12
2-14-95 821

UCRL-ID-108314 Rev. 1

Preliminary Waste Form Characteristics Report Version 1.0

Ray B. Stout
Herman Leider

Date Written: October 1991
Date Published: December 1994

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This document was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor the University of California nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately own rights. Reference herein to any specific commercial products, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or the University of California. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or the University of California, and shall not be used for advertising or product endorsement purposes.

This report has been reproduced
directly from the best available copy.

Available to DOE and DOE contractors from the
Office of Scientific and Technical Information
P.O. Box 62, Oak Ridge, TN 37831
Prices available from (615) 576-8401, FTS 626-8401

Available to the public from the
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Rd.,
Springfield, VA 22161

Preliminary

Waste Form Characteristics Report

Editors:

Ray B. Stout

Herman R. Leider

University of California/LLNL

Livermore, CA 94551

Version 1.0

October 11, 1991

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

BJ

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

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).

Ray B. Stout
Herman R. Leider

This work performed under the auspices of the U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Yucca Mountain Project, by the Lawrence Livermore National Laboratory under contract number W-7405-ENG-48.

YUCCA MOUNTAIN PROJECT
PRELIMINARY
WASTE FORM CHARACTERIZATION REPORT
VERSION 1.0

Preface.....	ii
Table of Contents	iv
1. Introduction.....	(V 1.0, 14 p)
1.1 Overview.....	1-1
1.2 Technical Objectives.....	1-1
1.3 Quality Objectives.....	1-3
1.4 Types of Waste Forms.....	1-4
1.5 Spent Fuel Waste Forms.....	1-4
1.6 Physical Inventory.....	1-5
1.7 Radionuclides.....	1-6
1.8 Decay Heat and Criticality	1-7
1.9 Radiation Field.....	1-10
1.10 Hardware	1-13
1.11 Modeling.....	1-13
1.12 Burnup Models.....	1-14
1.13 Glass Modeling Status.....	1-14
2. Design Data for Waste Forms.....	(V 1.0, 1 p)
2.1 Spent Fuel Waste Form.....	(V 1.0, 0 p)
2.1.1 Radionuclide Content.....	(V 1.0, 1 p)
2.1.1.1 Present Inventory	(V 1.0, 24 p)
2.1.1.2 Projected Inventory.....	(V 1.0, 19 p)
2.1.1.3 Radionuclide Activity vs. History.....	(V 1.0, 17 p)
2.1.1.4 Decay Heat vs. Time.....	(V 1.0, 17 p)
2.1.1.5 Fission Gas Release Distribution.....	(V 1.0, 18 p)
2.1.2 Structural Characteristics and Dimension.....	(V 1.0, 1 p)
2.1.2.1 Fuel Assemblies	(V 1.0, 15 p)
2.1.2.2 PWR Fuel	(V 1.0, 9 p)
2.1.2.3 BWR Fuel.....	(V 1.0, 8 p)
2.1.2.4 Non-Zircaloy Clad Fuel.....	(V 1.0, 1 p)
2.1.2.5 Hardware	(V 1.0, 13 p)
2.1.3 Repository Response.....	(V 1.0, 0 p)
2.1.3.1 Cladding Degradation	(V 1.0, 5 p)
2.1.3.2 UO ₂ Oxidation in Fuel.....	(V 1.0, 2 p)
2.1.3.3 Gaseous Radionuclide Release from Cladding.....	(V 1.0, 2 p)
2.1.3.4 Gaseous Radionuclide Release from UO ₂ Fuel.....	(V 1.0, 5 p)
2.1.3.5 Dissolution Radionuclide Release from UO ₂	(V 1.0, 31 p)
2.1.3.6 Soluble-Precipitated/Colloidal Species	(V 1.0, 25 p)
2.1.3.7 Radionuclide Release from Hardware.....	(V 1.0, 26 p)

2.2	Glass Waste Form.....	(V 1.0, 1 p)
2.2.1	Radionuclide Content.....	(V 1.0, 0 p)
2.2.1.1	Present Inventory.....	(V 1.0, 6 p)
2.2.1.2	Projected Inventory.....	(V 1.0, 8 p)
2.2.1.3	Radioactivity and Decay Heat vs. Time.....	(V 1.0, 9 p)
2.2.1.4	Glass Species Composition Statistics.....	(V 1.0, 17 p)
2.2.1.5	Fracture/Fragmentation Statistics.....	(V 1.0, 3 p)
2.2.2	Repository Response.....	(V 1.0, 0 p)
2.2.2.1	Gaseous Release from Glass.....	(V 1.0, 1 p)
2.2.2.2	Dissolution Radionuclide Release from Glass.....	(V 1.0, 8 p)
2.2.2.3	Soluble-Precipitated/Colloidal Species.....	(V 1.0, 7 p)
2.3	Special Cases Waste Forms.....	(V 1.0, 1 p)
2.3.1	Damaged Spent Fuel.....	(V 1.0, 2 p)
2.3.2	Non-LWR Spent Fuel.....	(V 1.0, 9 p)
3.	Scientific Basis for Predictive Model Development.....	(V 1.0, 0 p)
3.1	Spent Fuel Cladding Failure.....	(V 1.0, 5 p)
3.1.1	Experimental Parameters for Failure Models.....	(V 1.0, 7 p)
3.1.2	Failure Models.....	(V 1.0, 1 p)
3.2	Spent Fuel Oxidation.....	(V 1.0, 3 p)
3.2.1	Experimental Parameters for Oxidation Models.....	(V 1.0, 21 p)
3.2.2	Oxidation Models.....	(V 1.0, 18 p)
3.3	Spent Fuel Fission Gas Release.....	(V 1.0, 0 p)
3.3.1	Experimental Parameters for Fission Gas Release.....	(V 1.0, 6 p)
3.3.2	Release Models.....	(V 1.0, 3 p)
3.4	Spent Fuel Dissolution.....	(V 1.0, 2 p)
3.4.1	Experimental Parameters for Dissolution.....	(V 1.0, 3 p)
3.4.1.1	Dissolution Rates.....	(V 1.0, 6 p)
3.4.1.2	Solubility Limits.....	(V 1.0, 2 p)
3.4.1.3	Solubility Limiting Phases.....	(V 1.0, 4 p)
3.4.2	Dissolution Models.....	(V 1.0, 5 p)
3.5	Glass Dissolution.....	(V 1.0, 1 p)
3.5.1	Experimental Parameters for Glass Dissolution.....	(V 1.0, 3 p)
3.5.2	Glass Dissolution Models.....	(V 1.0, 9 p)
3.6	Other Release Sources of Radionuclides.....	(V 1.0, 0 p)
3.6.1	Crud.....	(V 1.0, 2 p)
3.6.2	Hardware.....	(V 1.0, 1 p)
3.6.3	Cladding.....	(V 1.0, 3 p)
	Bibliography.....	(V 1.0, 4 p)
	Appendix A. Physical Properties of Materials.....	(to be provided)

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 ^{60}Co 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

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.

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

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.

Figure 1. Quantities of Domestic LWR Spent Fuel

K.J. Noz, 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

1988 EIA Data

Figure 3.2 Fuel Rods by Assembly Class

1988 EIA Data

(suollIW)
 Rods Fuel

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)	Assemblies	BWR Metric Tons	Assemblies	PWR 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

Figure 3.3 BWR Discharges by Year

1988 EIA DATA

Figure 3.4 PWR Discharges by Year

1988 EIA DATA

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

Table 4.1 Summary of the Quantities of LWR Spent Fuel.

		Historical or Projected					
		Through 1988			Projected		
		Reactor Type		Summary	Reactor Type		Summary
		BWR	PWR		BWR	PWR	
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		Summary
		BWR	PWR	
No. of Assemblies	Total	160946	127492	288438
Mass, Mtihm	Total	28841	55067	83908

Table 43 Summary LWR Spent Fuel Burnup, Enrichment, and Age.

		Historical or Projected					
		Through 1988			Projected		
		Reactor Type		Summary	Reactor Type		Summary
		BWR	PWR		BWR	PWR	
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

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

		Summary		
		Reactor Type		Summary
		BWR	PWR	
Burnup, Mwd/Mt	Minimum	0	3000	0
	MEAN	30172	39410	36234
	Maximum	47000	67000	67000
	Standard Deviation	32110	31340	34685
Enrichm- ent	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
Dischar- ge 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
Fl. 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 Bumup 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.

HEDL 8303-271.10

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.

HEDL 8303-271.7

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.

HEDL 8303-271.12

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		1%	5%	10%	25%	Median 50%		Mode	75%	90%	95%	99%	Maximum 100%	
		0%														
BWR	Historical	0.0		0.7	1.8	2.1	2.2	2.5	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	3.2	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

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	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 ELA 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

Figure 3.1 Assemblies by Assembly Class

1988 EIA Data

Figure 3.2 Fuel Rods by Assembly Class

1988 EIA Data

(suoiliv)
 Rod Fuel of requen

Figure 3.3 BWR Discharges by Year

1988 EIA DATA

Figure 3.4 PWR Discharges by Year

1988 EIA DATA

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

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

		Historical or Projected					
		Through 1988			Projected		
		Reactor Type		Summary	Reactor Type		Summary
		BWR	PWR		BWR	PWR	
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

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
Enrichm- ent	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
Dischar- ge Date	Minimum	1968	1970	1968
	MEAN	2001	2002	2002
	Maximum	2029	2036	2036

HEDL 8303-271.10

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 $\geq 1\%$ 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 $\geq 1\%$ 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 $\geq 1\%$ 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
Pu147	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 $\geq 1\%$ of total) for PWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 40,000 MWd/MTIHM.

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

TABLE 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 ²³⁵U. (Adapted from Reference 3.)

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

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

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

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

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)

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	-	-	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

^aNuclides contributing >0.1% are listed.

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

Table 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)

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	-	-	2.66E-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

^aNuclides contributing >0.1% are listed.

^bThe 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

^aNuclides contributing >0.1% are listed.

^bThe 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)

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	-	-	2.32E-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

^aNuclides contributing >0.1% are listed.

^bThe 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 $\geq 1\%$ 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 $\geq 1\%$ of total) for BWR Spent Fuel as a function of Initial Enrichment and Decay Time for a burnup of 30,000 MWd/MTIHM.

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

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

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

Table 4.20

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

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

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

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

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

^aNuclides contributing >0.1% of total are listed.

^bOnly activation products contribute to this nuclide.

^cBoth activation and fission products contribute to this nuclide.

^dA.P. = Activation products.

^eF.P. = Fission products.

^fA.+D. = Actinides plus daughters.

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

^aNuclides contributing >0.1% of total are listed.

^bOnly activation products contribute to this nuclide.

^cBoth activation and fission products contribute to this nuclide.

^dA.P. = Activation products.

^eF.P. = Fission products.

^fA.+D. = Actinides plus daughters.

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)

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	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-110 ^m	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-137 ^m	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

^aNuclides contributing >0.1% of total are listed.

^bOnly activation products contribute to this nuclide.

^cBoth activation and fission products contribute to this nuclide.

^dA.P. = Activation products.

^eF.P. = Fission products.

^fA.+D. = Actinides plus daughters.

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

^aNuclides contributing >0.1% of total are listed.

^bOnly activation products contribute to this nuclide.

^cBoth activation and fission products contribute to this nuclide.

^dA.P. = Activation products.

^eF.P. = Fission products.

^fA.+D. = Actinides plus daughters.

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.

HEDL 8303-271.2

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.)

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

- 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 5.1. 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.

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.

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. (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.

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. (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 resistant 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.

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. (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

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

Table 5.3. Listing of Assembly Types by Assembly Class.

<u>Assembly Class</u> Assembly Type	Rodarray Code	Assemblies In Storage	Status	Comments
<u>BABCOCK & WILCOX 15 X 15</u>				
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.
<u>BABCOCK & WILCOX 17 X 17</u>				
B&W 17 X 17 B&W Mark C	B1717B	4	Discharged	Lead Assembly
<u>COMBUSTION ENGINEERING 14 X 14</u>				
CE 14 X 14 CE	C1414C	2810	Incore	
CE 14 X 14 ANF	C1414A	323	Incore	
CE 14 X 14 WE	C1414W	189	Incore	
<u>COMBUSTION ENGINEERING 16 X 16</u>				
CE 16 X 16 CE San Onofre	C1616C	1043	Incore	
CE 16 X 16 CE ANO2	C1616C		Incore	
<u>COMBUSTION ENGINEERING 16 X 16 SYSTEM 80</u>				
CE 16 X 16 CE System 80	C1616CS8	188	Incore	
<u>GENERAL ELECTRIC BWR/2,3</u>				
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	

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

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

<u>Class</u> <u>Type</u>	<u>Rodarray</u> <u>Code</u>	<u>Assemblies</u> <u>In Storage</u>	<u>Status</u>	<u>Comments</u>
<u>L ELECTRIC BWR/4,5,6</u>				
J/4-6 7 X 7 GE-2	G4607G2	1142	Discharged	
J/4-6 7 X 7 GE-3a	G4607G3A	3752	Discharged	
J/4-6 7 X 7 GE-3b	G4607G3B	1184	Discharged	
J/4-6 8 X 8 GE-4a	G4608G4A	1784	Discharged	
J/4-6 8 X 8 GE-4b	G4608G4B	1787	Discharged	
J/4-6 8 X 8 GE-5	G4608G5	3455	Incore	
J/4-6 8 X 8 GE Pressurized	G4608GP	6591	Incore	
J/4-6 8 X 8 GE Barrier	G4608GB	775	Incore	
J/4-6 8 X 8 GE-8a	G4608G8A		Incore	
J/4-6 8 X 8 GE-8b	G4608G8B		Incore	
J/4-6 8 X 8 GE-9a	G4608G9A		Lead Assembly	
J/4-6 8 X 8 GE-9b	G4608G9B		Lead Assembly	
J/4-6 8 X 8 GE-10	G4608G10		Lead Assembly	
J/4-6 9 X 9 GE-11	G4609G11		Future	
R/4-6 8 X 8 ANF	G4608A		Incore	
R/4-6 8 X 8 ANF Pressurized	G4608AP		Incore	
R/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	
<u>GHOUSE 14 X 14</u>				
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	
<u>IGHOUSE 15 X 15</u>				
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	
<u>NGHOUSE 17 X 17</u>				
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	
<u>I TEXAS</u>				
H TEXAS 17 X 17 WE	WST17W		Incore	

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

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

Assembly Class Assembly Type	Rodarray Code	Assemblies In Storage	Status	Comments
<u>BIG ROCK POINT</u>				
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	
<u>DRESDEN 1</u>				
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	
<u>FORT CALHOUN</u>				
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	
<u>HUMBOLDT BAY</u>				
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	
<u>HADDAM NECK</u>				
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
<u>INDIAN POINT 1</u>				
INDIAN POINT 13 X 14 B&W	XIP14B	0	Reprocessed	
INDIAN POINT 13 X 14 WE	XIP14W	160	Discharged	

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

<u>Assembly Class</u> Assembly Type	Rodarray Code	Assemblies In Storage	Status	Comments
<u>LACROSSE</u>				
LACROSSE 10 X 10 AC	XLC10L	155	Discharged	
LACROSSE 10 X 10 ANF	XLC10A	178	Discharged	
<u>PALISADES</u>				
PALISADES 15 X 15 CE	XPA15C	273	Discharged	
PALISADES 15 X 15 ANF	XPA15A	324	Incore	
<u>ST.LUCIE 2</u>				
ST. LUCIE 2 16 X 16 CE	XSL16C	236	Incore	
<u>SAN ONOFRE 1</u>				
SAN ONOFRE 1 14 X 14 WE	XSO14W	468	Incore	
<u>YANKEE ROWE</u>				
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	

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
<u>Fuel Zone</u>						
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	Microbrazed 50	2.6	1.2	--	--	--
Miscellaneous	SS 304 ^c	9.9	4.6	--	--	--
<u>Fuel-gas plenum zone</u>						
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
<u>End fitting zone</u>						
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

^aSource: 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.

^bAssumed to be discarded with fuel assembly, channels are often reused with fresh fuel.

^cDistributed 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 Engineering SYSTEM80 12-Rod Full-Length Control Element

Composition:

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

Used at the Following Reactors:

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

Used with the Following Fuel Assembly Types:

Vendor	Array	Version
Combustion Engineering	16 x 16	System 80

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

Radiological Description Report

Page 1

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

ISOTOPIC COMPOSITION

Used for 7 cycles (77,000 MWd/MTIHM)

5 years after discharge

Weight: 97.170 kg

Volume of metal: 0.013289 Cu. Meters

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

Used for 10 cycles (111,000 MWd/MTIHM)

5 years after discharge

Weight: 97.170 kg

Volume of metal: 0.013289 Cu. Meters

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

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

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

Radiological Description Report

Page 2

Combustion Engineering SYSTEM80 12-Rod Full-Length Control Element

PHOTON SPECTRA

<u>Mean Energy(MeV)</u>	<u>Photons/second (77,000 MWd/MTIHM)</u>	<u>Photons/second (110,000 MWd/MTIHM)</u>
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)

<u>Material</u>	<u>Total Weight (kg)</u>	<u>Zone</u>
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.

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)
<u>As Integral Part of Fuel Assys</u>					
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
<u>Compacted</u>					
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
<u>Uncompacted</u>					
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
<u>Westinghouse</u>	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 ₄ C/Ag-In-Cd	53 Full Length; 8 Part Length ^b
	17x17	161	8	24	100	Ag-In-Cd	
<u>B&W</u>	15x15	160	4	16	130	Ag-In-Cd	61
<u>Combustion Engineering</u>	16x16	253	8 ^c	4	95	Inconel 625	13
	16x16	253	8 ^c	12	192	B ₄ C	48
	16x16	181	8 ^c	5	92	Inconel 625	8
	16x16	181	8 ^c	5	72	B ₄ C	83
	16x16	181	8 ^c	5	91	Inconel 625	8
	16x16	181	8 ^c	5	71	B ₄ C	73
	16x16	163	8 ^c	5	83	Inconel 625	8
	16x16	163	8 ^c	5	66	B ₄ C	83
	14x14	161	8 ^c	5	105	B ₄ C	12
	14x14	161	8 ^c	5	82	B ₄ C	8
	14x14	161	8 ^c	5	63	B ₄ C	12
	14x14	161	8 ^c	5	77	B ₄ C	65
	14x14	152	8 ^c	5	63	B ₄ C	4
	14x14	152	8 ^c	5	67	B ₄ C	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.

<u>Class Name</u>	Hardware Weight	Hardware Composition
<u>Assembly Type Name</u>		
<u>B&W 15 x 15 Assembly Class</u>		
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
<u>B&W 17 x 17 Assembly Class</u>		
B&W 17 x 17 B&W Mark C	42.3 kg	28% Zirc, 25% Inc, 47% SS
<u>CE 14 x 14 Assembly Class</u>		
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
<u>CE 16 x 16 Assembly Class</u>		
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
<u>CE 16 x 16 System 80 Assembly Class</u>		
CE 16 x 16 CE System 80	44.0 kg	44% Zirc, 13% Inc, 43% SS
<u>WE 14 x 14 Assembly Class</u>		
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
<u>WE 15 x 15 Assembly Class</u>		
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
<u>WE 17 x 17 Assembly Class</u>		
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*

	Westinghouse		Combustion Engineering		Babcock & Wilcox		Exxon
	15 x 15	17 x 17	14 x 14	16 x 16	15 x 15	17 x 17	15 x 15
Rod Array							
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 ^a	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 ^a	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 FWR 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								
Number/assembly	None	None	3	None	None	None	None	None
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

^aSource: L. Iyengar, Westinghouse Electric Corporation, letter to J. W. Roddy, Oak Ridge National Laboratory, December 17, 1984.^bAll 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 ₂ O ₃	Al ₂ O 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

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

^bGuide tubes may be used to guide the control rod assembly or to contain instrumentation which is located in the center guide tube.

^cNot 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	-

^aSource: M. G. Andrews, C-E Power Systems, Combustion Engineering, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, February 11, 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

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

^bNot 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 FWRs
used by Babcock and Wilcox^a

Design component	Rod array		
	Standard	Long life	
	15 × 15		17 × 17
<u>Control rod assembly</u>			
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 ₄ C
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
<u>Burnable poison rod assembly</u>			
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 ₂ O ₃ -B ₄ C		Al ₂ O ₃ -B ₄ C
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

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

^bNiCrMoCb 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	—

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

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

Design component	Rod array			
	14 x 14	15 x 15	17 x 17	14 x 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

^aSource: G. J. Busselman, Exxon Nuclear Company, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, March 28, 1985.

^bProduced only for Combustion Engineering.

^cNot 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.

<u>Class Name</u>				
Assembly Type Name	Hardware Weight ^a	Hardware Materials		Comments
<u>GE BWR/2,3 Assembly Class</u>				
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		
<u>GE BWR/4,5,6 Assembly Class</u>				
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)

<u>Fuel Rod Array</u>	<u>7 x 7</u>	<u>7 x 7R</u>	<u>8 x 8</u>	<u>8 x 8R</u>
Introduction Date	1966	1968	1972	1973
Fuel Rod OD (cm)	1.430	1.448	1.430	1.252
Fuel Rod ID (cm)	1.268	1.242	1.080	1.064
Nominal Cladding Thickness (mil)	32	35.5	37	34
Nominal Diametral Gap (mil)	11	12	12	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.

TABLE 3
MECHANICAL DESIGN PARAMETERS OF BWR FUEL ASSEMBLIES*

Rod Array	General Electric		Exxon
	7 x 7	8 x 8	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 <ul style="list-style-type: none"> — Initial commercial BWRs — First internal steam separation
BWR/2	1963	Oyster Creek <ul style="list-style-type: none"> — The first turnkey plant — Elimination of dual cycle
BWR/3	1965	Dresden-2 <ul style="list-style-type: none"> — The first jet pump application — Improved emergency core cooling system (ECCS)
BWR/4	1966	Browns Ferry <ul style="list-style-type: none"> — Increased power density 20%
BWR/5	1969	Zimmer <ul style="list-style-type: none"> — Improved safeguards — Valve flow control
BWR/6	1972	BWR/6 <ul style="list-style-type: none"> — 8 × 8 fuel bundle — Added fuel bundles, increased output — Improved recirculation system performance — Improved ECCS performance — Reduced fuel duty

^aSource: E. D. Fuller, J. R. Finney, and H. E. Streeter, BWR/6 Nuclear System from General Electric — A Performance Description, NEDO-10569A, April 1972.

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 2.10. 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

^aSource: 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 x 7	8 x 8	8 x 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

^aSource: 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
Glenn 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

^aSource: G. J. Busselman, Exxon Nuclear Company, Inc., letter to J. W. Roddy, Oak Ridge National Laboratory, March 28, 1985.

^bNot 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

^aSource: Allis-Chalmers, Initial Testing of the La Crosse Boiling Water Reactor, ACNP-67533, December 1967.

^bNot available.

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.

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

	<u>Westinghouse</u>	<u>B&W</u>
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	3
	(10 sets for life of reactor)	(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

	<u>Fuel Array</u>	<u>Total Length (in)</u>	<u>Spider Length (in)</u>	<u>Total Weight (lb)</u>	<u>Poison</u>	<u>No. Poison Rods</u>	<u>No. Orifice Rods</u>
<u>Westinghouse</u>							
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 ₄ C	4	12
WABA 12-Rod Neutron Poison	14x14	154	4	28	B ₄ C	12	4
WABA 16-Rod Neutron Poison	14x14	154	4	34	B ₄ C	16	0
WABA 8-Rod Neutron Poison	15x15	154	4	23	B ₄ C	8	12
WABA 20-Rod Neutron Poison	15x15	154	4	41	B ₄ C	20	0
WABA 4-Rod Neutron Poison	15x15	147	4	17	B ₄ C	4	16
WABA 12-Rod Neutron Poison	15x15	147	4	28	B ₄ C	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 ₄ C	4	20
WABA 16-Rod Neutron Poison	17x17	154	4	37	B ₄ C	16	8
WABA 24-Rod Neutron Poison	17x17	154	4	51	B ₄ C	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
<u>B&W</u>							
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 ₄ C; Al ₂ O ₃	16	0
<u>Combustion Engineering</u>							
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

TABLE 3-7
SUMMARY COMPARISON OF ATTRIBUTES OF THIMBLE PLUG ASSEMBLIES IN PWRs^a

	<u>Fuel Array</u>	<u>Total Length (in)</u>	<u>Spider Length (in)^b</u>	<u>Total Weight (lb)</u>	<u>No. Orifice Rods</u>
<u>Westinghouse</u>					
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
<u>B&W</u>					
Standard	15x15	16	5	16	16
<u>Combustion Engineering</u>					
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

<u>Component</u>	<u>Items</u>
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)
<u>As Integral Part of Fuel Assys</u>					
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
<u>Compacted</u>					
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
<u>Uncompacted</u>					
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 MFA HARDWARE
 (Shipped Integral to Fuel Assembly and Separately in Both Canned and Uncanned & 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 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	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	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	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	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	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	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	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 ^e	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		<u>95,105</u>		<u>25,795</u>		<u>14,511</u>		<u>18,526</u>		<u>18,526</u>

^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	Microbrazed 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	--

^aSource: 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.

^bValue 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

^aSource: E. M. Greene, Spent Fuel Data for Waste Storage Programs, HEDL-TME 79-20, September 1980.

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

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 ·day)		Oxidation Depth (μm)*	
	Eq. 1	Eq. 2	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/dm²· day) = $1.12 \times 10^8 \exp (-12,529/T)$ (Ref. A1)

Eq 2. Weight gain rate (mg/dm²· day) = $1.53 \times 10^{10} \exp (-15,590/T)$ (Ref. A3)

T = temperature in °K

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

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

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

Reported at MCC Burnup and Fission Gas Release Distribution Workshop, Seattle, WA, June 27-28, 1990.

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).

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

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 ($<200^\circ\text{C}$). 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

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

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

³ R. E. Einzinger 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.

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

CONSTANT TEMPERATURE TESTS, THICK OXIDE

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.

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

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 ≥ 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

Percent of Fission Gas Release versus Local Temperature

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

Percent of Fission Gas Release versus Local Temperature

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

REVISED ANS-5.4

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

C. N. Wilson
Pacific Northwest Laboratory

C. J. Bruton
Lawrence Livermore National Laboratory

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.

* This work was performed under the auspices of the U. S. Department of Energy (DOE) by Lawrence Livermore National Laboratory under Contract No. W-7405-Eng-48, and by Pacific Northwest Laboratory operated for the DOE by Battelle Memorial Institute under Contract No. DE-AC06-76RLO-1830.

Table 1. Bare Fuel Test Identification

<u>Identification</u>	<u>Description</u>	<u>Starting Fuel Wt. (g)</u>
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 4.4×10^{-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 μ m 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 Analysis²

Component	Concentration (μ g/ml)	Component	Concentration (μ g/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 μ m 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 μ m 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 μ m 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 μ m 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 Cycle 1, 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 $0.4 \mu\text{m}$ 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 specimen 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 from 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 EQ3/6⁵ 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 CO₂ gas fugacity and two different O₂ 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 simulations 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

Actinide	Measured ^(a)		EQ3/6 ^(b)				Phase
			25°C		90°C		
	25°C	85°C	-0.7	-12.0	-0.7	-12.0	
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 = haiweite; 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 predicted 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 thermodynamic 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 $\text{Am}(\text{OH})\text{CO}_3$ is predicted to control Am concentration at 25°C and $\text{Am}(\text{OH})_3$ 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 $\text{PuO}_2 \cdot x\text{H}_2\text{O}$ 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 $\text{PuO}_2 \cdot x\text{H}_2\text{O}$ 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 $\text{Pu}(\text{OH})_4$, 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 $\text{Pu}(\text{OH})_4$, becoming closer to PuO_2 with aging. Equilibrium with amorphous $\text{Pu}(\text{OH})_4$ 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 Bruno⁹ 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 CO₃⁻. 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 Lemire¹⁰ 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 1987⁶ using an older database. The species (UO₂)₃(OH)₇⁻ and (UO₂)₂(OH)₃CO₃⁻ were omitted from version 3270 of the EQ3/6 database because their validity was questioned. UO₂(CO₃)₂⁻² and UO₂(CO₃)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 Ryan⁸ 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 $\text{Am}(\text{OH})\text{CO}_3$ or $\text{Am}(\text{OH})_3$. 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 of 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 $\text{Pu}(\text{OH})_4$. 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 l/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 CFR 60.113¹¹ 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

<u>Actinide</u>	<u>Concentration Log(M)</u>	<u>Log (Release)*</u>
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 l/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 $\text{Am}(\text{OH})\text{CO}_3$ and $\text{Am}(\text{OH})_3$.
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.

REFERENCES

1. R. A. Van Konynenburg, C. F. Smith, H. W. Culham and C. H. Otto Jr., "Behavior of Carbon-14 in Waste Packages for Spent Fuel in a Repository in Tuff," Scientific Basis for Nuclear Waste Management VIII, C. M. Jantzen, J. A. Stone and R. C. Ewing, eds., Materials Research Soc., Pittsburgh, PA, 44:405-412 (1985).
2. J. M. Delany, Reaction of Topopah Spring Tuff with J-13 Water: A Geochemical Modeling Approach Using the EQ3/6 Reaction Path Code, Lawrence Livermore National Laboratory, UCRL-53631 (1985).
3. C. N. Wilson, Results from Cycles 1 and 2 of NNWSI Series 2 Spent Fuel Dissolution Tests, Hanford Engineering Development Laboratory, HEDL-TME 85-22 (1987).
4. C. N. Wilson, "Summary Results from the Series 2 and Series 3 NNWSI Bare Fuel Dissolution Tests," Scientific Basis for Nuclear Waste Management IX, M. J. Apted and R. E. Westerman, eds. Materials Research Soc., Pittsburgh, PA, 112:473-483 (1987).
5. T. J. Wolery, Calculation of Chemical Equilibrium Between Aqueous Solution and Minerals: The EQ3/6 Software Package, Lawrence Livermore National Laboratory, UCRL-52658 (1979).
6. C. J. Bruton and H. F. Shaw, "Geochemical Simulation of Reaction Between Spent Fuel Waste Form and J-13 Water at 25°C and 90°C," Scientific Basis for Nuclear Waste Management IX, M. J. Apted and R. E. Westerman, eds. Materials Research Soc., Pittsburgh, PA, 112:485-494 (1987).
7. R. J. Lemire, An Assessment of the Thermodynamic Behavior of Neptunium in Water and Model Groundwater from 25°C to 150°C, Atomic Energy of Canada Limited, Whiteshell Nucl. Res. Estbl. AECL-7817 (1984).
8. D. Rai and J. L. Ryan "Crystallinity and Solubility of Pu(IV) Oxide and Hydrated Oxide in Aged Aqueous Suspensions," *Radiochem. Acta*, 30:213-216 (1982).
9. I. Puigdomenech and J. Bruno, Modelling Uranium Solubilities in Aqueous Solutions: Validation of Thermodynamic Data Base for the EQ3/6 Geochemical Codes, SKB technical report 88-21 (1988).
10. R. J. Lemire, Effects of High Ionic Strength Groundwaters on Calculated Equilibrium Concentrations in the Uranium-Water System, Atomic Energy of Canada Limited, Whiteshell Nucl. Res. Estbl. AECL-9549 (1988).
11. Code of Federal Regulations, "Disposal of High-Level Radioactive Wastes in Geological Repositories — Licensing Procedures, Title 10, Ch. 1, Pt. 60, Sec. 60.113 (1983).

