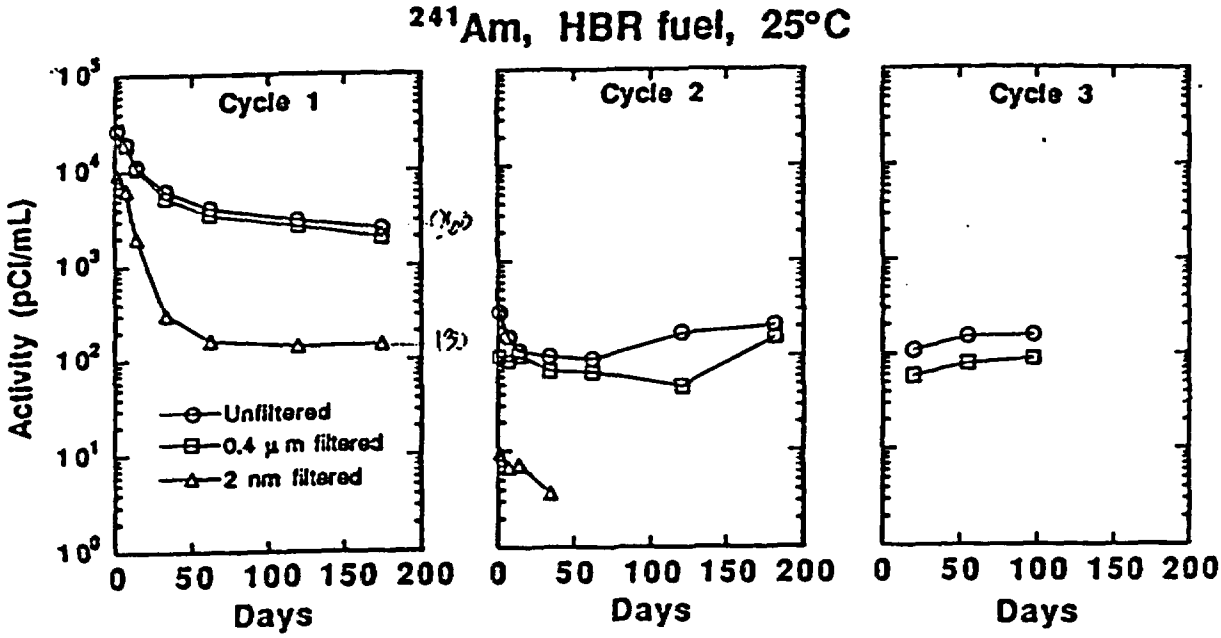


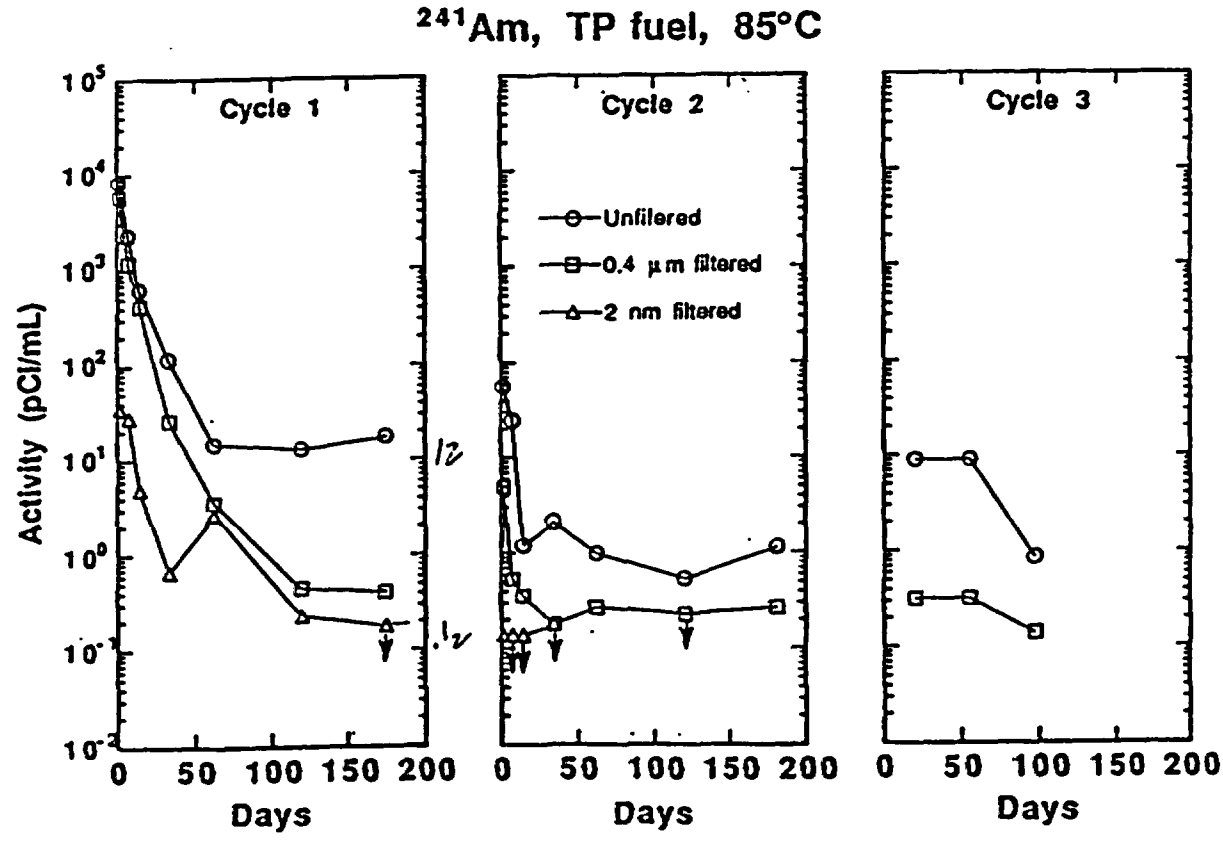
C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.



