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Office of Environmental Management

DOE Spent Nuclear Fuel Information In Support of TSPA - SR

August 1999

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DOE Spent Nuclear Fuel Information In Support of TSPA - SR

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DOE Spent Nuclear Fuel Information In Support of TSPA - SR

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August 1999

This report was 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 and Description. However, the data presented in the report are unqualified.

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ABSTRACT

The Office of Civilian Radioactive Waste Management (RW) has started the site recommendation (SR) effort to show that Yucca Mountain could be selected as the first geologic repository for spent nuclear fuel (SNF) and high-level waste. One component of the site recommendation 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. 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-SR evaluation. A number of parameters are needed in the TSPA-SR 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-SR data were requested. Thus, expert judgement and opinion were used to determine a best property value. The performance of the DOE-owned SNF will be published as part of the TSPA-SR report.

The DOE Spent Nuclear Fuel Information in Support of TSPA-SR Report contributors

The information contained in this report came from a number of people working on the DOE Spent Nuclear Fuel Program. (Listed in alphabetical order).

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ACRONYMS

ATR	Advanced Test Reactor
ACRR	Annular Core Research Reactor
BAPL	Bettis Atomic Power Laboratory
BOL	beginning of life
BWR	boiling-water reactor
DBE	design basis events
DFA	driver fuel assembly
DOE	Department of Energy
DRCT	Dry Rod Consolidation Test
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
FGE	fissile gram equivalent
FRR	foreign research reactor
FSV	Fort St. Vrain
HEU	high enriched uranium (>20% ^{235}U equivalent)
HIC	high integrity container
HLW	high-level waste
HTGR	High Temperature Gas Cool Reactor
HWCTR	Heavy Water Components Test Reactor
ICD	Interface Control Document
INEEL	Idaho National Engineering and Environmental Laboratory
LA	license application
LDP	large disposal package

LEU	low enriched uranium (<5% ^{235}U equivalent)
LWBR	light water breeder reactor
LWR	light water reactor
M&O	management and operation (Contractor)
MCO	multi-canister overpack
MEU	medium enriched uranium (>5% <20% ^{235}U equivalent)
MGDS	mined geologic disposal system
MOX	mixed oxide fuel
MTHM	metric tons heavy metal
MURR	Missouri University Research Reactor
MWd/kgU	SNF burnup in megawatt-day/kilograms uranium
NNPP	Naval Nuclear Propulsion Program
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
PA	performance assessment
PB	Peach Bottom
PNNL	Pacific Northwest National Laboratory
PBF	Power Burst Facility
PWR	pressurized water reactor
QARD	Quality Assurance Requirements Description
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

SR	site recommendation
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-SR

1. OBJECTIVE AND SCOPE

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 repository license application (LA) together with the commercial SNF and high level waste (HLW). Before the LA, Congress mandated the Office of Civilian Radioactive Waste Management (RW) to prepare the viability assessment (VA). The VA was to serve two purposes: (1) to provide information on the progress of the Yucca Mountain Site Characterization Project, and (2) to identify critical issues that need to be addressed so the Secretary of Energy can decide whether or not to recommend the Yucca Mountain site as the repository. In 1998, the Office of Civilian Radioactive Waste Management (RW) management and operation (M&O) contractor TRW Environmental Safety Systems (TESS) completed the *Viability Assessment of a Repository at Yucca Mountain*, DOE/RW-0508 [DOE 1998a,[1]]. As part of the effort, TESS also helped the DOE's Office of Environmental Management (EM) by including the DOE-owned SNF in the total system performance assessment viability assessment (TSPA-VA).

Volume 4 of the VA contained RW's plan and cost estimate for the remaining work required to complete and submit a LA to the Nuclear Regulatory Commission (NRC). In the plan, RW presented the rationale for the technical work needed to complete the LA. The technical work was divided into three major areas: (1) postclosure safety case, (2) preclosure safety case, and (3) additional work needed to complete design decisions. Figure 1-1 shows a summary representation of the technical work needed to complete the LA.

Since the VA analyses, RW has started the site recommendation (SR) effort to show that Yucca Mountain should be selected as the first geologic repository for SNF and HLW. A TSPA will also be part of the SR effort. The TSPA-SR will incorporate all the comments received from the TSPA-VA review by the NRC and peer review panel. To ensure that the DOE-owned SNF continues to be acceptable for disposal in the repository, it will also be included in the TSPA-SR evaluation.

A number of parameters are needed in the TSPA-SR models to predict the performance of the SNF materials placed into the potential repository. This report documents the basis and/or derivation for each of these parameters.

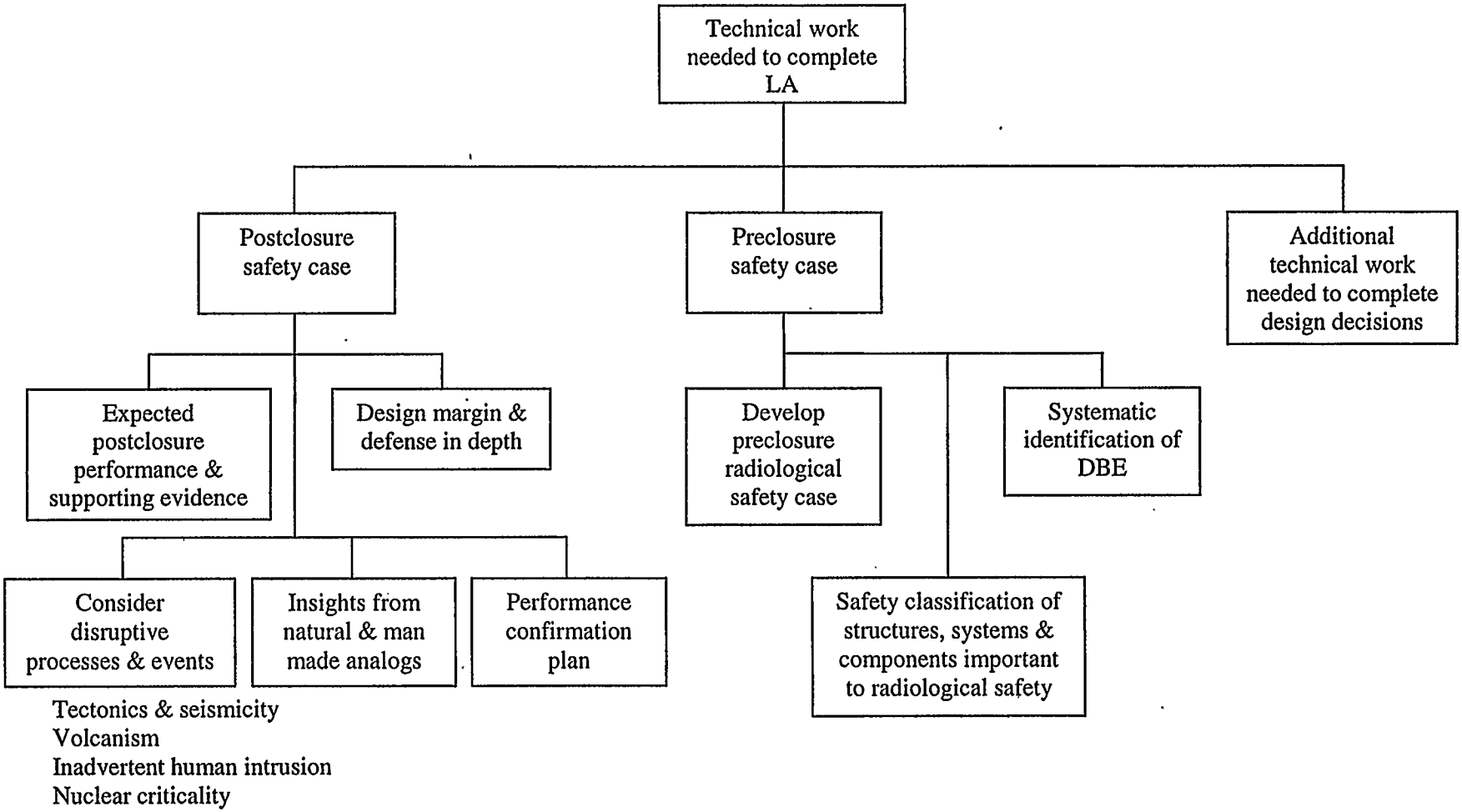

Figure 1-1. Repository technical work needed to complete SR/LA

- From viability assessment of a Repository at Yucca Mountain, DOE/RW-0508, Volume 4.

2. RESPONSIBILITIES/INTERFACES

The National Spent Nuclear Fuel (NSNF) Program will be responsible for interfacing with all the DOE sites to gather all the latest information on the various DOE SNF and providing the information to the RW M&O for inclusion into the VA, SR, and future LA TSPA analyses.

3. SPENT NUCLEAR FUEL DATA

All data came from the DOE sites, various DOE publications, and other commercial publications as indicated in this report. These data are "existing information" on the DOE SNF. As currently agreed by the RW M&O in the January 14, 1999 NSNFP/YMP Bimonthly Meeting minutes and stated in the RW Quality Assurance Requirements and Description DOE/RW-0333P Revision 8 Supplement III, existing DOE SNF information may be used for the purposes of repository SR/LA.

4. COMPUTER CODE/SOFTWARE

In preparing this report, the following computer software was used. Microsoft® WORD and Excel 97 SR2 program loaded on a DELL OptiPlex GXMT 5166. The computer has been certified to be year 2000 (Y2K) ready according to Lockheed Martin Idaho Technologies Company's (the INEEL management and operation contractor) Y2K desktop ready plan.

Computer software:	Microsoft® WORD and Excel 97 SR2
Computer Hardware:	DELL OptiPlex GXMT 5166

5. DOE SNF GROUPING AND RATIONALE

5.1 Background on DOE SNF Grouping

Since the inception of the DOE SNF program, the management of DOE SNF in a small number of groups has been discussed and reviewed in a number of published reports and meeting documents. The report titled *Grouping Method to Minimize Testing for Repository Emplacement of DOE SNF,* published in January 1997 is one such example [DOE-EM 1997a[2]]. The various reports provide the background on the many DOE SNF types (more than 200) located at the various DOE sites and present the reasons for grouping of DOE-owned SNF for specific proposes, such as repository disposition. However, at the time, none of the reports nor the meetings were developed or conducted in a manner consistent with the RW quality requirements [i.e., meeting the requirements of the Quality Assurance Requirements and Description (QARD) document.] DOE-EM 1997a suggested 11 groups to represent the DOE-owned SNF for the purpose of performance assessment and gave reasons for selecting 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, as well as the decision to expand grouping DOE SNF for other analyses needs in support of the repository license. The November 17–18, 1998, DOE SNF Grouping Meeting was one such example [DOE EM 1998a[3]].

The main goal of grouping the DOE SNF is to supply characterization data in a cost effective manner to support DOE SNF management and disposal without increased risk to the public, environment, or worker safety. As indicated in the DOE-EM 1997a report, the data needs required to meet the NRC and Environmental Protection Agency (EPA) regulations were evaluated. In the DOE-EM 1997a report, 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.

From these 34 DOE groups, it was determined that the groups may be further reduced to support both TSPA and criticality analyses. Specifically, the 34 groups of SNF were further reduced to 16 groups for the TSPA and 13 groups for criticality analyses purposes. The preliminary rationale used to reduce the groups further for TSPA was provided in the report titled *DOE SNF Information in Support of TSPA-VA* [DOE EM 1998b[4]]. The condensed DOE SNF groups, the TSPA groups, and criticality analyses groups are shown in Figure 5-1 and Table 5-1. The representative fuel in each condensed group was selected based generally on the quantity of the SNF within that specific group. These DOE SNF groups were used in the TSPA-VA analyses. U-metal fuel was used as the design basis DOE fuel that bounds the entire DOE SNF inventory.

On November 17–18, 1998, a follow-on DOE SNF grouping meeting was held by the NSNF Program to further refine and document the grouping process based on RW's plan and

VA
DOE SNF Grouping
for Total System Performance Assessment (TSPA)
and
Criticality Analyses

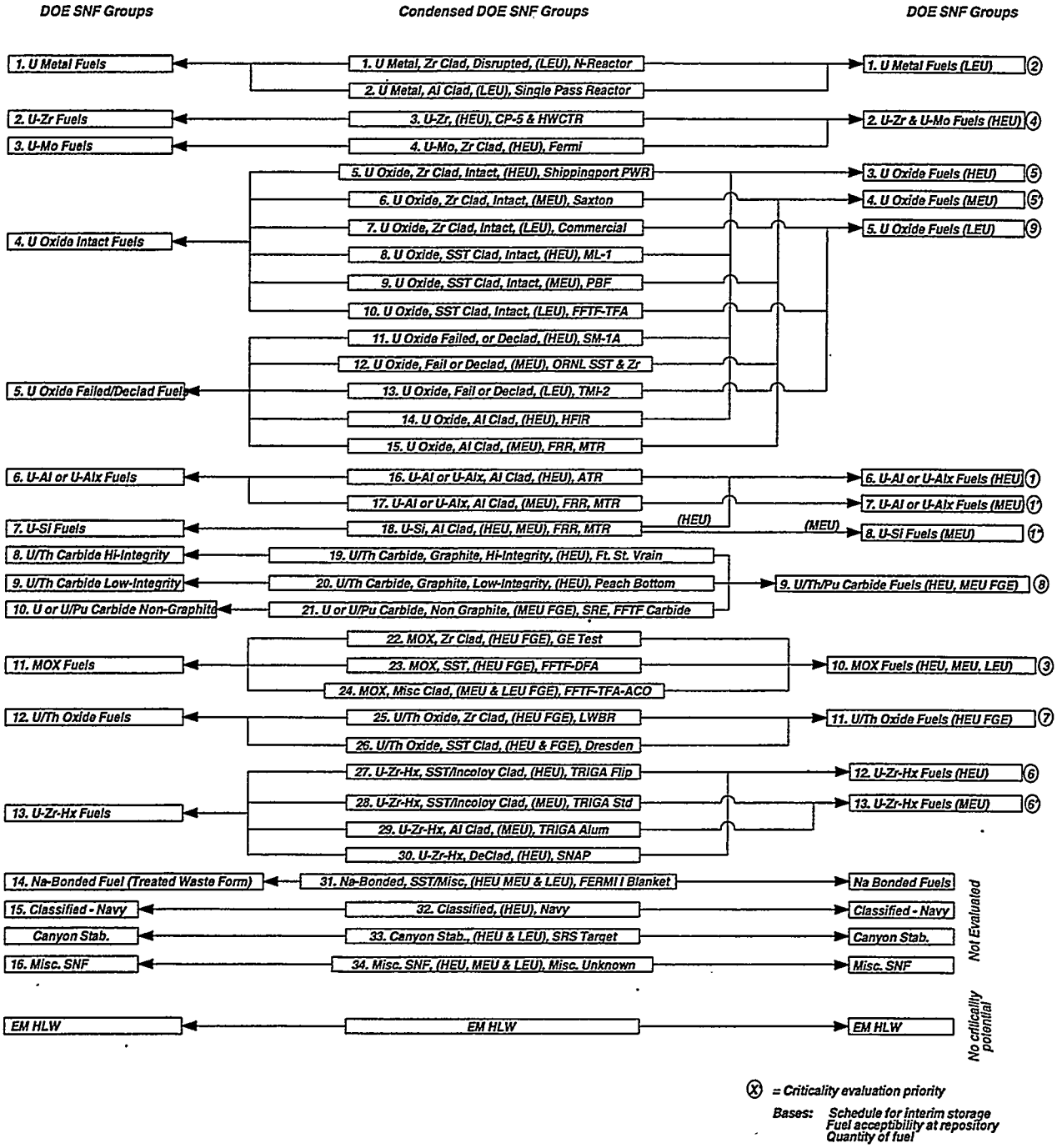

Figure 5-1. DOE SNF condensed groups, groups for TSPA-VA, and criticality analyses in FY 1998.

Table 5-1. DOE SNF fuel groups used in the TSPA-VA in FY 1998.

Fuel Group	Fuel Matrix	Typical Fuel in the Group	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 pressurized water reactor (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 (FRR) fuel	
8	U/Th carbide high-integrity	Fort St. Vrain (FSV) fuel	
9	U/Th carbide low-integrity	Peach Bottom (PB) fuel	
10	U or U/Pu carbide nongraphite	Fast Flux Test Facility (FFTF) carbide fuel	
11	Mixed oxide (MOX) fuel	FFTF oxide fuel	
12	U/Th oxide	Shippingport light water breeder reactor (LWBR) Fuel	
13	U-Zr-Hx	Training Research Isotopes — General Atomic (TRIGA) fuel	
15	Classified-Navy	Navy	Info by Navy
16	Misc. SNF	Misc. fuel	

cost estimate for the remaining work required to complete and submit an LA to the NRC. Based on RW's LA plan described in the VA, grouping of DOE SNF was discussed and expanded to support licensing analyses in three major areas. They are criticality, design basis events, and performance assessment. The detailed rationale for grouping DOE SNF for each of these areas are covered in the report DOE SNF Grouping in Support of Criticality, DBE, and TSPA-LA, DO/SNF/REP-0046 [DOE-EM 1999a[5]]. For performance assessment analyses, the DOE grouping team concluded that the 34 intermediate condensed groups of the DOE SNF continues to support RW's postclosure safety case and the DOE SNF could be refined and represented by 13 groups instead of 16 groups for the purpose of postclosure performance assessment. Subsequent to the November meeting, the team suggested that the plutonium/uranium nitride fuels be placed in the unknown group due to their small quantity and the uranium beryllium oxide fuels be placed into the uranium oxide group because of its similar characteristics. Thus, for the purpose of all future performance assessment, the DOE SNF grouping has been refined to 11 TSPA groups. The 11 DOE SNF groups and representative fuels for the SR/LA are indicated on Figure 5-2 and Table 5-2. Section 5.2 below further discusses the reasons for the grouping selections.

The Navy fuels, Group 1, was placed in its own criticality, DBE, and TSPA group for several reasons including the following:

- The design of naval SNF is significantly different than other DOE SNF designs.
- Because of its robust design, naval SNF will remain virtually intact well beyond several hundred-thousand years and its impact on repository performance will occur much later than other DOE SNF designs.
- The design of naval SNF is classified.

The organizational interface between the OCRWM and the Naval Nuclear Propulsion Program (NNPP) is documented in the OCRWM/NNPP Memorandum of Agreement [DOE 1998b[6]].

5.2 Reason for Group Selection for the DOE SNF

The SR and LA long-term performance predictions necessitate a certain knowledge of the fuel to provide a reasonable assurance that a repository at Yucca Mountain will adequately protect the safety and health of the public and the environment. As indicated in the LA plan overview (TSPA-VA, Volume 4), the repository safety strategy, based on investigation of the Yucca Mountain site, has identified four key attributes that are important to meeting the postclosure performance objectives. The four attributes are (1) limiting water contacting waste packages, (2) long waste package lifetime, (3) low rate of release of radionuclides from breached waste packages, and (4) radionuclide concentration reduction during transport from the waste packages. These key attributes can be associated with 19 principal factors that are important to the postclosure performance. The four key attributes and 19 principal factors are indicated on Table 5-3. As part of the November 1998 meeting, the grouping team has determined that the inclusion of DOE SNF in the repository will affect only very few of the principal factors identified by RW's repository safety strategy.

DOE SNF will have a different dissolution rate as compared with the commercial SNF and HLW, as well as possible differences in Neptunium solubility, colloid formation and transport through and out of the engineered barrier system. In addition, due to a general lower thermal output, DOE SNF may have some very minor impacts to water contacting the waste packages. However, with the small quantity of

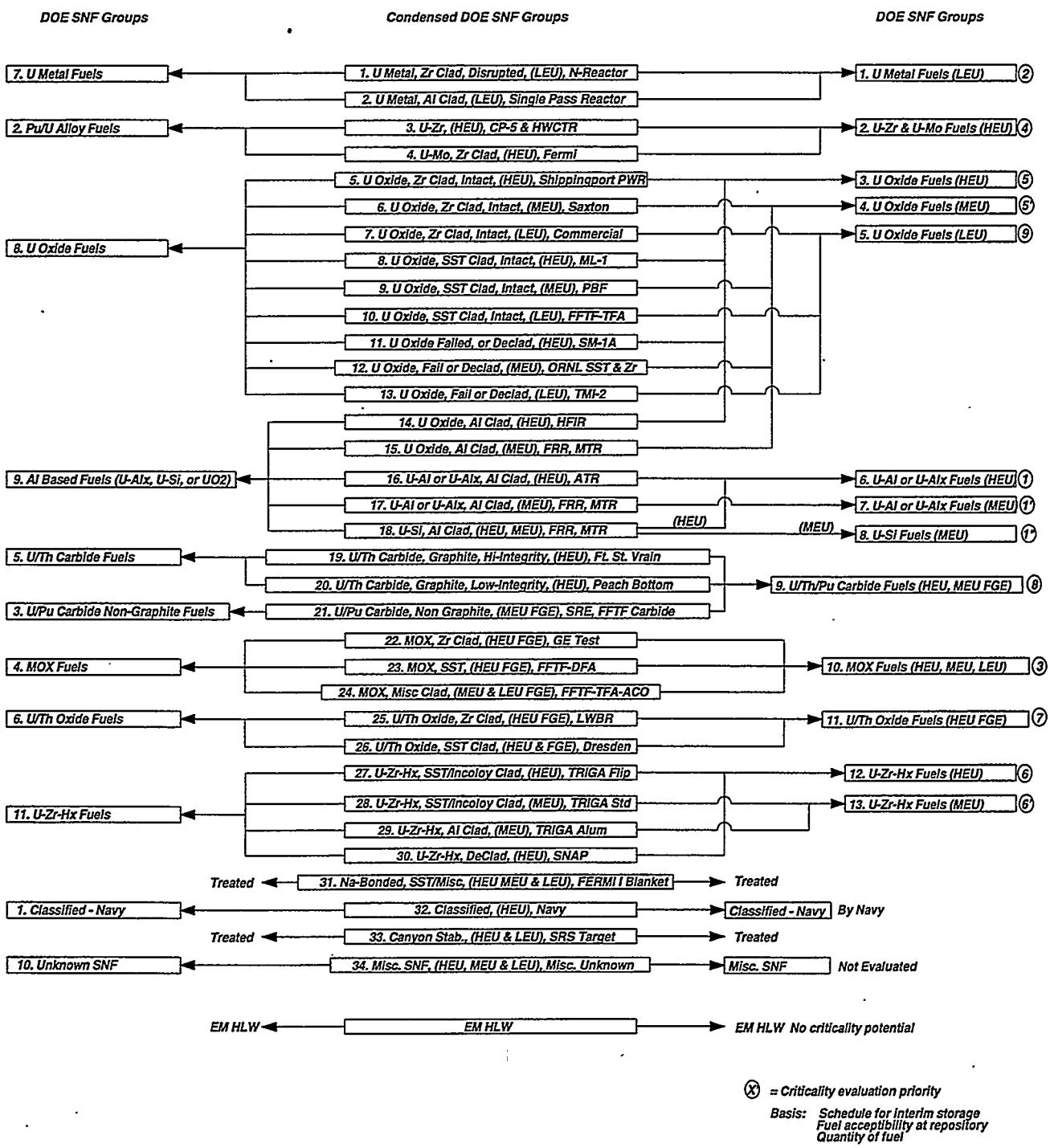

Figure 5-2. DOE SNF condensed groups, groups for TSPA-SR/LA, and criticality analyses in FY 1999.

Table 5-2. DOE SNF groups used in the TSPA-SR/LA in FY 1999.

Fuel Group	Fuel Matrix	Typical Fuel in the Group	Comment
1	Classified	Navy [151][Note below]	Info by Navy
2	Pu/U alloy	FERMI Core 1 and 2 (Standard fuel subassembly) [456]	
3	Pu/U carbide	FFTF-TFA-AC-3 [319]	
4	MOX and Pu oxide	FFTF-DFA/TDFA [71]	
5	U/Th-carbide	FSVR [86]	
6	U/Th oxide	Shippingport LWBR Reflect. IV [371]	
7	U-metal	N-Reactor fuel [147]	
8	U oxide	TMI-2 core debris [229]	
9	Aluminum-based fuel (UAlx, U_3Si_2, U oxide in Al)	FRR pin cluster U_3Si_2-leu Cananda [660]	
10	Unknown	Miscellaneous fuel [366]	
11	U-Zr-Hx	TRIGA (Aluminum) [235]	

Note: The number in [ ] is the fuel identification used in the DOE SNF Database, Version 3.4.0.

DOE fuels, the thermal differences should be negligible. These principal areas for which DOE fuel may have an affect have been shaded in yellow on Table 5-3.

In addition, the grouping team evaluated the parameter and properties of the DOE SNF important to performance, as well as the performance period they affect for the criticality, DBE, and TSPA-SR/LA analyses. The parameters and properties important to performance are indicated on Figure 5-3. Both Table 5.3 and Figure 5-3 suggested that fuel matrix, fuel cladding, fuel condition, fuel enrichment, and burnup should be considered in fuel grouping purpose. For the postclosure performance period, the grouping team concluded that fuel matrix would be the only parameter needed to group DOE SNF to support postclosure analyses. Figure 5.4 summarizes the reasons for not including the other parameters in the PA grouping consideration. Analyses completed as part of the TSPA-VA have demonstrated that DOE SNF can be represented by a surrogate spent fuel with properties like the U-metal fuel [DOE 1998a[1]]. Additional sensitivity analyses will be conducted in the SR and LA to show that this continues to be true. Thus, for the base case TSPA, a U-metal surrogate will be used to represent all DOE SNF (except the Naval fuel).

Based on the above, the methodology used in the development of DOE SNF groups for PA is shown in Figure 5-5. Using fuel matrix as the variable, all the DOE SNF could be placed into 11 groups as indicated on Table 5.2. Discussion of each DOE SNF group is presented in Section 5.3 below.

Table 5-3. RW's key attributes and principal factors affecting repository performance.

