DOE/SNF/REP-027
Revision 0

United States Department of Energy

Office of Environmental Management

DOE Spent Nuclear Fuel Information In Support of TSPA - VA

September 1998

U.S. Department of Energy
Assistant Secretary for Environmental Management
Office of Spent Fuel Management and Special Projects

DOE/SNF/REP-0027
Revision 0

DOE Spent Nuclear Fuel Information In Support of TSPA - VA

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September 1998

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.

Prepared for the
U.S. Department of Energy
Assistant Secretary for Environmental Management
Under DOE Idaho Operations Office
Contract No. DE-AC07-94ID13223

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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ACKNOWLEDGMENTS

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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,

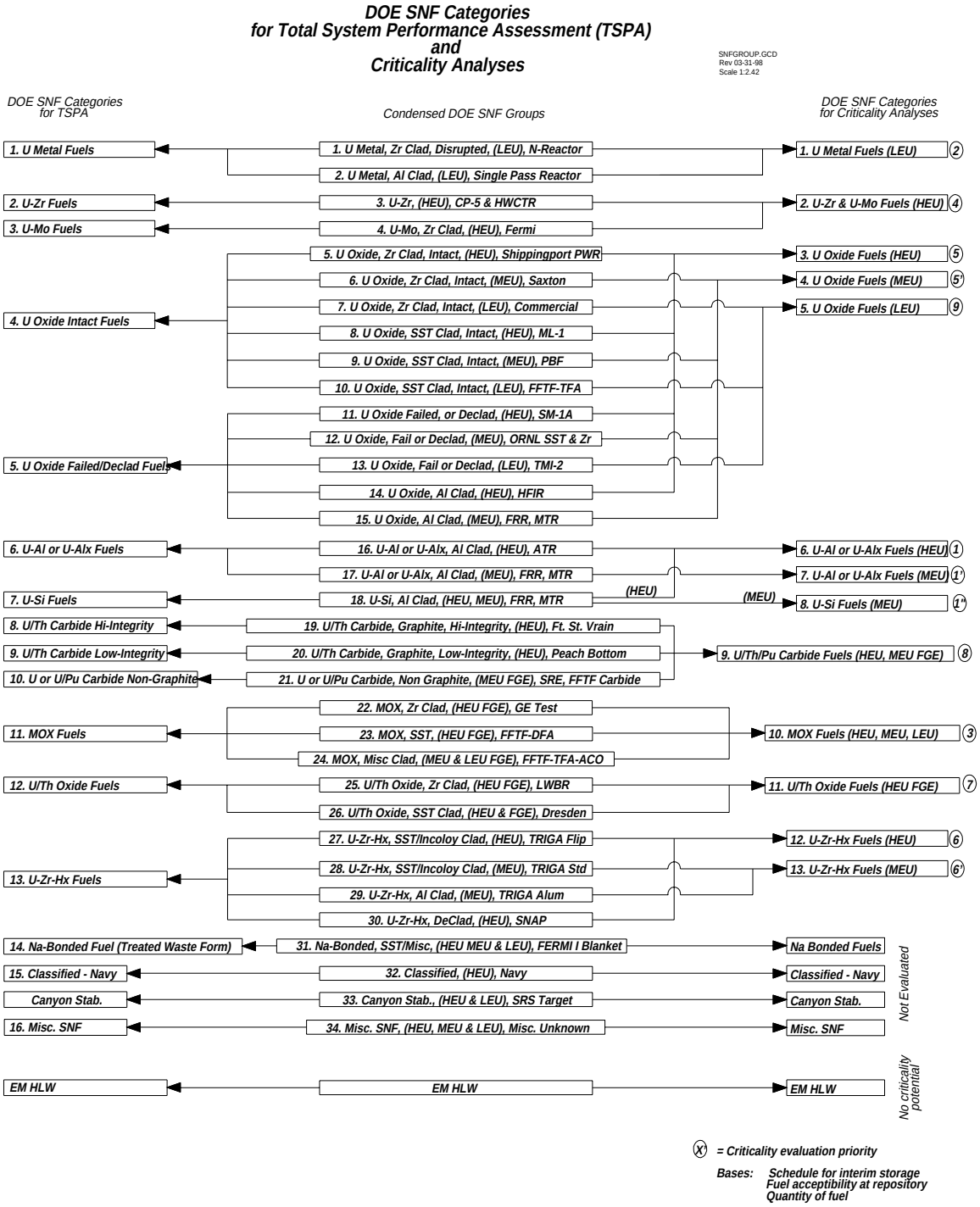

Figure 2-1. DOE SNF condensed groups, categories for TSPA, and criticality analyses.

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.