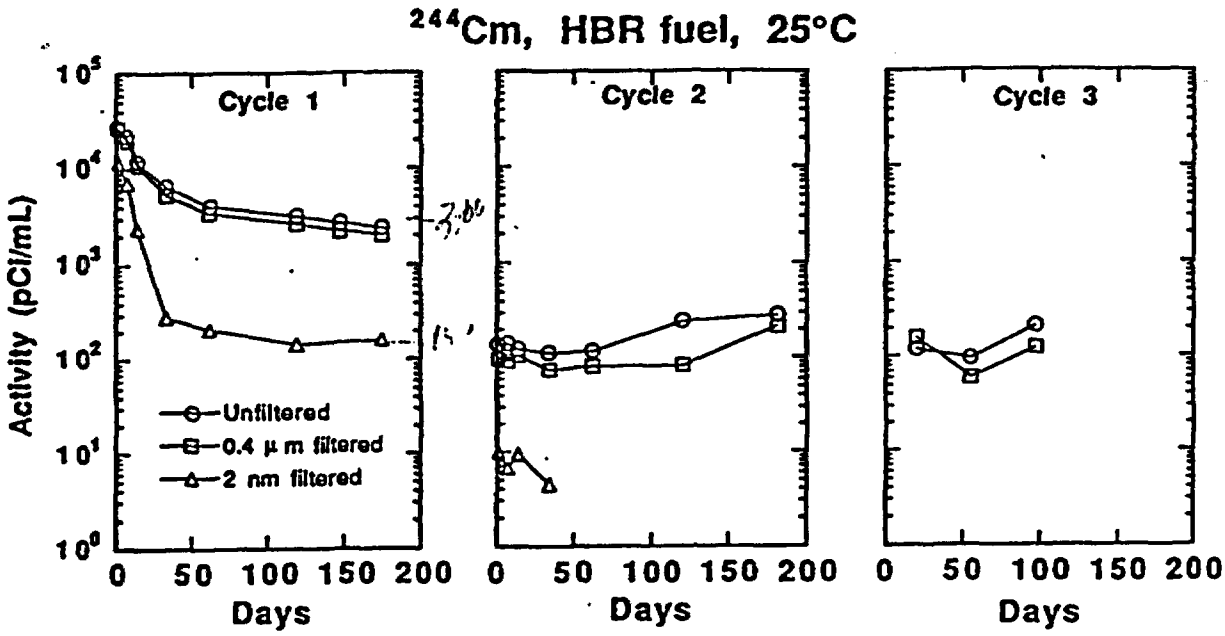


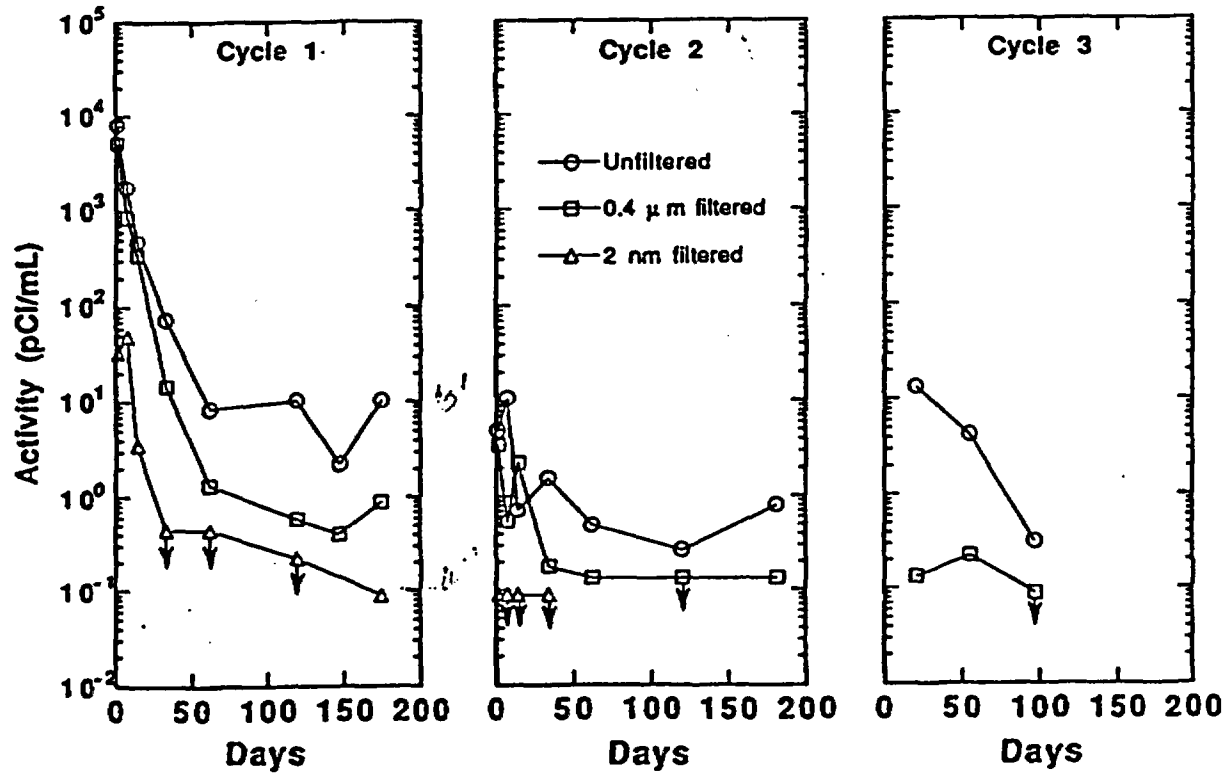
C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.





C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.





C.N. Wilson, *Indications for the Formation of Pu, Am, and Cm Colloids in Semi-Static Spent Fuel Dissolution Tests*, presented at the 1990 Spent Fuel Workshop, September, 1990.

2.1 Spent Fuel Waste Form

2.1.1 Radionuclide Content

2.1.1.1 Present Inventory

2.1.1.2 Projected Inventory

2.1.1.3 Radionuclide Activity vs. History

2.1.1.4 Decay Heat vs. Time

2.1.1.5 Fission Gas Release Distribution

2.1.2 Structural Characteristics and Dimension

2.1.2.1 Fuel Assemblies

2.1.2.2 PWR Fuel

2.1.2.3 BWR Fuel

2.1.2.4 Non-Zircaloy Clad Fuel

2.1.2.5 Hardware

2.1.3 Repository Response

2.1.3.1 Cladding Degradation

2.1.3.2 UO_2 Oxidation in Fuel

2.1.3.3 Gaseous Radionuclide Release from Cladding

2.1.3.4 Gaseous Radionuclide Release from UO_2 Fuel

2.1.3.5 Dissolution Radionuclide Release from UO_2

2.1.3.6 Soluble-Precipitated/Colloidal Species

2.1.3.7 Radionuclide Release from Hardware

Table 3.5. 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)

Isotope[a]	Time since discharge (years)					
	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.6. 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)

	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.7. 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)

Isotope[a]	Time since discharge (years)					
	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

Table 3.8. 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)

	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.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.