Lawrence Livermore National Laboratory

LLYMP9101029
January 22, 1991

WBS 1.2.2.3.1.1
QA

SEPDB Administrator
Sandia National Laboratory
Organization 6310
P.O. Box 5800
Albuquerque NM 87185

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

<u>Species</u>	<u>Upper Limit Steady-State Concentration (μg/ml)</u>	
	<u>25°C</u>	<u>85°C</u>
U	≤ 5	≤ 0.5
²³⁹⁺²⁴⁰ Pu	≤ 5x10 ⁻³	≤ 6x10 ⁻⁵
²⁴¹ Am	≤ 3x10 ⁻⁴	≤ 1.5x10 ⁻⁷
²⁴⁴ Cm	≤ 1.2x10 ⁻⁵	≤ 2.4x10 ⁻⁹
²³⁷ Np	≤ 4x10 ⁻⁴	≤ 1.4x10 ⁻³

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

Table 2. Solubility Data Sources

<u>Species</u>	<u>References</u>	
	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 repository 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, $\text{UO}_3 \cdot 2\text{H}_2\text{O}$.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 Germany¹³ 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 $1.2\text{--}1.7 \times 10^{-12} \text{ g cm}^{-2} \text{ sec}^{-1}$. This is similar to the rate for UO_2 in DIW at 25°C at $\sim 5 \times 10^{-12} \text{ g cm}^{-2} \text{ sec}^{-1}$. Given the great variability in other reported values⁴ 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:

$$X(t) = X_0 - \left(\frac{Q}{\rho} \right) t, \quad (1)$$

where

- $X(t)$ = the characteristic dimension as a function of time
- X_0 = 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_0 is then

$$t_{\infty} = \frac{X_0 \rho}{Q} \quad (2)$$

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) ($2X_0$)	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 $X_0 = 0.35$ cm. Here V_0 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

$$V_0 = kX_0^3 \text{ and } V(t) = kX^3(t)$$

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.

$$\frac{V(t)}{V_0} = \left(\frac{X(t)}{X_0} \right)^3 = 1 - 3 \left(\frac{Q}{X_0 \rho} \right) t + 3 \left(\frac{Q}{X_0 \rho} \right)^2 t^2 - \left(\frac{Q}{X_0 \rho} \right)^3 t^3, \quad (3)$$

and the dissolution rate is $-\frac{d\left(\frac{V(t)}{V_0}\right)}{dt}$

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

$$\text{Rate } (t=0) = 3 \left(\frac{Q}{X_0 \rho} \right)$$

and the extrapolated time for total dissolution is

$$t_{\infty}^* = \frac{X_0 \rho}{3Q}$$

In Figure 4 we show that the rate of dissolution relative to the initial rate varies with time for both the system with $X_0 = 0.35$ cm and for the distribution of Table I.

The measured dissolution rate for UO_2^{15} and spent fuel¹⁶ 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

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

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).

References

1. Amell, A. R., and Langmuir, D., "Factors Influencing the Solution Rate of Uranium Dioxide Under Conditions Applicable to In-Situ Leaching", Bureau of Mines Open File Report 84-79, U.S. Department of the Interior - Bureau of Mines, (1978). (Readily Available)
2. Parks, G. A., and Pohl, D. C., "Hydrothermal Solubility of Uraninite", *Geochim. Cosmochim. Acta* 52, 863 (1988). NNA.910821.0003
3. Bruno, J., Casas, I., and Puigdomenech, I., "The Kinetics of Dissolution of $\text{UO}_2(\text{s})$ Under Reducing Conditions", *Radiochimica Acta* 44/45, 11 (1988). NNA.910821.0004
4. Grambow, B., "Spent Fuel Dissolution and Oxidation. An Evaluation of Literature Data", SKB Technical Report 89-13 (1989). NNA.891013.0094
5. Grandstaff, D. E., "A Kinetic Study of the Dissolution of Uraninite", *Econ. Geo.* 71, 1493, (1976). NNA.911025.0061
6. Schortmann, W. E. and De Sesa, M. A., "Kinetics of the Dissolution of Uranium Dioxide in Carbonate-Bicarbonate Solutions", *Proc. 2nd International United Nations Conference Peaceful Uses of Atomic Energy, United Nations, Geneva*, 3, 333 (1958). NNA.910821.0005
7. Pearson, R. L., and Wadsworth, M. E., "A Kinetic Study of the Dissolution of UO_2 in Carbonate Solution", *Trans. Metal Soc. AIME* 212, 294 (1958). NNA.910821.0006
8. Habashi, F., and Thurston, G. A., "Kinetics and Mechanisms of the Dissolution of Uranium Dioxide", *Energ. Nucl.* 14, 238 (1967). NNA.910821.0007
9. Aronson, S., "Oxidation and Corrosion of Uranium Dioxide in Uranium Dioxide: Properties and Nuclear Applications", J. Belle, ed., United States Atomic Energy Commission, 377 (1961). NNA.9110234.0060
10. Einziger, R. E., "Test Plan for Long-Term, Low-Temperature Oxidation of BWR Spent Fuel", PNL-6427, Pacific Northwest Laboratory, (1988). NNA.890224.0045

11. Aronson, S., "Oxidation of UO_2 in Water Containing Oxygen", Bettis Tech. Rev., Westinghouse Atomic Power Div., Report WAPD-BT-10, 93 (1958). NNA.911025.0062
12. Wadsten, T., "The Oxidation of Polycrystalline Uranium Dioxide in Air at Room Temperature", T. Nucl. Mat. 64, 315 (1977). (Readily Available)
13. Wvertz, R. and Ellinger, M., "Source Term for the Activity Release from a Repository for Spent LWR Fuel", Mat. Res. Soc. Symp. Proc 50, 393 (1985). (Readily Available)
14. Gray, W. and Wilson, C., "Effects of Water Composition and Temperature on the Dissolution Rate of UO_2 " presented at 1990 Spent Fuel Workshop, Gull Harbor, Manitoba, Canada (1990). NNA.910821.0008
15. Gray, W., and Strachan, D., "Spent Fuel Grain Boundary Inventory and Testing the Congruency of UO_2 Matrix Dissolution of Spent Fuel", presented at the 1990 Spent Fuel Workshop, Gull Harbor, Manitoba, Canada (1990). NNA.910821.0009
16. Stout, R. B., Shaw, H. F. and Einziger, R. E., "Statistical Model for Grain Boundary and Grain Volume Oxidation Kinetics in UO_2 Spent Fuel", Lawrence Livermore National Laboratory UCRL-100859, September 1989. NNA.891031.0015
17. Stout, R. B., Kansa, E., Buchanan, H. C., Einziger, R. E. and Thomas, L. E., "Spent Fuel Waste Form Characteristics: Grain and Fragment Size Statistical Dependence for Oxidation Studies", Lawrence Livermore National Laboratory UCRL-104932, December 1990. (Readily Available)
18. Stout, R. B. et al, "Spent Fuel Waste Form Characteristics: Grain and Fragment Size Statistical Dependence for Dissolution Response", Lawrence Livermore National Laboratory Report UCRL-104931, December 1990. (Readily Available)
19. Van Luik, A. E., et al, "Spent Nuclear Fuel as a Waste Form for Geologic Disposal: Assessment and Recommendations on Data and Modeling Needs", Pacific Northwest Laboratory Report, PNL-6329 (UC-70), September 1987. (Readily Available)

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 UO₂ Surface

Phase	Formula	Appearance
Schoepite	UO ₃ •2H ₂ O	Dark yellow crystals
Dehydrated Schoepite	UO ₃ •0.8H ₂ O	Yellow crystals with reflective face
Compreignacite	K ₂ U ₆ O ₁₉ •11H ₂ O	Yellow crystals
Uranophane	Ca(UO ₂) ₂ (SiO ₃) ₂ (OH) ₂ •5H ₂ O	Fine white needles
Boltwoodite	K(H ₃ O)UO ₂ (SiO ₄)•nH ₂ O	Yellow crystals
Sklodowskite	Mg(UO ₂) ₂ (SiO ₃ OH) ₂ •5H ₂ O	Fine needles
Becquerelite	CaU ₆ O ₁₉ •10H ₂ O	Dark yellow crystals
Fluoropolymer	Not determined	White feathers

J.K. Bates, *Identification of Secondary Phases Formed During Unsaturated Reaction of UO₂ with EJ-13 Water*, Materials Research Society Symposium proceedings 176, 499 (1990).

Table 2.

U		
Solid	Concentration	
	mg/l	molality (moles/kg)
Haiweeite $\text{Ca}(\text{UO}_2)_2\text{Si}_6\text{O}_{15}\cdot 5\text{H}_2\text{O}$	0.1641E-3	0.6893E-9
Soddyite $(\text{UO}_2)_2\text{SiO}_4\cdot 2\text{H}_2\text{O}$	0.015	0.6096E-7
Sklodowskite $\text{Mg}(\text{H}_3\text{O})_2(\text{UO}_2)_2\cdot (\text{SiO}_4)_2\cdot 4\text{H}_2\text{O}$	11.05	0.4642E-4
CaUO_4	12.59	0.5289E-4
Schoepite $\text{UO}_3\cdot 2\text{H}_2\text{O}$	38.90	0.1634E-3
$\text{UO}_2(\text{OH})_2(\text{beta})$	56.73	0.2383E-3
Uranophane $\text{Ca}(\text{UO}_2)_2(\text{SiO}_3)_2(\text{OH})_2$	142.48	0.5986E-03

Table 3.

Np		
Solid	Concentration	
	mg/l	molality (moles/kg)
NpO_2	0.59	0.2468E-5
$\text{NpO}_2(\text{OH})(\text{am})$	129.39	0.5459E-3
$\text{NaNpO}_2\text{CO}_3\cdot 3.5\text{H}_2\text{O}$	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
$\text{PuO}_2(\text{OH})_2$	0.015	0.6204E-7
$\text{Pu}(\text{OH})_4$	27.97	0.1146E-3

Table 5.

Am		
Solid	Concentration	
	mg/l	molality (moles/kg)
AmOHCO_3	0.0041	0.1696E-7
$\text{Am}(\text{OH})_3$	8.42	0.3464E-4
$\text{Am}(\text{OH})_3(\text{am})$	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.

U Concentration vs. pH in J-13 Water

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

2.1.3.6-3

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

GETSOL: Mon Jul 30 13:11:24 1990

Program Soluplot

bruton /usr/bruton/soluplot 1001.sgl

Mon Nov 26 15:36:51 1990

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

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

Pu Concentration vs. pH in J-13 Water

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

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

Am Concentration vs. pH in J-13 Water

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

Figure 5.12 The solubility of UO₂(am) (both in diluted solutions and in 0.5 M NaClO₄) as a function of pH at 25°C.

Figure 5.14 The solubility of crystalline UO₂(s) versus pH at 100°C.

I. Puigdomenech and J. Bruno, *Modeling Uranium Solubilities in Aqueous Solutions: Validation of a Thermodynamic Data Base for the EQ3/6 Geochemical Codes*, SKB technical report 88-21, October 1988.

Figure 5.15 The solubility of crystalline $\text{UO}_2(\text{s})$ versus pH at 200°C.

Figure 5.16 The calculated solubility of crystalline $\text{UO}_2(\text{s})$ in water at 1 atm $\text{H}_2(\text{g})$ versus T, compared with experimental literature values (for $\text{UO}_2(\text{c})$ 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 UO₂(OH)₂(c), schoepite and Na₂U₂O₇(c) as a function of pH.

I. Puigdomenech and J. Bruno, *Modeling Uranium Solubilities in Aqueous Solutions: Validation of a Thermodynamic Data Base for the EQ3/6 Geochemical Codes*, SKB technical report 88-21, October 1988.

Figure 5.6 The calculated solubility of schoepite and UO₂(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 $\text{UO}_2(\text{OH})_2(\text{c})$ as a function of pH.

I. Puigdomenech and J. Bruno, *Modeling Uranium Solubilities in Aqueous Solutions: Validation of a Thermodynamic Data Base for the EQ3/6 Geochemical Codes*, SKB technical report 88-21, October 1988.

Figure 5.7 The solubility of rutherfordine ($UO_2CO_3(c)$) in dilute solutions as a function of pH at $p_{CO_2} = 1 \text{ atm}$ 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 $p_{CO_2} = 0.97$ and 0.098 atm .

Figure 5.10 The solubility of rutherfordine ($\text{UO}_2\text{CO}_3(\text{c})$) in dilute solutions as a function of pH at $p_{\text{CO}_2} = 1 \text{ atm}$ and 50°C .

Figure 5.11 The calculated solubility of rutherfordine ($\text{UO}_2\text{CO}_3(\text{c})$) in water as a function of temperature at the given values for the partial pressure of $\text{CO}_2(\text{g})$ (p_{CO_2}) compared with experimental results of Sergeyeva et al (1972).

I. Puigdomenech and J. Bruno, *Modeling Uranium Solubilities in Aqueous Solutions: Validation of a Thermodynamic Data Base for the EQ3/6 Geochemical Codes*, SKB technical report 88-21, October 1988.

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.

^{241}Am Activities in O/M = 2.21 Test (25°C)

^{241}Am Activities in O/M = 2.33 Test (25°C)

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.

U, TP fuel, 25°C

U, TP fuel, 85°C

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.

^{237}Np , HBR fuel, 25°C

^{237}Np , HBR fuel, 85°C

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.

$^{239+240}\text{Pu}$, HBR fuel, 25°C

$^{239+240}\text{Pu}$, TP fuel, 85°C

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.

^{241}Am , HBR fuel, 25°C

^{241}Am , TP fuel, 85°C

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.

^{244}Cm , HBR fuel, 25°C

^{244}Cm , TP fuel, 85°C

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 FWR
(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-93 ^m ^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-110 ^m	3.72E+3	-	-	-	-	-
Sn-119 ^m ^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-126 ^m	-	-	-	1.46E+0	1.37E+0	7.35E-1
Te-125 ^m ^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-137 ^m	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-144 ^m	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

^aNuclides contributing >0.1% are listed.

^bBoth activation and fission products contribute to this nuclide.

^cOnly activation products contribute to this nuclide.

^dA.P. = Activation products.

^eF.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 (G1/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)

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	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-93 ^m ^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-110 ^m	1.52E+3	-	-	-	-	-
Sn-119 ^m ^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-126 ^m	-	-	-	7.71E-1	7.24E-1	3.88E-1
Te-125 ^m ^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-137 ^m	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-144 ^m	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

^aNuclides contributing >0.1% are listed.

^bBoth activation and fission products contribute to this nuclide.

^cOnly activation products contribute to this nuclide.

^dA.P. = Activation products.

^eF.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-93 ^m	-	-	-	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-110 ^m	1.63E+3	-	-	-	-	-
Sn-119 ^m	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-126 ^m	-	-	-	8.82E-1	8.28E-1	4.44E-1
Te-125 ^m	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-137 ^m	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-144 ^m	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

^aNuclides contributing >0.1% are listed.

^bBoth activation and fission products contribute to this nuclide.

^cOnly activation products contribute to this nuclide.

^dA.P. = Activation products.

^eF.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)

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	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-93 ^m ^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-110 ^m	1.05E+3	-	-	-	-	-
Sn-119 ^m ^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-126 ^m	-	-	-	6.20E-1	5.83E-1	3.12E-1
Te-125 ^m ^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-137 ^m	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-144 ^m	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

^aNuclides contributing >than 0.1% are listed.

^bBoth activation and fission products contribute to this nuclide.

^cOnly activation products contribute to this nuclide.

^dA.P. = Activation products.

^eF.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.

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

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

1. Parameter of Principal Interest

Leaching rates from activated spent fuel hardware.

2. Test Conditions and Other Qualifying Data

The basis of the estimate of the number and type of fuel assemblies is the 1989 ELA's no new orders case (ref.1 is the annual estimate based on data through 1988: the report incorporating the 1989 data has not yet been published). The no new orders case uses data provided by the utilities on the RW-859 annual report. The projected number of fuel assemblies assumes that the reactors run to their anticipated end of life and are then shutdown. Fuel assemblies are assumed to be discharged on a routine basis, and the type of assemblies are assumed to be the type currently in use, unless information has been provided by the utilities to the contrary. The projection does not account for future design changes that are inevitable. For example, the predominant fuel type for boiling water reactors (BWRs) is currently an 8X8 configuration. Fuel assemblies with 9X9 and 10X10 configurations are being marketed and are expected to constitute a significant portion of the BWR fuel in the future. This, and other similar situations, are not accounted for in these estimates. Additionally, higher burnup of fuels is expected in the future. This would decrease the number of assemblies requiring disposal. This unknown is one of the greatest sources of uncertainty.

The mass and composition of the fuel assemblies was taken from ref.2. The distribution of materials in the core was also obtained from that report. This distribution was necessary in order to account for differing activation levels in the core (e.g. top end fittings vs. material in the active core). A small number of assemblies are not included in these estimates. They include some of the very odd fuel, such as West 15X16, that constitutes less than 1% of the manufacturers projected production.

Estimates of the numbers of non-fuel bearing components were taken from ref.3. Table 4-1 of that report provided estimates of the number of NFBC requiring disposal. Currently, efforts are underway to better define these estimates, but the results of that work will not be available until FY91. The estimates used in this report will likely prove to be

conservative. Estimates of the masses and compositions of the non-fuel bearing components was compiled using data in ref.2, 4, & 5.

The activation of the hardware was estimated using the ORIGEN2 computer code, and incorporating the appropriate scaling factors from ref.2. PWR hardware was activated to a burnup of 33,000 MWD/MTU and BWR to 27,500 MWD/MTU. These values reflect the average burnup expected from these fuel types. Non-fuel bearing components used fluxes commensurate with those burnups, but for component specific residence times. ORIGEN2 was then used to decay the radionuclides out to one million years. The results of these calculations were folded with the material quantities estimated above, and total inventories, as a function of time were developed. These are provided in the attached tables.

Surface area estimates were developed based on detailed measurements of a number of available hardware components. These included both top and bottom end fittings of a GE 8X8 BWR, West 14X14 PWR (top only), West 17X17 PWR, and CE 14X14 PWR. Measurements of grid spacers for these assembly types were also made. These represent a large fraction of the fuel assembly types that were considered. The results of these measurements were used to infer surface areas of fuel assembly types that were not available for measurements. An example of this is the surface area of a West 15X15 assembly. From drawings in ref.2, it can be seen that a 15X15 assembly is similar to a 14X14 assembly with an extra row of fuel pins added to two sides. To estimate the surface area of the end fittings and grid spacers, the measured values from the 14X14 are scaled up by a factor of $(15 \times 15) / (14 \times 14)$. For the guide tubes, the surface area of a single guide tube multiplied by the actual number in each fuel type is used (i.e. 20 in the 15X15 vs. 16 in the 14X14).

In evaluating the leaching of radionuclides from the hardware, the thickness of the materials is important. Though this data was not part of the scope of work, some generalizations can be made that should prove helpful. The end fittings are the most massive pieces. They are generally cast stainless steel, and their thickness varies between 1/8" to 5/8" in individual pieces. Grid spacers, made either of Inconel or Zircaloy, are thin pieces of sheet metal. The thickness of the individual strips are approximately 0.025" in the samples we have measured. The guide tubes, mostly Zircaloy but some stainless steel, are only slightly thicker, at about 0.035" - 0.045". Overall, the bulk of the Zircaloy and Inconel are relatively

thin. Leaching and/or corrosion would occur from both sides of the material. Given the high surface to volume ratio for the bulk of the Zircaloy and Inconel, a higher percentage of their inventory would be expected to be released earlier than the stainless steel end fittings. Leaching from the guide tubes and grid spacers would occur twice as fast as from the fuel cladding, because the grid spacers and guide tubes will corrode and leach from both sides.

Table IV provides the total inventory of actinides and transuranics for the no new orders case that can be used as a comparison to the results in Tables II-A through II-J for the spent fuel hardware and Tables III-A through III-J7 for the non-fuel bearing components. Overall, the number of curies associated with the actinides and transuranics is several orders of magnitude more than the activation products.

List of Tables

Table I	Repository Hardware Inventory Characteristics
Table II-A	Radioactivity At Discharge in Spent Fuel Hardware
Table II-B	100 Year Radioactivity in Spent Fuel Hardware
Table II-C	200 Year Radioactivity in Spent Fuel Hardware
Table II-D	500 Year Radioactivity in Spent Fuel Hardware
Table II-E	1,000 Year Radioactivity in Spent Fuel Hardware
Table II-F	2,000 Year Radioactivity in Spent Fuel Hardware
Table II-G	5,000 Year Radioactivity in Spent Fuel Hardware
Table II-H	10,000 Year Radioactivity in Spent Fuel Hardware
Table II-I	100,000 Year Radioactivity in Spent Fuel Hardware
Table II-J	1 Million Year Radioactivity in Spent Fuel Hardware
Table III-A	Radioactivity At Discharge in Non-Fuel Bearing Components
Table III-B	100 Year Radioactivity in Non-Fuel Bearing Components
Table III-C	200 Year Radioactivity in Non-Fuel Bearing Components
Table III-D	500 Year Radioactivity in Non-Fuel Bearing Components
Table III-E	1,000 Year Radioactivity in Non-Fuel Bearing Components
Table III-F	2,000 Year Radioactivity in Non-Fuel Bearing Components
Table III-G	5,000 Year Radioactivity in Non-Fuel Bearing Components
Table III-H	10,000 Year Radioactivity in Non-Fuel Bearing Components
Table III-I	100,000 Year Radioactivity in Non-Fuel Bearing Components
Table III-J	1 Million Year Radioactivity in Non-Fuel Bearing Components
Table IV	Spent Fuel Activity Inventory

TABLE I
Repository Hardware Inventory Characteristics

MATERIAL	Stainless Steel	Inconel	Zircaloy
Surface Area (M ²)	652,000	1,480,000	400,000
Weight (MT)	3,260	750	6,800
Total Activity At Discharge (Ci)	3,130,000	17,600,000	14,000,000
Total Activity 100 Years (Ci)	1,490,000	8,410,000	7,740
Total Activity 200 Years (Ci)	720,000	4,050,000	6,140
Total Activity 500 Years (Ci)	105,000	587,000	5,210
Total Activity 1,000 Years (Ci)	34,300	190,000	4,980
Total Activity 2,000 Years (Ci)	31,300	177,000	4,680
Total Activity 5,000 Years (Ci)	27,800	167,000	3,960
Total Activity 10,000 Years (Ci)	23,900	152,000	3,190
Total Activity 100,000 Years (Ci)	9,270	54,800	1,980
Total Activity 1 Million Years (Ci)	5	25	1,290

Table II-A**Radioactivity at Discharge in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.43E-03	0.00E+00	0.00E+00	1.43E-03
C 14	1.22E+04	1.18E+04	2.85E+03	2.68E+04
CL 36	1.09E+01	1.05E+01	4.14E+01	6.28E+01
NI 59	2.20E+04	1.27E+05	1.91E+01	1.49E+05
CO 60	0.00E+00	0.00E+00	1.93E+06	1.93E+06
NI 63	3.10E+06	1.75E+07	2.73E+03	2.06E+07
SR 90	0.00E+00	0.00E+00	1.04E+01	1.04E+01
Y 90	0.00E+00	0.00E+00	1.20E+07	1.20E+07
ZR 93	0.00E+00	2.75E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	1.82E-02	6.84E+01	6.84E+01
NB 94	8.62E+01	4.56E+04	3.91E+02	4.61E+04
MO 93	0.00E+00	8.98E+02	0.00E+00	8.98E+02
TC 99	0.00E+00	4.80E+01	0.00E+00	4.80E+01
SN121M	0.00E+00	0.00E+00	4.67E+03	4.67E+03
-----	-----	-----	-----	-----
Total	3.13E+06	1.76E+07	1.40E+07	3.48E+07

Table II-B**100 Year Radioactivity in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.43E-03	0.00E+00	0.00E+00	1.43E-03
C 14	1.20E+04	1.16E+04	2.82E+03	2.64E+04
CL 36	1.09E+01	1.05E+01	4.14E+01	6.28E+01
NI 59	2.20E+04	1.27E+05	1.91E+01	1.49E+05
CO 60	0.00E+00	0.00E+00	3.73E+00	3.73E+00
NI 63	1.46E+06	8.22E+06	1.29E+03	9.68E+06
SR 90	0.00E+00	0.00E+00	9.66E-01	9.66E-01
Y 90	0.00E+00	0.00E+00	9.66E-01	9.66E-01
ZR 93	0.00E+00	2.75E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	2.59E-01	9.77E+02	9.77E+02
NB 94	8.59E+01	4.55E+04	3.90E+02	4.59E+04
MO 93	0.00E+00	8.80E+02	0.00E+00	8.80E+02
TC 99	0.00E+00	4.82E+01	0.00E+00	4.82E+01
SN121M	0.00E+00	0.00E+00	1.17E+03	1.17E+03
-----	-----	-----	-----	-----
Total	1.49E+06	8.41E+06	7.74E+03	9.91E+06

Table II-C

**200 Year Radioactivity in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
BE 10	1.43E-03	0.00E+00	0.00E+00	1.43E-03
C 14	1.19E+04	1.15E+04	2.78E+03	2.61E+04
CL 36	1.09E+01	1.05E+01	4.14E+01	6.28E+01
NI 59	2.20E+04	1.27E+05	1.91E+01	1.49E+05
CO 60	0.00E+00	0.00E+00	7.24E-06	7.24E-06
NI 63	6.86E+05	3.87E+06	6.06E+02	4.56E+06
SR 90	0.00E+00	0.00E+00	8.94E-02	8.94E-02
Y 90	0.00E+00	0.00E+00	8.94E-02	8.94E-02
ZR 93	0.00E+00	2.75E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	2.61E-01	9.82E+02	9.82E+02
NB 94	8.57E+01	4.53E+04	3.89E+02	4.58E+04
MO 93	0.00E+00	8.63E+02	0.00E+00	8.63E+02
TC 99	0.00E+00	4.82E+01	0.00E+00	4.82E+01
SN121M	0.00E+00	0.00E+00	2.91E+02	2.91E+02
Total	7.20E+05	4.05E+06	6.14E+03	4.78E+06

Table II-D

**500 Year Radioactivity in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
BE 10	1.43E-03	0.00E+00	0.00E+00	1.43E-03
C 14	1.15E+04	1.11E+04	2.68E+03	2.52E+04
CL 36	1.09E+01	1.05E+01	4.14E+01	6.28E+01
NI 59	2.19E+04	1.26E+05	1.90E+01	1.48E+05
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	7.16E+04	4.04E+05	6.32E+01	4.75E+05
SR 90	0.00E+00	0.00E+00	7.08E-05	7.08E-05
Y 90	0.00E+00	0.00E+00	7.08E-05	7.08E-05
ZR 93	0.00E+00	2.75E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	2.61E-01	9.82E+02	9.82E+02
NB 94	8.48E+01	4.49E+04	3.85E+02	4.53E+04
MO 93	0.00E+00	8.13E+02	0.00E+00	8.13E+02
TC 99	0.00E+00	4.81E+01	0.00E+00	4.81E+01
SN121M	0.00E+00	0.00E+00	4.54E+00	4.54E+00
Total	1.05E+05	5.87E+05	5.21E+03	6.97E+05

Table II-E**1000 Year Radioactivity in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.43E-03	0.00E+00	0.00E+00	1.43E-03
C 14	1.08E+04	1.04E+04	2.53E+03	2.37E+04
CL 36	1.09E+01	1.05E+01	4.13E+01	6.27E+01
NI 59	2.18E+04	1.26E+05	1.89E+01	1.47E+05
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	1.66E+03	9.33E+03	1.46E+00	1.10E+04
SR 90	0.00E+00	0.00E+00	4.80E-10	4.80E-10
Y 90	0.00E+00	0.00E+00	4.80E-10	4.80E-10
ZR 93	0.00E+00	2.75E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	2.61E-01	.82E+02	9.82E+02
NB 94	8.33E+01	4.41E+04	3.78E+02	4.46E+04
MO 93	0.00E+00	7.37E+02	0.00E+00	7.37E+02
TC 99	0.00E+00	4.81E+01	0.00E+00	4.81E+01
SN121M	0.00E+00	0.00E+00	4.42E-03	4.42E-03
-----	-----	-----	-----	-----
Total	3.43E+04	1.90E+05	4.98E+03	2.30E+05

Table II-F**2000 Year Radioactivity in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.43E-03	0.00E+00	0.00E+00	1.43E-03
C 14	9.55E+03	9.23E+03	2.24E+03	2.10E+04
CL 36	1.09E+01	1.05E+01	4.12E+01	6.25E+01
NI 59	2.16E+04	1.25E+05	1.88E+01	1.46E+05
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	8.85E-01	4.99E+00	7.81E-04	5.87E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	2.74E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	2.61E-01	9.81E+02	9.82E+02
NB 94	8.05E+01	4.26E+04	3.66E+02	4.31E+04
MO 93	0.00E+00	6.04E+02	0.00E+00	6.04E+02
TC 99	0.00E+00	4.79E+01	0.00E+00	4.79E+01
SN121M	0.00E+00	0.00E+00	4.18E-09	4.18E-09
-----	-----	-----	-----	-----
Total	3.13E+04	1.77E+05	4.68E+03	2.13E+05

Table II-G**5000 Year Radioactivity in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.43E-03	0.00E+00	0.00E+00	1.43E-03
C 14	6.64E+03	6.42E+03	1.56E+03	1.46E+04
CL 36	1.08E+01	1.04E+01	4.09E+01	6.21E+01
NI 59	2.11E+04	1.21E+05	1.83E+01	1.42E+05
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	1.35E-10	7.62E-10	1.19E-13	8.97E-10
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	2.74E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	2.60E-01	9.80E+02	9.80E+02
NB 94	7.27E+01	3.85E+04	3.30E+02	3.89E+04
MO 93	0.00E+00	3.33E+02	0.00E+00	3.33E+02
TC 99	0.00E+00	4.74E+01	0.00E+00	4.74E+01
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-----	-----	-----	-----	-----
Total	2.78E+04	1.67E+05	3.96E+03	1.98E+05

Table II-H**10,000 Year Radioactivity in Spent Fuel Hardware
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.42E-03	0.00E+00	0.00E+00	1.42E-03
C 14	3.63E+03	3.51E+03	8.50E+02	7.98E+03
CL 36	1.07E+01	1.03E+01	4.05E+01	6.14E+01
NI 59	2.02E+04	1.16E+05	1.75E+01	1.36E+05
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	2.73E-01	1.03E+03	1.03E+03
NB 93M	0.00E+00	2.60E-01	9.78E+02	9.78E+02
NB 94	6.13E+01	3.24E+04	2.78E+02	3.28E+04
MO 93	0.00E+00	1.24E+02	0.00E+00	1.24E+02
TC 99	0.00E+00	4.67E+01	0.00E+00	4.67E+01
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-----	-----	-----	-----	-----
Total	2.39E+04	1.52E+05	3.19E+03	1.79E+05

Table II-I

100,000 Year Radioactivity in Spent Fuel Hardware
Ci

Isotope	SS	Inc	Zr-4	Total
BE 10	1.37E-03	0.00E+00	0.00E+00	1.37E-03
C 14	6.77E-02	6.55E-02	1.59E-02	1.49E-01
CL 36	8.67E+00	8.35E+00	3.29E+01	4.99E+01
NI 59	9.25E+03	5.33E+04	8.03E+00	6.26E+04
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	2.62E-01	9.88E+02	9.88E+02
NB 93M	0.00E+00	2.49E-01	9.39E+02	9.39E+02
NB 94	2.84E+00	1.50E+03	1.29E+01	1.52E+03
MO 93	0.00E+00	2.23E-06	0.00E+00	2.23E-06
TC 99	0.00E+00	3.48E+01	0.00E+00	3.48E+01
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	9.27E+03	5.48E+04	1.98E+03	6.61E+04

Table II-J

1 Million Year Radioactivity in Spent Fuel Hardware
Ci

Isotope	SS	Inc	Zr-4	Total
BE 10	9.26E-04	0.00E+00	0.00E+00	9.26E-04
C 14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CL 36	1.09E+00	1.05E+00	4.14E+00	6.28E+00
NI 59	3.80E+00	2.19E+01	3.30E-03	2.57E+01
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	1.75E-01	6.57E+02	6.57E+02
NB 93M	0.00E+00	1.66E-01	6.24E+02	6.24E+02
NB 94	1.28E-13	6.76E-11	5.80E-13	6.83E-11
MO 93	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TC 99	0.00E+00	1.86E+00	0.00E+00	1.86E+00
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	4.89E+00	2.51E+01	1.29E+03	1.32E+03

TABLE III-A**Radioactivity at Discharge in Non-Fuel Bearing Components
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.31E-06	0.00E+00	0.00E+00	1.31E-06
C 14	1.21E+01	3.89E+02	7.03E+03	7.43E+03
CL 36	1.09E-02	3.47E-01	1.03E+02	1.04E+02
NI 59	2.12E+01	4.21E+03	4.30E+01	4.28E+03
CO 60	0.00E+00	0.00E+00	4.83E+06	4.83E+06
NI 63	3.08E+03	5.75E+05	6.75E+03	5.85E+05
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.36E-03	2.77E+03	2.77E+03
NB 93M	0.00E+00	6.19E-04	1.85E+02	1.85E+02
NB 94	8.64E-02	1.51E+03	9.96E+02	2.51E+03
MO 93	0.00E+00	2.99E+01	0.00E+00	2.99E+01
TC 99	0.00E+00	1.24E+00	0.00E+00	1.24E+00
SN121M	0.00E+00	0.00E+00	1.16E+04	1.16E+04
-----	-----	-----	-----	-----
Total	3.11E+03	5.81E+05	4.86E+06	5.44E+06