Attributes of the repository safety strategy	Principal factors affecting postclosure repository performance	Does DOE SNF have an affect on the principal factor
Limited water contacting waste package	Precipitation and infiltration into the mountain	No
	Percolation to depth	No
	Seepage into drifts	No
	Effects of heat and excavation on flow	Negligible
	Dripping onto waste packages	No
	Humidity and temperature at waste packages	Negligible
Long waste package lifetime	Chemistry of water on waste packages	No
	Integrity of outer carbon steel waste package barrier	No
	Integrity of inner corrosion resistant waste package barrier	No
Low rate of release of radionuclides from breached waste packages	Seepage into waste packages	No
	Integrity of spent fuel cladding	No, no cladding credit for DOE SNF (except Navy)
	Dissolution of spent fuel and glass waste forms	Yes
	Neptunium solubility	Possible
	Formation and transport of radionuclide-bearing colloids	Possible
	Transport through and out of engineered barrier system	Possible
Radionuclide concentration reduction during transport from waste packages	Transport in the unsaturated zone	No
	Flow and transport in the saturated zone	No
	Dilution from pumping	No
	Biosphere transport and uptake	No

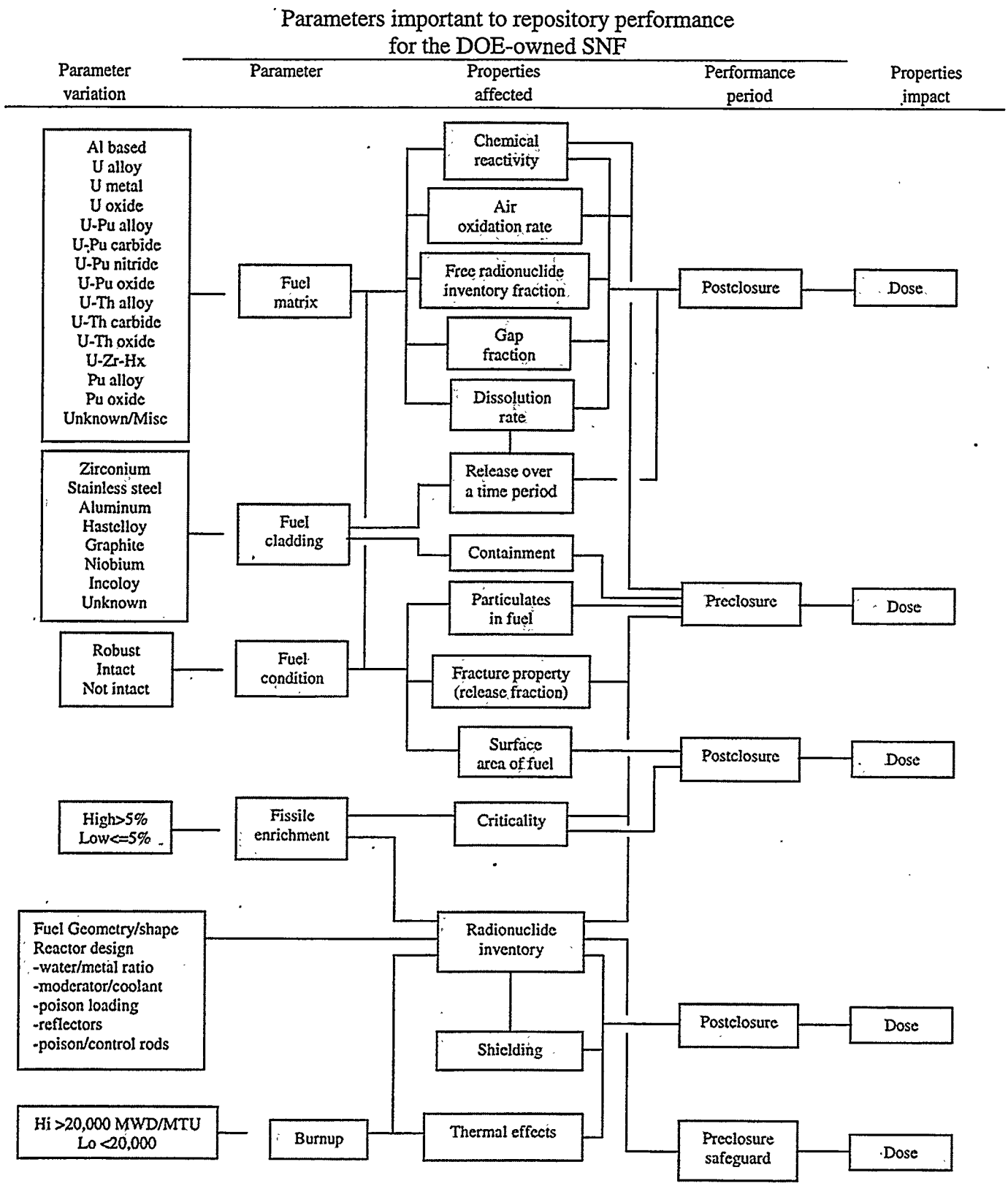

Figure 5-3. DOE SNF parameters and properties for criticality, DBE, and TSPA-SR analyses.

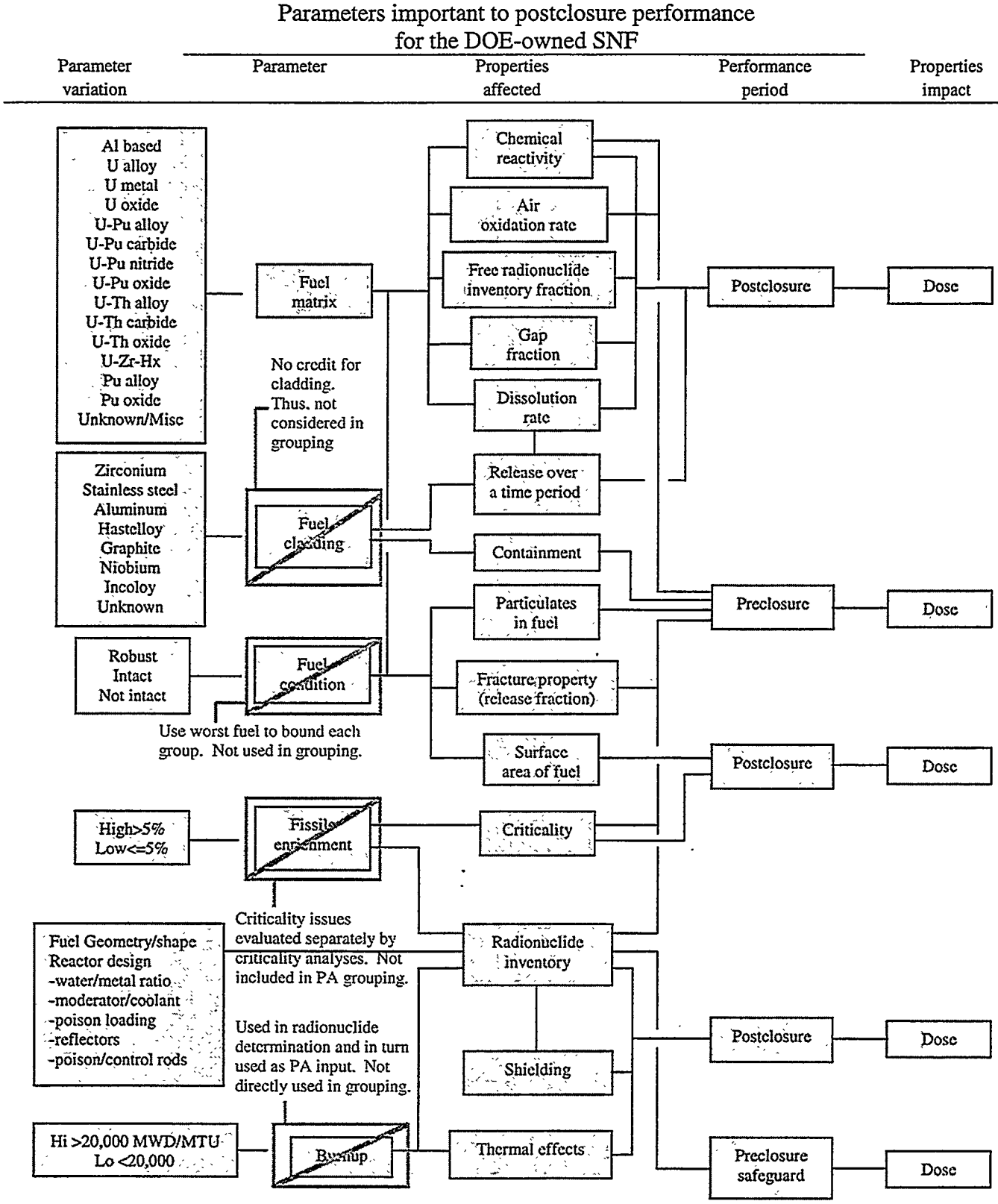

Figure 5-4. DOE SNF parameters used in grouping DOE SNF for TSPA-SR analyses.

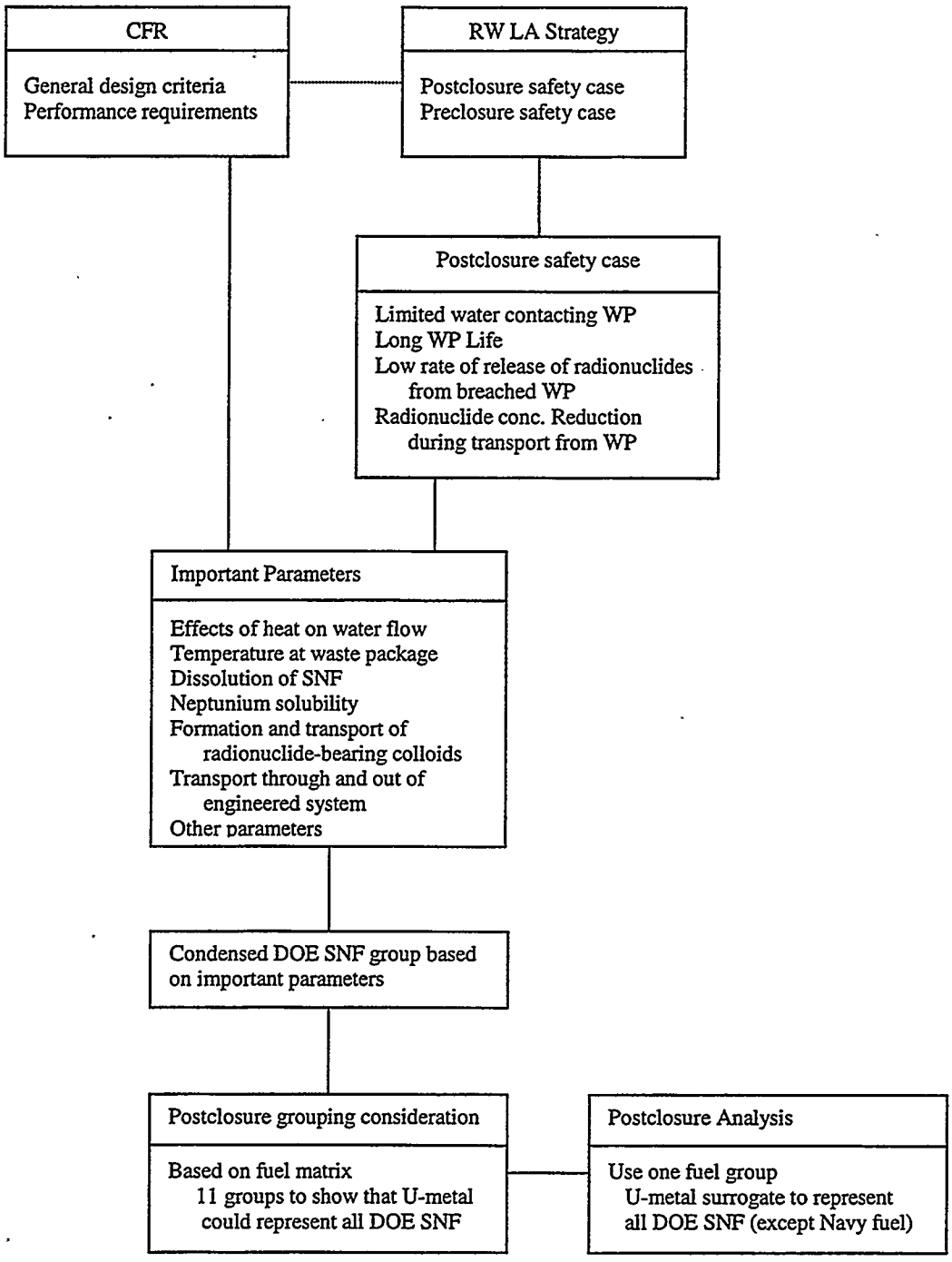

Figure 5-5. DOE SNF postclosure grouping methodology.

5.3 DOE SNF Grouping Basis

DOE has more than 200 varieties of SNF. Based on the discussion in Section 5.2, for the purpose of postclosure safety, all the DOE SNF may be placed into 11 performance assessment groups. The following sections briefly describe each of the 11 groups.

5.3.1 Grouping Discussion

As indicated previously, 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 sections provide a short discussion of the DOE SNF in each of the 11 groups.

Fuel Grouping for Performance Assessment

TSPA Groups—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 dissolution 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 dissolution, 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 have 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 and discussions covered in Section 5.2, the following fuel groups were determined to be appropriate for the purpose of TSPA evaluations. A DOE SNF release rate test program is in progress to confirm that the groups selected are appropriate.

Group 1 - Classified Navy: Because of the classified nature of the Navy fuel, it was placed in its own group and all information concerning this group will be provided by the Navy and will not be addressed here. Refer to Section 5.1 for discussions concerning Navy fuel.

Group 2 - Plutonium/Uranium Alloy Fuels: The Pu/U alloy fuels are placed into this group because of the alloy microstructure and its effects on grain boundary attacks, stress fractures, and crazing. Take U-Zr alloy fuel as an example, 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 also stabilize the uranium metal and thus this group could perform differently than the U-metal fuels. On the other hand, a study on unirradiated U-Mo fuels indicated that uranium alloyed with 10 wt% molybdenum corroded at only 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.

The center fuel section of the Fermi driver fuel (U-Mo alloy fuel type) subassembly makes up over 40% in metric tons of heavy metal (MTHM) of the Pu/U alloy fuel group. 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 wt% molybdenum. The zirconium-clad Heavy Water Components Test Reactor (HWCTR) driver assemblies (U-Th alloy fuel type) makes up ~24% of the MTHM of the Pu/U alloy fuel group. Enrichments are typically 80% ^{235}U. The uranium in the U-Th HWCTR assemblies is alloyed

with over 99 wt% thorium. The Annular Core Research Reactor (ACRR), uses a U-Zr alloy fuel, expects to generate ~26% of the MTHM of the Pu/U alloy fuel group by the year 2035. Enrichment is expected to be about 12% ^{235}U.

Group 3 - Plutonium/Uranium Carbide Fuel: This group consists primarily of fuels from the Fast Flux Test Facility (FFTF). The FFTF fuels are either UC pellet or UC spheres with helium or sodium bonded between the fuel and clad. It is uncertain as to the performance of the carbide particles as compared to the FSV fuels. Thus, this fuel was placed into its own group. The release rate of this group may be 100 times the pure U-metal fuel. Effective enrichments (including the ^{239}Pu) vary from about 10 to 18% ^{235}U.

Group 4 - 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 group. Since the fuels were fabricated using similar techniques as the commercial oxide fuels, performance of the MOX fuels should be very similar. Because of the high plutonium content as compared to the U-oxide fuel, this fuel was placed into its own group.

Group 5 - Uranium/Thorium Carbide Fuel: This group primarily consists of fuel from the FSV reactor and fuels from Core 1 and 2 of the PB reactor. A small amount of fuel from the General Atomic Gas-Cooled Reactor is also included in this group. 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. The fuel particles in the FSV and PB Core 2 fuel assemblies are in excellent condition. However, the fuel particles in the PB Core 1 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 group may be 10 times the U-metal rate because of the possible water/carbide reaction. Effective enrichment (including the ^{233}U) level at the end of life varies from about 78 to 83% ^{235}U.

Group 6 – Uranium/Thorium Oxide Fuel: Shippingport LWBR fuels make up the major inventory of the fuel in Group 6. 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 dissolve at a different rate than the standard U-oxide fuel and thus this fuel was placed into its own group.

Group 7 - Uranium Metal Fuels: The majority of this group 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.

Group 8 - Uranium Oxide Fuel: This group consists of the fuels removed from commercial reactors or test fuel with uranium oxide matrices similar to commercial spent fuels. In addition, 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 are also included in this group. This group is modeled as performing like the commercial SNFs, 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 release rate for fuels with the same matrix, enrichments from the typical of ~1–2%

commercial range (such as TMI Reactor fuels) to the 93% ^{235}U fuel from the High Flux Isotope Reactor and Shippingport PWR are included in this group.

Group 9 - Aluminum-Based Fuel (Uranium Aluminide Fuel, Uranium Silicide, and Uranium Oxide in Aluminum): This group consists of fuels with the: (1) uranium-aluminide dispersed in a continuous aluminum phase, (2) uranium-silicide dispersed in a continuous aluminum phase, and (3) uranium oxide dispersed in a continuous aluminum phase. This group should 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 (FRR) fuels make up a large part of the aluminum-based fuel. Enrichment level varies from about 11 to 93% ^{235}U with the majority of the silicide fuels having less than 20% ^{235}U.

Group 10 - Unknown Fuel: The DOE fuels with unknown matrices are placed in this group. Because of the potential 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 group. Based on the group inventory, the U-metal dissolution model is believed to well represent the DOE SNF in this group.

Group 11 - Uranium Zirconium Hydride Fuel: Group 11 contains fuel with the uranium/zirconium hydride matrix. Fuels from the TRIGA reactors make up the majority of the fuel in this group. The uranium-zirconium hydride in this group 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. Because of the unique uranium/zirconium hydride matrix, it was placed in its own group. This fuel matrix is expected to perform much better than the standard U-oxide fuel.

Criticality Analyses Groups[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 groups 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 Figures 5-1 and 5-2, 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 group.

Like the TSPA groups, the criticality analyses will not include the sodium-bonded and classified Navy SNF. The sodium-bonded fuel will be treated before 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.

6. SNF DISSOLUTION MODELS

With each group 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 group is discussed below. Two points in the grouping discussion 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 dissolution 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 groups [Fillmore[7]]. The suggested wet dissolution models are presented in Table 6-1. 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.

6.1 Dissolution Model for Group 1 — Classified Navy Fuel

Because of the classified nature of the Navy fuel, it was placed in its own group and all the dissolution information concerning this group will be provided by the Navy and will not be addressed here. See Section 5.1 for a brief discussion of Navy fuels.

6.2 Dissolution Model for Group 2 — Plutonium/Uranium Alloy Fuels

There are a number of different kinds of Pu/U alloy fuels in the DOE SNF inventory. The Fermi Core 1 and 2 fuels make up nearly 40% (in terms of MTHM) of this group. 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. Yemel'yanvo and Yevstyukhin also indicated 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 [Yemel'yanvo[8]].

A paper published by Waber in 1958 also covers the corrosion properties of various uranium alloys [Waber[9]]. 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

Table 6-1. DOE SNF wet dissolution models.

Fuel group	Fuel matrix	Typical fuel in the group	Wet dissolution model
1	Classified-Navy	Navy	Model by Navy
2	Pu/U alloy	FERMI Core 1 and 2 standard fuel assembly (Enrico Fermi Reactor) Fuel	10x U-metal model
3	U or U/Pu carbide	Fast Flux Test Facility (FFTF-TFA-AC-3) carbide fuel	100x U-metal model
4	MOX	Fast Flux Test Facility (FFTF DFA/TDFA) oxide fuel	Commercial model
5	U/Th carbide	Fort St. Vrain (FSV) fuel	10x U-metal model
6	U/Th oxide	Shippingport LWBR fuel	Ceramic model
7	U-metal	N-Reactor fuel	U-metal model
8	U-oxide	Three Mile Island (TMI) fuel Shippingport PWR fuel	Commercial model
9	Al-based fuel	Foreign Research Reactor (FFR) fuel	0.1x U-metal model
10	Unknown SNF	Unknown fuel	Instantaneous
11	U-Zr-Hx	Training Research Isotopes — General Atomic (TRIGA) fuel	0.1x Commercial model

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.

Besides the zirconium-clad Fermi Core 1 and 2 driver, sectioned, sodium worth, core foil (Fermi Core), and HWCTR U-Mo assemblies make up the rest of the U-Mo alloy fuels. The driver fuel makes up about 39% 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 before final disposal. The driver fuel enrichments are typically about 25% ^{235}U. The uranium in the Fermi driver center fuel sections is alloyed with 10 wt% molybdenum [LUNFIS[10]].

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 zircaloy sleeve by coextrusion of the fuel alloy slugs in Zr tubing 1,600 °F at which time a metallurgical 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. [LUNFIS[10]].

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 6-1. 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.

Uranium-zirconium alloy fuels make up the next largest quantity of the Pu/U alloy 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 [Yemel'yanvo[8]]. 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 [Bauer[11]].

Uranium-zirconium fuels, which make up over 30% in MTHM in this group, consist of stainless steel-cladded ACRR assemblies. Enrichments are typically 12% ^{235}U. The uranium in the ACRR driver assemblies is alloyed with 90.7 wt% zirconium [Directory[12]]. A plot of the Bauer corrosion data on 90.7 wt% zirconium and 9.3 wt% uranium is shown in Figure 6-2. To provide a comparison with the U-metal corrosion rate, all the U-metal reaction data have been included for reference.

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 6-2, 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.

Similarly, 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 Pu/U alloy corrosion at this time in the TSPA-SR/LA. As better Pu/U alloy information becomes available, it will be included in the licensing application process.

6.3 Dissolution Model for Group 3 —Plutonium/Uranium Carbide Fuel

This group consists primarily of fuels from the FFTF (over 70% by MTHM). The Sodium Reactor Experiment (SRE) fuel makes up the rest of the group. 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 group 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 [Tripler[13]]. This group, as indicated by Tripler, may perform much worse than the pure U-metal fuel. For the purpose of TSPA-SR/LA, Fillmore suggested that 100 times the uranium metal dissolution rate be used to conservatively represent this group [Fillmore[7]].

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

6.4 Dissolution Model for Group 4 — 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 group (over 83% by MTHM). Cleveland, in the *Plutonium Handbook,* reported that PuO_2 prepared at high temperature dissolved very slowly even in HNO_3-HF acid solutions [Cleveland[14]]. Sasahara reported a low fission gas release rate from MOX fuel was 1.7% as compared to the UO2 fuel at 4.8% [Sasahara[15]]. 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 conservatively represent the MOX fuel in the TSPA-SR/LA analysis.

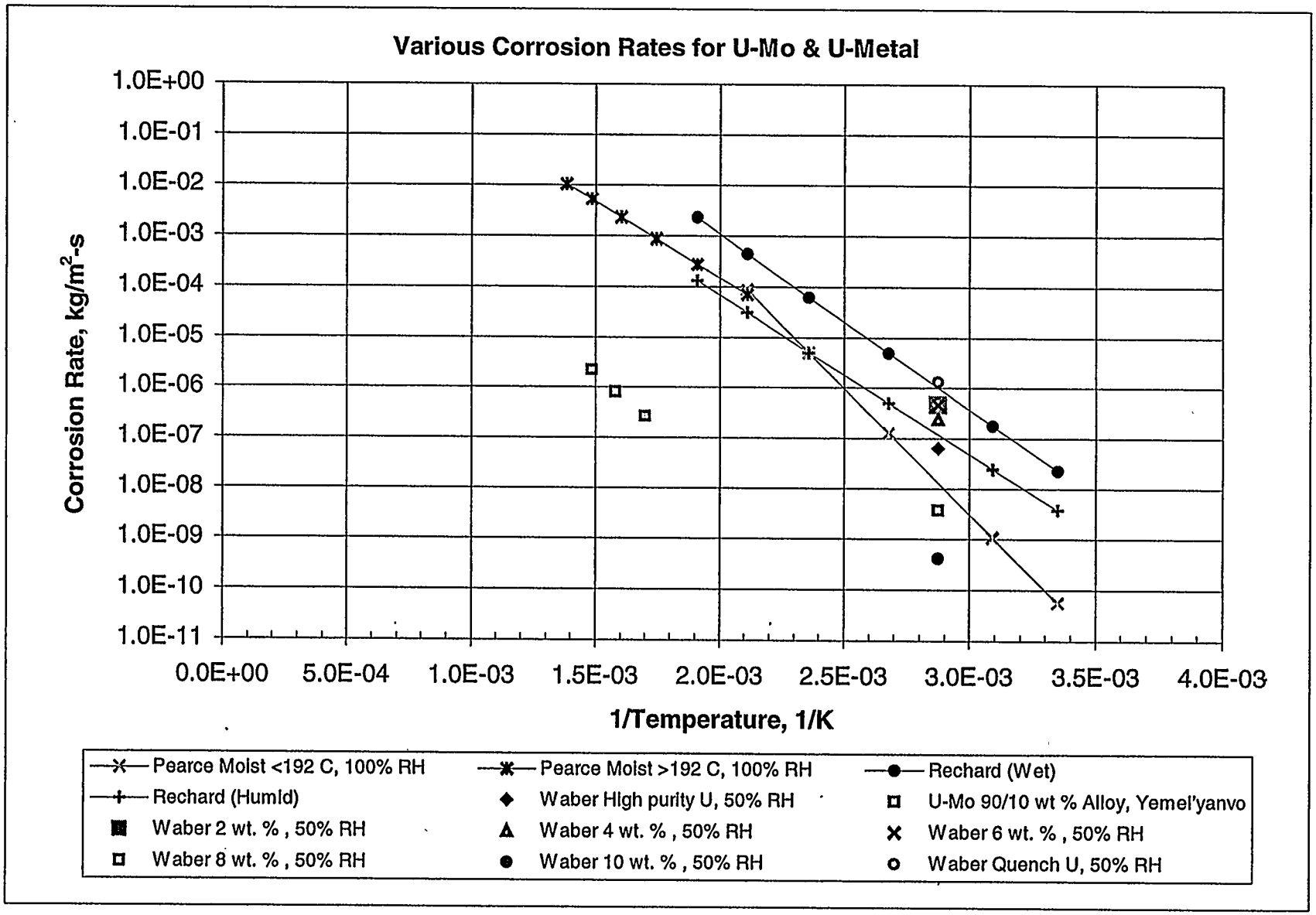

Figure 6-1. U-Mo and U-metal corrosion rate plotted with temperature.