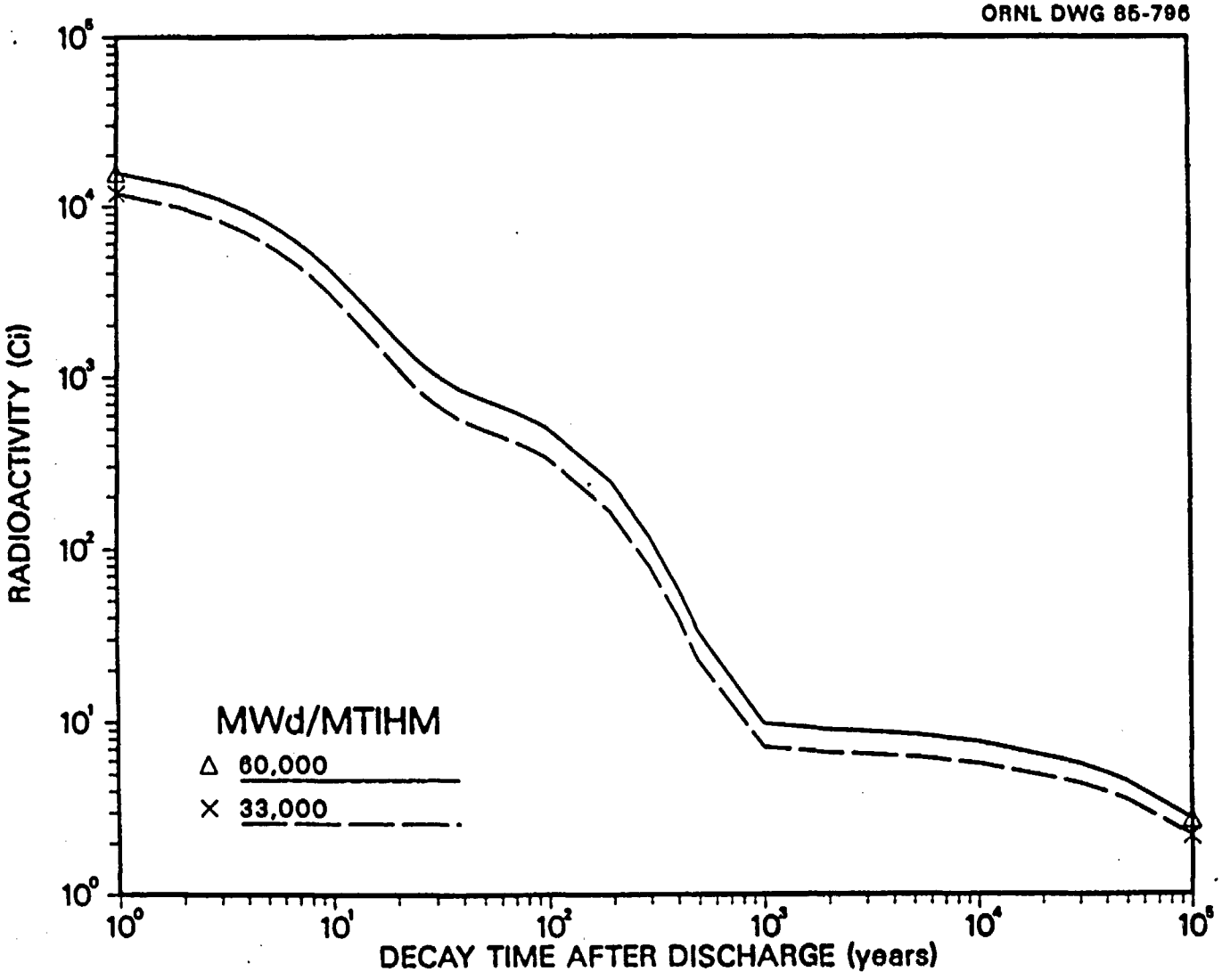


Fig. 4.1. Radioactivity from irradiated grid spacers, springs, braze, end pieces, and miscellaneous SS-304 for a PWR.