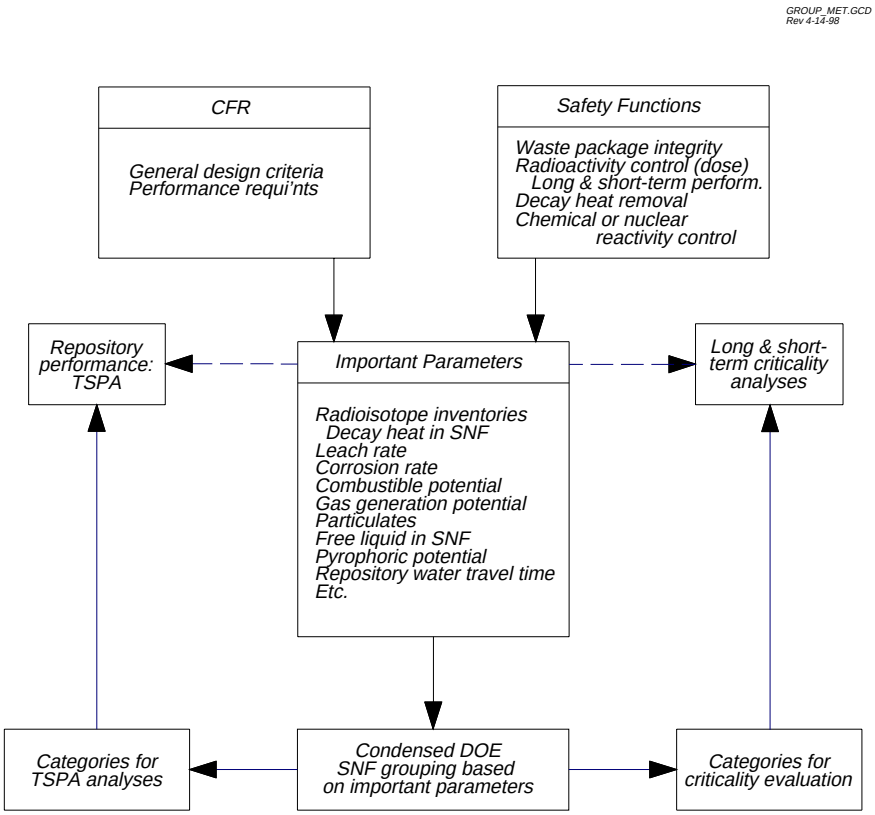

Figure 2-2. DOE SNF grouping methodology.

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.

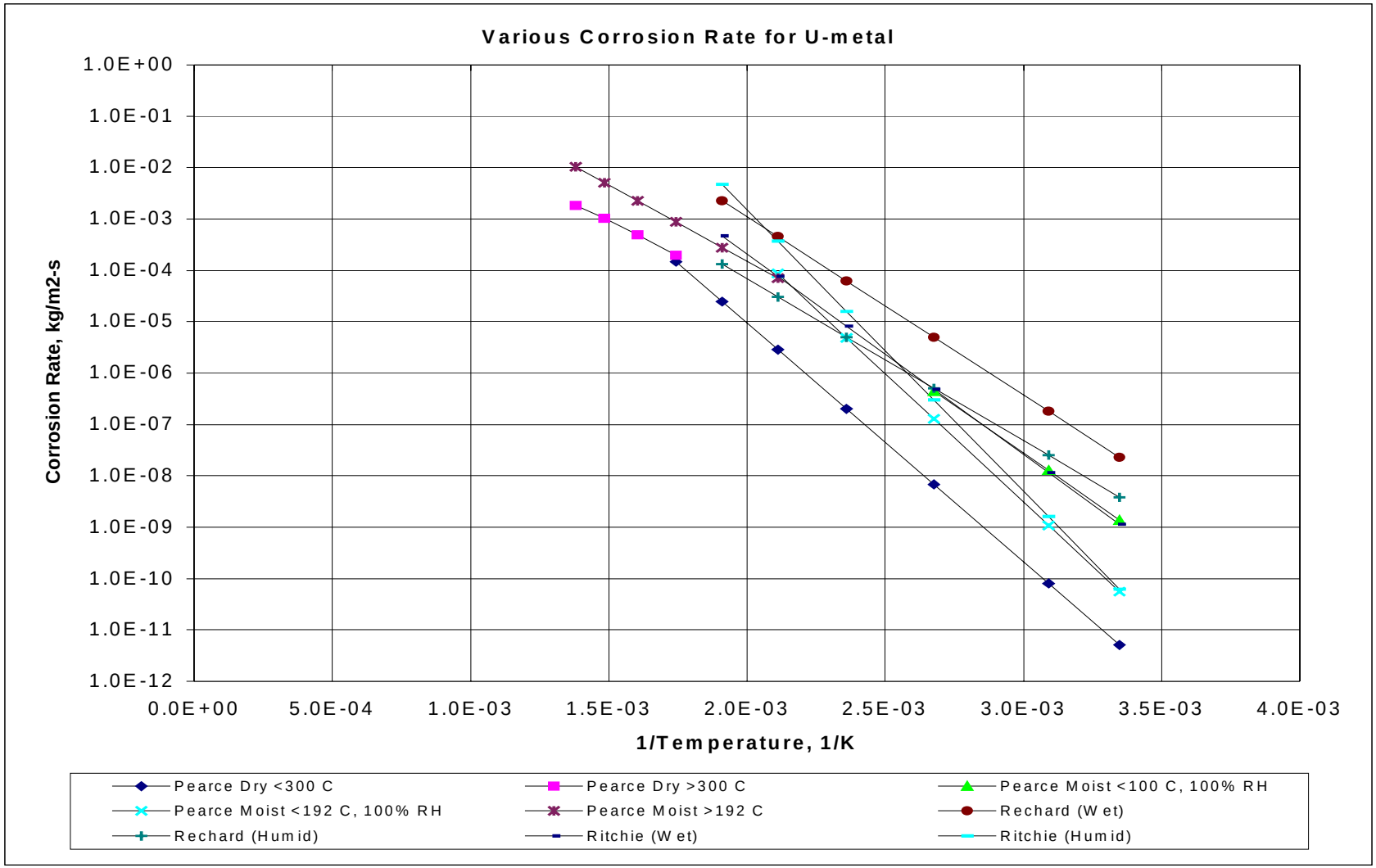

Figure 3-1. Various U-metal reaction rate equations plotted with temperature.