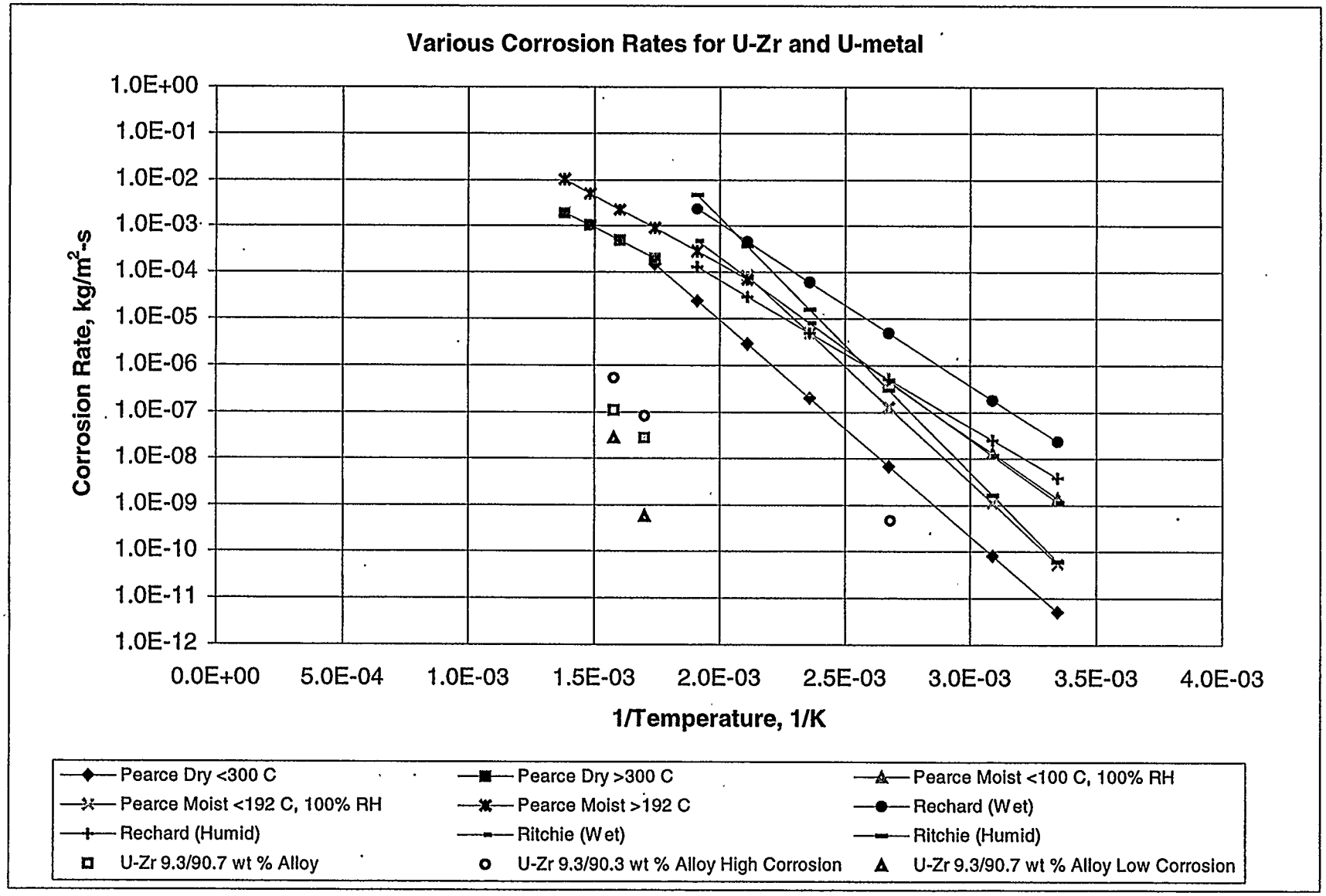

Figure 6-2. U-Zr and U-metal corrosion rate plotted with temperature.

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.

6.5 Dissolution Model for Group 5 — Uranium/Thorium Carbide Fuel

Fuel from the FSV reactor makes up ~90% (in terms of MTHM) of this group. The fuels from the Core 1 and 2 of the PB reactor and a small amount of fuel from the General Atomic Gas-Cooled Reactor make up the rest of this group. 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 the FSV and PB fuels is that the PB particles lack the silicon carbide coating. The fuel particles in all of the FSV 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 [Tripler[13]]. 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 [Yemel'yanvo[8]].

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 [Lotts[16]]. He reported that the graphite oxidation rate is extremely slow and estimated that it will take 3.6×10^9 years to oxidize 0.5 cm of graphite and will take only 5×10^5 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 [Rechard[17]] for both wet oxic and humid oxic conditions. The proposed equation is:

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

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 dependence at repository conditions)

C = 1, (linear corrosion kinetics)

D = 1, which is assumed to be conservative

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

The rest of this group consists of fuels from the Core 1 of the PB reactor. The fuels are also in the form of carbide particles coated with pyrolytic carbon, bonded together by a carbonaceous matrix material. 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 [Fillmore[7]].

No repository data are available on the dissolution rate of this group. 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 for the entire group due to the breached particles in the PB Core 1 fuel assemblies [Fillmore[7]]. Figure 6-3 is a plot of the silicon carbide and U-metal corrosion rates. Ten times the U-metal dissolution rate appears to be conservative with the large quantity of intact FSV fuel elements.

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.

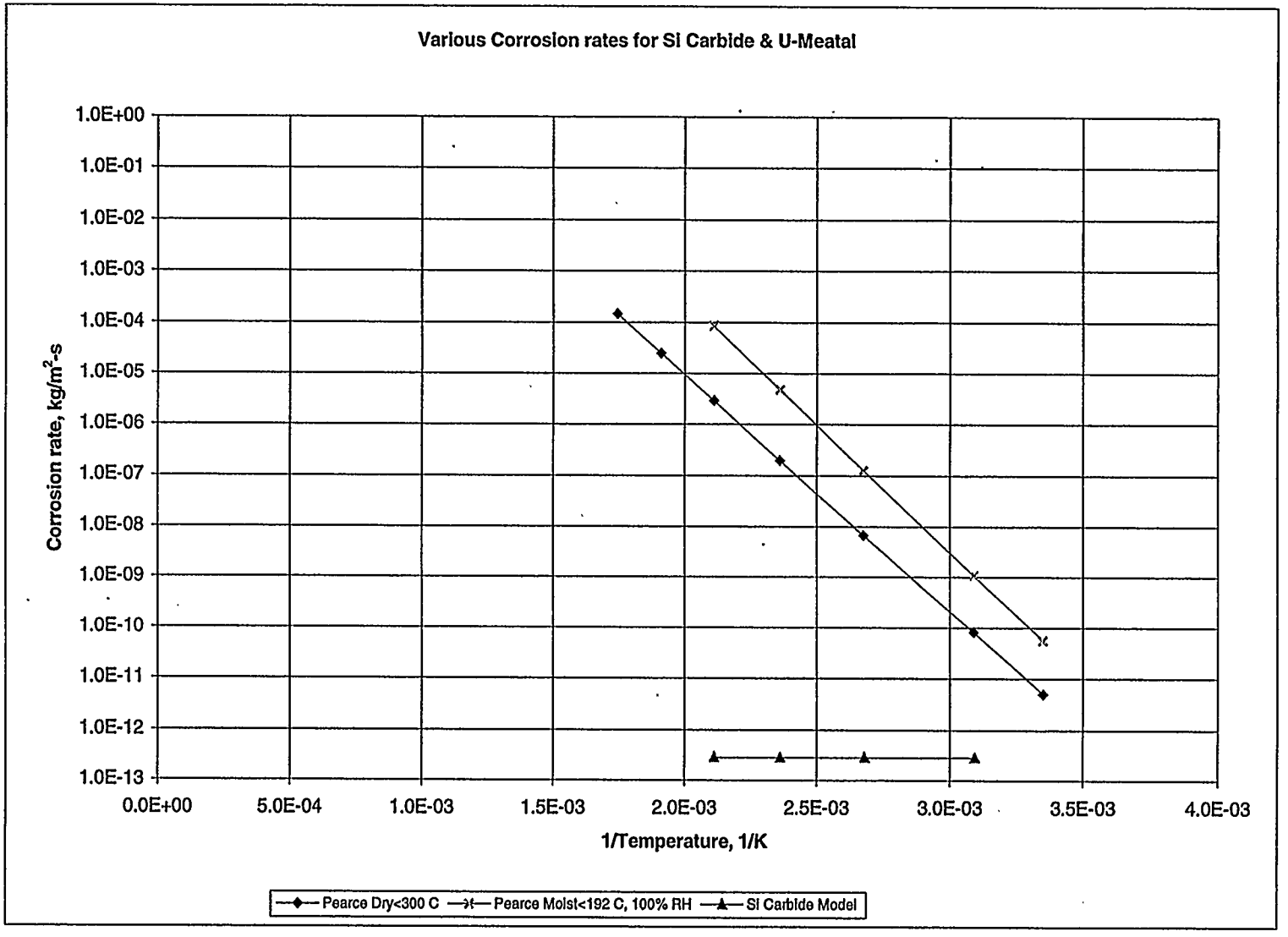

Figure 6-3. SiC and U-metal corrosion rate plotted with temperature.

6.6 Dissolution Model for Group 6 — Uranium/Thorium Oxide Fuel

Shippingport LWBR fuels make up the major inventory of the fuel in Group 6; specifically, it makes up over 86% of the group's inventory by MTHM. The remainder of the fuels in the group 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 group. 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 [Clayton[18]]. 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 [Brookins[19]].

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 [Duguid[20]]. The proposed ceramic model is indicated as Equation 2 below. The results from the BAPL report are plotted in Figure 6-4. 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.

In a letter report, Lappa suggested that Equation (2) be used to represent ceramic materials [Lappa[21]].

$$Q = Q_0 + \theta + SFt / A \qquad (2)$$

Where:

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

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

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

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

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

A = 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 (2). 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 [Ringwood[22]]. The Ringwood equation is indicated below.

$$R = \alpha 10^{-\beta(1000/T)} \tag{3}$$

Where:

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

T = temperature (°K)

α, β = constants ($\alpha = 0.082$, $\beta = 1.0$ based on available data).

The leaching rate for the Ringwood equation is also plotted in Figure 6-4 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-SR/LA, the Ringwood equation was selected to conservatively model the dissolution rate of Group 6. As better information becomes available, this model will be updated accordingly.

6.7 Dissolution Model for Group 7 — Uranium Metal Fuels

The zirconium-clad N-Reactor fuels, with a small amount of aluminum-clad Single Pass Reactor fuel, make up this group. The N-Reactor fuel elements consist of two concentric tubes made of uranium metal coextruded 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 [Hanford[23]]. The fuel's preirradiation ^{235}U enrichment is below 2%. Appendix A.1.1 presents a more detailed description of this group.

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. [Wilkinson[24]]. Ritchie performed similar data reviews during the 1980s [Ritchie[25,26]]. 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 [Pearce[27]]. 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 [Ballinger[28]].

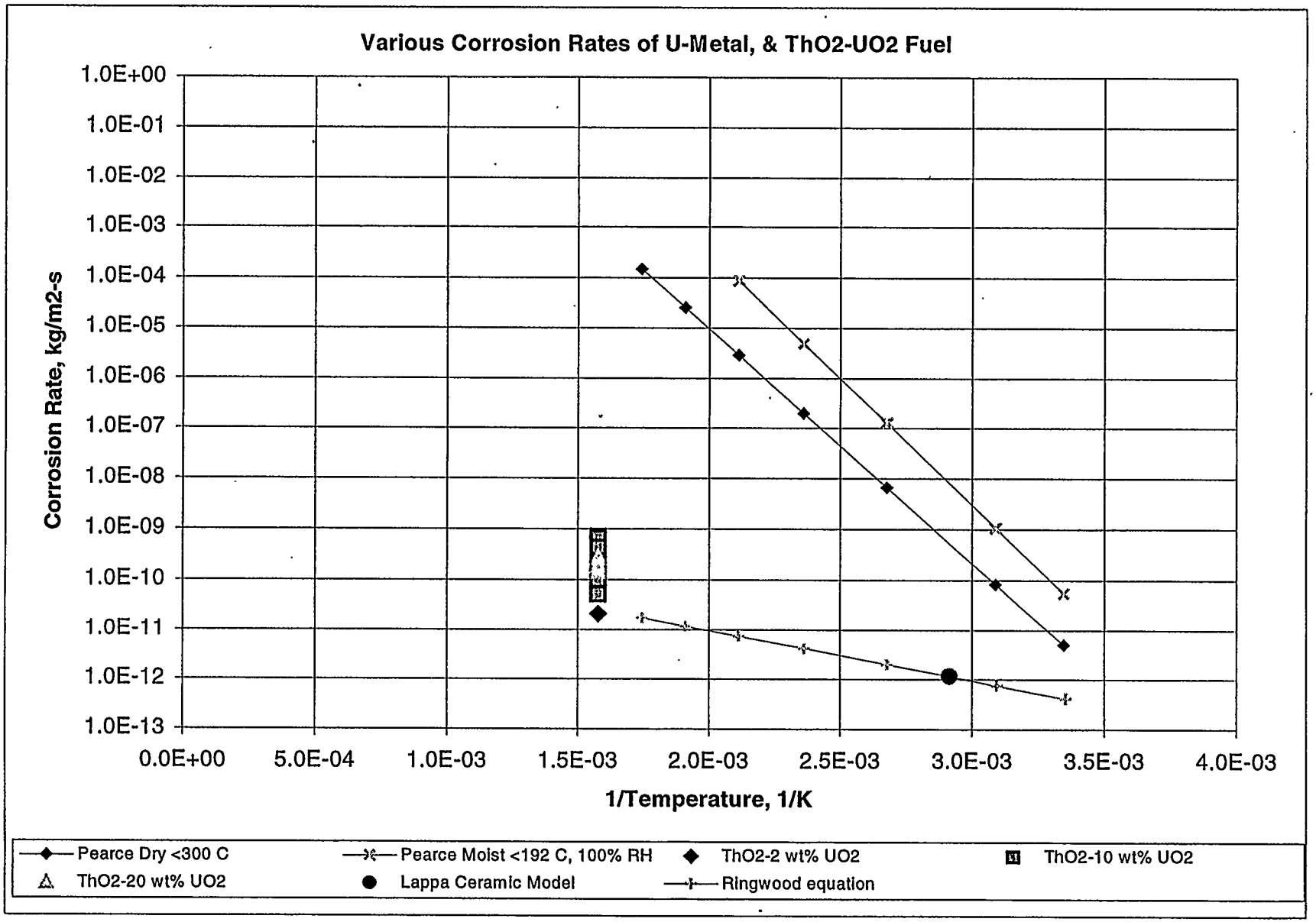

Figure 6-4. Th/U Oxide, and U-metal Corrosion Rate Plotted with Temperature.

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 increases), an equation of generalized corrosion was also proposed to represent the dissolution of the DOE SNF in an unsaturated Tuff repository by Rechard [Rechard[17]]. This generalized equation is Equation (4):

$$M = A \cdot e^{-B/T} \cdot (t_2^C - t_1^C) \cdot D \cdot E \cdot SA \qquad (4)$$

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
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^{2}s 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
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 6-5 [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 6-5, 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 before canisterization and the 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-SR/LA 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-SR/LA base case, a single DOE SNF fuel type represented the entire DOE SNF inventory. Based on the 1997 TSPA sensitivity analysis of DOE SNF [Duguid[20]], 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.

6.8 Dissolution Model for Group 8 — Uranium Oxide Fuels

This group 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 ~178 MTHM, over 81 MTHM (breached fuels) come from commercial reactors such as TMI. Over 67 MTHM (intact fuels) comes from commercial reactors such as Ginna operated by the Rochester Gas and Electric and Surry 2 operated by Virginia Power. Because of similar construction, a very small quantity of uranium beryllium oxide fuels is included in this group. This group 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 [Fillmore[7]]. Another potential difference is that some of the fuels in this group 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.

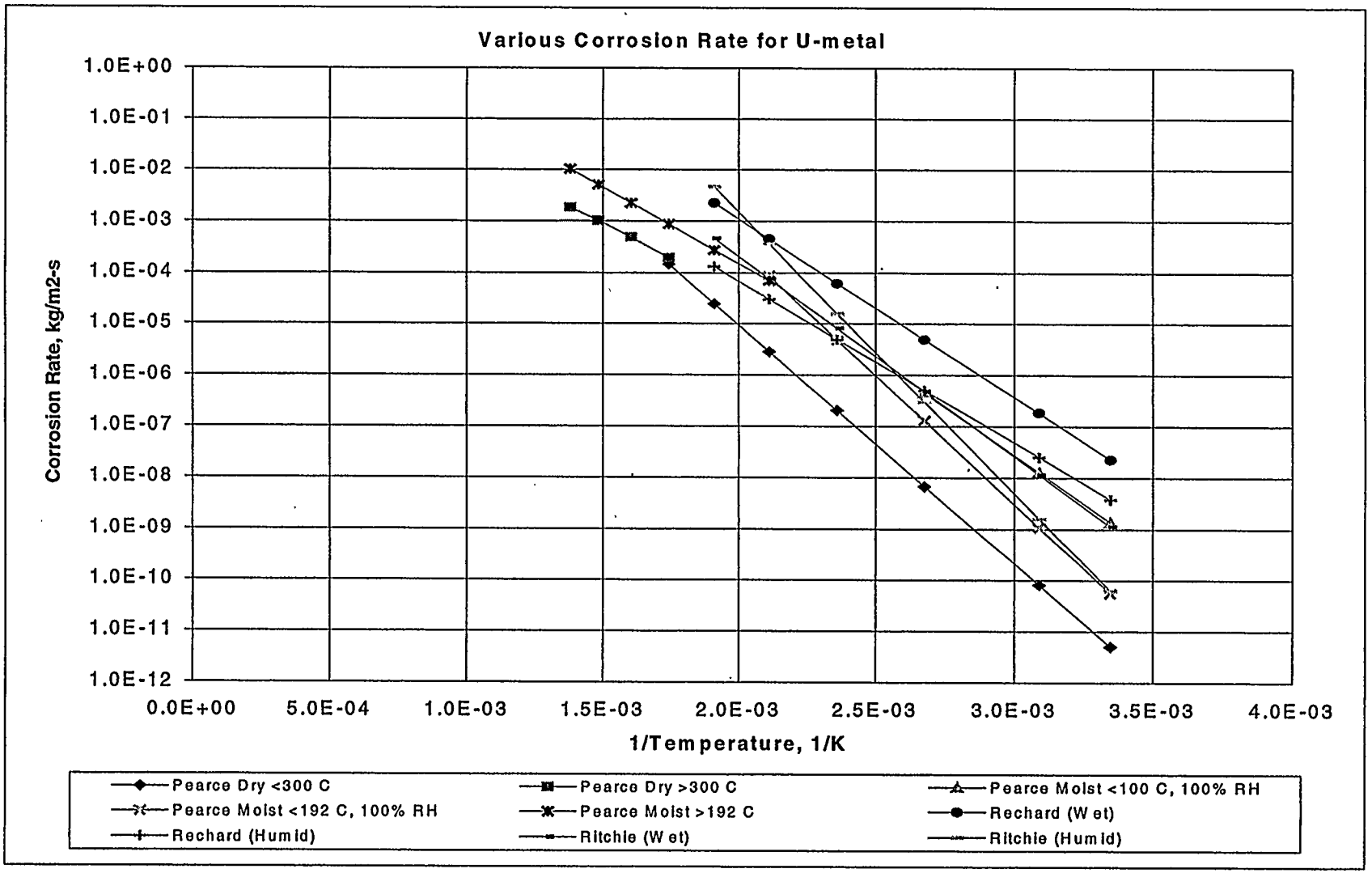

Figure 6-5. Various U-metal reaction rate equations plotted with temperature.

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 flowthrough 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 [Stout[29]] 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-SR/LA to conservatively bound the commercial SNF with burnup >30,000 MW days/kgU:

$$\log_{10}(Rate) = a_0 + [a_1 \cdot ((1/T) - c_1)] + [a_2 \cdot (\log_{10}(CO_3) - c_2)] + [a_3 \cdot (\log_{10}(O_2) - c_3)] + [a_4 \cdot (pH - c_4)] + [a_5 \cdot ((\log_{10}(CO_3) - c_2)^2 - c_5)] + [a_6 \cdot ((1/T) - c_1) \cdot (\log_{10}(O_2) - c_3)] \quad (5)$$

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-SR/LA are as follows:

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, the intact fuel in this group could be conservatively modeled as performing like the commercial SNFs and the commercial dissolution model could be used to represent the dissolution rate of Group 8.

However, this group also includes the fuels removed from commercial reactors or test fuels with uranium oxide matrices that have been damaged, experienced failed cladding, or are decladded. As indicated earlier, over 81 MTHM are from commercial reactors such as TMI Reactor fuels. This group contains enrichments from the typical 1–2% commercial ranges (such as TMI Reactor fuels) to the 93% fuel from the High Flux Isotope Reactor. With the disrupted fuels from TMI, to be conservative, they should be modeled as performing like the commercial SNFs, 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.

For TSPA-SR/LA, this group is conservatively modeled as performing like the commercial spent nuclear fuels and the commercial dissolution model was used to represent the dissolution rate. However, 100 times the commercial SNF surface area was used to represent the fuels in this group. The rationale concerning the selection of the fuel surface areas will be discussed in a later section. 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.

6.9 Dissolution Model for Group 9 — Aluminum-Based Fuels

This group consists of fuels with the uranium-aluminide, uranium silicide, and uranium oxide particles dispersed in a continuous aluminum phase. Fuels from FRRs make up over 36% in MTHM of the uranium silicide fuels in this group. The ATR fuel makes up over 11% in MTHM of the uranium-aluminide fuel in this group. Enrichment level varies from about 8 to 93%. The aluminum-based 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 [Sindelar[30]].

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 [Sindelar[30]]. 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 6-6 to represent the corrosion rates of the Al-10 wt% U and aluminum cladding alloys.

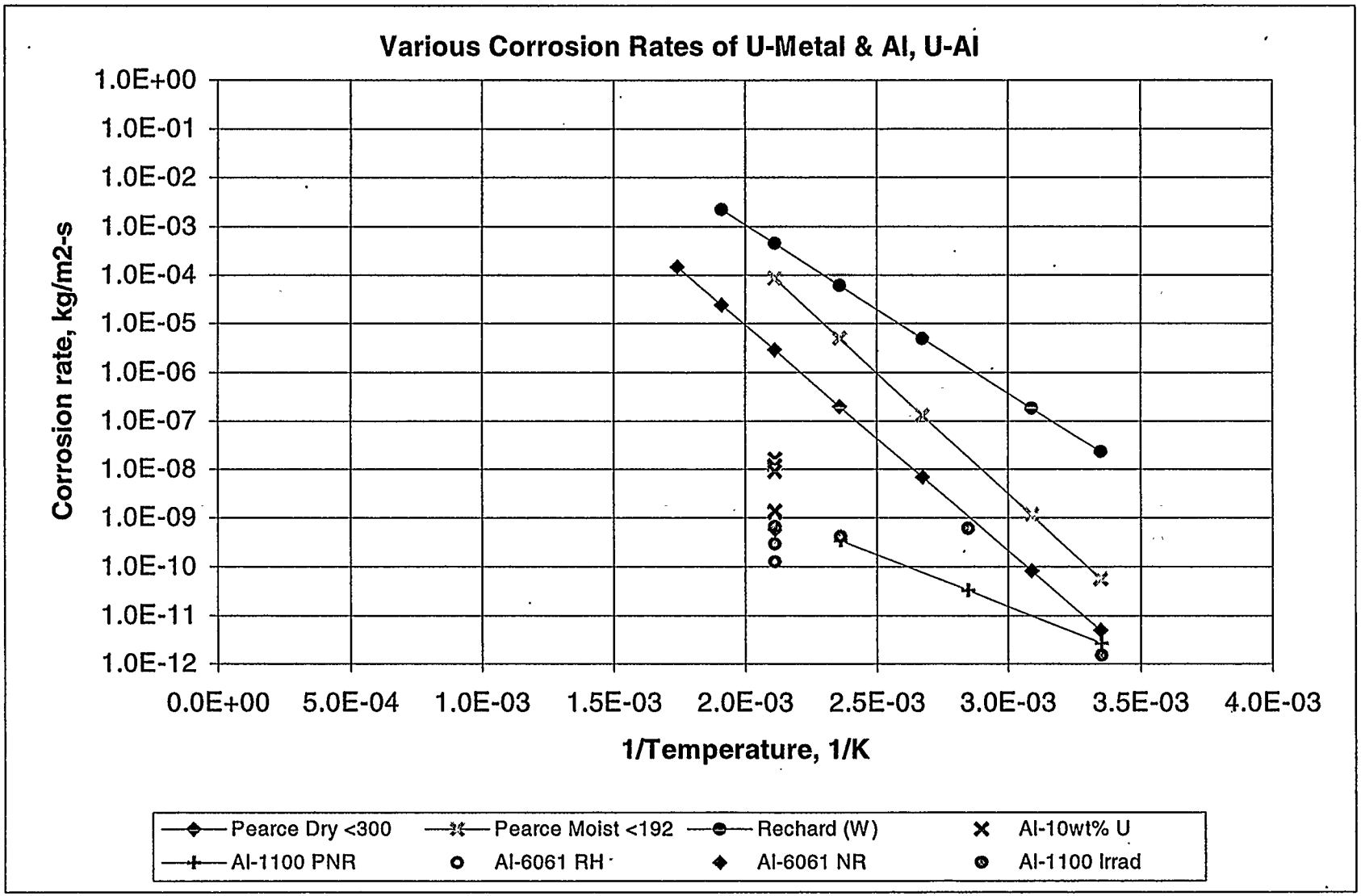

Figure 6-6. Al, U-Alx, and U-metal corrosion rate plotted with temperature.

Since the SRS reported that the 1100 and 6061 Al alloys should follow a parabolic corrosion behavior [Sindelar[30]], 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 Group 7 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.

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 dissolution information becomes available, it will be incorporated for use in the license application TSPA analyses.

The aluminum-based fuel also consists of fuel with uranium-silicide dispersed in a continuous aluminum phase. Foreign research reactor (FRR) fuels make up a large part of the U-Si fuels in this group. Enrichment level varies from ~8 to ~93%. But the majority of the fuels' enrichment is less than 20%. Wilkinson [Wilkinson[24]] 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 [Faraday[31]] 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 [Snyder[32]] 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, and 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 6-7 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. 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.