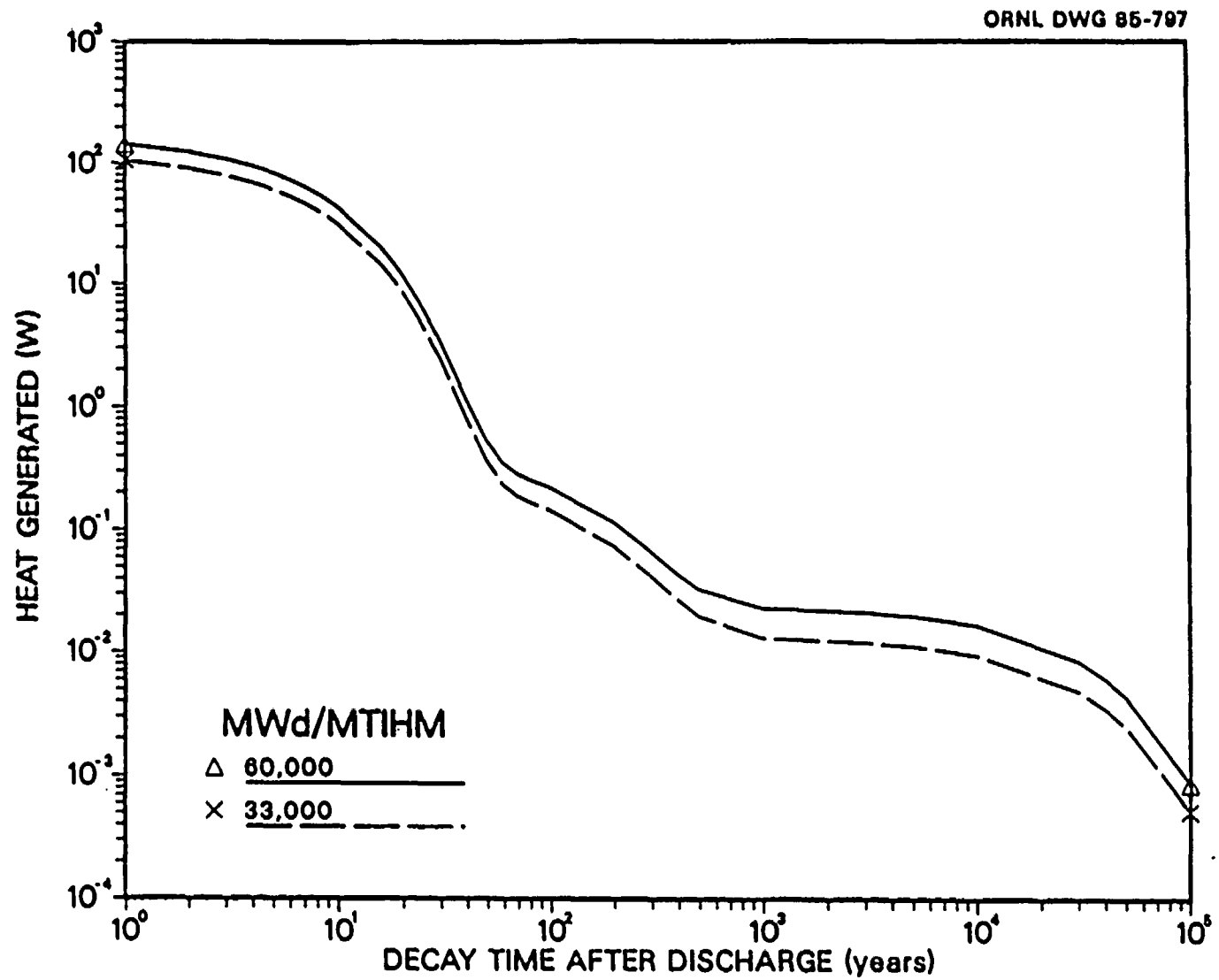


Fig. 4.2. Heat generated from irradiated grid spacers, springs, braze, end pieces, and miscellaneous SS-304 for a PWR.