TABLE III-B**100 Year Radioactivity in Non-Fuel Bearing Components
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.31E-06	0.00E+00	0.00E+00	1.31E-06
C 14	1.19E+01	3.84E+02	6.94E+03	7.34E+03
CL 36	1.09E-02	3.47E-01	1.03E+02	1.04E+02
NI 59	2.12E+01	4.21E+03	4.29E+01	4.27E+03
CO 60	0.00E+00	0.00E+00	9.35E+00	9.35E+00
NI 63	1.45E+03	2.71E+05	3.18E+03	2.75E+05
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.36E-03	2.77E+03	2.77E+03
NB 93M	0.00E+00	8.84E-03	2.62E+03	2.62E+03
NB 94	8.61E-02	1.51E+03	9.92E+02	2.50E+03
MO 93	0.00E+00	2.93E+01	0.00E+00	2.93E+01
TC 99	0.00E+00	1.25E+00	0.00E+00	1.25E+00
SN121M	0.00E+00	0.00E+00	2.91E+03	2.91E+03
-----	-----	-----	-----	-----
Total	1.48E+03	2.77E+05	1.96E+04	2.98E+05

TABLE III-C**200 Year Radioactivity in Non-Fuel Bearing Components
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.31E-06	0.00E+00	0.00E+00	1.31E-06
C 14	1.18E+01	3.79E+02	6.86E+03	7.25E+03
CL 36	1.09E-02	3.47E-01	1.03E+02	1.04E+02
NI 59	2.12E+01	4.21E+03	4.29E+01	4.27E+03
CO 60	0.00E+00	0.00E+00	1.81E-05	1.81E-05
NI 63	6.83E+02	1.27E+05	1.50E+03	1.30E+05
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.35E-03	2.77E+03	2.77E+03
NB 93M	0.00E+00	8.89E-03	2.63E+03	2.63E+03
NB 94	8.58E-02	1.50E+03	9.89E+02	2.49E+03
MO 93	0.00E+00	2.87E+01	0.00E+00	2.87E+01
TC 99	0.00E+00	1.25E+00	0.00E+00	1.25E+00
SN121M	0.00E+00	0.00E+00	7.27E+02	7.27E+02
-----	-----	-----	-----	-----
Total	7.16E+02	1.34E+05	1.56E+04	1.50E+05

TABLE III-D**500 Year Radioactivity in Non-Fuel Bearing Components
Ci**

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.31E-06	0.00E+00	0.00E+00	1.31E-06
C 14	1.14E+01	3.66E+02	6.62E+03	7.00E+03
CL 36	1.08E-02	3.47E-01	1.03E+02	1.04E+02
NI 59	2.11E+01	4.20E+03	4.28E+01	4.26E+03
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	7.12E+01	1.33E+04	1.56E+02	1.35E+04
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.35E-03	2.77E+03	2.77E+03
NB 93M	0.00E+00	8.89E-03	2.63E+03	2.63E+03
NB 94	8.49E-02	1.49E+03	9.79E+02	2.47E+03
MO 93	0.00E+00	2.71E+01	0.00E+00	2.71E+01
TC 99	0.00E+00	1.24E+00	0.00E+00	1.24E+00
SN121M	0.00E+00	0.00E+00	1.13E+01	1.13E+01
-----	-----	-----	-----	-----
Total	1.04E+02	1.94E+04	1.33E+04	3.28E+04

TABLE III-E

1,000 Year Radioactivity in Non-Fuel Bearing Components
Ci

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.31E-06	0.00E+00	0.00E+00	1.31E-06
C 14	1.07E+01	3.44E+02	6.23E+03	6.58E+03
CL 36	1.08E-02	3.46E-01	1.03E+02	1.03E+02
NI 59	2.10E+01	4.18E+03	4.26E+01	4.24E+03
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	1.65E+00	3.07E+02	3.61E+00	3.12E+02
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.35E-03	2.77E+03	2.77E+03
NB 93M	0.00E+00	8.88E-03	2.63E+03	2.63E+03
NB 94	8.35E-02	1.46E+03	9.63E+02	2.43E+03
MO 93	0.00E+00	2.45E+01	0.00E+00	2.45E+01
TC 99	0.00E+00	1.24E+00	0.00E+00	1.24E+00
SN121M	0.00E+00	0.00E+00	1.10E-02	1.10E-02
-----	-----	-----	-----	-----
Total	3.35E+01	6.32E+03	1.27E+04	1.91E+04

TABLE III-F

2,000 Year Radioactivity in Non-Fuel Bearing Components
Ci

Isotope	SS	Inc	Zr-4	Total
-----	-----	-----	-----	-----
BE 10	1.31E-06	0.00E+00	0.00E+00	1.31E-06
C 14	9.47E+00	3.05E+02	5.52E+03	5.83E+03
CL 36	1.08E-02	3.46E-01	1.03E+02	1.03E+02
NI 59	2.09E+01	4.14E+03	4.22E+01	4.20E+03
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	8.80E-04	1.64E-01	1.93E-03	1.67E-01
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.35E-03	2.77E+03	2.77E+03
NB 93M	0.00E+00	8.88E-03	2.63E+03	2.63E+03
NB 94	8.07E-02	1.41E+03	9.30E+02	2.34E+03
MO 93	0.00E+00	2.01E+01	0.00E+00	2.01E+01
TC 99	0.00E+00	1.24E+00	0.00E+00	1.24E+00
SN121M	0.00E+00	0.00E+00	1.04E-08	1.04E-08
-----	-----	-----	-----	-----
Total	3.04E+01	5.88E+03	1.20E+04	1.79E+04

TABLE III-G

5,000 Year Radioactivity in Non-Fuel Bearing Components
Ci

Isotope	SS	Inc	Zr-4	Total
BE 10	1.31E-06	0.00E+00	0.00E+00	1.31E-06
C 14	6.59E+00	2.12E+02	3.84E+03	4.06E+03
CL 36	1.07E-02	3.43E-01	1.02E+02	1.03E+02
NI 59	2.03E+01	4.03E+03	4.11E+01	4.10E+03
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	1.34E-13	2.51E-11	2.95E-13	2.55E-11
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.33E-03	2.76E+03	2.76E+03
NB 93M	0.00E+00	8.87E-03	2.62E+03	2.62E+03
NB 94	7.28E-02	1.28E+03	8.40E+02	2.12E+03
MO 93	0.00E+00	1.11E+01	0.00E+00	1.11E+01
TC 99	0.00E+00	1.23E+00	0.00E+00	1.23E+00
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	2.70E+01	5.54E+03	1.02E+04	1.58E+04

TABLE III-H

10,000 Year Radioactivity in Non-Fuel Bearing Components
Ci

Isotope	SS	Inc	Zr-4	Total
BE 10	1.30E-06	0.00E+00	0.00E+00	1.30E-06
C 14	3.60E+00	1.16E+02	2.10E+03	2.22E+03
CL 36	1.06E-02	3.39E-01	1.01E+02	1.01E+02
NI 59	1.95E+01	3.86E+03	3.94E+01	3.92E+03
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	9.31E-03	2.76E+03	2.76E+03
NB 93M	0.00E+00	8.85E-03	2.62E+03	2.62E+03
NB 94	6.14E-02	1.08E+03	7.08E+02	1.78E+03
MO 93	0.00E+00	4.12E+00	0.00E+00	4.12E+00
TC 99	0.00E+00	1.21E+00	0.00E+00	1.21E+00
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	2.31E+01	5.06E+03	8.32E+03	1.34E+04

TABLE III-I
100,000 Year Radioactivity in Non-Fuel Bearing Components
Ci

Isotope	SS	Inc	Zr-4	Total
BE 10	1.25E-06	0.00E+00	0.00E+00	1.25E-06
C 14	6.72E-05	2.16E-03	3.91E-02	4.14E-02
CL 36	8.63E-03	2.76E-01	8.21E+01	8.24E+01
NI 59	8.93E+00	1.77E+03	1.81E+01	1.80E+03
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	8.94E-03	2.65E+03	2.65E+03
NB 93M	0.00E+00	8.50E-03	2.51E+03	2.51E+03
NB 94	2.84E-03	4.98E+01	3.28E+01	8.25E+01
MO 93	0.00E+00	7.42E-08	0.00E+00	7.42E-08
TC 99	0.00E+00	9.00E-01	0.00E+00	9.00E-01
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	8.94E+00	1.82E+03	5.29E+03	7.13E+03

TABLE III-J
1 Million Year Radioactivity in Non-Fuel Bearing Components
Ci

Isotope	SS	Inc	Zr-4	Total
BE 10	8.50E-07	0.00E+00	0.00E+00	8.50E-07
C 14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CL 36	1.09E-03	3.47E-02	1.03E+01	1.04E+01
NI 59	3.66E-03	7.27E-01	7.42E-03	7.39E-01
CO 60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 93	0.00E+00	5.95E-03	1.76E+03	1.76E+03
NB 93M	0.00E+00	5.65E-03	1.67E+03	1.67E+03
NB 94	1.28E-16	2.24E-12	1.48E-12	3.72E-12
MO 93	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TC 99	0.00E+00	4.81E-02	0.00E+00	4.81E-02
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	4.75E-03	8.22E-01	3.44E+03	3.44E+03

Table IV
Spent Fuel Activity Inventory
CI

Time	PWR	BWR	Total
Discharge	9.94E+12	4.06E+12	1.40E+13
100 Years	2.27E+09	9.31E+08	3.20E+09
200 Years	4.47E+08	1.88E+08	6.35E+08
500 Years	1.64E+08	7.25E+07	2.37E+08
1,000 Years	9.75E+07	4.33E+07	1.41E+08
2,000 Years	5.40E+07	2.41E+07	7.81E+07
5,000 Years	3.57E+07	1.60E+07	5.17E+07
10,000 Years	2.59E+07	1.17E+07	3.76E+07
100,000 Years	2.64E+06	1.14E+06	3.78E+06
1 Million Yrs	1.08E+06	4.67E+05	1.54E+06

References

- 1 SR/CNEAF/90-01, Spent Nuclear Fuel Discharges from U.S. Reactors 1988, prepared by the Energy Information Administration, U.S. Department of Energy, March 1990.**
- 2 DOE/RW-0184, Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation, prepared by Oak Ridge National Laboratory, December 1987.**
- 3 ORNL/Sub/86-SA094/8, Acceptance of Non-Fuel Assembly Hardware by the Federal Waste Management System, prepared by E.R. Johnson Associates, Inc., March 1990.**
- 4 PNL-6046, Spent Fuel Disassembly Hardware and Other Non-Fuel Bearing Components: Characterization, Disposal Cost Estimates, and Proposed Repository Acceptance Criteria, A.T. Luksic, et al, October 1986**
- 5 PNL-6906, Spent Fuel Assembly Hardware: Characterization and 10CFR61 Classification for Waste Disposal, A.T. Luksic, et al, June 1989.**

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

2.2 Glass Waste Form

The glass form is the most probable disposal form for the various High-Level Wastes (HLW) other than Spent Fuel. HLW originates from domestic fuel reprocessing plants that are both commercial and defense related.

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

ORNL DWG 80-8260

*Includes spacing between fuel assembly rods.
 Does not include DOE spent fuel to be reprocessed.

Fig. 0.1. Volumes of commercial and DOE wastes and spent fuel accumulated through 1989.

ORNL DWG 80-8261

*Does not include DOE spent fuel to be reprocessed.

Fig. 0.2. Radioactivities of commercial and DOE wastes and spent fuel accumulated through 1989.

Table 6.2. Dimensions, weights, and radioactivity of canisters

	West Valley Demonstration Project	Savannah River Site	Hanford Site	Idaho Nat'l Engineering Laboratory
Outside diameter, cm	61	61	61	61
Overall height, cm	300	300	300	300
Material	SS	SS	SS	SS
Wall thickness, mm	3.4	9.5	9.5	9.5
Weights (kg)				
Canister	252	500	500	500
Glass or ceramic	1900	1682	1650	1825
Total	2152	2182	2150	2325
Curies per canister ^a	114,700	234,400	298,000 ^b	109,000
Watts per canister ^a	342	709	869 ^c	339

^aThese are estimated maximum values from ORIGEN2 calculations based on radionuclide compositions supplied by the sites. Curies and watts shown are at time of filling the canister, except for West Valley Demonstration Project where the values shown are for the start of year 1992. For West Valley Demonstration Project, maximum values are assumed to be 110% of average values, and average values are based on the Revision 7 mass balance (Crocker 1989a). Maximum values for the defense sites do not necessarily represent initial operations.

^bBased on Mitchell and Nelson 1988, maximum case.

^cBased on ORIGEN2 calculations using Mitchell and Nelson maximum case.

Table 6.3. West Valley Demonstration Project.
High-level waste form and canister characteristics^a

Waste form	Borosilicate glass in sealed canister
Canister material	Stainless steel type 304L
Borosilicate glass density, g/cm ³ at 25°C	2.7
Weights per canister:	
Empty canister, kg	234
Cover, kg	18
Borosilicate glass, kg	<u>1,900</u>
Total loaded weight, kg	2,152
Canister dimensions:	
Outside diameter, cm	61
Height overall, cm	300
Wall thickness, cm	0.34
Radionuclide content, ^b curies per canister (1992) ^b	
Average ^c	104,300
Maximum ^c	114,700
Thermal power, watts per canister (1992) ^b	
Average ^c	311
Maximum ^c	342

^aSource: Crocker 1989a and ORNL calculations based on Rev. 7 mass balance.

^bQuantities shown are at 85% fill. Curies and watts per canister are for the start of year 1992.

^cMaximum activity is assumed to be 110% of average.

Table 6.4. Savannah River Site. High-level waste form and canister characteristics^a

	Canister 85% fill
Canister inside volume, m ³	0.736
Glass volume at average fill temperature (see note b), m ³	0.626
Glass density at average fill temperature (see note b), g/cm ³	2.69
Glass weight, kg	1,682
Canister weight, kg	500
Gross weight, kg	2,182
Total activity, curies	234,000 ^c
Decay heat, watts	690 ^c

^aSources: DWPF Basic Data Report, DPSP 80-1033, Rev. 91, April 1985

^bThe average fill temperature (i.e. the average temperature of the glass upon completion of filling to 85% of canister volume) is 825°C. The glass volume per canister when cooled to 25°C is about 0.59 m³. The density of the glass is about 2.69 g/cm³ at 825°C and 2.85 g/cm³ at 25°C (SRP 1987).

^cThese figures are the ones given in DPSP 80-1033, Rev. 91. The corresponding figures calculated by ORIGEN2 are 234,400 Ci and 709W, as shown in Table 3.3.4. Activity and decay heat (thermal power) are at the time of filling the canister and are based on the maximum case, i.e. 5-yr old sludge and 15-yr old supernate.

Table 6.5. Hanford Site. High-level waste form and canister characteristics^a

Waste form	Borosilicate glass in sealed steel canister
Canister material	Type 304L stainless steel
Weights per canister	
Empty canister, kg	500
Borosilicate glass, kg	<u>1650</u>
Total loaded weight, kg	2150
Canister dimensions	
Outside diameter, cm	61
Height overall, cm	300
Wall thickness, cm	0.95
Inside volume, m ³	0.736
Glass volume at average fill temperature, m ³	0.626 ^b
Radionuclide content, curies per canister ^c	
Nominal	137,000
Maximum	298,000
Thermal power, watts per canister ^c	
Nominal	389
Maximum	869

^aSources: Wolfe 1985, White 1986, Mitchell and Nelson 1988.

^bCanister is filled to 85% of volume at average fill temperature of 825°C.

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

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

Table 6.6. Idaho National Engineering Laboratory. High level waste form and canister characteristics^a

Waste form	Glass-ceramic blocks in sealed canister
Canister material	Stainless steel type 304L
Glass-ceramic density, g/cm ³	3.2
Weights per canister:	
Empty canister, kg	500
Glass-ceramic, kg	<u>1825</u>
Total loaded weight, kg	2325
Waste loading in glass-ceramic, wte	70 ^b
Glass-ceramic volume per canister, m ³	0.57 ^b
Canister dimensions:	
Outside diameter, cm.	61
Height overall, cm.	300
Wall thickness, cm.	0.95
Radionuclide content, curies/canister	108,900 ^c
Heat generation rate, watts/canister	339 ^c

^aBased on the following assumptions:

1. Glass-ceramic form is chosen for HLW immobilization. The term "glass-ceramic" denotes an immobilized waste form consisting of a glass phase dispersed in a ceramic phase.

2. Canister load is equivalent to 1277 kg calcine.

3. Calcine is 3 years old at time of immobilization.

4. Canister is similar in dimensions to DWPF canister.

5. Radionuclide content of calcine is as shown in IDO-10105 (see Table 3.5.3).

^bReference: Berreth 1987.

^cAt time of immobilization. Quantities shown are estimated maximum values; average values are expected to be considerably less.

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

Fig. 0.3. Projections of annual volume additions for various wastes and spent fuel – DOE and commercial fuel cycle.

Fig. 0.4. Projections of annual radioactivity additions for various wastes and spent fuel – DOE and commercial fuel cycle.

Fig. 0.5. Projections of accumulated volumes for various wastes and spent fuel – DOE and commercial fuel cycle.

Fig. 0.6. Projections of accumulated radioactivity for various wastes and spent fuel – DOE and commercial fuel cycle.

Table 6.7. Projected annual number of canisters of immobilized HLW produced at each site^{a,b}

Calendar year	WVDP	SRS	Hanford	INEL	Total
1989	0	0	0	0	0
1990	0	0	0	0	0
1991	0	0	0	0	0
1992	0	136	0	0	136
1993	25	308	0	0	333
1994	200	376	0	0	576
1995	50	410	0	0	460
1996	0	410	0	0	410
1997	0	383	0	0	383
1998	0	369	0	0	369
1999	0	369	0	0	369
2000	0	342	240	0	582
2001	0	342	370	0	712
2002	0	342	345	0	687
2003	0	342	185	0	527
2004	0	302	370	0	672
2005	0	273	370	0	643
2006	0	273	80	0	353
2007	0	273	0	0	273
2008	0	32	0	0	32
2009	0	0	0	0	0
2010	0	0	0	0	0
2011	0	0	0	0	0
2012	0	0	0	500	500
2013	0	0	0	600	600
2014	0	0	0	700	700
2015	0	0	0	1000	1000
2016	0	0	0	1000	1000
2017	0	0	0	1000	1000
2018	0	0	0	1000	1000
2019	0	0	0	1000	1000
2020	0	0	0	1000	1000
Total	275	5282	1960	7800	15,317

^aSources: WVDP - Crocker 1989, 1989a, 1990

SRS - Garvin 1990

HANF - Turner 1990

INEL - Berreth 1990

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

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

Figure 6.1 Annual Canister Production for each Site

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

Figure 6.2 Total Annual Canister Production

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

Table 6.8. Projected cumulative production of canisters of immobilized HLW at each site^{a,b}

Calendar year	WVDP	SRP	Hanford	INEL	Total
1989	0	0	0	0	0
1990	0	0	0	0	0
1991	0	0	0	0	0
1992	0	136	0	0	136
1993	25	444	0	0	469
1994	225	820	0	0	1045
1995	275	1230	0	0	1505
1996	275	1640	0	0	1915
1997	275	2023	0	0	2298
1998	275	2392	0	0	2667
1999	275	2761	0	0	3036
2000	275	3103	240	0	3618
2001	275	3445	610	0	4330
2002	275	3787	955	0	5017
2003	275	4129	1140	0	5544
2004	275	4431	1510	0	6216
2005	275	4704	1880	0	6859
2006	275	4977	1960	0	7212
2007	275	5250	1960	0	7485
2008	275	5282	1960	0	7517
2009	275	5282	1960	0	7517
2010	275	5282	1960	0	7517
2011	275	5282	1960	0	7517
2012	275	5282	1960	500	8017
2013	275	5282	1960	1100	8617
2014	275	5282	1960	1800	9317
2015	275	5282	1960	2800	10317
2016	275	5282	1960	3800	11317
2017	275	5282	1960	4800	12317
2018	275	5282	1960	5800	13317
2019	275	5282	1960	6800	14317
2020	275	5282	1960	7800	15317

^aSources: WVDP - Crocker 1989, 1989a, 1990.

SRS - Garvin 1990.

HANF - Turner 1990.

INEL - Berreth 1990.

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

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

Figure 6.3 Cumulative Canister Production
for each Site

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

Figure 6.4 Cumulative Canister Production
for all Sites

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

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

Figure 6.5 Radioactivity and Thermal Power Per Canister

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

Table 6.9. West Valley Demonstration Project. Calculated radioactivity and thermal power per HLW canister^a

Decay time after 1990, years	Average radioactivity per canister (Ci)	Average thermal power per canister (W)
0	109,600	326
1	106,900	319
2	104,300	311
5	97,080	290
10	86,230	258
15	76,660	230
20	68,180	205
30	53,970	164
50	33,890	105
100	10,730	37
200	1,260	8.9
300	291	5.4
350	202	4.8
500	128	3.7
1,000	63	1.9
1,050	60	1.8
2,000	26.7	0.70
5,000	15.8	0.33
10,000	13.2	0.26
20,000	10.3	0.18
50,000	6.8	0.08
100,000	5.2	0.04
500,000	3.8	0.04
1,000,000	3.1	0.03

^aCalculations made with ORIGEN2 code based on data supplied by WVDP (Crocker 1989). Canister contains 1900 kg of HLW glass. Initial time point (0 years) is at the start of year 1990. The material balance used by WVDP for this case (Revision 7, October 1989) shows 484,000 kg of total glass and a total radioactivity of 27.9×10^6 Ci at the start of year 1990 in the HLW to be vitrified. Data are for the average canister and do not take into account possible variations in melter feed and fill level.

Figure 6.6 Radioactivity and Thermal Power Per Canister.

WEST VALLEY DEMONSTRATION PROJECT

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

Table 6.10. Savannah River Site. Calculated radioactivity and thermal power per HLW canister.^a

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

^aBased on 5-yr cooled sludge and 15-yr cooled supernate. Calculations made by ORIGEN2 code based on data supplied by SRS (Basic Data Report, DPSP-80-1033, Rev. 91, April 1985). Canister is filled to 85% of capacity and contains 1682 kg of glass.

^bYears after vitrification.

^cRadioactivity and thermal power include contributions of actinides and activation products as well as fission products.

Figure 6.7 Radioactivity and Thermal Power Per Canister.

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

Table 6.11. Hanford Site. Calculated radioactivity and thermal power per HLW canister^a

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

^aCalculations made by ORIGEN2 code based on data supplied by HANF (Mitchell and Nelson 1988). Canister is filled to 85% of capacity and contains 1650 kg of HLW glass made from neutralized current acid waste (NCAW). Data are shown for two cases, the nominal case and the maximum case. The maximum case is based on a 21-month cooling time from fuel reprocessing to HWVP.

^bYears after vitrification.

Figure 6.8 Radioactivity and Thermal Power Per Canister.

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

Table 6.12. Idaho National Engineering Laboratory. Calculated radioactivity and thermal power per HLW canister^a

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

^aResults of ORIGEN2 calculations based on glass-ceramic form, assuming 1277 kg of calcine per canister (1825 kg of glass-ceramic per canister), with the initial radionuclide composition shown in Table 3.5.3.

Figure 6.9 Radioactivity and Thermal Power Per Canister.

IDAHO NATIONAL ENGINEERING LAB

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

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

Table 6.13. Radioisotope composition of West Valley
Demonstration Project Vitrifified High-Level Waste^a

Radioisotope	Grams/canister	Curies/canister	Watts/canister
Fe-55	0.1104E-02	0.2760E+01	0.9313E-04
Co-60	0.2679E-02	0.3030E+01	0.4666E-01
Ni-59	0.5491E+01	0.4160E+00	0.1650E-04
Ni-63	0.4895E+00	0.3020E+02	0.3039E-02
Se-79	0.1980E+00	0.1380E-01	0.3431E-05
Sr-90	0.1928E+03	0.2630E+05	0.3048E+02
Y-90	0.4833E-01	0.2630E+05	0.1456E+03
Zr-93	0.4257E+03	0.1070E+01	0.1242E-03
Nb-93m	0.2529E-02	0.7150E+00	0.1265E-03
Tc-99	0.2524E+02	0.4280E+00	0.2144E-03
Ru-106	0.1655E-04	0.5540E-01	0.3290E-05
Rh-106	0.1556E-10	0.5540E-01	0.5307E-03
Pd-107	0.8416E+02	0.4330E-01	0.2563E-05
Cd-113m	0.3845E-01	0.8340E+01	0.1402E-01
Sn-121m	0.1160E-02	0.6860E-01	0.1373E-03
Sn-126	0.1441E+02	0.4090E+00	0.5095E-03
Sb-125	0.2769E-01	0.2860E+02	0.8929E-01
Sb-126	0.6852E-06	0.5730E-01	0.1057E-02
Sb-126m	0.5206E-08	0.4090E+00	0.5201E-02
Te-125m	0.3885E-03	0.7000E+01	0.5876E-02
Cs-134	0.1569E-01	0.2030E+02	0.2063E+00
Cs-135	0.5505E+03	0.6340E+00	0.2113E-03
Cs-137	0.3252E+03	0.2830E+05	0.3126E+02
Ba-137m	0.4981E-04	0.2680E+05	0.1051E+03
Ce-144	0.8023E-06	0.2560E-02	0.1696E-05
Pr-144	0.3387E-10	0.2560E-02	0.1879E-04
Pm-146	0.9566E-04	0.4260E-01	0.2146E-03
Pm-147	0.3721E+00	0.3450E+03	0.1236E+00
Sm-151	0.1258E+02	0.3310E+03	0.3876E-01
Eu-152	0.8267E-02	0.1430E+01	0.1080E-01
Eu-154	0.1389E+01	0.3750E+03	0.3350E+01
Eu-155	0.2014E+00	0.9370E+02	0.6806E-01
Tl-207	0.1690E-09	0.3220E-01	0.9444E-04
Tl-208	0.4312E-10	0.1270E-01	0.2985E-03
Pb-209	0.1815E-09	0.8250E-03	0.9475E-06
Pb-211	0.1308E-08	0.3230E-01	0.9666E-04
Pb-212	0.2540E-07	0.3530E-01	0.6712E-04
Bi-211	0.7718E-10	0.3230E-01	0.1287E-02
Bi-212	0.2409E-08	0.3530E-01	0.5995E-03
Bi-213	0.4265E-10	0.8250E-03	0.3464E-05
Po-212	0.1274E-18	0.2260E-01	0.1196E-02
Po-213	0.6231E-19	0.7860E-03	0.3972E-04
Po-215	0.1095E-14	0.3230E-01	0.1440E-02
Po-216	0.1013E-12	0.3530E-01	0.1443E-02
At-217	0.5124E-15	0.8250E-03	0.3516E-04
Rn-219	0.2482E-11	0.3230E-01	0.1339E-02
Rn-220	0.3826E-10	0.3530E-01	0.1339E-02
Fr-221	0.4653E-11	0.8250E-03	0.3180E-04
Fr-223	0.1117E-10	0.4320E-03	0.1120E-05
Ra-223	0.6306E-06	0.3230E-01	0.1149E-02

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

Table 6.13. (continued)

Radioisotope	Grams/canister	Curies/canister	Watts/canister
Ra-224	0.2216E-06	0.3530E-01	0.1210E-02
Ra-225	0.2104E-07	0.8250E-03	0.5778E-06
Ra-228	0.2550E-04	0.5970E-02	0.4595E-06
Ac-225	0.1421E-07	0.8250E-03	0.2878E-04
Ac-227	0.4464E-05	0.3230E-03	0.1562E-06
Ac-228	0.2662E-08	0.5970E-02	0.5153E-04
Th-227	0.1034E-05	0.3180E-01	0.1159E-02
Th-228	0.4306E-04	0.3530E-01	0.1153E-02
Th-229	0.3877E-02	0.8250E-03	0.2521E-04
Th-230	0.1169E-01	0.2360E-03	0.6670E-05
Th-231	0.6657E-09	0.3540E-03	0.1984E-06
Th-232	0.5880E+05	0.6450E-02	0.1559E-03
Th-234	0.1356E-06	0.3140E-02	0.1271E-05
Pa-231	0.1264E+01	0.5970E-01	0.1796E-02
Pa-233	0.4422E-05	0.9180E-01	0.2081E-03
Pa-234m	0.4571E-11	0.3140E-02	0.1550E-04
U-232	0.1270E-02	0.2720E-01	0.8721E-03
U-233	0.3666E+01	0.3550E-01	0.1031E-02
U-234	0.2640E+01	0.1650E-01	0.4746E-03
U-235	0.1637E+03	0.3540E-03	0.9259E-05
U-236	0.1700E+02	0.1100E-02	0.2976E-04
U-238	0.9337E+04	0.3140E-02	0.7954E-04
Np-236	0.2823E+01	0.3720E-01	0.7494E-04
Np-237	0.1302E+03	0.9180E-01	0.2802E-02
Np-239	0.5861E-05	0.1360E+01	0.3283E-02
Pu-236	0.6209E-05	0.3300E-02	0.1147E-03
Pu-238	0.1904E+01	0.3260E+02	0.1079E+01
Pu-239	0.1028E+03	0.6390E+01	0.1967E+00
Pu-240	0.2053E+02	0.4680E+01	0.1455E+00
Pu-241	0.3076E+01	0.3170E+03	0.9815E-02
Pu-242	0.1668E+01	0.6370E-02	0.1879E-03
Am-241	0.6117E+02	0.2100E+03	0.6967E+01
Am-242	0.1435E-05	0.1160E+01	0.1315E-02
Am-242m	0.1204E+00	0.1170E+01	0.4616E-03
Am-243	0.6820E+01	0.1360E+01	0.4366E-01
Cm-242	0.2912E-03	0.9630E+00	0.3544E-01
Cm-243	0.1021E-01	0.5270E+00	0.1931E-01
Cm-244	0.3707E+00	0.3000E+02	0.1048E+01
Cm-245	0.2015E-01	0.3460E-02	0.1147E-03
Cm-246	0.1279E-02	0.3930E-03	0.1285E-04
Total	0.7029E+05	0.1096E+06	0.3260E+03

^aCalculated from data in WVDP Mass Balance, Revision 7 (Crocker 1989). This is based on 484,000 kg of total glass and 1900 kg of glass per canister, and represents the average canister composition. Radioactivity shown is as of the start of year 1990.