For the purpose of the TSPA-SR/LA analyses, 0.1 times the U-metal dissolution model was selected and used to bound Group 9. Further testing will have to be done on the aluminum-based fuel material. Flowthrough testing is completed and drip testing will be added to the FY 1999 release rate program. As the results of the release rate program become available, the aluminum-based fuel dissolution model will be revised and implemented into the repository license application.

6.10 Dissolution Model for Group 10 — Unknown Fuel

The DOE SNFs with unknown matrix are placed in this group. In addition, the very small quantity of uranium nitride fuel was also placed in this group. Because of the potential of varying matrices, cladding, and condition of this group of fuel, Fillmore suggested that this fuel group be bounded by the fuel properties of the U-metal DOE SNF [Fillmore[7]]. Note that this group makes up less than 0.2 % of the total DOE SNF inventory based on MTHM.

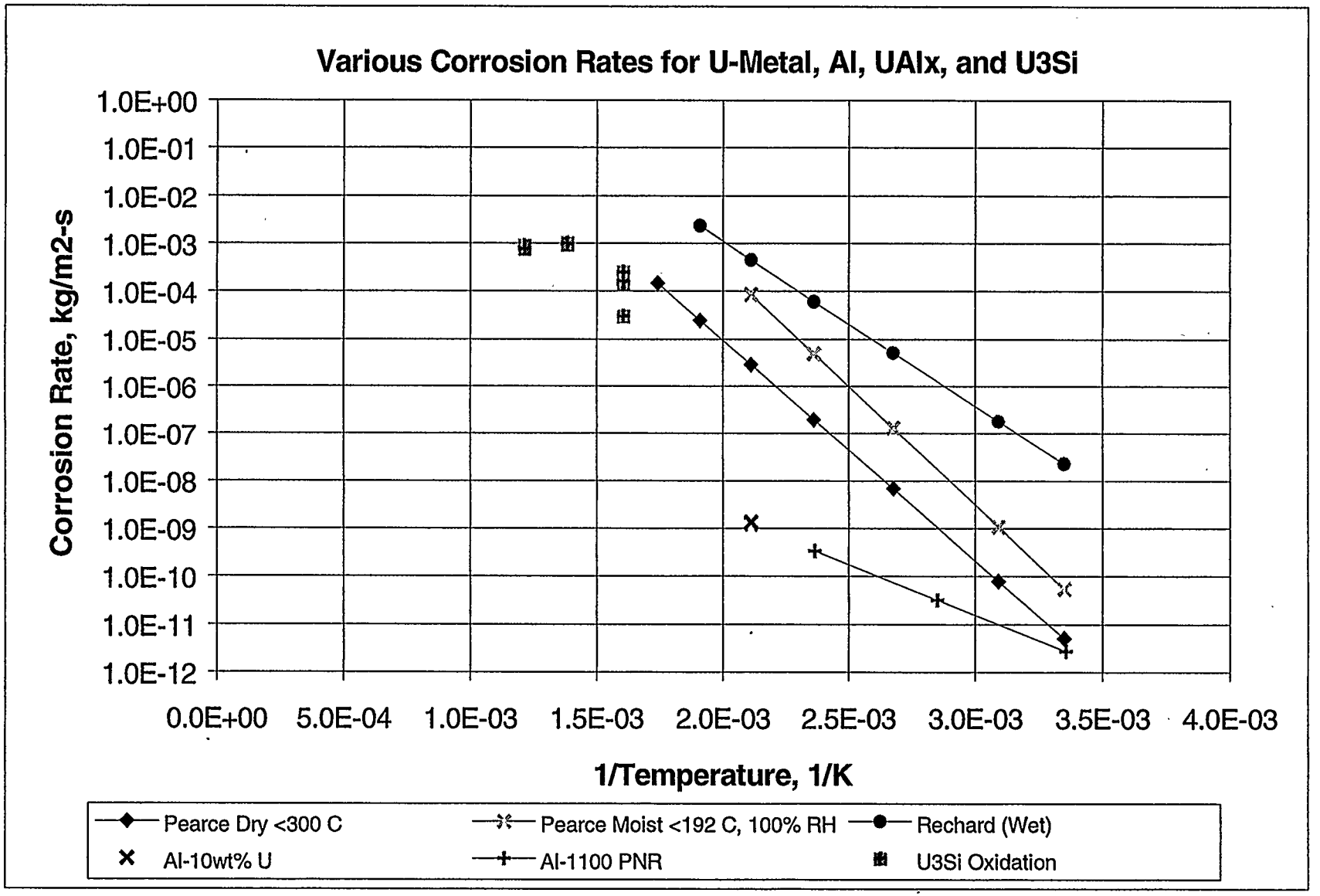

Figure 6-7. U_3Si Al, U-Alx, and U-metal corrosion rate plotted with temperature..

6.11 Dissolution Model for Group 11 — Uranium Zirconium Hydride Fuel

Group 11 contains the fuel with the uranium/zirconium hydride matrix. Fuels from the TRIGA reactors make up over 97% of the fuel in this group 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 group 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. Because of the unique uranium/zirconium hydride matrix, it was placed in its own group. 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 group.

7. OTHER DOE SNF PROPERTIES

A number of other parameters are needed in the TSPA-SR/LA models to predict the performance of the materials placed into the potential repository. The basis and/or derivation of each of these parameters are indicated in the following sections. Some properties were not readily available at the time of the TSPA-SR/LA 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.

7.1 DOE SNF Surface Area

The surface area for each DOE SNF group is derived in an engineering design file number EDF-NSNF-005 titled "Fuel Surface Area Calculation". 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 group was assumed to be the same. Where different geometries or dimensions exist in the same group, a dominant type was selected or average values were calculated for the entire group.

7.2 DOE SNF Volume

The volume for each DOE SNF group is derived in an engineering design file number EDF-NSNF-004 titled "Fuel Meat Volume Calculation". 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.

7.3 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 [Fillmore[7]]. 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 [Fillmore[7]]. Thus, Fillmore suggested for the purpose of the TSPA-SR/LA, the air oxidation rate for DOE SNF should also be set to zero.

7.4 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-SR/LA, Fillmore suggested that conservative estimates be made of the fraction of fuel cladding failed for the DOE SNF. Table 7-1 shows the conservative estimate of cladding failures for each DOE SNF group [Fillmore[7]]. 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 group, the fraction of cladding failed is one.

As shown in Table 7-1, some DOE SNF cladding is in good condition. However, no credit is currently claimed for fuel cladding as a barrier to releases for the DOE at this time.

Table 7-1. DOE SNF fraction of cladding failed.

Fuel group	Fuel type	Fraction of cladding failed, 0-1
1	Navy	by Navy
2	Pu/U Alloy	0.1
3	Pu/U Carbide	0.1
4	MOX	0.1
5	U/Th Carbide	0.6-0.8
6	U/Th Oxide	0.1
7	U-metal	1
8	U-Oxide	1
9	Aluminum-Based Fuel (UAlx, U_3Si_2, U Oxide in Al)	1
10	Unknown Fuel	1
11	U-Zr-Hx	0.1

7.5 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 before 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 before package and canister breach. The nongaseous 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) before packaging for repository disposal. The heating (from drying and conditioning) may release some of the nongaseous 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 7-2. See *DOE SNF Information Report in support of the TSPA-VA in the National SNF Program TSPA-VA* for more discussion [Fillmore[7]]. 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 7-2. DOE SNF free radionuclide inventory.

Fuel group	Fuel type	Free radionuclide inventory, 0-1
1	Navy	by Navy
2	Pu/U Alloy	0.00001
3	Pu/U Carbide	0
4	MOX	0.01-0.06
5	U/Th Carbide	0.1
6	U/Th Oxide	0
7	U-metal	0.001
8	U-Oxide	0.01-0.06
9	Aluminum-Based Fuel (UAlx, U_3Si_2, U Oxide in Al)	0.0001
10	Unknown Fuel	0.001
11	U-Zr-Hx	0.00001

7.6 DOE SNF Gap Inventory

The gap referred to here is between the fuel meat and the cladding. The inventory fraction is the fraction 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 coextruded 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 [Fillmore[7]]. Fillmore's proposed gap inventory fraction is indicated in Table 7-3. 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 7-3. DOE SNF fraction of gap inventory.

Fuel group	Fuel type	Fraction of gap inventory, 0-1
1	Navy	By Navy
2	Pu/U Alloy	0
3	Pu/U Carbide	0.01-0.06
4	MOX	0.01-0.06
5	U/Th Carbide	0.001
6	U/Th Oxide	0.01-0.06
7	U-metal	0
8	U-Oxide	0.01-0.06
9	Aluminum-Based Fuel (UAl_x, U_3Si_2, U Oxide in Al)	0
10	Unknown Fuel	0
11	U-Zr-Hx	0.00001

8. SNF PACKAGES

8.1 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 [Dreyfus[33]].

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 that make it unacceptable for repository dispositioning. 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 [SFD[34]]. 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 codispose the DOE SNF with the HLW in a large disposal package. The following sections describe the how the DOE fuels are packaged for disposal.

8.2 DOE SNF Disposal Configurations

8.2.1 SNF Standard Canisters

The DOE SNF will be placed into a standard SNF canister with approximate diameter of 18 inches (~450 mm) in both 118.1 inches and 179.9 inches (3,000 mm and 4,570 mm) lengths (See reference to the Interface Control Document in Section 9.0). This variety of fuel canister sizes, when placed with the HLW canisters, results in a variety of repository waste package combinations within each fuel group. Generally, fuel types (as determined by the originating reactor) within a fuel group 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 codisposal 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. 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.

A standard high integrity can (HIC) has been developed and may be used to repackage certain DOE fuel. The fuel to be repackaged may be in the form of fine particulate, declad fuel pieces, or test specimen. The HIC is approximate 5 inches in diameter constructed of Hastelloy C-22 thus it is very robust. The can will be built to various lengths to accommodate the variety of DOE fuels. The HIC will be placed into the 18 inches standard canisters for final disposal.

DOE EM, in cooperation 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 2 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-SR/LA 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 [Technical[35]]. These artificial loadings are not intended to be limiting values for any type or group 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

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

Using this proposed fuel loading for aluminum fuel and a number of criticality evaluations completed to date for DOE SNF, 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-SR/LA. 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). This variance will have to be proved acceptable in a criticality safety evaluation for these specific fuels before the licensing application.

HEU (>20%) not to exceed ~30 kg ^{235}U equivalent

MEU (>5%<20%) not exceed ~50 kg ^{235}U equivalent

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

As indicated earlier, the groups or fuel groups for the TSPA-SR/LA consist of one or more fuel types. These types may vary in terms of physical geometry, total mass, enrichment, or burnup. While other groupings may have segregated the fuels by cladding, the categorization of fuels for the TSPA-SR/LA 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 group. However, fuels from two different reactors within a given group 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 based on enrichments (For example, only fuel that was too large for an 18 inch diameter standard canister will be placed in a 24-inch canister). 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 [Miller[36]] 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 [WASRD[37]]. Based on the above, the DOE SNF groups are placed into the various canisters for repository disposal. Tables 8-1 and 8-2 summarize the canister size and count for each DOE SNF group based on 2,333 MTHM and ~2,500 MTHM respectively [DOE-EM 1998c[34]]. Detailed canister size and count from each site are available on an EXCEL spreadsheet and may be obtained from the NSNF Program. DOE EM plans to use 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) ~5 inches diameter HIC by various length for the fuel that has been degraded or SNF samples, (4) the MCO for the Hanford fuel (mainly N-Reactor), and (5) the large disposal package (LDP) for the commercial-like DOE SNF.

Table 8-1. DOE SNF canister size and count summary (2,333 MTHM).

Fuel group	Fuel matrix	~18" dia x 10' long	~18" dia x 15' long	~5" HIC x various length	~25" dia x 15' long (MCO)	PWR21 ~5.4' dia x 15' long	BWR ~5.4' dia x 15' long	
1	Classified-Navy	NA	NA	NA	NA	NA	NA	By Navy
2	Pu/U alloy	168	17	15	0	0	0	
3	Pu/U carbide	1	3	0	0	0	0	
4	MOX	19	564	50	0	0	1	
5	U/Th carbide	0	523	1	0	0	0	
6	U/Th oxide	17	47	0	0	0	0	
7	U-metal	13	5	0	371	0	0	
8	U-oxide	249	387	66	0	8	2	
9	Aluminum-based fuel	986	1	24	0	0	0	
10	Unknown fuel	2	2	5	0	0	0	
11	U-Zr-Hx	10	34	7	0	0	0	
Total		1,465	1,583	168	371	8	3	

Table 8-2. DOE SNF canister size and count summary (all ~2,500 MTHM).

Fuel group	Fuel matrix	~18" dia x 10' long	~18" dia x 15' long	~5" HIC various length	~25" dia x 15' long (MCO)	PWR21 ~5.4' dia x 15' long	BWR ~5.4' dia x 15' long	
1	Classified-Navy	NA	NA	NA	NA	NA	NA	By Navy
2	Pu/U alloy	180	18	16	0	0	0	
3	Pu/U carbide	1	3	0	0	0	0	
4	MOX	21	604	53	0	0	1	
5	U/Th carbide	0	560	1	0	0	0	
6	U/Th oxide	18	50	0	0	0	0	
7	U-metal	14	6	0	397	0	0	
8	U-oxide	266	415	71	0	8	2	
9	Aluminum-based fuel	1,057	1	25	0	0	0	
10	Unknown fuel	2	2	5	0	0	0	
11	U-Zr-Hx	11	37	7	0	0	0	
Total		1,570	1,696	178	397	8	3	

9. 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 [ICD[38]]. Tables 9-1 and 9-2 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 ICD. DOE EM plans to use five different disposal packages. They are as follows: (1) a 5 x 1 codisposal package with five HLW canisters and one ~18 inches diameter fuel canister in ~10 feet length, (2) a 5 x 1 codisposal package with five HLW canisters and one ~18 inches diameter fuel canister in ~15 feet length, (3) a 5 x 1 codisposal package with five HLW canisters and one 18 inches diameter by 15 feet long containing one or more various length ~5 inches diameter HIC, (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 9-1 through 9-3 show the nominal DOE SNF arrangement for the non-LDP disposal packages per the ICD.

An electronic word file named "WastePack.doc" shows the potential variability of the number of DOE SNF disposal packages. This file provides the basis for the disposal package count sensitivity analysis. The file is included in a diskette at the back of this report.

Table 9-1. DOE SNF codisposal size and package summary (2,333 MTHM).

Fuel group	Fuel matrix	5 HLW x 1 SNF x 10' long	5 HLW x 1 SNF x 15' long	5 HLW x 5" HIC x various length	No HLW x 4 MCO X15' long	PWR21 ~5.4' dia x 15' long	BWR ~5.4' dia x 15' long	
1	Classified-Navy	NA	NA	NA	NA	NA	NA	By Navy
2	Pu/U alloy	168	17	15	0	0	0	
3	Pu/U carbide	1	3	0	0	0	0	
4	MOX	19	564	50	0	0	1	
5	U/Th carbide	0	523	1	0	0	0	
6	U/Th oxide	17	47	0	0	0	0	
7	U-metal	13	5	0	93	0	0	
8	U-oxide	249	387	66	0	8	2	
9	Aluminum-based fuel	986	1	24	0	0	0	
10	Unknown Fuel	2	2	5	0	0	0	
11	U-Zr-Hx	10	34	7	0	0	0	
Total		1,465	1,583	168	93	8	3	

Table 9-2. DOE SNF codisposal size and package summary (all ~2,500 MTHM).

Fuel group	Fuel matrix	5 HLW x 1 SNF x 10' long	5 HLW x 1 SNF x 15' long	5 HLW x 5" HIC x various length	No HLW x 4 MCO X15' long	PWR21 ~5.4' dia x 15' long	BWR ~5.4' dia x 15' long	
1	Classified-Navy	NA	NA	NA	NA	NA	NA	By Navy
2	Pu/U Alloy	180	18	16	0	0	0	
3	Pu/U Carbide	1	3	0	0	0	0	
4	MOX	21	604	53	0	0	1	
5	U/Th Carbide	0	560	1	0	0	0	
6	U/Th Oxide	18	50	0	0	0	0	
7	U-metal	14	6	0	100	0	0	
8	U-oxide	266	415	71	0	8	2	
9	Aluminum-Based Fuel	1,057	1	25	0	0	0	
10	Unknown Fuel	2	2	5	0	0	0	
11	U-Zr-Hx	11	37	7	0	0	0	
Total		1,570	1,696	178	100	8	3	

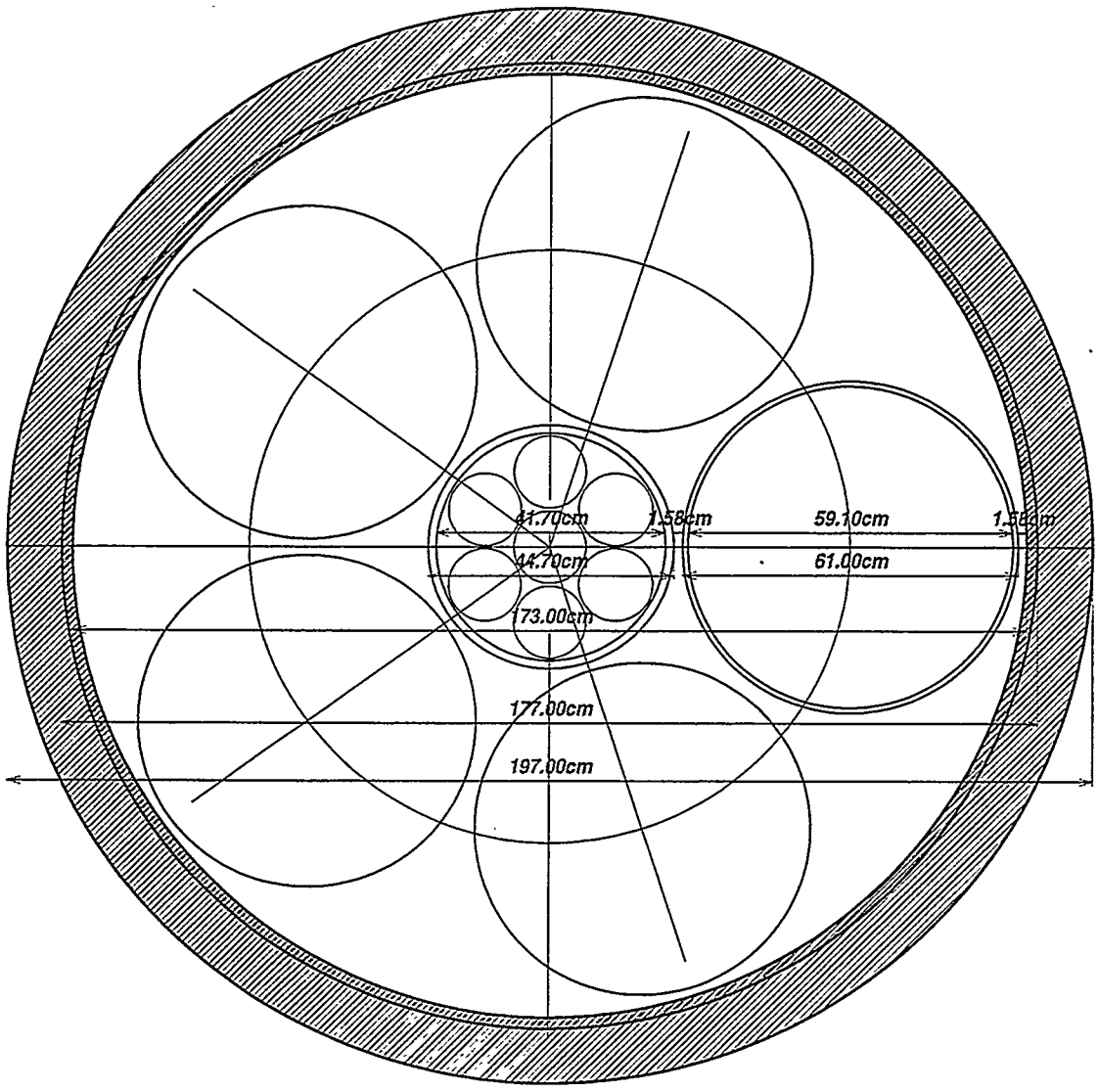

Figure 9-1. Proposed 5 (HLW) x 1 (SNF) codisposal package with HIC inside standard canister.

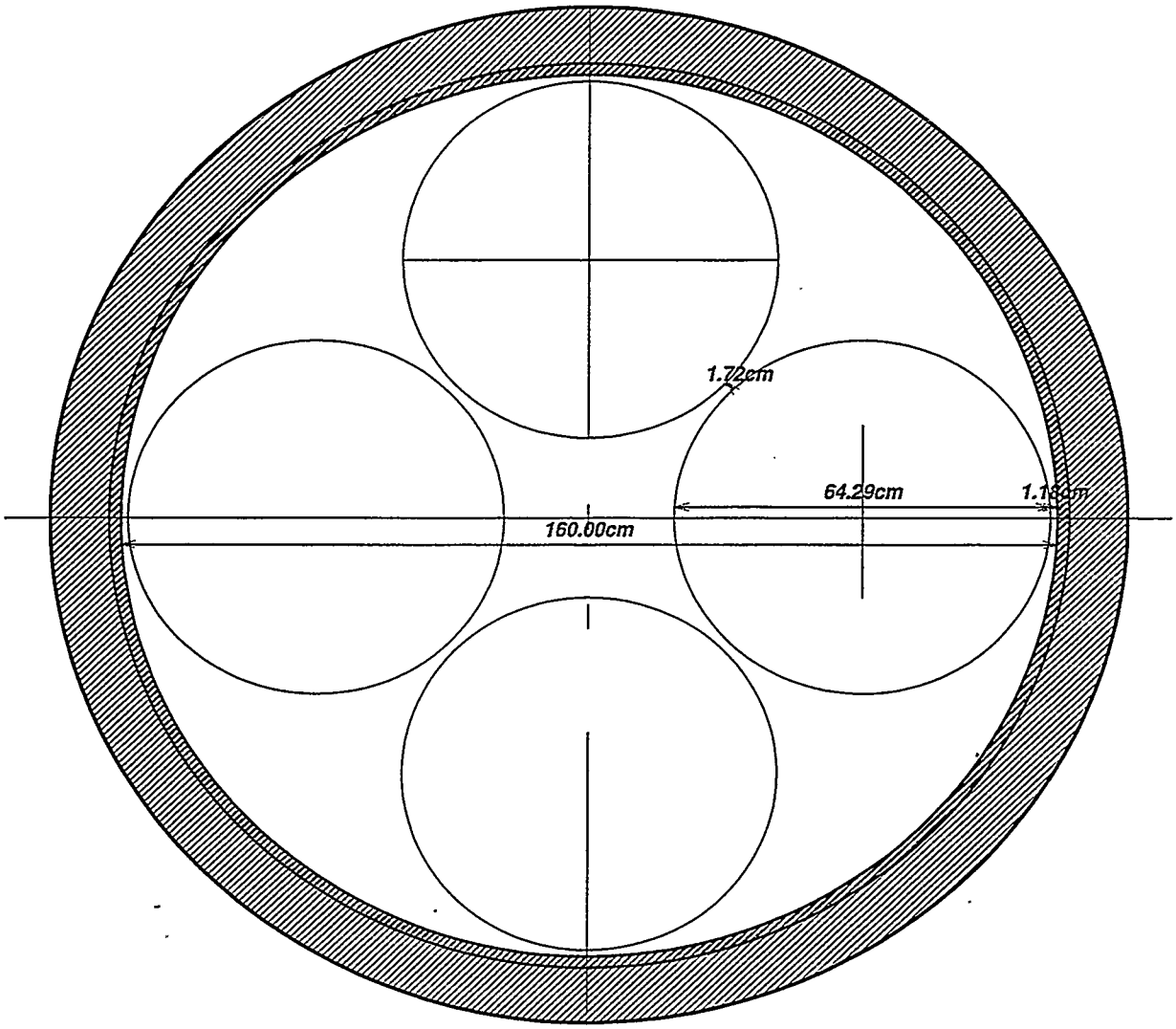

Figure 9-2. Proposed 0 (HLW) x 4 (MCO SNF) disposal package.

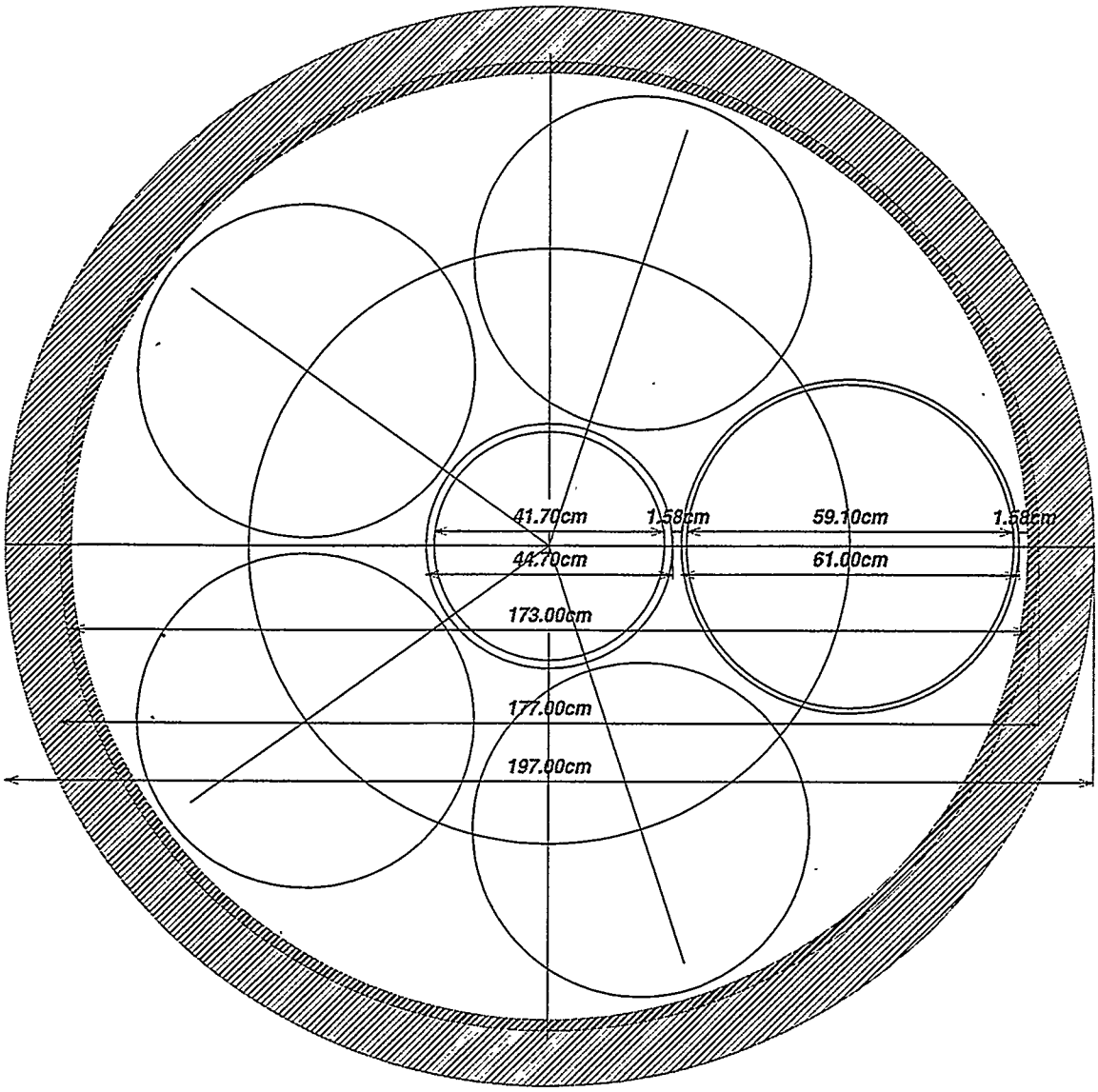

Figure 9-3. Proposed 5 (HLW) x 1 (SNF) codisposal package [both ~10 and 15 feet lengths].