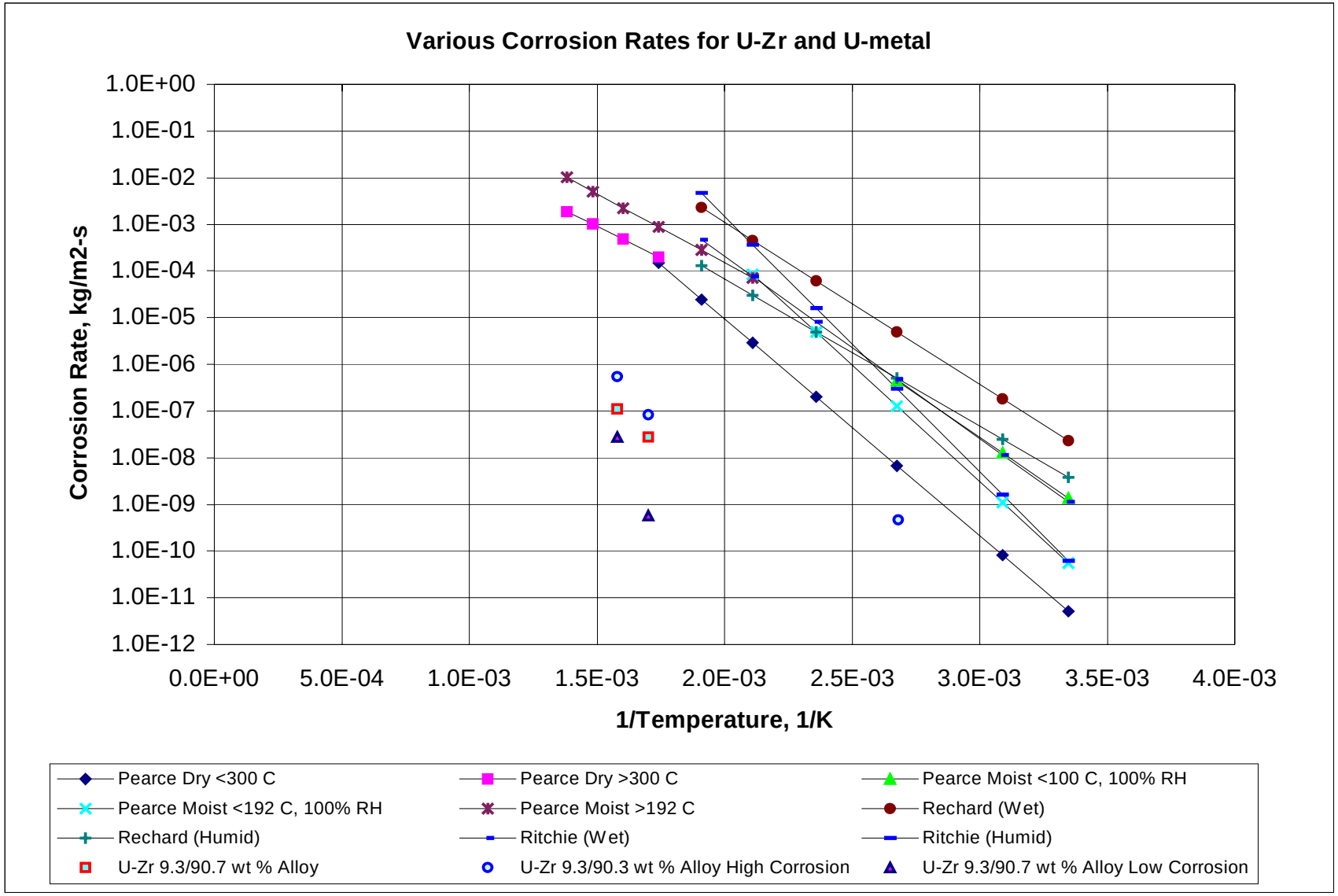

Figure 3-2. U-Zr and U-metal corrosion rate plotted with temperature.

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 ºF 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:

