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DOE Spent Nuclear Fuel Information In Support of TSPA - VA

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DOE Spent Nuclear Fuel Information In Support of TSPA - VA

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This report was not produced under a quality assurance program that satisfies the requirements of the National Spent Nuclear Fuel Program and DOE/RW-0333P, Office of Civilian Radioactive Waste Management Quality Assurance Requirements Description. Therefore, the data in the report are not considered qualified and are not to be relied upon to address safety and waste isolation issues until an accepted qualification process has been completed.

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ABSTRACT

RW has started the viability assessment (VA) effort to determine the feasibility of Yucca Mountain as the first geologic repository for spent nuclear fuel (SNF) and high-level waste. One component of the viability assessment will be a total system performance assessment (TSPA), based on the design concept and the scientific data and analysis available, describing the repository's probable behavior relative to the overall system performance standards. Thus, all the data collected from the Exploratory Studies Facility to-date have been incorporated into the latest TSPA model. In addition, the Repository Integration Program, an integrated probabilistic simulator, used in the TSPA has also been updated by Golder Associates Incorporated at December 1997. To ensure that the Department of Energy-owned (DOE-owned) SNF continues to be acceptable for disposal in the repository, it will be included in the TSPA-VA evaluation.

A number of parameters are needed in the TSPA-VA models to predict the performance of the DOE-owned SNF materials placed into the potential repository. This report documents all of the basis and/or derivation for each of these parameters. A number of properties were not readily available at the time the TSPA-VA data was requested. Thus, expert judgement and opinion was utilized to determine a best property value. The performance of the DOE-owned SNF will be published as part of the TSPA-VA report.

Each DOE site will be collecting better data as the DOE SNF program moves closer to repository license application. As required by the RW-0333P, the National Spent Nuclear Fuel Program will be assisting each site in qualifying the information used to support the performance assessment evaluations.

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CONTENTS

FIGURES

TABLES

ACRONYMS

ATR	Advanced Test Reactor
BAPL	Bettis Atomic Power Laboratory
BOL	beginning of life
DFA	driver fuel assembly
DOE	Department of Energy
EM	Office of Environmental Management
EPA	Environmental Protection Agency
ESF	Exploratory Studies Facility
FD-Hanford	Fluor-Daniel Hanford
FERMI	Enrico Fermi Reactor
FFTF	Fast Flux Test Facility
FRR	foreign research reactor
FSV	Fort St. Vrain
HEU	>20% ^{235}U equivalent
HLW	high-level waste
HTGR	High Temperature Gas Cool Reactor
HWCTR	Heavy Water Components Test Reactor
INEEL	Idaho National Engineering and Environmental Laboratory
LA	license application
LDP	Large Disposal Package
LEU	<5% ^{235}U equivalent
LWBR	Light Water Breeder Reactor
LWR	light water reactor
M&O	Management and Operation (Contractor)
MCO	multi-canister overpack
MEU	>5% <20% ^{235}U equivalent
MOX	mixed oxide

MTHM	metric tons heavy metal
MURR	Missouri University Research Reactor
MWd/kgU	SNF burnup in megawatt-day/kilograms uranium
NRC	Nuclear Regulatory Commission
NSNF	National Spent Nuclear Fuel (Program)
OD	outside diameter
ORIGEN	Oak Ridge Isotope Generation
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor
PB	Peach Bottom
PBF	Power Burst Facility
PWR	pressurized water reactor
RINSC	Rhode Island Nuclear Science Center
RW	Office of Civilian Radioactive Waste Management
SiC	silicon carbide
SNF	spent nuclear fuel
SPR	Single Pass Reactor
SRE	Sodium Reactor Experiment
SRS	Savannah River Site
TESS	TRW Environmental Safety Systems Incorporated
TFA	test fuel assembly
TMI	Three Mile Island
TREAT	Transient Reactor Test
TRIGA	Training Research Isotopes – General Atomic
TSPA	total system performance assessment
UC	uranium carbide
U-Mo	uranium molybdenum
U-Zr	uranium zirconium
VA	viability assessment
WSRC	Westinghouse Savannah River Company

DOE Spent Nuclear Fuel Information in Support of TSPA-VA

1. INTRODUCTION

For Department of Energy (DOE) spent nuclear fuel (SNF) to be considered for disposal in the repository, the performance of the packaged fuels must be evaluated in a performance assessment. In 1997, the Office of Civilian Radioactive Waste Management's (RW) management and operation (M&O) contractor TRW Environmental Safety Systems Incorporated (TESS) helped the DOE's Office of Environmental Management (EM) to conduct a total system performance assessment (TSPA) on the DOE-owned SNF. The analyses were conducted using an improved version of the TSPA model developed for the commercial spent nuclear fuels and high-level wastes (HLW) in fiscal year 1995. The results were very promising in that the majority of the DOE-owned SNF appears to be directly disposable in the repository.

Since those analyses, RW has started the viability assessment (VA) effort to determine the feasibility of Yucca Mountain as the first geologic repository for SNF and HLW. One component of the viability assessment will be a total system performance assessment, based on the design concept and the scientific data and analysis available, describing the repository's probable behavior relative to the overall system performance standards. Thus, all the data collected from the Exploratory Studies Facility (ESF) to-date have been incorporated into the latest TSPA model. In addition, the Repository Integration Program, an integrated probabilistic simulator, used in the TSPA has also been updated by Golder Associates Incorporated in December 1997. To ensure that the DOE-owned SNF continues to be acceptable for disposal in the repository, it will be included in the TSPA-VA evaluation.

A number of parameters are needed in the TSPA-VA models to predict the performance of the SNF materials placed into the potential repository. This report intends to document all of the basis and/or derivation for each of these parameters. A number of properties were not readily available at the time the TSPA-VA data was requested. Thus, expert judgement and opinion was utilized to determine a best property value. Each site will be collecting better data as the DOE SNF program moves closer to repository license application. As required by the RW-0333P, each of the sites will be qualifying the information used to support the performance assessment evaluations.

2. DOE SNF GROUPING AND RATIONALE

2.1 Background on DOE SNF Grouping

In January 1997, the DOE-EM/RW Repository Task Team published a report titled *Grouping Method to Minimize Testing for Repository Emplacement of DOE SNF* [Reference 35]. The report provided the background on the many DOE SNF types (more than 200) located at the various DOE sites and why grouping of DOE-owned SNF is necessary for repository disposition. In addition, the report also suggested 11 groups should represent the DOE-owned SNF and gave reasons for the 11 fuel groups. Since the publication of that report, more discussion has occurred in the DOE-EM SNF program and further refinement of the original grouping has been completed. This section will summarize the justifications for using 16 DOE SNF groups to represent the DOE SNF inventory for the repository TSPA.

The main goal of grouping the DOE SNF is to minimize the data-gathering effort to support DOE SNF management and disposal without increased risk to the public, environment, or worker safety. As indicated in the grouping report, the data needs required to meet the Nuclear Regulatory Commission (NRC) and Environmental Protection Agency (EPA) regulations were evaluated. Two fuel parameters, fuel matrix and cladding, were identified to have primary influence on the behavior of DOE SNF. These two are: (a) release rate, and (b) time-to-failure (i.e., the fuel's chemical and physical stability). Seven other parameters (burnup, initial enrichment, cladding integrity, fuel geometry, radionuclide inventory, fission gas release, and moisture content) were identified as having only secondary influences on fuel behavior.

Based on these findings, the report suggested grouping the DOE SNF into 11 groups for testing purposes. However, the 11 groups suggested are inconvenient for other analysis needs such as criticality evaluations in support of repository disposal.

Subsequent discussion among the DOE SNF programs proposed that the DOE SNF inventory be first reduced to 34 DOE SNF groups based on fuel matrix, cladding, cladding condition, and enrichment. These parameters are the basis used in selecting the SNF grouping as indicated in the center of Figure 2-1.

From these 34 DOE groups, it was determined that they may be further reduced to support both TSPA and criticality analyses. Specifically, the 34 groups of SNF were further reduced to 16 categories for the total system performance assessment and 13 categories for criticality analyses purposes. The rationale used to reduce the groups further for TSPA and criticality is provided below. The condensed DOE SNF groups, the TSPA categories, and criticality analyses categories are shown in Figure 2-1. The representative fuel in each condensed group was selected based generally on the quantity of the SNF within that specific group.

The TSPA-VA will evaluate 15 categories of DOE-owned SNF. The 15 fuel categories and their representative fuels are shown in Table 2-1. Although 16 fuel categories are listed on the table, category 14 will be treated (due to the reactive nature of the metallic sodium) prior to disposal and thus will not be included in the analyses. The Bettis Atomic Power Laboratory (BAPL) will be providing information concerning the classified naval fuels. Thus, category 15 will not be discussed in this document.

Evaluations of categories 1 through 13 and category 16 fuels will be completed in conjunction with high-level waste glass incorporated in the repository waste packages using a co-disposal concept,

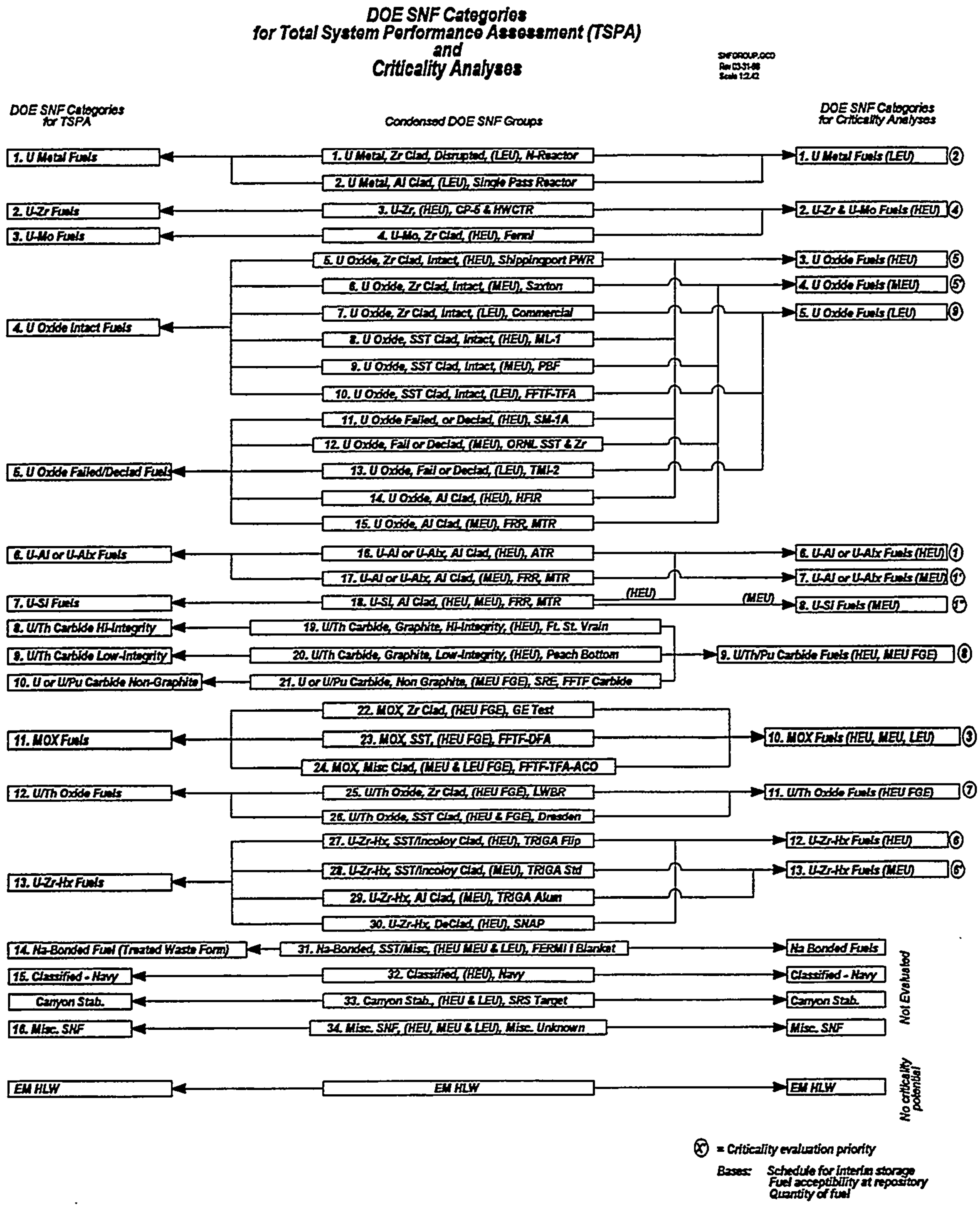

Figure 2-1. DOE SNF condensed groups, categories for TSPA, and criticality analyses.

Table 2-1. DOE spent nuclear fuel categories.

Fuel Category	Fuel Matrix	Typical Fuel in the Category	Comment
1	U-metal	N-Reactor fuel	
2	U-Zr	Heavy Water Components Test Reactor fuel	
3	U-Mo	FERMI (Enrico Fermi Reactor) Fuel	
4	U-oxide intact	Commercial PWR fuel Shippingport PWR fuel	
5	U-oxide failed/declad	Three Mile Island (TMI) fuel	
6	U-Al Or U-Alx	Advanced Test Reactor (ATR) fuel Foreign Research Reactor (FRR) fuel	
7	U-Si	Foreign Research Reactor (FFR) fuel	
8	U/Th carbide hi-integrity	Fort St. Vrain (FSV) fuel	
9	U/Th carbide low-integrity	Peach Bottom fuel	
10	U or U/Pu carbide nongraphite	Fast Flux Test Facility (FFTF) carbide fuel	
11	MOX	Fast Flux Test Facility (FFTF) oxide fuel	
12	U/Th oxide	Shippingport LWBR Fuel	
13	U-Zr-Hx	Training Research Isotopes- General Atomic (TRIGA) fuel	
14	Na-bonded	FERMI I Blanket	Will be treated. Not part of TSPA-VA analyses
15	Classified-Navy	Navy	Info by Navy
16	Misc. SNF	Misc. fuel	

if needed, as part of the TSPA-VA evaluation. The Navy fuels, category 15, will be considered in the TSPA based on the source term at the container boundary over the evaluation time period as provided by BAPL.

RW indicated recently that the TSPA-VA analyses would take some credit for the cladding on the commercial SNF. Depending on the availability of cladding information for DOE-owned fuels, DOE-EM may consider taking similar cladding credit for some of the DOE-owned SNF as well.

2.2 Reason for grouping the DOE SNF

The licensing application (LA) long-term performance predictions necessitate a certain knowledge of the fuel and provide the basis for the data needs. These data needs can be straightforward, such as dimensions, or can require significant technical insight, such as how the fuel behaves in the repository environment. The first type relates to the physical characteristic of the fuel, while the second type relates to the performance or behavioral characteristic of the fuel. Both types of characteristic information are needed for the TSPA to demonstrate that the DOE fuels do not increase the risk of higher dosees to the public in the postclosure period.

DOE has more than 200 varieties of SNF. Some of these varieties are quite close in terms of characteristics, while others vary considerably. It is too expensive and unnecessary for DOE to provide documentation in support of the TSPA for every individual fuel type. It is necessary to group these fuels to demonstrate that DOE SNF meets the long-term performance requirements as part of the repository licensing application for final disposal. Many of the 200 fuel types have a very limited volume or number of elements. With small numbers, bounding is more efficient. The intent of Section 2.3 is to present the basis for grouping of the DOE fuels so that the characteristics of limited numbers of DOE SNF will either bound or represent a particular characteristic of the whole group.

2.3 DOE SNF Grouping Basis

The NRC, DOE, and EPA regulations are the basis for the LA mentioned above. Based on these regulations, RW developed a document titled *OCRWM Data Needs for DOE Spent Nuclear Fuel* [Reference 36], which contained generic information needs that the owner of the DOE-owned SNF must provide. A total of 87 data needs were identified in the document. A number of the data needs were directly related to the performance, or properties, of the fuel. The remaining requirements apply to the manufacture of the SNF and expected performance of the canister and its components or other aspects of SNF disposal such as transportation. These requirements were based solely on regulatory requirements. The National Spent Nuclear Fuel (NSNF) Program met with the Hanford, Idaho National Engineering and Environmental Laboratory (INEEL), and Savannah River (SRS) sites' TSPA and criticality experts to determine fuel characteristics needed for demonstrating regulatory compliance in the typical analyses. Using the RW data needs document, the data are broken into: 1) physical characteristics such as dimensions, fuel meat volume, and void fractions; 2) radionuclide inventories; and 3) long-term degradation and failure rate. All the TSPA requirements were considered and covered by the regulatory needs, so TSPA requirements were not covered separately. Similarly, the criticality analyses needs could be broken into: 1) physical characteristics such as dimensions, fuel compositions, and cladding; 2) radionuclide inventories; and 3) long-term degradation and releases.

Based on these needs, the methodology used in the development of DOE SNF grouping is shown in Figure 2-2.

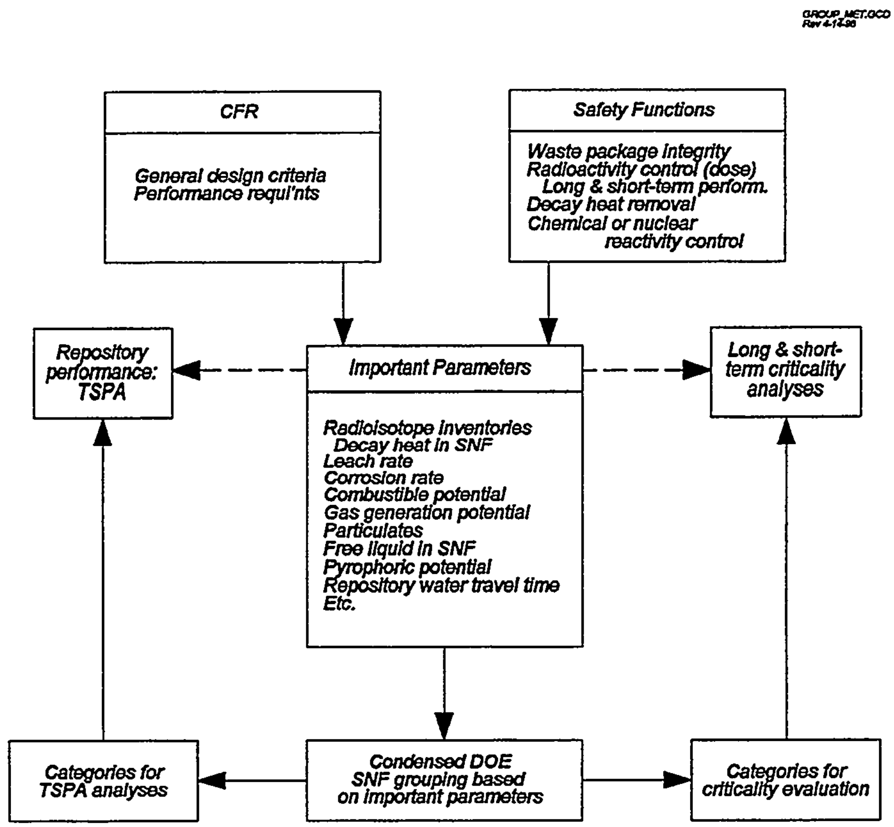

Figure 2-2. DOE SNF grouping methodology.

2.3.1 Grouping Rationale

The following sections provide the rationale for reducing the 34 SNF groups for the purpose of TSPA and criticality evaluation. As indicated above, the DOE SNF was placed into the 34 groups based on the fuel matrix, cladding, and enrichment. These parameters were selected based on their influence on the overall performance in the repository. The following summarizes the information for use in the TSPA-VA analyses.

Fuel Grouping for TSPA-VA.

TSPA Categories—Coupled with radionuclide inventory, the radionuclide release rate forms the source term for use in the TSPA to determine the dose to the public from SNF disposal. The radionuclide release rate is the product of the intrinsic release per unit surface area times the available surface area. For very dense fuel, where the grain boundary dissolution is not expected to be significantly different than matrix dissolution, the surface area is just the geometric area adjusted with some roughness factor. If leachant can possibly enter grain boundaries and/or separate grains, then surface area, and hence release rate, can be significantly increased. Both the intrinsic dissolution of the matrix and grain boundary effects are dependent on the microstructure of the fuel. Preliminary data on unirradiated fuel has indicated that the release mechanism and response to water conditions are significantly different for metal and oxide fuels (unpublished data from PNNL studies), which in turn are different from another matrix such as graphite or uranium zirconium hydride fuels. Based on these results, the following fuel categories were determined to be appropriate for the purpose of TSPA evaluations. A DOE SNF release rate test program has been initiated to confirm that the categories selected are appropriate.

Category 1 - Uranium Metal Fuels: The majority of this category consists of zirconium-clad N-Reactor fuel, with a small amount of aluminum-clad Single Pass Reactor fuel. Enrichments are below 2% ^{235}U. The majority of the fuels have low burnups. Some uranium target materials are also included in this group.

Category 2 - Uranium-Zirconium Fuels: The U-Zr fuels are placed into its category because of its microstructure at the grain boundaries. It is uncertain if there will be preferential attacks on the grain boundaries that could result in a large increase in surface area. However, the zirconium could stabilize the uranium metal and thus this category could perform differently than the U-metal fuels. The majority of this category consists of zirconium-clad Heavy Water Components Test Reactor (HWCTR) driver assemblies. Enrichments are typically 93% ^{235}U. The uranium in the HWCTR assemblies is alloyed with 90.7 wt% zirconium.

Category 3 - Uranium Molybdenum Fuels: Similarly, the U-Mo fuels are placed into their own category because of the uranium and molybdenum structure. One study on this unirradiated alloy indicated that uranium alloyed with 10% molybdenum corroded at 1% of the rate of pure uranium. But once corrosion starts, molybdenum causes stress fractures and crazing. This increases the matrix porosity and surface area and thus potentially increases the dissolution rate. This category consists of only the center fuel section of the Fermi driver fuel subassembly.

The lower and upper axial blankets have been cropped off and will be treated separately. Enrichments are typically ~25% ^{235}U. The uranium is alloyed with 10% molybdenum.

Category 4 - Uranium Oxide Intact Fuel: This category consists of the fuels removed from commercial reactors or test fuel with uranium oxide matrices similar to commercial spent fuels. This category is modeled as performing like the commercial spent nuclear fuels since all the fuels are fabricated using similar techniques and are all in the form of U-oxide. Since enrichment should not alter the dissolution rate for fuels with the same matrix, enrichments from the typical ~1–2% commercial ranges to the 93% ^{235}U fuel from the Shippingport Pressurized Water Reactor are included in this category.

Category 5 - Uranium Oxide Failed/Declad Fuels: This category consists of the fuels removed from commercial reactors or test fuels with uranium oxide matrices like the commercial spent fuels that have been damaged, have failed cladding, or are declad. This category is modeled as performing like the commercial spent nuclear fuels but potentially with a much higher fuel surface area due to the damage or the physical state (small pieces of disrupted fuel) of the fuel. Since enrichment should not alter the dissolution rate for fuels with the same matrix, enrichments from the typical of ~1–2% commercial range (such as Three Mile Island Reactor fuels) to the 93% ^{235}U fuel from the High Flux Isotope Reactor are included in this category.

Category 6 - Uranium Aluminum or Uranium Aluminide Fuel: This category consists of fuels with the uranium-aluminide dispersed in a continuous aluminum phase. This category may perform better than the pure U-metal fuel depending on the continuity of the primary aluminum phase and the release rate from each of the phases. Foreign research reactor fuels make up a large part of the uranium aluminide fuel in this category. Enrichment level varies from about 11% to 93% ^{235}U.

Category 7 - Uranium Silicide Fuel: This category consists of fuels with the uranium-silicide dispersed in a continuous aluminum phase. The U_3Si_2 fuel may perform differently than the uranium aluminide fuel and thus is placed in its own category. Depending on the continuity of the primary aluminum phase and the release rate from the U_3Si_2 phase, performance of this category may be better than the U-metal fuels. Foreign research reactor fuels make up a large part of the U-Si fuels in this category. Enrichment level varies from about 8% to 93% ^{235}U, but the majority of the enrichments are less than 20% ^{235}U.

Category 8 - Uranium/Thorium Carbide High Integrity Fuel: This category primarily consists of fuel from the Fort St. Vrain (FSV) reactor. The fuels from core 2 of the Peach Bottom (PB) reactor and a small amount of fuel from the General Atomic Gas-Cooled Reactor are also included in this category. The fuel is in the form of carbide particles coated with layers of pyrolytic carbon and silicon carbide (SiC) [Note: SiC coating is for the FSV only], bonded together by a carbonaceous matrix material. Two types of particles are used — fissile and fertile. The fissile particles contain thorium and ~93% enriched uranium. The fertile particles contain only thorium. One difference between the FSV and PB fuels is that the PB particles lack the silicon carbide coating. However, the fuel particles in these fuel assemblies are in excellent condition. Thus, the silicon carbide layer should provide a very slow release rate. This category should perform much better than the pure U-metal fuel. Effective enrichment (including the ^{233}U) level at the end of life varies from about 78% to 83% ^{235}U.

Category 9 - Uranium/Thorium Carbide Low Integrity Fuel: This category consists of fuels from core 1 of the PB reactor. Similar to category 8, the fuels are in the form of carbide particles coated with layers of pyrolytic carbon, bonded together by a carbonaceous matrix material. Two types of particles are used — fissile and fertile. Fissile particles contain thorium and ~93% enriched ^{235}U. The fertile particles contain only thorium. However, the fuel particles in these fuel assemblies are in poor condition. Some preliminary tests indicated that up to 60% of the particles may have been breached. Thus, the release rate of this may be 10 times the U-metal rate because of the possible water/carbide reaction.

Category 10 - Uranium and Uranium/Plutonium Carbide Nongraphite Fuel: This category consists primarily of fuels from the Fast Flux Test Reactor (FFTF). The FFTF fuels are mixed carbide fuel particles in a nongraphite matrix. It is uncertain as to the performance of the carbide particles without the presence of a graphite matrix and the silicon coating like the FSV fuels. Thus, this fuel was placed into its own category. This category may perform much worse than the pure U-metal fuel. Effective enrichments (including the ^{239}Pu) vary from about 10% to 18% ^{235}U.

Category 11 - Mixed Oxide Fuel: MOX fuels are composed of a mixture of uranium and plutonium oxides within various claddings. The uranium enrichment qualifies as "low" but the plutonium content increases the effective enrichment above 15% ^{235}U. The FFTF driver fuel assembly (DFA) and test fuel assembly (TFA) contributed to the large quantity of the fuel in this category. Since the fuels were fabricated using similar techniques as the commercial oxide fuels, performance of the MOX fuels should be very similar. Due to the high plutonium content as compared to the U-oxide fuel, this fuel was placed into its own category.

Category 12 - Thorium/Uranium Oxide Fuel: Shippingport Light Water Breeder Reactor (LWBR) fuels make up the major inventory of the fuel in category 12. The Shippingport LWBR was used to demonstrate the production of fissile ^{233}U from thorium in a water-cooled operating reactor. The fuel was made of uranium oxide, enriched up to 98% ^{233}U mixed with thorium oxide and made into cylindrically shaped ceramic pellets. These ceramic pellets are expected to perform differently than the standard U-oxide fuel and thus this fuel was placed into its own category.

Category 13 - Uranium Zirconium Hydride Fuel: Category 13 contains fuel with the uranium/zirconium hydride matrix. Fuels from the Training, Research, and Isotope General Atomic (TRIGA) reactors make up the majority of the fuel in this category. The uranium-zirconium hydride in this category provides the reactor with its built in control and inherent safety. The fuel consists of U-metal particles dispersed in zirconium hydride matrix, clad with aluminum, stainless steel, or Incoloy-800 with varying enrichment and weight percents of ^{235}U. Due to the unique uranium/zirconium hydride matrix, it was placed in its own category. This fuel matrix is expected to perform much better than the standard U-oxide fuel.

Category 14 - Sodium-Bonded Fuel: Due to the reactive nature of sodium, all the sodium-bonded fuel will be treated prior to disposal in the repository. Thus, this fuel was placed in its own category but the final waste form behavior will not be addressed here.

Category 15 - Classified Navy: Due to the classified nature of the Navy fuel, it was placed in its own category and all information concerning this category will be provided by the Navy and will not be addressed here.

Category 16 - Miscellaneous Fuel: The remainder of the DOE SNF that does not fit into the above categories is placed in this category. Due to the varying matrices, cladding, and condition of this group of fuel, the plan is to bound the fuel properties in the performance evaluation with the dissolution model that reasonably represents this category. Based on the category inventory, the U-metal dissolution model is believed to well represent the DOE SNF in this category.

Criticality Analyses Categories[a]—How the fuel degrades in the repository environment will affect its criticality risk. Thus, for criticality analyses, the 34 condensed fuel groups were further reduced into 13 categories based on the fuel matrices and enrichment. Although cladding could play an important role in extending the fuels' physical configuration, DOE EM has decided not to include cladding credit in the criticality analyses at this time. As indicated in Figure 2-1, criticality analyses of fuels with similar matrices (in terms of geologic time periods of over thousands of years) could be considered together. Thus, the actual number of criticality analyses may be further reduced to nine evaluations by combining the HEU and MEU fuels in the same category.

Like the TSPA categories, the criticality analyses will not include the sodium-bonded and classified Navy SNF. The sodium-bonded fuel will be treated prior to disposal and the Navy fuel criticality evaluation will be performed by the Navy.

[a] This brief discussion is to provide a general understanding as to how the criticality analyses grouping fits into the overall DOE SNF grouping methodology. Criticality analyses grouping will be further discussed as part of the individual criticality analyses that are in progress at this time.

3. SNF DISSOLUTION MODELS

With each category listed above, a dissolution model was used to represent the fuel's radionuclide release rate to the repository's unsaturated zone and eventual transport to the receptor. The rationale for selecting a dissolution model to represent the fuel category is discussed below. Two points in the grouping discussion that need to be revisited here. First, the radionuclide inventory and radionuclide release rate form the source term for use in the TSPA to determine the dose to the public from SNF disposal. Second, the radionuclide release rate is the product of the intrinsic release rate per unit surface area times the available surface area. Most DOE fuels are expected to have low specific surface area due to negligible swelling because of low burnup and negligible porosity due to manufacturing. Therefore, the surface area is just the geometric area adjusted with some roughness factor. We could make the assumption that the matrix dissolution will not be significantly different than the grain boundary dissolution. Based on the current understanding of the fuel properties, Fillmore suggested using the wet dissolution rate for the various DOE SNF categories [Reference 3]. The suggested wet dissolution models are presented in Table 3-1. The rationale for using each dissolution model is discussed below.

As indicated in the grouping discussion, the NSNF Program's release rate testing program, currently in progress, will confirm the dissolution model selected here. Each of these models will be revised as necessary to reflect the data collected in the release testing rate program.

3.1 Dissolution Model for Category 1 — Uranium Metal Fuels

The zirconium-clad N-Reactor fuels, with a small amount of aluminum-clad Single Pass Reactor fuel, makes up this category. The N-Reactor fuel elements consist of two concentric tubes made of uranium metal co-extruded into zircaloy-2 cladding. The density of the fuel matrix averages 18.96 gm/cc or 0.685 lb/in^3. The fuel matrix consists of a continuous metallic uranium structure [Reference 4]. The fuel's pre-irradiation ^{235}U enrichment is below 2%. Appendix A.1.1 presents a more detailed description of this category.

The uranium metal fuel's radionuclide release rate is expected to be very close to the uranium matrix dissolution or corrosion rate. Several authors have collated the available quantitative rate data for the reaction of unirradiated uranium in various environments. In *Uranium Metallurgy Volume II: Corrosion and Alloys*, Wilkinson presented the oxidation of uranium in a number of environments — in still air, humidity, steam, and for different temperatures with different gases, etc.[Reference 5]. Ritchie performed similar data reviews during the 1980s [References 6, 7]. In a more recent research report, *A Review of the Rates of Reaction of Unirradiated Uranium in Gaseous Atmosphere*, Pearce reviewed quantitative rate data for the reaction in dry and moist air, steam and carbon dioxide atmospheres, from room temperature to above the melting point of uranium [Reference 8]. A DOE report titled *An Independent Technical Assessment of the Dry Storage of N-Reactor Fuel* also shows a compilation of similar data for corrosion of uranium metal in water and water vapor [Reference 9].

Pearce, Ritchie, and Wilkinson generated reaction rate correlations (Arrhenius functions) for uranium reacting with dry oxygen and with water plus air. For material that follows the parabolic or cubic time dependence equation (rate of corrosion decreases as the thickness of the corrosion product

Table 3-1. DOE spent nuclear fuel wet dissolution models.

Fuel Category	Fuel Matrix	Typical Fuel in the Category	Wet Dissolution Model
1	U-metal	N-Reactor fuel	U-metal model
2	U-Zr	Heavy Water Components Test Reactor fuel	U-metal model
3	U-Mo	FERMI (Enrico Fermi Reactor) Fuel	10x U-metal model
4	U-oxide intact	Commercial PWR fuel Shippingport PWR fuel	Commercial model
5	U-oxide failed/declad	Three Mile Island (TMI) fuel	Commercial model
6	U-Al Or U-Alx	Advanced Test Reactor (ATR) fuel Foreign Research Reactor (FRR) fuel	0.1x U-metal model
7	U-Si	Foreign Research Reactor (FFR) fuel	0.1x U-metal model
8	U/Th carbide hi-integrity	Fort St. Vrain (FSV) fuel	Si carbide model
9	U/Th carbide low-integrity	Peach Bottom fuel	10x U-metal model
10	U or U/Pu carbide nongraphite	Fast Flux Test Facility (FFTF) carbide fuel	100x U-metal model
11	MOX	Fast Flux Test Facility (FFTF) oxide fuel	Commercial model
12	U/Th oxide	Shippingport LWBR fuel	Ceramic model
13	U-Zr-Hx	Training Research Isotopes- General Atomic (TRIGA) fuel	0.1x Commercial model
14	Na-bonded	FERMI I Blanket	Will be treated. Not part of TSPA-VA analyses
15	Classified-Navy	Navy	Model by Navy
16	Misc. SNF	Misc. fuel	U-metal model

increases), an equation of generalized corrosion was also proposed to represent the dissolution of the DOE SNF in an unsaturated Tuff repository by Rechard [Reference 10]. This generalized equation is Equation (1):

$$M = A \cdot e^{-B/T} \cdot (t_2^C - t_1^C) \cdot D \cdot E \cdot SA \qquad (1)$$

where:

M	=	mass of layer corroded in time step
A	=	Arrhenius-type pre-exponential term (kg/m^2s)
B	=	Arrhenius-type activation energy term (°K)
T	=	temperature of the material (°K)
t_2 and t_1	=	time at the beginning and end of the time step in seconds
C	=	time dependent term (reaction order, i.e., linear, parabolic)
D	=	saturation dependence term
E	=	oxygen concentration dependence term
SA	=	surface area of the layer

The uranium reaction rate portion of this equation (i.e., $A \cdot e^{-B/T}$) uses the data from the Wilkinson's book *Uranium Metallurgy Volume II: Corrosion and Alloys* for the Arrhenius fit. When the repository temperature is below 100 °C, wet oxic conditions are assumed and humid oxic conditions are assumed for all other times. Using this assumption and the Wilkinson data, the parameter values on the DOE-owned U-metal SNF are as follows:

For wet oxic conditions

A	=	9.4×10^3 kg/m^2s for wet oxic conditions,
B	=	7,970 °K for wet oxic conditions,
C	=	1 for wet oxic conditions (linear corrosion kinetics)
D	=	1 which is assumed to be conservative, and
E	=	0.2, the oxygen concentration term has been approximated by the mass fraction of air within the gas phase

For humid oxic conditions

A	=	1.35×10^2 kg/m^2s for humid oxic conditions,
B	=	7,240 °K for humid oxic conditions,

C = 1 for humid oxic conditions (linear corrosion kinetics)

D = 1 which is assumed to be conservative, and

E = 0.2, the oxygen concentration term has been approximated by the mass fraction of air within the gas phase

A plot of the U-metal reaction rate portion (i.e., $A \cdot e^{-B/T}$) of this generalized expression under the wet and humid oxic conditions is shown in Figure 3-1 [Rechard (Wet) and Rechard (Humid)]. Also plotted in this figure are the uranium rate equations proposed by Pearce as well as Ritchie to provide a reference. The Pearce expressions are considered to be the most extensive review of existing U-metal reaction data at this time. As such, they are presently viewed as the accepted rate equations although there are still uncertainties concerning its applicability to damaged fuels.