10. CALCULATING PACKAGE CURIE LOADING

10.1 DOE SNF Radionuclide Inventory

For the DOE SNF, one or more ORIGEN-2 runs were selected to estimate the total radionuclide inventory for each group. The specific ORIGEN-2 runs used to represent each group are indicated in the Table 10-1 below. As an example, for Group 7, the N-Reactor fuel ORIGEN run was used to represent the N-Reactor and the Single Pass Reactor fuels. A commercial PWR fuel ORIGEN run was 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. ORIGEN runs were performed by each site using input data gather by each site.

Table 10-1. ORIGEN-2 runs used in the DOE fuel group.

Fuel group	ORIGEN-2 runs used to represent various fuels in the group	Comment
1. Classified	Classified Navy	By Navy
2. Pu/U alloy	Advanced Test Reactor (ATR) fuel Enrico Fermi Reactor (FERMI) fuel	No ORIGEN runs available for U-Zr alloy fuels. ATR was used because the reactor and fuel characteristics were similar (i.e., HEU fuel, high burnup test reactor)
3. Pu/U carbide	FFTF carbide fuel FSV fuel ATR fuel	ATR was used to represent fuels with similar burnup
4. MOX	FFTF oxide fuel	
5. U/Th carbide	FSV fuel Peach Bottom fuel General Atomics-High Temperature Gas Cooled Reactor (GA-HTGR) fuel	
6. U/Th oxide	Shippingport LWBR fuel	ORIGEN runs for both seed and blanket
7. 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

Table 10-1. (continued).

Fuel group	ORIGEN-2 runs used to represent various fuels in the group	Comment
8. U-oxide	Commercial PWR fuel Commercial boiling water reactor (BWR) fuel Pathfinder fuel Power Burst Facility (PBF) fuel Pulstar Buffalo fuel Shippingport PWR Fuel TMI fuel Transient Reactor Test (TREAT) fuel FFTF oxide fuel ATR fuel Missouri University Research Reactor, (MURR) fuel Rhode Island Nuclear Science Center, (RINSC) fuel ORR fuel	
9. Al-based fuel (U-Alx, U_3Si_2, U-oxide in Al)	MURR fuel RINSC fuel ORR fuel	
10. Unknown fuel	ATR fuel Enrico Fermi Reactor (FERMI) fuel	No ORIGEN runs available for the unknown fuels. ATR was used to represent fuels with similar burnup and assume other fuels are from fast reactor (mostly from ANL-W) so Fermi core A-2 ORIGEN run was also used.
11. U-Zr-Hx	TRIGA fuel — General Atomic	ORIGEN runs for both STD and FLIP

As noted in the grouping discussion, Group 2 fuels consist of various uranium alloy fuels. For the U-Mo alloy fuels from the Fermi reactor, the inventory was represented by the Fermi ORIGEN run. For the U-Zr alloy fuels from reactor such as the HWCTR, ORIGEN-2 run from another group (ATR fuel) was used to estimate the inventory because no ORIGEN run was available for the U-Zr fuel in the group. In the future, the DOE SNF radionuclide inventories will be updated based on the source term determination effort presently in progress with participation from all the DOE sites and RW.

The total radionuclide inventory for each DOE SNF group 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. Three electronic EXCEL files named "11RW_2Input699B.xls", "11RW_Input399A.xls", and HIC_499WC.xls" contain the DOE SNF inventory and other parameters presented in this report are included in a diskette at the back of this report.

10.2 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 [CDB[39]]. 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-SR/LA. The radionuclide inventory for the HLW canister is shown in Table D-2 of Appendix D.

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DOE Spent Nuclear Fuel Information In Support of TSPA - SR

Appendix A
DOE SNF Group Description

A-1. DOE SNF GROUP DESCRIPTION

The following section describes the typical fuels within each of the DOE SNF groups and the various information for each of the fuel groups. The group title indicates the SNF matrix follow the dominant cladding material in the group. As an example, Group 7 consists of U metal matrix with the dominant cladding material of zirconium.

A-1.1 Group 1 Classified Navy Fuel

Typical fuel: Navy Fuel

Fuel Description

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

A-1.2 Group 2 Pu/U Alloy Fuel

Typical fuel: Fermi driver

Fuel Description

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

The radial blanket fuel subassembly is made up of an inlet nozzle, a lower axial blanket, a fuel section, and an upper axial blanket. The radial blanket fuel subassemblies were made up of 25 cylindrical rods fabricated from depleted U-Mo alloy, encased in stainless steel tubes and bonded with sodium. The radial blanket subassemblies are currently stored dry in ICPP-749. The radial blanket subassembly rods contain depleted uranium and sodium and thus will be treated before final disposition. Those rods are not part of the Group 2 inventory.

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

fuel pins were closely packed into the 2.646-inch square tube. The fuel pins were maintained on a square pitch of 0.200 inches in a cartridge made of stainless steel wires and plates.

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

Group 2 Pu/U Alloy Fuel Inventories/Information		**Comment**
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D Table D-1	
Composition	Nominal 25.69% in U-235, and 10 wt % Mo	
Matrix dissolution rate	Metal model x 10	See Section 3.2 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	1.2E-03	
Clad failure fraction	Assume 0.1	
Free radionuclide inventory fraction	0.00001	
Gap fraction	0	

Configuration and Package Count

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

	# 5x1 10 ft	# 5x1 15 ft	# HIC	# MCO 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 2 Pu/U Alloy Fuel						
- repository pkg count	168	17	15			
- HLW can count	840	85	75 (max)			
- SNF pkg count	168	17	15			

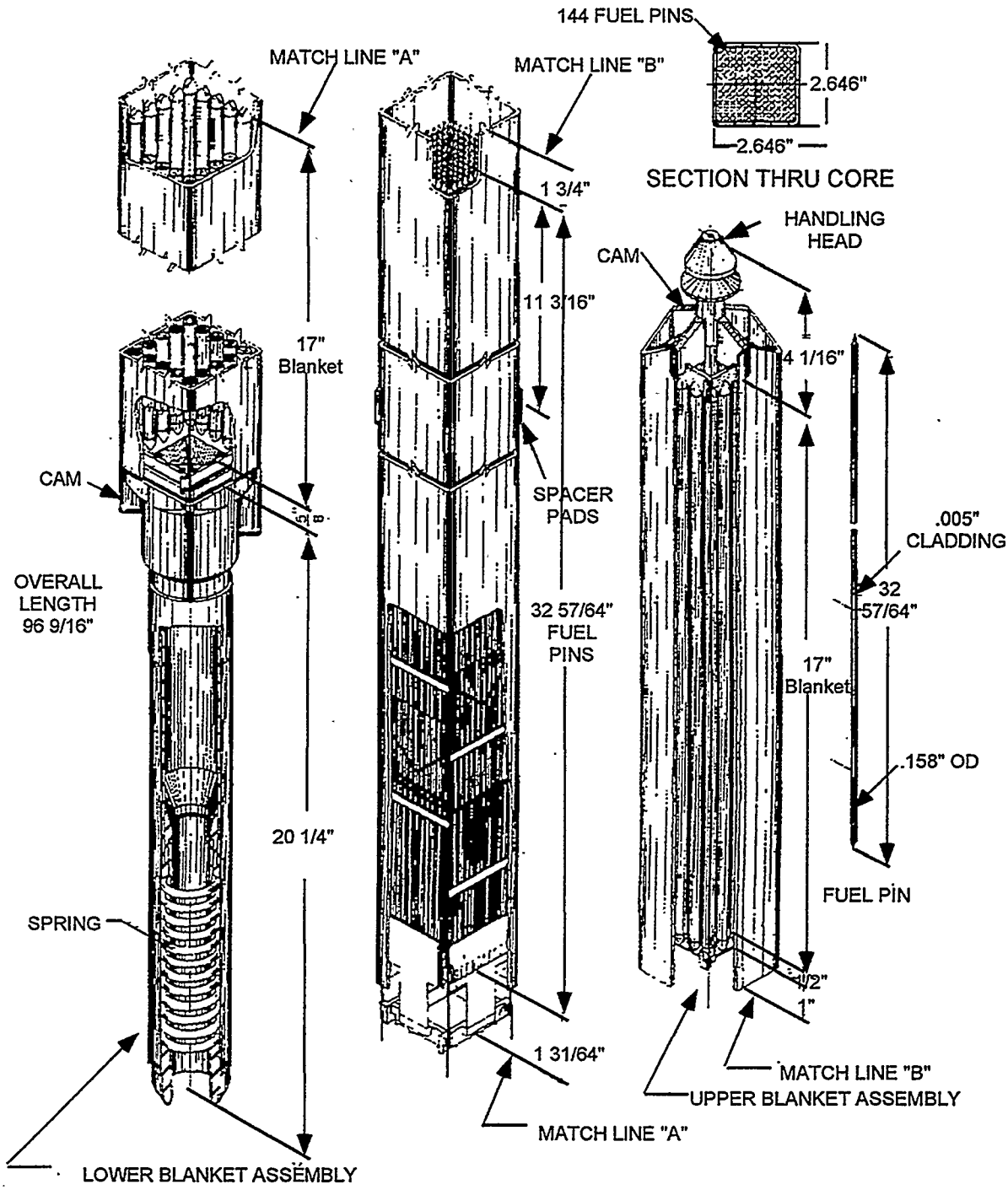


Figure A-1. Fermi driver fuel subassembly.

A-1.3 Group 3 Pu/U Carbide Fuel

Typical fuel: FFTF Carbide (MEU Fissile Gram Equivalent), SRE (MEU Fissile Gram Equivalent)

Fuel Description

Group 3 fuel is mixed carbide fuel in a nongraphite matrix. A number of the fuels were test fuel assemblies (TFAs) from the Fast Flux Test Facility (FFTF). FFTF was to provide testing capability for a wide range of development needs of the United States advanced reactor program. The mission of the FFTF included irradiation and evaluation of different types of fuel assemblies and different materials for fuel assembly construction. The purposes of the TFAs vary and a few examples are indicated below. However, in general, the TFAs support the fuel or material requirements for large-scale breeder reactors.

As an example, the FFTF-ACN-1 fuel in this group was tested to develop information on helium- and sodium-bonded mixed-carbide fuel pins with full length fuel columns at prototypic fluence and exposure conditions. Additionally, it tests the relative effects of 20% cold worked 316 SS and 25% cold worked D9 cladding on the carbide fuel pins. The assembly contains 18 sodium-bonded and 19 helium-bonded carbide fuel pins, enclosed in a 316 SS inner duct. The outer region contains 90 standard driver fuel pins and is enclosed by a D9 duct [Hanford[b]]. The test fuel assembly's (TFAs) configuration is similar to the FFTF driver fuels shown on Figures A.2 and A.3 under Group 4.

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

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

Group 3 Pu/U carbide Fuel Inventories/Information		**Comment**
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	Pu/U carbide	
Matrix dissolution rate	Metal model x 100	See Section 3.3 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	2.7E-03	
Clad failure fraction	Assume 0.1 failed	
Free radionuclide inventory fraction	0.0	
Gap fraction	0.01-0.06	

Configuration and Package Count

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

		# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 3	Pu/U carbide Fuel						
	- repository pkg count	1	3				
	- HLW can count	5	15				
	- SNF pkg count	1	3				

A-1.4 Group 4 MOX Fuel

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

Fuel Description

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%. The FFTF DFA and TFA contributed to a large quantity of the fuel in this group. The standard FFTF-DFA is hexagonally shaped and composed of 217 fuel pins. The assembly is 12 feet (3.6 m) long, 4.575 inches (11.6 cm) wide across the hexagon flats, 5.16 inches (13.1 cm) wide across the hexagon points, and weighs 381 pounds (173 kg). Figures A-14 and A-15 show the configuration of the standard FFTF-DFA fuel assembly [Hanford[b]].

Group 4 MOX Fuel Inventories/Information		Comment
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	Mixed oxide - U oxide and Pu oxide	
Matrix dissolution rate	Commercial model	See Section 3.4 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	4.0E-03	
Clad failure fraction	Assume 0.1	
Free radionuclide inventory fraction	0.01-0.06	
Gap fraction	0.01-0.06	

Configuration and Package Count

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

		# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 4	MOX Fuel						
	- repository pkg count	19	564	50			1
	- HLW can count	95	2,820	250			
	- SNF pkg count	19	564	50			1

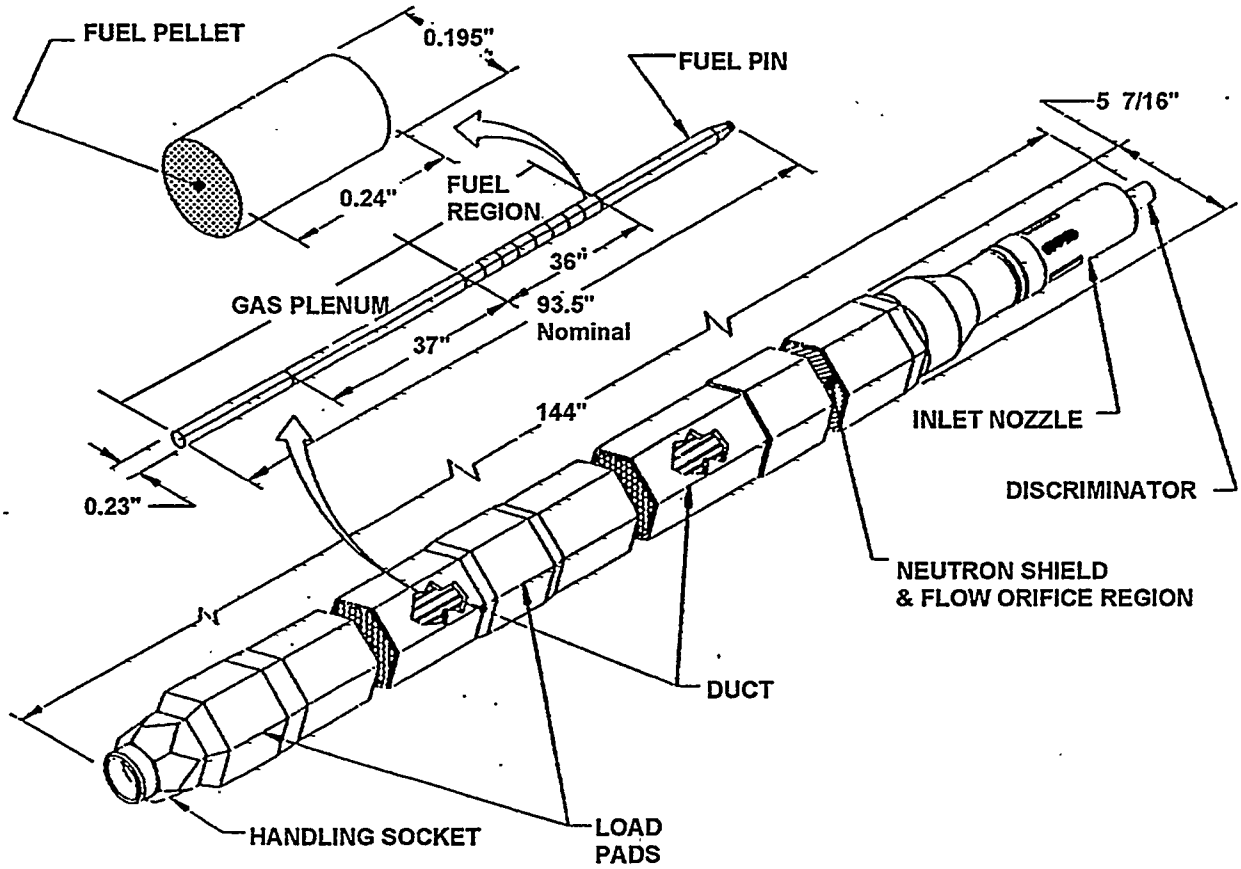
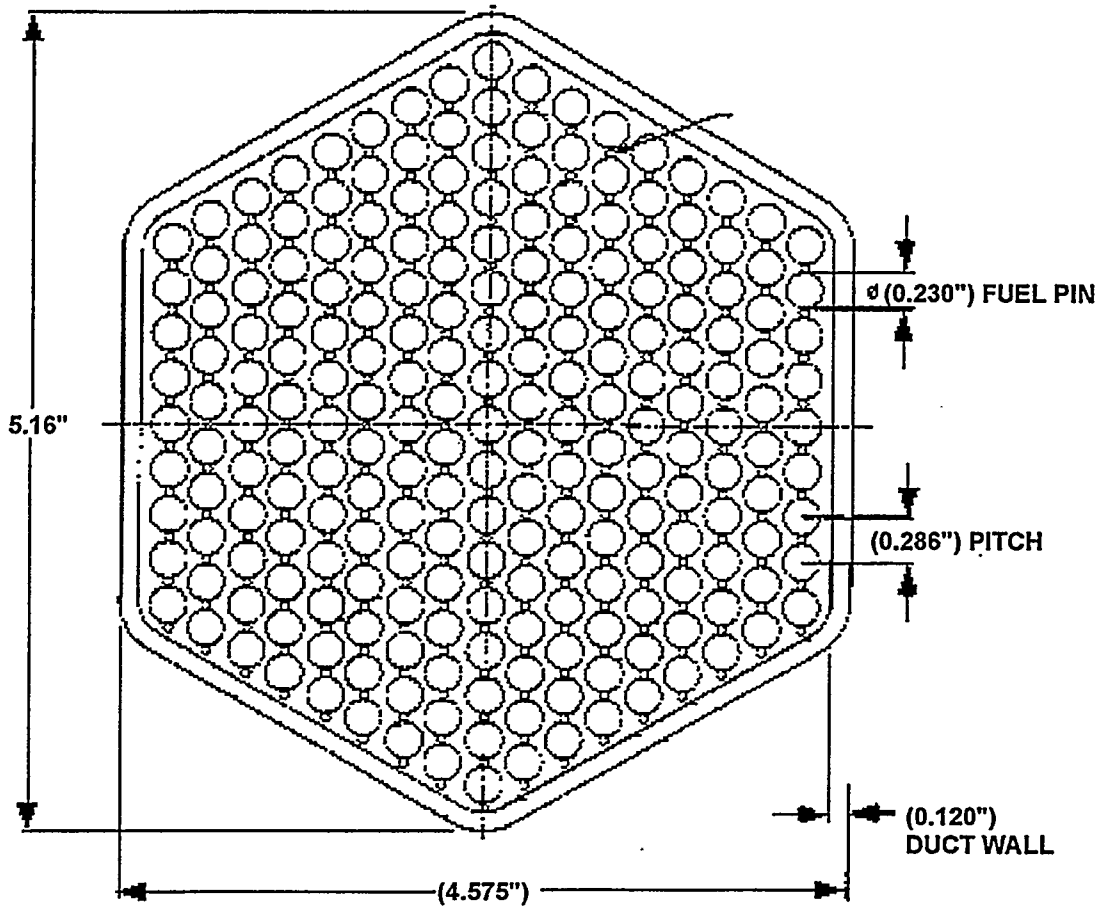


Figure A-2. FFTF standard driver fuel assembly.



FAST FLUX TEST FACILITY
FUEL PIN BUNDLE CROSS SECTION

Figure A-3. FFTF pin bundle cross section.

A-1.5 Group 5 U/Th Carbide Fuel

Typical fuel: Fort Saint Vrain (FSV)

Fuel Description

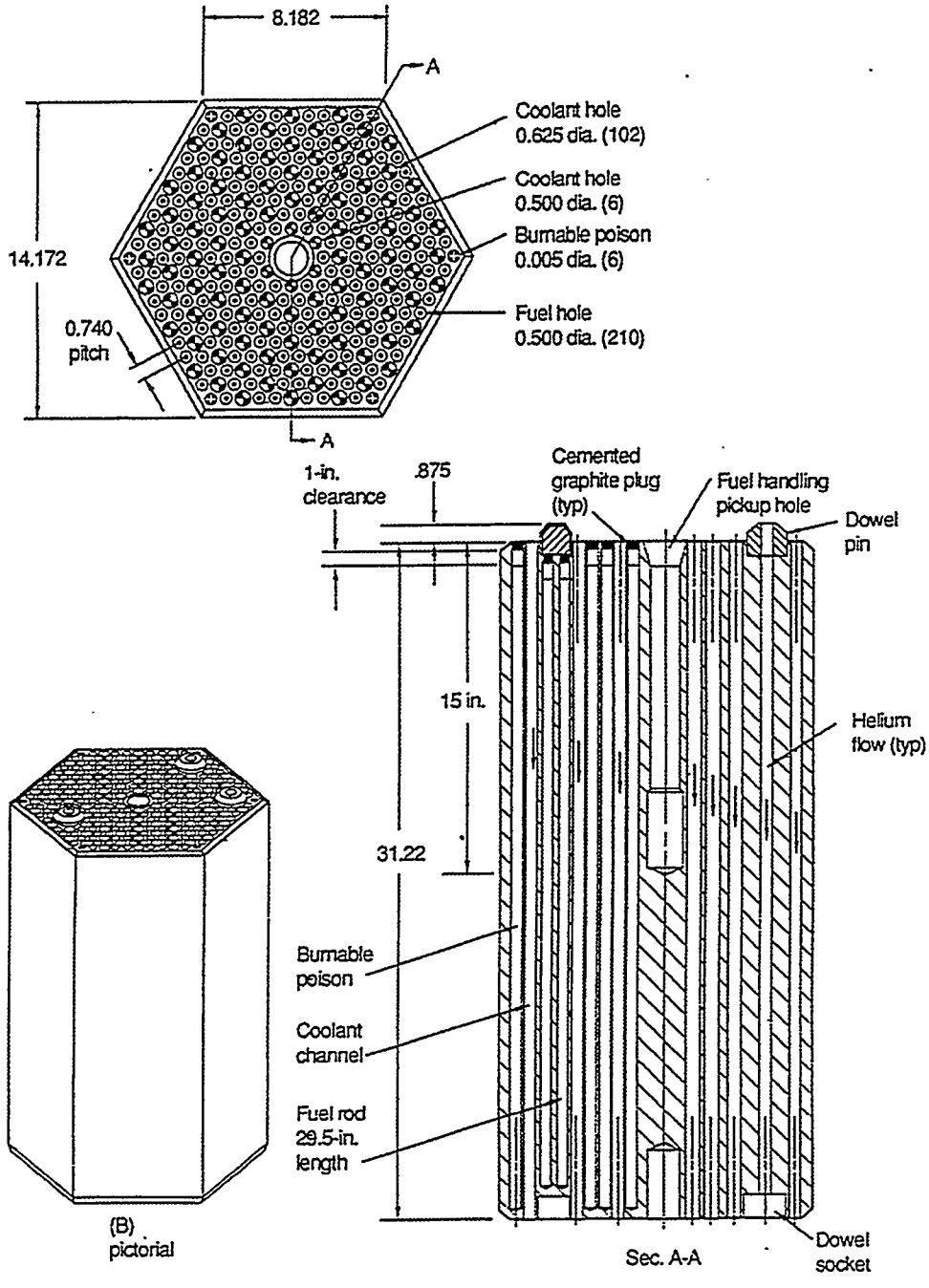
The FSV fuel is a graphite-based fuel that was used only in the Fort Saint Vrain Reactor. An assembly is composed of a hexagonal shaped graphite block drilled with 102 coolant holes and 210 fuel holes. The fuel is made of highly enriched uranium carbide and thorium carbide spheres coated with layers of pyrolytic carbon followed by a SiC protective outer coating, which is very durable, and an outer pyrolytic coating. The fuel spheres are sintered with carbon and formed into rods, called compacts, and then stacked into the fuel holes within large hexagonal blocks of graphite. The compacts are 0.5 inch (1.27 cm) in diameter and 2 inches (5.08 cm) long. These blocks are 14.172 inches (36 cm) across the flats, 8.102 inches (20.6 cm) on each side, and 31.22 inches (79.3 cm) long [LUNFIS[a]]. Figure A-4 shows the Fort Saint Vrain fuel assembly.

Group 5 U/Th carbide Fuel Inventories/Information		**Comment**
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	U/Th carbide	
Matrix dissolution rate	Si carbide model	See Section 3.5 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	2.2E-02	
Clad failure fraction	Assume 0.6-0.8	
Free radionuclide inventory fraction	0.1	
Gap fraction	0.001	

Configuration and Package Count

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

		# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 5	U/Th carbide Fuel						
	- repository pkg count		523	1			
	- HLW can count		2,615	5 (max)			
	- SNF pkg count		523	1			



Fort St. Vrain graphite fuel.

Figure A-4. Fort St. Vrain graphite fuel.