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

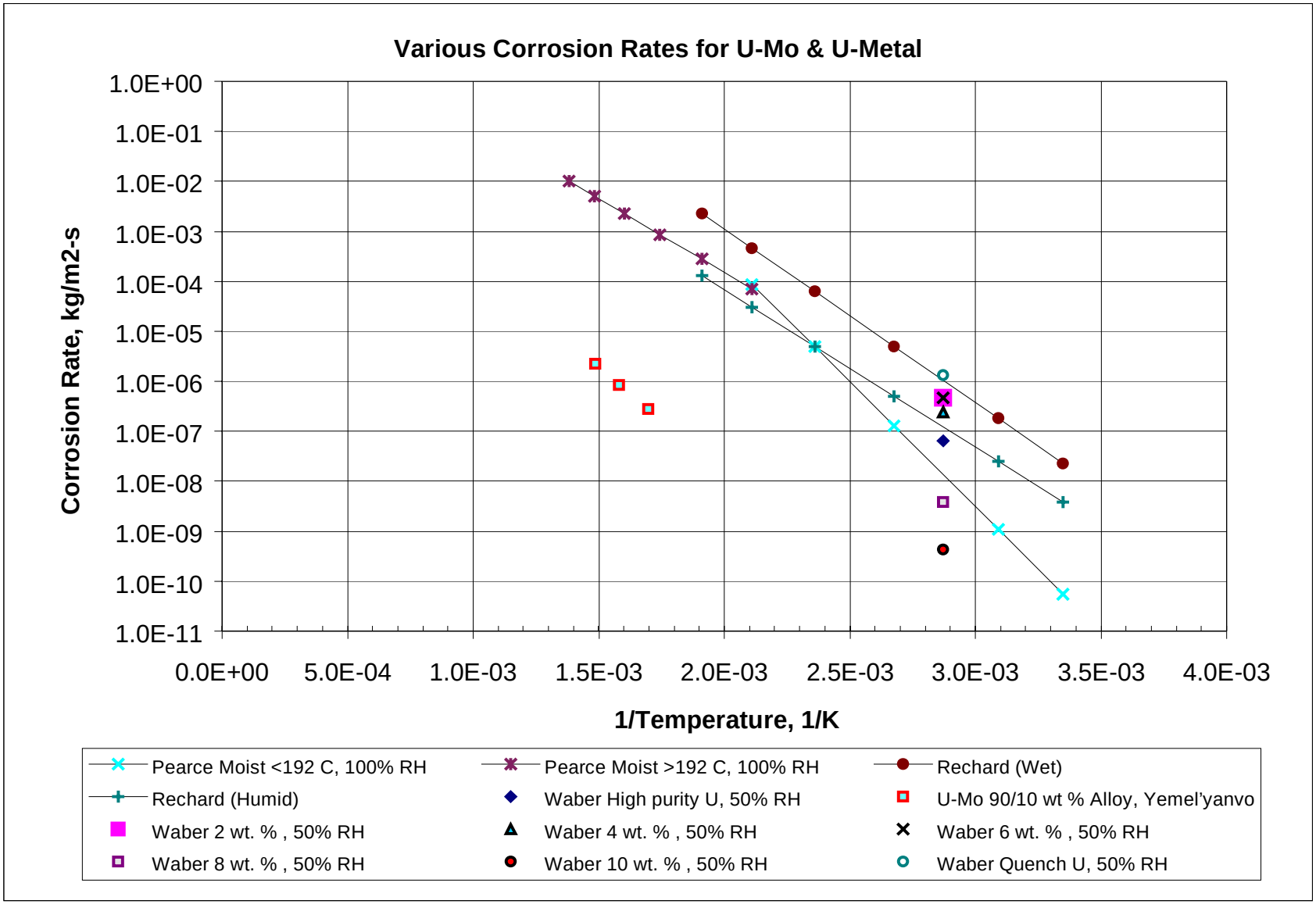

Figure 3-3. U-Mo and U-metal corrosion rate plotted with temperature.

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.

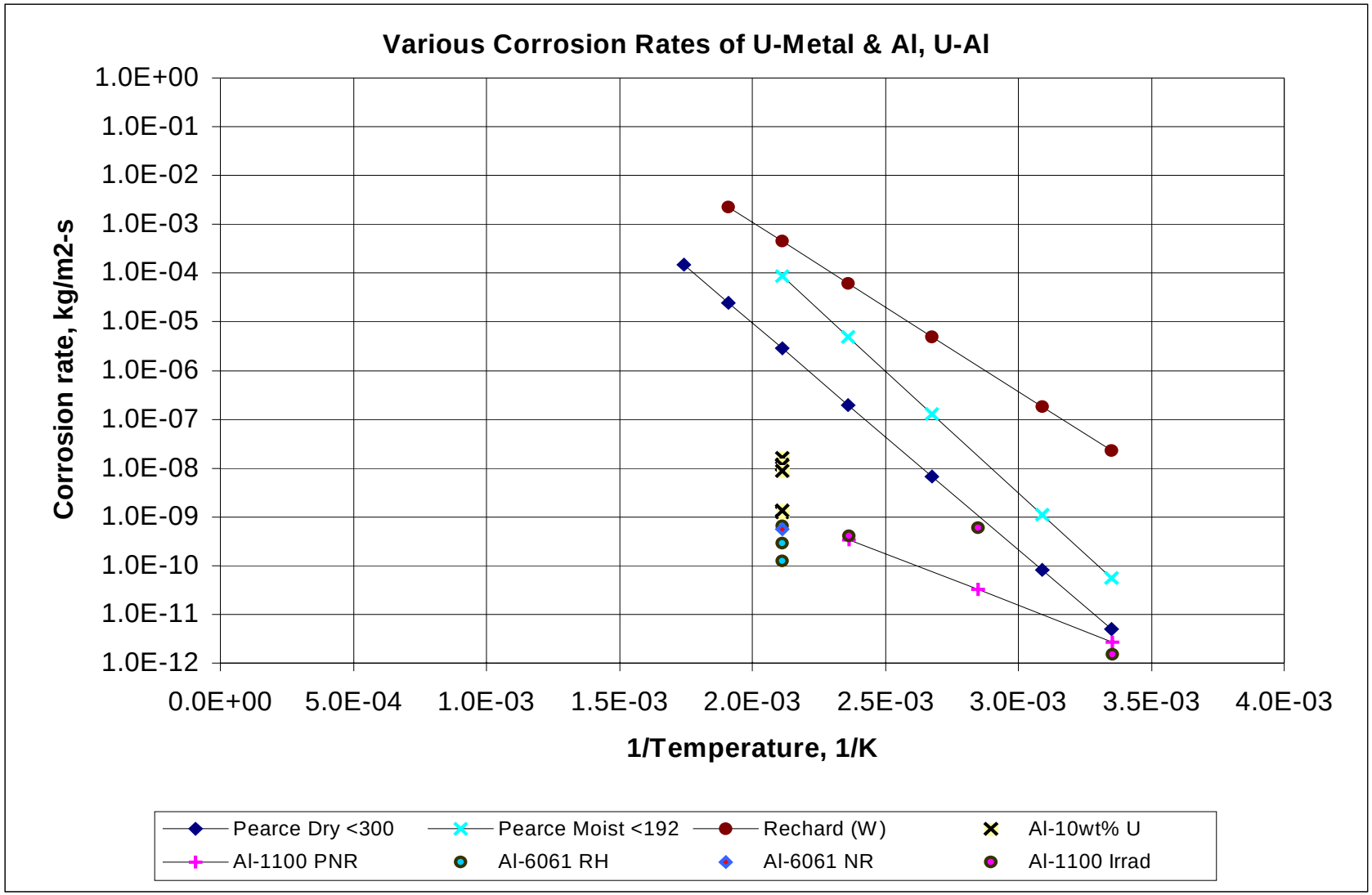

Figure 3-4. Al, U-Alx and U-metal corrosion rate plotted with temperature.