However, as indicated in Figure 3-1, the uranium reaction rate proposed by Rechard (based on the Wilkinson data) appears more conservative (i.e., faster) for all the conditions below ~100 °C. Similarly, the rate equation is more conservative for the wet oxic conditions up to ~200 °C. Since the DOE SNF will be dried prior to canisterization and possibility of DOE SNF encountering humid conditions above 100 °C will be unlikely (i.e., the disposal package will be intact for several thousand years and thus the fuel should be below 100 °C by the time the disposal package is breached), the uranium reaction rate proposed by Rechard was selected at this time for use in the TSPA-VA analysis.

When the U-metal release rate program confirms the reaction rate equation for the U-metal, the present U-metal rate equation will be updated for future repository license application purposes.

For the purpose of the TSPA-VA base case, a single DOE SNF fuel type represented the entire DOE SNF inventory. Based on the 1997 TSPA sensitivity analysis of DOE SNF [Reference 34], using the N-Reactor SNF to bound everything should be the most conservative. Thus, in the base case, the DOE SNF inventory was modeled as N-Reactor SNF using the U-metal dissolution model.

3.2 Dissolution Model for Category 2 — Uranium-Zirconium Fuels

The U-Zr fuels consist of uranium alloyed with zirconium. Yemel'yanvo and Yevstyukhin indicated that the addition of zirconium to uranium hardened it considerably and reduced its rate of creep. Both yield and ultimate tensile strength of the uranium-zirconium alloy peaks at a zirconium content of about 40–50 wt%. At this proportion, phase transformation is retarded so much that the γ-uranium becomes stable at room temperature. These alloys have increased corrosion resistance and greater creep resistance [Reference 11]. Similarly, a study conducted by Bauer evaluated the properties and behavior of U-Zr alloys confirmed most of these findings. In addition, Bauer included some limited corrosion data for the U-Zr alloy from various references [Reference 12].

Over 97% in metric tons of heavy metal (MTHM) of the fuel in this category consist of zirconium-cladded HWCTR driver assemblies. Enrichments are typically 93% ^{235}U. The uranium in the HWCTR driver assemblies is alloyed with 90.7 wt% zirconium [Reference 13]. A plot of the Bauer corrosion data on 90.7 wt% zirconium and 9.3 wt% uranium is shown in Figure 3-2. To provide a comparison with the U-metal corrosion rate, all the U-metal reaction data has been included for reference.

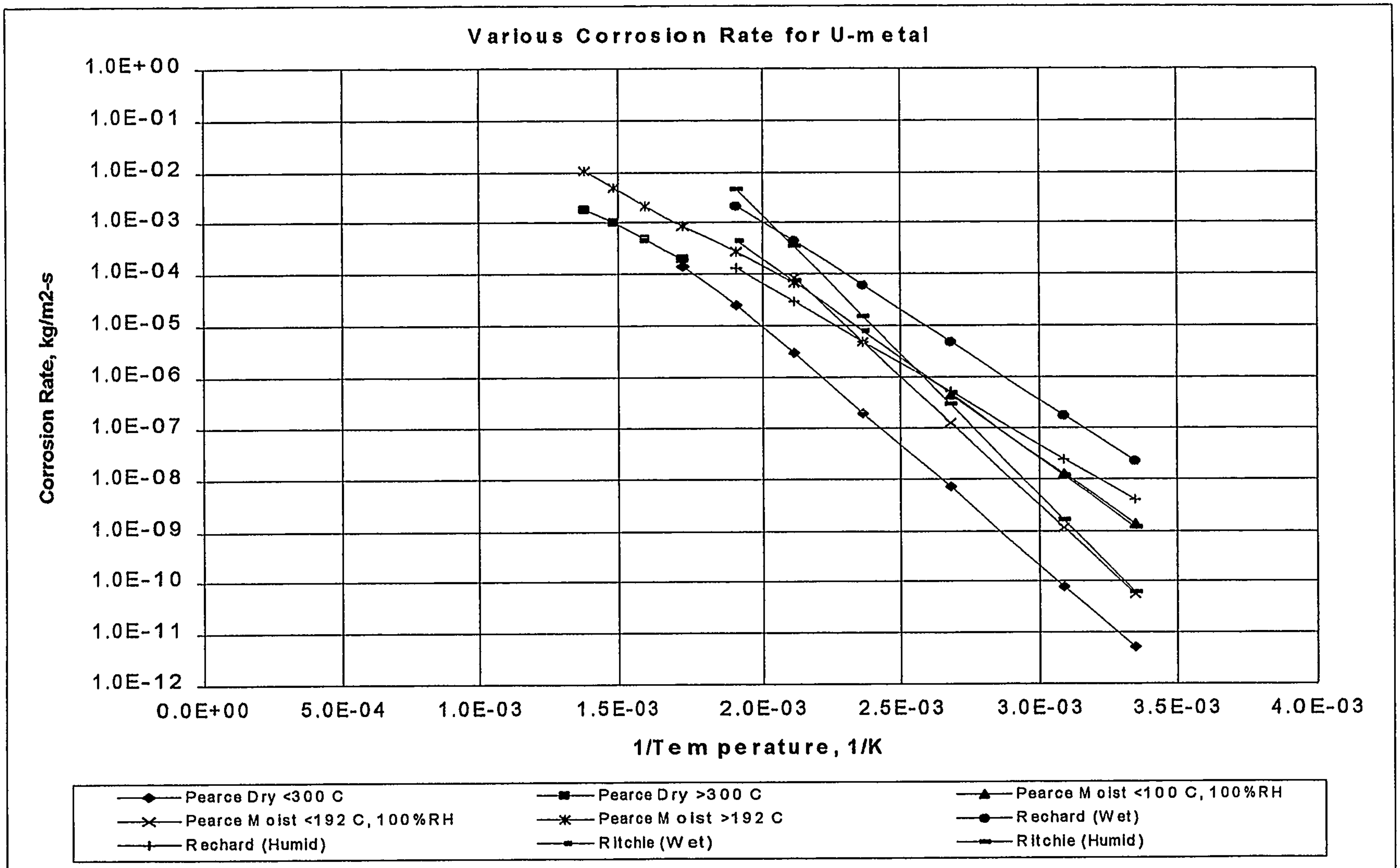

Figure 3-1. Various U-metal reaction rate equations plotted with temperature.

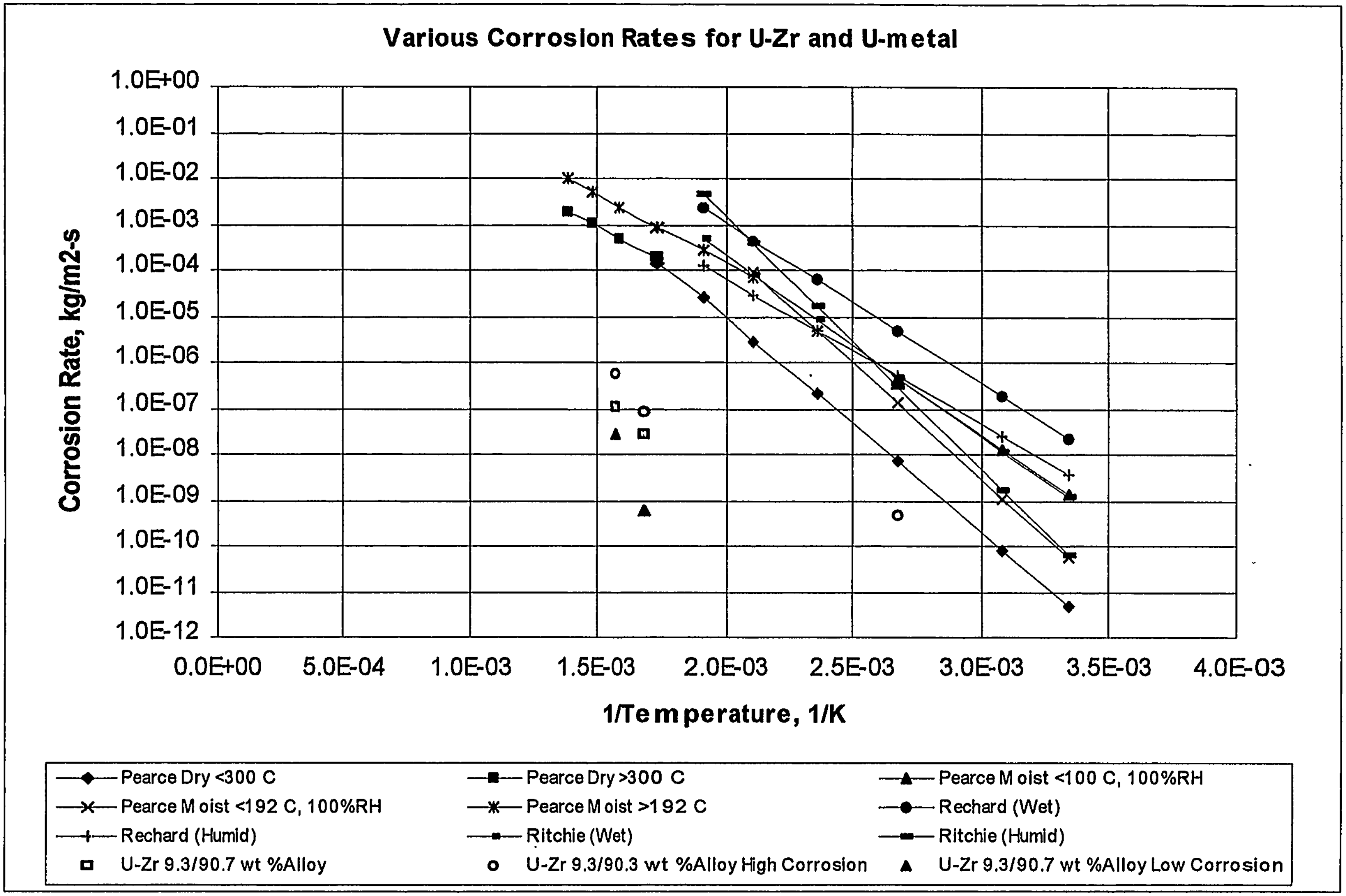

Figure 3-2. U-Zr and U-metal corrosion rate plotted with temperature.

Data have not been collected yet regarding the manufacturing process used to fabricate the U-Zr fuel assemblies or the effects of radiation on the phase transformation of this alloy. Thus, it is uncertain as to the corrosion properties of the U-Zr alloy because the unknown status of its phase at the time of disposal.

Based on the information collected to-date, the U-Zr fuel's radionuclide release rate is expected to be close to the U-Zr matrix dissolution or its corrosion rate. As indicated in Figure 3-2, the U-Zr alloy should perform better than the U-metal in the repository temperature ranges. However, with the potential that the U-Zr corrosion rate may be no better than the uranium metal itself depending on the phase of the alloy at the time of disposal, it was decided that the U-metal corrosion model will be used to bound the U-Zr alloy corrosion at this time in the TSPA-VA. As better U-Zr information becomes available, it will be included in the licensing application process.

3.3 Dissolution Model for Category 3 — Uranium Molybdenum Fuels

The uranium molybdenum (U-Mo) fuels consist of uranium alloyed with molybdenum. Yemel'yanvo and Yevstyukhin indicated that both tensile properties and creep improve with increase of the molybdenum content in the uranium. Several others have shown that the stable $\alpha + \gamma'$ phase for alloys containing 1–12 wt% molybdenum below 600 °C undergoes the observed reaction $\alpha + \gamma' \rightarrow \gamma$ when irradiated. This reaction reduces the quantity of sharply anisotropic α-phase with increase in the molybdenum content. Thus, it was concluded that alloying with molybdenum reduces the change in the shape of uranium samples under irradiation or thermal cycling over a wide temperature interval. An investigation also showed that the corrosion-resistance of heat-treated uranium-molybdenum increases sharply with increase in the molybdenum content. For γ-quenched alloys with 9–12 wt% molybdenum, the corrosion rate was quoted as 0.1 mg/cm^2-hr at 316 °C and 0.3 mg/cm^2-hr at 360 °C and 0.8 mg/ cm^2-hr at 400 °C [Reference 11].

A paper published by Waber in 1958 also covers the corrosion properties of various uranium alloys [Reference 14]. However, Waber reported that U-Mo alloys containing 6 wt% molybdenum (or less) show severe attack as compared to high-purity uranium after about a 10-month exposure to air containing 50% relative humidity at 75 °C. The alloys of lower wt% molybdenum appear to follow an accelerating rate law with a time dependence exponent of about 1.5. These samples also formed a powdery, layered corrosion product that expanded to more than three times the original height of the sample and thus tended to crack. Although Waber also reported that the 8 and 10 wt% molybdenum alloys show relatively good corrosion resistance, it is uncertain whether the apparent corrosion resistance of these specimens holds for exposures beyond ~10,000 hours.

All the fuel in this category consists of zirconium-clad Fermi core 1 and 2 driver, sectioned, sodium worth, or core foil fuels. The driver fuel makes up about 95% of the inventory based on MTHM. The Fermi driver fuel subassembly was designed with three active regions — a lower axial blanket, a fuel section, and an upper axial blanket. The lower and upper axial blanket subassemblies have been cropped off from the central core fuel section and are currently stored with the radial blanket subassemblies in ICPP-749 and will be treated prior to final disposal. Enrichments are typically about 25% ^{235}U. The uranium in the Fermi driver center fuel sections is alloyed with 10 wt% molybdenum [Reference 15].

Babcock and Wilcox Research Center, Battelle Memorial Institute, and Nuclear Metals Inc. developed the fabrication process. The procedure consisted of vacuum induction melting, casting, machining the uranium Mo alloy casting, encapsulating the fuel alloy slugs in a zircalloy sleeve by coextrusion of the fuel alloy slugs in Zr tubing 1,600 °F at which time a metalurgical bond was formed, and cold working with a rotary swager to the fuel pin's final dimension of 0.158 inches diameter. This procedure was used by the fuel fabricator, D.E. Makepeace, for production of two full core loadings. The fuel pin is made up of a solid uranium-molybdenum alloy fuel meat, 0.148 inches OD, metallurgically bonded to a Zr-4 tube. The fuel pins were originally fabricated in lengths of 12 feet or greater and were cut into 30.5 inches sections with the ends pointed by cold swaging. Following the sectioning, each pin was subjected to a heat treatment for stress relief. Next, prefabricated zirconium end caps were installed on either end of the pins and secured in place by cold swaging. [Reference 15].

Similar to the uranium or other uranium-alloyed fuel, the U-Mo fuel's radionuclide release rate is expected to be close to the U-Mo matrix dissolution or its corrosion rate. A plot of the corrosion rate mentioned by Yemel'yanvo and Yevstyukhin for U-Mo alloyed fuel is presented in Figure 3-3. Waber's corrosion data for 2, 4, 6 , 8, and 10 wt% molybdenum alloy in 50% RH was also included for 75 °C. The Pearce and Rechard U-metal corrosion rates were included for reference purposes.

As indicated on the figure, the heat-treated U-Mo alloys and the Waber 10% Mo alloy appear to perform better than the U-metal. However, since the potential exists that the U-Mo fuel may perform worse than the high-purity U-metal depending on the time period considered, it was decided that 10 times the U-metal corrosion model will be used to bound the U-Mo alloy corrosion at this time in the TSPA-VA. As better U-Mo fuel information becomes available, it will be included in the licensing application process.

3.4 Dissolution Model for Category 4 — Uranium Oxide Intact Fuels

This category consists of the fuels removed from commercial reactors or test fuel with uranium oxide matrices similar to RW's commercial SNF. Of the total inventory of ~98 MTHM, over 67 MTHM come from commercial reactors such as Ginna operated by the Rochester Gas and Electric and Surry 2 operated by Virginia Power. Thus, they should have the same aqueous dissolution and release rate responses as the commercial SNF being evaluated by RW. Fillmore indicated that a large number of the DOE test fuels have a ceramic matrix (e.g., the Shippingport PWR) and should have a much slower dissolution rate compared to the commercial oxide fuels [Reference 3]. Another potential difference is that some of the fuels in this category could be up to 93% enriched ^{235}U. Since enrichments should not alter the dissolution rate of fuels with the same matrix, the commercial dissolution model should be applicable to the highly enriched DOE test oxide fuels.

For the commercial SNF, RW's present approach is to obtain an experimental database of dissolution rates for a subset of specific spent fuels over a range of controlled, aggressive water chemistries and temperature. The database is a collection of measurements from flow-through tests on the dissolution of UO_2 and spent fuel (spanning a wide range of carbonate, oxygen, and pH values). These data are then used to evaluate empirical parameters in a rate law to describe the dissolution rate of the commercial SNF. Several dissolution models were presented in the *Waste Form Characteristics Report* version 1.2 [Reference 16] Section 3.4.2 (in the form of the Butler-Volmer equation). A final equation in the following form was selected for use in the TSPA-VA to conservative bound the commercial SNF with burnup >30,000 MW days/kgU:

$$\begin{aligned}\log_{10}(Rate) &= a_0 + [a_1 \cdot ((1/T) - c_1)] + [a_2 \cdot (\log_{10}(CO_3) - c_2)] + [a_3 \cdot (\log_{10}(O_2) - c_3)] \\ &+ [a_4 \cdot (pH - c_4)] + [a_5 \cdot ((\log_{10}(CO_3) - c_2)^2 - c_5)] + [a_6 \cdot ((1/T) - c_1) \cdot (\log_{10}(O_2) - c_3)]\end{aligned} \quad (2)$$

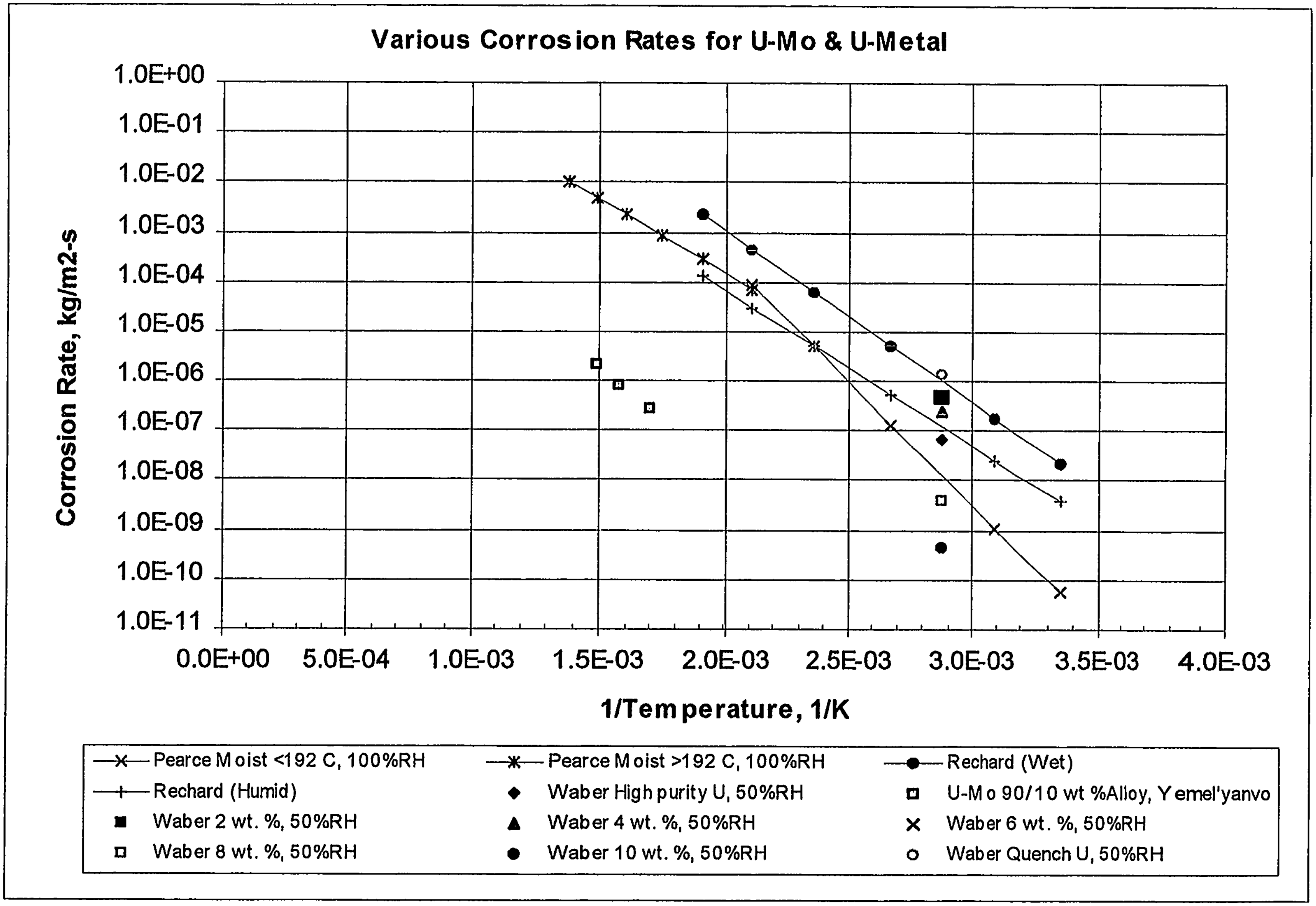

Figure 3-3. U-Mo and U-metal corrosion rate plotted with temperature.

Where

$a_0, \ldots, a_6$ = constants

$c_1, \ldots, c_5$ = mean value of variables under consideration

T = temperature (°K)

CO_3 = total carbonate concentration (mol/L)

pH = negative of the $\log_{10}$ of the hydrogen ion concentration in mol/L

O_2 = % oxygen concentration in the gas phase (atm)

Rate = mg/m^2-day

The constants and mean values used in the TSPA-VA are as follow:

a_0 = 0.5083

a_1 = -862.3339

a_2 = 0.0527

a_3 = 0.2915

a_4 = -0.1307

a_5 = -0.1381

a_6 = -781.7371

c_1 = 0.00311

c_2 = -2.51

c_3 = 0.071

c_4 = 8.89

c_5 = 0.74

Based on the above discussions, this category is conservatively modeled as performing like the commercial spent nuclear fuels and the commercial dissolution model was used to represent the dissolution rate of category 4. Since RW will be providing all the justification for the use of the model, no other discussion or work on the uranium oxide fuel is planned by DOE-EM to support the TSPA at this time.

3.5 Dissolution Model for Category 5 — Uranium Oxide Failed/Decladded Fuels

Like category 4, this category consists of the fuels removed from commercial reactors or test fuels with uranium oxide matrices that have been damaged, experienced failed cladding, or are decladded. Of the total inventory of ~87 MTHM, over 81 MTHM are from commercial reactors such as Three Mile Island Reactor fuels (TMI). Using similar arguments provided in category 4, this category is modeled as performing like the commercial spent nuclear fuels but potentially with a much higher fuel surface area due to the damage or the physical state (small pieces of disrupted fuel) of the fuel. This category contains enrichments from the typical 1–2% commercial ranges (such as TMI Reactor fuels) to the 93% fuel from the High Flux Isotope Reactor. Again, since enrichment level should not alter the dissolution rate for fuels with the same matrix, the commercial dissolution model was used to represent the category.

For TSPA-VA, this category is conservatively modeled as performing like the commercial spent nuclear fuels and the commercial dissolution model was used to represent the dissolution rate of category 5. However, 100 times the commercial SNF surface area was used to represent the fuels in this category. The rationale concerning the selection of the fuel surface areas will be discussed in a later section. As with category 4, no other discussion or work on the uranium oxide failed/decladded fuel is planned by DOE-EM to support the TSPA at this time.

3.6 Dissolution Model for Category 6 — Uranium Aluminum or Aluminide Fuels

This category consists of fuels with the uranium-aluminide dispersed in a continuous aluminum phase. Fuels from foreign research reactors make up a large part of the uranium-aluminide fuel in this category. Enrichment level varies from about 11% to 93%. The uranium-aluminide fuel may perform better than the pure U-metal fuel depending on the continuity of the primary aluminum phase, and the release rate from each of the phases. Aluminum corrosion studies have been conducted by various sites over the past number of years. The major concerns revolve around the long-term storage of aluminum-based fuels in wet and dry storage. SRS published a more recent report titled *Alternative Aluminum Spent Nuclear Fuel Treatment Technology Development Status Report* in April 1997 [Reference 17].

Section 3 of the SRS report describes the corrosion behavior of aluminum-10 wt% uranium (Al-10 wt% U) alloy in an autoclave at 200 °C under saturated vapor conditions and two aluminum cladding alloys under various conditions. The report indicated that for corrosion of the rolled samples of Al-10 wt% U, a large number of residual uranium aluminide particles remained, projecting from the metal matrix and scattered throughout the corrosion oxide layer. Based on this observation, the report concluded that the uranium aluminide may be more stable than aluminum and does not react, or reacts very slowly, in the 200 °C saturated vapor environment [Reference 17, Page 33]. Using this statement, the quasi linear portions of the reported weight gain for the Al cladding alloys and Al-10 wt% U alloy have been plotted in Figure 3-4 to represent the corrosion rates of the Al-10 wt% U and aluminum cladding alloys. Since the SRS reported that the 1100 and 6061 Al alloys should followed a parabolic corrosion behavior [Reference 17, Page 17], the quasi-linear portions of the weight gain should be a conservative representation of the Al materials corrosion process. From the figure, the Al cladding alloys and the Al matrix of the Al-10 wt% U appear to have corroded at a lower rate as compared to the U-metal of the category 1 SNF. Disregarding the rolled samples, the corrosion rates of Al-1100, Al-6061, and the Al-10 wt% U are over three orders of magnitude below the U-metal corrosion rate. For the purpose of the TSPA-VA analyses, 0.1 times the U-metal dissolution model was selected and used to bound category 6.

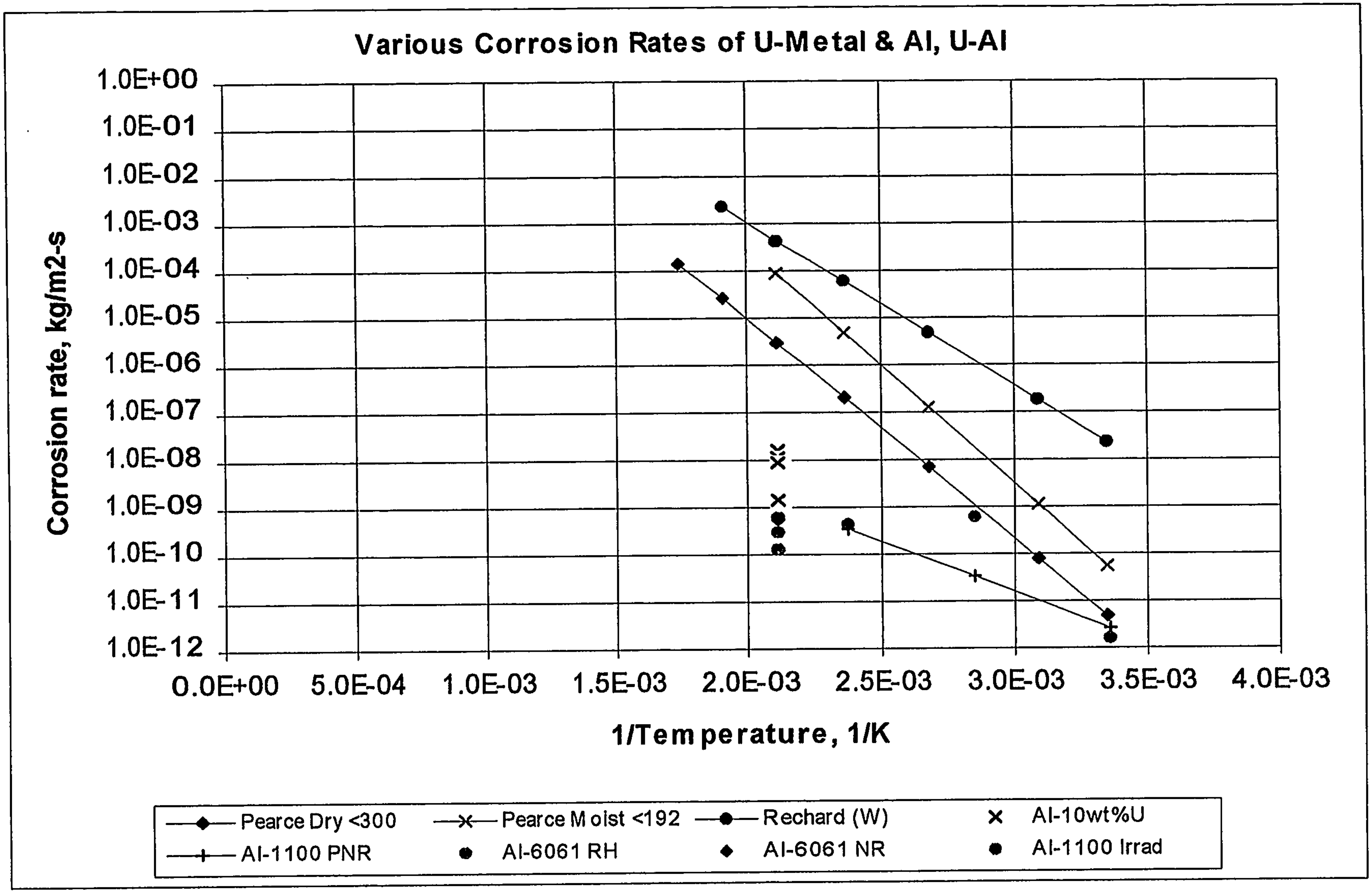

Figure 3-4. Al, U-Alx and U-metal corrosion rate plotted with temperature.

As indicated in the SRS report, the actual mechanisms causing the varying corrosion rates have not yet been determined and continuing Al fuel testing is currently in progress. These tests also include the more representative 18 and 33 wt% U making up the U-Alx fuel inventory. As better corrosion and dissolution information becomes available, it will be incorporated for use in the license application TSPA analyses.

3.7 Dissolution Model for Category 7 — Uranium Silicide Fuel

This category consists of fuels with the uranium-silicide dispersed in a continuous aluminum phase. Foreign research reactor fuels make up a large part of the U-Si fuels in this category. Enrichment level varies from ~8% to ~93%. But the majority of the fuels' enrichment is less than 20%. Based on the early findings for the uranium aluminide fuel, the U_3Si_2 may perform differently and thus is placed in its own category. Wilkinson [Reference 5] reported that the "Alloys in the range between U_3Si_2 and USi_2 are stable against atmospheric corrosion, and protective films are formed on these compounds in air when heated in the range 150 to 400 °C." Faraday [Reference 18] reported that U_3Si oxidation follows a three stage process where uranium reacted to form UO_2 and the U_3Si transforms to the other phases in the U_3Si -USi_x system (such as USi_2, U_3Si_2, or USi_3) depending on the temperature of the aqueous environment. The UO_2 reacts further to form U_3O_8. Faraday mentioned that "In general, U_3Si_2 particles did not appear to change in composition in advance of the corrosion front, since U_3Si_2 particles have been identified by the probe (microprobe) in the U_3Si matrix at the corrosion front." Snyder [Reference 38] reported silicide reaction results with oxygen that support a similar conclusion. Synder reported that, in order of increasing reaction rates, USi_3, U_3Si_2, USi_2 follow a parabolic rate law up to about 400 °C. Snyder also stated that "Therefore, in the initial reaction, these (USi_3, USi_2, U_3Si_2, and UAl_2) compounds are more oxidation resistant than uranium."

For comparison purposes, the oxidation rates of U_3Si in air reported by Faraday are plotted in Figure 3-5 with the U-metal, Al, and UAlx corrosion rates. As indicated, the oxidation rate of the U_3Si appears to be at least two orders of magnitude below the U-metal corrosion rate. Based on the discussion from the two reports above, U_3Si_2 appears to be less reactive than the U_3Si and thus should have an even lower corrosion rate than U_3Si. Thus, the 0.1 times the U-metal corrosion rate is used to bound the corrosion of U_3Si_2 for the purpose of TSPA-VA. Since the repository temperature will be much lower (~300 °C drift wall temperature), oxidation of U_3Si_2 may be relatively slow compared to the continuous aluminum phase.

Further testing will have to be done on the U_3Si_2 material. Flow through testing is currently in progress and drip testing will be added to the FY-1999 release rate program. As the results of the release rate program become available, the U_3Si_2 dissolution model will be revised and implemented into the repository license application.

3.8 Dissolution Model for Category 8 — Uranium/Thorium Carbide High Integrity Fuel

Fuel from the FSV reactor makes up ~95% (in terms of MTHM) of this category. The fuels from the core 2 of the PB reactor and a small amount of fuel from the General Atomic Gas-Cooled Reactor make up the rest of this category. The fuel is in the form of carbide particles coated with layers of pyrolytic carbon and SiC [Note: SiC coating is for the FSV only], bonded together by a carbonaceous matrix material. Two types of particles are used — fissile and fertile. The fissile particles contain thorium and ~93% enriched uranium. The fertile particles contain only thorium. One difference between

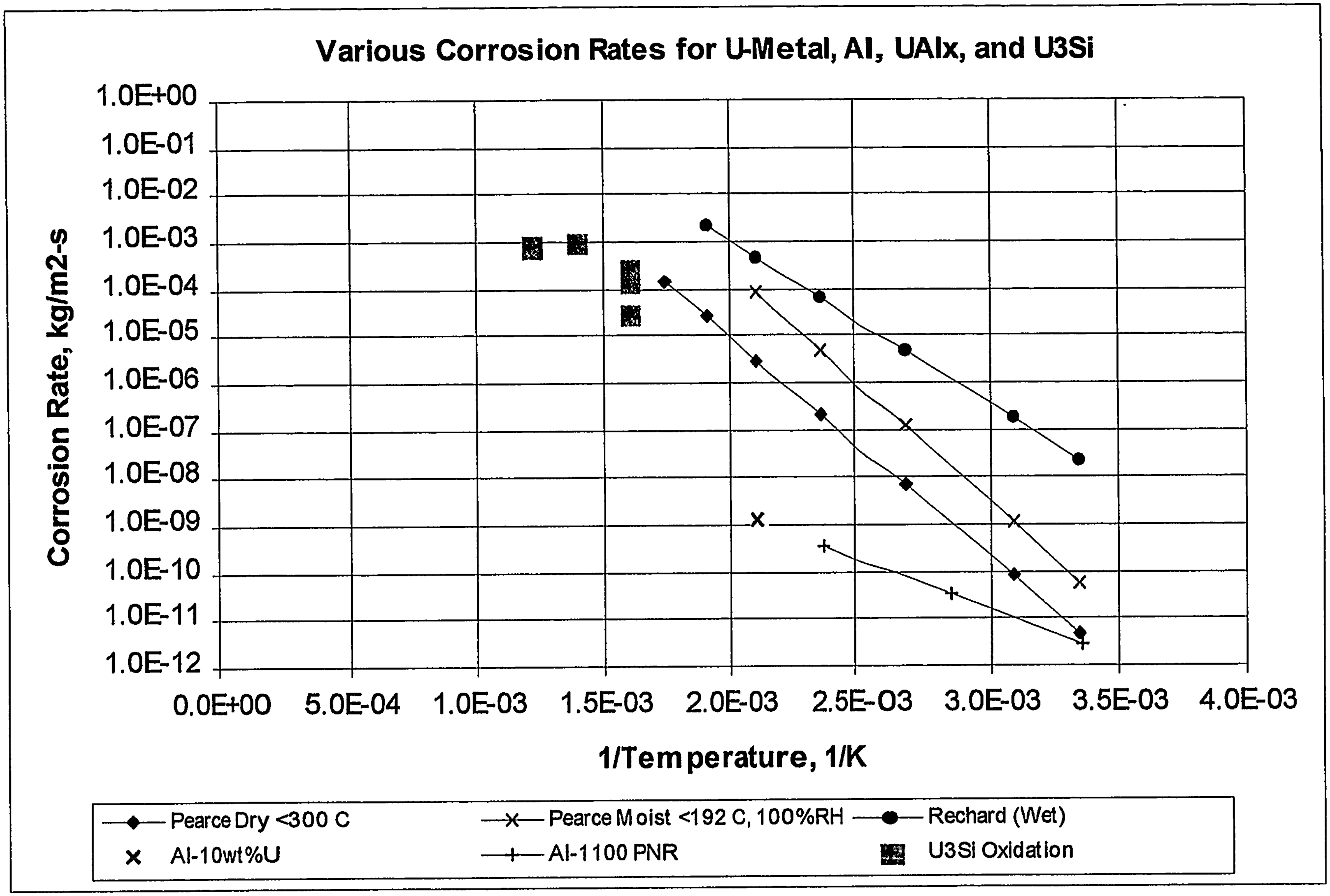

Figure 3-5. U_3Si Al, U-Alx and U-metal corrosion rate plotted with temperature .

the FSV and PB fuels is that the PB particles lack the silicon carbide coating. The fuel particles in all of the fuel assemblies are in excellent condition.