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

Table 6.14. Savannah River Site. Radioisotope
content per HLW canister^a

	Isotope	Curies/canister	Grams/canister
1	Cr-51	0.9312E-16	0.1008E-20
2	Co-60	0.1699E+03	0.1502E+00
3	Ni-59	0.2397E-01	0.3163E+00
4	Ni-63	0.2975E+01	0.4824E-01
5	Tl-208	0.1128E-02	0.3829E-11
6	U-232	0.1339E-01	0.6256E-03
7	U-233	0.1584E-05	0.1636E-03
8	U-234	0.3428E-01	0.5485E+01
9	U-235	0.1573E-03	0.7278E+02
10	U-236	0.1128E-02	0.1742E+02
11	U-238	0.1050E-01	0.3122E+05
12	Np-236	0.1744E-07	0.1323E-05
13	Np-237	0.8904E-02	0.1263E+02
14	Pu-236	0.1221E+00	0.2297E-03
15	Pu-237	0.8941E-11	0.7401E-15
16	Pu-238	0.1484E+04	0.8667E+02
17	Pu-239	0.1291E+02	0.2076E+03
18	Pu-240	0.8681E+01	0.3809E+02
19	Pu-241	0.1670E+04	0.1620E+02
20	Pu-242	0.1224E-01	0.3206E+01
21	Am-241	0.1102E+02	0.3210E+01
22	Am-242	0.1436E-01	0.1776E-07
23	Am-242m	0.1447E-01	0.1488E-02
24	Am-243	0.5788E-02	0.2902E-01
25	Cm-242	0.3495E-01	0.1057E-04
26	Cm-243	0.5565E-02	0.1078E-03
27	Cm-244	0.1076E+03	0.1329E+01
28	Cm-245	0.6715E-05	0.3910E-04
29	Cm-246	0.5342E-06	0.1739E-05
30	Cm-247	0.6604E-12	0.7116E-08
31	Cm-248	0.6864E-12	0.1614E-09
32	Se-79	0.1699E+00	0.2439E+01
33	Rb-87	0.8719E-06	0.9961E+01
34	Sr-89	0.4267E-04	0.1470E-08
35	Sr-90	0.4675E+05	0.3426E+03
36	Y-90	0.4786E+05	0.8795E-01
37	Y-91	0.7568E-03	0.3085E-07
38	Zr-93	0.1117E+01	0.4443E+03

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

Table 6.14. (continued)

	Isotope	Curies/canister	Grams/canister
39	Zr-95	0.1005E-01	0.4680E-06
40	Nb-94	0.9646E-04	0.5147E-03
41	Nb-95	0.2115E-01	0.5407E-06
42	Nb-95m	0.1247E-03	0.3272E-09
43	Tc-99	0.3079E+01	0.1816E+03
44	Ru-103	0.1684E-07	0.5217E-12
45	Ru-106	0.2252E+04	0.6729E+00
46	Rh-103m	0.1636E-07	0.5028E-15
47	Rh-106	0.2259E+04	0.6346E-06
48	Pd-107	0.1473E-01	0.2863E+02
49	Ag-110m	0.1258E+00	0.2647E-04
50	Cd-113	0.5009E-13	0.1472E+00
51	Cd-115m	0.1213E-08	0.4763E-13
52	Sn-121m	0.7902E-01	0.1336E-02
53	Sn-123	0.2549E+00	0.3101E-04
54	Sn-126	0.4415E+00	0.1556E+02
55	Sb-124	0.7123E-07	0.4071E-11
56	Sb-125	0.8496E+03	0.8226E+00
57	Sb-126	0.6159E-01	0.7365E-06
58	Sb-126m	0.4415E+00	0.5619E-08
59	Te-125m	0.2760E+03	0.1532E-01
60	Te-127	0.1202E+00	0.4555E-07
61	Te-127m	0.1228E+00	0.1302E-04
62	Te-129	0.3053E-11	0.1457E-18
63	Te-129m	0.4749E-11	0.1576E-15
64	Cs-134	0.3372E+03	0.2606E+00
65	Cs-135	0.9943E-01	0.8633E+02
66	Cs-136	0.7828E-39	0.1068E-43
67	Cs-137	0.4341E+05	0.4989E+03
68	Ba-136m	0.8607E-38	0.3195E-49
69	Ba-137m	0.4155E+05	0.7724E-04
70	Ba-140	0.1024E-35	0.1404E-40
71	La-140	0.4304E-36	0.7734E-42
72	Ce-141	0.3591E-10	0.1260E-14
73	Ce-142	0.9609E-05	0.4005E+03
74	Ce-144	0.9869E+04	0.3093E+01
75	Pr-143	0.1198E-33	0.1780E-38
76	Pr-144	0.9869E+04	0.1306E-03
77	Pr-144m	0.1187E+03	0.6545E-06

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

Table 6.14. (continued)

	Isotope	Curies/canister	Grams/canister
78	Nd-144	0.4860E-09	0.4110E+03
79	Nd-147	0.1261E-43	0.1570E-48
80	Pm-147	0.2419E+05	0.2609E+02
81	Pm-148	0.6975E-10	0.4243E-15
82	Pm-148m	0.1009E-08	0.4722E-13
83	Sm-147	0.2000E-05	0.8796E+02
84	Sm-148	0.5788E-11	0.1916E+02
85	Sm-149	0.1781E-11	0.7420E+01
86	Sm-151	0.2478E+03	0.9418E+01
87	Eu-152	0.3688E+01	0.2132E-01
88	Eu-154	0.6196E+03	0.2295E+01
89	Eu-155	0.4749E+03	0.1021E+01
90	Eu-156	0.5231E-31	0.9489E-36
91	Tb-160	0.1120E-05	0.9923E-10
Total		0.2344E+06	0.3427E+05

*Quantities shown are for sludge + supernate glass and are based on the DWPF Basic Data Report, DPSP 80-1033, Rev. 91, April 1985, assuming sludge aged an average of 5 years and supernate aged an average of 15 years, with a canister load of 3710 lb of glass (1682 kg). Radionuclide contents are at time of filling canister.

Table 6.15. Hanford Site. Radioisotope content
per HLW canister (NCAW glass)^a

Isotope	Curies/canister		Grams/canister	
	Nominal	Maximum	Nominal	Maximum
Fe-55	1.80E+01	1.41E+02	7.20E-03	5.64E-02
Ni-59	1.09E-01	1.36E-01	1.44E+00	1.80E+00
Co-60	1.50E+00	4.29E+00	1.33E-03	3.79E-03
Ni-63	1.21E+01	1.57E+01	1.96E-01	2.54E-01
Se-79	3.15E-03	3.90E-03	4.52E-02	5.60E-02
Sr-89	5.35E-13	6.52E-02	1.84E-17	2.24E-06
Sr-90	2.98E+04	4.18E+04	2.18E+02	3.06E+02
Y-90	2.98E+04	4.18E+04	5.48E-02	7.68E-02
Y-91	1.38E-10	7.26E-01	5.63E-15	2.96E-05
Nb-93m	6.16E-01	5.77E-01	2.18E-03	2.04E-03
Zr-93	1.05E+00	1.29E+00	4.18E+02	5.13E+02
Zr-95	2.92E-09	2.76E+00	1.36E-13	1.28E-04
Nb-95	6.73E-09	5.67E+00	1.72E-13	1.45E-04
Tc-99	7.51E+00	9.35E+00	4.43E+02	5.51E+02
Ru-103	3.37E-18	2.98E-04	1.04E-22	9.23E-09
Rh-103m	3.04E-18	2.69E-04	9.34E-26	8.27E-12
Ru-106	4.18E+01	4.99E+03	1.25E-02	1.49E+00
Rh-106	4.18E+01	4.99E+03	1.17E-08	1.40E-06
Pd-107	3.02E-02	4.07E-02	5.87E+01	7.91E+01
Ag-110m	2.22E-03	1.59E+00	4.67E-07	3.35E-04
Cd-113m	8.53E+00	1.46E+01	3.93E-02	6.73E-02
In-113m	1.01E-07	2.52E-02	6.04E-15	1.51E-09
Sn-113	1.01E-07	2.52E-02	1.01E-11	2.51E-06
Cd-115m	3.20E-18	8.78E-06	1.26E-22	3.45E-10
Sn-119m	6.80E-03	5.42E+00	1.52E-06	1.21E-03
Sn-121m	7.76E-02	1.06E-01	1.31E-03	1.79E-03
Sn-123	3.65E-05	2.89E+00	4.44E-09	3.52E-04
Sn-126	3.65E-01	4.60E-01	1.29E+01	1.62E+01
Sb-124	1.15E-14	3.50E-05	6.57E-19	2.00E-09
Sb-126	5.10E-02	6.48E-02	6.10E-07	7.75E-07
Sb-126m	3.65E-01	4.60E-01	4.65E-09	5.86E-09
Sb-125	2.54E+02	1.76E+03	2.46E-01	1.70E+00
Te-125m	6.20E+01	4.29E+02	3.44E-03	2.38E-02
Te-127	6.55E-06	2.95E+00	2.48E-12	1.12E-06
Te-127m	6.66E-06	3.00E+00	7.06E-10	3.18E-04
Te-129	3.14E-23	3.75E-07	1.49E-30	1.79E-14
Te-129m	4.82E-23	5.77E-07	1.60E-27	1.91E-11
I-129	1.29E-05	1.63E-05	7.31E-02	9.23E-02

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

Table 6.15. (continued)

Isotope	Curies/canister		Grams/canister	
	Nominal	Maximum	Nominal	Maximum
Cs-134	9.31E+01	1.20E+03	7.19E-02	9.27E-01
Cs-135	2.02E-01	2.51E-01	1.75E+02	2.18E+02
Cs-137	3.61E+04	5.10E+04	4.15E+02	5.86E+02
Ba-137m	3.40E+04	4.82E+04	6.32E-05	8.96E-05
Ce-141	2.93E-22	1.13E-05	1.03E-26	3.97E-10
Ce-144	8.00E+01	2.98E+04	2.51E-02	9.34E+00
Pr-144	8.00E+01	2.98E+04	1.06E-06	3.94E-04
Pr-144m	9.60E-01	3.58E+02	5.29E-09	1.97E-06
Pm-147	5.21E+03	3.97E+04	5.62E+00	4.28E+01
Pm-148m	6.23E-19	1.32E-05	2.92E-23	6.18E-10
Sm-151	6.98E+02	8.36E+02	2.65E+01	3.18E+01
Eu-152	1.40E-00	2.74E+00	8.09E-03	1.58E-02
Gd-153	1.35E-05	1.15E-02	3.83E-09	3.26E-06
Eu-154	1.45E+02	3.36E+02	5.37E-01	1.24E+00
Eu-155	1.37E+02	4.11E+02	2.94E-01	8.83E-01
Tb-160	9.49E-13	1.10E-04	8.41E-17	9.74E-09
U-234	4.57E-03	4.82E-03	7.31E-01	7.71E-01
U-235	1.91E-04	1.97E-04	8.83E+01	9.11E+01
U-236	4.21E-04	4.75E-04	6.51E+00	7.34E+00
U-238	3.51E-03	3.72E-03	1.04E+04	1.11E+04
Np-237	1.56E-01	1.99E-01	2.21E+02	2.82E+02
Pu-238	4.43E-01	7.68E-01	2.59E-02	4.48E-02
Pu-239	1.17E+00	1.41E+00	1.88E+01	2.27E+01
Pu-240	3.93E-01	5.42E-01	1.72E+00	2.38E+00
Pu-241	1.26E+01	2.58E+01	1.22E-01	2.50E-01
Pu-242	7.61E-05	1.31E-04	1.99E-02	3.43E-02
Am-241	2.84E+02	5.77E+02	8.27E+01	1.68E+02
Am-242	2.21E-01	4.14E-01	2.73E-07	5.12E-07
Am-243	3.79E-02	6.76E-02	1.90E-01	3.39E-01
Cm-242	1.82E-01	4.99E-01	5.50E-05	1.51E-04
Cm-244	5.03E+00	1.25E+01	6.22E-02	1.54E-01
Total	1.37E+05	2.98E+05	1.26E+04	1.40E+04

This table identifies the nominal and maximum activity of HWVP canisters at the time of vitrification. The maximum is principally based on close-coupling the final accumulated tank of NCAW (21 mo. from fuel discharge to HWVP). The overall waste loading is 0.031 kg non-volatile oxides/l (0.26 lb non-volatile oxides/gal) of feed. Canister contains 1650 kg of HLW glass (85% fill). Source: Mitchell and Nelson 1988.

Table 6.16. Idaho National Engineering Laboratory.
Radioisotope content per HLW canister^a

	Isotope	Curies/canister	Grams/canister
1	Sa-79	0.8173E-01	0.1173E+01
2	Rb-87	0.4597E-05	0.5252E+02
3	Sr-90	0.1660E+05	0.1217E+03
4	Y-90	0.1660E+05	0.3051E-01
5	Zr-93	0.3959E+00	0.1575E+03
6	Nb-93M	0.9577E-01	0.3387E-03
7	Tc-99	0.2682E+01	0.1582E+03
8	Ru-106	0.1239E+04	0.3701E+00
9	Rh-106	0.1239E+04	0.3479E-06
10	Pd-107	0.2554E-02	0.4965E+01
11	Sn-126	0.4086E-01	0.1440E+01
12	Sb-126M	0.4086E-01	0.5201E-09
13	Sb-126	0.4086E-01	0.4887E-06
14	Cs-134	0.4214E+04	0.3256E+01
15	Cs-135	0.9577E-01	0.8316E+02
16	Cs-137	0.1660E+05	0.1908E+03
17	Ba-137M	0.1532E+05	0.2848E-04
18	Ce-144	0.1047E+05	0.3282E+01
19	Pr-144	0.1047E+05	0.1386E-03
20	Pm-147	0.1532E+05	0.1653E+02
21	Sm-151	0.2171E+03	0.8250E+01
22	Eu-154	0.2299E+03	0.8513E+00
23	U-233	0.1532E-08	0.1583E-06
24	U-234	0.5491E-06	0.8785E-04
25	U-235	0.2299E-05	0.1063E+01
26	U-236	0.1277E-04	0.1973E+00
27	U-237	0.6130E-08	0.7507E-13
28	U-238	0.1277E-10	0.3797E-04
29	Np-237	0.6130E-04	0.8693E-01
30	Pu-238	0.8939E+02	0.5221E+01
31	Pu-239	0.8939E+00	0.1437E+02
32	Pu-240	0.8300E+00	0.3642E+01
33	Pu-241	0.2043E+03	0.1983E+01
34	Pu-242	0.2299E-02	0.6018E+00
35	Am-241	0.1162E+01	0.3385E+00
36	Am-243	0.1060E-01	0.5315E-01
37	Cm-242	0.8300E+00	0.2510E-03
38	Cm-244	0.6640E+00	0.8201E-02
Total		0.1088E+06	0.8315E+03

^aQuantities are at time of filling canister and are based on 3-yr old calcine immobilized in glass-ceramic with a load of 1277 kg of calcine per canister (1825 kg of glass-ceramic per canister). Based on IDO-10105 (1982) and Berreth 1986.

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

Table 6.17. West Valley Demonstration Project.
Chemical composition of reference HLW glass^a

Component	Nominal composition (wt%)	Range (wt%)	
AgO	0.0001	-	-
Al ₂ O ₃	2.8295	1.19	7.15
AmO ₂	0.0073	-	-
BaO	0.0540	0.04	0.08
B ₂ O ₃	9.9516	9.33	10.66
CaO	0.5993	0.39	0.93
CdO	0.0003	-	-
CeO ₂	0.0670	0.04	0.10
CrO ₂	0.0001	-	-
CoO	0.0002	-	-
Cr ₂ O ₃	0.3112	0.21	0.48
Cs ₂ O	0.0826	0.05	0.13
CuO	0.0001	-	-
Eu ₂ O ₃	0.0014	-	-
Fe ₂ O ₃	12.1570	8.32	18.50
Gd ₂ O ₃	0.0003	-	-
In ₂ O ₃	0.0001	-	-
K ₂ O	3.5733	3.36	3.84
La ₂ O ₃	0.0337	0.02	0.05
Li ₂ O	3.0315	2.84	3.25
MgO	1.3032	1.22	1.39
MnO ₂	1.3107	0.84	1.96
MoO ₃	0.0088	-	0.01
NaCl	0.0183	0.01	0.03
NaF	0.0013	-	-
Na ₂ O	10.9340	10.25	11.71
Nd ₂ O ₃	0.1209	0.08	0.19
NiO	0.3358	0.22	0.52
NpO ₂	0.0224	0.01	0.03
P ₂ O ₅	2.5084	0.21	3.16
PdO	0.0062	-	-
Pm ₂ O ₃	0.0003	-	-
Pr ₆ O ₁₁	0.0321	0.02	0.05
PuO ₂	0.0076	-	-
Rb ₂ O	0.0005	-	-

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

Table 6.17. (continued)

Component	Nominal composition (wt%)	Range (wt%)	
RhO ₂	0.0136	0.01	0.02
RuO ₂	0.0759	0.05	0.12
SO ₃	0.2164	0.14	0.33
Sb ₂ O ₃	0.0001	-	-
SeO ₂	0.0005	-	-
SiO ₂	44.8770	42.08	48.10
Sm ₂ O ₃	0.0267	0.02	0.04
SnO ₂	0.0006	-	-
SrO	0.0269	0.02	0.04
Tc ₂ O ₇	0.0021	-	-
ThO ₂	3.5844	1.83	6.56
TeO ₂	0.0028	-	-
TiO ₂	0.9800	0.92	1.05
UO ₂	0.5605	0.37	0.87
Y ₂ O ₃	0.0177	0.01	0.03
ZnO	0.0010	-	-
ZrO ₂	0.2943	0.19	0.45
Other	<u>0.0084</u>	-	-
Total	100.0000		

^aSource: Eisenstatt 1986. Reference glass composition is WV-205.

Table 6.18. Savannah River Site.
Chemical composition of HLW glass^a

Component	Water free wt %
Ag	0.05
Al ₂ O ₃	3.96
B ₂ O ₃	10.28
BaSO ₄	0.14
Ca ₃ (PO ₄) ₂	0.07
CaO	0.85
CaSO ₄	0.08
Cr ₂ O ₃	0.12
Cs ₂ O	0.08
CuO	0.19
Fe ₂ O ₃	7.04
FeO	3.12
K ₂ O	3.58
Li ₂ O	3.16
MgO	1.36
MnO	2.00
Na ₂ O	11.00
Na ₂ SO ₄	0.36
NaCl	0.19
NaF	0.07
NiO	0.93
PbS	0.07
SiO ₂	45.57
ThO ₂	0.21
TiO ₂	0.99
U ₃ O ₈	2.20
Zeolite	1.67
ZnO	0.08
Others	0.58
Total	100.00

^aSource: Baxter 1988.

Chemical Composition of Sludge-Precipitate Glass, Baxter, R. G., DP-1606, Rev. 2, Savannah River Plant, SC, December 1988, p. 49.

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

Table 6.19. Hanford Site.
Chemical compositions of HWVP reference HLW (NCAW),
substituted NCAW, frit, and borosilicate glass^a.

Component	Reference NCAW waste composition wt %	Substituted NCAW waste _b composition wt %	Frit composition wt %	Glass composition wt %
SiO ₂	2.9	3.0	67.25	51.3
B ₂ O ₃	0.0	0.0	12.75	9.6
Na ₂ O	10.5	10.7	10.25	10.4
Li ₂ O	0.0	0.0	5.0	3.8
CaO	0.3	0.3	3.75	2.9
MgO	0.2	0.3	1.0	0.8
Fe ₂ O ₃	44.0	44.4	--	11.1
Al ₂ O ₃	17.0	17.2	--	4.3
Cr ₂ O ₃	5.3	5.3	--	1.3
ZrO ₂	2.3	2.4	--	0.6
NiO	2.3	2.4	--	0.6
La ₂ O ₃	2.2	2.2	--	0.6
SO ₄	1.8	1.8	--	0.4
Nd ₂ O ₃	1.7	2.1	--	0.5
MoO ₃	1.2	1.2	--	0.3
F	1.2	1.2	--	0.3
CuO	0.6	0.6	--	0.1
TOC _c	0.6	0.6	--	--
MnO ₂	0.6	0.7	--	0.2
CeO ₂	0.6	0.7	--	0.2
RuO ₂	0.6	0.6	--	0.1
U ₃ O ₈	0.6	Sub Nd	--	--
Cs ₂ O	0.4	1.0	--	0.2
BaO	0.4	0.4	--	0.1
SrO	0.4	0.4	--	0.1
Pr ₂ O ₃	0.4	0.4	--	0.2
Tc ₂ O ₇	0.4	Sub Mn	--	--
Rb ₂ O	0.2	Sub Cs	--	--
Y ₂ O ₃	0.2	0.2	--	0.04
Sm ₂ O ₃	0.2	0.2	--	0.04
PdO	0.2	Del	--	--
Rh ₂ O ₃	0.2	Del	--	--
NpO ₂	0.1	Sub Ce	--	--
TeO ₂	0.1	Del	--	--
Pm ₂ O ₃	0.1	Sub Nd	--	--
BeO	0.1	Sub Mg	--	--
SeO ₂	0.03	Del	--	--
SnO ₂	0.02	Del	--	--
CdO	0.02	Del	--	--

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

Table 6.19. (continued)

Component	Reference NCAW waste composition wt %	Substituted NCAW waste, composition ^b wt %	Frit composition wt %	Glass composition wt %
Eu ₂ O ₃	0.02	Sub Nd	--	--
PuO ₂	0.02	Sub Ce	--	--
Am ₂ O ₃	0.02	Sub Nd	--	--
P ₂ O ₅	0.02	Del	--	--
Ag ₂ O	0.01	Del	--	--
Nb ₂ O ₅	0.01	Sub Mo	--	--
Cd ₂ O ₈	0.01	0.01	--	0.003
Ta ₂ O	0.01	Del	--	--
TiO ₂	0.01	Del	--	--
Total	100	100	100	100

^aSource: Mitchell 1986. Reference glass is HW-39. Data given are for a waste oxide loading of 25 wt% and are based on approximately 4-year old waste.

^bComponents marked sub were substituted as indicated. Components marked Del were deleted. TOC - total organic carbon

Table 6.20. Compositions of typical ceramic-based waste forms developed for immobilization of INEL calcined HLW^a

Formulation number	SiO ₂ (wt %)	Na ₂ O (wt %)	Li ₂ O (wt %)	B ₂ O ₃ (wt %)	Waste (wt %)
12	8.6	1.1	0.5	2.6	87.2
11	16.0	0.0	0.0	1.4	82.6
17	30.3	0.0	0.0	2.3	67.5
6	28.6	2.1	0.9	3.5	64.9
1	14.2	2.6	1.2	1.7	80.3

^aSource: Baker 1986.

PROJECTED DWPF WASTE GLASS COMPOSITIONS

MAJOR GLASS COMPONENTS weight %	CONSTITUENT SLUDGE TYPE					
	Blend ^d	Batch 1	Batch 2	Batch 3	Batch 4	Purex ^w
Al ₂ O ₃	3.98	4.87	4.46	3.25	3.32	2.89
B ₂ O ₃	8.01	7.69	7.70	7.69	8.11	10.21
BaSO ₄	0.27	0.22	0.24	0.26	0.38	0.29
CaO	0.97	1.17	1.00	0.93	0.83	1.02
CaSO ₄	0.077	0.12	0.11	0.10	0.0034	0.12
Cr ₂ O ₃	0.12	0.10	0.12	0.13	0.14	0.14
CuO	0.44	0.40	0.41	0.40	0.46	0.42
Fe ₂ O ₃	10.41	12.52	10.61	11.16	11.32	12.74
Group A ^a	0.14	0.099	0.14	0.10	0.20	0.078
Group B ^b	0.36	0.22	0.44	0.25	0.60	0.084
K ₂ O	3.86	3.49	3.50	3.47	3.99	3.58
Li ₂ O	4.40	4.42	4.42	4.42	4.32	3.12
MgO	1.35	1.36	1.35	1.35	1.38	1.33
MnO	2.03	2.06	1.62	1.81	3.08	1.99
Na ₂ O	8.73	8.62	8.61	8.51	8.88	12.14
Na ₂ SO ₄	0.10	0.10	0.12	0.096	0.13	0.12
NaCl	0.19	0.31	0.23	0.22	0.090	0.26
NiO	0.89	0.75	0.90	1.07	1.09	1.21
SiO ₂	50.20	49.81	50.17	49.98	49.29	44.56
ThO ₂	0.19	0.36	0.63	0.77	0.24	0.011
TiO ₂	0.90	0.66	0.67	0.66	1.02	0.65
U ₃ O ₈	2.14	0.53	2.30	3.16	0.79	2.89
Total	99.76	99.88	99.75	99.79	99.66	99.85

^a Group A: semi-volatile radionuclides (Se, Te, Rb, Mo, Tc)

^b Group B: nonvolatile radionuclides (e.g., Sm, Sn, Co, Np, Am, Cm)

^d The "Blend" is the current DWPF design-basis glass

^w The "Purex" glass is a possible "worst-case" composition

M.J. Plodinec, *Defense Waste Processing Facility High Level Waste Qualification Activities*, presented to the Nuclear Waste Technical Review Board, August, 1990.

ACCEPTABLE GLASSES

R.A. Palmer, *West Valley Demonstration Project High Level Waste Qualification Activities*, presented to the Nuclear Waste Technical Review Board, August, 1990.

DURABLE GLASS REGION

<u>GLASS OXIDES</u>	<u>COMPOSITION (WT. %)</u>	<u>LOWER BOUND</u>	<u>UPPER BOUND</u>
Si	43.1	38.0	45.0
Al	6.2	5.0	10.0
Fe	12.2	10.0	16.0
B	10.9	6.9	13.0
Na+K+Li	17.8	12.0	20.0
P	2.4	0.5	4.0
Mn	1.0	0.1	2.0

R.A. Palmer, *West Valley Demonstration Project High Level Waste Qualification Activities*, presented to the Nuclear Waste Technical Review Board, August, 1990.

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

Table 1. Summary of studies examining surface area increases due to thermal fracturing

Glass Composition	Glass block size (relative)	Surface Area Increase (relative to unfractured glass)	Reference
SRL211	large-scale	2 - 40	Smith and Baxter 1981
SRL211, SRL131	large-scale	7 - 18	Peters and Slate 1981
SRL211, SRL131	small-scale	0 - 18	Peters and Slate 1981
borosilicate	large-scale	9.0 - 16.3	Laude et al. 1982
SRL165	large-scale	25 - 35	Bickford and Pellarin 1987
borosilicate	small- to large scale	1.1 - 86	Faletti and Ethridge 1988
borosilicate	medium-scale	2.0 - 10	Lutze et al. 1986
R7T7	small-scale, 1:10	10 -12	Vernaz and Godon 1991
PNL76-375	large-scale	8 - 45	Martin 1985
PNL76-375	small-scale	1.1 - 12	Martin 1985
borosilicate	medium-scale	not measured	Keinzler 1989
BRETHLW borosilicate glass	medium-scale	not measured	Farnsworth et al. 1985

BICKFORD D. F. and PELLARIN D. J. (1987) Large scale leach testing of DWPF canister sections. in *Sci. Bas. for Nucl. Was. Mgmt. vol X*, eds. J. K. Bates and W. B. Seefeldt, Mat. Res. Soc. Symp. Proc. vol. 84, 509-518.

FALETTI D. W. and ETHRIDGE L. J. (1988) A method for predicting cracking in waste glass canisters. *Nucl. and Chem. Waste Mgmt.* 8, 123-133.

FARNSWORTH R. K., CHAN M. K. W. and SLATE S. C. (1985) The effect of radial temperature gradients on glass fracture in simulated high-level waste canisters. in *Sci. Bas. for Nucl. Was. Mgmt. vol VIII*, eds. Jantzen, Stone and Ewing, Mat. Res. Soc. Symp. Proc. vol. 44, 191-198.

KIENZLER B. (1989) Cooling and cracking of technical HLW glass products: experimental and numerical studies. in *Sci. Bas. for Nucl. Was. Mgmt.*

LAUDE F., VERNAZ E., and SAINT-GAUDENS M. (1982) Fracture appraisal of large scale glass blocks under realistic thermal conditions. in *Sci. Bas. for Nucl. Was. Mgmt. vol V*, ed. W. Lutze, Mat. Res. Soc. Symp. Proc. vol. 11, 239-247.

LUTZE W., MANARA A., MARPLES J. A. C., OFFERMANN P. and van ISEGHEM P. (1986) *Radioactive waste Management and Disposal*, Luxembourg, ed. R. Simon (Cambridge Univ. Press, Cambridge).

MARTIN D. M. (1985) *Fracture in Glass/High Level Waste Canisters*. NUREG/CR-4198. Engineering Research Inst., Iowa State Univ., Ames, ..

PETERS, R. D., and SLATE S. C. (1981) Fracturing of simulated high-level waste glass in canisters. *Nucl. Eng. and Design* 67, 425-445.

SMITH P. K. and BAXTER C. A. (1981) Fracture during cooling of cast borosilicate glass containing nuclear wastes. USDOE report DP-1602. E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, SC.

VERNAZ E. Y. and GODON N. (1991) Key parameters of glass dissolution in integrated systems. in *Scientific Basis for Nuclear Waste Mgmt. XIII* eds. T. A. Abrajano and Lawrence, in press.

Figure 1. Correlation between Relative Surface Area and Radial Temperature Difference at 500°C Centerline Temperature. From Faletti and Ethridge (1988).

FALETTI D. W. and ETHRIDGE L. J. (1988) A method for predicting cracking in waste glass canisters. *Nucl. and Chem. Waste Mgmt.* 8, 123-133.

Figure 2. Effect of Cooling Time on Relative Surface Area of Thermal Cracks in 17 cm and 60 cm Dia Canisters. From Peters and Slate (1981).

PETERS, R. D., and SLATE S. C. (1981) Fracturing of simulated high-level waste glass in canisters. *Nucl. Eng. and Design* 67, 425-445.

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

2.2.2.1 Gaseous Release from Glass

Internal pressure within the canister is due to the accumulation of helium from alpha emission of transuranic nuclides. A DWPF canister filled with waste glass produces about 0.32 cm^3 of helium per year at 40 C. The helium produced is assumed to diffuse through the glass into the void space above the solid glass surface. At the end of 1,000 years the 103-liter void space pressure has increased by only 0.05 psi. This negligible pressure buildup is of no concern in waste package design. For the case of a canister filled to 25.3 ft³ (733 L), the 23-liter void space pressure would increase by 0.2 psi.

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

RATE LAW USED TO MODEL DISSOLUTION KINETICS

- RATE LAW FOR RELEASE OF ELEMENT η_i IN GLASS

$$\frac{dn_i}{dt} = v_i A k_r \left(1 - \frac{Q}{K} \right)$$

S/A	SURFACE AREA OF GLASS VOLUME
Q	ACTIVITY PRODUCT FOR DISSOLUTION REACTION
v_i	STOICHIOMETRIC FACTOR
k_r	RATE CONSTANT
K	EQUILIBRIUM CONSTANT FOR DISSOLUTION REACTION

- EXPERIMENTS MUST PROVIDE k_r AND K

REGRESSION OF THE Si-NORMALIZED GLASS DISSOLUTION RATES ($\text{g}/\text{m}^2 \cdot \text{d}$) VS. *IN SITU* pH AT 25°C, 50°C AND 70°C

Simple glass dissolution rate vs. pH
normalized to Si release

EXAMPLE 2: STATIC LEACHING/SURFACE ANALYTICAL STUDIES

SOLUTION ANALYSES INDICATE THE REACTION OF BOTH GLASSES SLOWS WITH TIME, BUT DOES THAT TELL THE WHOLE STORY?

131 GLASS

165 GLASS

NOTE: WHILE 131 GLASS REACTION IS FASTER THAN 165 GLASS, U RELEASE IS SIMILAR FOR BOTH GLASSES

Input for simple glass model

Based on years of glass dissolution testing, the rate of release of radionuclides from the glass wasteform can be simplified to expressions involving three environmental parameters, temperature, solution pH, and solution silica concentration. Although many other environmental parameters (such as redox state) affect glass dissolution rates, a relatively simple model that incorporates just these three effects can provide fairly accurate estimates of radionuclide release rates from glass.

This treatment does not provide for the sequestering of radionuclides by precipitation in secondary phases or sorption onto solid surfaces. These phenomena have not yet been investigated to the extent that they can be treated in a rigorous quantitative model. Because sorption and precipitation act to reduce the rates of release of radionuclides, the treatment presented here therefore represents a conservative calculation.

Although glass composition also affects the release rate, we are assuming here that a glass of approximately the same durability as an SRL-165 or SRL-202 glass will be produced (about 53 wt. % silica). Significant deviations from this target composition would necessitate a re-evaluation of the rate equations given below, but probably not the basic assumptions and methodology.

The rate of release of a radionuclide from a borosilicate waste glass can be calculated from the equation:

$$dn_i/dt = \nu_i A k_r (1 - Q/K) \quad (1)$$

where n_i is the number of moles of radionuclide i released from the glass per unit time t , ν_i is the concentration of i in the glass, A is the surface area of the glass, k_r is the rate constant for glass dissolution, Q is the concentration of silica in solution, and K is the silica solubility product for the glass. Values for each of the parameters in equation can be estimated using the following methods:

1. The concentrations of radionuclides in the glasses (ν_i) must be obtained from the glass producers for actual glasses at the time of production. Estimates for these concentrations are available in Oversby (1984) but better estimates periodically become available from the glass producers at Savannah River Laboratories.
2. The surface area of glass (A) in the DWPF canister of dimensions 61 centimeters diameter and 300 centimeters height is about 5 m². After the glass is cooled, however,

thermal contraction causes the glass to crack, effectively increases the available surface area for reaction with water. Baxter (1983) estimates that the cracking process increases the surface area by a factor of about 25 times. An appropriate surface area per canister is therefore 125 m².

3. The rate constant for glass dissolution (r_k) is a function of temperature and pH. Laboratory measurements of the dissolution rate of an analog glass of SRL-165 have been made from pH 1-12 and temperatures of 25-70 °C (Knauss et al., 1990). The data are given in Table 1. A function that provides the rate constant as a function of temperature and pH has been regressed from these data and has the form:

$$r_k = -0.00172029 - 0.0231246 * T + 0.00148569 * T^2 - 0.0000113605 * T^3 \\ - 1.1558 * pH + 0.0812918 * pH^2 + 0.000137686 * pH^3$$

where T is temperature in degrees Celsius. The equation is valid up to 100°C over a pH range of 2 to 12.

4. The model assumes the effect of solution composition is limited to the effect of dissolved silica only. Other species are known to affect dissolution, but silica has the biggest effect (Grambow, 1987; Bourcier, 1991). A simple way to estimate the silica concentration in a groundwater at a potential repository site is to make the justifiable assumption that it is controlled by reactions of the groundwater with the local minerals present at the repository. At sites such as Yucca Mountain where significant amounts of water are not expected, the rock-water system will be rock dominated. The chemistry of evolved groundwater that has been heated and reacted at the waste repository site will be dominated by the local rock type, and less affected by its previous history and reactions with repository materials. However, a better evaluation of repository performance should take into account these other interactions as well.