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

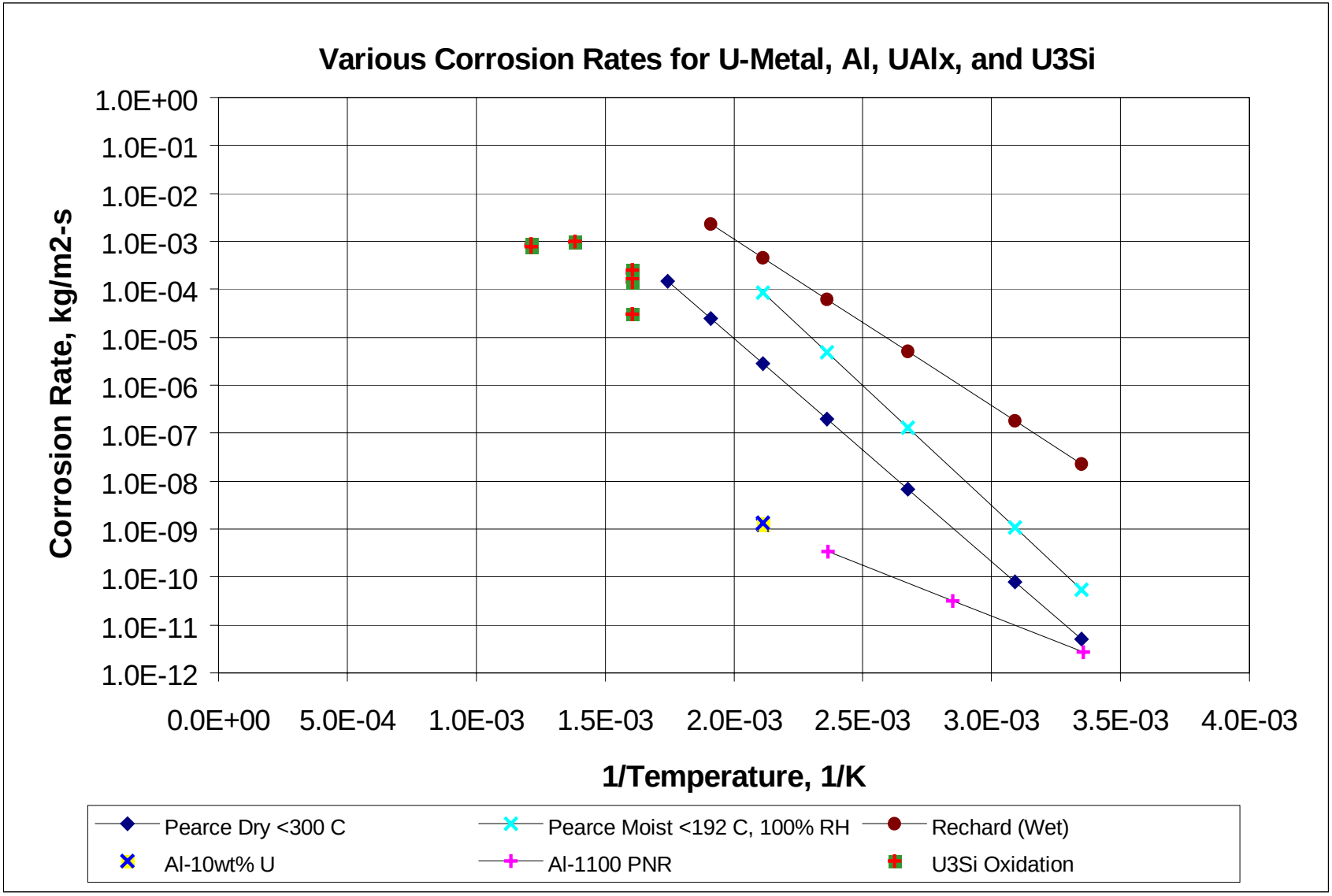

Figure 3-5. U_3Si Al, U-Alx and U-metal corrosion rate plotted with temperature .

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].