The pyrolytic carbon and silicon carbide layers on the FSV fuel assembly should provide a very slow release rate while the PB and General Atomic Gas-Cooled Reactor may be somewhat more reactive based on preliminary work done at the Battelle Memorial Institute in the late 1950s. Tripler reported that sintered compacts of UC and UC_2 disintegrated in boiling water (1 atm pressure) within an hour [Reference 19]. The disintegration was accompanied by rapid oxidation of the carbides.

In the same report, Tripler observed that UC_2 reacts with nitrogen and oxygen following a parabolic rate law in the range of 400 to 700 °C and 150 to 250 °C respectively. Tripler also reported that the reaction with water vapor follows the linear rate law from 50 to 200 °C. The reaction products consist of UN_x, UC, and UO_2. Yemel'yanvo and Yevstyukhin reported the decomposition of uranium carbides to U_3O_8 and CO_2 with damp air at 400 °C [Reference 11].

It is uncertain at this time what effects the layers of pyrolytic carbon and silicon carbide coating have on the carbide reaction studies by Tripler. Lotts compared the relative stability of High Temperature Gas Cooled reactor (HTGR) graphite to the light water reactor (LWR) fuels in ORNL/TM-12027 [Reference 20]. He reported that the graphite oxidation rate is extremely slow and estimated that it will take 3.6 x 10^9 years to oxidize 0.5 cm of graphite and will take only 5 x 10^5 years to uniformly oxidize a 25 mm thick LWR cladding. Based on Lotts information, an equation of generalized corrosion for the carbide fuel was also proposed to represent the dissolution of the silicon carbide by Rechard [Reference 10] for both wet oxic and humid oxic conditions. The proposed equation is:

$$M = A \cdot e^{-B/T} \cdot (t_2^C - t_1^C) \cdot D \cdot E \cdot M_{layer} \tag{3}$$

where:

M	=	mass of layer corroded in time step
A	=	Arrhenius-type pre-exponential term (1/s)
B	=	Arrhenius-type activation energy term (°K)
T	=	temperature of the material (°K)
t_2 and t_1	=	time at the beginning and end of the time step in seconds
C	=	time dependent term (reaction order, i.e., linear, parabolic)
D	=	saturation dependence term
E	=	oxygen concentration dependence term
M_{layer}	=	mass of the layer at time zero

Using the Lotts data, Rechard uses the following parameter values for the silicon carbide coating on the high integrity graphite SNF:

For both wet and humid oxic conditions:

A	=	3×10^{-12} /s,
B	=	0, (no temperature dependance at repository conditions)
C	=	1, (linear corrosion kinetics)
D	=	1, which is assumed to be conservative, and
E	=	0.2, the oxygen concentration term has been approximated by the mass fraction of air within the gas phase

For the purpose of the TSPA-VA, the corrosion rate of the silicon carbide was used to represent the SiC coating dissolution. The UC was assumed to react instantaneously when SiC coatings are breached. Figure 3-6 is a plot of the silicon carbide and U-metal corrosion rates. The NSNF Program's release rate testing program will be evaluating the carbide fuel's reactivity in the repository environment with respect to graphite reaction, not SiC reaction. As the results of the testing program become available, the carbide corrosion model will be updated as required.

3.9 Dissolution Model for Category 9 — Uranium/Thorium Carbide Low Integrity Fuel

This category consists of fuels from the core 1 of the PB reactor. Similar to category 8, the fuels are in the form of carbide particles coated with pyrolytic carbon, bonded together by a carbonaceous matrix material. Two types of particles are used — fissile and fertile particles. Fissile particles contain thorium and ~93% enriched uranium. The fertile particles contain only thorium. However, the fuel particles in these fuel assemblies are in poor condition. Fillmore indicated that up to 60% of the particles may have been breached [Reference 3].

No data are available on the oxidation rate of this fuel. Since the reaction rate of the UC_2 with water is expected to be rapid based on Tripler's observation, but moderated by the influx of water through the carbon matrix, Fillmore suggested that the dissolution rate should be treated as 10 times the value of the uranium metal dissolution rate.

3.10 Dissolution Model for Category 10 — Uranium and Uranium/Plutonium Carbide Nongraphite Fuel

This category consists primarily of fuels from the FFTF. Over 70% by MTHM are FFTF fuels. The Sodium Reactor Experiment (SRE) fuel makes up the rest of the category. The FFTF fuels are mixed carbide (Pu/U) fuel particles in a nongraphite matrix.

The SRE fuel elements are uranium carbide fuel in a nongraphite matrix. The fuel category has an effective enrichment (including the ^{239}Pu) from about 10% to 18%. It is uncertain as to the performance of the carbide particles without the presence of a graphite matrix and the silicon carbide coating like the FSV fuels. No data are available at this time other than the test conducted by Tripler. This category, as indicated by Tripler, may perform much worse than the pure U-metal fuel. For the purpose of TSPA-VA, Fillmore suggested that 100 times the uranium metal reaction rate be used to represent this category [Reference 3].

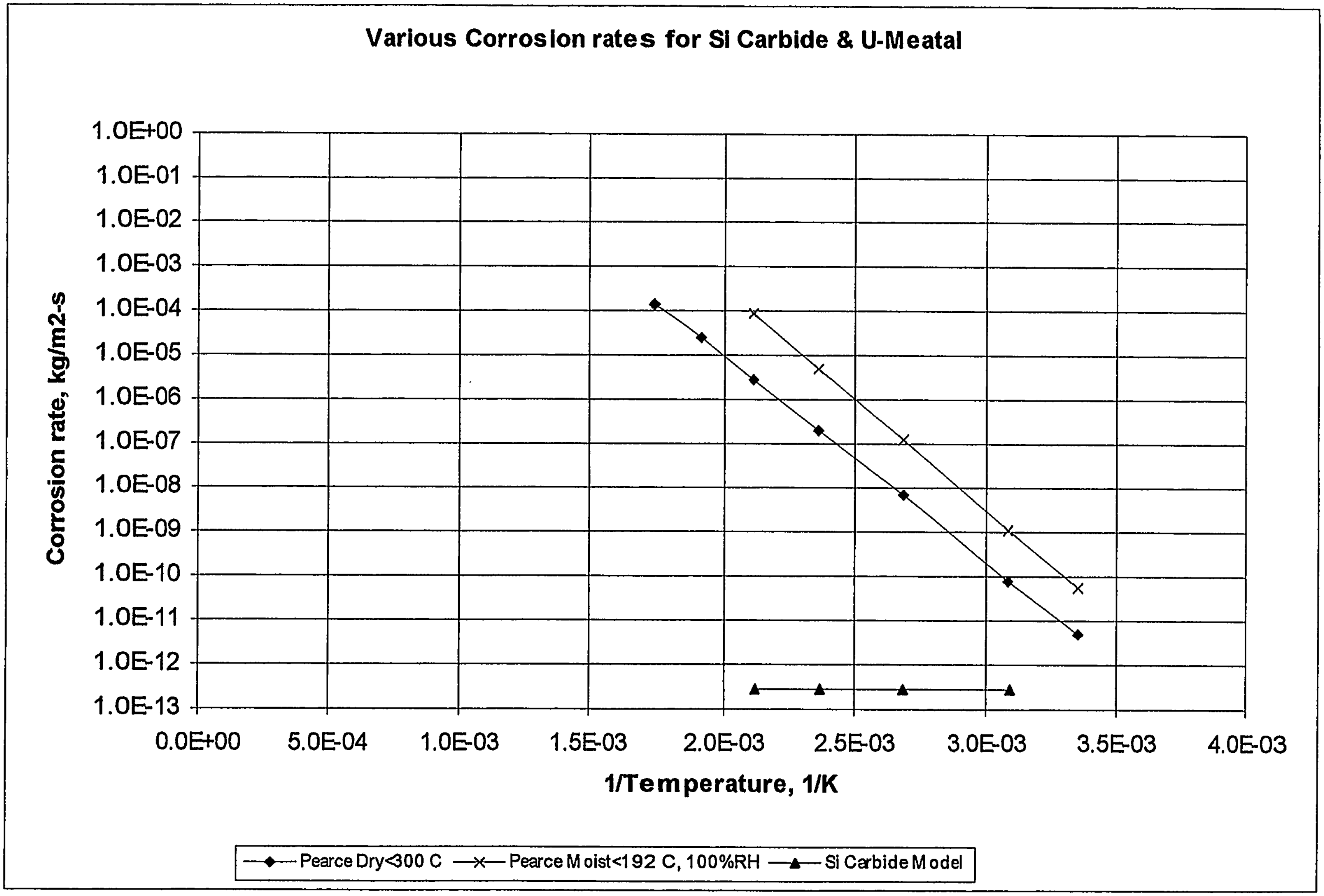

Figure 3-6. SiC, and U-metal corrosion rate plotted with temperature.

As better understanding of the carbide reaction becomes available, this reaction model will be revised accordingly. However, no testing of this fuel category is planned at this time.

3.11 Dissolution Model for Category 11 — Mixed Oxide Fuel

MOX fuels are composed of a mixture of uranium and plutonium oxides within various claddings. The uranium enrichment qualifies as "low" but the plutonium content increases the effective enrichment to above 15%. The FFTF DFA and TFA contributed to a large quantity of the fuel in this category. These driver fuel assemblies make up over 83% of the category in terms of MTHM. Cleveland in the *Plutonium Handbook* reported that PuO_2 prepared at high temperature dissolved very slowly even in HNO_3-HF acid solutions [Reference 21]. Sasahara reported a low fission gas release rate from MOX fuel was 1.7% as compared to the UO2 fuel at 4.8% [Reference 22]. Based on the readily available information, the MOX fuel appears to perform similarly or better than uranium oxide. However, without any definitive information on MOX fuel at this time, the uranium oxide model for the commercial SNF was selected to represent the MOX fuel in the TSPA-VA analysis.

The MOX fuel testing is currently part of the NSNF Program's release rate testing program. As these test data become available, the MOX fuel model will be revised to reflect the MOX reaction in the repository environment.

3.12 Dissolution Model for Category 12 — Thorium/Uranium Oxide Fuel

Shippingport LWBR fuels make up the major inventory of the fuel in category 12; specifically, it makes up over 86% of the category's inventory by MTHM. The remainder of the fuels in the category are from the Dresden Reactor. The Shippingport LWBR was used to demonstrate the production of fissile ^{233}U from thorium in a water-cooled operating reactor. The fuel was made of uranium oxide enriched up to 98% in ^{233}U mixed with thorium oxide made into cylindrically shaped ceramic pellets. The fuels contain between 1.19–3.67 wt% ^{233}U at the beginning of life (BOL). These ceramic pellets are expected to perform better than the standard U-oxide fuel and thus was placed into its own category. The BAPL conducted in-pile and out-of-pile corrosion behavior as part of the LWBR development program and published the results in WAPD-TM-1548 [Reference 23]. The study evaluated corrosion behavior of thoria (ThO_2) and thoria-urania (ThO_2-UO_2) materials, in the range of 2–30 wt% UO_2. Clayton (WAPD-TM-1548) reported that the LWBR type fuel has excellent corrosion resistance. The thoria's stability is also support by Brookins in his Eh-pH diagrams [Reference 32].

A ceramic model was suggested and used to represent the Th/U Oxide fuel in Total System Performance Assessment Sensitivity Studies of U.S. department of Energy Spent Nuclear Fuel [Reference 33]. The proposed ceramic model is indicated as Equation 5 below. The results from the BAPL report are plotted in Figure 3-7. All the information indicates a very low alteration for thoria-urania compound. As compared to the Pearce U-metal fuel corrosion data, the thoria-urania corrosion is over five orders of magnitude below it.

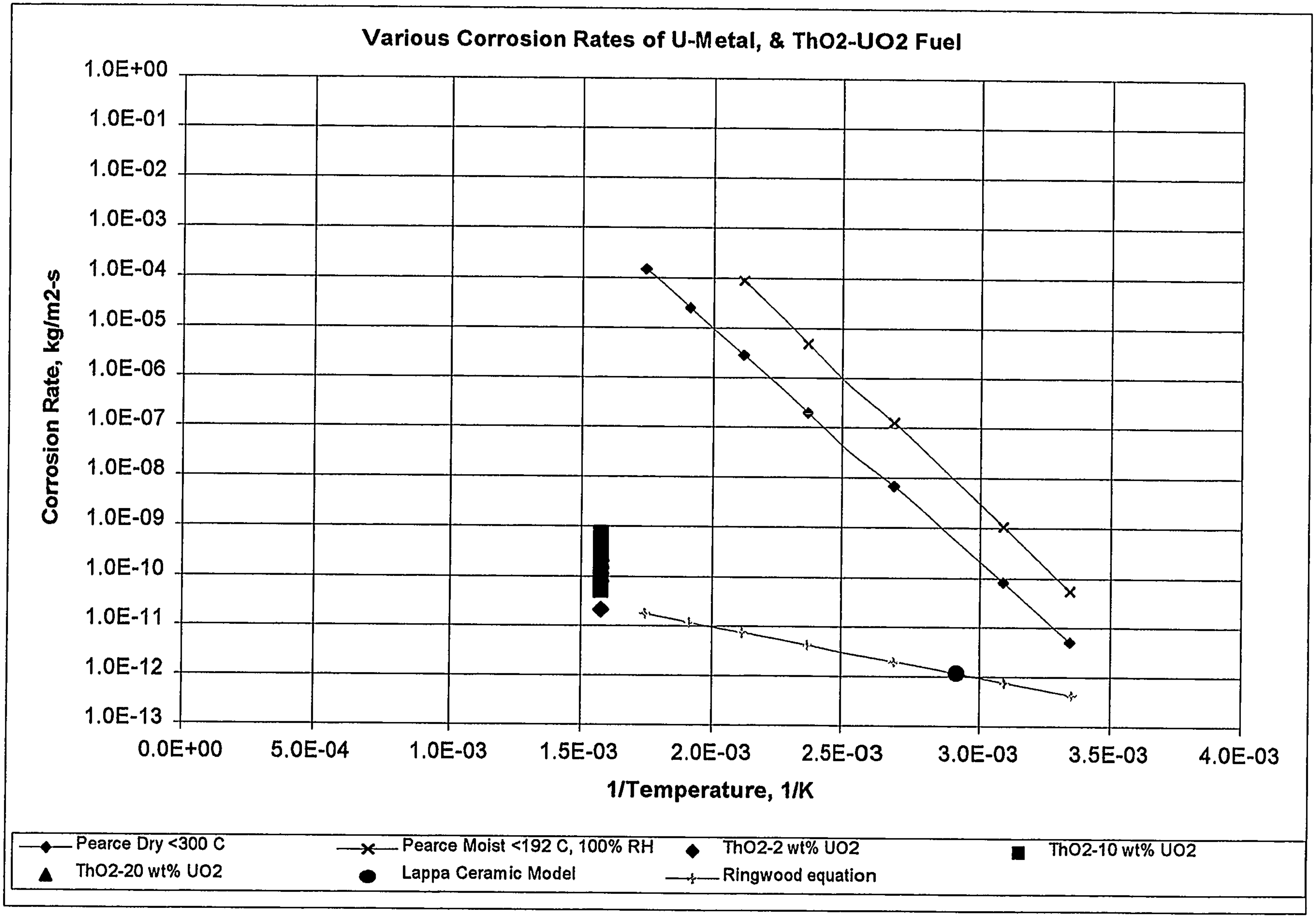

Figure 3-7. Th/U Oxide, and U-metal Corrosion Rate Plotted with Temperature

In a letter report, Lappa suggested that Equation 4 be used to represent ceramic materials [Reference 34].

$$Q = Q_0 + \theta + SFt / A \tag{4}$$

Where:

Q = the release per unit surface area (g/m^2).

Q_0 = the instantaneous release from grain boundaries and metastable phases (g/m^2).

θ = a complex kinetic function that accounts for ionic diffusion, selective matrix attack, etc. (g/m^2).

S = the solubility of matrix (g/m^3).

F = the ground-water flow rate (m^3/day).

A = the surface area of the matrix (m^2).

t = time (days).

Lappa stated that the long-term release from ceramic material such as Synroc is likely controlled by the third term in Equation 4. He indicated that using deionized water at 70 °C, the existing data support a matrix solubility of <0.007 g/m^3 based on a long-term leaching rate of less than 10^{-4} g/m^2-day (1.16×10^{-12} kg/m^2-s). As shown in the equation, the ceramic model is insensitive to the temperature range of the repository. Thus, if the reaction rate follows the same order as the U-metal matrix, the Lappa ceramic model appears to conservatively bound the data from the BAPL report.

The Lappa report also referenced a leaching rate equation proposed by Ringwood in which the leaching rate increases with increasing temperature [Reference 39]. The Ringwood equation is indicated below.

$$R = \alpha\, 10^{-\beta(1000/T)} \tag{5}$$

Where:

R = leaching rate (g/m^2-day)

T = temperature (°K)

α, β = constants (α = 0.082, β = 1.0 based on available data)

The leaching rate for the Ringwood equation is also plotted in Figure 3-7 for comparison purposes. Lappa indicated that "The effects of other factors such as leachant pH, water flow rate, and waste loading are either insignificant for the repository environment or not well understood at this time." For the purpose of TSPA-VA, the Ringwood equation was selected to model the dissolution rate of category 12. As better information becomes available, this model will be updated accordingly.

3.13 Dissolution Model for Category 13 — Uranium Zirconium Hydride Fuel

Category 13 contains the fuel with the uranium/zirconium hydride matrix. Fuels from the TRIGA reactors make up over 97% of the fuel in this category in terms of MTHM. The remainder of the fuels are from various research reactors such as the Atomic International Reactor. The uranium-zirconium hydride in this category provides the reactor with its built-in control and inherent safety. The fuel consists of a dispersion of U-metal particles in a ZrHx matrix. The fuels have various enrichments and loadings, and are clad with aluminum, stainless steel, or Incoloy-800. Due to the unique uranium/zirconium hydride matrix, it was placed in its own category. This fuel matrix is expected to perform much better than the standard U-oxide fuel. Thus, Fillmore suggested using 0.1 times the U-oxide model to represent this category.

3.14 Dissolution Model for Category 14 — Sodium Bonded Fuel

Due to its reactive nature of the metallic sodium, all the sodium-bonded fuel will be treated to remove the reactive characteristic prior to being disposed in the repository. Thus, a dissolution model of this fuel category will not be needed at this time. However, the final waste form dissolution and radionuclide release information will be required as part of the DOE HLW program.

3.15 Dissolution Model for Category 15 — Classified Navy Fuel

Due to the classified nature of the Navy fuel, it was placed in its own category and all the dissolution information concerning this category will be provided by the Navy and will not be addressed here.

3.16 Dissolution Model for Category 16 — Miscellaneous Fuel

The remainder of the DOE SNF that does not fit into the above categories is placed in this category. Due to the varying matrices, cladding, and condition of this group of fuel, Fillmore suggested that this fuel category be bounded by the fuel properties of the U-metal DOE SNF [Reference 3]. Note that this category makes up less than 0.5 % of the total DOE SNF inventory based on MTHM.

4. OTHER DOE SNF PROPERTIES

A number of other parameters are needed in the TSPA-VA models to predict the performance of the materials placed into the potential repository. The basis and/or derivation of each of these parameters is indicated in the following sections. Some properties were not readily available at the time of the TSPA-VA data request. Thus, expert judgment and opinion helped determine the best value. Each site will be collecting better data as the DOE SNF program moves closer to the repository license application. All of the information in support of the PA will be qualified according to the RW-0333P requirements by each of the sites.

4.1 DOE SNF Surface Area

The surface area for each DOE SNF category is derived in an engineering calculation number TSPA-VA-SURF-001 titled "Fuel Surface Area Calculation" and is summarized in the report by Fillmore [Reference 3]. This calculation is included as Appendix B. As indicated in the calculation, a roughening factor is added to the calculated surface area to account for the unevenness of the fuel surfaces. This parameter was based on the area and weight of the fuel meat. The calculations were simplified by the fact that the chemical form of the fuel meat within each category was assumed to be the same. Where different geometries or dimensions exist in the same category, a dominant type was selected or average values were calculated for the entire category.

4.2 DOE SNF Volume

The volume for each DOE SNF category is derived in an engineering calculation number TSPA-VA-VOL-001 titled "Fuel Meat Volume Calculation" and is summarized in the report by Fillmore [Reference 3]. This calculation is included as Appendix C. The volume of the fuel meat was based on MTHM/package and molecular weight and density of the fuel matrix. This volume does not include the void spaces between fuel plates (and rods) or fuel cladding.

4.3 DOE SNF Air Alteration Rate

Air alteration rate refers to the air oxidation rate of the DOE SNF under the repository conditions. For commercial SNF, this property was set to zero. Any value entered here is added to the wet dissolution rate at the time of the outer container failure. For DOE SNF, Fillmore indicated that the majority of air alteration rates for DOE SNF are unknown [Reference 3, page 2]. However, based on the experience at the wet and dry fuel storage facilities at various sites, it is generally agreed that the air alteration rate for the DOE SNF is insignificant as compared to its wet dissolution rate.

For the carbide fuels, uranium or thorium carbides reacting with air would produce uranium or thorium oxide that will dissolve much slower in the repository than the uranium or thorium carbide. Thus, neglecting the oxidation of the carbide is a conservative assumption [Reference 3, page 16 & 18]. Thus, Fillmore suggested for the purpose of the TSPA-VA, the air oxidation rate for DOE SNF should also be set to zero.

4.4 DOE SNF Cladding Failure

If the SNF cladding is in perfect condition, it will protect the fuel matrix materials from the repository environment after the container has been breached. Thus, no releases of radionuclide will occur nor will there be water available to alter the fuel matrix. The cladding is another layer of protection that must be degraded before the SNF matrix will see the repository environment. The majority of the commercial claddings are in good condition and thus RW is taking credit for it.

For DOE SNF, the cladding conditions for a number of fuels are not very well characterized at this time. In support of TSPA-VA, Fillmore suggested that conservative estimates be made of the fraction of fuel cladding failed for the DOE SNF. Table 4-1 shows the conservative estimate of cladding failures for each DOE SNF category [Reference 3]. The cladding failure fraction is an initial condition. Normal degradation processes are in effect from time zero or from canister breach, as appropriate. If the cladding is in perfect condition, the fraction of cladding failed is zero. If all the cladding has failed in a category, the fraction of cladding failed is one.

As shown in Table 4-1, some DOE SNF cladding is in excellent condition. However, no credit is currently claimed for fuel cladding as a barrier to releases for the DOE SNF except for the silicon carbide coating on the U/Th carbide high integrity fuel at this time.

Table 4-1. DOE SNF fraction of cladding failed.

Fuel Category	Fuel Type	Fraction of Cladding failed, 0-1
1	U-Metal	1
2	U-Zr	0.1
3	U-Mo	0.1
4	U-Oxide Intact	1
5	U-Oxide Failed/Declad	1
6	U-Al or U-Alx	1
7	U-Si	1
8	U/Th Carbide Hi-Integrity	0.01
9	U/Th Carbide Low-Integrity	0.6
10	U/Th Carbide nongraphite	0.1
11	MOX	0.1
12	U/Th Oxide	0.1
13	U-Zr-Hx	0.1
14	Na-Bonded	NA. Will be treated. Not part of TSPA-VA analyses
15	Navy	by Navy
16	Misc.	1

4.5 DOE SNF Free Radionuclide Inventory

This parameter describes the fraction of radionuclide inventory released from the fuel but still contained in the disposal package at the time the package is breached. Since the DOE-owned SNF will be sealed in canisters, the canister will also have to be breached prior to the free radionuclide inventory is available for immediate release. Because the DOE SNF, in most cases, has been stored for a long time (and in certain cases, the fuels have been breached) prior to repository package emplacement, most of the gaseous inventory available for immediate release would be gone prior to package and canister breach. The non-gaseous free radionuclide inventory fraction will depend on the fuel construction methods, the characteristics of the fuel matrix, the fuel storage condition, and the treatment of the fuel (such as drying and conditioning) prior to packaging for repository disposal. The heating (from drying and conditioning) may release some of the non-gaseous fission products from the matrix to the surface of the fuel and thus available for immediate transport. However, the free radionuclide fraction due to heating is going to be small compared to the total radionuclide inventory.

In addition, the conditions within the sealed repository disposal container are benign, and not likely to facilitate degradation of the fuel. For these reasons, the free fraction of the inventory in the DOE SNF will remain low. Fillmore evaluated various fuels in the DOE SNF inventory and suggested that they be set to values indicated in Table 4.2. See *DOE SNF Information Report in support of the TSPA-VA in the National SNF Program TSPA-VA* for more discussion [Reference 3]. If no radionuclide is available for immediate release, the fraction of free radionuclide is zero. If all of the radionuclide is available for immediate release, the fraction of free radionuclide is one.

Table 4-2. DOE SNF free radionuclide inventory.

Fuel Category	Fuel Type	Free Radionclide Inventory, 0-1
1	U-metal	0.001
2	U-Zr	0.00001
3	U-Mo	0.00001
4	U-oxide Intact	0
5	U-oxide failed/declad	0
6	U-Al or U-Alx	0.0001
7	U-Si	0.0001
8	U/Th carbide hi-integrity	0.00001
9	U/Th carbide low-integrity	0.1
10	U/Th carbide nongraphite	0
11	MOX	0
12	U/Th oxide	0
13	U-Zr-Hx	0.00001
14	Na-bonded	NA
15	Navy	by Navy
16	Misc.	0.001

4.6 DOE SNF Gap Inventory

The gap referred to here is between the fuel meat and the cladding. The inventory fraction is the faction of the fission product that has migrated from the fuel meat to the gap and is available for immediate release when the cladding is penetrated. This inventory may be specified separately for different isotopes. Some fuels are physically constructed so as to eliminate a gap region that could accumulate radionuclides. For instance, the N-Reactor fuel meat is co-extruded with the cladding. Fillmore evaluated DOE SNF construction and storage history and concluded that the majority of the DOE SNF will have zero gap inventory [Reference 3]. Fillmore's proposed gap inventory fraction is indicated in Table 4-3. Similar to the release fraction, if no radionuclide is available at the gap, the fraction of gap inventory is zero. If all of the radionuclide is in the gap, the fraction of gap inventory is one.

Table 4-3. DOE SNF fraction of gap inventory.

Fuel Category	Fuel Type	Fraction of Gap Inventory, 0-1
1	U-metal	0
2	U-Zr	0
3	U-Mo	0
4	U-oxide Intact	0.01–0.02
5	U-oxide failed/declad	0.0001
6	U-Al or U-Alx	0
7	U-Si	0
8	U/Th carbide hi-integrity	0
9	U/Th carbide low-integrity	0.001
10	U/Th carbide nongraphite	0.01–0.02
11	MOX	0.01–0.02
12	U/Th oxide	0.01–0.02
13	U-Zr-Hx	0.00001
14	Na-bonded	NA
15	Navy	by Navy
16	Misc.	0

5. SNF PACKAGES

5.1 DOE SNF Acceptance Basis

Allocation of repository space to DOE SNF and HLW glass has been identified as 10% of the 70,000 MTHM total allocated to high-level nuclear waste disposal in the repository under the Nuclear Waste Policy Act (1982) and its Amendment (1984). Within the 7,000 MTHM allocation, 1/3 of that inventory (or 2,333 MTHM) was to be dedicated to DOE-owned SNF. The balance of the allocation (4,667 MTHM equivalent) will be reserved for defense HLW placement within the repository [Reference 1].

The existing DOE SNF inventories include approximately 2,500 MTHM of fuels considered suitable for repository disposal. A small quantity of DOE SNF has been excluded from the ~2,500 MTHM inventory because it: (1) will be processed due to immediate vulnerabilities, or (2) will be treated due to fuel characteristics. In addition, for a number of the fuels (such as the Fort St. Vrain and several others) in the DOE EM inventory, portions of the fees for the repository have been paid. Thus, they will be deducted from the ~2,500 MTHM inventory making total direct disposal of all DOE SNF a possibility [Reference 2]. Finally, DOE RW has several other contracts similar to 10 CFR 961 with General Atomic and General Electric to take certain special fuels that are presently included in the DOE SNF inventory.

The current plan is to co-dispose the DOE SNF with the HLW in a large disposal package. The following sections describe the how the DOE fuels are packaged for disposal.

5.2 DOE SNF Disposal Configurations

5.2.1 SNF Canisters

The DOE SNF will be placed into individual fuel packages resulting in a combination of SNF canisters with approximate diameters of 18 inches, and 24 inches (~450 mm, and 610 mm) in both 118.1 inches and 179.9 inches (3,000 mm and 4,570 mm) lengths. This variety of fuel canister sizes, when placed with the HLW canisters, results in a variety of repository waste package combinations within each fuel category. Generally, fuel types (as determined by the originating reactor) within a fuel category will not be mixed in common SNF canisters. This approach may create a slight increase in the SNF canister count, and hence a corresponding increase in the HLW canisters needed to meet co-disposal requirements. However, such an approach does not affect the total MTHM.

Exceptions to the above rules include N-Reactor fuel and the intact commercial or commercial-like SNF from commercial reactors or test reactors such as the Big Rock Point and the Shippingport PWR blanket. The N-Reactor fuels will be placed into ~25 inches diameter (642.7 mm) multi-canister overpack (MCO) by 15 feet long canisters. The intact commercial-like DOE SNF will be shipped bare and thus will be placed into large disposal packages like the SNF from the commercial reactors at the repository.

DOE EM, in co-operation with RW M&O TESS, has been evaluating the fissile load limits for the DOE SNF (except the Navy fuel) in the past year and will continue with the analysis in the next two years. The evaluation will determine both the fissile loadings as well as the packaging requirements, such as basket configuration and filler materials for all DOE SNF types. Since no results were available at the time of the TSPA-VA data call, fissile loadings were selected for the DOE SNF canisters to determine how the package count might be affected. These load limits were adopted from an RW M&O study of aluminum fuel packaging and degradation scenarios [Reference 24]. These artificial loadings are not intended to be limiting values for any type or category of DOE SNF fuels proposed for repository disposal. As the evaluations on the fissile loading are completed, they will be used to determine the DOE SNF canister configuration and package counts.

The aluminum fuel study proposed the following package loading for the DOE-owned SNF based on the fuel enrichment level:

HEU (>20%) should not exceed 14.4 kg ^{235}U equivalent

LEU (>2%<20%) should not exceed 43 kg ^{235}U equivalent

VLEU (<2%) should not exceed 200 kg ^{235}U equivalent

Using this proposed fuel loading aluminum fuel, the following package loading for DOE SNF was developed to closely match the definition of LEU for commercial SNF and generally followed for use in the TSPA-VA. However, exceptions to these loading recommendations do exist for a small number of packages (i.e., some packages may exceed the proposed loading indicated below). As an example, a single element of the Shippingport LWBR in a disposal package will exceed the proposed limit unless the fuel is cut up in small sections for disposal. For the purpose of this evaluation, fuels like the Shippingport LWBR will be disposed intact (not cut up). This variance will have to be proved acceptable in a criticality safety evaluation for these specific fuels prior to the licensing application.

HEU (>20%) not to exceed 14.4 kg ^{235}U equivalent

MEU (>5%<20%) not exceed 43 kg ^{235}U equivalent

LEU (<5%) not exceed 200 kg ^{235}U equivalent

As indicated earlier, the categories or fuel groups for the TSPA-VA consist of one or more fuel types. These types may vary in terms of physical geometry, total mass, enrichment, or burn-up. While other groupings may have segregated the fuels by cladding, the categorization of fuels for the TSPA-VA resulted in analysis of fuels types by fuel matrix composition. No emphasis was placed on any further segregation by fuel cladding or enrichments within a given category. However, fuels from two different reactors within a given category were not "mixed" in the same SNF canister unless physical geometry, cladding, and BOL enrichments were similar. There were no attempts to load a variety of fuels in a canister to maximize fissile loading up to a prescribed limit or to minimize void volume.

Diameter differences in the SNF canisters are not dictated by anything other than the cross-section dimensions of the fuel to be loaded, and only secondarily by the fissile loads. Canister length will be determined by fuel length, with the majority of fuels destined for loading within 118.1 inches (3,000 mm) long canisters. Fuel canisters 179.9 inches (4,570 mm) long will be reserved for those fuels requiring the length to avoid disassembly. Selectively, the longer SNF canisters could also be used to stack shorter fuels. Co-disposal options for 179.9 inches (4,500 mm) SNF canisters should prove substantial since RW approved [Reference 37] the use of longer canisters in the HLW production facility intended for Hanford's liquid waste treatment facility.

Canister design will need to accommodate containment of the fuel load with a maximum pressure of 22 psia [Reference 25]. Based on the above, the DOE SNF categories are placed into the various canisters for repository disposal. Tables 5-1 and 5-2 summarize the canister size and count for each DOE SNF category based on 2,333 MTHM and ~2,500 MTHM respectively. Detailed canister size and count from each site are available on an EXCEL spread sheet and may be obtained from the NSNF Program. DOE EM plans to utilize five different containers. They are as follows: (1) ~18 inches diameter (17.6 inches OD, 0.59 inches thick wall) canister in ~10 feet length, (2) ~18 inches diameter canister in 15 feet length, (3) 24 inches diameter (24 inches OD, 0.375 inches thick wall) by 15 feet long canister for the fuel that does not fit into the ~18 inches diameter canister, (4) the MCO for the Hanford fuel (mainly N-Reactor), and (5) the large disposal package (LDP) for the commercial-like DOE SNF.

Table 5-1. DOE spent nuclear fuel canister size and count summary (2,333 MTHM).

Fuel Category	Fuel Matrix	~18" dia x 10' long	~18" dia x 15' long	~24" dia x 15' long	~25" dia x 15' long (MCO)	PWR21 ~5.4' dia x 15' long	
1	U-metal	2	4	0	380	0	
2	U-Zr	2	6	0	0	0	
3	U-Mo	66	0	0	0	0	
4	U-oxide intact	62	120	0	20	15	
5	U-oxide failed/declad	279	363	0	0	0	
6	U-Al Or U-Alx	628	31	0	0	0	
7	U-Si	154	47	0	0	0	
8	U/Th carbide hi-integrity	0	470	0	0	0	
9	U/Th carbide low-integrity	0	56	0	0	0	
10	U or U/Pu carbide nongraphite	3	2	0	0	0	
11	MOX	36	308	0	0	0	
12	U/Th oxide	14	9	44	0	0	
13	U-Zr-Hx	86	8	0	0	0	
14	Na-bonded	N/A	N/A	N/A	N/A	N/A	Will be treated
15	Classified-Navy	N/A	N/A	N/A	N/A	N/A	By Navy
16	Misc. SNF	23	19	0	0	0	
Total		1,355	1,443	44	400	15	

Table 5-2. DOE spent nuclear fuel canister size and count summary (All ~2,500 MTHM).