The potential repository at Yucca Mountain will be hosted by the Topopah Spring tuff. Experimental rock water interactions at elevated temperatures of samples of Topopah Springs tuff and J-13 well water from nearby have shown that the silica concentration appears to be controlled at saturation with respect to the silica polymorph cristobalite (Knauss et al., 1987). Therefore, for the value of Q in equation 1, we can use the value of the saturation concentration of silica with respect to the phase cristobalite. That value is given as a function of temperature in Table 2 (Walther and Helgeson, 1977).

5. The value of K , the thermodynamic solubility product for the glass, is a complex function of glass composition (Grambow, 1987). However, we can make a simple conservative estimate of its value by equating it to the solubility product of amorphous silica. Grambow (1987) has shown that empirically estimated values for K for a variety

Table 1. Dissolution Rate Constant for Glass

Temperature	pH	log rate (gm/m ² /day)
25	1	-1.25
25	2	-1.73
25	3	-2.21
25	4	-2.69
25	5	-3.17
25	7	-4.53
25	8	-4.02
25	9	-3.51
25	10	-3.0
25	12	-1.98
50	1	0.02
50	2	-0.68
50	3	-1.38
50	4	-2.08
50	5	-2.78
50	7	-3.43
50	8	-2.92
50	9	-2.41
50	10	-1.90
50	12	-0.88
70	1	0.51
70	2	-0.18
70	3	-0.87
70	4	-1.56
70	5	-2.25
70	6	-2.94
70	7	-2.3
70	8	-1.9
70	9	-1.5
70	10	-1.1
70	12	-0.3

of glass compositions are always less than but close to the value for amorphous silica. Because K corresponds physically to the silica-rich surface gel that forms on reacted glass surfaces, and it is extremely unlikely that this layer would ever be less stable than amorphous silica (because amorphous silica would then precipitate in the layer), we can use this assumption to provide values of K . These values are listed in column three of Table 2.

Table 2. Saturation Concentrations for Cristobalite and Amorphous Silica.

Temperature	Cristobalite	Amorphous Silica
	(log molal)	
0	-3.89	-2.99
25	-3.45	-2.71
60	-3.02	-2.43
90	-2.75	-2.26
100	-2.68	-2.20
150	-2.36	-1.98
200	-2.12	-1.80

Model Limitations

There are important limitations to the simple method for predicting borosilicate waste glass elemental releases presented here. The most important is that the method ignores all other solution chemistry other than solution pH and solution silica concentration. Some experiments have shown that glass dissolution rates can be orders of magnitude slower in the presence of certain cations such as magnesium (Barkatt et al., 1989). The effect may be due either to the poisoning of the glass surface by adsorbed magnesium or the precipitation of a magnesium-rich phase that armors the glass from further reaction. Other surface chemical reactions could increase the rate of reaction.

Another limitation of this model is that it does not account for the incorporation of radionuclides in secondary phases. Although experimental work has shown that radionuclides are incorporated into such phases (J. K. Bates, pers. com.), we cannot yet quantify the process and incorporate it into chemical models of dissolution.

The model also ignores any glass reaction in humid environments that may take place during shipping and storage of the glass before reaction with liquid water. Vapor phase hydration of glass in humid environments is known to occur (Diebold and Bates, 1986). The subsequent effects on glass reaction with liquid water present are currently under investigation.

Finally, the effects of other repository materials must be included in the model. Canister corrosion, the effects of backfill and other synthetic materials in the repository, and the radiation field must all be coupled to the glass dissolution model.

References

- Barkatt, A., E. Saad, R. Adiga, W. Sousanpour, Al. Barkatt, M. Adel-Hadadi, J. O'Keefe, and S. Alterescu (1989) Leaching of natural and nuclear waste glasses in sea water. *Applied Geochem.* 4:593-603.
- Baxter, R. G. (1983) Description of defense waste processing facility reference waste form and cannister. Savannah River Plant Report, DP-1606 rev. 1.
- Bourcier, W. L. (1991) Overview of chemical modeling of nuclear waste glass dissolution. *Mat. Res. Soc. Symp. Proc.* v. 212, p. 3-18
- Diebold, F. E. and J. K. Bates (1986) Glass-water vapor interaction. *Advances in Ceramics*, Vol. 20, Nuclear Waste Management II, p. 515-522.
- Grambow, B. (1987) Nuclear waste glass dissolution: mechanism, model and application. JSS Project Technical Report 87-02.
- Knauss, K. G., W. L. Bourcier, K. D. McKeegan, C. I. Merzbacher, S. N. Nguyen, F. J. Ryerson, D. K. Smith, H. C. Weed, and L. Newton (1990) Dissolution kinetics of a simple analogue nuclear waste glass as a function of pH, time and temperature. *Mat. Res. Soc. Symp. Proc.* v. 176, p. 371-381.
- Knauss, K. G., W. J. Beiriger, and D. W. Peiffer (1987) Hydrothermal interaction of solid wafers of Topopah Spring Tuff with J-13 water at 90 and 150°C using Dickson-type gold bag rocking autoclaves: long-term experiments. Lawrence Livermore National Laboratory UCRL-53722.
- Oversby, V. M. (1984) Reference waste forms and packing material for the Nevada Nuclear Waste Storage Investigations Project. Lawrence Livermore National Laboratory UCRL-53531, 26 p.
- Walther, J. V., and H. C. Helgeson (1977) Calculation of the thermodynamic properties of aqueous silica and the solubility of quartz and its polymorphs at high temperatures and pressures. *Amer. J. Sci.* 277:1315-135.

2.2 Glass Waste Form

2.2.1 Radionuclide Content

2.2.1.1 Present Inventory

2.2.1.2 Projected Inventory

2.2.1.3 Radioactivity and Decay Heat vs. Time

2.2.1.4 Glass Species Composition Statistics

2.2.1.5 Fracture/Fragmentation Statistics

2.2.2 Repository Response

2.2.2.1 Gaseous Release from Glass

2.2.2.2 Dissolution Radionuclide Release from Glass

2.2.2.3 Soluble-Precipitated/Colloidal Species

Table 1. Description, Purpose, and Status of Parametric Experiments and N2 and N3 Unsaturated Tests

Experiment #	Description	Purpose	Status (Tests in Progress)	Reaction Rate (g/m ² /d)
P-II	Regular-sized glass waste form, no ss holder, 0.075 mL J-13/ 3.5 days, continuous and batch experiments	To study the release from glass only	7 years	0.010
P-III	Half-sized glass waste form, ss holder, 0.075 mL and EJ-13/ 3.5 days, continuous and batch experiments	To study the effect of changing the waste form surface area by reducing the as-cast surface area by half	8.5 years	0.005
P-IV	Half-sized glass waste form, ss holder, 0.0375 mL and EJ-13/ 3.5 days, continuous and batch experiments	To study the effect of drop volume by reducing the amount of water added and the as-cast surface area by half	6 years	0.014
P-V	Regular-sized glass waste form, ss holder, 0.075 mL and EJ-13/ 14 days, continuous and batch experiments	To study the effect of lengthening the time interval between water additions	5.5 years	0.002
P-VII	Regular-sized glass waste forms in presensitized ss holders (heat #699980), 0.075 mL EJ-13/ 3.5 days, continuous and batch experiments	To study the effect of presensitized ss waste form holder	3.5 years	0.017
P-VIII	Regular-sized glass waste forms in presensitized ss holders (heat #22841), 0.075 mL EJ-13/ 3.5 days, continuous and batch experiments	To study the effect of presensitized ss waste form holder	5 years	0.003
Test #				
N2	Regular-sized glass waste forms in presensitized ss holders (heat #699980), 0.075 mL EJ-13/ 3.5 days, continuous and batch experiments	QA I execution of Unsaturated Test on SRL 105 glass doped with Np, Pu, and Am	5 years	0.008
N3	Regular-sized glass waste forms in presensitized ss holders (heat #699980), 0.075 mL EJ-13/ 3.5 days, continuous and batch experiments	QA I execution of Unsaturated Test on ATM-10 glass doped with Np, Pu, and Am	3.5 years	0.011

A.B. Woodland, J.K. Bates and T.J. Gerding, *Parametric Effects on Glass Reaction in the Unsaturated Test Method*, Argonne National Laboratory Report, in review.

Table 2. Estimated Forward and Final Rates Based on Static Leach Testing

Glass Type	Reference	Initial Rate g/(m ² ·d)	Final Rate (g/m ² ·d)
SRL 131	4, 5, 6	4.2	0.03
SRL 131 (Purex)	7	1.0	
SRL 131/11	8	0.05	
SRL 165	9	0.8	0.02
SRL 165/42	8	0.05	
SRL 202	7	0.2	
PNL 76-68	4,9	1.7	0.008, 0.08
JSS-A	4	1.5	0.003
EMS-11	9	0.08	0.002
SAN 60	10		0.2
SM 58	10		0.015

4. B. Grambow and D. M. Strachan, "A Comparison of Nuclear Waste Glasses by Modeling," Pacific Northwest Laboratory report PNL-6698 (1988).
5. J. Bates, D. Lam, and M. Steindler, Mater. Res. Soc. Symp. Proc. Vol. 15, 183 (1983).
6. G. G. Wicks, J. A. Stone, G. T. Chandler, and S. Williams, "Long-Term Leaching Behavior of Simulated Savannah River Plant Waste Glass. Part 1: MCC-1 Leachability Results, Four-Year Leaching Data," Savannah River Laboratory report DP-1728 (1986).
7. W. L. Ebert, Argonne National Laboratory, personal communication (1991).
8. J. K. Bates et al., "Unsaturated Glass Testing for DOE Program in Environmental Restoration and Waste Management, Annual Report, October 1989-September 1990," Argonne National Laboratory report ANL-90/40 (1991).
9. B. P. McGrail, D. M. Strachan, M. J. Apted, D. W. Engel, and P. W. Eslinger, "Preliminary Assessment of the Controlled Release of Radionuclides from Waste Packages Containing Borosilicate Waste Glass," PNL Draft Report (1990).
10. J. J. Hazer, Argonne National Laboratory, personal communication (1991).

Table 3. The Rate of Reaction of Glass in Vapor

Glass	Temp (°C)	Initial Rate (t^1)	Initial Rate ($t^{1/2}$)	Long-term rate	Surface finish	Reference
SRL 202U	200	0.3 $\mu\text{m/day}$	0.9 $\mu\text{m}/\sqrt{\text{day}}$	0.9 $\mu\text{m/day}$	600 grit	7
SRL 202U	200	1.25 $\mu\text{m/day}$	---		600 grit	7
SRL 202U	200	0.3 $\mu\text{m/day}$		5 $\mu\text{m/day}$	240 grit	7
SRL 202U	90	---	---	0.4 $\mu\text{m/day}$	---	extrapolated
SRL 165U	200	2.5 $\mu\text{m/day}$	8 $\mu\text{m}/\sqrt{\text{day}}$	7.5 $\mu\text{m/day}$	240 grit	7
SRL 165U	175	3.75 $\mu\text{m/day}$	10 $\mu\text{m}/\sqrt{\text{day}}$	7.5 - 15 $\mu\text{m/day}$	240 grit	7
SRL 165U	200	0.25 $\mu\text{m/day}$	0.9 $\mu\text{m}/\sqrt{\text{day}}$	0.4 - 1.6 $\mu\text{m/day}$	600 grit	7
SRL 165	187	---	---	5.8 $\mu\text{m}/\sqrt{\text{day}}$	600 grit	10
SRL 165U	175	0.6 $\mu\text{m/day}$	1.9 $\mu\text{m}/\sqrt{\text{day}}$	0.6 - 1.2 $\mu\text{m/day}$	600 grit	7
SRL 165U	125	0.2 $\mu\text{m/day}$	0.8 $\mu\text{m}/\sqrt{\text{day}}$	0.2 - 0.5 $\mu\text{m/day}$	600 grit	7
SRL 165U	90	---	---	0.2 $\mu\text{m/day}$	---	extrapolated
SRL 131	240	---	11.2 $\mu\text{m}/\sqrt{\text{day}}$	21 $\mu\text{m/day}$	600 grit	11
SRL 131	202	---	5.8 $\mu\text{m}/\sqrt{\text{day}}$	7.5 $\mu\text{m/day}$	600 grit	11
SRL 131	202	---	6.1 $\mu\text{m}/\sqrt{\text{day}}$	8 $\mu\text{m/day}$	600 grit	12
SRL 131	120	---	0.8 $\mu\text{m}/\sqrt{\text{day}}$		600 grit	11
SRL 131	75	---	---	0.009 $\mu\text{m/day}$	600 grit	11
SRL 131	90	---	0.32 $\mu\text{m}/\sqrt{\text{day}}$	0.03 $\mu\text{m/day}$	---	extrapolated

7. W. L. Ebert, Argonne National Laboratory, personal communication (1991).
10. J. J. Mazer, Argonne National Laboratory, personal communication (1991).
11. J. K. Bates, L. J. Jardine, and M. J. Steindler, "The Hydration of Nuclear Waste Glass: An Interim Report," Argonne National Laboratory report ANL-82-11 (1982).
12. T. A. Abrajano, Jr., J. K. Bates, and J. J. Mazer, "Aqueous Corrosion of Natural and Nuclear Waste Glasses. II. Mechanisms of Vapor Hydration of Nuclear Waste Glasses," J. Non-Cryst. solids 108, 269-288 (1989).

Figure 1. Cumulative Lithium Release from the PVII Tests

Figure 2. Arrhenius Plots for Vapor Hydration Reaction.
 (a) Final Rate. (b) Initial Rate.

7. W. L. Ebert, Argonne National Laboratory, personal communication (1991).
10. J. J. Mazer, Argonne National Laboratory, personal communication (1991).
11. J. K. Bates, L. J. Jardine, and M. J. Steindler, "The Hydration of Nuclear Waste Glass: An Interim Report," Argonne National Laboratory report ANL-82-11 (1982).
12. T. A. Abrajano, Jr., J. K. Bates, and J. J. Mazer, "Aqueous Corrosion of Natural and Nuclear Waste Glasses. II. Mechanisms of Vapor Hydration of Nuclear Waste Glasses," J. Non-Cryst. solids 108, 269-288 (1989).

Table 2. Statistical Analysis of Triplicate Leach Results for Gamma Irradiated SRP 165 Glass^a. Taken from Bibler¹⁵

Dose, gray	Average Concentration (ppm), Standard Deviation and Student T Value											
	B			Si			Na			Li		
	Aver.	Dev.	T ^b	Aver.	Dev.	T ^b	Aver.	Dev.	T ^b	Aver.	Dev.	T ^b
Unirrad.	15.3	0.15		109.8	0.52		56.4	0.71		17.8	0.23	
4.0 x 10 ⁶	14.6	0.34	-3.1	104.7	3.1	-2.8	55.2	1.2	-1.6	17.4	0.36	-1.9
4.7 x 10 ⁷	15.2	0.06	-1.1	108.1	0.42	-4.4	56.9	0.11	1.1	18.4	0.03	4.0
3.1 x 10 ⁸	15.5	0.37	0.8	110.1	4.0	0.14	57.9	1.5	1.9	18.4	0.19	3.0

^aProduct Consistent Test: 1.3 grams 100-200 mesh glass leached for 7 days in 13 mL of deionized water at 90°C.

^bT value for one-sided Student's T test. If T > +2.1, then the data are precise enough to establish to the 95% confidence level that the irradiated glass is leaching greater than the unirradiated glass.

Figure 5. Generation of Nitric Acid in Thin Water Films Due to (a) Gamma Radiation and (b) Alpha Radiation

Figure 6. Normalized Boron Mass Loss from SRL U Glass vs. the Square Root of the Reaction Time for EJ-13 Plus SRL U Glass: (a) irradiated, with tuff (▲) or without tuff (■); (b) nonirradiated, without tuff (▼) or with tuff (●). Taken from Ebert.²⁶

2.3 Special Cases Waste Forms

2.3.1 Damaged Spent Fuel

2.3.2 Non-LWR Spent Fuel

2.3 Special Cases Waste Forms

Waste Forms that may require special handling or may require special processing or containerizing.

2.3 Special Cases Waste Forms

2.3.1 Damaged Spent Fuel

2.3.2 Non-LWR Spent Fuel

TABLE J-11.1

TYPICAL FUEL ASSEMBLY PARAMETERS*

VENDOR	B&W	B&W	C-E	C-E	W	W	W	ENC	ENC	GE	GE	GE
Fuel Rod Array	15x15	17x17	14x14	16x16	14x14	15x15	17x17	15x15	8x8	7x7	8x8	8x8 R
Reactor Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR
Assemblies per Core	177	205	217	177	121	193	193	193	560	764	560	560
Fuel Rod Locations Per Assembly	225	289	196	256	196	225	289	225	64	49	64	64
Fuel Rods Per Assembly	208	264	176	236	179	204	264	204	60	49	63	62
Empty Locations Per Assembly	17	25	5	5	17	21	25	21	4	NONE	1	2
Rod Pitch, mm (in.)	14.4 (0.568)	12.7 (0.501)	14.7 (0.580)	12.9 (0.5063)	14.1 (0.556)	14.3 (0.563)	12.6 (0.496)	14.3 (0.563)	16.3 (0.642)	18.7 (0.738)	16.3 (0.640)	16.3 (0.640)
System Pressure, MPa (psia)	15.2 (2200)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	7.14 (1035)	7.14 (1035)	7.14 (1035)	7.14 (1035)
Core Average Power Density, kW/liter	91.4	107.3	78.5	96.4	95.6	98.1	104.7	98.1	40.57	50.732	50.51	49.15
Average LHGR, kW/M (kW/ft)	20.3 (6.20)	18.8 (5.73)	20.0 (6.09)	17.5 (5.34)	20.3 (6.20)	22.0 (6.70)	17.8 (5.44)	22.0 (6.70)	15.2 (4.63)	23.1 (7.049)	17.9 (5.45)	17.7 (5.38)
Axial Peak LHGR, in an Average Rod, W/M (kW/ft)	24.41 (7.44)	22.57 (6.88)	24.00 (7.31)	21.00 (6.41)	24.36 (7.44)	26.40 (8.04)	21.36 (6.53)	26.40 (8.04)	18.24 (6.02)	27.72 (9.16)	21.48 (7.09)	21.24 (6.99)
Max. Peak LHGR, kW/M (kW/ft)	53.0 (16.16)	49.9 (15.20)	53.5 (16.3)	42.7 (13.0)	56.8 (17.3)	61.7 (18.8)	44.6 (13.6)	51.9 (15.83)	47.6 (14.5)	60.2 (18.35)	44.0 (13.4)	44.0 (13.4)
Max. Fuel Temp., °C (°F)	2340 (4245)	2090 (4155)	2140 (3890)	1880 (3420)	2260 (4100)	2340 (4250)	1870 (3400)	2200 (3997)	2040 (3700)	2440 (4430)	1830 (3325)	1890 (3435)
Core Average Enrichment, wt% ²³⁵ U	3.00	2.67	2.35	2.36	2.90	2.80	2.60	3.02	2.65	2.19	1.80	1.99
Max. Local Exposure, MWd/MTU GJ/kgU	55,000 4752	55,000 4752	50,000 4320	55,000 4752	50,000 4320	50,000 4320	50,000 4320	47,500 4104	35,000 3024	40,000 3456	40,000 3456	45,000 3888
Cladding Material	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-2	Zry-2	Zry-2	Zry-2

TABLE J-11.1 (Continued)

VENDOR	Fuel Rod Length, m (in.)	Active Fuel Height, m (in.)	Plenum Length, m (in.)	Fuel Rod OD, mm (in.)	Cladding ID, mm (in.)	Cladding Thickness, mm (in.)	Diameter Cap, microns (mil)	Fuel Pellet Diameter, mm (in.)	Fuel Pellet Length, mm (in.)	Fuel Pellet Density, STD
B&W	3.904 (153.688)	3.602 (141.8)	0.298 (11.27)	9.627 (0.430)	9.576 (0.377)	0.673 (0.0265)	213.4 (8.4)	9.362 (0.3686)	15.240 (0.600)	95
B&W	3.864 (152.125)	3.632 (143)	0.242 (9.52)	9.627 (0.379)	9.407 (0.331)	0.610 (0.0240)	198.1 (7.8)	8.209 (0.3232)	9.525 (0.375)	95
C-E	3.71 (145.9)	3.47 (136.7)	0.22 (8.6)	11.18 (0.440)	9.66 (0.388)	0.660 (0.026)	216 (8.5)	9.64 (0.3795)	16.51 (0.650)	94.75
C-E	4.09 (161.02)	3.81 (150)	0.25 (10.00)	9.70 (0.382)	8.43 (0.332)	0.635 (0.025)	178 (7.0)	8.26 (0.325)	9.91 (0.390)	95
H	3.87 (152.36)	3.66 (144)	0.18 (6.99)	10.72 (0.422)	9.48 (0.3734)	0.617 (0.0243)	190 (7.5)	9.29 (0.3659)	15.24 (0.600)	94
H	3.80 (149.7)	3.66 (144)	0.21 (8.2)	10.72 (0.422)	9.48 (0.3734)	0.617 (0.0243)	190 (7.5)	9.29 (0.3659)	15.24 (0.600)	95
H	3.85 (151.6)	3.65 (143.7)	0.16 (6.3)	9.50 (0.374)	8.16 (0.329)	0.572 (0.0225)	165 (6.5)	8.19 (0.3225)	13.46 (0.530)	95
ENC	3.86 (152.0)	3.66 (144)	0.17 (6.8)	10.77 (0.424)	9.25 (0.364)	0.762 (0.030)	190 (7.5)	9.06 (0.3565)	6.93 (0.273)	94
ENC	3.99 (156.92)	3.66 (144)	0.27 (10.63)	12.74 (0.5015)	10.91 (0.4295)	0.914 (0.036)	254 (10.0)	10.66 (0.4195)	8.13 (0.320)	95
GE	4.09 (161.1)	3.66 (144)	0.41 (16.0)	14.30 (0.563)	12.68 (0.499)	0.813 (0.032)	305 (12.0)	12.37 (0.487)	12.70 (0.500)	95
GE	4.08 (161.1)	3.71 (146)	0.36 (14.0)	12.52 (0.493)	10.80 (0.425)	0.864 (0.034)	229 (9.0)	10.57 (0.416)	10.67 (0.420)	95
GE	4.20 (165.4)	3.81 (150)	0.25 (10.0)	12.27 (0.483)	10.64 (0.419)	0.813 (0.032)	229 (9.0)	10.41 (0.410)	10.41 (0.410)	95

2.3 Special Cases Waste Forms

2.3.1 Damaged Spent Fuel

2.3.2 Non-LWR Spent Fuel

Table 7-1. Number of research and test reactors in each fuel type category

Fuel type	University/ educational	Private research and test	Government- owned (DOE)	Government- owned (non-DOE)
MTR-plate type, U-Al alloy, high enrichment	15	2	14	1
TRIGA (U-Zr hydride fuel)	16	4	2	3
UO ₂ -polyethylene disks or blocks	4	0	0	0
PULSTAR and other low-enriched pin type	3	2	0	0
Liquid fuels (aqueous solutions)	0	0	1	0
U-Mo alloy, high-enriched (93.2%)	0	0	4	2
FFTF (UO ₂ -PuO ₂)	0	0	1	0
Miscellaneous	0	0	26	0
Totals	38	6	48	6

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

Table 7-2. Summary of non-LWR spent fuels

Reactor or site	Estimated quantities		
	End of 1989	Annual rate	End of 2020
HTGR Reactors			
Fort St. Vrain (elements)	732	0 ^a	2214 ^b
Peach Bottom I			
Core I (elements)	819	0	819
Core II (elements)	820	0	820
Research and Test Reactors^c			
MTR Plate	--	-	20,000 ^d
TRIGA	--	-	4,500
UO ₂ /Polyethylene	--	-	87
PULSTAR	--	-	170
FFTF (assemblies)	170	30-45	677 ^e
Miscellaneous (kg HM)^g			
ANL West	311		
Babcock & Wilcox	88		
Battelle-PNL	2,348		
HEDL	263 ^h		
INEL	39,508 ⁱ		
LANL	38		
ORNL	1,254		
SRS	19,110		

^aReactor was shut down in 1989. No further refueling is expected.

^bIncludes final discharge of full core.

^cTotal through 2020, including fuels in reactors at that time. Quantities shown are numbers of individual fuel elements, except for the FFTF.

^dThis is expected to be reprocessed and disposed of as defense HLW.

^eThrough year 2003; does not include final core discharge.

^gReported as kg of heavy metal (U plus Pu plus Th). Data are from Integrated Data Base for 1990.

^hIncludes some FFTF and TRIGA fuels.

ⁱNot including Shippingport LWBR fuel (982 kg U, mostly U-233, and 56,167 kg Th), 17 Turkey Point 3 assemblies and 69 VEPCO assemblies being used for dry consolidation testing, HTGR fuel, Pulstar and TRIGA fuel, and TMI-2 spent fuel and core debris.

Table 7-5. Estimated number of canisters required for repository disposal of various non-LWR and special LWR spent fuels^a

	<u>Number of fuel assemblies</u>		Estimated fuel assemblies per canister	Estimated number of canisters required
	In storage, 1988	Total as of year 2020 ^b		
<u>24-in. diam x 12 ft canisters</u>				
Fort St. Vrain	732	2214	4	554
Peach Bottom-1	1639	1639	12	138
TRIGA	800	4500	112	40
PULSTAR	24	170	48	4
CEUSP material	401 ^c	401 ^c	24	17
Fermi-1 blanket	510	510	12	43
Elk River	188	188	12	16
EBWR	300	300	24	13
Canned fuel at B&W	58 ^d	58 ^d	24	3
Saxton	14 ^e	14 ^e		4
Other	-	-	-	60 ^f
Total	4,666	9,994		892
<u>28-in. diam x 15 ft canisters</u>				
VEPCO	69	69	48	15
Turkey Point	20	20	4	5
Dresden 1	20	20	8	3
Shippingport LWBR	65	65	1	65
TMI-2 (estimated)	-	-		350
Other	-	-	-	40 ^f
Total	174	174		478
Total number of canisters				1,370

^aOnly the major non-LWR and special LWR fuels are listed. An allowance is included for minor fuels not specifically listed.

^bReloads and in-core fuel are included in totals.

^cCEUSP material is stored in 3.5-in. diameter x 24 in. cans; numbers shown are numbers of cans.

^dThere are 58 cans of LWR fuel at B&W, Lynchburg. Cans are 4.25-in. diameter x 33 in. long.

^eQuantities shown are numbers of cans.

^fAn allowance is included here for fuels not specifically listed.

^gSome of these assemblies have been compacted.

Table 7-6. Radioactivity and decay heat of Fort St. Vrain spent fuel per MTIHM

Time after discharge, years	Radioactivity (Ci/MTIHM)	Decay heat (W/MTIHM)
1	3.36E06	1.0E04
10	9.82E05	2.5E03
100	1.17E05	4.0E02
1,000	4.42E02	1.5E01
10,000	1.22E03	3.0E01
100,000	1.40E03	3.0E01
1,000,000	6.17E01	2.0E00

Table 7-7. Estimated radioactivity and decay heat per canister of Fort. St. Vrain fuel

Time after discharge, years	Radioactivity (Ci/canister)	Decay heat W/canister
1	75,600	225
10	22,000	56
100	2,600	9
1,000	10	0.3
10,000	27	0.7
100,000	32	0.7
1,000,000	1.4	0.05

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

Table 7-8. Estimated radioactivity and decay heat per canister of FLIP TRIGA fuel

Time after discharge (years)	Radioactivity (Ci/canister)	Decay heat (W/canister)
1	9.0E04	380
5	2.0E04	100
10	1.5E04	80
100	1.6E03	40
1,000	68	2
10,000	9	0.5
100,000	1	0.04
1,000,000	0.16	0.02

Table 7-9. Estimated radioactivity and decay heat per canister of PULSTAR fuel

Time after discharge (years)	Radioactivity (Ci/canister)	Decay heat (W/canister)
1	1.7E05	750
10	3.0E04	82
100	3.0E03	21
1,000	140	4
10,000	35	1
100,000	4	0.08
1,000,000	0.4	0.04

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

Table 7-10. Projected volumes of miscellaneous wastes^a

	Estimated total in 2020 (m ³)	Est. annual rate in 2020 (m ³)
OCRWM-generated TRU waste	600 ^b	60-260 ^c
Commercial TRU waste		
West Valley decommissioning	300	0
Other decommissioning	680	TBD
Abnormal reactor operations	70 ^d	10-30
Industrial/institutional	TBD	10-40
Reactor decommissioning	15060 ^e	46 ^f
Radioisotope capsules	500 ^g	0
Routine reactor operations ^h	8,000	15 ^b
Totals	11,710	282-532

^aData are given in m³. One 2-ft by 12-ft canister holds about 1 m³. "TBD" means to be determined.

^bDepends on startup date for these facilities; 2010 was assumed.

^cFrom dry rod consolidation. The upper limit is a conservative (high) estimate of HEPA filter usage.

^dQuantity estimated based on two abnormal reactor operations (at Oyster Creek and TMI-2).

^eAssumes 65 have been decommissioned.

^fAssumes 2 per year.

^gAssumes that 90% of existing capsules are packages in canisters by 1995; later packaging would result in fewer canisters because of the decreased thermal output per capsule.

^hBased on estimated quantity of 3 m³ per GW(e)-yr being GTCC, and an EIA projection of 52 GW(e) installed capacity in 2020 (no new orders case).

Table 7-11. Volumes and Activities of Decommissioned LWR
Activated Metals^a

Component	Constr. ^b Material	Disposal Volume ^c (m ³)	Activity (Ci)	Disposal Concentration (Ci/m ³)
<u>Reference BWR:</u>				
Steam separator assembly	S	10	9,600	960
Fuel support pieces	S	5	700	140
Control rods and in-core instruments	S	15	189,000	12,600
Control rod guide tubes	S	4	100	25
Jet pump assemblies	S	14	20,000	1,429
Top fuel guide	S	24	30,000	1,254
Core support plate	S	11	650	59
Core shroud	S	47	6,300,000	134,043
Reactor vessel wall	C	8	2,160	270
Total		138	6,552,310	
<u>Reference PWR:</u>				
Pressure vessel cylindrical wall	C	108	19,170	178
Vessel head	C	57	<10	0.18
Vessel bottom	C	57	<10	0.18
Upper core support assembly	S	11	<10	0.91
Upper support columns	S	11	<100	9.1
Upper core barrel	S	6	<1,000	167
Upper core grid plate	S	14	24,310	1,736
Guide tubes	S	17	<100	6
Lower core barrel	S	91	651,000	7,154
Thermal shields	S	17	146,000	8,594
Core shroud	S	11	3,431,100	311,909
Lower grid plate	S	14	553,400	39,529
Lower support columns	S	3	10,000	333
Lower core forging	S	31	2,500	81
Miscellaneous internals	S	23	2,000	87
Reactor cavity liner	S	14	<10	0.7
Total		485	4,840,820	

a Source: Oztunali 1986.

b Construction material symbols: S - stainless steel, C - carbon steel.

c Disposal volumes include the disposal container after the activated metal components have been cut into manageable pieces.

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

Table 7-12. Radioactivity and thermal power of canisters within strontium and cesium capsules^a

Decay time (years)	Strontium canister (4 capsules)		Cesium canister (4 capsules)	
	Curies	Watts	Curies	Watts
0	412,400	1,380	371,000	918
5	366,300	1,230	330,000	817
10	325,200	1,090	294,000	727
20	256,300	860	233,000	577
50	125,500	420	117,000	290
100	38,200	128	36,800	91
200	3,530	12	3,650	9
300	327	1.1	360	0.9
1,000	1.9E-05	6.4E-08	3.4E-05	8.4E-08

^aBased on ORIGEN2 calculations. Radioactivity and thermal power include the contributions of the daughter isotopes Y-90 and Ba-137M. Starting point for decay time is December 1985. The assumed thermal limits at a decay time of 10 years are 1,170 W/canister for Sr capsules and 800 W/canister for Cs capsules (Coony 1987).

Table 8-1 Average Properties of LWR Spent Fuel

BWR Spent Fuel Average Properties:	Historical	Projected	Total
Burnup (GWd/MTU)	21	33	30
Enrichment (%)	2.3	3.2	3.0
Discharged (year)	1981	2007	2001
Thermal Power (W/MT)			
in 2010 (over 5 years old)	--	--	
in 2020 (over 5 years old)	--	--	
in 2050 (all fuel)	--	--	
PWR Spent Fuel Average Properties:	Historical	Projected	Total
Burnup (GWd/MTU)	29	42	39
Enrichment (%)	2.9	3.9	3.7
Discharged (year)	1982	2007	2002
Thermal Power (W/MT)			
in 2010 (over 5 years old)	--	--	
in 2020 (over 5 years old)	--	--	
in 2050 (all fuel)	--	--	

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

Major Contributors	Est. Canisters
HTGRs	
Fort St. Vrain	554
Peach Bottom-1	138
Degraded LWR Fuel	
TMI-2	350
Other Contributors	
Shippingport LWBR	65
Fermi-1 Blanket	43
TRIGA	40
All Others	245

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

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.1 Spent Fuel Cladding Failure

The Zircaloy cladding that already exists on the fuel rod may be an important barrier that contributes in a performance evaluation of waste forms to their ability to meet the radionuclide release requirement. Several potential mechanisms of Zircaloy cladding degradation during dry storage have been identified, such as hydride reorientation, stress corrosion cracking, creep, and creep fracture.