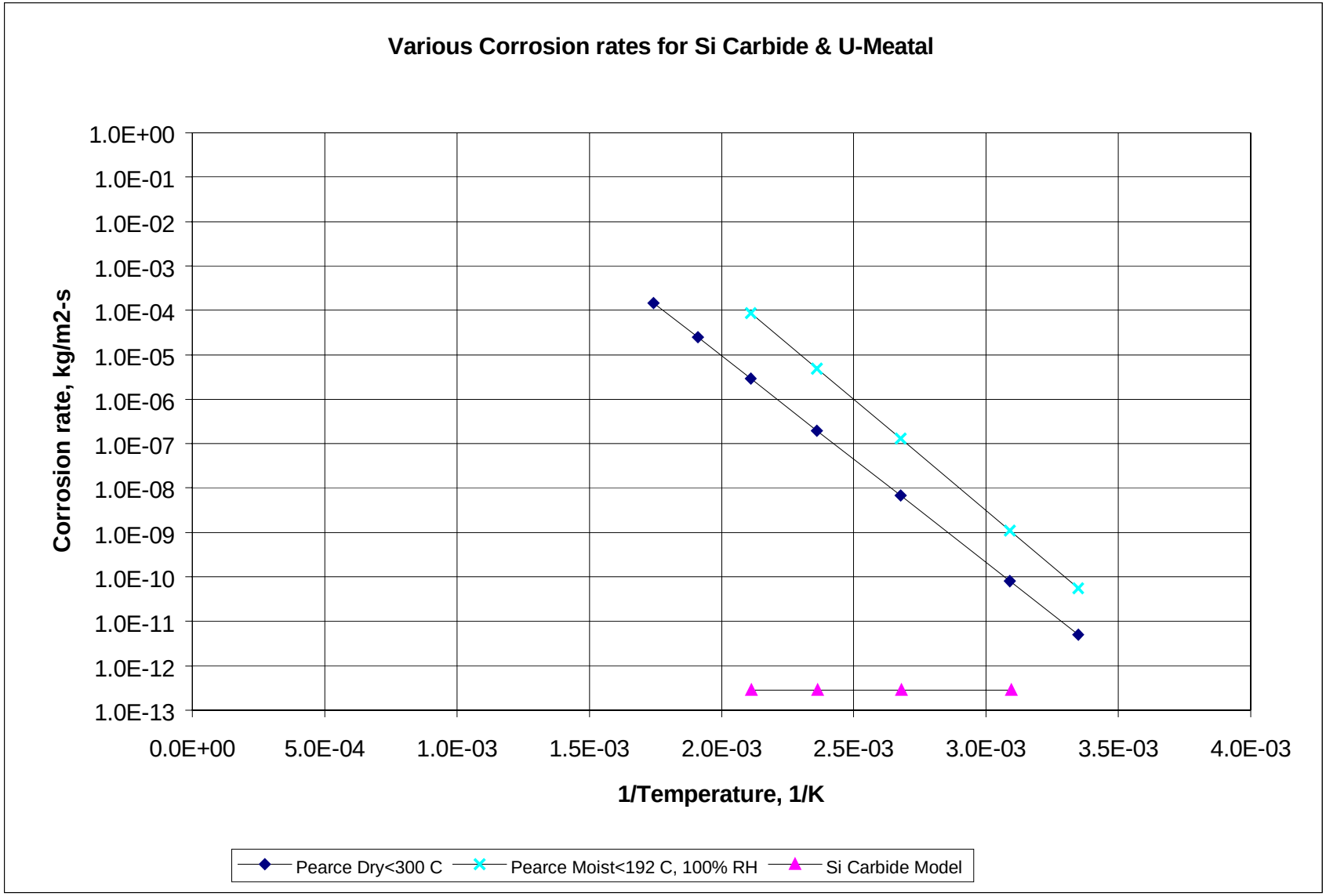

Figure 3-6. SiC, and U-metal corrosion rate plotted with temperature.

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.

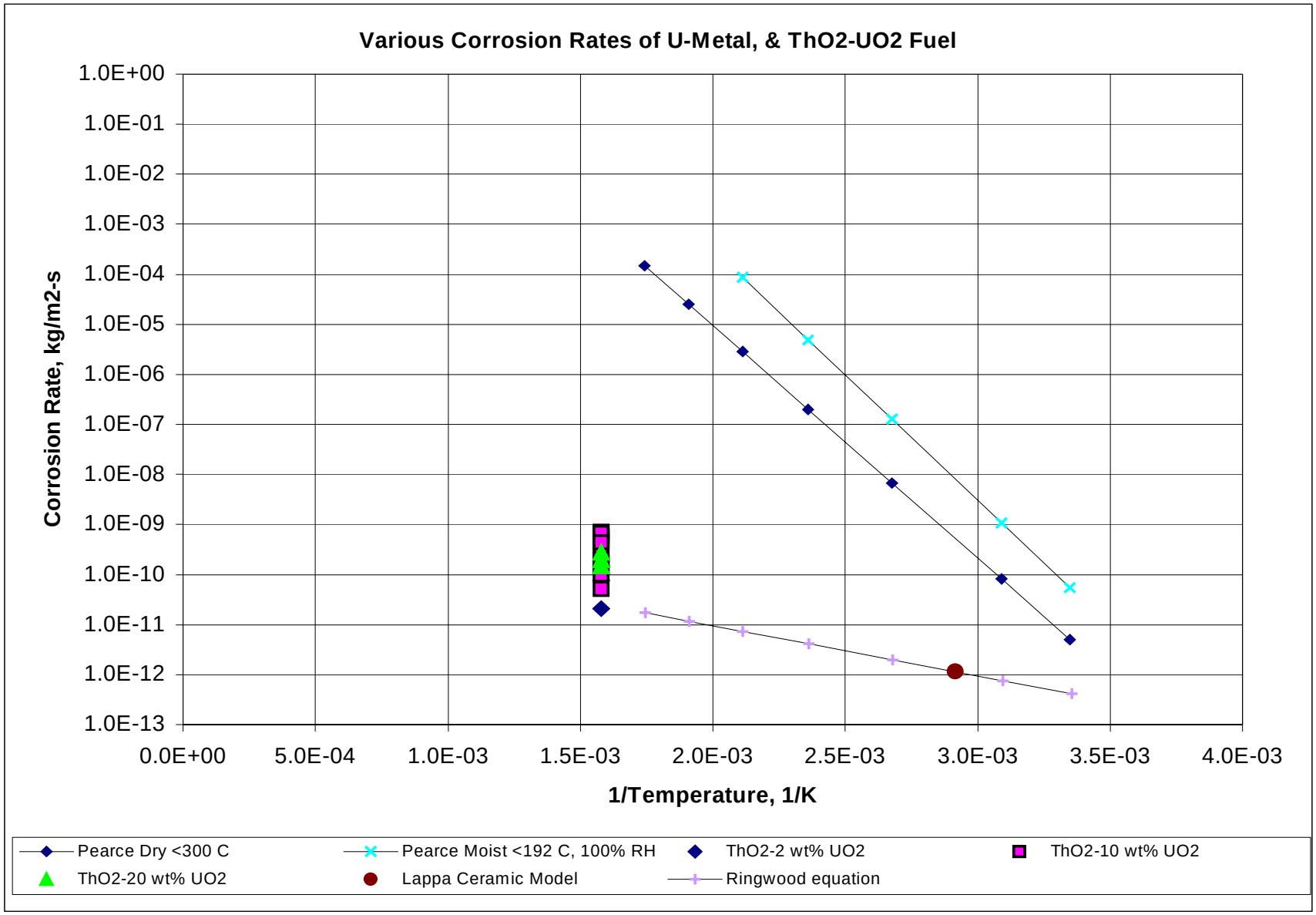

Figure 3-7. Th/U Oxide, and U-metal Corrosion Rate Plotted with Temperature

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).
2	=	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)
∀, ∃	=	constants (∀ = 0.082, ∃ = 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	