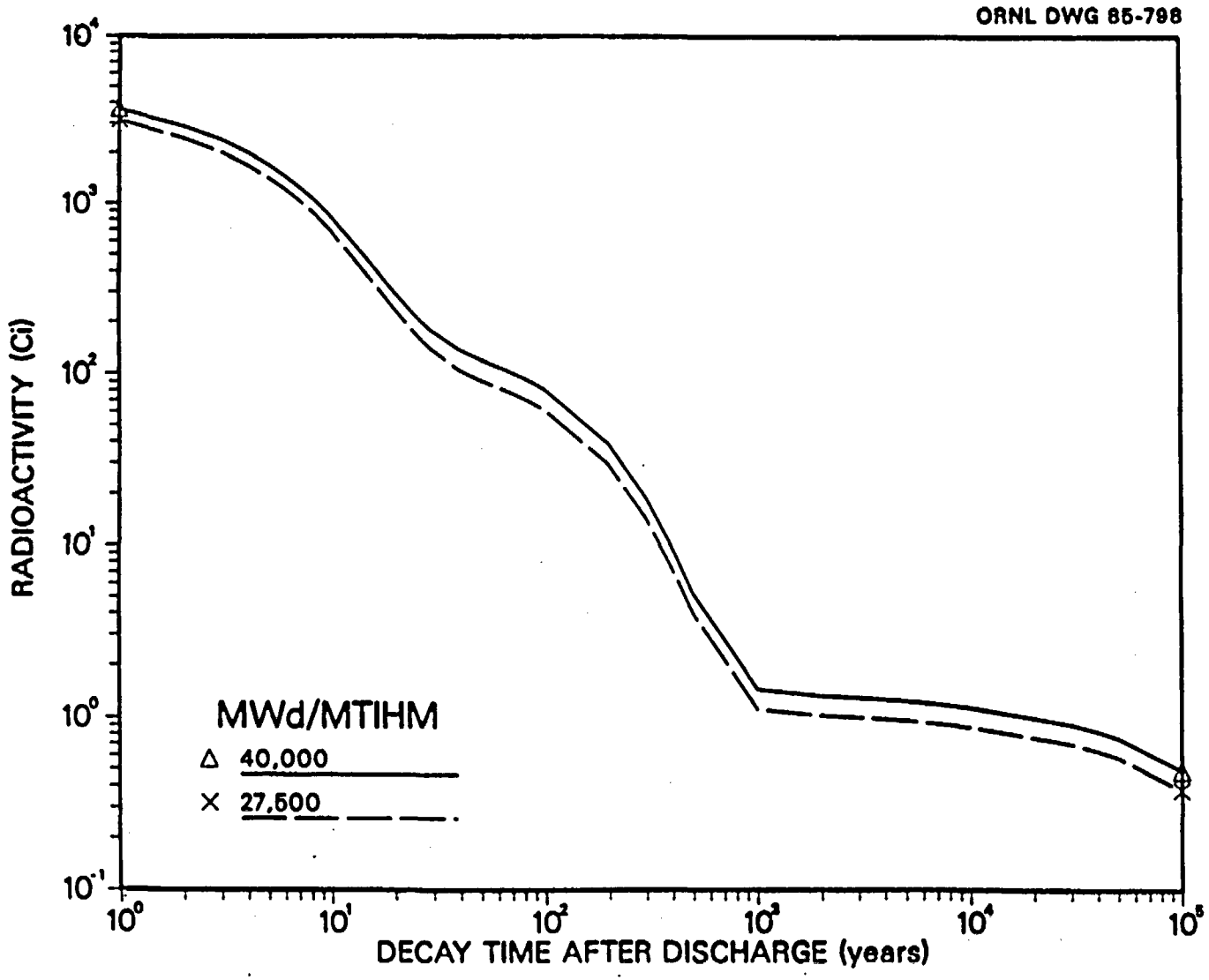


Fig. 4.3. Radioactivity from irradiated grid spacers, springs, and end pieces for a BWR.