At the present time three distinct time periods can be identified during repository storage:

1. A high-temperature period (above 250°C) during which the container is probably unbreached, the fuel rods are surrounded by inert gas or air, and no liquid water is present.
2. An intermediate-temperature period (250° to 100°C) when the container is probably unbreached and liquid water (from breached fuel rods containing water) may be present in contact with the fuel rods (90 to 1000 years).
3. A lower-temperature period (below 100°C) when the container may be breached and air, water vapor, and liquid water may be in contact with the spent fuel rods.

Cladding Failure Model

Model equations. LLNL is currently formulating quasi-static rate of displacement and rate of stress equations for a Zircaloy tube with an adjacent thin zirconium oxide film. The equations include contributions from elastic, creep, thermal, and hydride precipitation dependent strains. Given initial conditions and expected repository environmental histories to derive boundary conditions, we will integrate the set of rate equations to assess cladding failure. Our simplest and most conservative modeling assumption, with respect to cladding failure, is a stress or strain limit at which the zirconium oxide film fractures. This failure modeling concept is motivated because of the known large volume change that occurs when zirconium in the Zircaloy is oxidized to zirconium oxide. Hence, the oxide film is expected to remain in a compressive state of hoop stress for the expected large number of fuel rods that initially have low fission gas releases (<1%). This compressive hoop stress will prevent stress corrosion cracking from being initiated. In addition to the strain contribution of hydride precipitate, another potential mechanism of cladding failure is fluoride-Zircaloy corrosion, which is not stress dependent. The available data for fluoride-Zircaloy corrosion are preliminary, but suggest a pin-hole pitting uniformly distributed on the surface.

There have been no activities to evaluate stainless steel cladding failure response. The amount of stainless steel cladding is very small compared to the amount of Zircaloy cladding.

Measured quantities. Initial conditions are required for a rate displacement-stress formulation; thus, measurements for the initial dimensions and state of stress of the Zircaloy and zirconium oxide film are necessary to characterize the cladding. The boundary condition for a rate displacement-stress formulation of the cladding are the inside and outside pressure histories. This means that the fission gas content (released from UO_2 spent fuel matrix) inside the cladding is required for the fuel rods. This data will also have a statistical character because the different UO_2 fuels and burnup cycles may result in different amounts of fission gas content in the fuel rods. We would like to see this expressed as a probabilistic density function $f(g,b,t)$, where the density function, f , is the number of fuel rods per unit fission gas content per unit burnup with fission gas content, g , and burnup, b , at time, t . At $t=0$, this density function would characterize the initial distribution of fission gas content in fuel rods emplaced in the repository. At later times, the fission gas content may increase because of helium produced due to decaying actinides. Thus, to predict the expected stress state in the Zircaloy and in the oxide film, the expected fission gas released from the UO_2 matrix is a required initial condition that must be measured.

Considerable information on material properties for the elastic strain, creep strain, thermal strain, and fracture responses of Zircaloy cladding is available from reports and analysis for the Dry Storage Spent Fuel Program, reactor design documents, and the open literature. Some of this data may require additional confirmation tests for purposes of QA Level I input to models and analysis. Much of the data is not particularly useful because of the relatively low temperatures expected in a repository compared to in-reactor temperatures and the tensile hoop stress state expected in a repository compared to in-reactor compressive hoop stress state. Also, some testing to establish material properties of zirconium oxide failure is anticipated; again, this may be confirmation tests. The problem of hydride precipitation strains will require measurement of the initial concentrations of hydrogen as well as low temperature hydride platelet orientation statistics in each class of Zircaloy cladding. Additional testing and model development for hydride precipitation and re-orientation and its associated dependence on the state of stress are currently being planned. The initial hydrogen content data may have a statistical character similar to the fission gas content data. Hence, we are interested in it being expressed as a probabilistic density function, $h(H, a_r, a_\theta, b)$, where the density function, h , is the number of fuel rods per unit hydrogen content per unit size in the radial direction per unit size in the theta direction per unit burnup with hydrogen content H , a_r length of hydride platelets in the radial direction, a_θ length of hydride platelets oriented in the theta (hoop) direction, and burnup, b . Note that time is not a variable here as we do not anticipate additional hydrogen pickup by the cladding in the low temperature environment of the repository.

With this statistical information on initial hydrogen content and low temperature orientation, a time-dependent model is being planned to predict the precipitation kinetics of hydride platelets and the effect of stress on the hydride

platelet orientation as the repository temperature decreases. The strain contribution from hydride platelet precipitation is a required part of the model development for assessing probable cladding failure rate.

Finally, the proposed quasi-static displacement-stress rate model assumes that the Zircaloy cladding and its adjacent zirconium oxide film are initially pristine. This assumption must be supported as part of the MCC characterization of spent fuel rods. Certainly there will be a statistical character to the initial qualities of the fuel rods with respect to defects and surface flaws. We will need to have data on flaw size, flaw surface density, etc. We plan to support tests that will subject flawed and defected cladding to temperature and stress states that will provide failure data for additional failure rate models for the number of fuel rods that are not initially pristine.

The following list of references address the progress that has been accomplished in testing, modeling, and understanding the complexities of spent fuel cladding failure response and a range of environmental conditions that may need to be addressed in the design of a geological repository.

Ardell, A.J. "On the Calculation of Melting Temperatures for Low-Temperature Phases of Polymorphic Metals," *Acta Metall.*, 11, 591-594 (June, 1963).

Blackburn, L.D., et al. "Maximum Allowable Temperature for Storage of Spent Nuclear Fuel: An Interim Report," HEDL-TME 78-37, UC-70 (1978).

Chen, I.-W. "Implications of Transformation Plasticity in ZrO_2 -Containing Ceramics: II, Elastic-Plastic Indentation," *J. Am. Ceram. Soc.* 69(3), 192 (1986).

Christian, J.W. "Phase Transformations," in *Physical Metallurgy*, R.W. Chan, ed., North-Holland Pub. Co., Ch. 10, pp. 443 (1965).

deGroot, S.R. "Thermodynamics of Irreversible Processes," North-Holland Pub. Co. Amsterdam (1957).

Dieter, G.E. *Mechanical Metallurgy*, second edition, McGraw Hill (1976).

Einzig, E.R., D.M. Bozi, and A.K. Miller. "Transactions," *Waste Form Development and Processing*, American Nuclear Soc. p. 131 (1980).

Einzig, R.E. and R. Kohli. "Low-Temperature Rupture Behavior of Zircaloy-Clad Pressurized Water Reactor Spent Fuel Rods Under Dry Storage Conditions," *Nuc. Tech.*, 67, 107 (1984).

Ells, C.E. "Hydride Precipitates in Zirconium Alloys," *N. Nuc. Mat.*, 28, 129 (1968).

Ells, C.E. "The Stress Orientation of Hydride in Zirconium Alloys," J. Nuc. Mat., 34, 306 (1970).

Eringen, A.C. "Mechanics of Continua," John Wiley Pub. New York (1967).

Farwick, D.G. and R.A. Moen. "Properties of Light Water Reactor Spent Fuel Cladding," Westinghouse Hanford Report HEDL-TME 78-70 (Aug. 1979).

Gibbs, J. "The Scientific Papers of J. Willard Gibbs," Vol. 1, Dover Pub. Inc., New York (1961).

Hardy, H.K. and T.J. Heal. "Report on Precipitation," in Progress in Metal Physics, 5, 143, 2nd printing, Pergamon Press (1961).

Khan, M.A., N.H. Madsen, and B.A. Chin. "Fracture Predictions in Zircaloy Fuel Cladding: Effects of Radiation on Materials," Twelfth International Symposium, ASTM Stp 870, Philadelphia, pp. 642-655 (1985).

Leger, M. and A. Donner. "The Effect of Stress on Orientation of Hydrides in Zirconium Alloy Pressure Tube Materials," Canadian Meta. Qu., 24, 235 (1984).

Lupis, C.H.P. "Chemical Thermodynamics of Materials," Elsevier Science Pub., New York (1983).

Lustman, B. and F. Kerze, Jr., eds. "The Metallurgy of Zirconium," McGraw Hill, first edition, p. 354 (1955).

Miller, A.K. "Application of the SCCIG Model to Dry Storage of Spent Fuel," Workshop on Spent Fuel Integrity in Dry Storage, Seattle, WA, (Jan 20, 1982).

Onsager, L. "Reciprocal Relations in Irreversible Processes; I and II," Phys. Rev. 37 405, and 38 2265 (1931).

Ostberg, G. "Crack Propagation in Hydrided Zircaloy-2," Int. J. Fract. Mech., 4 95 (1968).

Parfenov, B.G., V.V. Gerasimov, and G.I. Venddiktova. "Corrosion of Zirconium and Zirconium Alloys," (translated from Russian), Israel Program for Scientific Translations, Jerusalem, p. 87 (1969).

Pearson, W.B. "Handbook of Lattice Spacings and Structures of Metals and Alloys," 1, 130.

Pescatore, C., M.G. Cowgill, and T.M. Sullivan. "Zircaloy Cladding Performance Under Spent Fuel Disposal Conditions," Brookhaven National Laboratory Report BNL-52235 (Progress Report May 1 - October 31, 1989).

Rothman, A.J. "Potential Corrosion and Degradation Mechanisms of Zircaloy Cladding on Spent Nuclear Fuel in a Tuff Repository," UCID-20172 (Sept. 1984).

Sawatzky, A. J. Nucl. Mat, 2,321 (1960).

Smith, C.F. and W.B. Crandall, J. Amer. Ceramic Soc., 47(12), 624-627 (Dec., 1964).

Smith, H.D. "Initial Report on Stress Corrosion Cracking Experiments Using Zircaloy-4 Spent Fuel Cladding Rings," Westinghouse Hanford Company, Richland, WA.

Smith, H.D. "Zircaloy Cladding Corrosion Degradation in a Tuff Repository," HEDL-7455, Rev. 1, p. 12 (July, 1985).

Stehle, H., F. Garzarolli, A.M. Garde, and P.G. Smerde. "Zirconium in the Nuclear Industry," Sixth International Symposium, ASTM STP 824, pp. 483-506 (1984).

Stout, R.B. "A Deformation and Thermodynamic Model for Hydride Precipitation Kinetics in Spent Fuel Cladding," Lawrence Livermore National Laboratory, UCRL-100860, (submitted to Symp. on Scientific Basis for Nuclear Waste Management XII, Boston MA, Nov. 27-30, 1989).

Stout, R.B. "Deformation and Thermodynamic Response for a Crack Dislocation Model of Brittle Fracture," Eng. Fract. Mech., 19, 545 (1984).

Stout, R.B. "Modeling the Deformations and Thermodynamics for Materials Involving a Dislocation Kinetics," Cryst. Latt. Def. 9, 65 (1981).

Yaggee, F.L., R.F. Mattas, and L.A. Neimark. "Characterization of Irradiated Zircaloys" Susceptibility to Stress Corrosion Cracking," NP-1557, Research Project 1027, prepared by Argonne National Laboratory for EPRI, pp. 4-27, Fig. 4-7 (Oct., 1980).

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

OBSERVED RELATIONSHIP: $\log t_f$ (TIME TO FAILURE) vs. STRESS (% OF YIELD STRESS)

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

Fig. 2. Effect of temperature on the tensile properties of H.B. Robinson spent fuel cladding. Strain rate = 0.025 / min. A.A. Bauer, L.M. Lowry, "Tensile Properties and Annealing Characteristics of H.B. Robinson Spent Fuel Cladding," Nuclear Technology, Vol. 41, Dec. 1978, pp. 359-372.

Fig. 3. Effect of strain rate on the tensile properties of H.B. Robinson spent fuel cladding. Test temperature was 371°C. A.A. Bauer, L.M. Lowry, "Tensile Properties and Annealing Characteristics of H.B. Robinson Spent Fuel Cladding," Nuclear Technology, Vol. 41, Dec. 1978, pp. 359-372.

Fig. 4. Response of spent fuel cladding to transient annealing. Test temperature was 371°C. (a) Yield strength and uniform elongation, and (b) ultimate strength and total elongation.
 A.A. Bauer, L.M. Lowry, "Tensile Properties and Annealing Characteristics of H.B. Robinson Spent Fuel Cladding," Nuclear Technology, Vol. 41, Dec. 1978, pp. 359-372.

Fig. 5. Response of spent fuel cladding to isothermal annealing. Test temperature was 371°C. (a) Yield strength and uniform elongation, and (b) ultimate strength and total elongation.

A.A. Bauer, L.M. Lowry, "Tensile Properties and Annealing Characteristics of H.B. Robinson Spent Fuel Cladding," Nuclear Technology, Vol. 41, Dec. 1978, pp. 359-372.

TABLE A1
RATES AND OXIDATION DEPTHS OCCURRING IN THE
LOW-TEMPERATURE CORROSION OF ZIRCALOY CLADDING

Temperature (°C)	Corrosion Rate (mg/dm ² ·day)		Oxidation Depth (μm)*	
	Eq. 1	Eq. 2	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.

METHOD FOR CORRELATING MAXIMUM OXIDE THICKNESS WITH BURNUP

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.1.2 Failure Models

Both the experimental testing and the model development activities for cladding failure response are incomplete. At this time only the report by L. Santanam, H. Shaw, and B.A. Chin (Modeling of Zircaloy Cladding Degradation Under Repository Conditions, Lawrence Livermore National Laboratory Report UCRL-100211, July, 1989) contains a preliminary analysis for Zircaloy cladding failure during the high temperature time period. The analysis used a deformation and fracture map methodology that is an extension of the methods applied to analyze Zircaloy cladding deformation and fracture for the dry storage of spent fuel waste form program established by the nuclear utility industry. Substantial testing and model development activities remain to be completed to support creditable repository design that would include the cladding as a long term barrier to the UO_2 spent fuel waste form. It is believed, however, that the Zircaloy-clad fuel rods that have low internal pressures will have low failure rate through both the high temperature and intermediate temperature periods.

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.2 Spent Fuel Oxidation

In any proposed nuclear waste repository for spent fuel from nuclear power reactors, the potential release rates of many radionuclides over a 10,000 year design lifetime depends on the oxidation rate and the oxidation state of any irradiated UO_2 fuel pellets that may be exposed to the atmosphere. This is because UO_2 spent fuel can oxidize into $\text{U}_4\text{O}_9 \rightarrow \text{U}_3\text{O}_8 \rightarrow \text{UO}_3$, and possibly other oxides, which could influence the surface area and the dissolution rates of any spent fuel that may be exposed to water in a repository. Therefore, experiments to provide both data and a physical basis for rational model development for UO_2 oxidation kinetics are necessary in order to eventually predict potential radionuclide release rates from spent fuel in a repository.

Results from tests at this time imply that the grain boundaries of irradiated UO_2 oxidized more rapidly than the grain volumes at low temperatures (less than 200°C). This two-rate oxidation process of UO_2 spent fuel is difficult to represent mathematically because classical diffusion models with their associated classical initial conditions and external boundary conditions do not physically describe certain geometrical aspects of the experimental observations.

Furthermore, the vast majority of the spent fuel rods placed in the repository will have intact Zircaloy cladding, but approximately 0.01% of the rods will contain cladding defects, usually in the form of small splits or pin holes. Some of the breached rods may contain water. If cladding with small breaches is to provide a barrier function, then it will be necessary to determine if fuel oxidation occurs rapidly enough under repository conditions to split the cladding and expose additional fuel with an oxidation state higher than UO_2 before significant credit can be taken for pin-hole-defected cladding as a barrier to radionuclide release. Thus,

spent fuel oxidation time response is also an input function for modeling the extent and amount of exposed UO_2 spent fuel in failed cladding.

Spent LWR fuel consists primarily of UO_2 pellets, whose density is 92-95% of the theoretical density, enclosed in a Zircaloy sheath. When UO_2 oxidizes in excess oxygen, it passes through certain possibly metastable states, such as U_4O_9 and U_3O_8 , before it totally oxidizes to UO_3 . The rate at which the fuel oxidizes through the various phases depends on the temperature. The rate may also depend on the moisture content of the atmosphere, previous radiation history of the fuel, and radiation level during storage. The densities of the phases range from a high of 10.3 g/cm^3 for 93% dense UO_2 to 7.3 g/cm^3 for UO_3 . Until U_3O_8 , with a density of 8.3 g/cm^3 , is formed, intermediate phases have densities approximately equal to that of UO_2 . Therefore, as the UO_2 oxidizes through U_3O_8 , the fuel pellets will swell and put a tensile hoop stress on the cladding. Several studies have shown that cladding placed under a hoop stress, caused by the formation of U_3O_8 , will enlarge existing breaches and, in some cases, will fracture where there had been on previous pin hole (small) breaches.

In the following Section 3.2.1, information and data are provided from TGA (Thermogravimetric Apparatus) Tests and ODB (Oven Dry Bath) tests. The temperature-time response with spent fuel oxidation testing below 260°C remains to be completed. Above 260°C, it appears that the oxidation time response is sufficiently rapid to be instantaneous relative to repository time duration. Thus, atmospherically exposed UO_2 spent fuel in failed waste packages as a model for kinetics, transforms instantly to U_3O_8 (or UO_3) above 260°C.

The temperature-time response of oxidizing UO_2 below 260°C will be described with a preliminary model in Section 3.2.2. This model represents two time sub-

domains. The first sub-domain is the time interval for oxidizing spent fuel to a U_4O_9 lattice structure and to attain the O to U of ~ 2.4 plateau that has been observed. The time interval will be evaluated based on the time for the U_4O_9 front to propagate to the center of a spent fuel grain. This is a conservative model as it assumes that all grain boundaries oxidized and crack open instantaneously when the UO_2 spent fuel is initially exposed to atmospheric oxygen. The model can also describe the weight gain time response during partial oxidation of UO_2 grains to U_4O_9 grains.

The second sub-domain is the time interval on the ~ 2.4 O/U plateau before a transition away from the plateau level appears. This transition in oxygen weight gain is conjectured to be the result of the initial formation of the U_3O_8 phase, which would occur most likely on the outer boundary of the existing U_4O_9 grains. At the present time, no data are available on the subsequent time response or geometrical character of the U_3O_8 oxidation response. Thus, it is conservative to assume that the total time interval for UO_2 spent fuel to transform to U_3O_8 is the sum of the two sub-domain time intervals. However, it is conservative to assume that the total time interval for UO_2 spent fuel to transform to U_3O_8 is the sum of the two sub-domain time intervals (time to reach U_4O_9 plus time to initiate the U_3O_8 phase). Note that a critical part of the assumption is that an individual grain must all be at an O/U of ~ 2.4 before the phase transition to U_3O_8 can be initiated. Once the relatively low (compared to UO_2) density state of U_3O_8 has been attained, the dissolution/release performance of spent fuel has been significantly decreased because of the large ($\sim$ potentially three orders of magnitude) increase in exposed spent fuel surface area, relative to the initial surface area of fragmented spent fuel pellets. Thus, the time-temperature-phase transformation responses for the different spent fuel oxidation processes can significantly impact the potential release rate of radionuclides from spent fuel waste forms.

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.2.1 Experimental Parameters for Oxidation Models

The testing activities to determine oxidation response are thermogravimetric analysis (TGA) method and oven dry bath (ODB) method. Both methods provide measurements of weight gain due to oxidation of the sample during the time interval of the testing and under controlled temperature and some controlled atmospheric gas variations. In the case of TGA testing, the sample size is small, around 200 mg, and this initial weight is typical of an average fragment from a spent fuel pellet. In the case of ODB testing, the sample size is considerably larger, around 10 grams of spent fuel. (The larger initial sample weights of ODB testing provides a well-controlled procedure to obtain oxidized spent fuel samples for future oxidized spent fuel dissolution testing.) In each testing method, the weight gain time response is measured, and samples can be obtained for microscopic examinations at various oxidation stages (times) during the weight gain time response. The weight gain time response is usually reported as the oxygen to heavy metal atomic ratio, with an O to U or O to M ratio notation (the first is oxygen to uranium and the second is oxygen to heavy metal, which ideally would include all actinide atoms but is dominated by the uranium atomic number density). For example, an O/U of 2.0 is UO_2 . The microscopic analyses are performed on samples to identify the sequence of crystallographic lattice structures that occur during the oxidation of spent fuel. From the weight gain measurements and the phase identification analysis, the oxidation response for the existing test matrix of spent fuel samples has shown that UO_2 spent fuel transforms first to a non-stoichiometric U_4O_9 lattice structure phase with an O/U of ~ 2.4 at temperatures below 200 C. Transitions to higher oxidation phases (U_3O_8 and UO_3) have not yet been observed in ODB tests below 200 C. Higher temperature TGA and ODB testing are being initiated to determine the phases and the kinetics of oxidation plus phase transformation mechanisms. The critical temperature range to establish oxidation response and phase change kinetics (stable versus metastable transformations) is between 200 C and 260 C. Both TGA

and ODB testing activities are in progress to provide additional data in the 200 C to 260 C temperature interval. Above 260 C, the rate of spent fuel oxidation and the phase transformations proceed rapidly to U_3O_8 and UO_3 lattice structures in short periods of time (weeks/years) relative to repository disposal time periods (100 to 1000 years).

The following set of figures present a visual and brief statement format that provides information and data obtained with the existing spent fuel oxidation testing methods. The oxidation response of spent fuel is not believed to be a radionuclide release process (there is a possible gaseous release mechanism which remains to be characterized). Rather, the oxidation response of spent fuel is considered primarily a degradation process which transforms the physical and chemical state of the waste form. As a result, the oxidation phases UO_2 , U_4O_9 , U_3O_8 , or UO_3 have potentially different intrinsic dissolution rates that determine the aqueous release rate response. In addition to the dissolution rates, the potential magnitude of surface area exposed greatly increases as UO_2 oxidizes to the higher phases. In going from UO_2 to U_4O_9 , grain boundaries between grain volumes crack open because of the slight volume decrease during this phase transformation. Furthermore, the phase transformations from the U_4O_9 lattice structure to U_3O_8 and UO_3 have significant volume increases which can microcrack and flake grain volumes to smaller particles and/or powdered forms. Thus, the aqueous radionuclide release rate is potentially increased for higher oxidized phases if groundwater access occurs to wet the increased exposed surfaces of oxidized spent fuel.

SPENT FUEL OXIDATION RESPONSE

FUEL PELLETS, NOMINALLY 0.5cm TO 0.6cm RADIUS AND ~2cm LENGTH, FRACTURE INTO FRAGMENTS DUE TO THERMAL STRAINS DURING FIRST FULL POWER CYCLE.

FUEL FRAGMENTS OXIDIZE AFTER CLADDING BREACH.

MODEL - OXIDATION KINETICS DEPEND STATISTICALLY ON FRAGMENT SIZES AND SHAPES IN A TEST SAMPLE; ANY FRAGMENT CAN BE SUBDIVIDED INTO DIFFERENT SIZED PYRAMIDAL VOLUME SUBSETS TO OBTAIN A STATISTICAL DISTRIBUTION FUNCTION.

Fragment cross-section

3.2.1-3

Figure 3.2.1-1

Spent fuel waste form characteristics observations

- **UO₂ fuel pellets (Initially ~0.5 cm radius and ~1–3 cm length) break into fragments during reactor operation. This fragmentation increases the surface area of spent fuel for oxidation and dissolution responses.**
- **A fragment of UO₂ spent fuel will oxidize to higher oxidation weight gains (UO_{2+x}) and at low temperatures other oxidation state phases (U₄O₉, U₃O₇, U₃O₈, UO₃ plus possible hydrates).**
- **A fragment of spent fuel (oxidized or not) will dissolve in aqueous solutions.**

Why Study Spent Fuel Oxidation?

- Small fraction ($< 0.1\%$) of rods will enter repository breached and be available for oxidation when container is compromised
- Cladding corrosion may lead to additional breaches
- High temperature data indicate that low-density U_3O_8 can form, destroying fuel and cladding
- Four oxidation effects:
 - Change phase of fuel
 - Open additional internal fuel surfaces to leachant
 - Release trapped fission gas
 - Split cladding; change path for radioisotope release

Question?

**$\Delta(O/M)$ as a function of time, temperature,
and atmosphere**

Basis for YMP Spent Fuel Oxidation Testing from Early Work

- **Temperature was an important variable**
- **Effect of atmospheric moisture and burnup is uncertain**
- **Low-temperature oxidation data were not available**
- **Assumed UO_2 and spent fuel had similar oxidation behavior**

TGA Apparatus

HEDL 8703-143.1

TGA DATA ANALYSIS

ASSUMPTIONS

1. UNIFORM SPHERICAL GRAINS
2. GRAINS OXIDIZE INDEPENDENTLY
3. PLANAR OXIDATION FRONT

DESCRIPTION

$$1 - [1 - 3 \Delta(O/M)]^{1/3} = (k't)^{1/2}$$

WHERE

$\Delta(O/M)$ = CHANGE IN OXYGEN TO METAL RATIO
 k' = OXIDATION RATE CONSTANT
 t = OXIDATION TIME

FITTING TGA DATA TO OBTAIN RATE CONSTANT

3.2.1-9

Figure 3.2.1-7

TGA Oxidation Summary

- Different oxidation behavior in unirradiated UO_2
- Spent fuel oxidation is a two-step process: oxygen penetration of the grain boundaries followed by oxidation of the bulk grains
- Arrhenius dependence on temperature. The activation energy is consistent with O_2 diffusion into UO_2 .
- Moisture level has little effect
- Oxidation more rapid at the pellet surface
- The majority of the mechanistic data comes from the microstructural examination of the oxidized fuel

Dry-Bath Oxidation Program

- **To provide rate data for oxidation model**
- **Determine long-term oxidation behavior**
- **Source of fuel for leach testing**

32.1-11

Figure 3.2.1-9

Test Variables

Temperature: 195°, 175°, 130°, 110°C

Dew Points: -55, +80°C

**Sample Configuration: As-irradiated fragments
pulverized fuel**

**Fuels: HB Robinson PWR (ATM-101)
Turkey Point PWR
Cooper BWR (ATM-105)
Calvert Cliffs PWR (ATM-103, -104, -106)**

Grain Size Range: 5 to 30 μm

Burnup Range: 25 to 48 GWd/MTU

FGR Range: 0.1% to 18%

Current Test Times: Up to 40 kh (5.0 yr)

Oxidation of LWR Spent Fuel

3.2.1-13

Figure 3.2.1-11

Figure 3.2.1-12

Change in O/M with Time, 175°C

3.2.1-15

Figure 3.2.1-13

Oxidation of Fragments of Turkey Point Fuel

Data are from Reference: R. E. Einziger and R. V. Strain, *Nuc. Technol.*, 75:82 (1986).

Figure 3.2.1-14

Generalized Spent Fuel Oxidation Curve

δ = time on O/M 2.4 plateau until new phase transition

3.2.1-17

Figure 3.2.1-15

Figure 3.2.1-16

Time Required to Propagate Existing Cladding Defects

Figure 3.2.1-17

Preliminary Conclusions

1. Spent fuel has different oxidation behavior than unirradiated UO_2
2. Fuel variability affects oxidation rate in a transitory manner
3. After transient, all tested fuels show similar oxidation behavior
4. No effect of atmospheric moisture
5. Test temperatures too low for oxidation beyond $\text{UO}_{2.4}$
6. Only UO_2 and U_4O_9 phases found at $\text{O/M} < 2.4$
7. At 95°C , >2000 years to reach plateau

Information Needs:

- 1. No tests of high-burnup or Gd-containing fuels**
- 2. Long-term stability of $\text{UO}_{2.4}$
Thermodynamics to aid modeling?**
- 3. Oxidation kinetics beyond $\text{UO}_{2.4}$ (to U_3O_8)**
- 4. Tests on low burnup fuel (12 GWd/MTU) ?**
- 5. Leaching studies from oxidized and
non-oxidized fuel**

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.2.2 Oxidation Models

Below 260°C Repository UO₂ Spent Fuel Oxidation Response

For UO₂ spent fuel exposed to atmospheric oxygen at temperatures less than 260°C, the elapsed time to oxidize to a new U₄O₉ phase and a weight gain "oxide" of UO_{2.4} depends exponentially on temperature and on the grain size of the UO₂. For simplified model development purposes, it is assumed that this oxidation process for UO₂ spent fuel is such that all grain boundaries oxidize and crack open instantaneously. Following the opening of grain boundaries, each grain volume surface is exposed to oxygen and an oxidation front propagates into each grain volume. Behind this front, the "oxide phase" has a crystal lattice structure of U₄O₉ but an oxygen weight gain that corresponds to UO_{2.4} "oxide." The following rate of propagation for the oxidation front into a grain volume is based on preliminary data that are currently available. This rate of front propagation is empirically estimated (Einziger et al., 1992) to be represented by

$$\dot{H}(t) = \sqrt{2k}/2\sqrt{t} \quad 3.2.2 (1)$$

Equation 3.2.2(1) can be integrated over time to obtain the thickness of the U₄O₉ oxidation zone around a grain for a constant temperature history as

$$H(t) = \sqrt{2kt} \quad 3.2.2 (2)$$

where $k(\mu\text{m}/\text{h}) = 1.04 \times 10^8 \exp(-24.0 \text{ kcal.mol}^{-1}/RT)$

$R = 1.986 \text{ cal/mol/}^\circ\text{K}$

$T = \text{temperature in } ^\circ\text{K}$

$t = \text{time (h) in hours}$

For a grain of nominal dimension $2H_0$, the time interval at constant temperature for a UO₂ grain to change phase to the U₄O₉ phase at an O/M of ~2.4 is

$$t_{2.4} = H_o^2 / 2k \quad 3.2.2 (3)$$

H_o in units of μm (10^{-6} meter)

$t_{2.4}$ in unit of hours

In the more general case, the above equation for $\dot{H}$ can be used in evaluating U_4O_9 weight gain time response expressions provided in the description that follows the elapsed time response modeling. The nominal dimension H_o for this weight gain response model is obtained by subdividing the grain volumes into pyramidal sub-volumes. For time-dependent temperature histories, the closed-form integration of Equation 3.2.2 (1) cannot be performed, and it will be necessary to use a numerical integration method.

After a grain has attained an O/M plateau at ~ 2.4 , there appears to be a transition time interval denoted as δ (see Section 3.2.1) before significant subsequent oxidation weight gains occur. The dependence of this time interval on temperature has not yet been well established, but has been conjectured to have the function form:

$$\delta = \delta_o \exp (Q/RT) \quad 3.2.2 (4)$$

where consistent dimensions for the parameters would be

$\delta_o =$ units of hours

$Q =$ units of kcal/mol

$R = 1.986 \text{ cal/mol/}^\circ\text{K}$

At this time, sufficient experimental data are not available to provide estimated values for δ_o and Q at temperatures below 260°C . If such values were available, then it is conjectured that the next oxidation phase, namely U_3O_8 , (or $\text{UO}_{2.66}$), begins to form and initiates the transition from the plateau. Then the elapsed time until

the appearance of the transition from the O/M ~2.4 plateau can be denoted as $t_{2.66}$ and would be given by

$$t_{2.66} \cong t_{2.4} + \delta \quad (\text{hours}) \quad 3.2.2 (5)$$

This is the time at which U_3O_8 is conjectured to be initially forming on the boundary of U_4O_9 grains. Although an estimate for the δ elapsed time is not available from existing data, previous experimental work on defected cladding and splitting of the cladding after formation of U_3O_8 at higher temperature will be used to provide a direct estimate for the elapsed time $t_{2.66}$. These data (Einziger and Strain, 1986), which have been augmented with other data, are shown in Figure 3.2.2-1. The top line will be taken as a $t_{2.66}$ elapsed time line for purposes here. Note that it is not a lower bound line to the $t_{2.66}$ elapsed time for temperatures above 260°C, and that extrapolation is necessary for temperatures below 260°C. Nonetheless, for conceptual design and preliminary performance assessments, this will provide an approximation for the initial formation of the U_3O_8 phase of spent fuel. The equation of the line has the form

$$t_{2.66} = t_{2.66(0)} \exp (Q/RT) \quad 3.2.2 (6)$$

and estimated values and definitions for parameters in Equation 3.2.2 (6) are

$$t_{2.66(0)} = 1.37 \times 10^{-15} \text{ hour}$$

$$Q = 44,100 \text{ cal/mole}$$

$$R = 1.986 \text{ cal/mole/}^\circ\text{K}$$

$$T = \text{temperature } ^\circ\text{K}$$

For UO_2 spent fuel first exposed to the atmosphere, and held at a fixed temperature of 150°C, the extrapolated elapsed time for U_3O_8 to initially form is approximately 9.75×10^7 hours or 1.1×10^4 years. The subsequent rate response of the U_4O_9 grains oxidizing to U_3O_8 remains to be experimentally established.