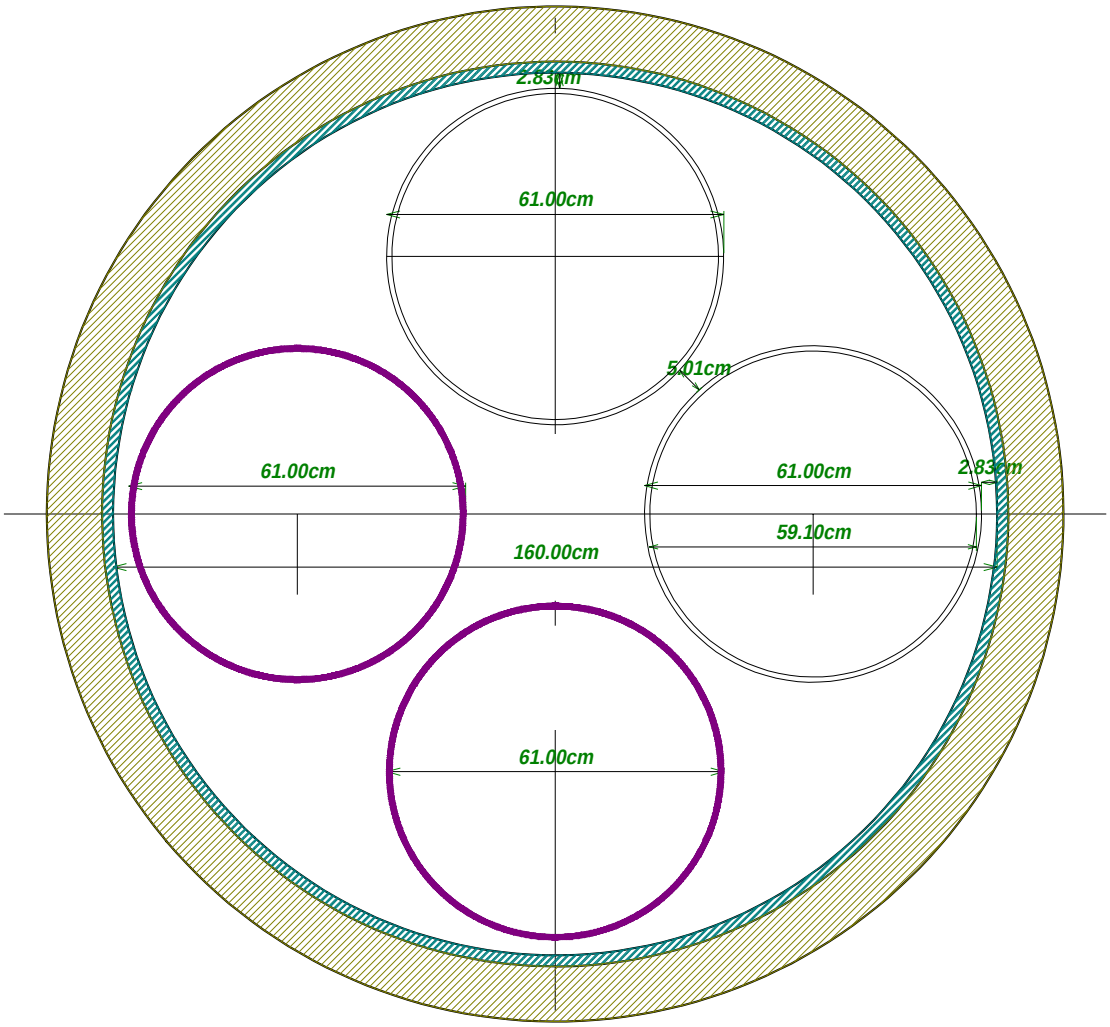

Figure 6-1. Proposed 3 (HLW) x 1 (SNF) co-disposal package.