Fuel Category	Fuel Matrix	~18" dia x 10' long	~18" dia x 15' long	~24" dia x 15' long	~25" dia X 15' long (MCO)	PWR21 ~5.4' dia x 15' long	
1	U-metal	2	4	0	404	0	
2	U-Zr	2	6	0	0	0	
3	U-Mo	70	0	0	0	0	
4	U-oxide intact	66	127	0	20	16	
5	U-oxide failed/declad	298	388	0	0	0	
6	U-Al Or U-Alx	673	33	0	0	0	
7	U-Si	165	50	0	0	0	
8	U/Th carbide hi-integrity	0	503	0	0	0	
9	U/Th carbide low-integrity	0	60	0	0	0	
10	U or U/Pu carbide nongraphite	3	2	0	0	0	
11	MOX	38	329	0	0	0	
12	U/Th oxide	15	9	47	0	0	
13	U-Zr-Hx	92	8	0	0	0	
14	Na-bonded	N/A	N/A	N/A	N/A	N/A	Will be treated
15	Classified-Navy	N/A	N/A	N/A	N/A	N/A	By Navy
16	Misc. SNF	24	20	0	0	0	
Total		1,448	1,539	47	424	16	

6. REPOSITORY DISPOSAL PACKAGES

RW presently is considering approximately 13 disposal package designs to accommodate both the commercial as well as DOE-owned SNF. For the DOE SNF (the Navy is responsible for the Navy fuel), RW plans to place it in several waste package designs as indicated in the Interface Control Document [Reference 26]. Tables 6-1 and 6-2 summarize the disposal canisters the DOE SNF will be placed into for eventual disposal in the repository. Compatibility with the mined geologic disposal system (MGDS) has been given preliminary acceptance by the Yucca Mountain Repository through agreement set forth by the same Interface Control Document. DOE EM plans to utilize five different disposal packages. They are as follows: (1) a 5 x 1 co-disposal package with five HLW canisters and one ~18 inches diameter fuel canister in ~10 feet length, (2) a 5 x 1 co-disposal package with five HLW canisters and one ~18 inches diameter fuel canister in ~15 feet length, (3) a 3 x 1 co-disposal package with three HLW canisters and one 24 inches diameter by 15 feet long canister for the fuel that does not fit into the ~18 inches diameter canister, (4) a 0 x 4 disposal package with no HLW canisters and four MCO for the Hanford fuel (mainly the N-Reactor), and (5) the LDP for the commercial like DOE SNF. Figures 6-1 through 6-3 show the nominal DOE SNF arrangement for the non-LDP disposal packages.

Table 6-1. DOE spent nuclear fuel co-disposal size and package summary (2,333 MTHM).

Fuel Category	Fuel Matrix	5 HLW x 1 SNF x 10' long	5 HLW x 1 SNF x 15' long	3 HLW x 1 SNF x 15' long	No HLW x 4 MCO X15' long	Commercial PWR LDP	
1	U-metal	2	4	0	95	0	
2	U-Zr	2	6	0	0	0	
3	U-Mo	66	0	0	0	0	
4	U-oxide intact	62	120	0	5	15	
5	U-oxide failed/declad	279	363	0	0	0	
6	U-Al Or U-Alx	628	31	0	0	0	
7	U-Si	154	47	0	0	0	
8	U/Th carbide hi-integrity	0	470	0	0	0	
9	U/Th carbide low-integrity	0	56	0	0	0	
10	U or U/Pu carbide nongraphite	3	2	0	0	0	
11	MOX	36	308	0	0	0	
12	U/Th oxide	14	9	44	0	0	
13	U-Zr-Hx	86	8	0	0	0	
14	Na-bonded	N/A	N/A	N/A	N/A	N/A	Will be treated
15	Classified-Navy	N/A	N/A	N/A	N/A	N/A	By Navy
16	Misc. SNF	23	19	0	0	0	
Total		1,355	1,443	44	100	15	

Table 6-2. DOE spent nuclear fuel co-disposal size and package summary (All ~2,500 MTHM).

Fuel Category	Fuel Matrix	5 HLW x 1 SNF x 10' long	5 HLW x 1 SNF x 15' long	3 HLW x 1 SNF x 15' long	No HLW x 4 MCO X15' long	Commercial PWR LDP	
1	U-metal	2	4	0	101	0	
2	U-Zr	2	6	0	0	0	
3	U-Mo	70	0	0	0	0	
4	U-oxide intact	66	127	0	5	16	
5	U-oxide failed/declad	298	388	0	0	0	
6	U-Al Or U-Alx	673	33	0	0	0	
7	U-Si	165	50	0	0	0	
8	U/Th carbide hi-integrity	0	503	0	0	0	
9	U/Th carbide low-integrity	0	60	0	0	0	
10	U or U/Pu carbide nongraphite	3	2	0	0	0	
11	MOX	38	329	0	0	0	
12	U/Th oxide	15	9	44	0	0	
13	U-Zr-Hx	92	8	0	0	0	
14	Na-bonded	N/A	N/A	N/A	N/A	N/A	Will be treated
15	Classified-Navy	N/A	N/A	N/A	N/A	N/A	By Navy
16	Misc. SNF	24	20	0	0	0	
Total		1,448	1,539	0	106	16	

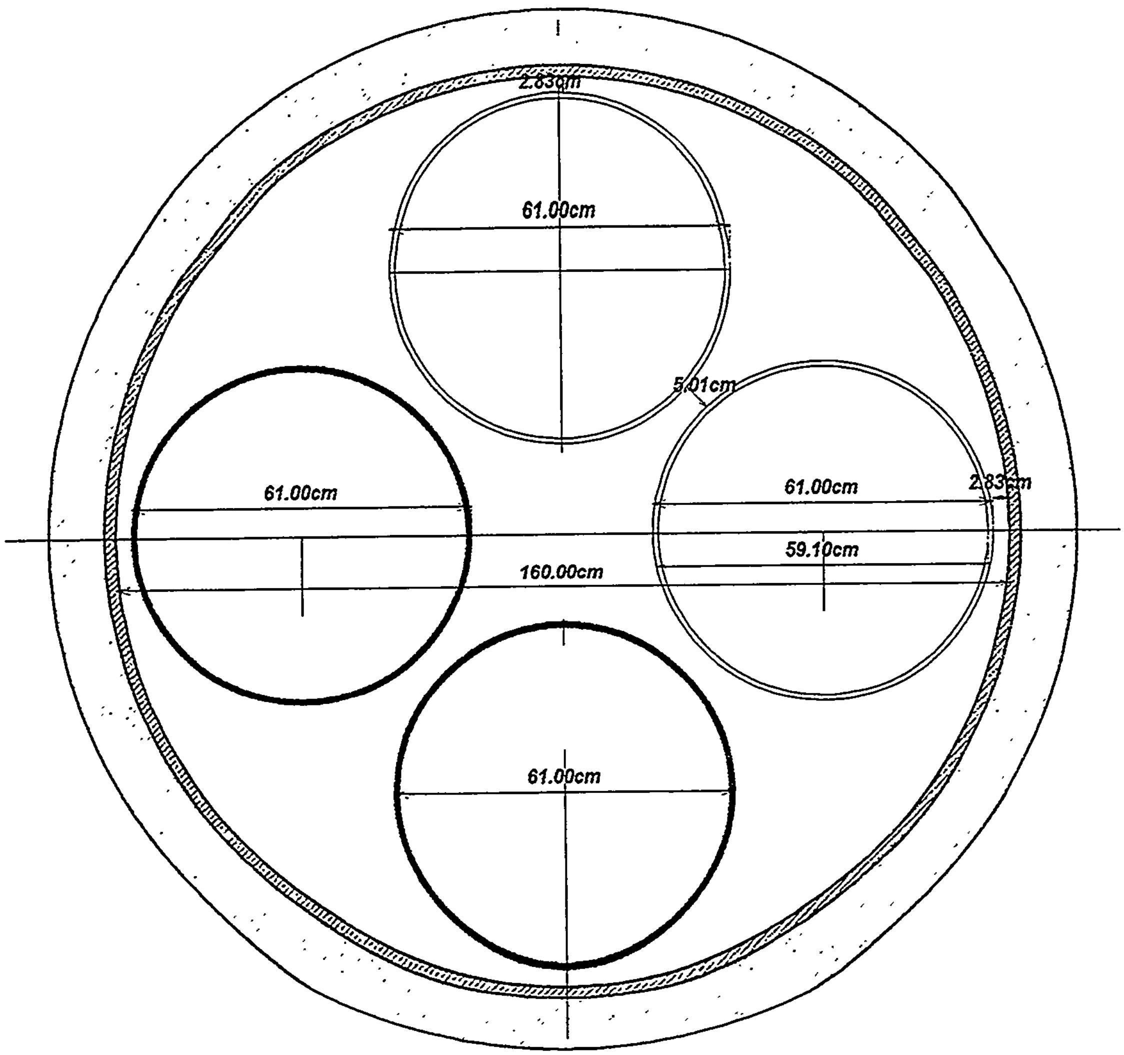

Figure 6-1. Proposed 3 (HLW) x 1 (SNF) co-disposal package.

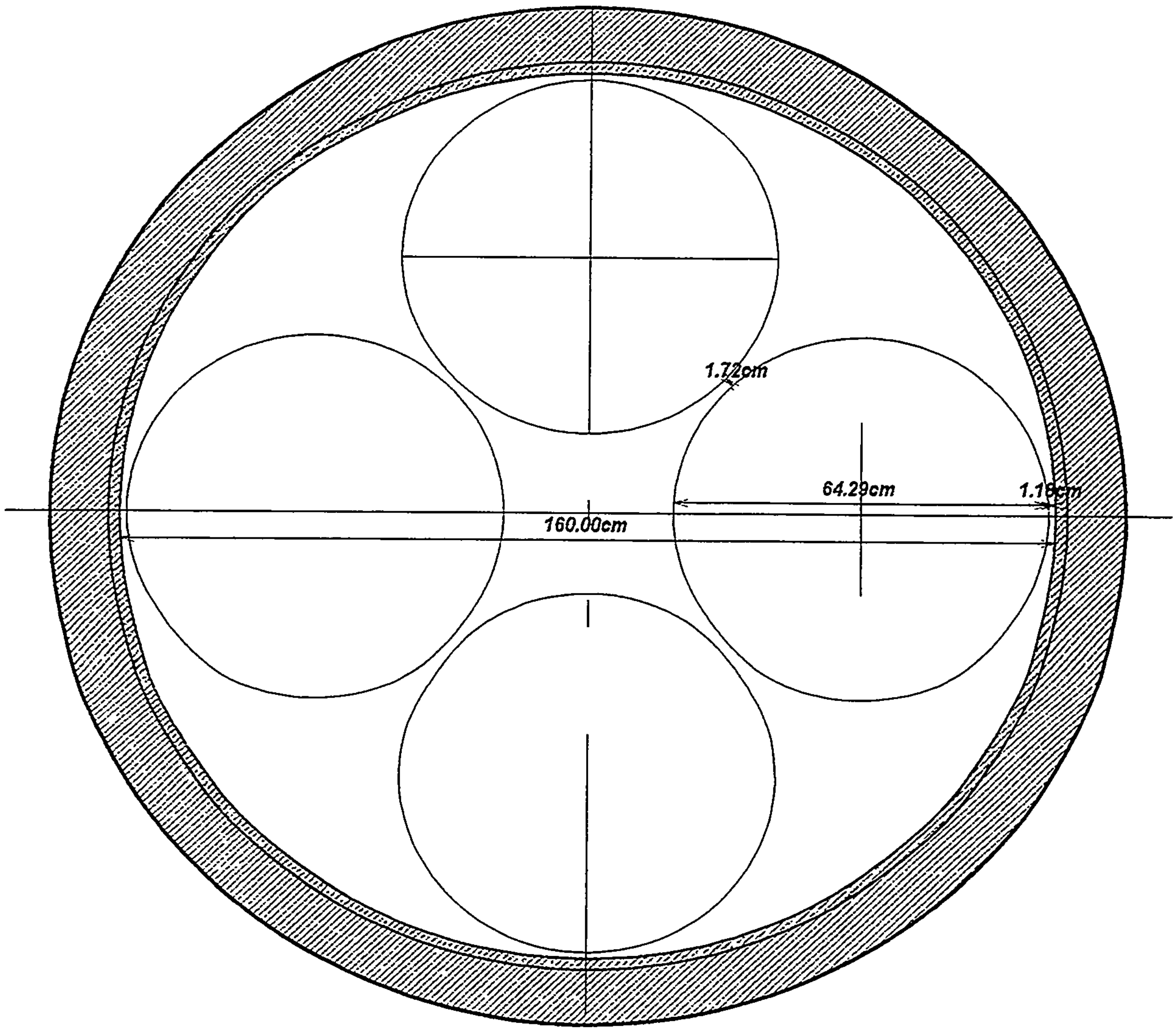

Figure 6-2. Proposed 0 (HLW) x 4 (MCO SNF) disposal package.

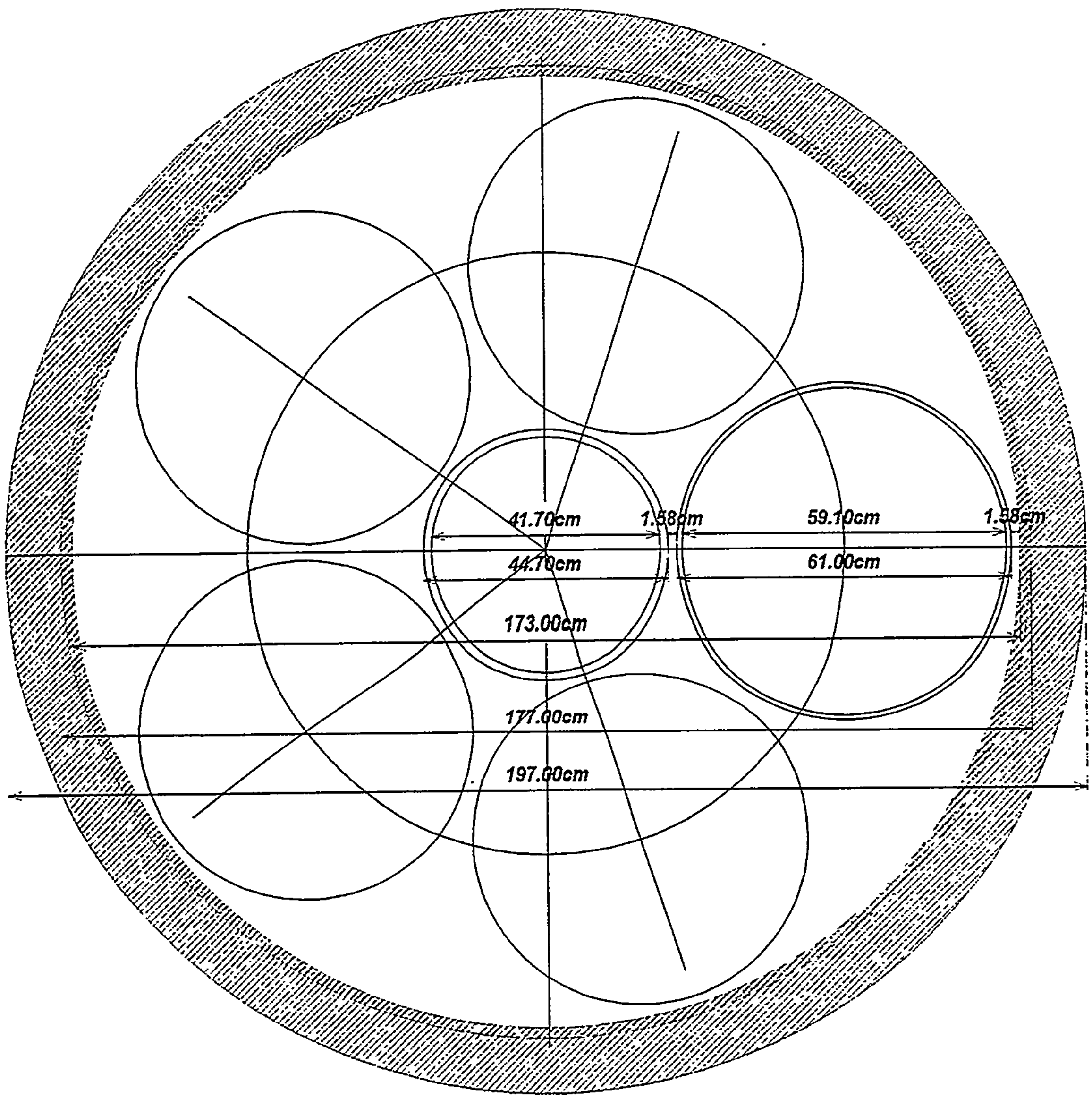

Figure 6-3. Proposed 5 (HLW) x 1 (SNF) co-disposal package [both ~10 and 15 feet lengths].

7. CALCULATING PACKAGE CURIE LOADING

7.1 DOE SNF Radionuclide Inventory

For the DOE SNF, one or more ORIGEN-2 runs were selected to estimate the total radionuclide inventory for each category. The specific ORIGEN-2 runs used to represent each category are indicated in the Table 7-1 below. As an example, for category 1, the N-Reactor fuel ORIGEN runs was used to represent the N-Reactor and the Single Pass Reactor fuels. A commercial PWR fuel ORIGEN run is used to represent the Heavy Water Components Test Reactor fuels. Similarly, the Oak Ridge Research Reactor fuel ORIGEN run is used to represent the EBR-II Targets and core filters.

Table 7-1. ORIGEN-2 runs used in the DOE fuel category.

Fuel Category	ORIGEN-2 Runs used to Represent various fuels in the category	Comment
1. U-metal	Commercial PWR fuel N-Reactor fuel Oak Ridge Research Reactor (ORR) fuel	N-Reactor fuel ORIGEN run was used to represent the Single Pass Reactor fuels
2. U-Zr	Advanced Test Reactor (ATR) fuel	No ORIGEN runs available. ATR was used because the reactor and fuel characteristics were similar (i.e., HEU fuel, high burnup test reactor)
3. U-Mo	Enrico Fermi Reactor (FERMI) fuel	
4. U-oxide (intact)	Commercial PWR fuel Commercial BWR fuel Pathfinder fuel Power Burst Facility (PBF) fuel Pulstar Buffalo fuel Shippingport PWR Fuel Transient Reactor Test (TREAT) fuel Fast Flux Test Facility (FFTF) oxide fuel ATR fuel	
5. U-oxide (failed or decladded)	Commercial PWR fuel Pulstar Buffalo fuel Three Mile Island (TMI) fuel PBF fuel ATR fuel Missouri University Research Reactor (MURR) fuel Rhode Island Nuclear Science Center (RINSC) fuel ORR fuel	

Table 7-1. (continued).

Fuel Category	ORIGEN-2 Runs used to Represent various fuels in the category	Comment
6. U-Al or U-Alx	MURR fuel RINSC fuel ORR fuel	
7. U-Si	MURR fuel RINSC fuel ORR fuel	
8. U/Th carbide hi-integrity	Fort St. Vrain (FSV) fuel Peach Bottom fuel General Atomics-High Temperature Gas Cooled Reactor (GA-HTGR) fuel	
9. U/Th carbide low-integrity	Peach Bottom fuel	
10. U/Th carbide nongraphite	Fast Flux Test Facility (FFTF) carbide fuel FSV fuel ATR fuel	
11. MOX	Fast Flux Test Facility (FFTF) oxide fuel	
12. U/Th Oxide	Shippingport Light Water Breeder Reactor (LWBR) fuel	ORIGEN runs for both seed and blanket
13. U-Zr-Hx	Training Research Isotopes- General Atomic (TRIGA) fuel	ORIGEN runs for both STD and FLIP
14. Na-bonded	Na-bonded fuel	SNF will be treated before disposal.
15. Classified	Classified Navy	By Navy
16. Misc.	N-Reactor fuel Fast Flux Test Facility (FFTF) oxide Fuel MURR fuel RINSC fuel ORR fuel Commercial PWR fuel ATR fuel FERMI fuel	

As noted in the table, category 2 fuels are represented by an ORIGEN-2 run from another category (ATR fuel) because no ORIGEN run was available for the category. In the future, the DOE SNF radionuclide inventories will be updated as the sites complete more ORIGEN runs for their fuels and this table will be updated accordingly.

The total radionuclide inventory for each DOE SNF category is shown in Table D-1 of Appendix D. A more detailed DOE SNF radionuclide inventory listing is in an EXCEL spreadsheet and is available from the NSNF Program.

7.2 HLW Radionuclide Inventory

The radionuclide inventory for the HLW canister was from the RW M&O 1995 TSPA report. According to TSPA-95, the inventory used was from the report *Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation*, DOE/RW-0184 published in 1987 [Reference 27]. Since the 1995 TSPA report radionuclide inventory was based on 118 inches (3,000 mm) long, 24 inches (610 mm) diameter standard canisters, for those SNF/HLW package combinations using 177 inches (4,500 mm) HLW canisters, the inventory may be obtained by multiplying the 118 inches long canister's inventory by 1.5. The inventory from RW M&O 1995 TSPA report is off by a factor of four and was corrected and used in the TSPA-VA. The radionuclide inventory for the HLW canister is shown in Table D-2 of Appendix D.

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Appendix A

DOE SNF Category Description

A-1. DOE SNF CATEGORY DESCRIPTION

The following section describes the typical fuels within each of the DOE SNF categories and the various informations for each of the fuel categories. The category title indicates the SNF matrix follow by the dominant cladding material in the category. As an example, category 1 consists of U-metal matrix with the dominant cladding material of zirconium.

A-1.1 Category 1 U-metal/zirconium

Typical fuel: N-reactor

Fuel Description

The N-Reactor fuel elements consist of two concentric tubes made of uranium metal co-extruded into Zircaloy-2 cladding. There are two basic types of fuel elements differentiated by their uranium enrichment Mark IV fuels elements a pre-irradiation enrichment of 0.947% U-235 in both tubes and an average uranium weight of 50 pounds (22.7 kg). The Mark IV fuels have an outside diameter of 2.4 inches (6.1 cm) and a length of 17.4, 13.2, 24.6, or 26.1 inches (44, 59, 62 or 66 cm). Mark IA fuel elements have a pre-irradiation enrichment of 1.25% U-235 in the outer tube and 0.947% U-235 in the inner tube. They have an average uranium weight of 35.9 pounds (16.3 kg). Mark IA fuels have an outside diameter of 2.1 inches (6.1 cm) and a length of 14.9, 19.6, or 20.9 inches (38, 50, or 53 cm) [Reference 4].

The degraded condition of the N-Reactor fuels has created a vulnerability issue relative to their continued storage in a water environment. Breach of the fuel element cladding and long-term water storage has created an apparent uranium hydride formation. The original proposed remediation of these fuels includes drying and controlled oxidation of the hydride to an oxide for interim storage in a package labeled as a multi-canister overpack (MCO) [Reference 28]. However, the current plans are limited to only cold vacuum drying of the N-Reactor fuels prior to placing them into the MCO. The MCO has experienced evolutionary design changes; the basic unit will contain a close packed arrangement of either Mark IV or Mark IA fuels. While the original concept of the MCO is not intended as a repository-approved disposal package, no alternative or proposed package exists at this time. The physical size of the MCO is akin to the standard HLW glass package, and will therefore be modeled as a 4-pack within the repository overpack.

Each MCO consists of a 24 inches (61 cm) outer diameter shell that is 164 inches (416.6 cm) long. The package has a 0.375 inches (0.95 cm) wall thickness, and uses 304L stainless steel construction. The approximate mass of the empty MCO is 3,900 pounds (1,700 kg).

Category 1 U-metal Fuel Inventory/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	Breached fuel cladding uranium metal with possible oxide surface coating	
Matrix dissolution rate	Metal model	Section 3.1
Surface area (m^2/g)	7.0E-05	
Clad failure fraction	Assume 100% failed	
Free radionuclide inventory fraction	0.1%	
Gap fraction	0%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 1 SNF (based on 2,333 MTHM).

	# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 1 U metal, failed clad					
- repository pkg count	2	4		95	
- HLW can count	10	20		0	
- SNF canister count	2	4		380	

Tables A-1 and A-2 provide a summary listing of the various chemical components associated with the typical N-Reactor fuels. Figure A-1 depicts a typical N-Reactor fuel element; Figures A-2 through A-4 depict proposed layout of N-Reactor fuel packaging within an MCO as it was evaluated in the performance assessment.

Table A-1 N-Reactor fuel element description.

	Mark IV				Mark IA		
Pre-irradiation enrichment of U235	0.947% Enriched				1.25-0.947% Enriched		
Type-Length code [a]	E	C	S	A	M	F	T
Outer length (cm)	66.3	62.5	58.9	44.2	53.1	49.8	37.8
Element diameter (cm)							
1. Outer of outer	6.15				6.1		
2. Inner of outer	4.32				4.5		
3. Outer of inner	3.25				3.18		
4. Inner of inner	1.22				1.12		
Cladding weight (kg)							
I. Outer element	1.09	1.04	0.99	0.79	0.88	0.83	0.66
2. Inner element	0.55	0.52	0.50	0.40	0.24	0.51	0.40
Weight of uranium in outer (kg)							
1. (0.947% 235U)	15.96	15.01	14.15	10.48			
2. (1.25% 235U)					11.07	10.39	7.85
Weight of uranium Inner (kg) 0.947%	7.48	7.03	6.62	4.94	5.49	5.12	3.90
Weighted average of uranium in element (kg)	22.68				16.28		
Ratio of Zircaloy-2 to uranium (kg/MT)	70.0	70.8	71.6	77.1	85.5	86.3	90.4
Weighted ave. (kg/MT)	63.76				77.73		
% of total elements	63				37		
% of length type of each fuel	78	10	7	5	87	10	3
Displacement Volume(l/MT uranium)	66.77				66.77		

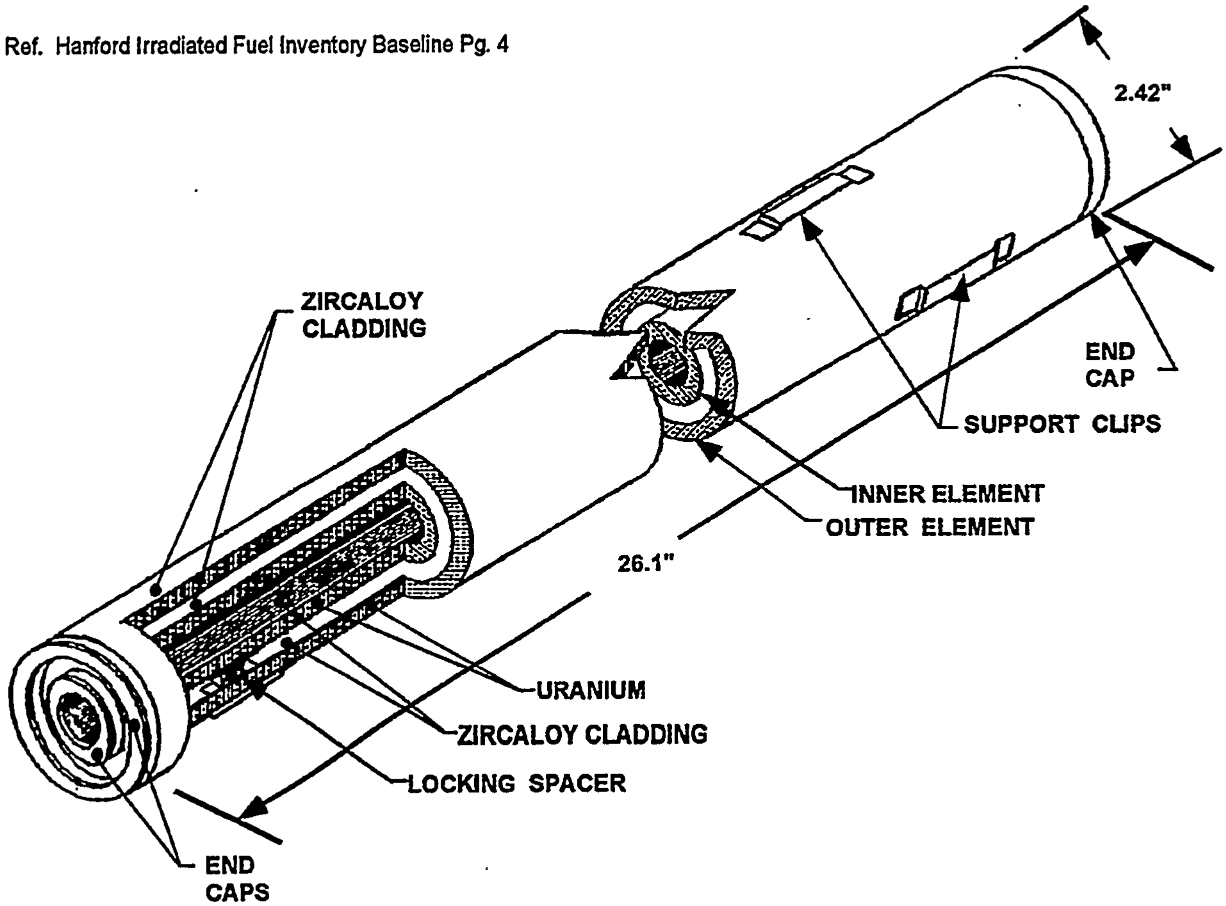
a. Letter code differentiates the different lengths of the Mark IV or Mark IA fuel elements, i.e., a type "E" element is 66.3 cm long. [Hanford Irradiated Fuel Inventory Baseline]

Table A-2 Chemical composition of 105-N-Reactor fuel elements.[a]

Element	Uranium Alloy 601	Zircaloy-2	Braze Filler
Aluminum	700-900	75	145
Beryllium	10	—	4.75-5.25 wt %
Boron	0.25	0.50	0.50
Cadmium	0.25	0.50	0.50
Carbon	365-735	275	500
Chromium	65	0.05-0.15 wt %	0.05-0.15 wt %
Cobalt	—	10	20
Copper	75	50	60
Hafnium		200	200
Hydrogen	2.00	25	50
Iron	300-400	0.07-0.20 wt %	0.06-0.21 wt %
Lead	—	100	130
Magnesium	25	20	60
Manganese	25	50	60
Molybdenum	—	50	50
Nickel	100	0.03-0.08 wt %	0.03-0.08 wt %
Nitrogen	75	80	200
Oxygen	—	—	2300
Silicon	124	100	250
Sodium	—	20	20
Tin	—	1.20-1.70 wt %	1.14-1.70 wt %
Titanium	—	50	50
Tungsten		50	100
Uranium	Balance	3.50	4
Vanadium	—	50	50
Zirconium	65	Balance	Balance

a. Concentrations given in parts per million (ppm) maximum or ppm range, unless indicated otherwise.

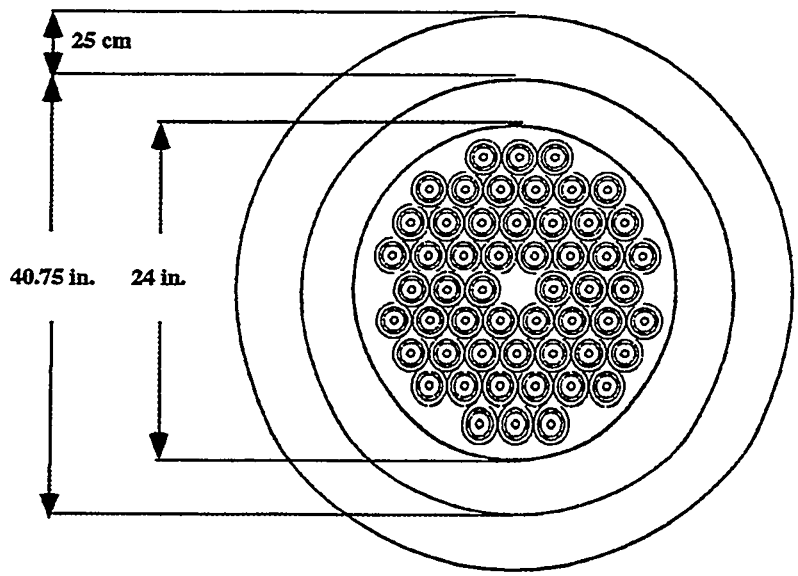
* Ref. Hanford Irradiated Fuel Inventory Baseline Pg. 4



105-N REACTOR MARK IV FUEL ELEMENT ASSEMBLY

Figure A-1. N-Reactor Mark IV fuel element assembly.

Loading Arrangement for Mark IV Fuel in MCO Container.



Loading Arrangement for Mark IA Fuel in MCO Container.

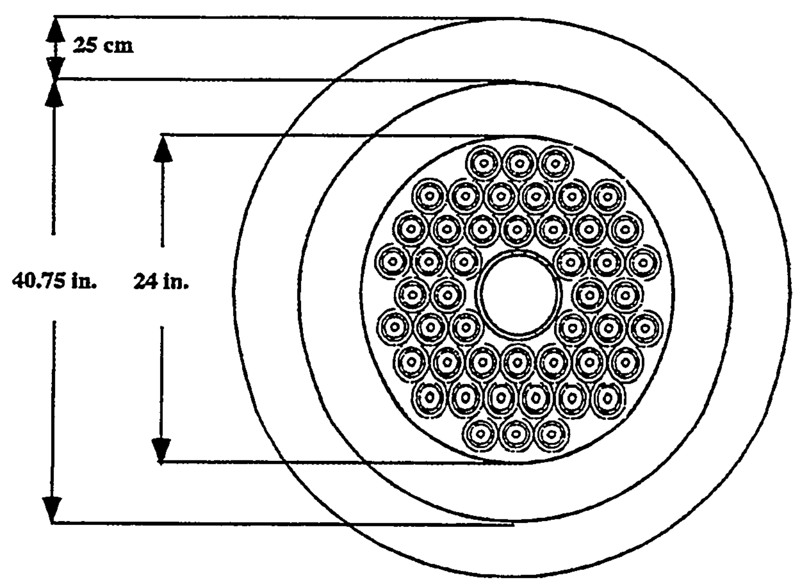


Figure A-2. (Top) Loading arrangement for Mark IV fuel in MCO container.
Figure A-3. (Bottom) Loading arrangement for Mark 1A fuel in MCO container.

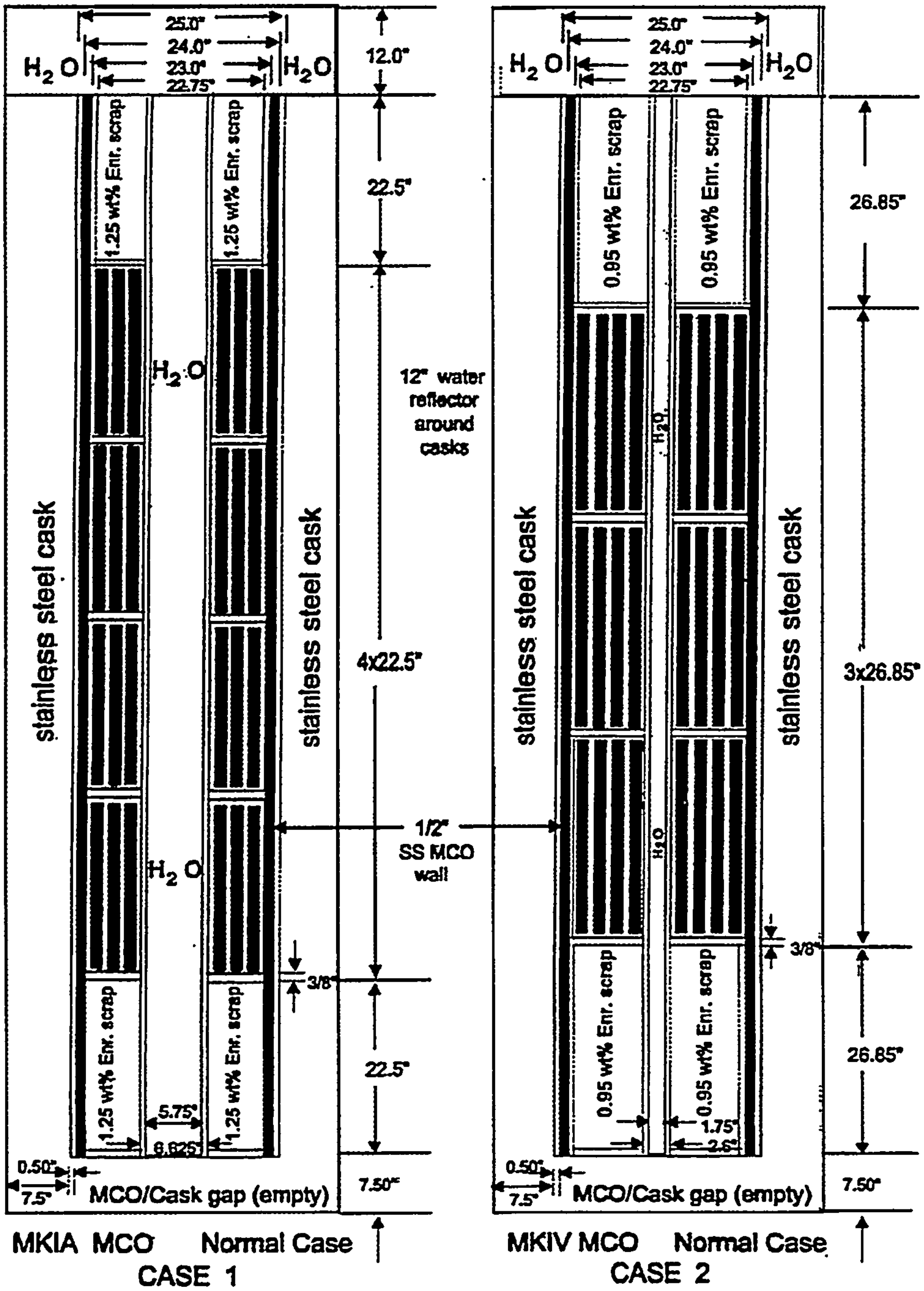


Figure A-4. MCO axial geometry layout.