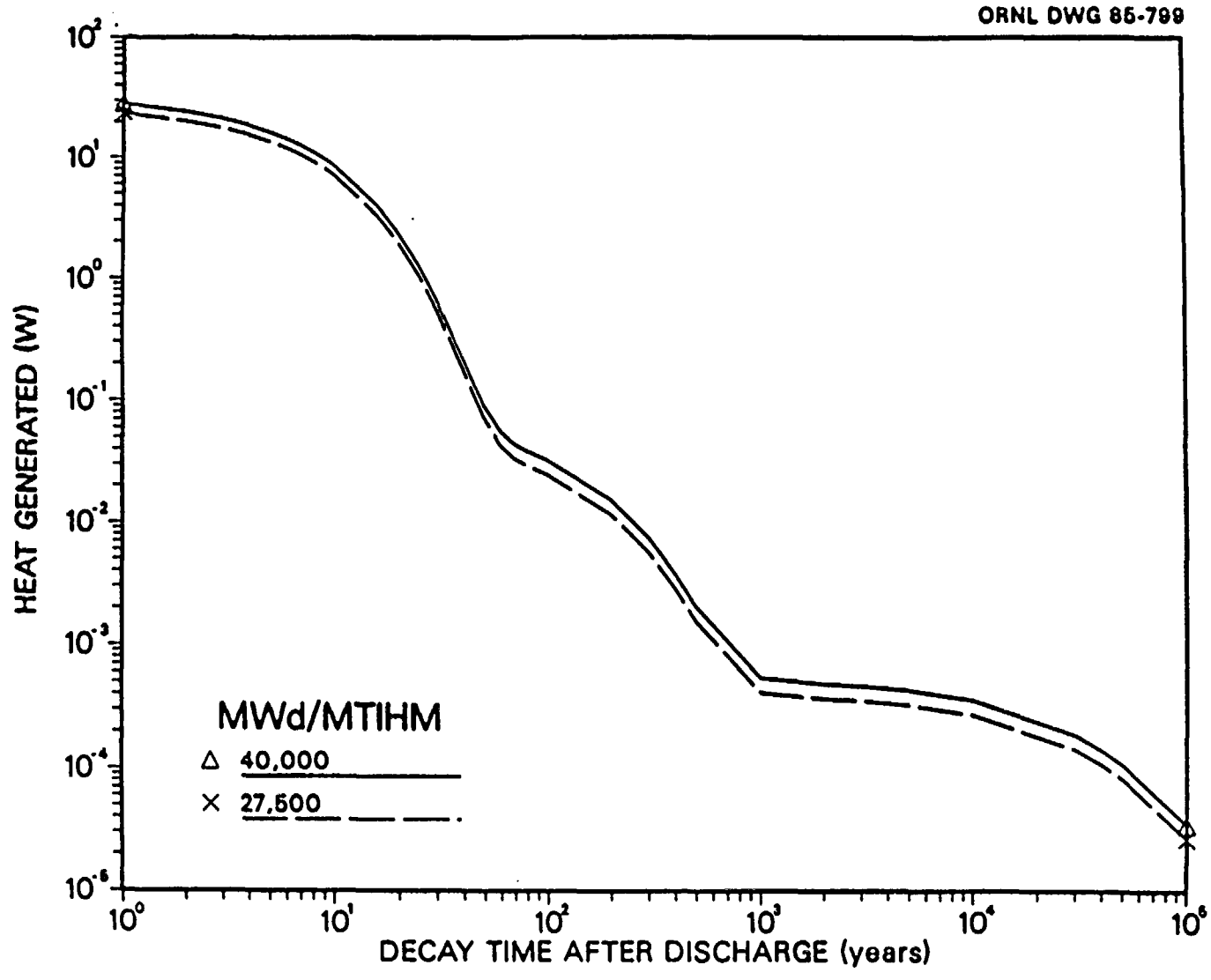


Fig. 4.4. Heat generated from irradiated grid spacers, springs, and end pieces for a BWR.