A-1.6 Group 6 U/Th Oxide Fuel

Typical fuel: LWBR (HEU Fissile Gram Equivalent), Dresden (HEU Fissile Gram Equivalent)

Fuel Description

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

Group 6 U/Th oxide Fuel Inventories/Information		**Comment**
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	U/Th oxide	
Matrix dissolution rate	Ceramic model	See Section 3.6 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	3.6E-04	
Clad failure fraction	Assume 0.1	
Free radionuclide inventory fraction	0.0	
Gap fraction	0.01-0.06	

Configuration and Package Count

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

		# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 6	U/Th oxide Fuel						
	- repository pkg count	17	47				
	- HLW can count	85	235				
	- SNF pkg count	17	47				

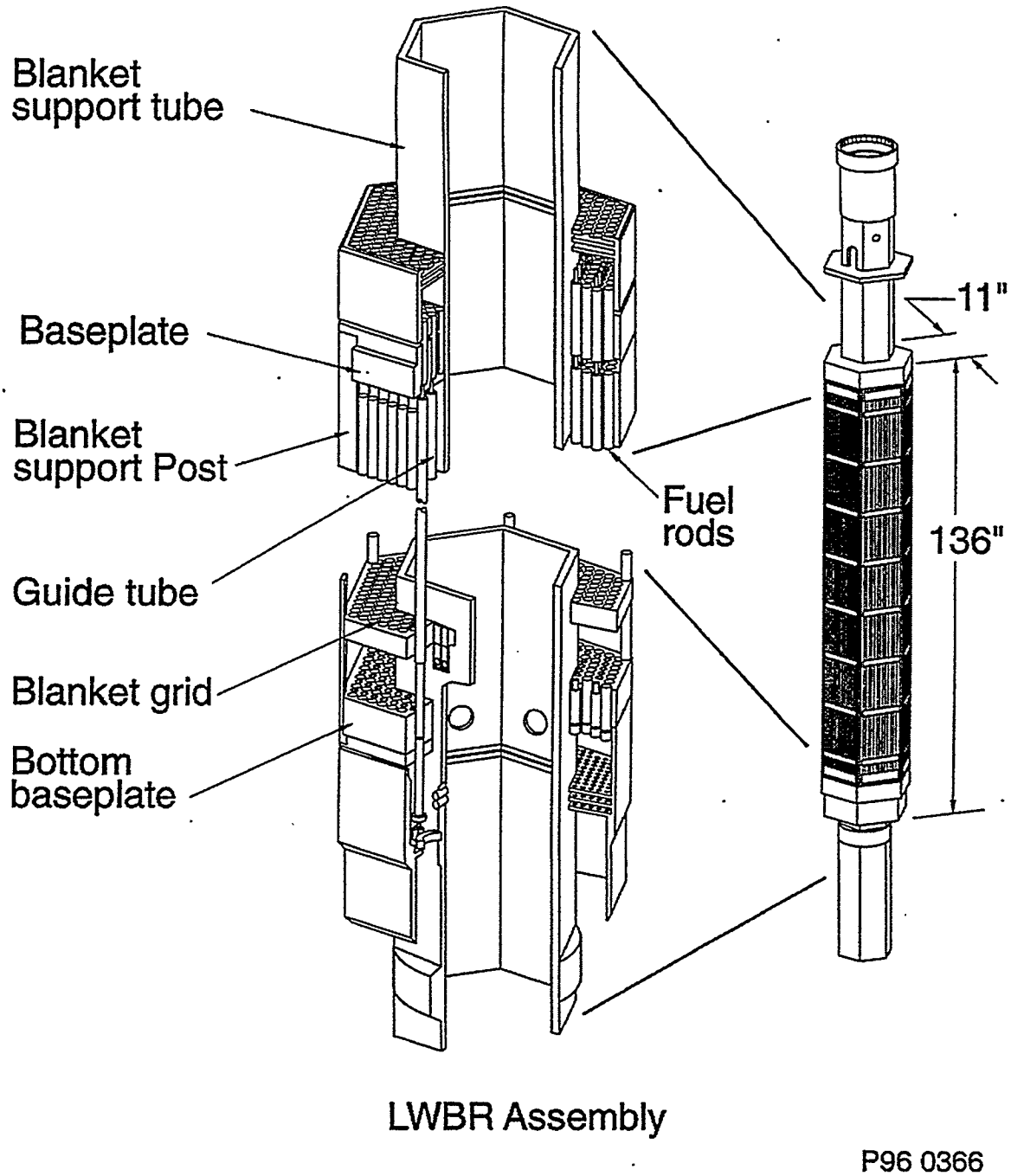


Figure A-5. Shippingport LWBR fuel assembly.

A-1.7 Group 7 U metal Fuel

Typical fuel: N-reactor

Fuel Description

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

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

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

Group 7 U metal Fuel Inventory/Information		**Comment**
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	Breached fuel cladding uranium metal with possible oxide surface coating	
Matrix dissolution rate	Metal model	See Section 3.7 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	7.0E-05	
Clad failure fraction	Assume 1	
Free radionuclide inventory fraction	0.001	
Gap fraction	0	

Configuration and Package Count

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

	# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 7 U metal Fuel						
- repository pkg count	13	5		93		
- HLW can count	65	25		0		
- SNF pkg count	13	5		93		

Tables A-1 and A-2 provide a summary listing of the various chemical components associated with the typical N-Reactor fuels. Figure A-6 depicts a typical N-Reactor fuel element; Figures A-7 through A-9 depict proposed layout of N-Reactor fuel packaging within an MCO as it was evaluated in the performance assessment.

Table A-1. N-Reactor fuel element description.

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

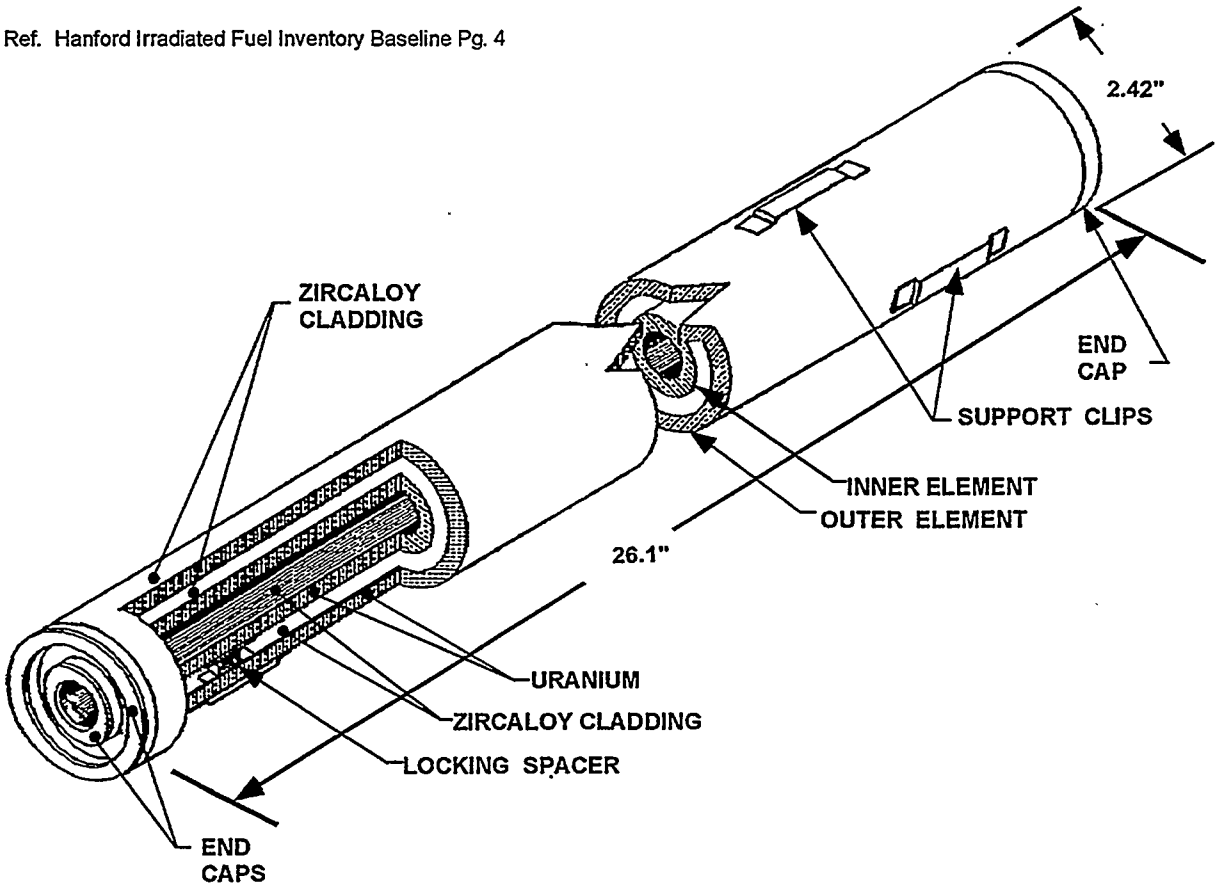
a. Letter code differentiates the different lengths of the Mark IV or Mark IA fuel elements, i.e., a Type "E" element is 66.3 cm long. [Hanford Irradiated Fuel Inventory Baseline]

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

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

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

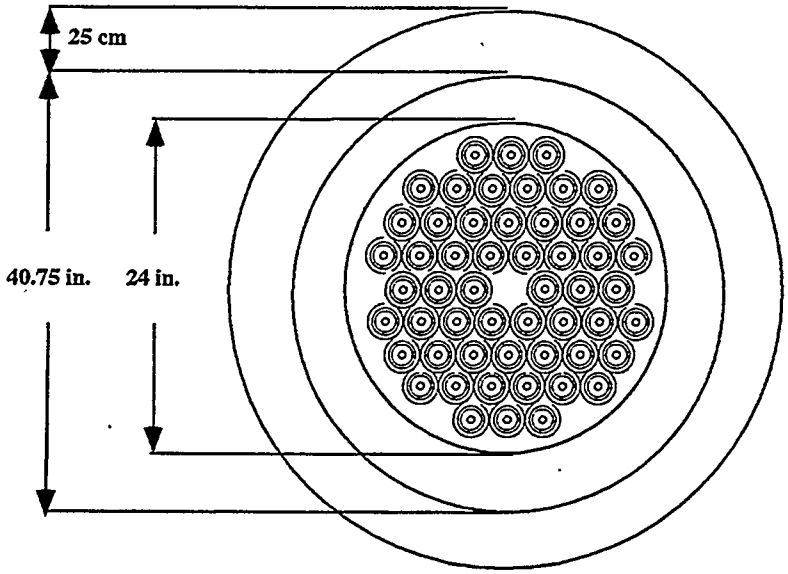
* Ref. Hanford Irradiated Fuel Inventory Baseline Pg. 4



105-N REACTOR MARK IV FUEL ELEMENT ASSEMBLY

Figure A-6. N-Reactor Mark IV fuel element assembly.

Loading Arrangement for Mark IV Fuel in MCO Container.



Loading Arrangement for Mark IA Fuel in MCO Container.

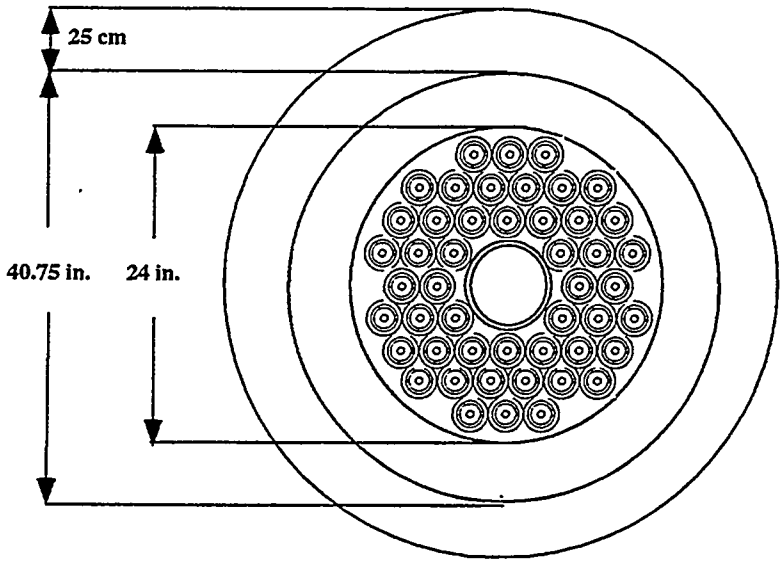


Figure A-7. (Top) Loading arrangement for Mark IV fuel in MCO container.
Figure A-8. (Bottom) Loading arrangement for Mark 1A fuel in MCO container.

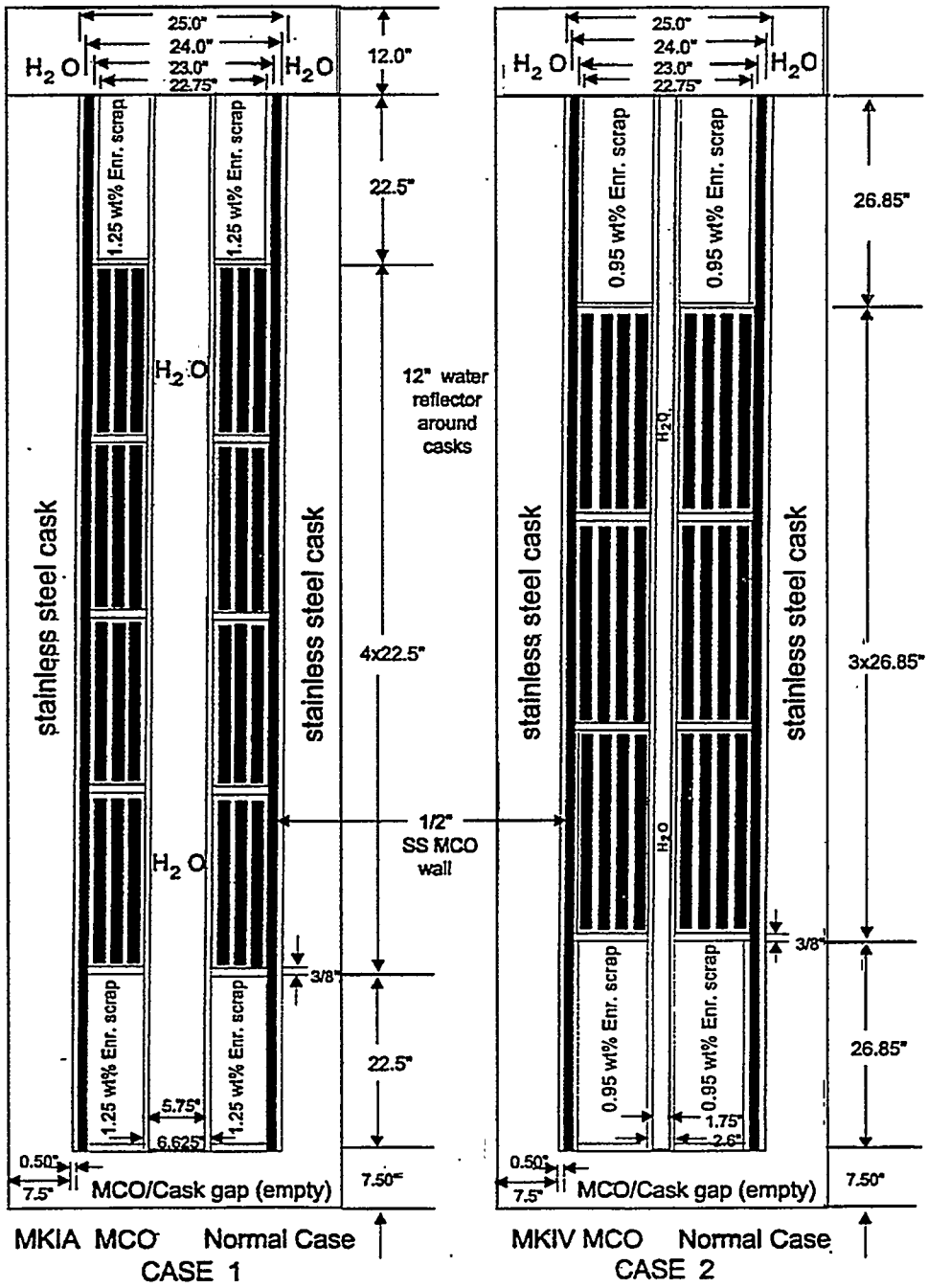


Figure A-9. MCO axial geometry layout.

A-1.8 Group 8 U Oxide Fuel

Typical fuel: TMI-2 (LEU), SM-1A, ORNL SST & Zr (MEU), Shippingport (HEU), commercial (LEU), Saxton (MEU), ML-1 (HEU), PBF (MEU), FFTF-TFA (LEU)

Fuel Description

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

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

Primary issues related to packaging this fuel group for disposal related to: (1) packaging for criticality control and (2) drying material to prevent gas generation. Figure A-10 shows the canister configuration for the Three Mile Island, Unit 2 (TMI-2).

The remainder of the fuels in this group generally have the characteristics found in most of the commercial fuels (PWR and BWR). The fuel was received for examination and testing under a variety of programs. These fuels have ended up in the DOE SNF inventory. As an example, the commercial fuels were brought to the DOE site for examination or testing programs, while some were reconfigured for the Dry Rod Consolidation Test (DRCT) at the INEEL. The reconfiguration involved consolidating the fuel by removing the rods and placing them into canisters so as to double the number of rods in a volume equal to a standard commercial fuel assembly. Other examination or testing involved taking some of the assemblies and rods apart for postirradiation examination. The fuel compositions, properties, and conditions are identical to the commercial fuel.

The Power Burst Facility (PBF) was used to test fuel materials and the driver fuel was included in the Group 8 inventory. The PBF driver core fuel contains a pelletized ternary fuel (UO_2-ZrO_2-CaO-18.5% enriched) surrounded by a helium gas annulus, an insulator sleeve of (ZrO_2-CaO), and cladded with 304L stainless steel. This fuel is similar to commercial fuel that is made by pressing the uranium oxide into pellets. The pellets are loaded into zircaloy or stainless steel tubes [LUNFIS[a]].

Other fuels such as the Shippingport PWR Core 2 Seed 2 were also included in the Group 8 inventory. The Shippingport PWR was built to demonstrate the concept of a light water, slow breeder reactor using a commercial type pressurized water reactor (PWR). This was a joint AEC/Navy project that was designed for development and demonstration purposes of this type of reactor. Bettis Atomic Power Laboratory designed the reactor. The Naval Reactors Group of the AEC directed the project, and the power was distributed by Duquesne Light Company. The Navy's NRF and ECF facilities received the fuel after it was removed from the core. The Navy played a large part in all aspects of this reactor. Shippingport was designed and built to test different core designs and explore operating variables for large-scale nuclear reactors. The reactor was of the seed and blanket type and began operation with the first core (PWR-C1) in December 1957. The seed element was a zircaloy-clad plate-type fuel, while the

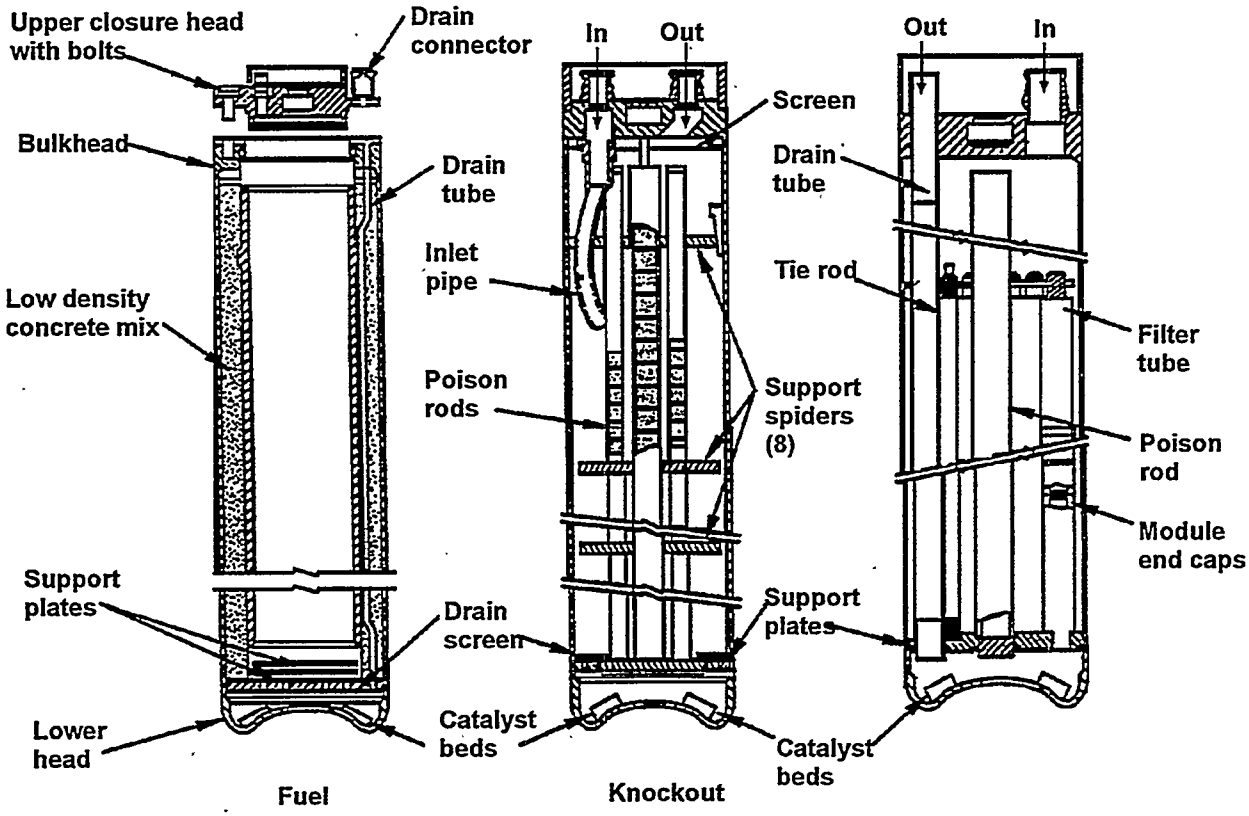


Diagram of the Three TMI-2 Canister Types

Figure A-10. TMI-2 canister types.

blanket fuel was in the form of pellets placed inside short (~10 inches) Zircaloy-2 tubes. The basic component of the seed elements was the fuel plate. A plate was formed by sandwiching an enriched (~93%) U-Zr alloy strip between two zircaloy-2 cover plates and four side strips. Figure A-11 shows the Shippingport PWR fuel subassembly [LUNFIS[a]].

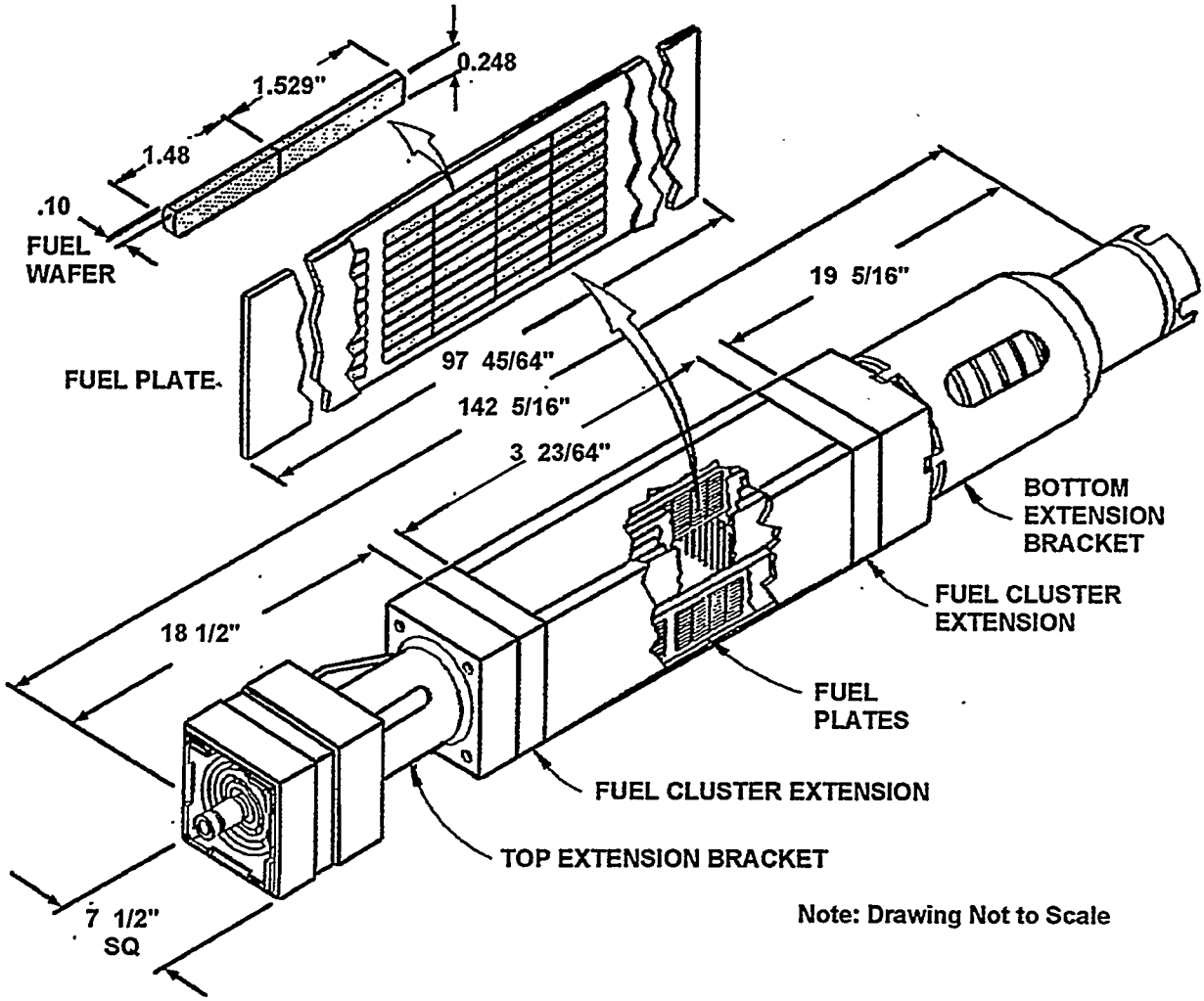


Figure A-11. Shippingport Core 2 Blanket Fuel Assembly.