A-1.2 Category 2 U-Zr/Zirconium

Typical fuel: HWCTR, CP-5

Fuel Description

The Heavy Water Components Test Reactor (HWCTR) is a tank-type, fully enriched (93%) uranium, heavy water moderated and cooled reactor. The purpose of the reactor was to test fuel elements, materials, and components for heavy water reactors at power reactor conditions. The reactor had a nominal thermal power of 61 MW. The driver fuel elements are located around the outside part of the reactor with up to 12 of the test fuel elements placed in the center of the reactor.

The driver fuels are tube type design with 2.3 inches (5.84 cm) outside diameter, 1.96 inches (4.98 cm) inside diameter and 113 inches (287 cm) long. The fuel meat is 0.137 inch (0.348 cm) thick consisting of 93% enriched uranium alloyed with 90.7 wt % zirconium. Figure A-5 contains a section view of the driver element [Reference 13]. The test elements are made of natural or slightly enriched uranium metal or uranium oxide. Thus, they are not included in this category.

Category 2 U-Zr Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	93% enriched uranium alloyed with 90.7 wt % zirconium	
Matrix dissolution rate	Metal model	Section 3.2
Surface area (m^2/g)	6.5E-03	
Clad failure fraction	Assume 10% failed	
Free radionuclide inventory fraction	0.001%	
Gap fraction	0%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 2 SNF (based on 2,333 MTHM).

	# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 2 U-Zr, Zr clad					
- repository pkg count	2	6			
- HLW can count	10	30			
- SNF pkg count	2	6			

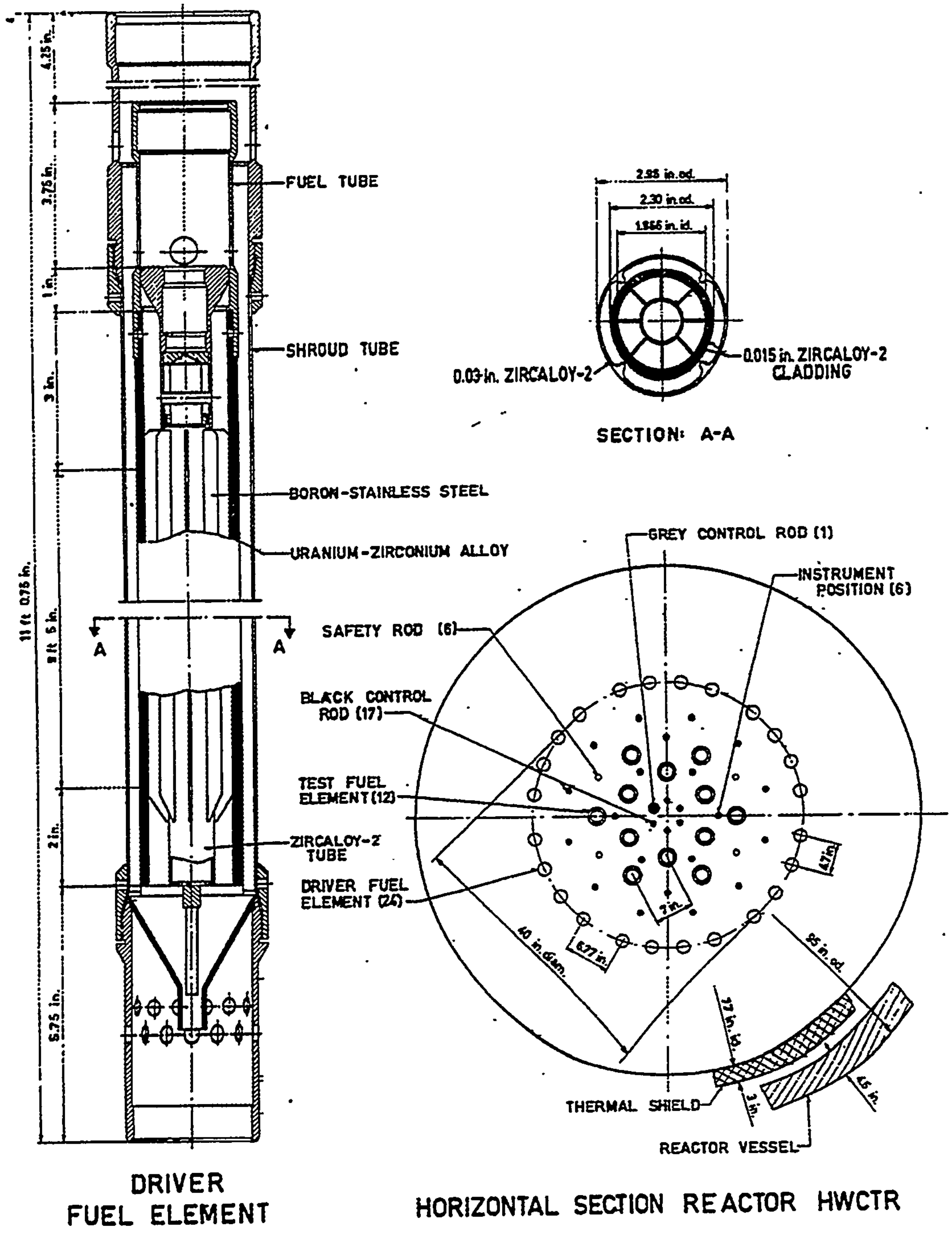


Figure A-5. Section view of the HWCTR driver fuel and reactor.

A-1.3 Category 3 U-Mo/Zirconium

Typical fuel: Fermi

Fuel Description

Fermi was a sodium-cooled fast breeder reactor with intermediate sodium loops, sodium-to-water steam generators, and an associated steam-driven turbine-generator. The lower reactor section of the reactor vessel has a 9.5 feet (289.56 cm) outside diameter and is 96.5 inches (245.11 cm) in height. Core and blanket subassemblies are housed within the lower reactor vessel and are cooled by sodium that flows from the bottom of the lower reactor through the subassemblies and up into the upper reactor vessel. Each subassembly has a nozzle attached to the bottom end for insertion into the two 2 inches support plates spaced 14 inches apart. The core and blanket of Fermi were made up of 2.646 inches (6.72-cm) square driver core and blanket subassemblies positioned to approximate a right circular cylinder approximately 80 inches in diameter and 70 inches tall. Figure A-6 shows the configuration of the core subassembly. The reactor core region was 30.5 inches in diameter and 31.2 inches tall and was completely enclosed by a thick breeder blanket that was designed to give a high breeding ratio and provide shielding.

The radial blanket fuel subassembly is made up of an inlet nozzle, a lower axial blanket, a fuel section, and an upper axial blanket. The radial blanket fuel subassemblies were made up of 25 cylindrical rods fabricated from depleted U-Mo alloy, encased in stainless steel tubes and bonded with sodium. The radial blanket subassemblies are currently stored dry in ICPP-749. The radial blanket subassembly rods contain depleted uranium and sodium and thus will be treated prior to final disposition. Those rods are not part of the category 3 inventory.

The Fermi driver fuel subassembly was designed with three active regions — a lower axial blanket, a fuel section, and an upper axial blanket. The lower and upper axial blanket subassemblies have been cropped off from the central core fuel section and are currently stored with the radial blanket subassemblies in ICPP-749 and will be treated prior to final disposal. A type 347 stainless steel square tube measuring 2.646 inches square with a 0.096 inch wall thickness was used as the outside structure to hold the three regions together. The fuel section contained 140 fuel pins, made up of 25.69% enriched uranium-molybdenum alloy. Four stainless steel structural support pins were inserted into the corner positions of the 12 x 12 array to add structural support to the fuel section and the fuel subassembly. The fuel pins were closely packed into the 2.646 inches square tube. The fuel pins were maintained on a square pitch of 0.200 inches in a cartridge made of stainless steel wires and plates.

The fuel pin is made up of a solid uranium-molybdenum alloy fuel meat, 0.148 inch in diameter, metallurgically bonded to a zirconium tube. The fuel material is 90 weight percent uranium that has been enriched to a nominal 25.69 percent in U-235, and 10 weight percent molybdenum. The fuel pins were originally fabricated in lengths of 12 feet or greater and were later cut into 30.5 inches sections with the ends pointed by cold swaging. Following the sectioning, each pin was subjected to heat treatment to provide for stress relief. Next, prefabricated zirconium caps were placed on the end of each pin and secured in place by cold swaging. The total length of the fuel pins including the zirconium endcaps is 32.78 inches. A slot was made in the bottom cap of the fuel pin for anchoring purposes [Reference 15].

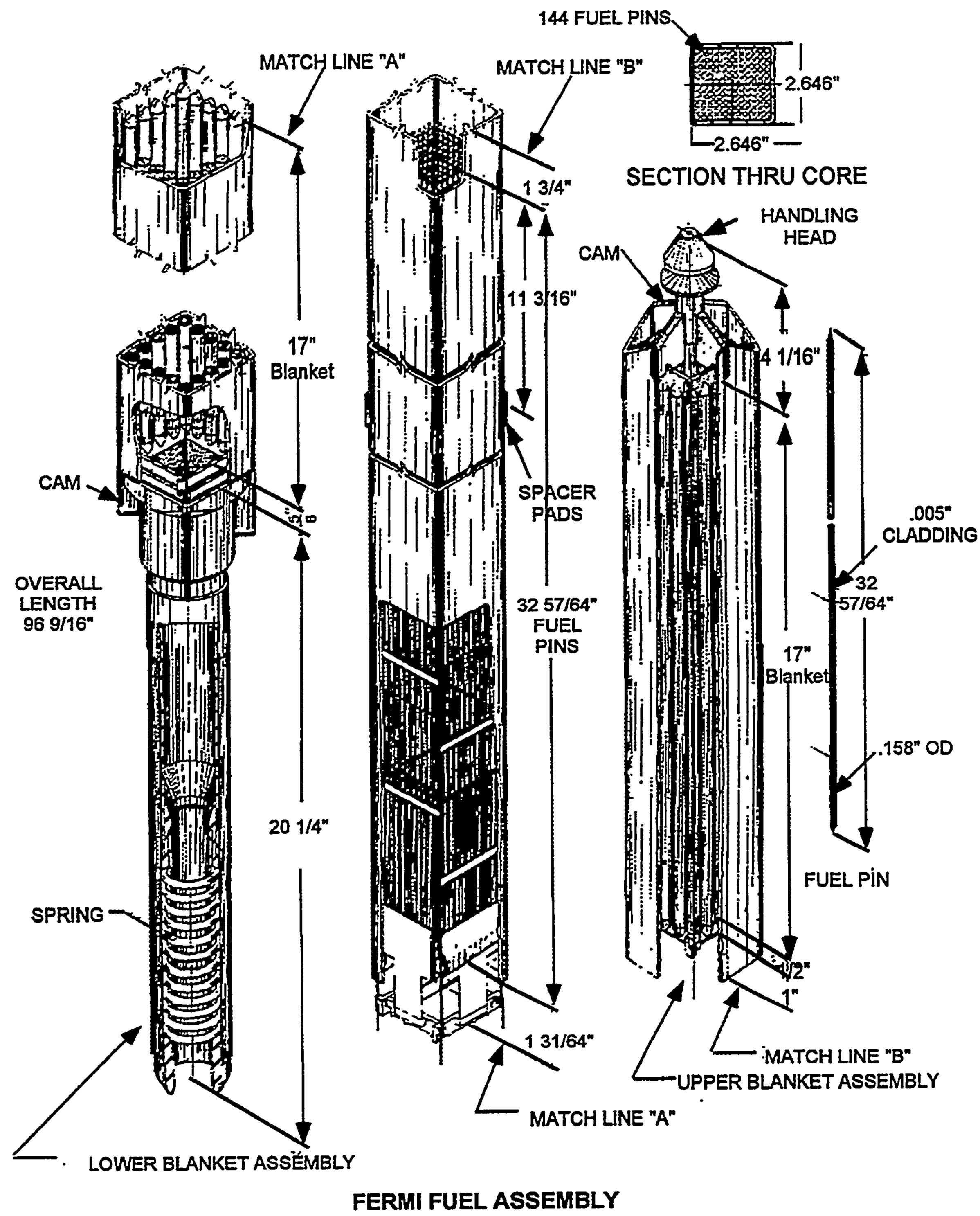


Figure A-6. Fermi driver fuel subassembly.

Category 3 U-Mo Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	Nominal 25.69% in U-235, and 10 wt % Mo	
Matrix dissolution rate	Metal model x 10	Section 3.3
Surface area (m^2/g)	4.0E-04	
Clad failure fraction	Assumé 10% failed	
Free radionuclide inventory fraction	0.001%	
Gap fraction	0%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 3 SNF (based on 2,333 MTHM).

	# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 3 U-Mo, Zr clad					
- repository pkg count	66				
- HLW can count	330				
- SNF pkg count	66				

A-1.4 Category 4 U Oxide/Zirconium & Stainless Steel

Typical fuel: Shippingport (HEU), commercial (LEU), Saxton (MEU), ML-1 (HEU), PBF (MEU), FFTF-TFA (LEU)

Fuel Description

The fuels in this category generally have the characteristics found in most of the commercial fuels (PWR and BWR). For one reason or another, these fuels have ended up in the DOE SNF inventory. As an example, the commercial fuels were brought to the DOE site for examination or testing programs while some were reconfigured for the Dry Rod Consolidation Test (DRCT) at the INEEL. The reconfiguration involved consolidating the fuel by removing the rods and placing them into canisters so as to double the number of rods in a volume equal to a standard commercial fuel assembly. Other examination or testing involved taking some of the assemblies and rods apart for post-irradiation examination. The fuel compositions, properties, and conditions are identical to the commercial fuel [Reference 29].

The Power Burst Facility (PBF) was used to test fuel materials and the driver fuel was included in the category 4 inventory. The PBF driver core fuel contains a pelletized ternary fuel (UO2-ZrO2-CaO-18.5% enriched) surround by a helium gas annulus, an insulator sleeve of (ZrO2-CaO), and cladded with 304L stainless steel. This fuel is similar to commercial fuel that is made by pressing the uranium oxide into pellets. The pellets are loaded into stainless steel tubes [Reference 15].

Another fuels such as the Shippingport PWR Core 2 Seed 2 were also included in the category 4 inventory. The Shippingport PWR was built to demonstrate the concept of a light water, slow breeder reactor using a commercial type pressurized water reactor (PWR). This was a joint AEC/Navy project that was designed for development and demonstration purposes of this type of reactor. Bettis Atomic Power Laboratory designed the reactor. The Naval Reactors Group of the AEC directed the project, and the power was distributed by Duquesne Light Company. The Navy's NRF and ECF facilities received the fuel after it was removed from the core. The Navy played a large part in all aspects of this reactor. Shippingport was designed and built to test different core designs and explore operating variables for large-scale nuclear reactors. The reactor was of the seed and blanket type and began operation with the first core (PWR-C1) in December 1957. The seed element was a zircaloy-clad plate-type fuel while the blanket fuel was in the form of pellets placed inside short (~10 inches) Zircaloy-2 tubes. The basic component of the seed elements was the fuel plate. A plate was formed by sandwiching an enriched (~93%) U-Zr alloy strip between two zircaloy-2 cover plates and four side strips. Figure A-7 shows the Shippingport PWR fuel subassembly [Reference 15].

Category 4 U-Oxide Intact Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	Pressed uranium oxide pellets	
Matrix dissolution rate	Commercial model	Section 3.4
Surface area (m^2/g)	9.5E-04	
Clad failure fraction	Assume 100% failed	
Free radionuclide inventory fraction	0.00%	
Gap fraction	1-2%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 4 SNF (based on 2,333 MTHM).

	# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 4 U oxide, Zr/SST clad					
- repository pkg count	62	120		5	15
- HLW can count	310	600		0	0
- SNF canister count	62	120		20	15

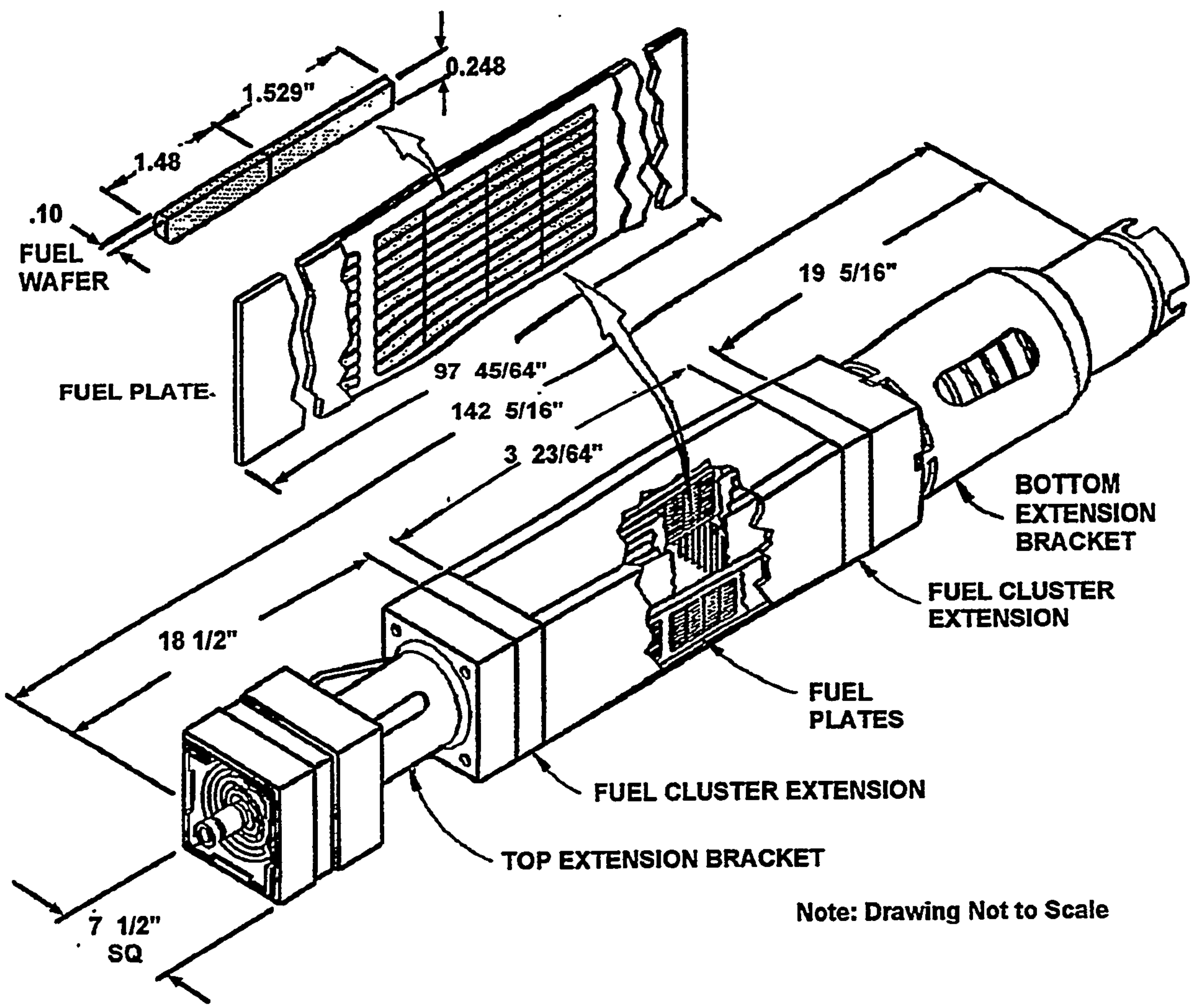


Figure A-7. Shippingport Core 2 Blanket Fuel Assembly

A-1.5 Category 5 U Oxide/Failed Clad or Aluminum

Typical fuel: SM-1A, ORNL SST & Zr (MEU), TMI-2 (LEU), HFIR, FRR, MTR

Fuel Description

The fuels in this category represent those materials that are already damaged, disrupted, or considered the least robust in terms of immediate fissile and fission product movement upon package breach. Many of the fuels in this category have been disrupted from their original configuration for number of reasons such as operational activities, testing, accidents, or destructive examination.

The bulk of this category consists of the packaged TMI-2 debris. The fuel was a typical commercial pressurized water nuclear reactor fuel until it melted in a reactor accident. It now consists of materials with sizes ranging from fines to nearly intact assemblies. Some of which have been melted and cooled. The fuel debris was placed into three types of stainless steel canisters: filter canister that contain the fines, knockout canisters that contain gravel consistency materials, and fuel canisters that contain large pieces of melted or unaffected assemblies. The materials have been extensively characterized as part of the TMI-2 reactor analysis [Reference 15].

Primary issues related to packaging this fuel category for disposal related to: (1) packaging for criticality control, and (2) drying material to prevent gas generation. Figure A-8 shows the canister configuration for the Three Mile Island unit 2 (TMI-2).

Category 5 U-Oxide Failed Clad Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	Pressed uranium oxide pellets	
Matrix dissolution rate	Commercial model	Section 3.5
Surface area (m^2/g)	9.5E-02	
Clad failure fraction	Assume 100% failed	
Free radionuclide inventory fraction	0.00%	
Gap fraction	0.01%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 5 SNF (based on 2,333 MTHM).

	# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 5 U oxide, mixed clad					
- repository pkg count	279	363			
- HLW can count	1,395	1,815			
- SNF pkg count	279	363			

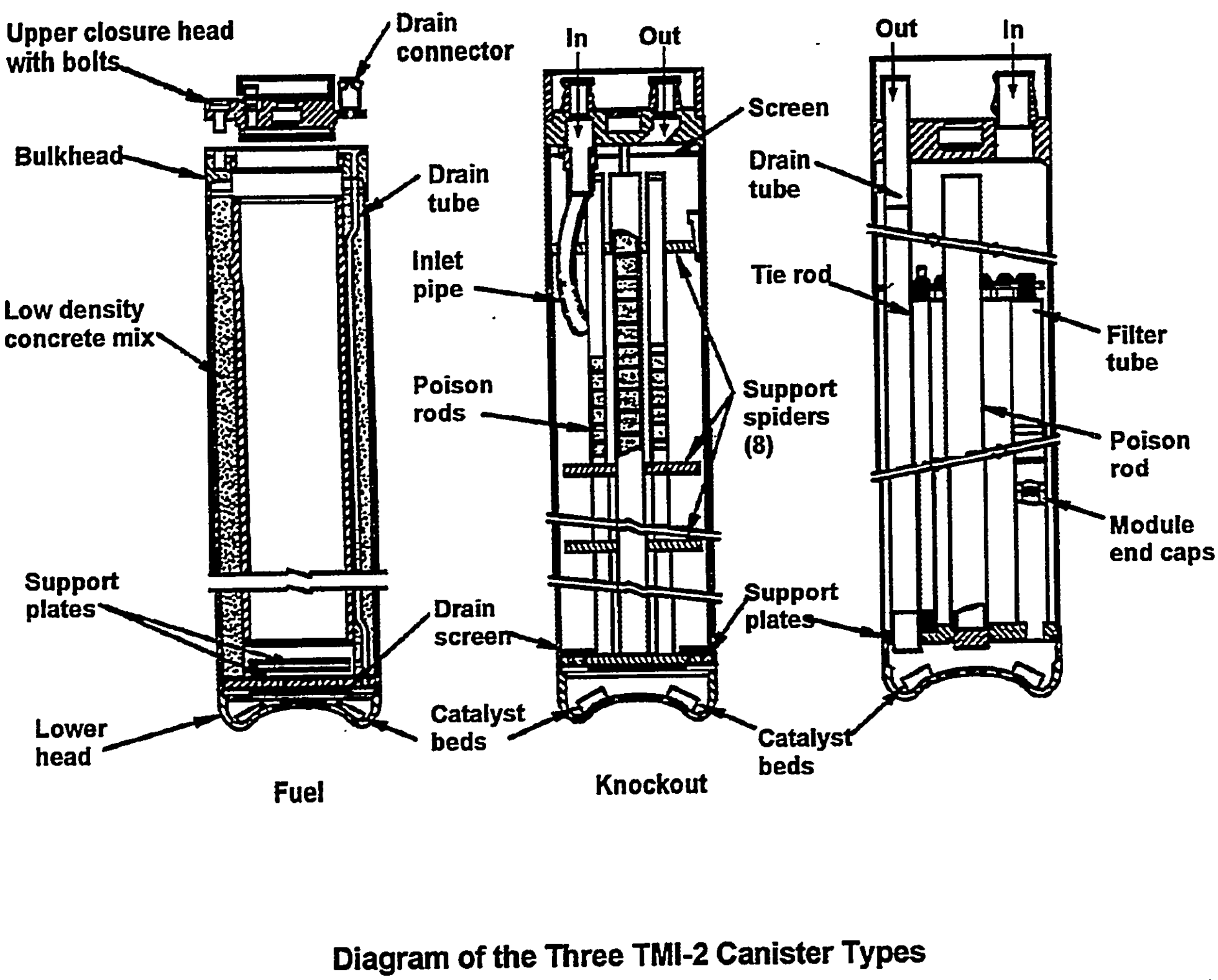


Figure A-8. TMI-2 canister types.

A-1.6 Category 6 U-Al_x / Aluminum

Typical fuel: ATR (HEU), MTR, FRR (MEU)

Fuel Description

This category includes fuels composed of a uranium-aluminum alloy. The cladding is assumed to be intact at this time, but is not considered to be a very durable material in long-term storage conditions in wet environments.

The typical Advanced Test Reactor (ATR) fuel element consists of 19 curved aluminum-clad fuel plates swaged into two non-fueled aluminum side plates. The 19 curved (concentric) aluminum-clad UAlx fuel plates form a pie-shaped geometry. The fuel meat consists of UAlx, boron, and aluminum particles mixed together and pressed into a 0.015 inch thick plate and clad with a 6061 aluminum foil (nominally 15 mils). The uranium and poison loadings are varied among the fuel plates giving a total U-235 loading of 1,075 grams per fuel element [Reference 15]. Figure A-9 shows the ATR fuel configuration. MTR is of a similar plate design but uses flat than curved plates.

Other UAlx fuels are similarly constructed and generic fuel information is indicated below.

Category 6 U-Al_x Al Clad Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	UAlx dispersed in aluminum	
Matrix dissolution rate	Metal model x 0.1	Section3.6
Surface area (m^2/g)	7.4E-03	
Clad failure fraction	Assume 100% failed	
Free radionuclide inventory fraction	0.01%	
Gap fraction	0	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-disposed the category 6 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 6	U alloy, aluminum clad					
	- repository pkg count	628	31			
	- HLW can count	3140	155			
	- SNF pkg count	628	31			

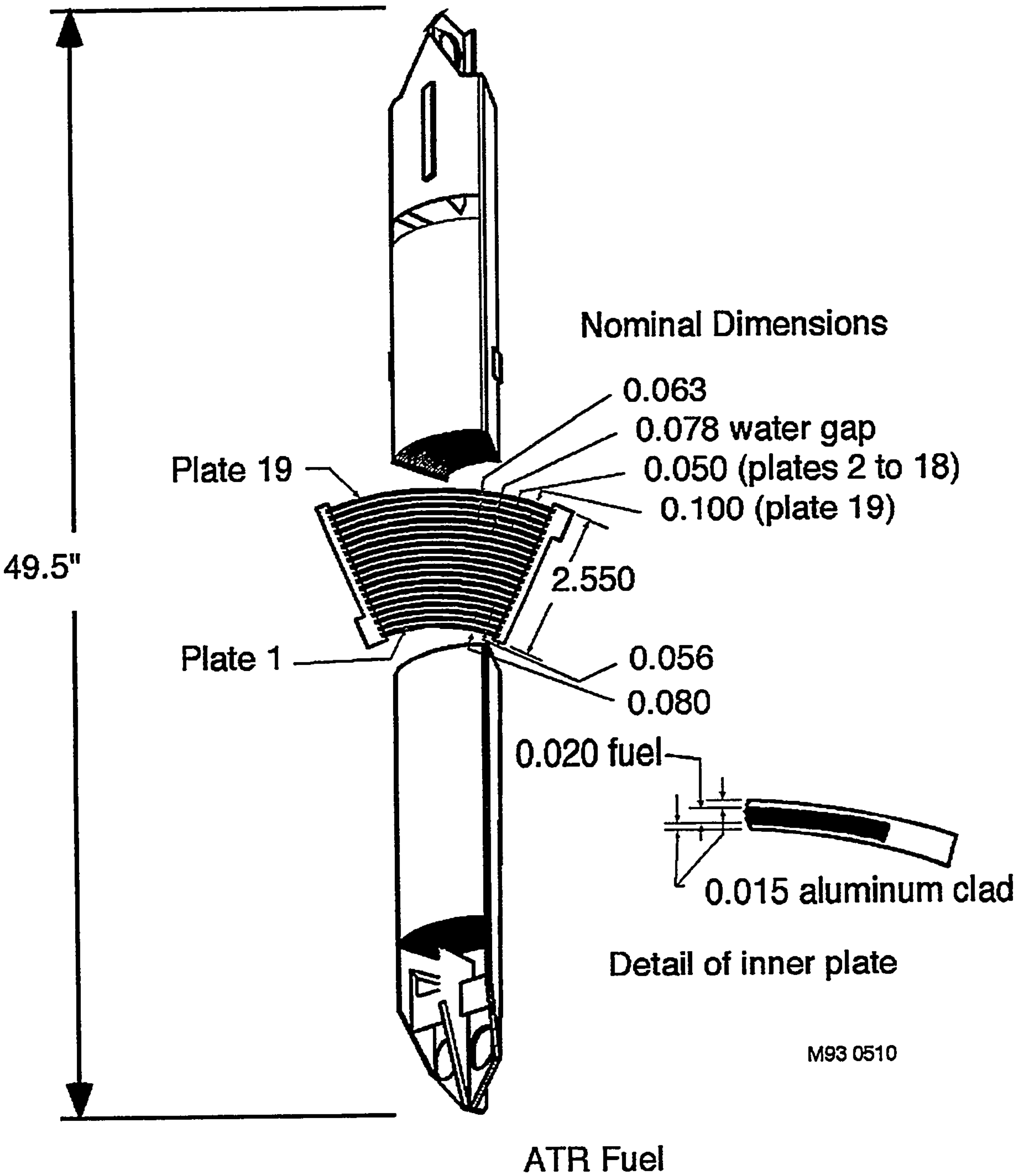


Figure A-9. ATR fuel element.

A-1.7 Category 7 U-Si

Typical fuel: MTR, FRR (HEU, MEU)

Fuel Description

The typical fuels in this category come from material test reactors and foreign research reactors (FRR). Most foreign research reactors will continue to operate during the next several years. Foreign reactor reactors use a number of different fuel designs. These designs can be placed into five groups: (1) plate-type design, (2) concentric tube-type design, (3) pin-type design, (4) special-type design, and (5) rod-type design. The various designs are shown in the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, Appendix B, DOE/EIS-0218F February 1996.

The plate type design is described here since it is used in the majority of the FRR fuels. The thermal power of these reactors ranges from 1 MW to 50 MW. Each fuel assembly contains from 6 to 23 plates and an initial U-235 content of 37 to 420 grams. The fuel matrix consists of U-Si dispersed in aluminum. Figure A-10 shows the plate-type MTR element.

Category 7 U-Si Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	U-Si dispersed in aluminum	
Matrix dissolution rate	Metal model x 0.1	Section 3.7
Surface area (m^2/g)	1.4E-02	
Clad failure fraction	Assume 100% failed	
Free radionuclide inventory fraction	0.01%	
Gap fraction	0	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 7 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 7	U silicide, aluminum clad					
	- repository pkg count	154	47			
	- HLW can count	770	235			
	- SNF pkg count	154	47			

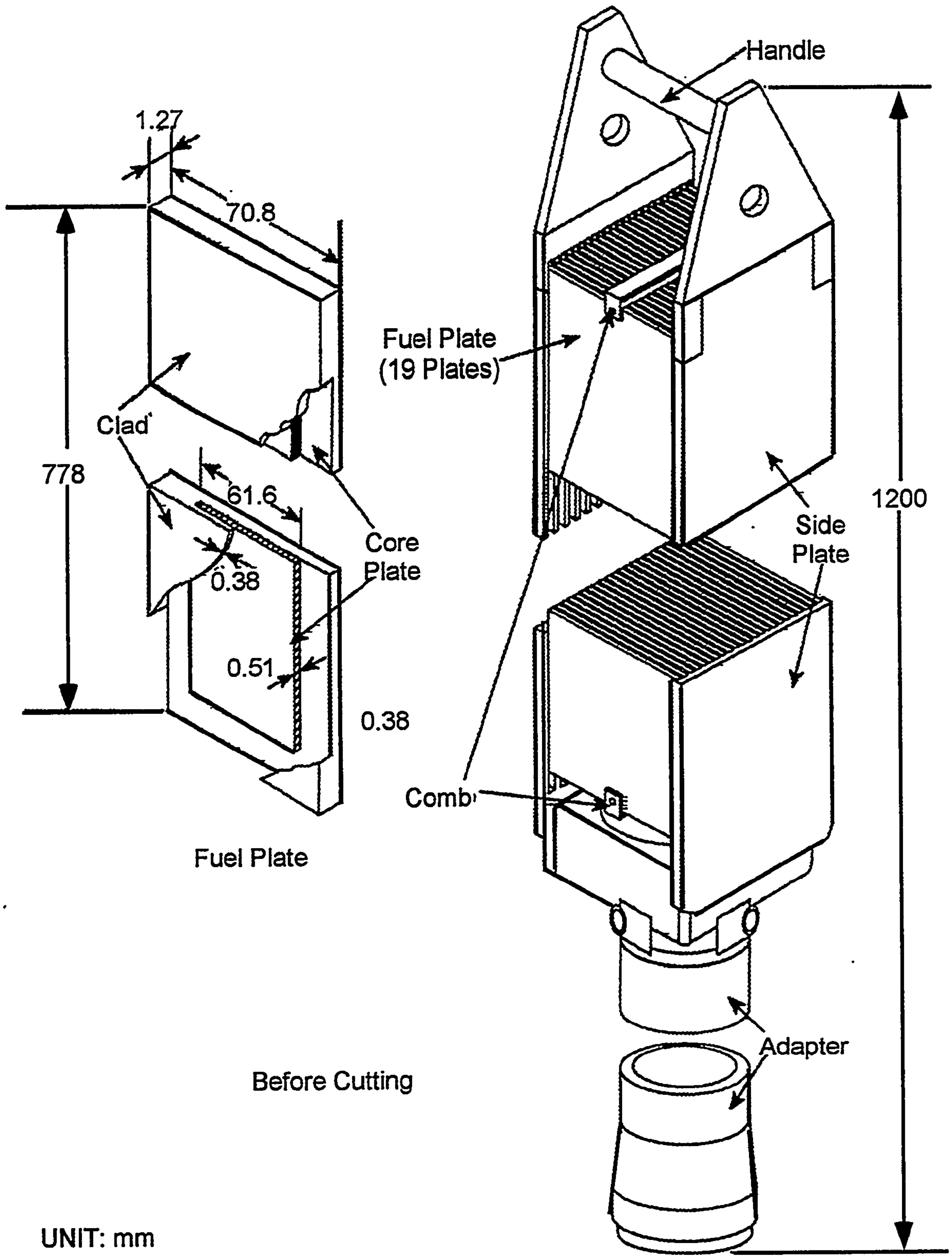


Figure A-10. Typical (boxed-type/flat plate) FRR fuel element.