Time Required to Propagate Existing Cladding Defects

Figure 3.2.2-1

Once the U_3O_8 oxidation rate response has been established, and if a U_3O_8 front also propagates as an oxidation front into the U_4O_9 grains, then the following model of oxidation weight gain time response can be applied (Stout, et al. 1990). This model, as noted above, subdivides a grain into pyramidal subvolumes, and assumes that an oxidation front (either the U_4O_9 or the U_3O_8) propagates into the interior of the grain such that the front remains geometrically similar to the original grain boundary. This rate response model can be used to predict the extent of partial oxidation at low temperatures; for example, some U_4O_9 and UO_2 (or U_3O_8 and U_4O_9) may exist simultaneously at temperatures below 100°C when water would be first potentially available. The application of the oxidation weight gain response would predict the amount of spent fuel inventory which exists in each known oxidation phase. This information, along with intrinsic dissolution rate data for each phase, would be used to provide estimates of the radionuclide release rate over the long time periods (thousands of years) of interest in performance assessment and waste package design analysis.

Oxidation weight gain response depends on oxidation phase, oxidation front motion and surface area

Oxidation phase ~ amount (mass) of O₂ added per unit volume of UO₂ in a fragment.

Oxidation front motion ~ velocity of front (cm/unit time).

Surface area ~ area at oxidation front; decreases in time as front propagates into a fragment.

Weight gain ~ $\left[\begin{array}{c} \text{Mass of O}_2 \\ \text{added at front} \end{array} \right] \cdot \left[\begin{array}{c} \text{Front} \\ \text{velocity} \end{array} \right] \cdot \left[\begin{array}{c} \text{Surface area} \\ \text{at front} \end{array} \right]$

Grain set decomposed to pyramidal volume subsets

A set of grain volumes (in cross section)

Put a point at the center of each grain, and decompose into a set of pyramids (triangles in cross section).

3.2.2-7

Figure 3.2.2-3

Density function: probable number of grain pyramids

- Exists a large number of grain pyramids, many of which will be of the same "size" (compact domain set).

A "size" can be identified by attributes (a, b, c), as illustrated below.

Let $G(\underline{x}, \underline{t}, \underline{a}, \underline{b}, \underline{c})$ denote the probably number of pyramids of size (a, b, c) in a unit spatial volume of grains about point $\underline{x}$ at time $\underline{t}$.

Grain volume oxidation front

Pyramidal volume in an oxidizing grain and its associated physical attributes.

3.2.2-9

Figure 3.2.2-5

Density function: probable number of grain pyramids

- Exists a large number of grain pyramids, many of which will be of the same "size" (compact domain set).

A "size" can be identified by attributes ($\underline{a}$, $\underline{b}$, $\underline{c}$), as illustrated below.

Let $G(\underline{x}, \underline{t}, \underline{a}, \underline{b}, \underline{c})$ denote the probably number of pyramids of size ($\underline{a}$, $\underline{b}$, $\underline{c}$) in a unit spatial volume of grains about point $\underline{x}$ at time $\underline{t}$.

Figure 3.2.2-6

Fragment set decomposed to pyramidal volume subsets

A set of fragments (in cross section)

Put a point at the center of each fragment, and decompose into a set of pyramids (triangles in cross section).

3.2.2-11

Figure 3.2.2-7

Physical attributes of pyramidal volumes for fragments

Each fragment pyramid characterized by a set of vector attributes ($\underline{p}$, $\underline{q}$, $\underline{h}$)

Figure 3.2.2-8

Fragment pyramidal volume oxidation front

Pyramidal volume in an oxidizing fragment and its associated physical attributes.

oxygen weight gain rate

$$\dot{O}(t) = N e_{ijk} p_j q_k h_i (1-H)^2 \dot{H}$$

Figure 3.2.2-9

Density function: probable number of fragment pyramids

- Exists a large number of fragment pyramids, many of which will be of the same "size" (compact domain set).

A "size" can be identified by attributes (p , q , h), as illustrated below.

Let $F(\underline{x}, t, p, q, h)$ denote the probably number of pyramids of size (p, q, h) in a unit spatial volume of fragments about point $\underline{x}$ at time t .

Figure 3.2.2-10

Oxygen weight gain for a density of fragments

The probable number of pyramidal volume of species $\underline{f} \sim (\underline{p}, \underline{q}, \underline{h}, H, \dot{H})$ in a $d\underline{x}$ volume at time t is $F(\underline{x}, t, \underline{f}) d\underline{f}d\underline{x}$;

Then the oxygen weight gain for this number is

$$\dot{O}(\underline{x}, t, \underline{f}) = N e_{ijk} p_j q_k h_l (1-H)^2 \dot{H} F(\underline{x}, t, \underline{f}) d\underline{f}d\underline{x}$$

Integrate over the domain of fragment species $\{\underline{f}\}$ and the spatial volume of B of fragments

$$\dot{O}(t) = \int \int_{B\{\underline{f}\}} N e_{ijk} p_j q_k h_l (1-H)^2 \dot{H} F(\underline{x}, t, \underline{f}) d\underline{f}d\underline{x}$$

The above formula for the case of a distribution of grain volume oxidizing would replace the fragment density function F with a grain density $G(\underline{x}, t, \underline{a}, b, c, h, \dot{h})$ to become

$$\dot{O}(t) = \int \int_{B\{g\}} N e_{ijk} a_j b_k c_i (1-h)^2 \dot{h} G(\underline{x}, t, g) dg d\underline{x}$$

$$\dot{h} = 2.12 \times 10^{13} \text{ (m/hr) } \exp (-27000/RT)$$

where $\dot{h}$ is from Section 3.2.2

Illustration of oxidation responses from fragment statistics

Figure 3.2.2-12

R.B. Stout, et al., "Spent Fuel Waste Form Characterization: Grain and Fragment Size Statistical Dependence for Oxidation Response, LLNL Report UC RL-107910, May, 1991.

Adachi, T., M. Ohnuki, et al., "Dissolution Study of Spent PWR Fuel: Dissolution Behavior and Chemical Properties on Insoluble Residues," *J. Nucl. Mater.*, **174**, 60-71 (1990). (Readily available.)

Aronson, S., R.B. Roof, and J. Belle. "Kinetic Study of the Oxidation of Uranium Oxide," *J. Chem. Phys.* **27**, 137 (1957). (Readily available)

Boltzmann, L. "Lectures on Gas Theory," translated by S.G. Brush, Univ. of California Press, Berkeley, CA (1964). NNA.920401.0117

Carslaw, H.S. and J.C. Jaeger. "Conduction of Heat in Solids," second edition, Oxford University Press, New York (1959). NNA.900522.0259

Einzig, R.E. and H.C. Buchanan. "Long-Term, Low Temperature Oxidation of PWR Spent Fuel," Westinghouse Hanford Co. Report, WHC-EP-0070 (June, 1987). NNA.900620.0297

Einzig, R.E. and R.E. Woodley. "Predicting Spent Fuel Oxidation States in a Tuff Repository," Westinghouse Hanford Co. Report HEDL-SA-3627, (April, 1985). NNA.870915.0073

Einzig, R.E. "Technical Test Description of Activities to Determine the Potential for Spent Fuel Oxidation in a Tuff Repository," Westinghouse Hanford Co. Report HEDL-7540 (June, 1985). NNA.920302.0060

Einzig, R.E. "Test Plan for Series 2 Thermogravimetric Analyses of Spent Fuel Oxidation," Westinghouse Hanford Co. Report HEDL-7556 (February, 1986). NNA.920302.0061

Einzig, R.E., "Test Plan for Long-Term, Low-Temperature Oxidation of Spent Fuel, Series 1," Westinghouse Hanford Co. Report HEDL-7560 (June, 1986). NNA.920302.0062

Einzig, R.E. and R.V. Strain, *Nucl. Technol.*, **75**, 82 (1986). (Readily available)

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

Finch, R., and R. Ewing, "Alteration of Natural UO_2 Under Oxidizing Conditions from Shinkolobwe, Katanga, Zaire: A Natural Analogue for the Corrosion of Spent Fuel," *Radiochim. Acta*, **52/53**, 395-401 (1991). NNA.900507.0149

Gleiter, H. and B. Chalmers. "High-Angle Grain Boundaries," in *Prog. in Mat. Sci.*, Vol. 16, Pergamon Press, New York (1972). (Readily available)

Grambow, B. "Spent Fuel Dissolution and Oxidation: An Evaluation of the Literature," SKB Tech. Rept. 89-13, Svensk Kärnbränslehantering AB, Stockholm, 42 pgs. (1989). NNA.891013.0094

Grambow, B., R. Forsyth, et al. "Fission Product Release from Spent UO_2 Fuel Under Uranium-Saturated Oxidic Conditions," *Nucl. Tech.*, **92**, 204-213 (November, 1990). (Readily available)

Olander, D.R. "Combined Grain-Boundary and Lattice Diffusion in Fine-Grained Ceramics," *Advances in Ceramics*, **17**, 271 (1986). (Readily available)

Slattery, J.C. *Momentum, Energy and Mass Transfer in Continua*, R.E. Krieger Pub. Co., New York (1978). NNA.920401.0115

Stout, R.B., H.F. Shaw, and R.E. Einziger, LLNL Report UCRL-100859, September, 1989. (Readily available)

Stout, R.B., E. Kansa, R.E. Einziger, H.C. Buchanan, and L.E. Thomas, LLNL Report UCRL-104932. (Readily available)

Stout, R.B. "Statistical Model for Particle-Void Deformation Kinetics in Granular Materials During Shock Wave Propagation," Lawrence Livermore National Laboratory Report UCRL-101623 (July, 1989). NNA.900517.0267

Thomas, L., O. Slagle, and R. Einziger, "Nonuniform Oxidation of LWR Spent Fuel in Air," *J. Nucl. Mat.*, **184** 117-126 (1991). NNA.910509.0071

Thomas, L., R. Einziger, and R. Woodley, "Microstructural Examination of Oxidized Spent PWR Fuel by Transmission Electron Microscopy," *J. Nucl. Mat.*, **166**, 243-251 (1989). NNA.900709.0482

Thomas, L.E., R.E. Einziger, and R.E. Woodley. "Microstructural Examinations of Oxidized Spent Fuel by Transmission Electron Microscopy," *J. Nucl. Mat.*, in press. YMPO Accession NML 880707.0043.

Wood, P. and G.H. Bannister. "Investigation of the Mechanism of UO_2 Oxidation in Air: The Role of Grain Size," Proc. Workshop on Chemical Reactivity of Oxide Fuel and Fission Product Release, K.A. Simpson and P. Wood, eds., Berkeley Nuclear Lab., U.K., p. 19 (1987). NNA.920401.0116

Woodley, R.E., R.E. Einziger, and H.C. Buchanan. "Measurement of the Oxidation of Spent Fuel Between 140 and 225 C by Thermogravimetric Analysis," Westinghouse Hanford Co. Report WHC-EP-0107 (Sept., 1988). NNA.880927.0069

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.3.1 Experimental Parameters for Fission Gas Release

The Materials Characterization Center has estimated distributions of burnup and fission gas release for the current and projected spent fuel inventories through 2020.¹ This was done both to help assure that the current suite of Approved Testing Materials is representative of the spent fuel inventory and to help define additional spent fuel ATM needs.

From the developed distributions it was concluded that the current ATM's may be considered representative, in terms of fission gas release and burnup, of nearly 100% of the spent fuel inventory discharged through 1988. However, those ATM's may be considered representative of only 61% of the total projected inventory discharged through 2020. That is because 39% of the inventory is projected to have burnup levels in excess of 45 MWd/kgM while none of the ATM's have burnups in excess of 45 MWd/kgM.

Noting that there are no ATM's representative of high burnup spent fuel, it is recommended that the next ATM to be acquired be representative of a modern fuel design (e.g., BWR 8x8 barrier or PWR 17x17) and have the highest possible burnup. It should also have low fission gas release so as to be representative of the large volume of fuel in the low fission gas release/high burnup category of spent fuel. A second ATM to be acquired should have the characteristics of high fission gas release/high burnup to be both representative of the other currently unrepresented category of spent fuel and provide an ATM that would be bounding of the expected spent fuel characteristics.

¹ M. E. Cunningham, et al., "The Impact of Burnup and Fission Gas Release Distributions of the U.S. Spent Fuel Inventory on the Selection of Spent Fuel Test Materials for the U.S. Geological Repository Project," PNL report in preparation, September, 1990.

The current and above-proposed ATM's will be representative of standard design, non-failed LWR spent fuel. Fuel that will still not be represented by ATM's will include stainless steel-clad fuel, fuel that failed either in-reactor or during interim storage, and miscellaneous test and experimental fuel. It is estimated that these two fuel types will account for 2-5% of the total emplacement inventory.

One significant spent fuel classification that is not included in a burnup-fission gas release distribution is fuel that contains a burnable neutron poison. However, the MCC does have an early vintage Gd_2O_3 burnable poison fuel in its inventory and if a modern high burnup BWR fuel assembly is acquired, modern burnable poison fuel would be part of such an assembly.

Example of Justification for Grouping Varying Rod Designs

LHGR, Burnup, and FGR Correlation for BWR Rod-Types

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.

LHGR, Burnup, and FGR Correlation for PWR Rod-Types

39006088.4

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.3.2 Gaseous Radionuclide Release

Fission gas release has been predicted using a "standard" model, the so-called ANS-5.4 model,¹ as revised by C.E. Beyer of PNL.²

Similar fission gas release curves were presented at the "Status and Future Directions of Spent Fuel ATM Acquisition and Characterization" meeting at PNL March 28-29, 1989, by C.E. Beyer of the MCC. These curves were generated using the revised ANS 5.4 Gas Release Model and we have fit these curves to a very simple and easily used equation for burnup ≥ 20 MWd/kgM and for gas release $\leq 60\%$. This equation is

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

We have calculated points at 30 and 40 MWd/kgM and superimposed them on Beyer's curves to show the agreement. Additional curves for 30, 50, and 60 MWd/kgM calculated using our simple expression are also plotted.

¹ Method for Calculating Fractional Release of Volatile Fission Products from Oxide Fuel," ANSI/ANS-5.4-1982.

² Memo from C.E. Beyer (PNL) to J.C. Vogtlewede (USNRC), May 24, 1982.

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

Percent of Fission Gas Release versus Local Temperature

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

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.4 Spent Fuel Dissolution

The dissolution of a waste form and the associated release of the included radionuclides will be limited by two boundary conditions in a geologic repository: (1) in the case of contact with fast moving water, the dissolution rate of the waste form will determine the rate at which radionuclides are released, (2) in the case of contact with slowly moving water (slow relative to the dissolution rate), the rate of release of radionuclides will be determined by their solubility under the prevailing conditions. Thus, in order to assess performance of a repository, both rates of dissolution and solubility limits should be available.

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 Amell 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 of 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 unconstrained or which simulated complex repositorial conditions. The results of such studies are difficult

The results are equivocal due to the difference in experimental designs, the diverse history of the fuel samples, the formation of secondary phases during the tests, and the complexity of the solution and surface chemistry of UO_2 . Data indicate that UO_2 is easily oxidized to U_4O_9 and U_3O_8 in air^{9,10} and can be further oxidized to either U_3O_8 ^{9,10,11} or schoepite, $\text{UO}_3 \cdot 2\text{H}_2\text{O}$.¹² The UO_2 surface oxidation leads to higher leach rates because of higher dissolution rates of U_3O_8 , U_3O_8 , or schoepite relative to that of UO_2 and because of the increase of surface area of the fuels due to surface cracking.

- ¹ A.R. Amell, and D. Langmuir, "Factors Influencing the Solution Rate of Uranium Dioxide Under Conditions Applicable to In-Situ Leaching," Bureau of Mines Open File Report 84-79, U.S. Dept. of Interior, Bureau of Mines (1978).
- ² G.A. Parks, and D.C. Pohl, "Hydrothermal Solubility of Uraninite," *Geochim. Cosmochim. Acta*, 52, 863 (1988).
- ³ J. Bruno, I. Casas, and I. Puigdomenech, "The Kinetics of Dissolution of $\text{UO}_2(\text{s})$ Under Reducing Conditions," *Radiochim. Acta*, 44/45, 11 (1988).
- ⁴ B. Grambow, "Spent Fuel Dissolution and Oxidation. An Evaluation of Literature Data," SKB Technical Report 89-13 (1989).
- ⁵ D.E. Grandstaff, "A Kinetic Study of the Dissolution of Uraninite," *Econ. Geo.*, 71, 1493 (1976).
- ⁶ W.E. Schortmann, and M.A. DeSesa, "The Kinetics of the Dissolution of Uranium Dioxide in Carbonate-Bicarbonate Solutions," *Proc. 2nd Intern. United National Conf. Peaceful Uses of Atomic Energy, United Nations, Geneva*, 3, 333 (1958).
- ⁷ R.L. Pearson and M.E. Wadsworth, "A Kinetic Study of the Dissolution of UO_2 in Carbonate Solution," *Trans. Metal. Soc. AIME*, 212, 294 (1958).
- ⁸ F. Habashi and G.A. Thurston, "Kinetics and Mechanisms of the Dissolution of Uranium Dioxide," *Energ. Nucl.* 14, 238 (1967).
- ⁹ S. Aronson, "Oxidation and Corrosion of Uranium Dioxide in Uranium Dioxide: Properties and Nuclear Applications," J. Belle, ed., U.S. Atomic Energy Comm., 377 (1961).
- ¹⁰ R.E. Einziger, "Test Plan for Long-Term, Low-Temperature Oxidation of BWR Spent Fuel," PNL-6427, Pacific Northwest Laboratory (1988).
- ¹¹ S. Aronson, "Oxidation of UO_2 in Water Containing Oxygen," Bettis Tech. Rev., Westinghouse Atomic Power Div., Report WAPD-BT-10, 93 (1958).
- ¹² T. Wadsten, "The Oxidation of Polycrystalline Uranium Dioxide in Air at Room Temperature," *T. Nucl. Mat.*, 64, 315 (1977).

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

Spent Fuel Dissolution Release Rate

Cladding gap, fragment surface, (grain boundary and areas),
grain inventory releases illustrated

SOLUBLE RADIONUCLIDE RELEASE

- **Rapid release of "gap inventories" with initial water contact (days)**
- **Preferential release from grain boundaries and other sources of radionuclide concentration (years)**
- **Releases are controlled by matrix dissolution after exposed grain boundaries and other sources of radionuclide concentration become depleted** (assuming fuel is not substantially degraded by oxidation)

A SCHEMATIC VIEW OF SF DISSOLUTION*

*L.H. JOHNSON AND D.W. SHOESMITH, "RADIOACTIVE WASTE FORMS FOR THE FUTURE,"
W. LUTZE AND R.C. EWING, EDS., ELSEVIER (1988) P. 686

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.4.1.1 Dissolution Rates

Recent measurements on both UO_2 and spent fuel (SF) under comparable conditions have provided dissolution rates for UO_2 between 25°C and 85°C in waters of various composition and for SF on deionized water (DIW) at 25°C. These experiments were done in equilibrium with air. The results are shown in figures 1 and 2. The rate of dissolution of SF in DIW at 25°C is $1.2\text{--}1.7 \times 10^{-12} \text{ g cm}^{-2} \text{ sec}^{-1}$ as compared to UO_2 in DIW at 25°C at $\sim 5 \times 10^{-12} \text{ g cm}^{-2} \text{ sec}^{-1}$. Given the great variability in other reported values this is reasonable agreement. In fact, the observed dissolution rate for SF in 25°C is about the same as of UO_2 in (DIW + Ca + Si), a simulation of ground water.

The measured dissolution rates for UO_2 and spent fuel 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, 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 used are derived from the bottom curve in figure 1. The rate equation used is

$$G(t) (\text{g cm}^{-2} \text{ sec}^{-1}) = 6.43 \times 10^{-9} \exp\left(-\frac{4740}{RT(K)}\right) \quad (5)$$

A model for the 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. This implies that the particle geometry is retained.

Leider, H.R., et al. "Estimating the Time for Dissolution of Spent Fuel Exposed to Unlimited Water," LLNL Report UCRL-ID-107289, December, 1991. (See Section 2.1.3.5 for more complete discussion.)

We can describe the change in characteristic dimension of a SF particle (a sort of "radius"), X as follows:

$$X(t) = X_o - \left(\frac{G}{\rho} \right) t$$

where $X(t)$ = the characteristic dimension as a function of time

X_o = the original dimension (half of the actual size)

t = time

G = dissolution rate per unit area

ρ = density

The time for complete dissolution of a particle, of original size X_o , is then $t_{\infty} = \frac{X_o \rho}{G}$.

Table 1

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

Table II

<u>Temperature (°C)</u>	<u>Dissolution Time (years)</u>	
	t_{∞}^*	t_{∞}
25	8.0×10^3	5.5×10^4
85	2.2×10^3	1.5×10^4

Figure 1

Gray, W. and Wilson, C., "Effects of Water Composition and Temperature on the Dissolution Rate of UO_2 ," presented at 1990 Spent Fuel Workshop, Gull Harbor, Manitoba, Canada (1990) NNA.910821.0008.

Figure 2

Gray, W. and Wilson, C., "Spent Fuel Grain Boundary Inventory and Testing the Congruency of UO_2 Matrix Dissolution of Spent Fuel," presented at 1990 Spent Fuel Workshop, Gull Harbor, Manitoba, Canada (1990) NNA.910821.0009.

Figure 4

Figure 3. Dissolution Rate for Set of Fragments

Leider, H.R., S.N. Nguyen, R.B. Stout, H.C. Weed, "Estimating the Time for Dissolution of Spent Fuel Exposed to Unlimited Water," LLNL Report UCRL-ID-107289, Dec. 1991.

TABLE 3.6. Particle-Size Distribution of Fuel Rods from NRU LOCA Test HT-3
(Rausch 1984)

Rod Section	Total Weight of Fuel, g	% of Total Sample Weight Retained on Each Sieve Size							Receiver ($<300 \mu\text{m}$ or $<0.0117 \text{ in.}$)
		No. 3-1/2 (5.6 mm or 0.233 in.)	No. 5 (4.00 mm or 0.157 in.)	No. 7 (2.80 mm or 0.111 in.)	No. 10 (2.00 mm or 0.0787 in.)	No. 18 (1.00 mm or 0.0394 in.)	No. 50 (300 μm or 0.0117 in.)		
205 (1)(a)	243.753	15.98	69.84	12.69	0.71	0.23	0.25	0.31	
385 (1)	225.212	19.88	65.57	13.35	0.71	0.08	0.19	0.22	
305 (2)	253.991	18.66	61.03	15.83	2.20	1.13	0.73	0.42	
3E5 (3)	254.497	18.88	58.28	20.2	1.53	0.44	0.27	0.31	
5C5 (3)	270.867	28.49	54.95	14.92	0.74	0.28	0.24	0.30	
5D5 (3)	246.704	20.13	64.58	13.61	0.64	0.31	0.32	0.42	
3B5A (3)	136.590	8.43	70.17	18.97	1.22	0.37	0.33	0.52	
3B5B (11)	146.223	49.83	45.00	4.52	0	0.07	0.26	0.33	

(a) Number in parentheses indicates the number of whole pellets before size analysis.

TABLE 3.7. Particle-Size Distribution of Fuel Fragments from H. B. Robinson
Spent Fuel with a Burnup of 28 MWd/kgM (Katayama, Bradley and
Harvey 1980)

Sieve Number	Sieve Opening, mm	Weight, g	Fraction Retained
3	6.73	0	0
4	4.73	192.883	0.1007
5	4.00	634.765	0.3331
10	2.00	1031.170	0.5384
20	0.841	35.205	0.01038
40	0.420	11.242	0.005369
60	0.250	4.979	0.002599
80	0.177	1.424	0.0007434
100	0.149	1.042	0.0005440
140	0.105	1.204	0.0006286
200	0.074	0.769	0.0004015
200	0.074	0.737	0.0003848

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.4.1.2. Solubility Limits

Attached are solubility data developed 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. (See also Section 2.1.3.5 for additional explanation. In cases where several values from different samples with different geometrics 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.

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 $M_i = C_i \Phi t$.

Table 1. Solubility Data, C_i

<u>Species</u>	<u>Upper Limit Steady-State Concentration (μg/ml)</u>	
	<u>25°C</u>	<u>85°C</u>
U	≤ 5	≤ 0.5
²³⁹⁺²⁴⁰ Pu	≤ 5x10 ⁻³	≤ 6x10 ⁻⁵
²⁴¹ Am	≤ 3x10 ⁻⁴	≤ 1.5x10 ⁻⁷
²⁴⁴ Cm	≤ 1.2x10 ⁻⁵	≤ 2.4x10 ⁻⁹
²³⁷ Np	≤ 4x10 ⁻⁴	≤ 1.4x10 ⁻³

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.4.1.3 Solubility Controls on Radionuclide Concentrations in Solution: Preliminary Results for U, Np, Pu, and Am

Radionuclide concentrations in solution are limited by the precipitation of solids. This is a presentation of calculations of the dissolved concentrations of the radionuclides U, Np, Pu, and Am in equilibrium with potential radionuclide-bearing solids in J-13 water at 25 C. Elemental concentrations vary as a function of solution composition, Eh and pH, among other parameters. To illustrate the potential impact of such variations, the dependance of radionuclide concentrations is solution as well as the identity of the radionuclide-bearing precipitate are calculated as a function of pH. These calculations can be used as a first approximation to estimate potential ranges in radionuclide concentrations in solution.

The chemical composition of J-13 water used in the calculations is given in Table 1. The redox potential of J-13 water was determined by assuming equilibrium with the atmosphere at an oxygen partial pressure of 0.2 bars. At a pH of 7.6, this corresponds to an Eh of 0.77 volts.

Geochemical modeling codes EQ3NR ver. 3245R123, and EQ6 ver. 3245R118, supported by EQLIB version 3245R152 and the thermodynamic data base DATA0.com.R6, were used to make the calculations. All calculations were carried out at 25°C.

Table 1.

J-13 Water		
Element	Concentration	
	mg/l	molality (moles/kg)
Li	0.042	6.053E-6
Na	43.9	1.909E-3
K	5.11	1.307E-4
Ca	12.5	3.119E-4
Mg	1.92	7.897E-5
Sr	0.035	3.995E-7
Al	0.012	4.447E-7
Fe	0.006	1.074E-7
Si	27.0	9.613E-4
NO₃	9.6	1.548E-4
F	2.2	1.158E-4
Cl	6.9	1.946E-4
HCO₃	125.3	2.054E-3
SO₄	18.7	1.947E-4
pH	7.6	
* average J-13 water analysis of LLNL laboratory supply (Table 1, Delany, 1985)		

Table 2.

U		
Solid	Concentration	
	mg/l	molality (moles/kg)
Haiweeite $\text{Ca}(\text{UO}_2)_2\text{Si}_6\text{O}_{15}\cdot 5\text{H}_2\text{O}$	0.1641E-3	0.6893E-9
Soddyite $(\text{UO}_2)_2\text{SiO}_4\cdot 2\text{H}_2\text{O}$	0.015	0.6096E-7
Sklodowskite $\text{Mg}(\text{H}_3\text{O})_2(\text{UO}_2)_2\text{--}(\text{SiO}_4)_2\cdot 4\text{H}_2\text{O}$	11.05	0.4642E-4
CaUO_4	12.59	0.5289E-4
Schoepite $\text{UO}_3\cdot 2\text{H}_2\text{O}$	38.90	0.1634E-3
$\text{UO}_2(\text{OH})_2(\text{beta})$	56.73	0.2383E-3
Uranophane $\text{Ca}(\text{UO}_2)_2(\text{SiO}_3)_2(\text{OH})_2$	142.48	0.5986E-03

Table 3.

Np		
Solid	Concentration	
	mg/l	molality (moles/kg)
NpO_2	0.59	0.2468E-5
$\text{NpO}_2(\text{OH})(\text{am})$	129.39	0.5459E-3
$\text{NaNpO}_2\text{CO}_3\cdot 3.5\text{H}_2\text{O}$	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
$\text{PuO}_2(\text{OH})_2$	0.015	0.6204E-7
$\text{Pu}(\text{OH})_4$	27.97	0.1146E-3

Table 5.

Am		
Solid	Concentration	
	mg/l	molality (moles/kg)
AmOHCO_3	0.0041	0.1696E-7
$\text{Am}(\text{OH})_3$	8.42	0.3464E-4
$\text{Am}(\text{OH})_3(\text{am})$	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.

TABLE II
Phases Identified on Reacted UO₂ Surface

Phase	Formula	Appearance
Schoepite	UO ₃ •2H ₂ O	Dark yellow crystals
Dehydrated Schoepite	UO ₃ •0.8H ₂ O	Yellow crystals with reflective face
Compreignacite	K ₂ U ₆ O ₁₉ •11H ₂ O	Yellow crystals
Uranophane	Ca(UO ₂) ₂ (SiO ₃) ₂ (OH) ₂ •5H ₂ O	Fine white needles
Boltwoodite	K(H ₃ O)UO ₂ (SiO ₄)•nH ₂ O	Yellow crystals
Sklodowskite	Mg(UO ₂) ₂ (SiO ₃ OH) ₂ •5H ₂ O	Fine needles
Becquerelite	CaU ₆ O ₁₉ •10H ₂ O	Dark yellow crystals
Fluoropolymer	Not determined	White feathers

J.K. Bates, *Identification of Secondary Phases Formed During Unsaturated Reaction of UO₂ with EJ-13 Water*, Materials Research Society Symposium proceedings 176, 499 (1990).

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

Release rate response—depends on inventory, dissolution front motion, surface area, and solubility

For the case of highly soluble species (CS, I, Sr, Tc, etc.):

Inventory ~ amount of a radionuclide species per unit mass

**Dissolution response ~ material removal (mass) per unit area
per unit time**

**Surface area ~ area of fragments size distribution decreases
as fragments dissolve over time.**

Release rate ~ [Inventory] • [dissolution] • [area]

**Release rate
spent fuel ~ [gap inventory] • [gap dissolution] • [gap area]**

$$\begin{aligned} &+ \left[\begin{array}{c} \text{grain boundary} \\ \text{inventory} \end{array} \right] \cdot \left[\begin{array}{c} \text{grain boundary} \\ \text{dissolution} \end{array} \right] \cdot \left[\begin{array}{c} \text{grain boundary} \\ \text{area} \end{array} \right] \\ &+ \left[\begin{array}{c} \text{grain volume} \\ \text{inventory} \end{array} \right] \cdot \left[\begin{array}{c} \text{grain volume} \\ \text{dissolution} \end{array} \right] \cdot \left[\begin{array}{c} \text{grain volume} \\ \text{area} \end{array} \right] \end{aligned}$$

Oxidation front and dissolution front analog

Grain boundary/volume oxidation front motion through a pellet fragment has geometrical and model development similarities to an idealized dissolution surface motion progressing into a fragment.

This means that model development concepts for oxidation kinetics can be also applied to dissolution response for a distribution of fragments.

Fragment pyramidal volume dissolution front

Pyramidal volume in a dissolving fragment and its associated physical attributes.

Dissolution rate on a grain volume and grain boundary surface

Fragment surface $\underline{p} \times \underline{q}$ with grain statistical density $G(g)$.

- I_{VK} – Grain volume inventory species K^{th} radionuclide.
- $\dot{H}_{Vh}$ – Dissolution rate of grain volumes exposed on surface $\underline{p} \times \underline{q}$ of fragment.
- I_{BK} – Grain boundary inventory species K^{th} radionuclide.
- $\dot{H}_{Bh}$ – Dissolution rate of grain boundary exposed on surface $\underline{p} \times \underline{q}$ of fragment.