Group 8 U Oxide Fuel Inventories/Information		Comment
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	Pressed uranium oxide pellets	
Matrix dissolution rate	Commercial model	
Surface area (m^2/g)	4.0E-01	See Section 3.8 of the DOE SNF Information in Support of TSPA-SR report for detail
Clad failure fraction	Assume 1	
Free radionuclide inventory fraction	0.01-0.06	
Gap fraction	0.01-0.06	

Configuration and Package Count

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

	# 5x1 10 ft	# 5x1 15 ft	HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 8 U oxide Fuel						
- repository pkg count	249	387	66		8	2
- HLW can count	1,245	1,935	330			
- SNF pkg count	249	387	66		8	2

A-1.9 Group 9 Aluminum based fuel

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

Fuel Description

This group consists of fuels with the (1) uranium-aluminide dispersed in a continuous aluminum phase, (2) uranium-silicide dispersed in a continuous aluminum phase, and (3) uranium oxide dispersed in a continuous aluminum phase. The cladding is assumed to be intact at this time, but is not considered to be a very durable material in long-term storage conditions in wet environments. The nature of the cladding suggests application of a lower allowable centerline temperature (~200 C) within a waste package.

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

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

Group 9 Aluminum Based Fuel Inventories/Information		**Comment**
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	UAlx dispersed in aluminum	
Matrix dissolution rate	Metal model x 0.1	See Section 3.9 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	6.5E-03	
Clad failure fraction	Assume 1	
Free radionuclide inventory fraction	0.0001	
Gap fraction	0	

Configuration and Package Count

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

		# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 9	Aluminum Based Fuel						
	- repository pkg count	986	1	24			
	- HLW can count	4,930	5	120 (max)			
	- SNF pkg count	986	1	24			

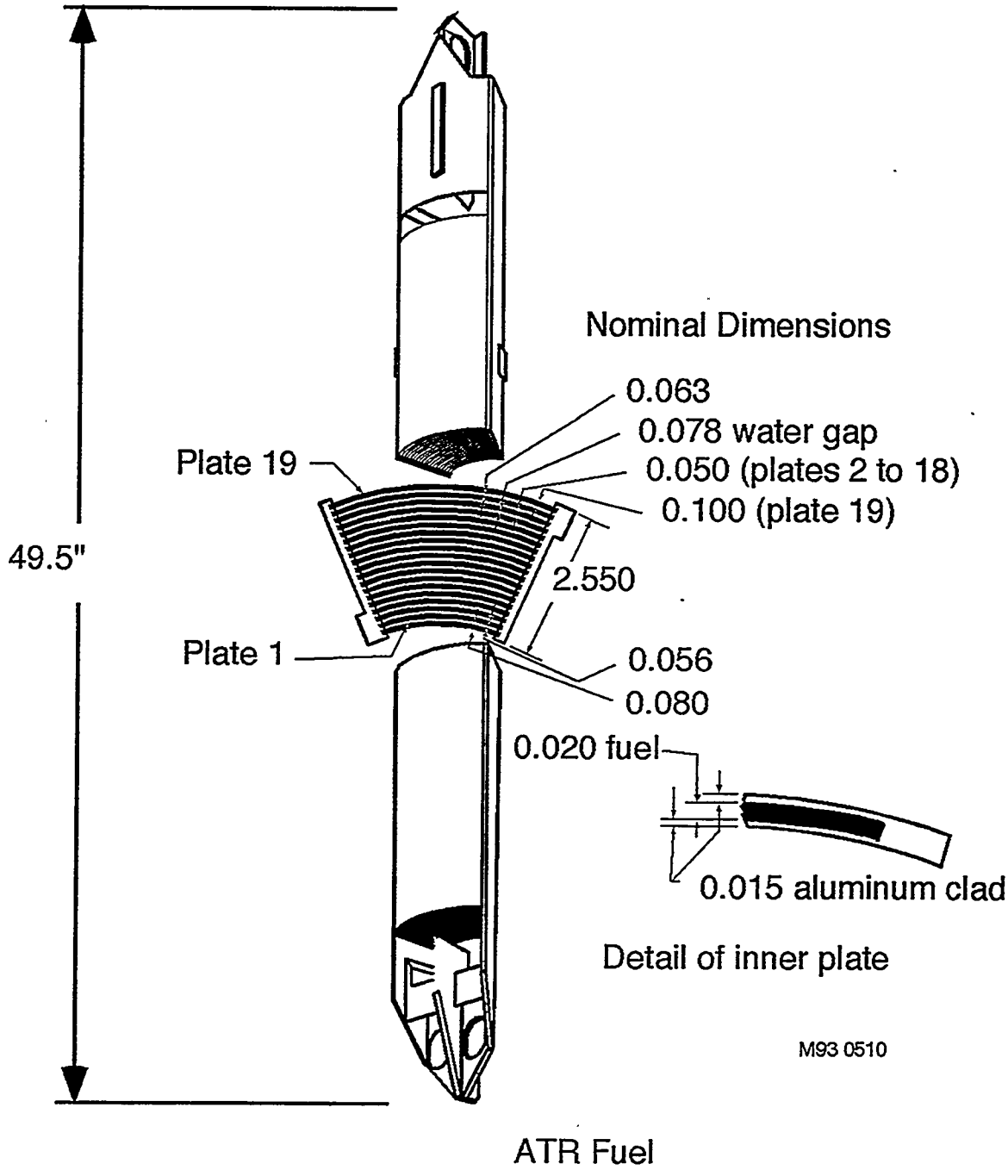


Figure A-12. ATR fuel element.

A-1.10 Group 10 Miscellaneous DOE SNF

Typical fuel: Miscellaneous DOE SNF

Fuel Description

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

Group 10 Miscellaneous DOE SNF Inventories/Information		Comment
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	Miscellaneous compositions	
Matrix dissolution rate	Metal model	See Section 3.10 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	4.0E-01	
Clad failure fraction	Assume 1	
Free radionuclide inventory fraction	0.001	
Gap fraction	0	

Configuration and Package Count

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

		# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 10	Miscellaneous DOE SNF						
	- repository pkg count	2	2	5			
	- HLW can count	10	10	25 (max)			
	- SNF pkg count	2	2	5			

A-1.11 Group 11 U-Zr Hydride Fuel

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

Fuel Description

Group 11 contains the fuel with the uranium/zirconium hydride matrix. Fuels from the TRIGA reactors make up the majority of the fuels in this group. The Training, Research, Isotope General Atomics (TRIGA) research reactor have been in use since 1957 throughout the United states and more than 20 countries worldwide. The TRIGA reactors are water-cooled, graphite and water reflected, pool-type research reactors that have steady-state and pulsing capabilities. There are six TRIGA reactors developed by General Atomic, each having different experimental facility features to accommodate a user's specific needs.

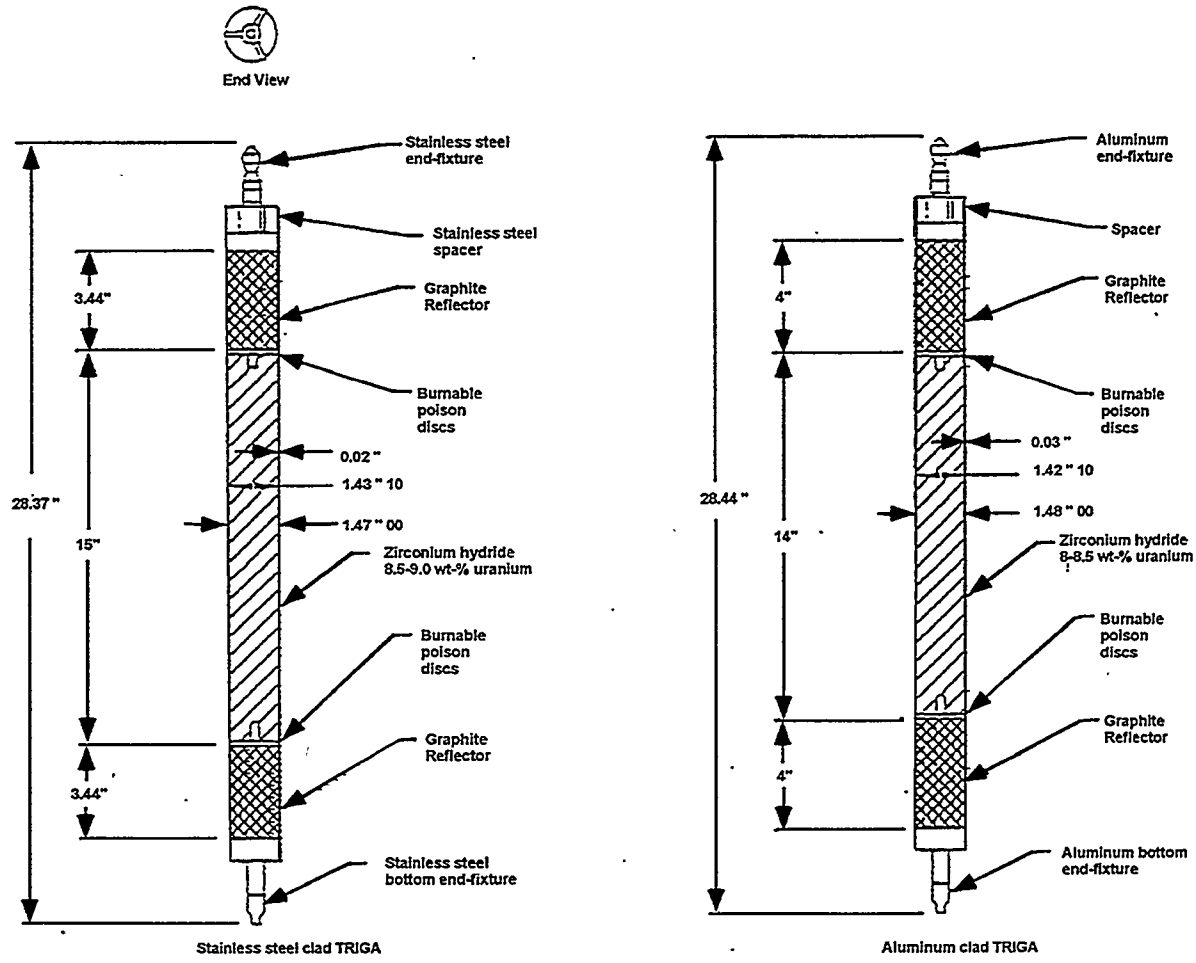
Like all the fuels in this group, TRIGA fuel elements are made of a uranium-zirconium hydride matrix that provides the reactor with its built in control and inherent safety. They are solid homogeneous all clad with aluminum, stainless steel, or incoloy-800 and varying enrichment and weight percent of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

Group 11 U-Zr Hydride Fuel Inventories/Information		**Comment**
Radionuclide inventory (41 isotopes)	Refer to TSPA group listing data – Appendix D, Table D-1	
Composition	U-Zr hydride	
Matrix dissolution rate	Commercial model x 0.1	See Section 3.11 of the DOE SNF Information in Support of TSPA-SR report for detail
Surface area (m^2/g)	1.0E-04	
Clad failure fraction	Assume 0.1	
Free radionuclide inventory fraction	0.00001	
Gap fraction	0.00001	

Configuration and Package Count

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

		# 5x1 10 ft	# 5x1 15 ft	# HIC	# 0x4 15 ft	PWR21 x 15 ft	BWR x 15 ft
Group 11	U-Zr hydride Fuel						
	- repository pkg count	10	34	7			
	- HLW can count	50	170	35			
	- SNF pkg count	10	34	7			



Standard TRIGA Fuel Element

Figure A-13. Standard TRIGA fuel element.

REFERNCES

a. Lockheed-Martin Unclassified Nuclear Fuel Information System (LUNFIS), INEEL Spent Nuclear Fuel Program.

b. Hanford Irradiated Fuel Inventory Baseline, WHC-SD-CP-TI-175, February 1993.

c. Dry Storage of N Reactor Fuel Independent Technical Assessment, Appendix G - Corrosion and Dry Storage of N Reactor Fuel - R. G. Ballinger, B. Johnson, and K. A. Simpson.

DOE Spent Nuclear Fuel Information In Support of TSPA - SR

Appendix B
DOE SNF Surface Area Calculation

TITLE: Fuel Surface Area Calculation

DOCUMENT IDENTIFIER: EDF-NSNF-005

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Fuel Surface Area Calculation

Revision 00

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Calculation Executive Summary

This calculation estimates the Department of Energy (DOE) spent nuclear fuel (SNF) fuel surface area. The DOE SNF fuel surface area will be used in the total system performance assessment site recommendation (TSPA-SR), and the total system performance assessment license application (TSPA-LA) to determine the radionuclide releases (dose contribution) from the DOE SNF. The inputs used in this calculation are traceable to the DOE SNF sites and compliance to the RW/0333P requirements has not been determined at this time. This calculation was performed under the control of the NSNF QA program using appropriate procedure(s). This calculation and spreadsheets at the end of the calculation are on a CD-R disk.

CONTENTS

TABLE

ACRONYMS

ATR	Advanced Test Reactor
DOE	Department of Energy
FERMI	Enrico Fermi Reactor
FFTF-DFA	Fast Flux Test Facility-Driver Fuel Assembly
FRR	foreign research reactor
FSVR	Fort St. Vrain Reactor
heu	high enriched uranium (>20% U-235 equivalent)
HWCTR	Heavy Water Component Test Reactor
leu	low enriched uranium (<5% U-235 equivalent)
LWBR	light water breeder reactor
INEEL	Idaho National Engineering and Environmental Laboratory
meu	medium enriched uranium (>5% <20% U-235 equivalent)
MOX	mixed oxide
MTHM	metric tons of heavy metal
MWd/MTHM	SNF burnup in megawatt-days/metric ton heavy metal
PWR	pressurized water reactor
QA	Quality Assurance
QARD	Quality Assurance and Requirements Description
ROD	Record of Decision
RW	Office of Civilian Radioactive Waste Management
SNF	spent nuclear fuel

TMI	Three Mile Island
TRIGA	Training, Research, and Isotope General Atomic
TSPA	Total System Performance Assessment
SPA-SR	Total System Performance Assessment-Site Recommendation
$UZrH_x$	uranium zirconium hydride
Y2K	year 2000

1.0 PURPOSE

The Total System Performance Assessment-Site Recommendation (TSPA-SR) modeling logic is being refined by the Office of Civilian Radioactive Waste Management (RW), and the Department of Energy (DOE) Spent Nuclear Fuel (SNF) Programs are providing new inputs to support the refined detail in the model. In particular, the DOE SNF dissolution rates are being refined and will require the surface area of fuel per gram of fuel material. This calculation is performed on the basis of a specific surface area in square meters per gram of fuel meat (m2/g).

2.0 METHOD

The calculation approach was based on surface area estimate by simple geometric shapes using a specific representative fuel's dimensions in each group (neglecting cladding thickness). To allow for surface roughness effects, the surface areas were in most cases increased by a factor of 5. This should be a conservative value that will exceed the actual fuel area.

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

For the uranium and thorium carbide graphite fuel, the calculation was performed by Bob Kirkham of the INEEL SNF Program. The result of his calculation was used here.

3.0 ASSUMPTIONS

- Fuels listed as intact in the SNF Data Base are assumed to be intact.
- For each fuel group, the surface area of the fuel per gram of fuel material will be based on a simple geometric shape for the entire group.
- A factor of 5 due to surface roughness will be included in the surface area calculation where appropriate.

4.0 USE OF COMPUTER SOFTWARE

In performing this engineering calculation, the following computer software was used. Microsoft® Excel 97 SR2 program load on a DELL OptiPlex GXMT 5166 was used to generate the template tables and fuel radionuclide inventory at the end of the calculation. The computer has been certified to be year 2000 (Y2K) ready according to Lockheed Martin Idaho Technologies Company's (the INEEL management and operation contractor) Y2K desktop ready plan.

Computer software: Microsoft® Excel 97 SR2
Computer Hardware: DELL OptiPlex GXMT 5166

5.0 CALCULATION

5.1 DOE SNF Grouping Proposed for Site Recommendation

Table 1 is a list of the fuel groups that have been proposed for used in the total system performance assessment site recommendation (TSPA-SR). The groups were purposed based on the fuel matrix (compound). In the table, the MTHM and fuel package count came from the spread sheet 11RW_input399.xls that summarizes MTHM and package in each of the DOE SNF group based on inputs from the DOE SNF Data Base version 3.4.0 and the EIS ROD responsible sites (Hanford, INEEL, and SRS). The proposed groups have been reduced from 15 DOE SNF groups in the viability assessment to 11 groups indicated.

Table 1 Fuel Group, MTHM, and Package Count

Fuel Group	Fuel Matrix	Typical Fuel in the group	MTHM/Packages/Comment (based on ~2,496 MTHM)
1	*Classified*	NAVAL [151]	65 MTHM By Navy
2	*Pu/U Alloy*	FERMI CORE I & 2 (STD FUEL SUBASSEMBLY) [456]	9.1 MTHM 213 packages
3	*Pu/U Carbide*	FFTF-TFA-AC-3 [319]	0.1 MTHM 3 packages
4	*Pu/U Oxide (MOX) & Pu Oxide*	FFTF-DFA/TDFA [71]	12.4 MTHM 677 packages
5	*Th/U Carbide*	FSVR [86]	26.3 MTHM 561 packages
6	*Th/U Oxide*	SHIPPINGPORT LWBR REFLCT. IV [371]	50.4 MTHM 67 packages
7	*U Metal*	N REACTOR [147]	2127.2 MTHM 120 packages(4MCO/pkg)
8	*U Oxide*	TMI-2 CORE DEBRIS [229]	182.4 MTHM 760 packages
9	*Aluminum Based Fuel (UAlx, U3Si2, U Oxide in Al)*	FRR PIN CLUSTER U3SI2-LEU CANADA [660]	16 MTHM 1081 packages
10	*Unknown*	MISCELLANEOUS FUEL [366]	4.5 MTHM 9 packages
11	*U-Zr-Hx*	TRIGA (ALUM) [235]	1.62 MTHM 53 packages

5.2 Example Calculations

Following are examples of fuel surface area calculation for the various DOE SNF groups. These example calculations show how the surface area of each group will be estimated.

U metal Fuel

Representative Type: N reactor

Dimension of the N-reactor fuel:
Table 2 shows the dimensions of the N-reactor fuel. The source of the table is the Hanford Irradiated Fuel Inventory Baseline, WHC-SD-CP-TI-175, February 1993.

The dimensions of the Mark IV fuel were arbitrary selected for the U metal fuel surface area calculation. The Mark IV fuel element has the following properties.

Design:	Double annulus design
Length:	66.3 cm long (longest fuel length)
Outer annulus outside diameter:	6.15 cm OD
Outer annulus inside diameter:	4.32 cm ID
Inner annulus outside diameter:	3.25 cm OD
Inner annulus inside diameter:	1.22 cm ID
Uranium loading (Total)	23.68 kg U per element

Surface Area Calculation

Surface area of two annular sections

$$Surface\ Area = Length * \pi * [OD_{outer} + ID_{outer} + OD_{inner} + ID_{inner}] + 2 * \frac{\pi}{4} * (OD_{outer}^2 - ID_{outer}^2) + \frac{\pi}{2} * (OD_{inner}^2 - ID_{inner}^2)$$

$= 66.3\ [cm]\ \pi\ [6.15+4.32+3.25+1.22]\ [cm] + \pi/2*(6.15^2-4.32^2) + \pi/2*(3.25^2-1.22^2)$
$\cong 3{,}156\ cm^2$

Table 2 N Reactor fuel element description.

	Mark IV				Mark IA		
Pre-irradiation enrichment of U235	0.947% Enriched				1.25-0.947% Enriched		
Type-Length code [a]	E	C	S	A	M	F	T
Outer length (cm)	66.3	62.5	58.9	44.2	53.1	49.8	37.8
Element diameter (cm)							
1. Outer of outer	6:15				6.1		
2. Inner of outer	4.32				4.5		
3. Outer of inner	3.25				3.18		
4. Inner of inner	1.22				1.12		
Cladding weight (kg)							
I. Outer element	1.09	1.04	0.99	0.79	0.88	0.83	0.66
2. Inner element	0.55	0.52	0.50	0.40	0.24	0.51	0.40
Weight of uranium in outer (kg)							
1. (0.947% 235U)	15.96	15.01	14.15	10.48			
2. (1.25% 235U)					11.07	10.39	7.85
Weight of uranium Inner (kg) 0.947%	7.48	7.03	6.62	4.94	5.49	5.12	3.90
Weighted average of uranium in element (kg)	22.68				16.28		
Ratio of Zircaloy-2 to uranium (kg/MT)	70.0	70.8	71.6	77.1	85.5	86.3	90.4
Weighted ave. (kg/MT)	63.76				77.73		
% of total elements	63				37		
% of length type of each fuel	78	10	7	5	87	10	3
Displacement Volume(l/MT uranium)	66.77				66.77		

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

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{3{,}156\ cm^2 * [\frac{1\ m^2}{10{,}000\ cm^2}]}{22.68\ kg * 1{,}000\ \frac{g}{kg}} \cong 1.4 x 10^{-5}\ \frac{m^2}{g}$$

With the roughness factor of 5, we have 1.4 x 10^{-5} m^2/g x 5 (roughness) ≅ **7 x 10^{-5} m^2/g**

U Zr Alloy Fuel

***Representative Type*:** HWCTR

Dimension of the HWCTR fuel:
Below are the dimensions of the HWCTR fuel. The source is the Director of Nuclear Reactors, Vol V, published by the International Atomic Energy Agency, Vienna, 1964, page 229. The uranium loading is the average value from the SNF Data Base records 117,and 783.

The dimensions of the HWCTR fuel were selected for the U Zr Alloy fuel surface area calculation. The HWCTR fuel element has the following properties.

Design:	Annulus element design
Length:	290 cm long
Annulus outside diameter:	5.84 cm OD
Annulus inside diameter	4.98 cm ID
Uranium loading EOL (Average)	4,259 g U per element

Surface Area Calculation

Surface area of annulus

$$Surface\ Area = Length * \pi * [OD + ID] + 2 * \frac{\pi}{4} * (OD^2 - ID^2)$$

$$= 290\ [cm]\ \pi\ [5.84+4.98]\ [cm] + (\pi/2)*(5.84^2-4.98^2) \cong 9{,}866\ cm^2$$

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{9{,}866\ cm^2 * [\frac{1\ m^2}{10{,}000\ cm^2}]}{4{,}259\ g} \cong 2.3 \times 10^{-4}\ \frac{m^2}{g}$$

With the roughness factor of 5, we have 2.3 x 10^{-4} m^2/g x 5 (roughness) ≅ **1.2 x 10^{-3} m^2/g**

U Mo Alloy Fuel

***Representative Type*:** Fermi

Dimension of the Fermi fuel:

Below are the dimensions of the Fermi fuel. The source is the Fermi (UMo) Fuel Characteristics for Disposal Criticality Analysis, DOE/SNF/REP-035 pages 4, and 7.

The dimensions of the Fermi fuel pin were selected for the U Mo Alloy fuel surface area calculation. The Fermi fuel pin has the following properties.

Design:	Pin type design
Length:	~84 cm long
Pin outside diameter:	~0.4 cm OD
U loading (Total)	133.9 g U per pin

Surface Area Calculation

Surface area of pin

$$Surface\ Area = Length * \pi * [OD] + 2 * \frac{\pi}{4} * (OD^2)$$

$$= 84\ [cm]\ \pi\ [0.4]\ [cm] + 2*\pi/4*(0.4^2) \cong 105.8\ cm^2$$

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{105.8\ cm^2 * [\frac{1\ m^2}{10{,}000\ cm^2}]}{133.9\ g} \cong 7.9 \times 10^{-5}\ \frac{m^2}{g}$$

With the roughness factor of 5, we have 7.9×10^{-5} $m^2/g \times 5$ (roughness) $\cong$ **4×10^{-4} m^2/g**

U Oxide Fuel

***Representative Type*:** Commercial

The source of the surface area for the commercial fuel is the Viability Assessment of a Repository at Yucca Mountain, Volume 3 Total System Performance Assessment, September 10, 1998, page 4-13.

Surface Area per gram of fuel material

Since the U Oxide fuel in the DOE SNF inventory is the same as the RW commercial fuel, the same surface area as commercial reactor fuel used by RW will be used for the DOE U Oxide fuel. See reference TSPA-VA page indicated above.

Surface area $\cong$ **4×10^{-3} m^2/g**

Failed U Oxide Fuel

***Representative Type*:** TMI

The TMI fuel has the same properties as Commercial fuel except that the fuels have been disrupted from the accident. The source of the surface area is the Viability Assessment of a Repository at Yucca Mountain, Volume 3 Total System Performance Assessment, September 10, 1998.

Surface Area per gram of fuel material

Since the failed U Oxide fuel in the DOE SNF inventory is the same as the RW commercial fuel, the same surface area as commercial reactor fuel used by RW will be used for the DOE U Oxide fuel. However, because of the accident, a number of the fuel elements have been damaged and in some cases, in pieces or as fines. Thus, 100 time the surface area as the commercial reactor fuel was selected to represent the failed DOE U Oxide fuel. See Parameter Selection for Department of Energy Spent Nuclear Fuel to be used in the Yucca Mountain Viability Assessment for more detail.

Surface area $\cong 4 \times 10^{-1}$ m^2/g

UAlx Alloy Fuel

***Representative Type*: ATR**

Dimension of the ATR fuel:
Below are the dimensions of the ATR fuel. The source is the Director of Nuclear Reactors, Vol V, published by the International Atomic Energy Agency, Vienna, 1964, page 108. The uranium loading is from the INEEL letter JAH-195-85 Attachment page 1.

The dimensions of the ATR fuel were selected for the U Alx Alloy fuel surface area calculation. The ATR fuel element has the following properties.