A-1.8 Category 8 U/Th Carbide (Hi-Integrity)/Graphite

Typical fuel: Fort Saint Vrain (FSV)

Fuel Description

The FSV fuel is a graphite-based fuel that was used only in the Fort Saint Vrain Reactor. An assembly is composed of a hexagonal shaped graphite block drilled with 102 coolant holes and 210 fuel holes. The fuel is made of highly enriched uranium carbide and thorium carbide spheres coated with layers of pyrolytic carbon followed by a SiC protective outer coating, which is very durable, and an outer pyrolytic coating. The fuel spheres are sintered with carbon and formed into rods, called compacts, and then stacked into the fuel holes within large hexagonal blocks of graphite. These blocks are 14.172 inches (36 cm) across the flats, 8.102 inches (20.6 cm) on each side, and 31.22 inches (79.3 cm) long [Reference 15]. Figure A-11 shows the Fort Saint Vrain fuel assembly.

Category 8 U/Th carbide (high-integrity) Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	U/Th carbide	
Matrix dissolution rate	Si carbide model	Section 3.8
Surface area (m^2/g)	2.2E-02	
Clad failure fraction	Assume 1% Failed	
Free radionuclide inventory fraction	0.001%	
Gap fraction	0	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 8 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 8	Th/U carbide, graphite (FSV)					
	- repository pkg count		470			
	- HLW can count		2350			
	- SNF pkg count		470			

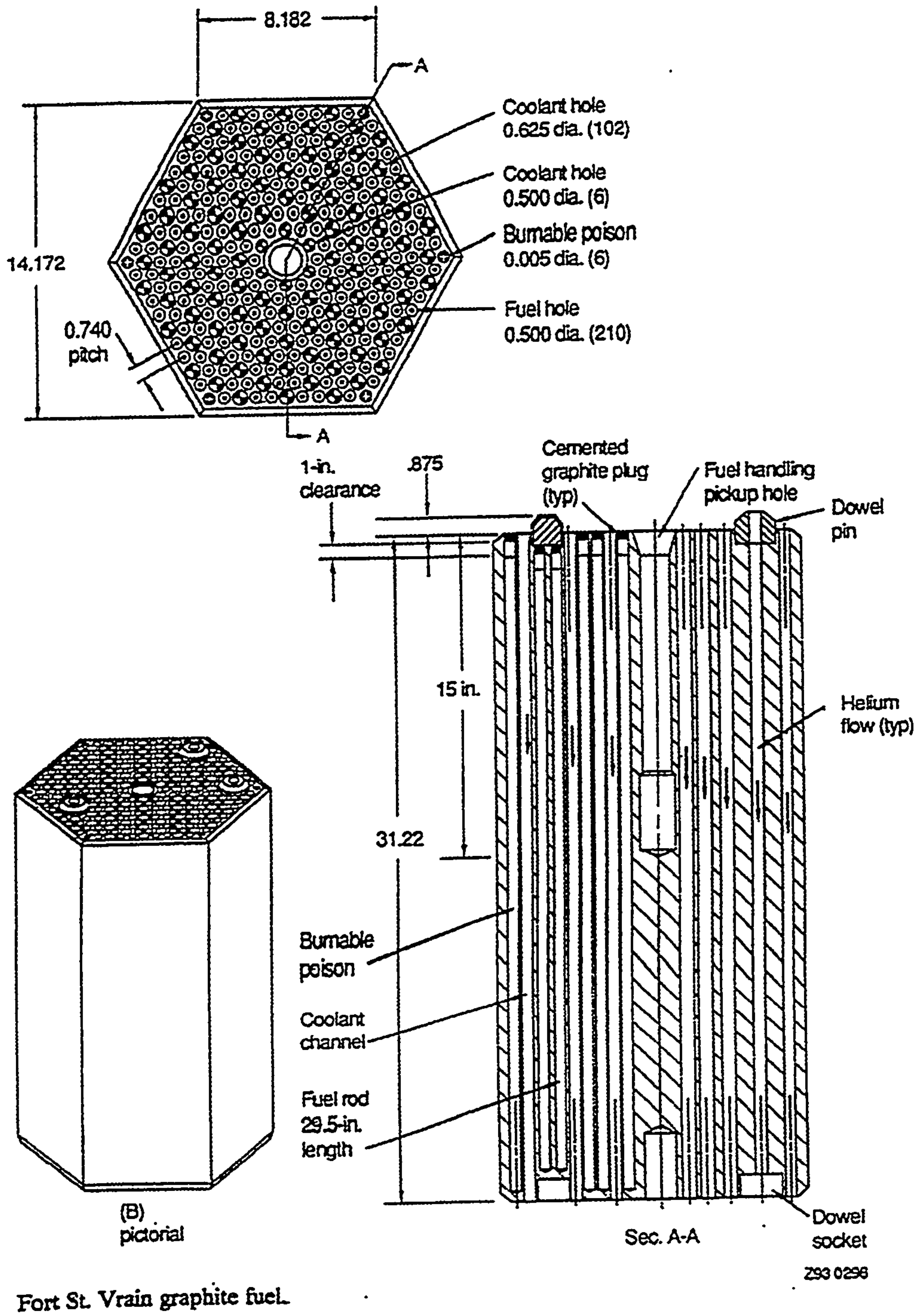


Figure A-11. Fort St. Vrain graphite fuel.

A-1.9 Category 9 U/Th Carbide (Low-Integrity)/Graphite

Typical fuel: Peach Bottom

Fuel Description

The Peachbottom (PB) Core 1 is a graphite based fuel that is made of mixed uranium carbide and thorium carbide particles ranging from 295 to 630 microns in diameter and coated with pyrolytic carbon. However, there is no a SiC protective outer coating on the fuel particles. The particles are formed into annular compacts 2.98 inches (7.6 cm) high with a center hole diameter of 1.75 inches (4.45 cm) and an outside diameter of 2.7 inches (6.86 cm). the compacts are stacked on a 30 inches (76.2 cm) long graphite spine. Three units make up the 90 inches (228.6 cm) long fuel section. An annular low-permeability graphite sleeve is slipped over the fuel compacts [Reference 15]. The failure rate of the particles is estimated to be considerable higher than the FSV fuel particles. Figures A-12 and A-13 show the PB fuel assembly.

Category 9 U/Th Carbide (low-integrity) Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	U/Th carbide	
Matrix dissolution rate	Metal model x 10	Section 3.9
Surface area (m^2/g)	2.2E-02	
Clad failure fraction	Assume 60% failed	
Free radionuclide inventory fraction	10%	
Gap fraction	0.1%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 9 SNF (based on 2,333 MTHM).

	# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 9 Th/U carbide, graphite (PB)					
- repository pkg count	56				
- HLW can count	280				
- SNF pkg count	56				

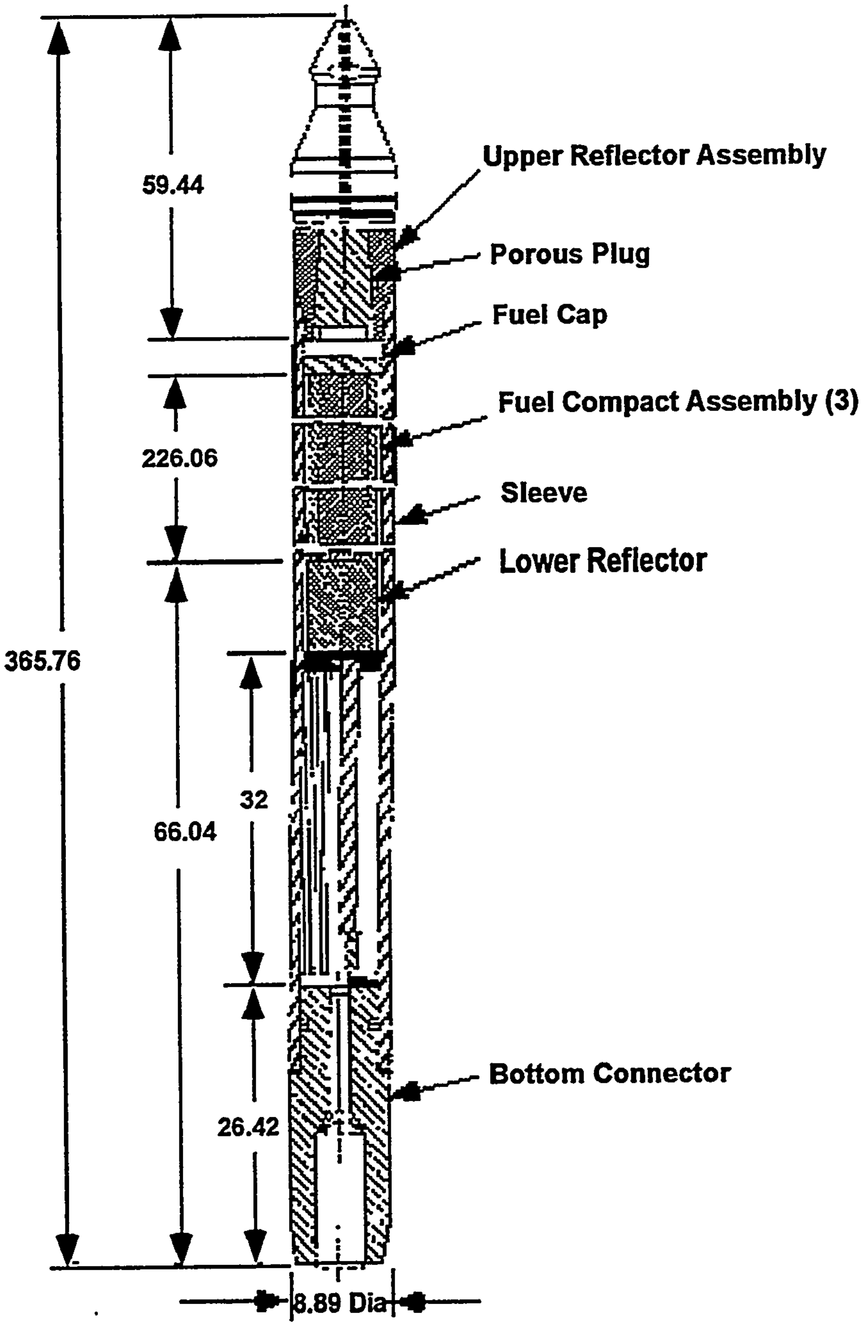


Figure A-12. Peach Bottom Unit 1, Core 1 fuel element.
(Drawing not to scale; dimensions in centimeters)

Fuel Type	Number of Elements Required
1	54
2	564
3	84
4	102
	804

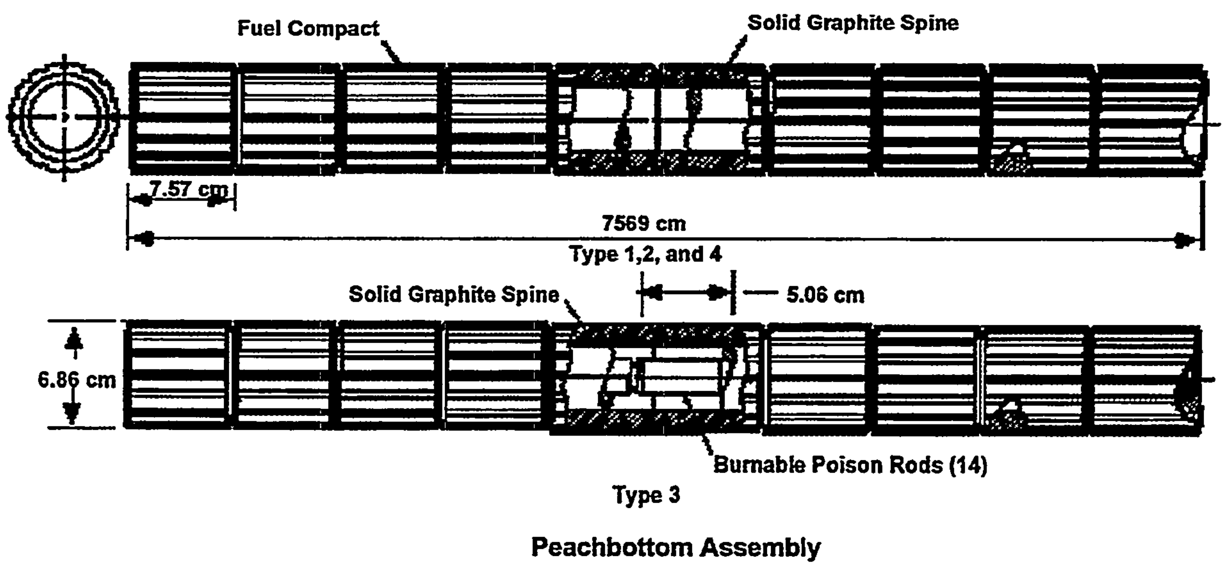


Figure A-13. Peach Bottom Unit 1, Core 1 fuel compacts
(Drawing not to scale)

A-1.10 Category 10 U and U/Pu Carbide/Nongraphite

Typical fuel: SRE (MEU FGE), FFTF Carbide (MEU FGE)

Fuel Description

Category 10 fuels are mixed carbide fuel in a nongraphite matrix. A number of the fuels were test fuel assemblies (TFAs) from the Fast Flux Test Facility (FFTF). FFTF was to provide testing capability for a wide range of development needs of the United States advanced reactor program. The mission of the FFTF included irradiation and evaluation of different types of fuel assemblies and different materials for fuel assembly construction. The purposes of the TFAs vary and a few examples are indicated below. However, in general, the TFAs support the fuel or material requirements for large scale breeder reactors.

As an example, the FFTF-ACN-1 fuel in this category was tested to develop information on helium- and sodium-bonded mixed-carbide fuel pins with full length fuel columns at prototypic fluence and exposure conditions. Additionally, it tests the relative effects of 20% cold worked 316 SS and 25% cold worked D9 cladding on the carbide fuel pins. The assembly contains 18 sodium-bonded and 19 helium-bonded carbide fuel pins, enclosed in a 316 SS inner duct similar to the SRF-3. The outer region contains 90 standard driver fuel pins and is enclosed by a D9 duct [Reference 4]. The test fuel assembly's (TFAs) configuration is similar to the FFTF driver fuels shown on Figures A.15 and A.16 under category 11. The rods containing metallic sodium are not part of this category.

Another fuel assembly, the FFTF-AC-3 was tested in cooperative effort of the United States and Swiss governments and was part of the advanced liquid metal fast breeder reactor fuel program. The test compared performance of 66 pins containing pelletized fuel with that of 25 sphere-pac fuel pins at typical conditions of the breeder reactor. The pins are D9-clad, wire-wrapped, and were housed in a D9 duct. The fuel is mixed plutonium-uranium carbide with plutonium enrichments of 19.1% for the sphere-pac fuel and 19.7% for the pelletized fuel [Reference 4].

The FFTF-FC-1 assembly was tested to establish performance characteristics of a full size carbide fuel assembly. The assembly contains 91 large diameter [0.37 in (0.94 cm)], D9 clad, wire-wrapped, helium-bonded fuel pins. The plutonium enrichment is 21.4 % in uranium carbide, with 6.5 inches (16.5 cm) top and bottom blankets [Reference 4].

Category 10 U-Si Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	U & U/Pu carbide	
Matrix dissolution rate	Metal model x 100	Section 3.10
Surface area (m^2/g)	2.6E-03	
Clad failure fraction	Assume 10% failed	
Free radionuclide inventory fraction	0.00%	
Gap fraction	1-2%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 10 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 10	Pu carbide, SST clad					
	- repository pkg count	3	2			
	- HLW can count	15	10			
	- SNF pkg count	3	2			

A-1.11 Category 11 MOX/(Zr) (SST) (other)

Typical fuel: GE Test ((HEU FGE), FFTF-DFA (HEU FGE, FFTF-TFA-ACO (LEU & MEU FGE)

Fuel Description

MOX fuel is a blend of uranium dioxide and plutonium dioxide within various claddings. The uranium enrichment qualifies as "low" but the plutonium content increases the effective enrichment above 15%. The Fast Flux Test Facility (FFTF) driver fuel assembly (DFA) and test fuel assembly (TFA) contributed to large quantity of the fuel in this category. The standard FFTF-DFA is hexagonally shaped composed of 217 fuel pins. The assembly is 12 feet (3.6 m) long, 4.575 inches (11.6 cm) wide across the hexagon flats, 5.16 inches (13.1 cm) wide across the hexagon points, and weight 381 pounds (173 kg).

The driver fuel pins are 0.23 inch (0.58 cm) in diameters, approximately 93.5 inches (2.37 m) long and have a 36 inches (91 cm) fuel bearing region, which is centered 65.5 inches (1.66 m) from the bottom end of the fuel assembly. Each fuel pin is helically wrapped with a 0.056 inch (0.14 cm) diameter steel wire to provide lateral spacing along its length.

The fuel region contain approximately 150 pressed and sintered, mixed uranium-plutonium oxide pellets and has two 0.4 inch (1 cm) uranium oxide pellets on each end for temperature insulation. The mixed uranium-plutonium oxide is at a nominal theoretical bulk density of 90.4% and the uranium oxide in the insulator pellets is at a nominal theoretical bulk density of 95%. Figures A-14, and A-15 show the configuration of the standard FFTF-DFA fuel assembly [Reference 4].

Category 11 MOX Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	Mixed oxide - U oxide and Pu oxide	
Matrix dissolution rate	Commercial model	Section 3.11
Surface area (m^2/g)	9.5E-04	
Clad failure fraction	Assume 10% failed	
Free radionuclide inventory fraction	0.00%	
Gap fraction	1-2%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 11 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 11	Pu/U oxide, Zr/SST clad					
	- repository pkg count	36	308			
	- HLW can count	180	1,540			
	- SNF pkg count	36	308			

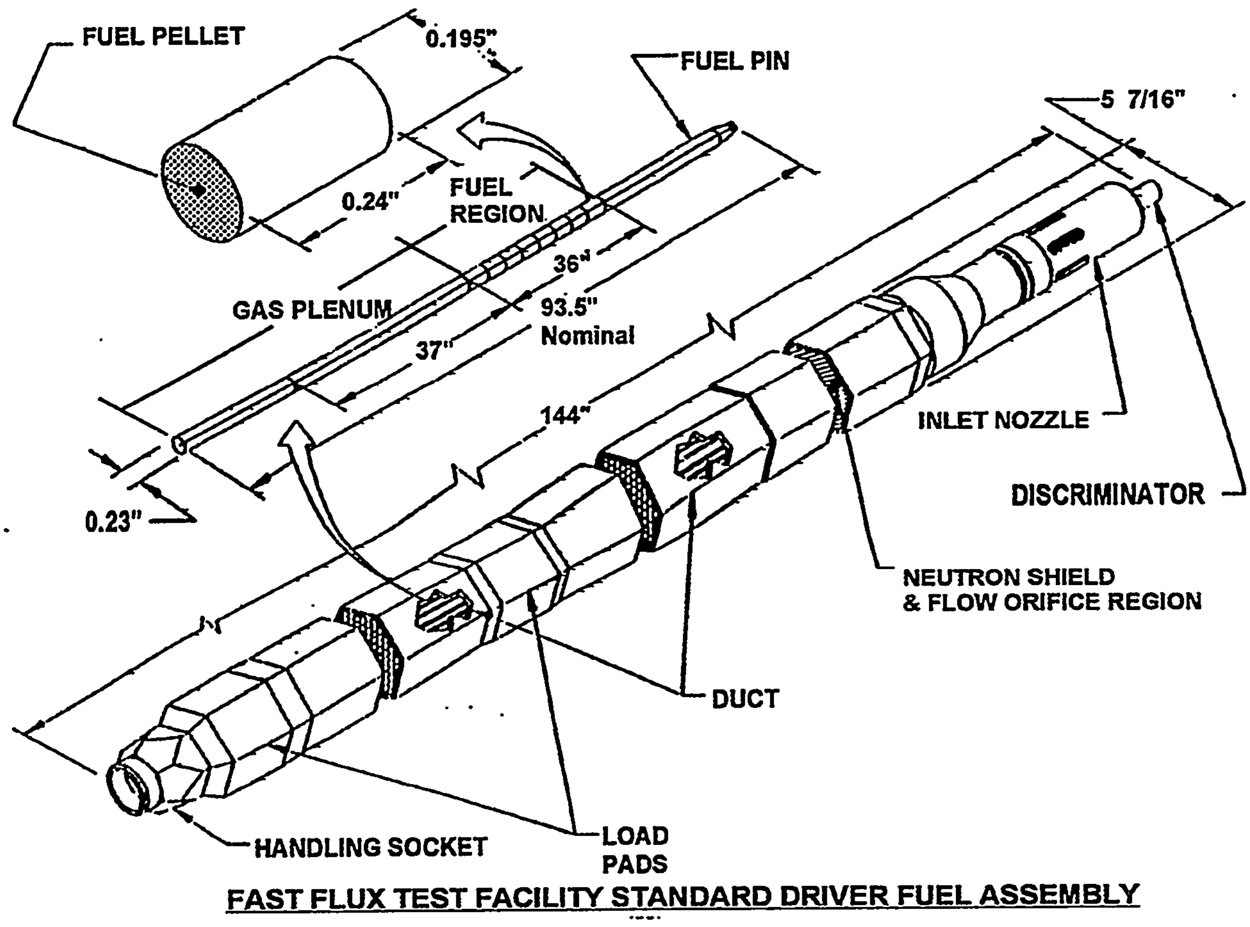


Figure A-14. FFTF standard driver fuel assembly.

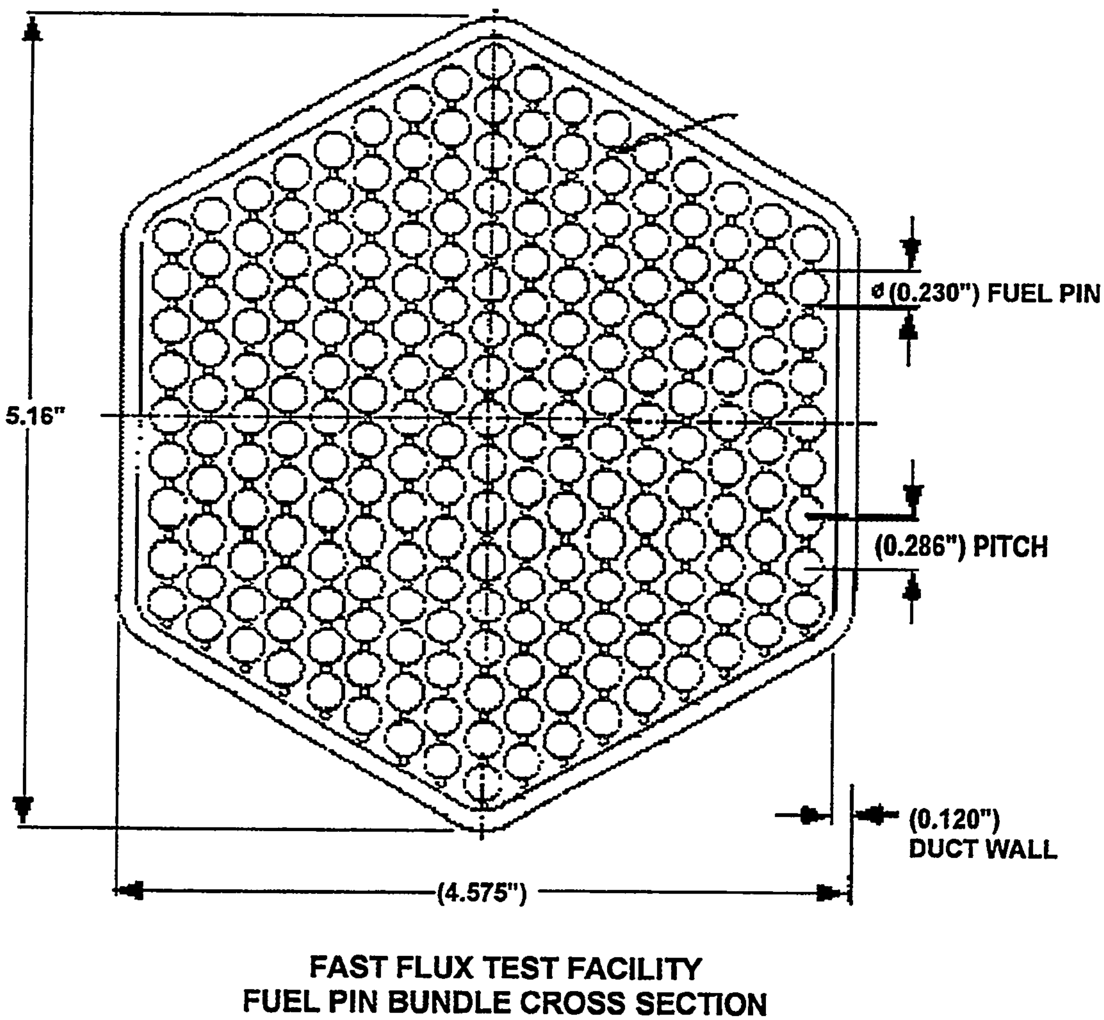


Figure A-15. FFTF pin bundle cross section.

A-1.12 Category 12 U/Th Oxide / (Zr) (SST)

Typical fuel: LWBR (HEU FGE), Dresden (HEU FGE)

Fuel Description

Shippingport Light water Breeder Reactor (LWBR) fuel makes up the major inventory of the fuel in category 12. The Shippingport LWBR was used to demonstrate the production of fissile uranium 233 from thorium in a water-cooled operating reactor. The fuel was made of uranium oxide, enriched up to 98% in uranium 233 (with a very small amount of U-235) mixed with thorium oxide made into cylindrically shaped ceramic pellets. The pellets were loaded into 0.3 in diameter zircaloy-4 tubes whose ends are capped and seal welded. These tubes were made into assemblies. The LWBR has four different types of assemblies: 12 seed assemblies used HEU to produce power, 12 blanket assemblies were used to capture neutrons and convert the thorium to uranium 233, and 9 type IV reflector assemblies and 6 type V reflector assemblies were used to reflect neutrons back into the reactor. The seed assemblies [beginning of life (BOL)] contain 3.67 wt % U-233. The standard blanket (BOL) contain 1.19-1.23 wt % U-233. The power flattening blanket (BOL) contain 2.06-2.08 wt % U-233. Figure A-16 shows the configuration of the Shippingport LWBR assembly.

Category 12 U/Th oxide Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	U/Th oxide	
Matrix dissolution rate	Ceramic model	Section 3.12
Surface area (m^2/g)	5.0E-04	
Clad failure fraction	Assume 10% failed	
Free radionuclide inventory fraction	0.00%	
Gap fraction	1-2%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 12 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 12	Th/U oxide, Zirconium					
	- repository pkg count	14	9	44		
	- HLW can count	70	45	132		
	- SNF canister count	14	9	44		

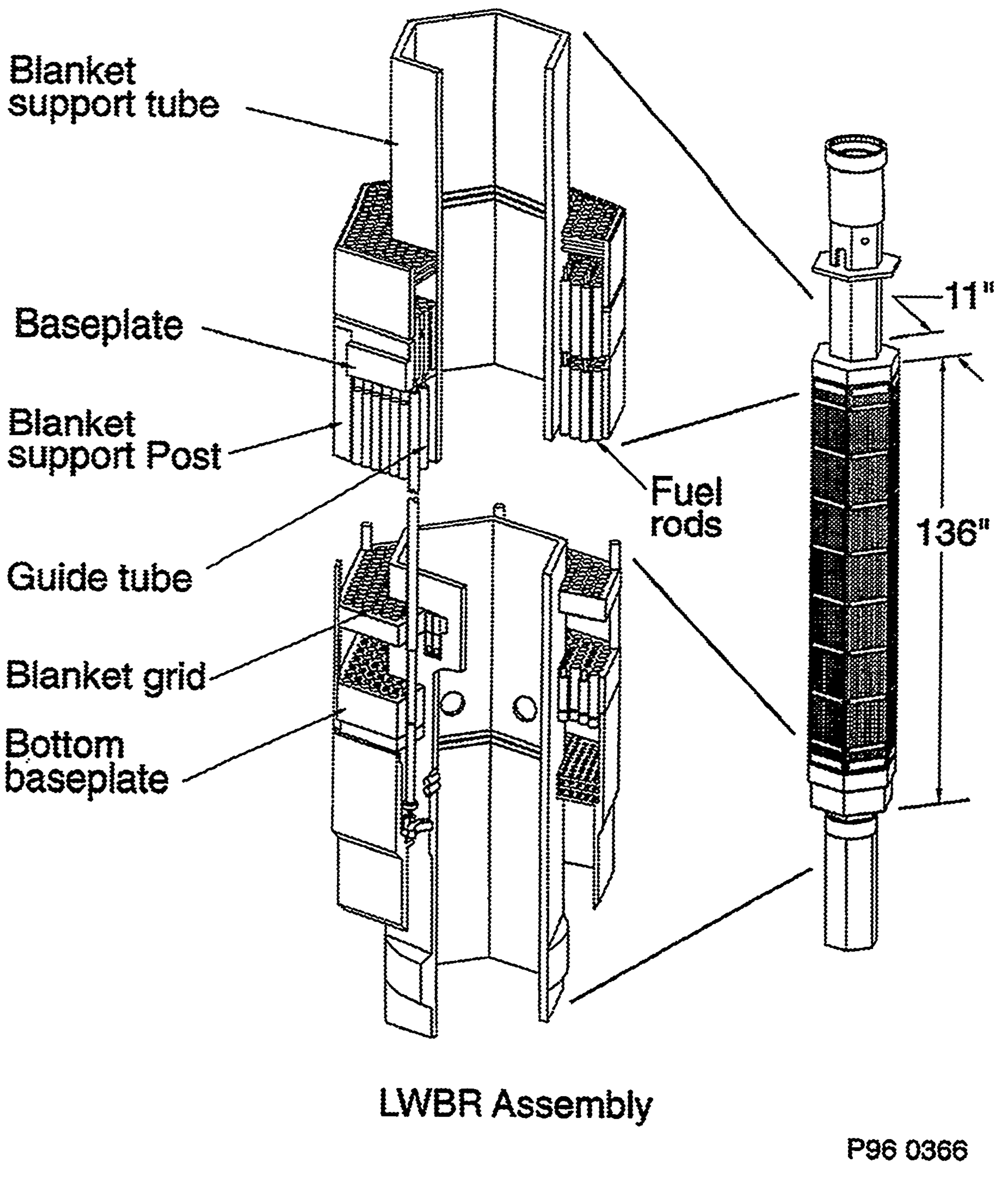


Figure A-16. Shippingport LWBR fuel assembly.

A-1.13 Category 13 U-Zr Hydride / (SST) (Incaloy) (other)

Typical fuel: TRIGA Flip (HEU), TRIGA Std. (MEU), TRIGA Alum (MEU), SNAP (HEU)

Fuel Description

Category 13 contains the fuel with the uranium/zirconium hydride matrix. Fuels from the TRIGA reactors make up the majority of the fuels in this category. The Training, Research, Isotope General Atomics (TRIGA) research reactor have been in use since 1957 throughout the United states and more than 20 countries world-wide. The TRIGA reactors are water-cooled, graphite and water reflected, pool-type research reactors that have steady-state and pulsing capabilities. There are six TRIGA reactors developed by General atomic, each having different experimental facility features to accommodate a user's specific needs.

Like all the fuels in this category, TRIGA fuel elements are made of a uranium-zirconium hydride matrix that provides the reactor with its build in control and inherent safety. They are solid homogeneous all clad with aluminum, stainless steel, or incoloy-800 and varying enrichment and weight percent of U-235 [Reference 30]. Figure A-17 shows a typical configuration of the TRIGA fuel assembly.

Category 13 U-Zr Hydride Fuel Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	U-Zr hydride	
Matrix dissolution rate	Commercial model x 0.1	Section 3.13
Surface area (m^2/g)	1.0E-04	
Clad failure fraction	Assume 10% failed	
Free radionuclide inventory fraction	0.001%	
Gap fraction	0.001%	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 13 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 13	U-Zr hydride, mixed clad					
	- repository pkg count	86	8			
	- HLW can count	430	40			
	- SNF pkg count	86	8			

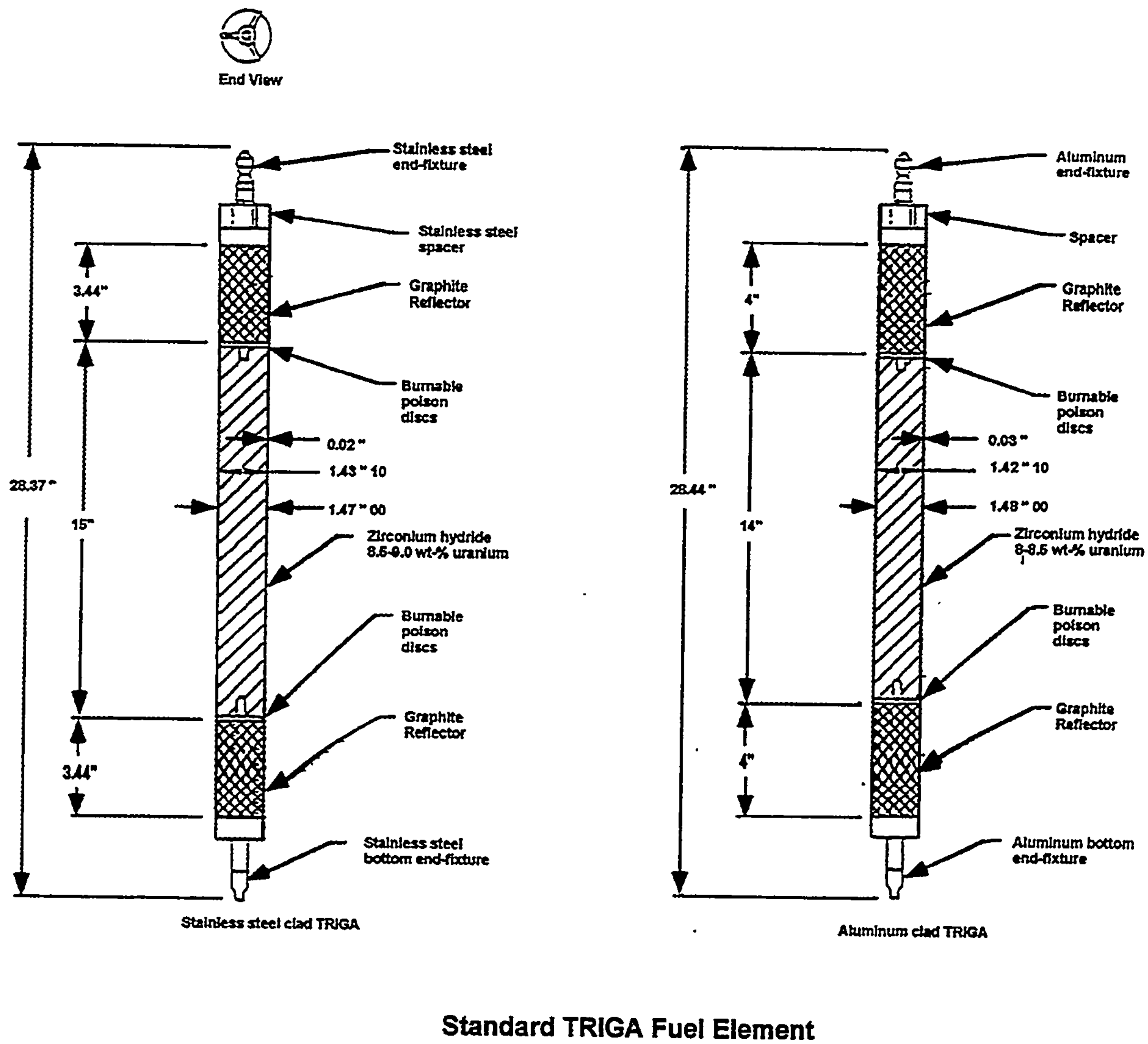


Figure A-17. Standard TRIGA fuel element.