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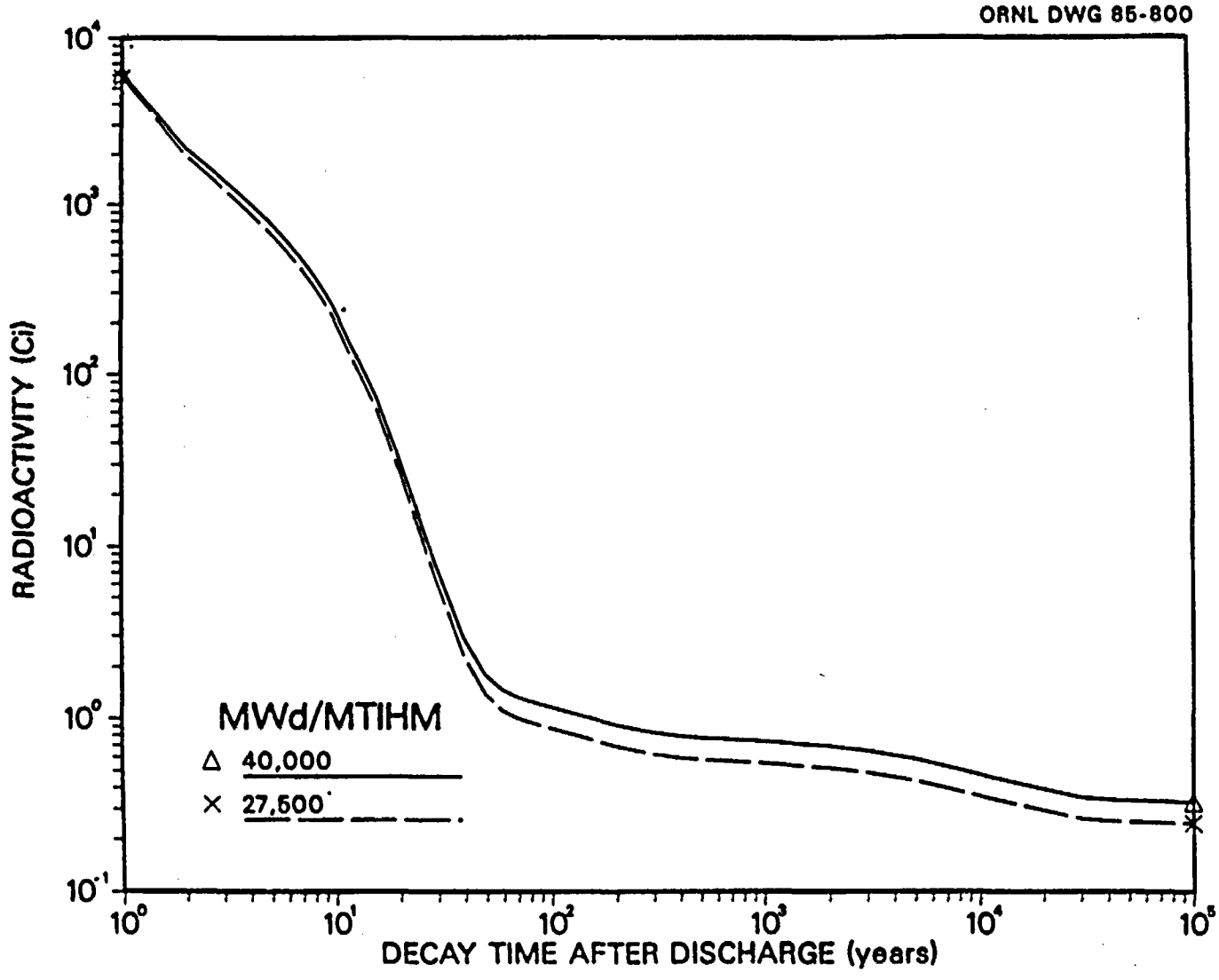

Fig. 4.5. Radioactivity from irradiated fuel channels from a BWR.