Statistical representation of grain and fragment geometrical influences on dissolution rate

$$\begin{aligned}
 \dot{D}_K(\underline{x}, t, f) df d\underline{x} = & I_{VK} \dot{H}_V [h_\ell e_{\ell jk} p_j q_k - \int_{\{g\}} (h_\ell e_{\ell mn} w c_m e_{ijk} a_j b_k p_l q_n \\
 & + h_\ell e_{\ell mn} w c_n e_{ijk} a_j b_k q_l p_m) G(\underline{x}, t, g) dg] (1-H_V)^2 F(\underline{x}, t, f) df d\underline{x} + \\
 & I_{BK} \dot{H}_B \int_{\{g\}} (e_{\ell mn} w c_m e_{ijk} a_j b_k p_l q_n h_\ell + e_{\ell mn} w c_n e_{ijk} a_j b_k q_l p_m h_\ell \\
 & + e_{jln} w c_j e_{\ell km} a_k b_m p_l q_n h_\ell) G(\underline{x}, t, g) dg df (1-H_B)^2 F(\underline{x}, t, f) df d\underline{x}
 \end{aligned}$$

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.5 Glass Dissolution

Hydrolysis of Si-O bonds by water initiates glass dissolution and causes the formation of a hydrous reacted layer. Elements released during hydrolysis diffuse outwards into solution while the hydrous altered layer re-polymerizes. The overall rate of reaction appears to be controlled by the rate of dissolution of the re-polymerized hydrous surface layer. Secondary phases consisted of elements present in the leachate and elements released from the glass precipitate from solution or segregate from amorphous material on the glass surface. These alteration layers do not appear to provide a transport barrier.

Current models for glass dissolution combine a rate equation derived from irreversible thermodynamics with reaction path computer codes that account for solution speciation and precipitation of solids. Although these models account for the major features observed in short-term dissolution tests of waste glasses, there remain uncertainties when extrapolating these models to long time periods. The most critical of these uncertainties is that of the nature of the chemical process which determines the long-term dissolution rate of the glass.

Features observed on reacted glass surfaces.

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

Table I. Compositions in weight percent of several nuclear waste glasses and basalt glass. SRL-165, JSS-A, PNL 76-68, SRL-131, basaltic glass.

	SRL-165-U	JSS-A	PNL 76-68	SRL-131	basaltic glass
Al ₂ O ₃	4.08	4.9	0.7	3.1	14
B ₂ O ₃	6.76	14.4	9.3	9.9	—
BaO	0.06	—	0.5	0.1	—
CaO	1.62	4.0	2.4	1.0	11
CeO ₂	—	0.9	0.9	0.2	—
Cs ₂ O	0.07	1.4	0.9	0.3	—
Fe ₂ O ₃	11.3	2.9	9.3	14.3	4
FeO	0.35	—	—	—	8
K ₂ O	—	—	—	0.1	0.2
La ₂ O ₃	—	0.9	4.1	0.3	—
Li ₂ O	4.18	2.0	—	3.9	—
MgO	0.70	—	—	1.2	7
MnO	2.27	—	.04	4.2	.2
MoO ₃	—	1.7	1.9	—	—
Na ₂ O	10.8	9.8	13.9	14.8	2.5
NiO	0.85	—	—	1.5	—
SiO ₂	52.8	45.2	41.5	38.6	51
SrO	.11	—	0.4	0.1	—
TiO ₂	0.14	—	3.0	0.8	2
ZnO	0.04	2.5	4.6	—	—
ZrO ₂	0.66	2.6	1.8	0.3	—
U ₃ O ₈	0.96	0.5	—	1.6	—
P ₂ O ₅	0.29	—	0.7	0.1	0.1

(a) Log plot of extent of reaction (measured as cumulative hydrogen consumption) for leaching of sodium silicate glass in water and D₂O. The separation in the leaching curves remained constant through both stages of reaction, the curved and linear parts of the release curves shown in (b), indicating no change in rate-controlling mechanism throughout the reaction

Cumulative release of silica from SRL-165 glass leached in 0.003 molal sodium bicarbonate (solid diamonds). Curves are regressed on data using equations for diffusion (rate = $A + Bt^{1/2}$) and surface affinity control (rate = $Ak_f(1-Q/K)$), where A and B are fitting parameters.

pH dependence of dissolution rate of Na-Ca-Al-B-Si glass determined from flow-through constant pH dissolution tests

W.L. Bourcier, "Overview of Chemical Modeling of Nuclear Waste Glass Dissolution," UCRL-JC-104531 preprint, Nov. 1990. Prepared for presentation at the Materials Research Society 1990 Fall Meeting, Boston, MA.

Hydration of the surface and ion exchange of alkalis

Diffusion layer thickens until rate of diffusion of alkalis equals rate of dissolution of network structure of gel

Diffusion and gel layers migrate into the glass maintaining a nearly constant thickness (steady state)

Schematic diagram showing evolution of glass alteration layers terminating with a steady state condition where the thickness of the diffusion layer remains constant.

Hydrogen depth profiles in soda-lime glass hydrated in water at 90°C for times up to 540 hours. Steady state is indicated by unchanging profile over last 100 hours.

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.5.2 Glass Dissolution Model

The rate equation used in the general model (W. L. Bourcier, Overview of Chemical Modeling of Nuclear Waste Glass Dissolution, *Mat. Res. Soc. Symp. Proc.*, 212, 3-18, 1991) of glass dissolution analysis is

$$\frac{dc_i}{dt} = \frac{A}{V} vk_+(a_{H^+})^n (1 - e^{Aff/RT}) \quad (1)$$

where

- c_i = concentration of species i in solution
- A/V = surface area over solution volume (cm^{-1})
- v = stoichiometric factor for species e' in glass waste form
- k_+ = forward rate constant ($\text{moles}/\text{cm}^2/\text{sec}$)
- n = exponent for pH dependence of rate
- Aff = affinity to dissolve (kcal/mole)
- R = gas constant
- T = temperature, Kelvins

to calculate the dissolution rate of the glass. Equation 1 is derived by starting with the general rate equation

$$\frac{dc_i}{dt} = \frac{A}{V} vk_+(a_{H^+})^n + \frac{A}{V} vk_-(a_{H^+})^n \quad (2)$$

where k_+ is the forward rate constant (dissolution) and k_- is the backward rate constant (dissolution). The principle of detailed balancing states that the net rate of a reaction is the difference between the rates of the forward and reverse reactions. At equilibrium, therefore, the rates are equal and $K_{eq} = \frac{k_+}{k_-}$

Equation 1, is derived assuming the principle of detailed balancing applies and that the driving force for a reaction is proportional to the extent of disequilibrium, or affinity of the reaction. The further from equilibrium, the stronger is the driving force. The origin of this rate equation is important because implicit in its use is the assumption that a *precipitation rate for glass* is not only possible, but must occur as the glass approaches "saturation."

Fortunately, application of this model to glass dissolution can be justified because the model is applied to the dissolution reaction of the gel layer and not the glass. A reverse reaction to precipitate the gel layer from solution is possible and highly likely. The affinity effect on glass dissolution corresponds to the effect of increasing the concentrations of species in solution, in particular silica, on the rate of precipitation of the hydrated silica-rich gel layer at the gel-water interface.

The models incorporate rate equation 1 into a general reaction path program that provides for speciation of the aqueous phases, and precipitation of secondary phases as they become saturated, and the effects of surface layer formation on solution chemistry.

A simplified method of calculating glass release rates is described in the following:

March 17, 1993.

TO: Ray Stout
FROM: William Bourcier
SUBJECT: Draft input for LLNL PA calculation, Glass Wasteform

Experimental and modeling work on borosilicate glass to date shows that the important parameters which need to be considered in order to predict radionuclide release rates from glass are temperature, exposed surface area, solution pH, and dissolved silica concentration in solution. Below we have supplied the equations and parameters needed to calculate conservative release rates of radionuclides from glass. We also include suggestions as to how to further simplify the model to make it appropriate for input into a first-cut comprehensive performance assessment model of a repository.

The rate equation commonly used to describe glass dissolution is:

$$R = sk_{T,pH}(1 - Q/K) \quad (1)$$

where R is the dissolution rate of glass in grams/day, s is the surface area of glass exposed to solution in m^2 , k is the rate constant for glass which is primarily a function of temperature and pH, Q is the concentration of dissolved silica, and K a thermodynamic parameter for borosilicate glass that is approximated here as the solubility product for amorphous silica. Each of these parameters must be known or estimated in order to calculate radionuclide release rates from glass. Suggested values for each of these parameters are as follows:

Surface Area, s

The surface area of uncracked glass in a DWPF canister is about 5m^2 . However, after being poured and cooled, the glass undergoes fracturing. Estimates for the increase in glass surface area due to fracturing range from 2 to 100 times the geometric surface area. A reasonable and conservative value to use is 25 (Baxter, 1983). A surface area of $125\text{m}^2/\text{canister}$ is therefore recommended.

Rate Constant, k

The rate constant has been measured over a range of pH and temperature conditions. The following data and regression equation describe the rate constant as a function of pH and temperature and is valid from 10 to 100°C and pH values from 1-12:

Table 1. $\log_{10}$ glass dissolution rate in $\text{g}/\text{m}^2/\text{day}$.

pH	T = 25	50	70
1	-1.25	0.02	0.51
2	-1.73	-0.68	-0.18
3	-2.21	-1.38	-0.87
4	-2.69	-2.08	-1.56
5	-3.17	-2.78	-2.25
6			-2.94
7	-4.53	-3.43	-2.3
8	-4.02	-2.92	-1.9
9	-3.51	-2.41	-1.5
10	-3	-1.9	-1.1
12	-1.98	-0.88	-0.3

$$\log_{10}\text{Rate}(\text{g glass}/\text{m}^2/\text{day}) = -0.00172029 - 0.0231246T + 0.00148569T^2 - 1.13605 \times 10^{-5}T^3 - 1.1558\text{pH} + 0.0812918\text{pH}^2 + 0.000137686\text{pH}^3 \quad (T \text{ in degrees C}).$$

Solution Chemistry, Q and K

The major effect of groundwater chemistry on the glass dissolution rate (other than pH) is the concentration of dissolved silica. In this simple model, Q equals the concentration of dissolved silica in the water contacting the glass. The local groundwater chemistry in the vicinity of the repository will likely be dominated by the host rocks (Wilder, 1992) and the silica concentration is therefore expected to be close to cristobalite saturation at the ambient temperature. Cristobalite is a common constituent of the host rocks at Yucca Mountain. Table 2 lists concentrations of silica in equilibrium with cristobalite at temperatures from 0 to 150°C from the thermodynamic database SUPCRT92 (Johnson, Oelkers, and Helgeson, 1992).

"K" in equation (1) for the waste glass is assumed equal to the equilibrium constant for amorphous silica in this simple model. K actually varies as a function of glass composition, but for most waste glass compositions, the experimentally determined value of K is of the same general magnitude but less than the value of K for amorphous silica. Our simplification therefore gives conservative estimates. Table 2 lists values of $\log_{10}K$ (in molality) for temperatures from 0 to 150°C. As an example, at 60°C, $Q/K = 10^{-3.02}/10^{-2.43} = 0.26$. The term $(1-Q/K) = (1-0.26)$ or 0.74. The glass reaction rate therefore is about 74% of the rate under silica-free conditions.

Table 2. Cristobalite and Amorphous Silica Solubilities

T°C =	0	25	60	90	100	150
Cristobalite	-3.89	-3.45	-3.02	-2.75	-2.68	-2.36
Amorphous Silica	-2.99	-2.71	-2.43	-2.26	-2.20	-1.98

Solution pH

Experimental studies of tuff-water interactions have shown that the pH of reacted J-13 water maintains a pH slightly higher than neutral (Knauss, Beiriger and Peifer, 1987). Therefore, for anticipated repository conditions, a slightly alkaline pH of about 8 is recommended as a substitute for the lack of any more rigorous calculation of groundwater chemistry. This pH value should be used to estimate rate constants for glass dissolution from Table 1 (it should also be consistent with any data for solubility limited radionuclide concentrations which are also highly dependent on pH). Note however that glass dissolution rates and radionuclide release rates are very sensitive to pH and nothing more than a qualitative estimate of release rates is possible without a more rigorous treatment of solution chemistry in the repository performance assessment model.

Temperature Dependence of Glass Dissolution Rate

Experiments have shown that glass dissolution rates follow an Arrhenius relation where $\text{rate} \propto e^{-E/RT}$ where R is the gas constant, T is temperature (Kelvins) and the activation energy (E) is about 20 kcal/mole. This corresponds roughly to dissolution rate increasing by a factor of 3 for a ten degree rise in temperature. This simple rule can be used to describe the effect of temperature on glass dissolution rate if the data in Table 1 cannot be explicitly used.

Radionuclide Content of Glass

Table 3 lists anticipated radionuclide contents for SRL glasses. More information on glass compositions is provided in the Wasteform Characteristics Report. Conservative estimates for release rates for radionuclides from the glass wasteform are given by multiplying the glass dissolution rate (R) by the weight fraction of radionuclide in the glass from Table 3.

Example Calculation

What is the rate of release of ^{235}U from one canister of glass at 70°C in cristobalite-saturated groundwater of $\text{pH}=8$? The rate constant for glass dissolution at 70°C and $\text{pH} = 8$ is $10^{-1.9} \text{ g/m}^2/\text{day}$. The affinity term $(1-Q/K)$ has a value of $(1-10^{-2.93}/10^{-2.37})$ or 0.72. The bulk dissolution rate of glass is therefore $0.0091 \text{ g/m}^2/\text{day}$. Surface area for one canister is 125 m^2 , so that the total rate of glass dissolution is $1.13 \text{ g/day/canister}$. Predicted ^{235}U content of SRL waste glass is 72.78 g/canister . Total weight of glass in a canister is 1682 kg so that the weight fraction of ^{235}U is 4.3×10^{-5} . Release rate of ^{235}U is therefore $1.13 \times 4.3 \times 10^{-5} = 4.89 \times 10^{-5} \text{ g/day}$ or $.018 \text{ g/year}$.

Recommended Simplifications

Assume a constant pH of 8 and cristobalite saturation of the groundwater. Use Table 1 to provide the rate constant as a function of temperature at $\text{pH} = 8$. Use Table 2 to provide the factor that accounts for the lowering of glass dissolution rate due to dissolved silica. This provides a simple function of glass dissolution rate with temperature and no other variables need to be considered.

Limitations

This simplified treatment of estimating glass dissolution rates provides conservative estimates for release rates of radionuclides. It ignores solubility limits of some radioactive species (such as the actinides) and instead uses the conservative assumption that the radionuclides will be released as fast as the glass structure breaks down. Experiments have shown that the actinides more commonly are included in alteration phases at the surface of the glass either as minor components of other phases or as phases made up predominantly of actinides. We do not take any credit for this process in this simple treatment. In order to perform accurate

estimates of solubility-limited release rates, we need to know detailed information on water chemistry (pH, Eh, etc.) which demands a much more complex PA model that explicitly accounts for coupled chemical interactions between all of the repository materials (spent fuel, glass, metals, etc.).

This simple model ignores all solution chemistry other than pH and silica concentration. We know from a variety of experiments that species such as dissolved Mg and Fe can change glass dissolution rates by up to several orders of magnitude. Mg decreases the rate, Fe increases the rate. We do not account for effects such as these in this model.

We also ignore vapor phase alteration of the glass. If a canister containing glass is breached and humid air reaches the glass, the glass will react and form a thick alteration rind composed of hydrated glass and secondary phases. The durability of this material with respect to later contact with liquid water may be much greater or much less than the unaltered glass. We do not account for this effect here.

Table 3. Radioisotope content per HLW container for borosilicate glass from the Savannah River Site (from Wasteform Characteristics Report, Table 6.14). Contents in grams of each isotope. Mass of glass in each canister is 1682 kilograms. Only elements with more than 1 gram per canister are reported here.

<u>Isotope</u>	<u>g/canister</u>	<u>Isotope</u>	<u>g/canister</u>
U-234	.549e1	Tc-99	.182e3
U-235	.727e2	Pd-107	.286e2
U-236	.174e2	Sn-126	.156e2
U-238	.312e5	Cs-135	.863e2
Np-237	.126e2	Cs-137	.499e3
Pu-238	.867e2	Ce-143	.401e3
Pu-239	.208e3	Ce-144	.309e1
Pu-240	.381e2	Nd-144	.411e3
Pu-241	.162e2	Pm-147	.261e2
Pu-242	.321e1	Sm-147	.877e2
Am-241	.321e1	Sm-148	.192e2
Cm-244	.132e1	Sm-149	.742e1
Se-79	.243e1	Sm-151	.941e1
Rb-87	.996e1	Eu-154	.229e1
Sr-90	.343e3	Eu-155	.102e1
Zr-93	.444e3		

References

- Baxter, R. G. (1983). Description of defense waste processing facility reference waste form and canister. (Report No. DP-1606). Savannah River Plant.
- Johnson, J. W., E. H. Oelkers, and H. C. Helgeson, (1992) SUPCRT92: A software package for calculating the standard molal thermodynamic properties of minerals, gases, aqueous species, and reactions from 1 to 5000 bar and 0 to 1000°C. *Computers and Geosciences*, 18, 899-947.
- Knauss, K. G., W.J. Beiriger, and D. W. Peifer (1987). Hydrothermal interaction of solid wafers of Topopah Spring Tuff with J-13 water at 90° and 150°C using Dickson-type, gold-bag rocking autoclaves: Long-term experiments. (UCRL 53722). Lawrence Livermore National Laboratory.
- Wilder, D. G. E. (1992). Near-field environment report volume I: Technical basis for BES design (UCRL-LR No. 107476), Lawrence Livermore National Laboratory.

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.6.1 Crud

The corrosion products in the coolant of a power reactor form an activated radionuclide subset which, when deposited on the surfaces of a spent fuel assembly, are termed "crud." In reference (1), the crud subset of radionuclides with their specific surface activity was provided as "worst case" estimates for PWR and BWR assemblies; these were estimated irrespective of fuel manufacturer, handling of assemblies, reactor zone, axial location, and burnup. Table I below combines crud data from reference (1) and their half-life data from reference (2). Crud composition and structure are described extensively in reference (2) and summarized in reference (1) as being of two types:

- (a) a fluffy, easily removed crud found usually on BWR rods and is composed mainly of hematite (Fe_2O_3)
- (b) A tenacious, tightly bound crud found usually on PWR and is composed mainly of a nickel-substituted spinel ($\text{Ni}_x\text{Fe}_{3-x}\text{O}_4$ with $x \approx 1$).

Table I

Nuclide	Half-life (days)	Crud Activity Density $\mu\text{Ci}/\text{cm}^2$ (reactor shutdown)	
		PWR Fuel	BWR Fuel
^{51}Cr	28	391	35
^{54}Mn	312	380	635
^{58}Co	71	1400	63
^{59}Fe	45	300	87
^{60}Co	1924	140	1250
^{65}Zn	244	N/A	56
^{95}Zr	65	36	30

Because of the relatively short half-lives of the crud radionuclides, the release of crud appears as a potential problem primarily during transportation (Ref. 1 and 2) and preclosure radiological design of repository facilities (Ref. 3). From Table I, the longest half-life is ~5.3 yr for the crud radionuclide ^{60}Co , which at the end of 1,000 years would have an activity decay multiplier of 1.6×10^{-57} . Thus, the total activity of crud inventory would be small at 1,000 years, even though there exist several square miles of assembly surface (1 sq. mile $\sim 2.6 \times 10^{10} \text{ cm}^2$).

References

1. R.P. Sandoval, R.E. Einziger, H. Jordan, A.P. Malinauskas, and W.J. Mings, "Estimate of the CRUD Contribution to Shipping Cask Containment Requirements," *Nucl. Tech.*, **98**, 196 (1992).
2. R.P. Sandoval, R.E. Einziger, H. Jordan, A.P. Malinauskas, and W.J. Mings, "Estimate of the CRUD Contribution to Shipping Cask Containment Requirements," SAND88-1358, TCC-0811, Sandia National Laboratories (Jan. 1991).
3. L.J. Jardine, "Preclosure Radiological Safety Analysis for Normal Conditions of the Yucca Mountain Repository," SLTR87-7013, Sandia National Laboratories (April, 1988).

3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT

3.1 SPENT FUEL CLADDING FAILURE

3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS

3.1.2 FAILURE MODELS

3.2 SPENT FUEL OXIDATION

3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS

3.2.2 OXIDATION MODELS

3.3 SPENT FUEL FISSION GAS RELEASE

3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE

3.3.2 FISSION GAS RELEASE MODELS

3.4 SPENT FUEL DISSOLUTION

3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION

3.4.1.1 DISSOLUTION RATES

3.4.1.2 SOLUBILITY LIMITS

3.4.1.3 SOLUBILITY LIMITING PHASES

3.4.2 DISSOLUTION MODELS

3.5 GLASS DISSOLUTION

3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION

3.5.2 GLASS DISSOLUTION MODELS

3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES

3.6.1 CRUD

3.6.2 HARDWARE

3.6.3 CLADDING

3.6.2 Hardware

See Section 2.2.2.5

- 3. SCIENTIFIC BASIS FOR PREDICTIVE MODEL DEVELOPMENT
 - 3.1 SPENT FUEL CLADDING FAILURE
 - 3.1.1 EXPERIMENTAL PARAMETERS FOR FAILURE MODELS
 - 3.1.2 FAILURE MODELS
 - 3.2 SPENT FUEL OXIDATION
 - 3.2.1 EXPERIMENTAL PARAMETERS FOR OXIDATION MODELS
 - 3.2.2 OXIDATION MODELS
 - 3.3 SPENT FUEL FISSION GAS RELEASE
 - 3.3.1 EXPERIMENTAL PARAMETERS FOR FISSION GAS RELEASE
 - 3.3.2 FISSION GAS RELEASE MODELS
 - 3.4 SPENT FUEL DISSOLUTION
 - 3.4.1 EXPERIMENTAL PARAMETERS FOR DISSOLUTION
 - 3.4.1.1 DISSOLUTION RATES
 - 3.4.1.2 SOLUBILITY LIMITS
 - 3.4.1.3 SOLUBILITY LIMITING PHASES
 - 3.4.2 DISSOLUTION MODELS
 - 3.5 GLASS DISSOLUTION
 - 3.5.1 EXPERIMENTAL PARAMETERS FOR GLASS DISSOLUTION
 - 3.5.2 GLASS DISSOLUTION MODELS
 - 3.6 OTHER RELEASE SOURCES OF RADIONUCLIDES
 - 3.6.1 CRUD
 - 3.6.2 HARDWARE
 - 3.6.3 CLADDING

Interdepartmental letterhead

Mail Station L-352

CHEMISTRY & MATERIALS SCIENCE DEPARTMENT

Ext: 2-0456

Corrosion & Electrochemical Processes Section
Materials Division

MEMORANDUM - January 23, 1991
91-02RVKm

TO: Ray Stout, L-201
FROM: Rich Van Konynenburg *Rich Van Konynenburg*
SUBJECT: *Carbon-14 Information for Preliminary Waste Form Characteristics Document*

In response to your request, here is some information about carbon-14 associated with spent fuel:

According to my current best estimates, the inventory of carbon-14 in spent fuel is as follows:

BWR

Total: $3.71 \times 10^{-5} \frac{\text{curies per MTIHM}}{\text{MWd/MTIHM}}$

Zircaloy only: $1.38 \times 10^{-5} \frac{\text{curies per MTIHM}}{\text{MWd/MTIHM}}$

(Nearly half of the BWR zircaloy inventory is in the fuel channel.)

PWR

Total: $3.03 \times 10^{-5} \frac{\text{curies per MTIHM}}{\text{MWd/MTIHM}}$

Zircaloy only: $5.45 \times 10^{-6} \frac{\text{curies per MTIHM}}{\text{MWd/MTIHM}}$

Based on the work of Smith and Baldwin, as much as 2% of the total spent fuel ^{14}C inventory can be released at 350°C in air in 8 hours.

According to recent work in Germany (D. Kopp and H. Münzel, "Release of Volatile Carbon-14 Containing Products from Zircaloy," *J. Nucl. Mater.* 173, 1-6 (1990), oxygen is necessary for volatile ^{14}C release to occur.

To be conservative, one could assume that sufficient oxygen would be present in the argon fill gas in a waste container to bring about release of the entire 2% of the inventory into the fill gas prior to breach of the container.

RVK:mlm

Distribution

Author File

Ballou, L.

Clarke, W.

L-206

L-352

CONSTANT TEMPERATURE TESTS, THICK OXIDE

Harry 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

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

Adachi, T., M. Ohnuki, et al. "Dissolution Study of Spent PWR Fuel: Dissolution Behavior and Chemical Properties on Insoluble Residues," *J. Nucl. Mat.*, 174, 60-71 (1990).

Aronson, S., R.B. Roof, and J. Belle. "Kinetic Study of the Oxidation of Uranium Oxide," *J. Chem. Phys.* 27, 137 (1957).

Bates, J.K., "Identification of Secondary Phases Formed During Unsaturated Reaction of UO_2 with EJ-13 Water," Materials Research Society Symposium proceedings 176, 499 (1990). NAA.910128.0029

Bates, J.K., "Integrated Glass Alteration Tests," presented to the Nuclear Waste Technical Review Board, (August, 1990). NAA.920302.0059

Bates, J.K., et al., "NNWSI Waste Form Testing at Argonne National Laboratory Semi-Annual Report, January - June, 1988," UCRL-21000-88-1, (January, 1991). NAA.910128.0062

Beyer, C.E., in "Status and Future Direction of Spent Fuel ATM Acquisition and Characterization meeting" in Richland, Washington, (March, 1989). NAA.920401.0114

Boltzmann, L. "Lectures on Gas Theory," translated by S.G. Brush, Univ. of California Press, Berkeley, CA (1964). NAA.920401.0117

Bourcier, W.L., "Modeling of Glass Dissolution," presented to the Nuclear Waste Technical Review Board, (August, 1990). NAA.920302.0057

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

Carslaw, H.S. and J.C. Jaeger. "Conduction of Heat in Solids," second edition, Oxford University Press, New York (1959). NAA.900522.0259

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). NAA.900427.0017

Einzig, R.E. and H.C. Buchanan. "Long-Term, Low Temperature Oxidation of PWR Spent Fuel," Westinghouse Hanford Co. Report, WHC-EP-0070 (June, 1987). NAA.900620.0297

Einzig, R.E. and R.E. Woodley. "Predicting Spent Fuel Oxidation States in a Tuff Repository," Westinghouse Hanford Co. Report HEDL-SA-3627, (April, 1985). NAA.870915.0073

Einzig, R.E. "Technical Test Description of Activities to Determine the Potential for Spent Fuel Oxidation in a Tuff Repository," Westinghouse Hanford Co. Report HEDL-7540 (June, 1985). NAA.920302.0060

Einzig, R.E. "Test Plan for Series 2 Thermogravimetric Analyses of Spent Fuel Oxidation," Westinghouse Hanford Co. Report HEDL-7556 (February, 1986). NAA.920302.0061

Einzig, R.E., "Spent Fuel Oxidation," presented to Nuclear Waste Technical Review Board, (August, 1990). NAA.920302.0054

Einzig, R.E., "Test Plan for Long-Term, Low-Temperature Oxidation of Spent Fuel, Series 1," Westinghouse Hanford Co. Report HEDL-7560 (June, 1986). NAA.920302.0062

Finch, R. and R. Ewing, "Alteration of Natural UO_2 Under Oxidizing Conditions from Shinkolobwe, Katanga, Zaire: A Natural Analogue for the Corrosion of Spent Fuel," *Radiochim. Acta*, 52/53, 395-401 (1991). NAA.900507.0149

Gleiter, H. and B. Chalmers. "High-Angle Grain Boundaries," in *Prog. in Mat. Sci.*, Vol. 16, Pergamon Press, New York (1972).

Grambow, B. "Spent Fuel Dissolution and Oxidation: An Evaluation of the Literature," SKB Tech. Rept. 89-13, Svensk Kärnbränslehantering AB, Stockholm, 42 p. (1989). NAA.891013.0094

Grambow, B., R. Forsyth, et al. "Fission Product Release from Spent UO_2 Fuel Under Uranium-Saturated Oxidic Conditions," *Nucl. Tech.*, 92, 204-213 (November, 1990).

Gray, W. and Wilson, C., "Effects of Water Composition and Temperature on the Dissolution Rate of UO_2 ," presented at 1990 Spent Fuel Workshop, Gull Harbor, Manitoba, Canada, (1990). NNA.910821.0008

Gray, W. and Wilson, C., "Spent Fuel Grain Boundary Inventory and Testing the Congruency of UO_2 Matrix Dissolution of Spent Fuel," presented at 1990 Spent Fuel Workshop, Gull Harbor, Manitoba, Canada, (1990). NNA.910821.0009

Knauss, K.G., "Experimental Basis for Glass Dissolution," presented to the Nuclear Waste Technical Review Board, (August, 1990).

Leider, H.R., S.N. Nguyen, R.B. Stout, H.C. Weed, "Estimating the Time for Dissolution of Spent Fuel Exposed to Unlimited Water," LLNL Report #UCRL-ID-107289, (Dec., 1991). NNA.911211.0002

Notz, K.J., "Characteristics of Potential Repository Waste," DOE/RW-0184-R1, V.1 (draft), (July, 1990). NNA.900427.0056

Notz, K.J., T.D. Welch, R.S. Moore, and W.J. Reich, "Preliminary Waste Form Characteristics," ORNL-TM-11681 (draft) (September, 1990). NNA.901113.0010

Olander, D.R. "Combined Grain-Boundary and Lattice Diffusion in Fine-Grained Ceramics," *Advances in Ceramics*, 17, 271 (1986).

Palmer, R.A., "West Valley Demonstration Project High Level Waste Qualification Activities," presented to the Nuclear Waste Technical Review Board, (August, 1990). NNA.920302.0056

Plodinec, M.J., "Defense Waste Processing Facility High Level Waste Qualification Activities," presented to the Nuclear Waste Technical Review Board, (August, 1990). NNA.920302.0055

Puigdomenech, I., and J. Bruno, "Modeling Uranium Solubilities in Aqueous Solutions: Validation of a Thermodynamic Data Base for the EQ3/6 Geochemical Codes," SKB technical report 88-21, (October, 1988). NNA.890425.0186

Roddy, W.J., 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). NNA.891016.0145

Site Characterization Plan, Yucca Mountain Site, Nevada, Research and Development Area, DOE/RW-0199, Vol. III, Part A, (December, 1988).

Slattery, J.C. "Momentum, Energy and Mass Transfer in Continua," R.E. Krieger Pub. Co., New York (1978). NNA.920401.0115

Smith, H.D., "Spent Fuel Cladding Degradation," presented to the Nuclear Waste Technical Review Board, (August, 1990). NNA.920302.0052

Stout, R.B. and L. Thigpen. "A Model for the Deformations and Thermodynamics of Materials Involving a Microcrack Kinetics," *Cryst. Latt. Defects and Amorph. Mat.*, 10, 129 (1984).

Stout, R.B. "Modeling the Deformations and Thermodynamics for Materials Involving a Dislocation Kinetics," *Cryst. Latt. Defects*, 9, 65 (1981).

Stout, R.B. "Statistical Model for Particle-Void Deformation Kinetics in Granular Materials During Shock Wave Propagation," Lawrence Livermore National Laboratory Report UCRL-101623 (July, 1989). NNA.900517.0267

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

Thomas, L., O. Slagle, and R. Einziger, "Nonuniform Oxidation of LWR Spent Fuel in Air," *J. Nucl. Mat.*, 184 117-126 (1991). NNA.910509.0071

Thomas, L., R. Einziger, and R. Woodley, "Microstructural Examination of Oxidized Spent PWR Fuel by Transmission Electron Microscopy," *J. Nucl. Mat.*, 166, 243-251 (1989). NNA.900709.0482

Thomas, L.E., R.E. Einziger, and R.E. Woodley. "Microstructural Examinations of Oxidized Spent Fuel by Transmission Electron Microscopy," *J. Nucl. Mat.*, in press. NML 880707.0043.

Wilson, C.N., "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). NNA.920302.0064

Wilson, C.N., "Results from Cycles 1 and 2 on NNWSI Series 2 Dissolution Tests," HEDL-TME 85-22, (May, 1987). NNA.900716.0358

Wilson, C.N., "Results from the NNWSI Series 3 Spent Fuel Dissolution Tests," PNL 7170, (June, 1990). NNA.900329.0142

Wood, P. and G.H. Bannister. "Investigation of the Mechanism of UO₂ Oxidation in Air: The Role of Grain Size," Proc. Workshop on Chemical Reactivity of Oxide Fuel and Fission Product Release, K.A. Simpson and P. Wood, eds., Berkeley Nuclear Lab., U.K., p. 19 (1987). NNA.920401.0116

Woodley, R.E., "The Characteristics of Spent LWR Fuel Relevant to its Storage in Geologic Repositories," HEDL-TME 83-28, (October, 1983). NNA.870406.0337

Woodley, R.E., R.E. Einziger, and H.C. Buchanan. "Measurement of the Oxidation of Spent Fuel Between 140° and 225°C by Thermogravimetric Analysis," Westinghouse Hanford Co. Report WHC-EP-0107 (September, 1988). NNA.880927.0069