A-1.14 Category 14 Na-Bonded SNF

Typical fuel: Fermi Blankets

Fuel Description

Fermi was a sodium-cooled fast breeder reactor with intermediate sodium loops, sodium-to-water steam generators, and an associated steam-driven turbine-generator. For a detailed description of the fuel and reactor, see Appendix A.1.3 above.

This category consists of the lower and upper axial blanket subassemblies that have been cropped off from the central driver core fuel section and the radial blanket subassemblies. The typical blanket element of the Enrico Fermi Atomic Power Plant (FERMI I Blanket) consists of a U-10wt%-Mo alloy pin clad in a stainless steel tube with elemental sodium in the plenum spaces (of the tube) as a thermal bond. The geometry of the fuel element consists of 140 long, thin (0.443 in. - dia.) cylindrical pins closely packed to form 2 and 1/2 in. square bundles. The fuel meat consists of U-10wt%-Mo alloy composed of depleted uranium (0.36% U-235). They are stored in ICPP-749 and will be treated prior to final disposal.

A-1.15 Category 15 Classified Navy Fuel

Typical fuel: Navy Fuel

Fuel Description

Due to the classified nature of the Navy fuel, it was placed in its own category and all information concerning this category will be provided by the Navy and will not be addressed here.

A-1.16 Category 16 Miscellaneous DOE SNF

Typical fuel: Miscellaneous DOE SNF

Fuel Description

The remainder DOE SNF that does not fit into the above categories are placed in this category. Due to the varying matrices, cladding, and condition of this group of fuel, the plan is to bound the fuel properties in the performance evaluation with the worst performing DOE SNF.

Category 16 Miscellaneous DOE SNF Inventories/Information		
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data - Table D-1	
Composition	Miscellaneous compositions	
Matrix dissolution rate	Metal model	Section 3.16
Surface area (m^2/g)	7.5E-05	
Clad failure fraction	Assume 100% failed	
Free radionuclide inventory fraction	0.1%	
Gap fraction	0	

Configuration and Package Count

The following table shows the disposal configuration, repository package count, and HLW used to co-dispose the category 13 SNF (based on 2,333 MTHM).

		# 5x1 10 ft	# 5x1 15 ft	# 3x1 15 ft	# 0x4 15 ft	PWR21 x 15 ft
Category 16	Miscellaneous DOE SNF, mixed clad					
	- repository pkg count	23	19			
	- HLW can count	115	95			
	- SNF canister count	23	19			

Appendix B

DOE SNF Surface Area Calculation

Title: Fuel Surface Area Calculation
Calc. No: TSPA-VA-SA-002
Page 1 of

By: Dale Cresap
Aug 98
Reviewer:
Review date:

CALCULATION SET
TSPA-VA-SA-002
Rev. 0
QA: NA

The attached calculations determine the surface area of DOE fuels for the purpose of TSPA-VA modeling. They were performed by Dale Cresap on the basis of fuel drawings and design dimensions. The spreadsheet at the end of these calculations is on Dale Cresap's Macintosh under the name VAsurf_area.

Approved by: date:

By: Dale Cresap
Aug 98
Reviewer:
Review date:

Objective and Background:

The TSPA-VA modeling logic requires the fuel meat surface area to perform dissolution rate calculations. This calculation is performed on the basis of a specific surface area in square meters per gram of fuel meat (m2/g), and these calculations all report in these units.

Design Inputs and Their Sources:

Fuel design drawings and documents were used for these calculations.

Assumptions:

The calculational approach was based on surface area calculation by simple geometric shapes based on fuel dimensions. To allow for surface roughness effects, the surface areas were in most cases increased by a factor 5. This should be a conservative value that will exceed the actual fuel area.

This approach assumes that for TSPA-VA modeling all fuels in a category are sufficiently similar to be represented by a single value of surface area. In most cases, a single representative type was chosen and used to provide a value for the entire category.

Fuel Types:

The following list shows the TSPA fuel categories which were assigned by fuel composition.

By: Dale Cresap
Aug 98
Reviewer:
Review date:

1	U-metal
2	U-Zr
3	U-Mo
4	U-oxide intact
5	U-oxide failed
6	U-Al
7	U-Si
8	U-Th-C intact
9	U-Th-C fail
10	U-Pu-C non-g
11	MOX
12	U-Th oxide
13	U-Zr-Hx
16	Misc.

Category 1 U metal
Representative Type N reactor
26" long double annulus ID .48in, OD 1.28, ID 1.7, OD 2.42 (largest Mk4)
Area = 26 pi (.48+1.28+1.7+2.42) = 480 in2
U loading 51.7 lbs
480 in2/51.7lb x 2.205lb/1000g x (2.54/100 m/in)^2 =1.32 e-5m2/g
use 1.4 to include ends
1.4e-5m2/g x 5 (roughness)= 7e-5 m2/g

Category 2 U-Zr

Representative Type HWCTR
annulus: element dimensions 5.8 cm x 290 cm
area= pi x 2 x 5.8 x 290 cm2 x (1m2/1e4 cm2) = 1 m2
1 m2 / EOL U 777g = 1.3e-3 m2/g x 5 (roughness) = 6.5e-3 m2/g

Category 3 U-Mo
Representative Type Fermi
fuel pin .4 cm x 84 cm, 134 gU/pin
Area = 84 x pi x 0.4 x 1e-4m2/cm2 /134 g U = 8e-5 m2/g
times 5 (roughness) = 4e-4 m2/g

Category 4 U oxide
Representative Type commercial PWR
9.5e-4 m2/g by reference tspa va

Category 5 disrupted U oxide
Representative Type TMI commercial fuel
100x intact commercial fuel = 9.5e-2 m2/g

Category 6 Al
Representative Type ATR
126 cm fuel section based on ATR design drawings
19 plates from 5.8 to 10.8 cm wide
mean width(5.8+10.8)/2 = 8.3 cm
area/element= 19 plates x 2 sides x 8.3 cm wide x 126 cm = 4 m2
3.02 kg fuel matrix 4m2/3020g = .0013 m2/g
1.3e-3 m2/g x 5 (roughness factor) = 6.5e-3 m2/g
The actual value used for this category is 7.4 e-3 which is the weighted average of values provided by SRS for this category. The above calculation gives a reasonably close value for a check.

Category 7 U-Si
Representative Type FRR-MTR from RERTR
SRS provided value 1.4e-2 m2/g

By: Dale Cresap
Aug 98
Reviewer:
Review date:

Category 8 High Integrity U-Th C
Representative Type FSVR
2.2e-2 m2/g value provided by Bob Kirkham

Category 9 Low Integrity U-Th C
Representative Type Peachbottom 1
2.2e-2 m2/g by similarity to category 8

Category 10 Non-graphite UC
Representative Type FFTF carbide
particle size 200um density of UC2 is 11.3 g/cm3
Area of sphere = pi d2=pi(2e-4)^2= 1.3 m2
mass = volume x spg= pi x d^3/6 x spg=pi/6(2e-4)^3 x 1e6cm3/m3 = 5e-5 g
specific area is 2.6e-3 m2/g
no roughness appropriate for these small spheres

Category 11 MOX
Representative Type FFTF
considered similar to commercial fuel 9.5e-4m2/g

Category 12 U-Th oxide
Representative Type Shippingport LWBR
fuel assembly consists of 30 plates, 19cm x 248 cm
mass of fuel meat 225 kg U; 30 kg oxide
area: 30 x2(sides)x19x248/255kg x1e-3g/kg x1e-4m2/cm2=1e-4m2/g
roughness factor of 5 gives 5e-4 m2/g

Category 13 U-Zr-H
Representative Type Triga
1.435 in diameter 15 inch long rod with 195 g U rod is 8.5% U
area: 1.435 x pi x 15 = 67.6 in2
mass 195g/0.085 = 2294 g
spec area 67.6/2294 x (2.54/100)^2 m2/in2 = 2e-5 m2/g
with roughness factor of 5 = 1e-4 m2/g

Category 16 Miscellaneous Fuel
Representative Type n reactor
Therefore use same value as category 1
7e-5 m2/g

	fuel group	SA calc by shape, m2/g	roughness factor	surface area m2/g	old area bas	notes	MTHM	# pkgs	SA basis correct	VAarea/pkg sq meter
1	U-metal	1.4E-05	5	7.0E-05	MTHM	N-reactor	2122	107	1	1.4E+03
2	U-Zr	1.3E-03	5	6.5E-03	like cat 4	HWCTR	0.04	8	1.12	3.6E+01
3	U-Mo	8.0E-05	5	4.0E-04	U-Mo	FERMI	3.76	70	1	2.1E+01
4	U-oxide intact			9.5E-04	UO2	RW commercial fuel value	98.65	214	1.134	5.0E+02
5	U-oxide failed			9.5E-02	UO2	100 x cat 4	87	686	1.134	1.4E+04
6	U-Al	(DF 6.5e-3)		7.4E-03	U-Al matrix	wt avg of SRS cat recrd vals	8.74	706	2.61	1.2E+02
7	U-Si			1.4E-02	Like cat 6	wt avg of SRS cat recrd vals	11.55	215	3.2	8.5E+01
8	U-Th-C intact	2.2E-02	1	2.2E-02	MTHM	FSVR value from RJ Kirkham	24.67	503	1	1.1E+03
9	U-Th-C fail	2.2E-02	1	2.2E-02	like UO2	similar to 8	1.66	60	1	6.1E+02
10	U-Pu-C non-g	2.6E-03	1	2.6E-03	like UO2	FFTF	0.15	5	1.1	8.6E+01
11	MOX			9.5E-04	like UO2	FFTF MOX like cat 4	12.3	367	1.134	3.6E+01
12	U-Th oxide	1.0E-04	5	5.0E-04	MTHM	LWBR	49.63	71	1	3.5E+02
13	U-Zr-Hx	2.0E-05	5	1.0E-04	MTHM	TRIGA	2.02	100	1	2.0E+00
16	Misc.		5	7.0E-05		like cat 1	10.7	44	1	1.7E+01

Notes

1. TSPA model bases area on fuel meat.
2. The basic area calculations are based on the simple geometric forms of the fuel.
3. A surface roughness factor of 5 is applied as appropriate
4. MTHM values were corrected as required by molecular weight ratios, eg, UO2/U = (238+32)/238 = 1.134
5. The high correction factor for category 6 is based on an ATR element with UAlx-Al matrix.
6. Note that these calculations are for fuel only and do not include HLW.
7. Note that these are mean values for categories and do not reflect variation within a category.

DF val	notes	OLD SA/pkg	SA ratio	fuel vol/pkg
		sq meter	VA/Old	m3
335.	4 x MCO	264.41	5.25	1.0E+00
75.		0.26	139.45	7.8E-03
30.		5.40	3.98	5.0E-02
500.	not 21 pwr	28.54	17.40	4.8E-02
1200.	???	861.33	15.86	1.3E-02
0.	.17 low?	40.58	2.91	8.2E-03
420.		171.31	0.50	1.5E-02
320 to 1200		153.90	7.01	6.0E-03
336.		0.95	643.06	3.3E-03
83.		1.88	45.65	2.9E-03
500.	no 21 pwr	2.00	18.03	3.3E-03
25.		79.17	4.41	8.0E-02
9.		37.25	0.05	4.0E-02
335.	not like 1			2.5E-02

Appendix C

DOE SNF Volume Calculation

Title: Fuel Meat Volume Calculation	By: Dale Cresap
Calc. No: TSPA-VA-VOL-001	Date: 2-17-98
Page 1 of 10	Reviewer: WHL00
	Review date: 3/3/98

CALCULATION SET
TSPA-VA-VOL-001
Revision: 0
QA: N/A

Calculation Purpose

The attached calculations determine the volume of fuel meat per disposal package. The volume per disposal package information will be used in the total system performance assessment viability assessment (TSPA-VA) for radionuclide release from the DOE SNF. The calculation was preformed by Dale Cresap on February 17, 1998. The inputs used in this calculation do not meet the RW/0333P requirements and thus this calculation is not under the control of the QA program. The spreadsheet at the end of these calculations is on Dale Cresap's Macintosh under the name VAfuelvolume and in a floppy disk with this calculation

Approved by: Date:

1.0 Objective and Background

The TSPA-VA modeling logic is being revised by the Office of Civilian Radioactive Waste Management (RW), and the Department of Energy (DOE) Spent Nuclear Fuel (SNF) Programs are providing new inputs to support the increased detail in the model. In particular, the package water retention logic is being revised and will require the volume of fuel meat per disposal package. Several methods were considered for development of this data, and the method selected was based on mean disposal package loadings of heavy metal and the densities of fuel meat constituents appropriate for the composition of each category.

2.0 Design Inputs and Their Sources

- DOE SNF Data Base Version 3.0.0
- MTHM and the number of packages per fuel category came from the spread sheet RW_input198.xls
- Material density came from Denny Fillmore - See attachment A.

3.0 Assumptions

- For each fuel category, the volume of fuel per package will be based on the average MTHM of fuel per package.

- A 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc. This non-ideal density correction factor was not added to the U-Alx and U-Si fuel categories based on information provided by Allen Brewer of SRS. In addition, based on measurement done at Hanford, this non-ideal density correction for the FFTF fuel category was increased to 1.15 (This information was provided by Alan Carlson of Hanford).

4.0 Computer and Software

In performing this calculation, the following computer hardware and software were used. The EXCEL program was used to generate the summary table at the end of the calculation.

Computer Hardware: Macintosh Model Quadra 950
Computer software: EXCEL version 5

5.0 Calculation and Results

5.1 Fuel Types

The following list shows the TSPA fuel categories which were assigned by fuel composition. In the table, the MTHM and fuel package count came from the spread sheet RW_input198.xls that summarizes MTHM and package in each of the DOE SNF category based on inputs from the DOE SNF Data Base and the EIS ROD responsible sites (Hanford, INEEL, and SRS).

Table 1 Fuel Category, MTHM, and Package Count

Fuel Category	Fuel Matrix	Typical Fuel in the category	MTHM/Packages/Comment (based on 2,500 MTHM)
1	U-Metal	N-Reactor Fuel	2122.26 MTHM/ 107 pkg
2	U-Zr	Heavy Water Component Test Reactor Fuel	0.04 MTHM/ 8 pkg
3	U-Mo	FERMI (Enrico Fermi Reactor) Fuel	3.77 MTHM/ 70 pkg
4	U-Oxide Intact	Shippingport PWR Fuel Commercial PWR Fuel	98.68 MTHM/ 214 pkg
5	U-Oxide Failed/Declad	Three Mile Island (TMI)	87.02 MTHM/ 686 pkg
6	U-Al Or U-Alx	Advanced Test Reactor (ATR) Fuel	8.74 MTHM/ 706 pkg
7	U-Si	Foreign Research Reactor (FFR) Fuel	11.55 MTHM/ 215 pkg
8	U/Th Carbide Hi-Integrity	Fort Saint Vrain (FSV) Fuel	24.67 MTHM/ 503 pkg
9	U/Th Carbide Low-Integrity	Peach Bottom Fuel	1.66 MTHM/ 60 pkg
10	U or U/Pu Carbide Non-Graphite	Fast Flux Test Facility (FFTF) Carbide Fuel	0.15 MTHM/ 5 pkg
11	MOX	Fast Flux Test Facility (FFTF) Oxide Fuel	12.32 MTHM/ 367 pkg
12	U/Th Oxide	Shippingport LWBR Fuel	49.63 MTHM/ 71 pkg
13	U-Zr-Hx	Training Research Isotopes-General Atomic (TRIGA) Fuel	2.03 MTHM/ 100 pkg
14	Na-Bonded	FERMI I Blanket	Will be treated. Not part of TSPA-VA analyses
15	Classified-Navy	Navy	Info by Navy
16	Misc. SNF	Misc. Fuel	10.73 MTHM/ 44 pkg

By: Dale Cresap
Date: 2-17-98
Reviewer:
Review date: 3-3-98

5.2 Calculation Approach and Example Calculations

Calculation Approach

The calculational approach was based on the metric ton of heavy metal (MTHM) inventory from the DOE SNF Data Base and the number of packages estimated in each fuel category. One advantage of this method is that it depends only on MTHM and package count as measured fuel data. All other inputs are established physical constants such as molecular weight and density. The theoretical density is used in calculation. In practice, many fuels are fabricated as compacted particles of oxides or various mixtures, and these are pressed to a specified fraction of theoretical density, typically 98%. Frequently fuels also swell during irradiation, further reducing the density. These effects could increase the volume of the fuel by approximately 10%. For the purpose of this calculation, density reduction due to these processes of 10% will be included here (except no correction for the U-Alx, and U-Si, fuel caltegories and a 15% correction for the FFTF category as indicated under assumptions). The reason is that the TSPA-VA model calculates the water volume available for radionuclide removal based on the volume of the SNF matrix. The higher the volume of SNF, the larger the volume of water available for radionuclide removal from the SNF (See Attachment C, e-mail note from Vinod Vallikat of RW).

As necessary, the mean mass per package will be adjusted based on the form of fuel in the category to account for other materials such as oxides, carbides, etc. Since the TSPA-VA group the fuels by their chemical form, the fuel categories facilitated these calculations. A single chemical form will be assigned per category. This approach assumes that all fuels in a category are sufficiently similar to be represented by a single value.

Following are examples of volume calculation for the various DOE SNF form. These example calculations show how the volume of each category will be calculated. The complete DOE SNF volume calculations are performed using an EXCEL spread sheet based on the method discussed here and is included as Table 2 Computer Calculations.

U metal Fuel

The volume of any material may be determined if the mass and density of the material are known through equation 1, where ρ is the density of the material.

$$volume = [mass][\frac{1}{\rho}] \qquad (1)$$

From the DOE SNF Data Base, the quantity of heavy metal in category 1 is 2,122.26 metric tons. Heavy metal includes all the uranium, plutonium, and thorium that are in the category. After irradiation, the U metal SNF will also contain small quantities of plutonium and thorium. But for the purpose of this volume calculation, the heavy metal will be assumed to be 100% uranium. Based on this assumption, a density of 19 g/cc (or 19 metric ton/m^3) could be used to represent the U metal fuel.

In support of the TSPA-VA effort, the DOE sites have indicated that all the U metal fuels (category 1) could be placed into 107 packages for repository disposal (See Table 1 above). Thus, the average quantity of MTHM per package may be calculated using equation 2.

$$\frac{MTHM}{package} = \frac{[Total\ MTHM\ (category 1)]}{[Total\ No.\ packages\ (category 1)]} \qquad (2)$$

Since it was assumed that uranium makes up 100% of the heavy metal. The MT U in equation 3 would be the same as the total MTHM shown in equation 2 above.

$$\frac{MT\ U}{package} = \frac{[Total\ MTHM\ (category 1)]}{[Total\ No.\ packages\ (category 1)]} \qquad (3)$$

Based on equations 1, 3, and the density of U metal, equation 4 could be used to solve for the volume of U metal fuel per disposal package. As noted earlier, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.

$$\frac{volume}{pkg} = [\frac{MT\ U}{pkg}][\frac{1}{\rho(U)}] = [\frac{2122.26}{107}][\frac{MT\ U}{pkg}][\frac{1}{19}][\frac{m^3}{MT\ U}][1.1] = 1.1\times10^0\ \frac{m^3}{pkg} \qquad (4)$$

Uranium alloy Fuels

In a similar manner, the volume of any two or more materials may be expressed by equation 5 if the fraction of each material and its density are known.

$$volume = [mass][\frac{Fraction\ (A)}{\rho(A)}+\frac{Fraction\ (B)}{\rho(B)}+...+\frac{Fraction\ (n)}{\rho(n)}] \qquad (5)$$

By: Dale Cresap
Date: 2-17-98
Reviewer: [illegible]
Review date: 3-3-98

Since the alloy material is not a heavy metal, it will not be reported in the DOE SNF Data Base as total MTHM. One way of determining the volume of the alloy material is to first determine the mass fraction of the alloy material to the mass fraction of the uranium. The volume of the alloy may then be determined using equation 6 (based on equations 3 and 5 above).

$$volume = [MT\ U][\frac{(Alloy\ mass\ Fraction)}{(U\ mass\ Fraction)}\frac{1}{\rho(Alloy)}] \qquad (6)$$

In the case of U-Zr alloy fuel, Attachment A reported that uranium makes up 9.3% of the total mass and Zr makes up the 90.7% of the total mass. Using a Zr density of 6.49 g/cc (or 6.49 MT/m^3), 0.04 MTHM and 8 packages for U-Zr fuel (See category 2 Table 1 above), and the assumption that U makes up 100% of all the heavy metal, the volume per pacakge of the U-Zr fuel could be represented by equation 7. As with the metal fuel, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.

$$\frac{volume}{pkg} = [\frac{MT\ U}{pkg}][\frac{1}{\rho(U)}+\frac{(Alloy\ mass\ Fraction)}{(U\ mass\ Fraction)}\frac{1}{\rho(Alloy)}] \qquad (7)$$

or

$$\frac{volume}{pkg} = [\frac{0.04}{8}][\frac{MT\ U}{pkg}][\frac{1}{19.05}+\frac{(.907)}{(.093)}\frac{1}{6.49}][\frac{m^3}{MT\ U}][1.1] = 8.6\times10^{-3}\ \frac{m^3}{pkg}$$

This approach will also be used for category 3, with suitable alloy mass fractions and densities for the U-Mo fuels. See Table 2 Computer Calculations for the volume of the U-Mo fuel.

Fuel Meat Consisting of Uranium Oxide Compound

For oxide fuels, the mass of the fuel meat is greater than the heavy metal content because of the presence of oxygen. Therefore, to calculate the mass of the fuel meat from the heavy metal mass, the heavy metal basis must be adjusted by the molecular weight and atomic weight ratio indicated by equation 8.

$$oxide\ mass = [MTHM][\frac{MW\ (UO_2)}{atomic\ wt\ (U)}] \qquad (8)$$

or

$$oxide\ mass = [MTHM][\frac{(238\ +\ 32)}{238}]$$

or

$$oxide\ mass = [MTHM][1.134]$$

This factor is applied to categories 4 and 5 (UO_2 fuels) to arrive at the correct fuel meat mass. Then the appropriate oxide density is used to determine the volume of the fuel. Using the DOE SNF Data Base of 98.68 MTHM and 214 disposal packages for category 4 (See category 4 Table 1 above), and a uranium oxide density of 10.96 (See Attachment A), the volume of category 4 fuel may be determined using equation 9 below. As noted earlier, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.

$$\frac{volume}{pkg} = [\frac{98.68}{214}][\frac{MTHM}{pkg}][\frac{1.134}{10.96}][\frac{MT(UO_2)}{MTHM}][\frac{m^3}{MT(UO_2)}][1.1] = 5.3\times10^{-2}\ \frac{m^3}{pkg} \qquad (9)$$

For the volume of category 5 fuel, see Table 2 Computer Calculation at the end of this calculation.

Fuel Meat Consisting of Several Heavy Metal Compounds

In cases where more than one element contributes to MTHM, the relative mass fractions of each of the heavy metal compounds are required. In all cases, the mass fractions refer to the elements that comprise MTHM. For instance, U mass fraction = U mass/ MTHM. In category 8, for example, the fuel is 94.8% Th and the Th mass fraction is therefore 0.948 of the heavy metal inventory.

Similar to the oxide fuels, the mass of the fuel meat (in this case the U carbide fuel) is greater than the heavy metal content because of the presence of carbon. Therefore, to calculate the mass of the fuel meat from the heavy metal mass, the heavy metal basis must be adjusted. This adjustment is derived from the molecular weight ratio of the compound/metal, and converted to volume using the density of the compound similar to equation 9 above.

Equation 10 below shows this generic relationship for a fuel with two heavy metal compounds.

$$volume = [M][(\frac{X_1}{\rho_1})(\frac{MW_1}{AW_1})+(\frac{X_2}{\rho_2})(\frac{MW_2}{AW_2})] \qquad (10)$$

Where:

M	=	Metric tons of heavy metal
MW_1	=	molecular weight of compound 1
MW_2	=	molecular weight of compound 2
AW_1	=	atomic weight of heavy metal 1
AW_2	=	atomic weight of heavy metal 2
X_1	=	mass fraction of heavy metal 1
X_2	=	mass fraction of heavy metal 2
ρ_1	=	density of compound 1
ρ_2	=	density of compound 2

Using category 8 as an example, equation 11 below calculates the volume of the Fort Saint Vrain fuel that contain both uranium and thorium in the form of carbide. The DOE SNF Data Base shows that there are 24.67 MTHM and 503 disposal packages for category 8 (See category 8 Table 1 above). Of the total heavy metal, uranium makes up 5.2% and thorium makes up 94.8% of the mass (See Attachment A). The uranium and thorium carbides have a density of 11.28 and 8.96 g/cc (or 11.28 and 8.96 MT/m^3) respectively. To correct for the additional mass due to the carbon, the MW/AW for U carbide and Th carbide of 1.1 is included (i.e., UC_2/U = (238+24/238) = 1.1 and ThC_2/Th = (232+24/232) = 1.1). And finally, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.

$$\frac{volume}{pkg} = [\frac{24.67}{503}][\frac{MTHM}{pkg}][\frac{(0.052)(1.1)}{11.28}+\frac{(0.948)(1.1)}{8.96}][\frac{m^3}{MT}][1.1] = 6.6\times10^{-3}\ \frac{m^3}{pkg} \qquad (11)$$

A similar approach is used for categories 9, 10, 11, and 12. For the volume of category 9, 10, 11, and 12 fuels, see Table 2 Computer Calculation at the end of this calculation.

Calculations by Volumetric Methods

In some cases, the fuel composition was uncertain, and it was more straightforward to calculate a volume of fuel meat based on fuel drawings or specifications and attribute it to the Uranium content of the fuel. This was true of categories 6 (UAlx+Al), 7 (U-Si), and 13 (U-Zr-Hx).

By: Dale Cresap
Date: 2-17-98
Reviewer: [illegible]
Review date: 3/3/98

Attachment B contains the volumetric calculations for the three fuel types.

For the UAlx+Al fuel, the ATR fuel specifications were used to determine both the volume of fuel meat and MTHM per element. Attachment B calculation shows that for the ATR fuel, each element contains 0.000763 m^3 of fuel meat and 0.001156 MTHM. The DOE SNF Data Base shows that there are 8.74 MTHM and 706 disposal packages for category 6 (See category 6 Table 1 above). The volume per disposal package could be calculated using equation 12. As noted earlier, no density correction was added to the equation.

$$\frac{volume}{pkg} = [\frac{8.74}{706}][\frac{MTHM}{pkg}][\frac{0.000763}{0.001156}][\frac{m3/element}{MTHM/element}] = 8.2\times10^{-3}\ \frac{m^3}{pkg} \qquad (12)$$

For the USi fuel, the fuel information was based on a letter from James Snelgrove of Argonne National Laboratory (See Attachment B). The letter shows that for the USi fuel, the density of the fuel meat is 3.5 g(U)/cc (or 3.5 MT/m^3). The DOE SNF Data Base shows that there are 11.55 MTHM and 215 disposal packages for category 7 (See category 7 Table 1 above). The volume per disposal package could be calculated using equation 13. As noted earlier, no density correction was added to the equation.

$$\frac{volume}{pkg} = [\frac{11.55}{215}][\frac{MTHM}{pkg}][\frac{1}{3.5}][\frac{m3}{MTHM}] = 1.5\times10^{-2}\ \frac{m^3}{pkg} \qquad (13)$$

For the U-Zr-Hx fuel, the TRIGA fuel specifications were used to determine both the volume of fuel meat and MTHM per element. Attachment B calculation shows that for the TRIGA fuel, each element contains 388 cm^3 of fuel meat and 195 g U (or 0.5 MTHM/m^3). The DOE SNF Data Base shows that there are 2.02 MTHM and 100 disposal packages for category 13 (See category 13 Table 1 above). The volume per disposal package could be calculated using equation 14. As noted earlier, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.

$$\frac{volume}{pkg} = [\frac{2.02}{100}][\frac{MTHM}{pkg}][\frac{1}{0.5026}][\frac{m3}{MTHM}][1.1] = 4.4\times10^{-2}\ \frac{m^3}{pkg} \qquad (14)$$

Misc. Fuel was pro-rated on the same basis as categories 4 and 5

The value arrived at independently by RW for use with commercial oxide fuels of 1.1126 m^3/21 PWR element package equates to 0.115 m^3/MT. This agrees closely with the value of 0.113 m^3/MT by this method for the same type of fuel meat in category 4.

5.3 Computer calculations

The Table 2 Computer Calculation spreadsheet was made with Microsoft excel 5 on a Quadra Macintosh. Only simple arithmetic operations were used to calculate the results.

6.0 Attachment A

The following sheets provide densities and fuel composition values provided by Denny Fillmore. The physical constants were taken from the CRC Handbook of Physic and Chemistry Ed 57. The fuel composition data were taken from the task team report which was in turn taken from fuel receipt criteria and from The Research, Training, Test and Production Reactor Directory, 3rd edition, 1988.

7.0 Attachment B

The following sheets provide the hand calculations which determine the specific fuel meat volume values for categories 6, 7, and 13. The data for category 6 are taken from ATR fuel specifications. For category 7 the data are taken from a letter reference which is enclosed. TRIGA fuel specifications are used for category 13.

Review: [illegible]
Date: 3-3-98

Computer Calculations using EXCEL version 5

The following Table uses EXCEL to calculates the fuel volume of the various fuel categories based on the methods described above.

Review: 1/1/00
Date: 3-3-98

Table 2 Computer Calculation

TSPA Cat	Fuel Type	MTHM	Cat pkg total	Non-Ideal Density Correction	Fuel Vol/pkg, m3/pkg	molecular wt (compound)/ atomic wt (heavy metal) uranium	molecular wt (compound)/ atomic wt (heavy metal) thorium	molecular wt (compound)/ atomic wt (heavy metal) plutonium	Notes
1	U-Metal	2122.26	107	1.1	1.1E+00	NA	NA	NA	U density = 19
2	U-Zr	0.04	8	1.1	8.6E-03	NA	NA	NA	90.7% Zr, density Zr = 6.49
3	U-Mo	3.77	70	1.1	5.5E-02	NA	NA	NA	90% Mo, density = 10.2
4	U-Oxide Intact	98.68	214	1.1	5.3E-02	1.1	NA	NA	Oxide mass basis correction, density = 10.96
5	U-Oxide Failed/Declad	87.02	686	1.1	1.4E-02	1.1	NA	NA	Like category 4
6	U-Al or U-Alx	8.74	706	NA	8.2E-03	NA	NA	NA	Based on calc. of volume of UAlx+Al per element
7	U-Si	11.55	215	NA	1.5E-02	NA	NA	NA	Based on published density and U loading
8	U/Th Carbide Hi-Integrity	24.67	503	1.1	6.6E-03	1.1	1.1	NA	94.8% Th, carbide mass correction
9	U/Th Carbide Low-Integrity	1.66	60	1.1	3.6E-03	1.1	1.1	NA	82.6% Th, carbide mass correction
10	U/Th Carbide Non-graphite	0.15	5	1.1	3.3E-03	1.1	NA	NA	Carbide mass correction, density=11.28
11	MOX	12.32	367	1.15	4.0E-03	1.1	NA	1.1	Oxide mass correction, 15% Pu, density=11.46
12	U/Th Oxide	49.63	71	1.1	8.8E-02	1.1	1.1	NA	Oxide mass correction, 98.7% Th
13	U-Zr-Hx	2.03	100	1.1	4.4E-02	NA	NA	NA	Based on calc U-Zr-Hx vol and mass per element
15	Navy	63.00	NA	NA	NA	NA	NA	NA	NA
16	Misc.	10.73	44	1.1	2.8E-02	1.1	NA	NA	Same basis as category 4 and 5
otal		2496.25	3156						

Attachment A

Densities and Fuel composition Values provided by Denny Fillmore

Volume estimates for DOE SNF by group

Group 1- Uranium Metal
multiply the uranium metal mass by the density of uranium metal

$\rho = 19.05$ g/cc

Group 2- uranium zirconium alloy
determine the uranium-zirconium mass ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

9.3% U 90.7% Zr $\rho_U = 19.05$ $\rho_{Zr} = 6.49$

$\cancel{\rho = 7.66}$ $\frac{1.0}{\rho} = \frac{.093}{19.05} + \frac{.907}{6.49}$ $\rho = 6.91$ g/cc

Group 3 - Uranium moly alloy
determine the uranium-moly ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

10% U 90% Mo $\rho_U = 19.05$ $\rho_{Mo} = 10.2$

$\rho = 11.1$ $\frac{1.0}{\rho} = \frac{.1}{19.05} + \frac{.9}{10.2}$ $\rho = 11.2$

Group 4 - Intact uranium oxide
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

1% Pu 99% U

$\rho = 10.96$

Group 5 - failed uranium oxide
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

$\rho = 10.96$

ATR?

Group 6 - uranium alloy or uranium oxide or uranium aluminide in an aluminum matrix
determine the uranium-aluminum ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

$\rho_{UAl_x} = 6.3$ $\rho_{Al} = 2.7$ $\rho_{U_3O_8} = 8.3$ $U = 71\%$ $Al = 29\%$

use $\rho_U = 7.3$

$\rho = 5.97$

Group 7 - uranium silicide
determine the uranium-aluminum and silicon ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

7.5% U_3Si_2 of fuel matrix $\rho_{U_3Si_2} = 12.2$ $\rho_{Al} = 2.7$

93% U_3Si_2 is U

given by Fuel Fab $\rho = 3.5$

Group 8 - high integrity uranium thorium carbide
determine the uranium-thorium ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

mass split 5.2 U : 94.8 th $\rho_{UC_2} = 11.8$ $\rho_{ThC} = 8.96$

$\rho = 9.11$

Group 9 - low integrity uranium thorium carbide
determine the uranium-thorium ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

17.4% U 82.6% th $\rho_{UC_2} = 11.8$ $\rho_{ThC} = 8.96$

$\rho = 9.45$

Group 10 - Non graphite uranium and plutonium carbide
determine the uranium-plutonium ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

$\rho_{UC_2} = 11.28$

Group 11 - MOX
determine the uranium-plutonium ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

15% Pu 85% U $\rho_{PuO_2} = 11.46$ $\rho_{UO_2} = 10.96$

$\rho = 11.0$

Group 12 - Uranium thorium oxide
determine the uranium- thorium ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

1.3% U 98.7% Th $\rho_{UO_2} = 10.96$ $\rho_{ThO_2} = 9.86$

$\rho = 9.87$

Group 13 - Uranium Zirconium Hydride ~ just 1% H prime U-Zr
determine the uranium-zirconium ratio
determine a density of the fuel matrix using a weighted density based on the ratio
multiply the uranium metal mass by the density of the uranium metal

8.54% U 91.46% Zr $\rho_U = 19.05$ $\rho_{Zr} = 6.49$

$\rho = 7.56$

Group 16 - Miscellaneous

? PUNT

Attachment B

Hand Calculation for Categories 6, 7, and 13

Attachment B

Reviewed by:
date:

Category 6
The specific fuel volume for this category is based on ATR fuel which was selected as a typical representative for which data were available. The following values were taken from ATR fuel specifications, Fuel receipt criteria 10/92, letter JAH-195-85, letter HJR-11-78.