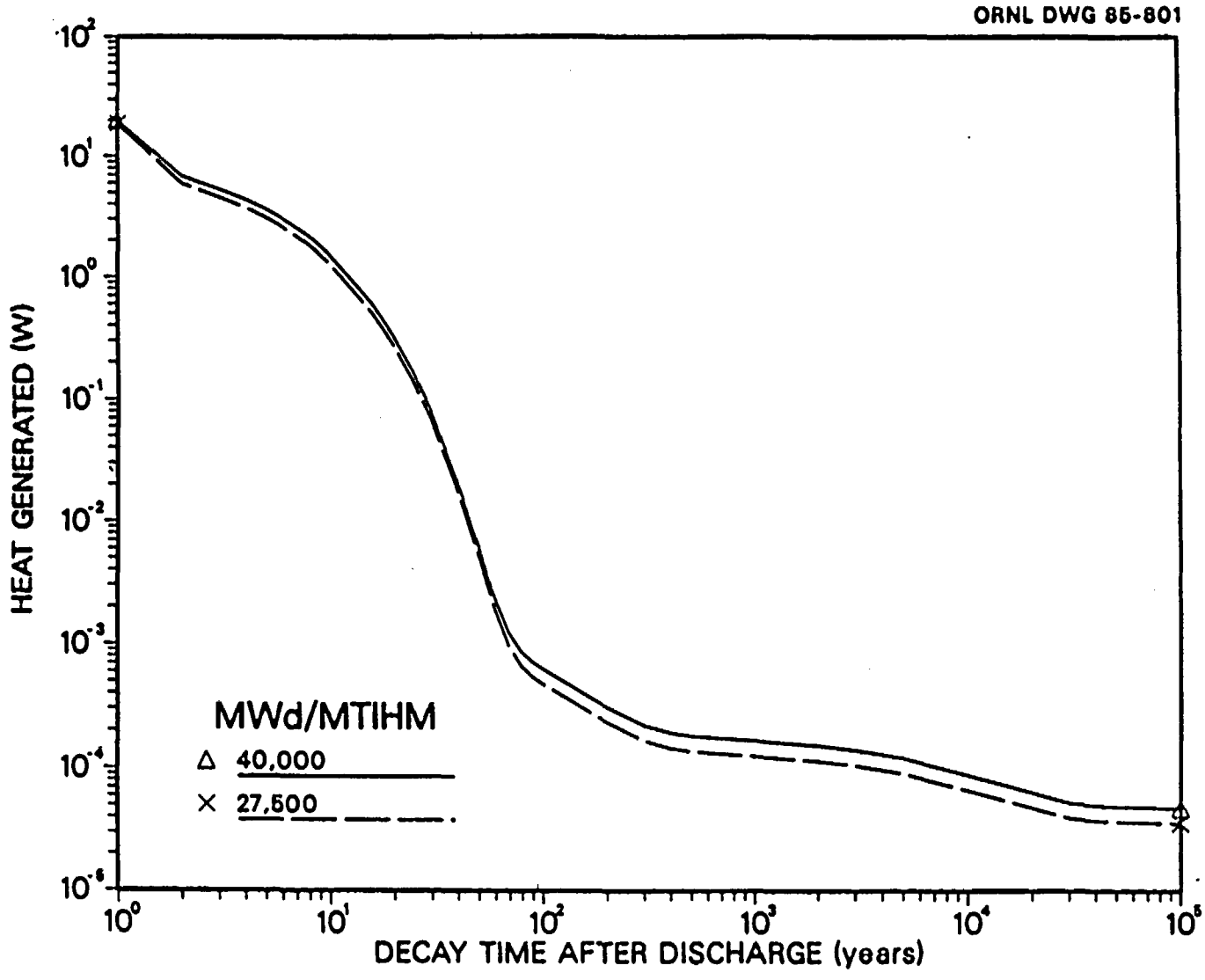

Fig. 4.6. Heat generated from irradiated fuel channels from a BWR.

W.J. Roddy, H.C. Claiborne, R.C. Ashline, P.T. Johnson, and B.T. Rhyne, *Physical and Decay Characteristics of Commercial LWR Spent Fuels*, ORNL/TM-9591/V.1, October, 1985.