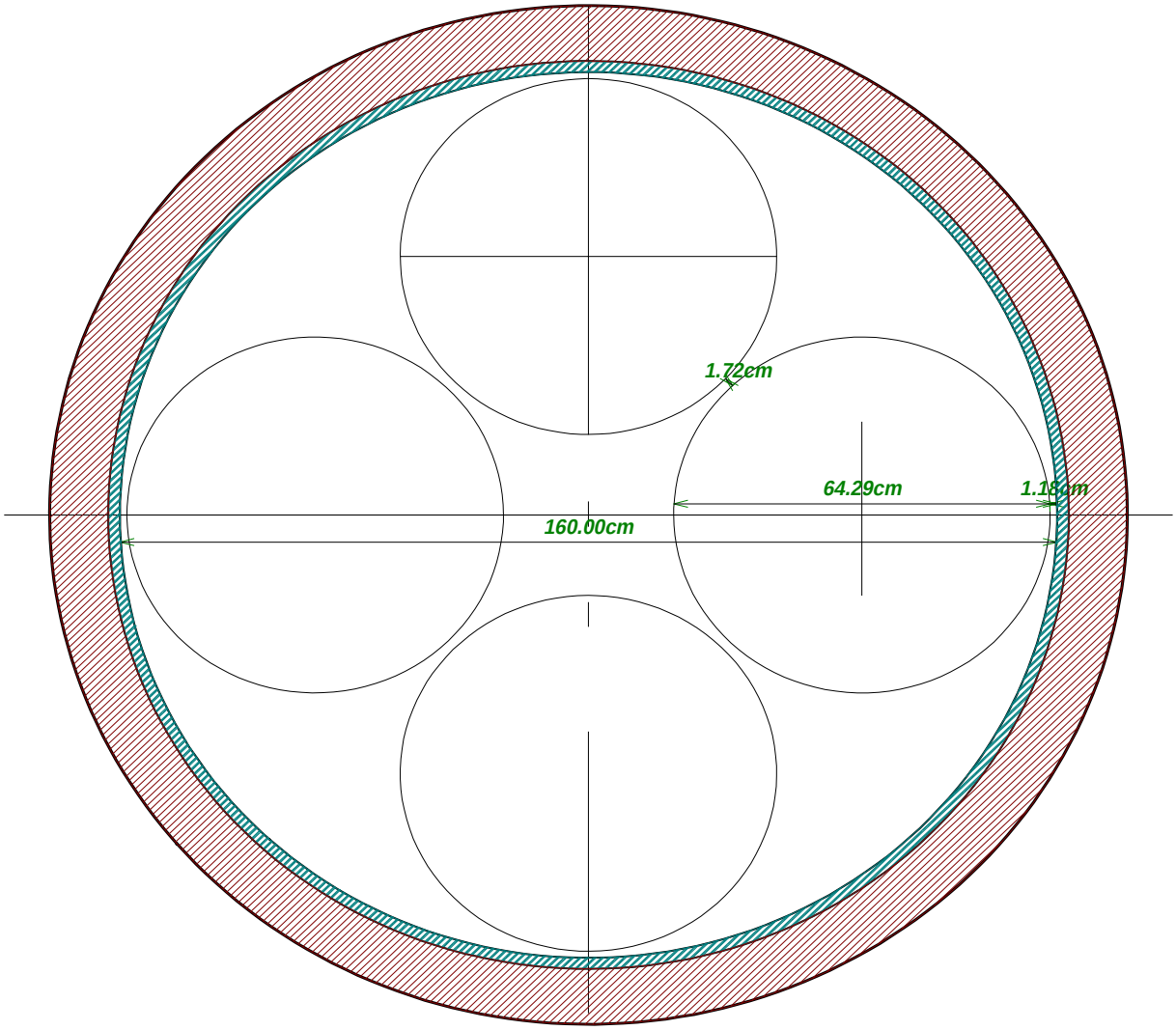

Figure 6-2. Proposed 0 (HLW) x 4 (MCO SNF) disposal package.

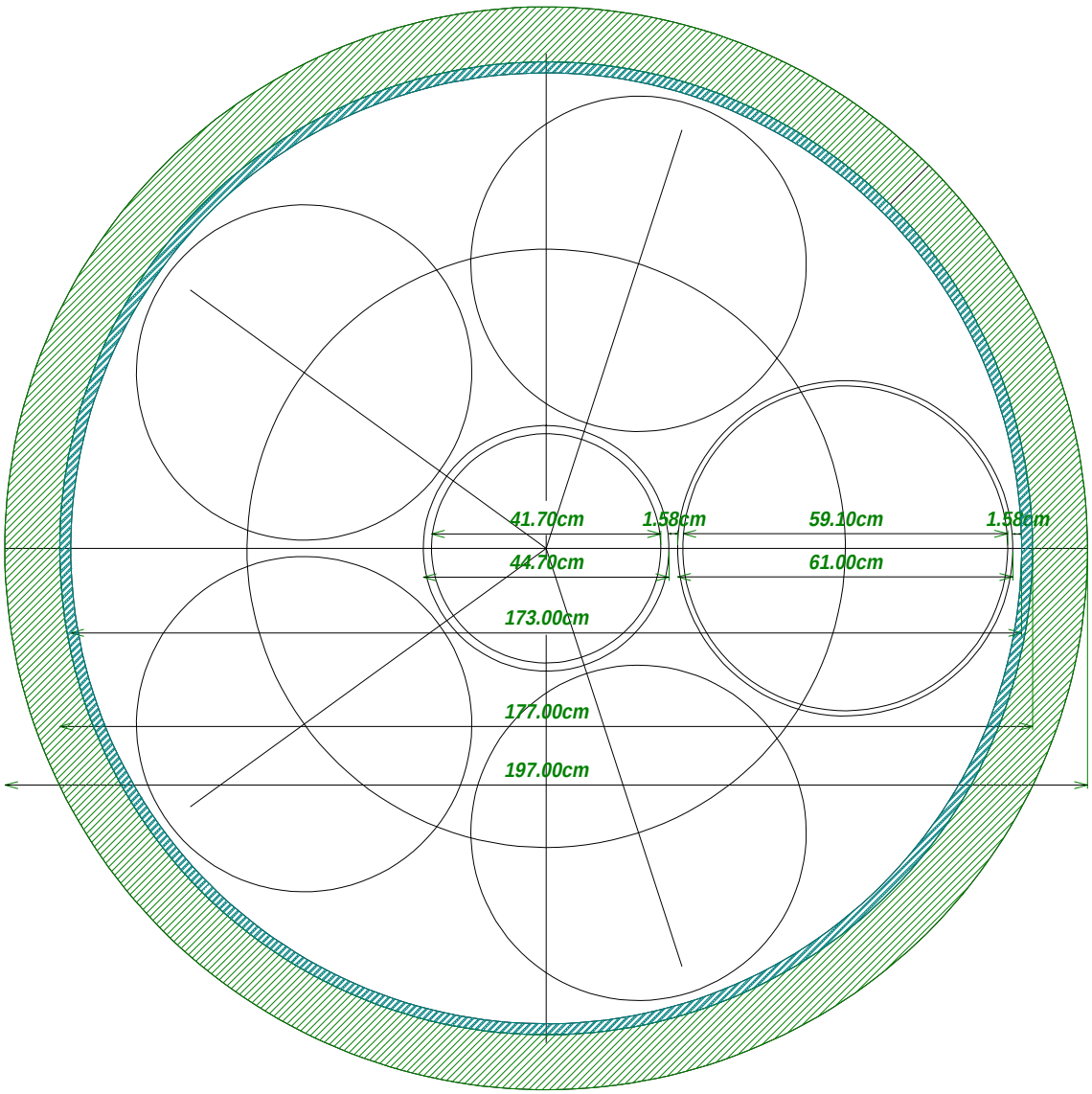

Figure 6-3. Proposed 5 (HLW) x 1 (SNF) co-disposal package [both ~10 and 15 feet lengths].

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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