ATR fuel contains 3.02 kg of matrix per assembly which has 1.156 kg U
The density of UAlx is 6.3 and of Al is 2.7
UAlx is 69% U
If UAlx *.69 = 1.156 kg, then UAlx = 1.67 kg
UAlx + Al = 3.02 kg, then Al = 3.02-1.675 = 1.345 kg

Volume of fuel meat per assy = Al + UAlx = 1.345/2.7 + 1.675/6.3 = .763 L

= .000763 m3/.001156 MTU

Category 7
The specific volume of U silicide fuels was based on the attached ANL reference letter from James Snelgrove dated 2/21/96
Uranium loading 3.5 g/cm3
fuel meat density 11.28 g/cm3
Fraction of Uranium in meat 3.5/11.28 = 0.31

volume of meat/pkg = MTHM/pkg/11.28/.31

Category 13
This category consists of TRIGA fuels. Data were taken from URANIUM-ZIRCONIUM HYDRIDE FUELS FOR TRIGA REACTORS", and "CHARACTERIZATION OF TRIGA FUEL". The cover sheets of these reports are attached.
Fuel is 8.5% U fuel meat section is 15 inches long 1.435 inch diameter and has a central Zr rod .225 inch diam. Uranium loading is 195 grams.

Volume = $\pi/4*(D^2-d^2)*h$ = $\pi/4*(1.435^2-.225^2)*15*2.54^3$ cm^3/in^3
=388 cm^3/.000195 MTU

ARGONNE NATIONAL LABORATORY
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February 21, 1996

Dr. Ratib A Karam
Neely Nuclear Research Center
Georgia Institute of Technology
900 Atlantic Drive
Atlanta, Georgia 30332-0425

Dear Ratib:

As we discussed during our telephone conversation earlier this morning, I do not think that the amount of void in the fuel meat is important to the performance of the U_3Si_2 fuel plates which will be used in your LEU fuel elements. In fact, in 1987 I recommended that the void content not be specified for the LEU standard U_3Si_2 fuel plate for use in university research reactors (see enclosed letter to Jerry Reed at INEL). As indicated in that letter, small deviations in the void content of a fuel plate will result in insignificant differences in plate thickness change owing to fuel swelling. I am also enclosing some pages from a recently published reference work discussing the basis for my conclusion. I will give some numbers for your fuel below.

Uranium density in the standard plate (ρ_U) = ~3.4-3.5 gU/cm^3--assume 3.5

Volume fraction of fuel in the meat (V_f) = 3.5/11.28 = 0.31

Volume fraction of void in the meat (V_v) = 0.035 (Using Eq. 2-1 of reference)

Fission density in fuel particle for 50% burnup = 2.5×10^{21} cm^{-3}

Fuel particle swelling for 50% burnup = 9% (Using Fig. 2-34 of reference)

Meat swelling = (Fuel particle swelling) x (Fuel volume fraction) - Void volume fraction
= 0.028 - V_v (Using simplified interpretation of Eq. 2-22 of reference)

Therefore, you will experience a negligible amount of swelling at 50% burnup. According to Fig. 2-5 of the reference, a decrease of 1% in the void content would actually enhance the thermal conductivity of the meat by about 5 W m^{-1} K^{-1}. As you can see, the standard fuel plate is expected to have a void content near the lower limit of the specification. Since there is some variation in void content owing to variations inherent in the fabrication process, it is not surprising that a number of plates would be out of spec.

If you have any questions, please call.

Sincerely,

Jim Snelgrove

James L. Snelgrove
Coordinator, Engineering Applications
RERTR Program

Enclosures (2)

Operated by The University of Ch

TRIGA® Reactors

UZR-28

URANIUM-ZIRCONIUM HYDRIDE FUELS

FOR

TRIGA® REACTORS

A Report Prepared for

LOCKHEED MARTIN IDAHO TECHNOLOGIES COMPANY

under

CONTRACT NO. C96-180484

June, 1997

GENERAL ATOMICS

GA Technologies

ORNL/Sub/86-22047/3
GA-C18542

CHARACTERIZATION OF TRIGA FUEL

by
N. TOMSIO

Report prepared by
GA Technologies Inc.
P.O. Box 85608
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GA PROJECT 3442
OCTOBER 1986

Attachment C

E-mail note from Vinod Vallikat of RW concerning the calculation of the water volume using the fuel rod volume.

Vinod_Vallikat@notes.ymp.gov on 02/23/98 10:21:14 AM

To: Henry H Loo/HENRY/LMITCO/INEEL/US
cc: Denzel L Fillmore/DFF/LMITCO/INEEL/US, Dale A Cresap/DCRESAP/LMITCO/INEEL/US
Subject: Re: SNF volume

Hi Henry,

In response to Denny Filmore's question, let me once again try to describe the method we are currently using to calculate the volume of water in the waste form mixing cell. But before that, let me explain how the dissolution rate and surface area information is used. As in previous model, we do still use the dissolution rate and specific surface area to calculate the matrix degradation rate.

i.e., dissolution rate (g/m2.yr) x specific surface area (m2/g) = matrix degradation rate (1/yr).

Now, coming to the water volume calculation. As you would have seen in the slide, I sent you,

Vrind = Vrod x k x t , where k is the matrix degradation rate in (1/yr) and t is the time in years.

Vrind < = Vrod

Vwater to calculate radionuclide concentration = Vrind x porosity x saturation.

So, this basically means the higher the volume of rod, the higher the volume of water.

Hope this helps clear some of your questions.

-Vinod

To: Vinod Vallikat
cc: DFF@inel.gov, DCRESAP@inel.gov
From: HENRY@inel.gov
Date: 02/23/98 08:51:28 AM MST
Subject: SNF volume

Vinod, There is still some confusion as to which way is more conservative. That is, a higher fuel volume or a lower fuel volume from the radionuclide release stand point (See Denny's question below). Could you conisder Denny's question below and maybe we could talk about the fuel volume question some more this afternoon?
Thanks,
Henry
--------------------- Forwarded by Henry H Loo/HENRY/LMITCO/INEEL/US on 02/23/98 08:43 AM ---------------------------

DFF@inel.gov on 02/23/98 07:51:12 AM

To: Henry H Loo/HENRY/LMITCO/INEEL/US
cc: Dale A Cresap/DCRESAP/LMITCO/INEEL/US
Subject: SNF volume

I have been calculating the volume of SNF fuel matrix in the waste package and need some further guidance.
I will estimate the volume based on the mass and density of the fuel matrix. There are some uncertainties in the values and I think that I need to give conservative answers (conservative is the case that give more release of radionuclides from the waste package. In this case is more volume of fuel matrix or less volume of fuel matrix conservative???
Because the model has changed, and I do not completely understand the new model I am having trouble reasoning it out. In the old model larger surface area gave a larger release because the rate depended on the surface area. Using that reasoning then larger volume would be conservative because it would have a larger surface area. However, if that is true why did we go away from surface area and consider volume. I can also see where smaller volume of fuel matrix might be considered conservative because smaller volume of fuel matrix give more volume of water in the flooded waste package to dissolve those species that are solubility limited.
Could you please check with Vinod and ask him to help us understand which approach is conservative and why? Let me know what he says so I can finish the estimates.

Appendix D

Radionuclide Inventory Summary of DOE SNF and HLW

Table D-1. DOE-owned SNF Radionuclide Inventory at the year 2030

COMPLEX CUM	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category
2,333 MTHM SNF, no HLW	1	1	2	2	3	3
	MTHM	packages	MTHM	packages	MTHM	packages
Hanford	1962.45	95				
INEEL	1.57	6	0.04	8	3.51	66
SRS	15.85	0				
Total	1979.88	101	0.04	8	3.51	66
	total curies	Ci/pkg	total curies	Ci/pkg	total curies	Ci/pkg
Isotopes						
AC227	9.3903E-03	9.2973E-05	3.5603E-08	4.4504E-09	4.5333E-04	6.8686E-06
AM241	4.8103E+05	4.7627E+03	2.8789E+00	3.5987E-01	2.9730E-03	4.5045E-05
AM242M	2.7934E+01	2.7657E-01	8.7230E-03	1.0904E-03	0.0000E+00	0.0000E+00
AM243	1.1808E+02	1.1691E+00	2.8415E-02	3.5519E-03	4.7793E-11	7.2413E-13
C14	6.1831E+02	6.1219E+00	6.1023E-05	7.6279E-06	1.4338E-01	2.1724E-03
CL36	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	3.6547E-04	5.5375E-06
CM244	3.5866E+03	3.5511E+01	1.1718E+00	1.4647E-01	4.4630E-12	6.7621E-14
CM245	1.5194E+00	1.5044E-02	5.9900E-05	7.4875E-06	1.2159E-17	1.8423E-19
CM246	2.2475E-01	2.2253E-03	4.1181E-06	5.1477E-07	4.0061E-21	6.0699E-23
CS135	7.4650E+01	7.3911E-01	5.7654E-02	7.2067E-03	2.9519E-01	4.4726E-03
CS137	8.3715E+06	8.2886E+04	5.5033E+04	6.8792E+03	0.0000E+00	0.0000E+00
I129	6.8555E+00	6.7876E-02	1.2205E-02	1.5256E-03	7.5554E-03	1.1448E-04
NB93M	3.4000E+02	3.3663E+00	6.4767E-02	8.0959E-03	3.0116E-01	4.5631E-03
NB94	2.3990E-03	2.3753E-05	1.2242E-05	1.5303E-06	5.4821E-02	8.3062E-04
NI 59	3.4735E+01	3.4392E-01	0.0000E+00	0.0000E+00	4.4981E-01	6.8153E-03
NI 63	3.2773E+03	3.2449E+01	0.0000E+00	0.0000E+00	9.3477E+00	1.4163E-01
NP237	7.2342E+01	7.1626E-01	1.4975E-01	1.8719E-02	2.1753E-02	3.2959E-04
PA231	2.4438E-02	2.4196E-04	1.7109E-06	2.1386E-07	1.2792E-03	1.9381E-05
PB210	1.3646E-07	1.3510E-09	6.4393E-12	8.0491E-13	9.3828E-09	1.4216E-10
PD107	1.4032E+01	1.3893E-01	8.5358E-03	1.0670E-03	8.3637E-03	1.2672E-04
PU238	1.0581E+05	1.0476E+03	3.2271E+02	4.0339E+01	1.0859E+00	1.6453E-02
PU239	2.1749E+05	2.1534E+03	8.0116E+00	1.0015E+00	1.2862E+02	1.9488E+00
PU240	1.2840E+05	1.2713E+03	4.5674E+00	5.7092E-01	4.0061E-01	6.0699E-03
PU241	1.6211E+06	1.6051E+04	1.5724E+03	1.9655E+02	3.9359E-02	5.9634E-04
PU242	6.7957E+01	6.7284E-01	6.8511E-03	8.5638E-04	2.4951E-09	3.7804E-11
RA226	1.9550E-03	1.9356E-05	1.6398E-11	2.0497E-12	4.2170E-08	6.3894E-10
RA228	1.5886E-07	1.5729E-09	9.2845E-13	1.1606E-13	2.6321E-07	3.9880E-09
SE79	1.0919E+02	1.0811E+00	2.2013E-01	2.7517E-02	1.0894E-01	1.6506E-03
SM151	1.3800E+05	1.3663E+03	1.9842E+02	2.4802E+01	4.5333E+02	6.8686E+00
SN126	1.4715E+02	1.4570E+00	1.9692E-01	2.4615E-02	2.4740E-01	3.7484E-03
SR90	6.6835E+06	6.6173E+04	5.2787E+04	6.5984E+03	0.0000E+00	0.0000E+00
TC99	3.2624E+03	3.2301E+01	7.4126E+00	9.2658E-01	2.9519E+00	4.4726E-02
TH229	1.6557E-05	1.6393E-07	7.9368E-10	9.9210E-11	1.5392E-07	2.3321E-09
TH230	1.8455E-03	1.8272E-05	6.4018E-08	8.0023E-09	7.4852E-06	1.1341E-07
TH232	2.1597E-07	2.1383E-09	1.5162E-11	1.8953E-12	2.7727E-07	4.2010E-09
U233	1.0444E-02	1.0341E-04	6.8136E-06	8.5170E-07	6.4309E-05	9.7438E-07
U234	8.2914E+02	8.2093E+00	6.4018E-03	8.0023E-04	3.2330E-02	4.8985E-04
U235	3.5538E+01	0.0000E+00	6.1023E-02	7.6279E-03	2.0558E+00	3.1148E-02
U236	1.3862E+02	1.3724E+00	2.5121E-01	3.1401E-02	1.1280E+02	1.7092E+00
U238	6.5690E+02	6.5040E+00	1.1231E-03	1.4039E-04	9.3477E-01	1.4163E-02
ZR93	4.2693E+02	4.2270E+00	1.1306E+00	1.4133E-01	4.3576E-01	6.6024E-03

Table D-1. (continued).

	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category
2,333 MTHM SNF, no HLW	4	4	5	5	6	6
	MTHM	packages	MTHM	packages	MTHM	packages
Hanford	17.06	19	0.15	1		
INEEL	75.00	183	78.03	380		
SRS			3.01	261	8.15	659
Total	92.06	202	81.18	642	8.15	659
	total curies	Ci/pkg	total curies	Ci/pkg	total curies	Ci/pkg
Isotopes						
AC227	3.6165E-02	1.7904E-04	1.0570E-02	1.6465E-05	2.4022E-04	3.6452E-07
AM241	3.3074E+05	1.6373E+03	5.6705E+03	8.8326E+00	2.3118E+03	3.5081E+00
AM242M	5.5682E+02	2.7565E+00	1.0965E+01	1.7080E-02	1.5904E+00	2.4133E-03
AM243	1.7726E+03	8.7754E+00	1.7371E+01	2.7057E-02	1.7161E+00	2.6041E-03
C14	2.1813E+01	1.0798E-01	8.3775E-01	1.3049E-03	6.9531E-04	1.0551E-06
CL36	7.3802E-02	3.6535E-04	1.3193E-02	2.0549E-05	0.0000E+00	0.0000E+00
CM244	7.4871E+04	3.7065E+02	7.2145E+02	1.1237E+00	1.4683E+01	2.2281E-02
CM245	3.0509E+01	1.5103E-01	2.9282E-01	4.5610E-04	1.2449E-03	1.8891E-06
CM246	5.1757E+00	2.5622E-02	4.9649E-02	7.7335E-05	5.9997E-05	9.1042E-08
CS135	3.7658E+01	1.8643E-01	1.6923E+01	2.6360E-02	1.9723E+01	2.9928E-02
CS137	6.1554E+06	3.0472E+04	2.7919E+06	4.3488E+03	2.6847E+06	4.0739E+03
I129	3.7870E+00	1.8748E-02	5.8318E-01	9.0838E-04	1.2785E+00	1.9400E-03
NB93M	1.4340E+02	7.0992E-01	8.5684E+00	1.3346E-02	1.0819E+01	1.6417E-02
NB94	1.5675E+00	7.7601E-03	3.2573E-02	5.0737E-05	2.1655E-03	3.2861E-06
NI 59	2.0025E+01	9.9136E-02	1.8142E-01	2.8259E-04	0.0000E+00	0.0000E+00
NI 63	6.3188E+04	3.1281E+02	1.9730E+01	3.0732E-02	1.5922E-19	2.4161E-22
NP237	3.8126E+01	1.8874E-01	2.3738E+00	3.6976E-03	8.1581E+00	1.2379E-02
PA231	6.3506E-02	3.1439E-04	1.9780E-02	3.0811E-05	2.9088E-03	4.4140E-06
PB210	1.9125E-05	9.4676E-08	4.0652E-06	6.3320E-09	1.4540E-08	2.2064E-11
PD107	1.0256E+01	5.0772E-02	6.4096E-01	9.9838E-04	6.4771E-01	9.8287E-04
PU238	2.4513E+05	1.2135E+03	4.4698E+03	6.9623E+00	1.2058E+04	1.8298E+01
PU239	3.2852E+04	1.6263E+02	9.7319E+03	1.5159E+01	1.6715E+03	2.5365E+00
PU240	5.0960E+04	2.5228E+02	3.4232E+03	5.3321E+00	8.6832E+02	1.3176E+00
PU241	2.9837E+06	1.4771E+04	2.2711E+05	3.5375E+02	4.3840E+04	6.6525E+01
PU242	1.9533E+02	9.6700E-01	2.3523E+00	3.6640E-03	6.8895E-01	1.0454E-03
RA226	5.5556E-05	2.7503E-07	1.5413E-05	2.4007E-08	2.6621E-07	4.0397E-10
RA228	2.8681E-02	1.4199E-04	8.5944E-03	1.3387E-05	5.1139E-10	7.7601E-13
SE79	4.0622E+01	2.0110E-01	1.2457E+01	1.9404E-02	3.4695E+01	5.2647E-02
SM151	1.1228E+05	5.5583E+02	1.9103E+04	2.9756E+01	2.9061E+04	4.4099E+01
SN126	5.1246E+01	2.5369E-01	9.0356E+00	1.4074E-02	1.1609E+01	1.7616E-02
SR90	4.5345E+06	2.2448E+04	2.5053E+06	3.9023E+03	2.5842E+06	3.9215E+03
TC99	1.4037E+03	6.9491E+00	3.2257E+02	5.0245E-01	7.1981E+02	1.0923E+00
TH229	8.5249E-02	4.2203E-04	2.4689E-02	3.8457E-05	1.5327E-06	2.3258E-09
TH230	7.8919E-03	3.9069E-05	2.2652E-03	3.5284E-06	1.2597E-04	1.9115E-07
TH232	3.0212E-02	1.4956E-04	9.0632E-03	1.4117E-05	1.3146E-08	1.9948E-11
U233	3.3056E+01	1.6364E-01	9.4539E+00	1.4726E-02	1.8473E-03	2.8032E-06
U234	3.5459E+01	1.7554E-01	8.1979E+00	1.2769E-02	1.5250E+00	2.3141E-03
U235	3.6199E+00	1.7920E-02	9.3301E+00	1.4533E-02	1.2560E+01	1.9059E-02
U236	2.9336E+01	1.4523E-01	9.0123E+00	1.4038E-02	2.5620E+01	3.8878E-02
U238	2.7604E+01	1.3665E-01	2.6256E+01	4.0898E-02	6.2237E-01	9.4441E-04
ZR93	1.9506E+02	9.6565E-01	3.8990E+01	6.0732E-02	7.0831E+01	1.0748E-01

Table D-1. (continued).

	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category
2,333 MTHM SNF, no HLW	7	7	8	8	9	9
	MTHM	packages	MTHM	packages	MTHM	packages
Hanford						
INEEL			23.01	470	1.55	56
SRS	10.78	201				
Total	10.78	201	23.01	470	1.55	56
	total curies	Ci/pkg	total curies	Ci/pkg	total curies	Ci/pkg
Isotopes						
AC227	4.7185E-05	2.3475E-07	0.0000E+00	0.0000E+00	1.5820E-01	2.8250E-03
AM241	7.6220E+03	3.7920E+01	1.7420E+03	3.7065E+00	1.4859E+02	2.6533E+00
AM242M	2.7496E+00	1.3679E-02	4.7175E-01	1.0037E-03	7.6154E-02	1.3599E-03
AM243	7.4983E+00	3.7305E-02	1.2450E+01	2.6489E-02	7.3827E-02	1.3183E-03
C14	3.6656E-03	1.8237E-05	1.0102E+02	2.1495E-01	2.0938E+00	3.7390E-02
CL36	0.0000E+00	0.0000E+00	1.2680E+00	2.6978E-03	5.9403E-02	1.0608E-03
CM244	6.2202E+01	3.0946E-01	3.9121E+02	8.3236E-01	1.9387E+00	3.4620E-02
CM245	4.1572E-03	2.0683E-05	6.5355E-02	1.3905E-04	2.2334E-04	3.9883E-06
CM246	3.0676E-04	1.5261E-06	3.2447E-02	6.9037E-05	7.2742E-06	1.2990E-07
CS135	3.9278E+00	1.9541E-02	8.1004E+00	1.7235E-02	1.4765E+00	2.6367E-02
CS137	1.7009E+06	8.4624E+03	1.1253E+06	2.3943E+03	1.0454E+05	1.8667E+03
I129	7.9505E-01	3.9555E-03	9.6882E-01	2.0613E-03	4.0946E-02	7.3118E-04
NB93M	6.5719E+00	3.2696E-02	4.3263E+00	9.2049E-03	2.5902E+00	4.6253E-02
NB94	1.5530E-03	7.7264E-06	6.2133E-02	1.3220E-04	2.7453E-02	4.9023E-04
NI 59	0.0000E+00	0.0000E+00	8.0773E+00	1.7186E-02	8.1737E-02	1.4596E-03
NI 63	0.0000E+00	0.0000E+00	1.9399E+02	4.1275E-01	8.4219E+00	1.5039E-01
NP237	3.7425E+00	1.8619E-02	7.6171E+00	1.6207E-02	4.1256E-01	7.3672E-03
PA231	5.9294E-04	2.9500E-06	8.7217E+00	1.8557E-02	2.6987E-01	4.8192E-03
PB210	2.4270E-09	1.2075E-11	2.1448E-03	4.5633E-06	1.4564E-05	2.6007E-07
PD107	7.3733E-01	3.6683E-03	4.0732E-01	8.6664E-04	2.6677E-02	4.7638E-04
PU238	5.8767E+03	2.9237E+01	3.7970E+04	8.0788E+01	1.0283E+03	1.8363E+01
PU239	4.6624E+03	2.3196E+01	1.0540E+02	2.2425E-01	2.2955E+01	4.0991E-01
PU240	3.3342E+03	1.6588E+01	1.7743E+02	3.7750E-01	1.7992E+01	3.2128E-01
PU241	1.4617E+05	7.2720E+02	0.0000E+00	0.0000E+00	1.6751E+03	2.9912E+01
PU242	3.2553E+00	1.6195E-02	0.0000E+00	0.0000E+00	2.3730E-02	4.2375E-04
RA226	4.7417E-08	2.3590E-10	2.2782E-03	4.8473E-06	5.2269E-05	9.3337E-07
RA228	2.3828E-10	1.1855E-12	3.0606E+00	6.5120E-03	1.3339E-01	2.3819E-03
SE79	2.1699E+01	1.0795E-01	1.3784E+01	2.9329E-02	7.5378E-01	1.3460E-02
SM151	7.1994E+03	3.5818E+01	2.1125E+04	4.4948E+01	1.4254E+03	2.5453E+01
SN126	8.5209E+00	4.2393E-02	6.4895E+00	1.3807E-02	6.9640E-01	1.2436E-02
SR90	1.5885E+06	7.9030E+03	1.0563E+06	2.2474E+03	9.8954E+04	1.7670E+03
TC99	4.5268E+02	2.2521E+00	3.4749E+02	7.3933E-01	2.2645E+01	4.0437E-01
TH229	2.5302E-07	1.2588E-09	1.3002E+01	2.7664E-02	3.8310E-01	6.8410E-03
TH230	2.4907E-05	1.2392E-07	8.9058E-01	1.8948E-03	7.3827E-03	1.3183E-04
TH232	6.2263E-09	3.0977E-11	2.3703E+00	5.0431E-03	1.4052E-01	2.5093E-03
U233	3.6792E-04	1.8304E-06	3.2217E+03	6.8547E+00	1.4657E+02	2.6173E+00
U234	3.6714E-01	1.8266E-03	2.6234E+02	5.5817E-01	2.4351E+01	4.3483E-01
U235	2.6725E+00	1.3296E-02	1.0125E+00	2.1543E-03	3.7844E-01	6.7579E-03
U236	1.2392E+01	6.1652E-02	9.9644E+00	2.1201E-02	9.5541E-01	1.7061E-02
U238	3.2699E+00	1.6268E-02	2.5083E-02	5.3369E-05	3.9550E-03	7.0626E-05
ZR93	4.3768E+01	2.1775E-01	5.0397E+02	1.0723E+00	3.5828E+00	6.3979E-02

Table D-1. (continued).

	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category	TSPA Category
2,333 MTHM SNF, no HLW	10	10	11	11	12	12
	MTHM	packages	MTHM	packages	MTHM	packages
Hanford	0.07	2	9.54	303		
INEEL	0.07	3	1.95	41	46.30	67
SRS						
Total	0.14	5	11.49	344	46.30	67
	total curies	Ci/pkg	total curies	Ci/pkg	total curies	Ci/pkg
Isotopes						
AC227	4.1494E-08	8.2988E-09	5.1412E-07	1.4945E-09	2.8429E+01	4.2431E-01
AM241	1.0951E+03	2.1902E+02	1.4758E+05	4.2901E+02	5.5099E+01	8.2237E-01
AM242M	1.9101E+00	3.8202E-01	2.5600E+02	7.4418E-01	5.7877E-01	8.6383E-03
AM243	3.1305E-02	6.2609E-03	6.1634E+01	1.7917E-01	1.1020E-01	1.6447E-03
C14	6.7410E-05	1.3482E-05	3.3962E-01	9.8727E-04	4.4264E+01	6.6066E-01
CL36	0.0000E+00	0.0000E+00	3.5869E-03	1.0427E-05	9.8622E-01	1.4720E-02
CM244	1.2920E+00	2.5840E-01	2.5894E+03	7.5273E+00	1.0603E+01	1.5825E-01
CM245	6.5878E-05	1.3176E-05	1.0602E+00	3.0819E-03	2.1947E-03	3.2756E-05
CM246	4.5348E-06	9.0697E-07	1.7996E-01	5.2314E-04	1.4492E-04	2.1630E-06
CS135	6.3324E-02	1.2665E-02	1.0954E+00	3.1842E-03	1.3288E+01	1.9834E-01
CS137	7.0071E+04	1.4014E+04	1.3696E+06	3.9814E+03	1.6437E+05	2.4533E+03
I129	1.3431E-02	2.6862E-03	1.2277E-01	3.5689E-04	7.3156E-01	1.0919E-02
NB93M	7.1495E-02	1.4299E-02	4.3220E+00	1.2564E-02	2.0373E+01	3.0407E-01
NB94	1.3482E-05	2.6964E-06	6.5155E-03	1.8941E-05	1.0279E+00	1.5342E-02
NI 59	0.0000E+00	0.0000E+00	7.0021E-01	2.0355E-03	3.3893E+00	5.0586E-02
NI 63	2.3152E+01	4.6304E+00	2.9668E+03	8.6245E+00	4.1301E+02	6.1643E+00
NP237	1.7345E-01	3.4690E-02	2.2103E+00	6.4252E-03	4.6764E-02	6.9798E-04
PA231	1.8931E-06	3.7862E-07	4.2507E-05	1.2357E-07	7.1767E+01	1.0712E+00
PB210	7.0985E-12	1.4197E-12	1.0602E-07	3.0819E-10	5.1858E-03	7.7399E-05
PD107	9.3965E-03	1.8793E-03	3.4983E-01	1.0170E-03	1.5881E-01	2.3704E-03
PU238	5.1577E+02	1.0315E+02	2.7090E+04	7.8749E+01	1.8243E+02	2.7228E+00
PU239	8.7962E+02	1.7592E+02	1.0946E+05	3.1821E+02	1.1807E+01	1.7622E-01
PU240	7.5835E+02	1.5167E+02	9.5197E+04	2.7674E+02	6.7600E+00	1.0090E-01
PU241	8.0133E+03	1.6027E+03	8.8553E+05	2.5742E+03	1.6067E+03	2.3980E+01
PU242	7.5782E-03	1.5156E-03	6.7836E+00	1.9720E-02	1.5141E-02	2.2598E-04
RA226	1.0523E-08	2.1046E-09	1.8186E-06	5.2866E-09	2.9818E-03	4.4505E-05
RA228	3.2121E-12	6.4243E-13	6.8957E-09	2.0046E-11	4.6764E+00	6.9798E-02
SE79	2.4257E-01	4.8515E-02	1.2469E+00	3.6249E-03	1.6298E+01	2.4326E-01
SM151	8.5300E+02	1.7060E+02	8.2785E+04	2.4065E+02	6.0192E+03	8.9838E+01
SN126	2.1704E-01	4.3408E-02	1.6307E+00	4.7405E-03	1.8289E+01	2.7297E-01
SR90	6.1409E+04	1.2282E+04	5.3486E+05	1.5548E+03	1.6900E+05	2.5224E+03
TC99	8.1709E+00	1.6342E+00	4.3407E+01	1.2618E-01	1.5141E+02	2.2598E+00
TH229	1.1407E-09	2.2814E-10	5.7799E-07	1.6802E-09	1.2038E+01	1.7968E-01
TH230	2.3787E-06	4.7575E-07	3.9722E-04	1.1547E-06	4.5561E-01	6.8001E-03
TH232	2.0328E-11	4.0655E-12	1.8673E-08	5.4283E-11	5.5562E+00	8.2928E-02
U233	8.0011E-06	1.6002E-06	3.1714E-04	9.2192E-07	7.8249E+03	1.1679E+02
U234	2.2943E-02	4.5885E-03	2.7207E+00	7.9090E-03	3.8847E+02	5.7980E+00
U235	6.7436E-02	1.3487E-02	7.3490E-02	2.1363E-04	2.6160E-02	3.9045E-04
U236	2.8151E-01	5.6302E-02	1.4092E+00	4.0964E-03	5.3710E-02	8.0164E-04
U238	1.2358E-03	2.4717E-04	9.2919E-01	2.7011E-03	8.5195E-04	1.2716E-05
ZR93	1.2461E+00	2.4921E-01	5.8031E+00	1.6870E-02	3.8013E+01	5.6736E-01

Table D-1. (continued).

	TSPA Category	TSPA Category	TSPA Category	TSPA Category		
2,333 MTHM SNF, no HLW	13	13	16	16		
	MTHM	packages	MTHM	packages		
Hanford	0.03	3	0.15	5	sum MTHM Hanf	1989.45
INEEL	1.86	91	7.14	37	sum MTHM INEEL	303.04
SRS			2.72		sum MTHM SRS	40.51
Total	1.89	94	10.01	42	sum MTHM cmplx	2333
						complex
	total curies	Ci/pkg	total curies	Ci/pkg		total curies
Isotopes						
AC227	5.2604E-06	5.5962E-08	2.9322E-05	6.9815E-07	AC227	2.8644E+01
AM241	1.8921E+01	2.0129E-01	4.9749E+03	1.1845E+02	AM241	9.8298E+05
AM242M	2.1002E-01	2.2342E-03	6.6225E+00	1.5768E-01	AM242M	8.6593E+02
AM243	2.3272E-02	2.4757E-04	2.0797E+01	4.9518E-01	AM243	2.0125E+03
C14	7.9288E+00	8.4349E-02	7.3275E-02	1.7446E-03	C14	7.9683E+02
CL36	2.3843E-01	2.5365E-03	1.8151E-04	4.3216E-06	CL36	2.6432E+00
CM244	6.3377E-01	6.7422E-03	8.2008E+02	1.9526E+01	CM244	8.3072E+04
CM245	1.2940E-05	1.3766E-07	3.0651E-01	7.2978E-03	CM245	3.3761E+01
CM246	3.6322E-07	3.8640E-09	5.1617E-02	1.2290E-03	CM246	5.7147E+00
CS135	3.0086E+00	3.2006E-02	5.8646E+00	1.3963E-01	CS135	1.8613E+02
CS137	2.8572E+05	3.0396E+03	3.7407E+06	8.9063E+04	CS137	2.8620E+07
I129	6.7552E-02	7.1864E-04	9.9049E-01	2.3583E-02	I129	1.6255E+01
NB93M	6.6039E-01	7.0254E-03	7.0807E+00	1.6859E-01	NB93M	5.4915E+02
NB94	4.3331E-01	4.6097E-03	1.5034E-03	3.5795E-05	NB94	3.2199E+00
NI 59	4.6361E+01	4.9321E-01	3.5309E-02	8.4068E-04	NI 59	1.1404E+02
NI 63	5.7337E+03	6.0997E+01	4.3675E+00	1.0399E-01	NI 63	7.5838E+04
NP237	1.7993E-01	1.9142E-03	1.0304E+01	2.4534E-01	NP237	1.4586E+02
PA231	8.4017E-05	8.9380E-07	4.4752E-04	1.0655E-05	PA231	8.0872E+01
PB210	1.2752E-10	1.3566E-12	3.2265E-08	7.6822E-10	PB210	7.3686E-03
PD107	5.5820E-02	5.9383E-04	7.8318E-01	1.8647E-02	PD107	2.8122E+01
PU238	1.9867E+02	2.1135E+00	2.2670E+04	5.3977E+02	PU238	4.6332E+05
PU239	4.2389E+02	4.5094E+00	2.2283E+03	5.3054E+01	PU239	3.7968E+05
PU240	1.6501E+02	1.7554E+00	1.5649E+03	3.7259E+01	PU240	2.8488E+05
PU241	1.2299E+04	1.3085E+02	1.5108E+05	3.5972E+03	PU241	6.0838E+06
PU242	2.2894E-02	2.4355E-04	3.0135E+00	7.1751E-02	PU242	2.7946E+02
RA226	2.1191E-10	2.2544E-12	1.7687E-07	4.2112E-09	RA226	7.3406E-03
RA228	3.2552E-07	3.4629E-09	6.4237E-10	1.5295E-11	RA228	7.9077E+00
SE79	1.2035E+00	1.2803E-02	1.9938E+01	4.7472E-01	SE79	2.7246E+02
SM151	2.2329E+03	2.3755E+01	1.6110E+04	3.8356E+02	SM151	4.3684E+05
SN126	1.1145E+00	1.1857E-02	1.4350E+01	3.4166E-01	SN126	2.7080E+02
SR90	2.7248E+05	2.8987E+03	3.5681E+06	8.4955E+04	SR90	2.3710E+07
TC99	4.0497E+01	4.3082E-01	5.8221E+02	1.3862E+01	TC99	7.3674E+03
TH229	1.0710E-07	1.1394E-09	3.7239E-07	8.8665E-09	TH229	2.5533E+01
TH230	3.5950E-07	3.8245E-09	4.9831E-05	1.1865E-06	TH230	1.3662E+00
TH232	9.6321E-07	1.0247E-08	4.2075E-09	1.0018E-10	TH232	8.1062E+00
U233	5.2415E-04	5.5761E-06	6.8161E-04	1.6229E-05	U233	1.1236E+04
U234	1.9678E-02	2.0934E-04	7.7030E-01	1.8340E-02	U234	1.5534E+03
U235	9.2152E-01	9.8034E-03	5.0201E+00	1.1953E-01	U235	7.3337E+01
U236	1.2375E+00	1.3165E-02	1.9069E+01	4.5402E-01	U236	3.6100E+02
U238	5.3935E-01	5.7378E-03	1.0095E+00	2.4037E-02	U238	7.1810E+02
ZR93	7.2660E+00	7.7298E-02	8.0618E+01	1.9195E+00	ZR93	1.4176E+03

Table D-2. HLW Inventory.

Isotope	DHLW Inventory (Ci/pkg)[a]	Isotope	DHLW Inventory (Ci/pkg)[a]
^{227}Ac	2.41E-03	^{239}Pu	1.89E+01
^{241}Am	3.46E+02	^{240}Pu	1.32E+01
^{242m}Am	8.24E-02	^{241}Pu	5.92E+02
^{243}Am	1.47E-01	^{242}Pu	2.01E-02
^{14}C	0.00E+00	^{226}Ra	3.75E-07
^{36}Cl	0.00E+00	^{228}Ra	0.00E+00
^{244}Cm	4.56E+01	^{79}Se	3.67E-01
^{245}Cm	2.26E-04	^{151}Sm	0.00E+00
^{246}Cm	2.56E-05	^{126}Sn	0.00E+00
^{135}Cs	4.60E-01	^{99}Tc	1.32E+01
^{129}I	7.60E-06	^{229}Th	6.04E-05
^{93m}Nb	2.19E+00	^{230}Th	4.96E-05
^{94}Nb	1.21E-04	^{232}Th	4.20E-04
^{59}Ni	1.08E-01	^{233}U	2.34E-03
^{63}Ni	0.00E+00	^{234}U	2.00E-01
^{237}Np	1.13E-01	^{235}U	3.17E-04
^{231}Pa	3.90E-03	^{236}U	1.74E-03
^{210}Pb	1.09E-07	^{238}U	1.51E-02
^{107}Pd	0.00E+00	^{93}Zr	2.80E+00
^{238}Pu	1.60E+03		

a. Assumed 4 canisters per container.

Source: RW M&O 1995 TSPA x4 [Reference 31].