Design:	Plate type design
Length:	~122 cm long
Width	5.8 to 10.8 cm wide
	(Mean width (5.8+10.8)/2 $\cong$ 8.3 cm
Number of plates	19 per assembly
Fuel loading (Total)	3,020 g per assembly

Surface Area Calculation

Surface area of plate

$$Surface\ Area = Length * width * 2\ sides * Number\ plates$$

$$= 122\ [cm]\ 8.3\ [cm]\ [2]\ [19] \cong 38{,}500\ cm^2$$

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{38{,}500\ cm^2 * [\frac{1\ m^2}{10{,}000\ cm^2}]}{3{,}020\ g} \cong 1.3 x 10^{-3}\ \frac{m^2}{g}$$

With the roughness factor of 5, we have 1.3 x 10^{-3} m^2/g x 5 (roughness) ≅ **6.5 x 10^{-3} m^2/g**

U_3Si_2 Alloy Fuel

***Representative Type*:** FRR-MTR

Dimension of the FRR-MTR fuel:
Below are the dimensions of the FRR-MTR fuel. The source is the DOE/EIS-0218F, Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, Appendix B page B-10, B-11.

The dimensions of the FRR-MTR fuel were selected for the U_3Si_2 Alloy fuel surface area calculation. The FRR MTR fuel element has the following properties.

Design:	Plate type design
Length:	~77.8 cm long
Width	7.08 cm wide
Number of plates	19 per assembly
Fuel loading (Total)	3,840 g per assembly (202 g/plate*19 plate)

Surface Area Calculation

Surface area of plate

$$Surface\ Area = Length * width * 2\ sides * Number\ plates$$

$$= 77.8\ [cm]\ 7.08\ [cm]\ [2]\ [19] \cong 21{,}000\ cm^2$$

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{21{,}000\ cm^2 * [\frac{1\ m^2}{10{,}000\ cm^2}]}{3{,}840\ g} \cong 5.5 \times 10^{-4}\ \frac{m^2}{g}$$

Adding the roughness factor of 5, we have 5.5×10^{-4} $m^2/g \times 5$ (roughness) $\cong$ **2.8×10^{-3} m^2/g**

<u>High Integrity UThC Fuel</u>

***Representative Type*:** FSV

Surface Area per gram of fuel material

The surface area of the FSV fuel was performed by Bob Kirkham of the INEEL Program. The calculation was reported in a letter and is indicated in the reference section.

Bob Kirkham estimated the surface of the FSV areas per g HM ranges from 5.9×10^{-3} to 2.2×10^{-2} m^2/g depending on the particle size distribution. For the purpose of the TSPA, the largest surface area was selected and indicated below.

Surface area $\cong$ **2.2×10^{-2} m^2/g**

<u>Low Integrity UThC Fuel</u>

***Representative Type*:** Peach Bottom Core I

Surface Area per gram of fuel material

The Peach Bottom fuel was fabricated in a similar manner as the FSV fuel. Thus, the surface area for the FSV fuel could also be applied here.

Bob Kirkham estimated the surface of the FSV areas per g HM ranges from 5.9×10^{-3} to 2.2×10^{-2} m^2/g depending on the particle size distribution. For the purpose of the TSPA, the largest surface area was selected and indicated below.

Surface area $\cong$ **2.2×10^{-2} m^2/g**

U/PuC Non Graphite Fuel

***Representative Type*:** FFTF Carbide

Dimension of the FFTF Carbide fuel:
Below are the dimensions of the FFTF Carbide fuel. The source is the Fuel fabrication processes, design and experimental conditions for the joint US-Swiss mixed carbide test in FFTF (AC-3 test), Journal of Nuclear Materials 201 (1993) 39-49, Figure 6.

The dimensions of the FFTF Mixed Carbide fuel (Sphere pac) were selected for the non-graphite carbide fuel surface area calculation. The FFTF Mixed Carbide fuel (Sphere pac) fuel pins have the following properties.

Design:	Spherical particles	
Particle Diameter:	Pu/UC-spheres	0.0805 cm
Particle specific gravity		12.97 g/cm^3

Surface Area Calculation

Surface area of sphere

$$Surface\ Area = (\pi)*(d^2)$$

$$= \pi\ 0.0805^2\ [cm^2] \cong 2.04 \times 10^{-2}\ cm^2$$

Volume of the sphere

$$Volume = \frac{4}{3}(\pi)*(R^3)$$

$$= (4/3)*(\pi)*(.04025)^3\ [cm^3] \cong 2.73 \times 10^{-4}\ cm^3$$

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{2.04x10^{-2}\ [cm^2]*\frac{1}{10{,}000}\left[\frac{m^2}{cm^2}\right]}{2.73x10^{-4}\ [cm^3]*12.97\left[\frac{g}{cm^3}\right]} \cong 5.8 \times 10^{-4}\ \frac{m^2}{g}$$

No roughness factor was used for the sphere. Surface area ≅ **5.8 x 10^{-4} m^2/g**

Mixed Oxide Fuel

***Representative Type*:** FFTF

Since the MOX fuel in the DOE SNF inventory is the similar to the RW commercial fuel, the same surface area as commercial reactor fuel will be used for the DOE MOX fuel. See reference under U Oxide fuel indicated above.

Surface Area per gram of fuel material

Surface area ≅ **4 x 10^{-3} m^2/g**

UTh Oxide Fuel

***Representative Type*:** Shippingport LWBR Fuel

Dimension of the LWBR fuel:
Below are the dimensions of the Shippingport LWBR fuel. The source of the table is the Fuel Summary Report: Shippingport Light Water Breeder Reactor, INEEL/EXT-98-00799 Rev.1 Tables 3-6, and 3-8.

The dimensions of the Shippingport LWBR fuel seed element was selected for the UTh oxide fuel surface area calculation. The Shippingport LWBR fuel seed element (UO_2-ThO_2 fuel pellet) has the following properties.

Design:	Pellets
Pellet dimensions (Seed)	0.64 cm Diameter, 1.562 cm long
Density of fuel meat (Seed)	10.035 g/cc

Surface Area Calculation

Surface area of the pellet

$$Surface\ Area = \pi * D * Length + 2 * (\pi * \frac{D^2}{4})$$

$$= \pi * 0.64\ [cm]\ 1.562\ [cm] + 2\ (\pi * 0.64^2/4)\ [cm^2] \cong 3.78\ cm^2$$

Volume Calculation

Volume of the pellet

$$Volume = \pi * \frac{D^2}{4} * Length$$

$$= \pi * (0.64^2/4) * 1.562\ [cm^2] \cong 0.5025\ cm^3$$

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{3.78\ [cm^2] * \frac{1}{10{,}000}\ [\frac{m^2}{cm^2}]}{0.525\ [cm^3] * 10.035\ [\frac{g}{cm^3}]} \cong 7.2x10^{-5}\ \frac{m^2}{g}$$

With the roughness factor of 5, we have 7.2 x 10^{-5} m^2/g x 5 (roughness) = **3.6 x 10^{-4} m^2/g**

U Zr H_x Fuel

***Representative Type*:** TRIGA

Dimension of the TRIGA fuel:
Below are the dimensions of the TRIGA fuel. The source of the information is the Lockheed Martin Unclassified Nuclear Fuel Information System (LUNFIS), INEEL Spent Fuel Program, TRIGA stainless fuel Detailed Fuel Information page.

The dimensions of the TRIGA Stainless fuel were selected for the U Zr H_x Alloy fuel surface area calculation. The TRIGA Stainless fuel element has the following properties.

Design:	Rod type design
Length:	~38.1 cm (15 inch)
Pin outside diameter:	~3.65 cm (1.435 in)
Fuel loading (Total)	2283 g per Rod

Surface Area Calculation

Surface area of pin

$$Surface\ Area = Length * \pi * [OD] + 2 * \pi * (\frac{D^2}{4})$$

$$= 38.1\ [\text{cm}]\ \pi\ [3.65]\ [\text{cm}] + 2*\pi*(3.65^2/4) \cong 457.8\ \text{cm}^2$$

Surface Area per gram of fuel material

$$\frac{Area}{g} = \frac{457.8\ cm^2 * [\frac{1\ m^2}{10{,}000\ cm^2}]}{2{,}283\ g} \cong 2.0 \times 10^{-5}\ \frac{m^2}{g}$$

With the roughness factor of 5, we have 2×10^{-5} $m^2/g \times 5$ (roughness) = **1×10^{-4} m^2/g**

Unknown Fuel

Representative Type: Miscellaneous Fuel

Dimension of the Miscellaneous fuel:
All the fuels in this group have unknown fuel properties. The largest entry is a Miscellaneous fuel presently located at ANL-W. Since no information is available on these fuels, the most conservative fuel group surface area should be used.

Surface Area per gram of fuel material

Since the fuels in this group have unknown properties, it is proposed that the surface area of the U Oxide (the largest in all of the DOE SNF group) be used in the TSPA evaluation..

Surface area = 4×10^{-1} m^2/g

6.0 RESULTS

The Attachment A Computer Calculation spreadsheet was made with Microsoft® Excel 97 SR2 program load on a DELL OptiPlex GXMT 5166. Only simple arithmetic operations were used to calculate the results.

The EXCEL spreadsheets used to perform the inventory calculation are also included on a CD-R.

7.0 REFERENCES

1. *SNF Data Base, Version 3.4.0 September 1998*
2. *CRC Handbook of Physic and Chemistry Ed 57*
3. *Hanford Irradiated Fuel Inventory Baseline, WHC-SD-CP-TI-175, February 1993*
4. *Director of Nuclear Reactors, Vol V, published by the International Atomic Energy Agency, Vienna, 1964*
5. *Fermi (UMo) Fuel Characteristics for Disposal Criticality Analysis, DOE/SNF/REP-035*
6. *Viability Assessment of a Repository at Yucca Mountain, Volume 3 Total System Performance Assessment, September 10, 1998*
7. *Surface Area of the FSV fuel Calculation, Bob Kirkham of the INEEL Program*
8. *INEEL letter JAH-195-85 Attachment page 1*
9. *DOE/EIS-0218F, Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, Appendix B page B-10, B-11*
10. *Fuel fabrication processes, design and experimental conditions for the joint US-Swiss mixed carbide test in FFTF (AC-3 test), Journal of Nuclear Materials 201 (1993) 39-49*
11. *Fuel Summary Report: Shippingport Light Water Breeder Reactor, INEEL/EXT-98-00799 Rev.1*
12. *Lockheed Martin Unclassified Nuclear Fuel Information System (LUNFIS), INEEL Spent Fuel Program*
13. *Characteristics of Potential Repository Wastes, DOE/RW-0184 R1 Volume 2 page 4.2-1*
14. *Parameter Selection for Department of Energy Spent Nuclear Fuel to be used in the Yucca Mountain Viability Assessment, INEEL/EXT-98-00666, June 1998*

8.0 ATTACHMENTS

Attachment A Computer Calculation of Surface Area

Attachment A contains a summary of spreadsheet that estimates the surface areas for all the DOE SNF groups. In cases where only the heavy metals were used in the surface area calculation, a correction factor is used to include other materials such as oxygen, carbon, zirconium, molybdenum, aluminum, silicon, and hydrogen in the fuel. The correction factors are shown under the column titled "Surface Area correction factor". The correction factors were developed in the sheet titled "Correct-factor" in the EXCEL file "SRfuelAreaB.xls."

Attachment A Computer Calculation of the SNF Surface Area

The following EXCEL spreadsheet summarizes the surface areas (per gram and package) used for the DOE SNF. For detailed calculation, see the spreadsheet file "SRfuelAreaB.xls."

TSPA Cat	Fuel Type	MTHM	Group Disposal pkg total	Surface Area Calc by geometry	Surface area basis	Roughness factor	Surface Area, m^2/g	Surface Area correction factor	Surface Area per Pkg m^2/pkg	Notes
1	Classified	65.00	NA	NA		NA	NA	NA	NA	
2	Pu/U Alloy	9.11	213		g U		1.2E-03	10.8	5.5E+02	
3	Pu/U Carbide	0.11	3	2.7E-03	g matrix	1	2.7E-03	1.1	1.0E+02	
4	Pu/U Oxide (MOX) & Pu Oxide	12.42	677		g U oxide		4.0E-03	1.1	8.3E+01	
5	Th/U Carbide	26.28	561	2.2E-02	g HM	1	2.2E-02	1.1	1.1E+03	
6	Th/U Oxide	50.35	67	7.2E-05	g matrix	5	3.6E-04	1.1	3.1E+02	
7	U Metal	2,127.24	120	1.4E-05	g U	5	7.0E-05	1.0	1.2E+03	
8	U Oxide	178.15	760		g U oxide		4.0E-01	1.1	1.1E+05	
9	Aluminium Based Fuel (UAlx, U3Si2, U Oxide in Al)	20.94	1,081		g matrix		6.5E-03	2.6	3.3E+02	
10	Unknown	4.54	9		g U oxide		4.0E-01	1.1	2.4E+05	
11	U-Zr-Hx	1.62	53	2.0E-05	g matrix	5	1.0E-04	11.7	3.6E+01	
Total		2,495.75	3,544							

DOE Spent Nuclear Fuel Information In Support of TSPA - SR

Appendix C
DOE SNF Volume Calculation

TITLE: Fuel Meat Volume Calculation

DOCUMENT IDENTIFIER: EDF-NSNF-004

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Fuel Meat Volume Calculation

Revision 00

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Calculation Executive Summary

This calculation estimates the Department of Energy (DOE) spent nuclear fuel (SNF) fuel meat volume. The DOE SNF fuel meat volume will be used in the total system performance assessment site recommendation (TSPA-SR), and the total system performance assessment license application (TSPA-LA) to determine the radionuclide releases (dose contribution) from the DOE SNF. The inputs used in this calculation are traceable to the DOE SNF sites and compliance to the RW/0333P requirements has not been determined at this time. This calculation was performed under the control of the NSNF QA program using appropriate procedure(s). This calculation and spreadsheets at the end of the calculation are on a CD-R disk.

CONTENTS

Tables

ACRONYMS

ATR	Advanced Test Reactor
DOE	Department of Energy
FERMI	Enrico Fermi Reactor
FFTF	Fast Flux Test Facility
FRR	foreign research reactor
FSVR	Fort St. Vrain Reactor
heu	high enriched uranium (>20% U-235 equivalent)
HWCTR	Heavy Water Component Test Reactor
leu	low enriched uranium (<5% U-235 equivalent)
LWBR	light water breeder reactor
INEEL	Idaho National Engineering and Environmental Laboratory
meu	medium enriched uranium (>5% <20% U-235 equivalent)
MOX	mixed oxide
MTHM	metric tons of heavy metal
MWd/MTHM	SNF burnup in megawatt-days/metric ton heavy metal
PWR	pressurized water reactor
QA	Quality Assurance
QARD	Quality Assurance and Requirements Description
ROD	Record of Decision
RW	Office of Civilian Radioactive Waste Management
SNF	spent nuclear fuel

TMI	Three Mile Island
TRIGA	Training, Research, and Isotope General Atomic
TSPA	Total System Performance Assessment
$UZrH_x$	uranium zirconium hydride
Y2K	year 2000

1.0 PURPOSE

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

2.0 METHOD

The calculation approach was based on the metric ton of heavy metal (MTHM) inventory from the DOE SNF Data Base and the number of packages estimated in each fuel group. One advantage of this method is that it depends only on MTHM and package count as measured fuel data. All other inputs are established physical constants such as molecular weight and density. The theoretical density is used in calculation. In practice, many fuels are fabricated as compacted particles of oxides or various mixtures, and these are pressed to a specified fraction of theoretical density, typically 98%. Frequently fuels also swell during irradiation, further reducing the density. These effects could increase the volume of the fuel by approximately 10%. For the purpose of this calculation, density reduction due to these processes of 10% will be included here (except no correction for the U-Alx, and U_3Si_2, fuel group and a 15% correction for the FFTF (MOX) group as indicated under assumptions). The reason is that the TSPA-SR model calculates the water volume available for radionuclide removal based on the volume of the SNF matrix. The higher the volume of SNF, the larger the volume of water available for radionuclide removal from the SNF (See Attachment C, e-mail note from Vinod Vallikat of RW).

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

3.0 ASSUMPTIONS

- DOE SNF Data Base Version 3.4.0
- MTHM and the number of packages per fuel group came from the spread sheet 11RW_input399.xls
- Material density came from Denny Fillmore - See attachment A.

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

- A 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc. This non-ideal density correction factor was not added to the U-Alx and U_3Si_2 fuel groups based on information provided by Allen Brewer of SRS. In addition, based on measurement done at Hanford, this non-ideal density correction for the FFTF fuel (MOX) group was increased to 1.15 (This information was provided by Alan Carlson of Hanford).

4.0 USE OF COMPUTER SOFTWARE

In performing this engineering calculation, the following computer software was used. Microsoft® Excel 97 SR2 program load on a DELL OptiPlex GXMT 5166 was used to generate the template tables and fuel radionuclide inventory at the end of the calculation. The computer has been certified to be year 2000 (Y2K) ready according to Lockheed Martin Idaho Technologies Company's (the INEEL management and operation contractor) Y2K desktop ready plan.

Computer software: Microsoft® Excel 97 SR2
Computer Hardware: DELL OptiPlex GXMT 5166

5.0 CALCULATION

5.1 DOE SNF Grouping Proposed for Site Recommendation

Table 1 is a list of the fuel groups that have been proposed for use in the total system performance assessment site recommendation (TSPA-SR). The groups were assigned by fuel composition. In the table, the MTHM and fuel package count came from the spread sheet 11RW_input399.xls that summarizes MTHM and package in each of the DOE SNF group based on inputs from the DOE SNF Data Base version 3.4.0 and the EIS ROD responsible sites (Hanford, INEEL, and SRS). The proposed groups have been reduced from 15 DOE SNF groups in the viability assessment to 11 group indicated.

Table 1 Fuel Group, MTHM, and Package Count

Fuel Group, MTHM, and Package Count			
Fuel Group	Fuel Matrix	Typical Fuel in the group	MTHM/Packages/Comment (based on ~2,496 MTHM, Packages rounded to whole #)
1	*Classified-Navy*	NAVAL [151][Note Below]	65 MTHM By Navy
2	*Pu/U Alloy*	FERMI CORE I & 2 (STD FUEL SUBASSEMBLY) [456]	9.11 MTHM 213 packages
3	*Pu/U Carbide*	FFTF-TFA-AC-3 [319]	0.11 MTHM 3 packages
4	*Pu/U Oxide (MOX) & Pu Oxide*	FFTF-DFA/TDFA [71]	12.42 MTHM 677 packages
5	*Th/U Carbide*	FSVR [86]	26.28 MTHM 561 packages
6	*Th/U Oxide*	SHIPPINGPORT LWBR REFLCT. IV [371]	50.35 MTHM 67 packages
7	*U Metal*	N REACTOR [147]	2127.24 MTHM 120 packages(4MCO/pkg)
8	*U Oxide*	TMI-2 CORE DEBRIS [229]	178.15 MTHM 760 packages
9	*Aluminum Based Fuel (UAlx, U3Si2, U Oxide in Al)*	FRR PIN CLUSTER U3SI2-LEU CANADA [660]	20.94 MTHM 1081 packages
10	*Unknown*	MISCELLANEOUS FUEL [366]	4.54 MTHM 9 packages
11	*U-Zr-Hx*	TRIGA (ALUM) [235]	1.62 MTHM 53 packages

Note:The number in [ ] is the fuel identification used in the DOE SNF Database Version 3.4.0

5.2 Example Calculations

Following are examples of volume calculation for the various DOE SNF forms. These example calculations show how the volume of each group will be calculated. Not all of the examples are

used in the TSPA-SR SNF volume determination. The complete DOE SNF volume calculations are performed using an EXCEL spreadsheet based on the method discussed here and are included as file SRfuelvol00.xls attached.

Example of U metal Fuel Volume Calculation

Fuel Consisting of U metal: N-Reactor Fuel

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

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

From the DOE SNF Data Base version 3.4.0, the quantity of heavy metal in group 7 is 2,127.2 metric tons. Heavy metal includes all the uranium, plutonium, and thorium that are in the group. After irradiation, the U metal SNF will also contain small quantities of plutonium and thorium. But for the purpose of this volume calculation, the heavy metal will be assumed to be 100% uranium. Based on this assumption, a density of 19.05 g/cc (or 19.05 metric ton/m^3) could be used to represent the U metal fuel.

In support of the TSPA-SR effort, the DOE sites have indicated that all the U metal fuels (group 7) could be placed into 120 packages for repository disposal (See Table 1 above). Thus, the average quantity of MTHM per package may be calculated using equation 2.

$$\frac{MTHM}{package} = \frac{[Total\ MTHM\ (group\ 7)]}{[Total\ No.\ packages\ (group\ 7)]} \qquad (2)$$

If uranium is assumed to makes up 100% of the heavy metal, the MT U in equation 3 would be the same as the total MTHM shown in equation 2 above.

$$\frac{MT\,U}{package} = \frac{[Total\ MTHM\ (group\ 7)]}{[Total\ No.\ packages\ (group\ 7)]} \qquad (3)$$

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

$$\frac{volume}{pkg} = [\frac{MT\,U}{pkg}][\frac{1}{\rho(U)}] = [\frac{2127.24}{120}][\frac{MT\,U}{pkg}][\frac{1}{19.05}][\frac{m^3}{MT\,U}][1.1] \cong 1.0x10^0\ \frac{m^3}{pkg} \qquad (4)$$

Example of U Alloy Fuel Volume Calculation

Fuel Consisting of Uranium alloy: U Zr Alloy Fuel

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

$$volume = [mass]\left[\frac{Fraction\,(A)}{\rho(A)} + \frac{Fraction\,(B)}{\rho(B)} + \ldots + \frac{Fraction\,(n)}{\rho(n)}\right] \quad (5)$$

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

$$volume = [MT\,U]\left[\frac{(Alloy\ mass\ Fraction)}{(U\ mass\ Fraction)}\ \frac{1}{\rho(Alloy)}\right] \quad (6)$$

In the case of U-Zr alloy fuel, Attachment A reported that uranium makes up 9.3% of the total mass and Zr makes up the 90.7% of the total mass. Using a Zr density of 6.49 g/cc (or 6.49 MT/m^3), 2.93 MTHM and 22 packages for U-Zr fuel (See FILE Srfuelvol00.xls under UZr fuel type), and the assumption that U makes up 100% of all the heavy metal, the volume per package of the U-Zr fuel could be represented by equation 7. As with the metal fuel, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation,

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

etc.

$$\frac{volume}{pkg} = \left[\frac{2.93}{22}\right]\left[\frac{MT\,U}{pkg}\right]\left[\frac{1}{19.05} + \frac{(.907)}{(.093)}\ \frac{1}{6.49}\right]\left[\frac{m^3}{MT\,U}\right][1.1] \cong 2.3x10^{-1}\ \frac{m^3}{pkg}$$

or

This approach will also be used for the other alloy fuels in group 2, with suitable alloy mass fractions and densities for the U-Mo, U-Th, and Pu-Fe fuels. See file Srfuelvol00.xls Computer

Calculations for the volume of the other alloy fuels. Since the fuel volume vary significantly depending on fuel types. Thus, an average volume per package of 3.6×10^{-2} m^3/pkg was calculated based on the quantities of the four types of fuel in the group. See the EXCEL spreadsheet for the calculation

Example of Oxide Fuel Volume Calculation

Fuel Consisting of Oxide: Uranium Oxide Fuel

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

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

or

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

or

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

This factor is applied to group 8 (UO_2 fuels) to arrive at the correct fuel meat mass. Then the appropriate oxide density is used to determine the volume of the fuel. Using the DOE SNF Data Base version 3.4.0, 178.15 MTHM and 760 disposal packages for group 8 (See group 8 Table 1 above), and a uranium oxide density of 10.96 (See Attachment A), the volume of group 8 fuel may be determined using equation 9 below. As noted earlier, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.
For the volume of other oxide fuels, see Table 2 Computer Calculation at the end of this calculation.

$$\frac{volume}{pkg} = [\frac{178.15}{760}][\frac{MTHM}{pkg}][\frac{1.134}{10.96}][\frac{MT(UO_2)}{MTHM}][\frac{m^3}{MT(UO_2)}][1.1] = 2.7 \times 10^{-2}\ \frac{m^3}{pkg} \qquad (9)$$

Example of Fuel Volume Calculation with Several Heavy Metal Compounds

Fuel Consisting of Several Heavy Metal Compounds: Ft. St. Vrain Fuel

In cases where more than one element contributes to MTHM, the relative mass fractions of each of the heavy metal compounds are required. In all cases, the mass fractions refer to the elements that comprise MTHM. For instance, U mass fraction = U mass/ MTHM. In group 5, for example, the fuel is 94.8% Th and the Th mass fraction is therefore 0.948 of the heavy metal inventory.

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

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

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

Where:

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

Using group 5 as an example, equation 11 below calculates the volume of the Fort Saint Vrain fuel that contain both uranium and thorium in the form of carbide. The DOE SNF Data Base shows that there are 26.3 MTHM and 561 disposal packages for group 5 (See group 5 Table 1 above). Of the total heavy metal, uranium makes up 5.2% and thorium makes up 94.8% of the mass (See Attachment A). The uranium and thorium carbides have a density of 11.8 and 8.96 g/cc (or 11.8 and 8.96 MT/m^3) respectively. To correct for the additional mass due to the carbon, the MW/AW for U carbide and Th carbide of 1.1 is included (i.e., UC_2/U = (238+24/238)

= 1.1 and $ThC_2/Th = (232+24/232) = 1.1$). And finally, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.

$$\frac{volume}{pkg} = [\frac{26.28}{561}][\frac{MTHM}{pkg}][\frac{(0.052)(1.1)}{11.8} + \frac{(0.948)(1.1)}{8.96}][\frac{m^3}{MT}][1.1] = 6.3x10^{-3}\ \frac{m^3}{pkg} \quad (11)$$

A similar approach is used for groups 2, 3, 4, and 6. For the volume of group 2, 3, 4, and 6 fuels, see file Srfuelvol00.xls Computer Calculation at the end of this calculation.

Example of Calculations by Volumetric Methods

In some cases, the fuel composition has a varying range or was uncertain. Thus, it was more straightforward to calculate a volume of fuel meat based on fuel drawings or specifications and attribute it to the Uranium content of the fuel. This was true of groups 9 (UAlx+Al, and U_3Si_2), and 11 (U-Zr-Hx). Attachment B contains the volumetric calculations for the three fuel types.

Aluminum Based Fuel

For the UAlx+Al and U_3Si_2 fuel, the ATR fuel specifications were used to determine both the volume of fuel meat and MTHM per element. Attachment B calculation shows that for the ATR fuel, each element contains 0.000763 m^3 of fuel meat and 0.001156 MTHM. The DOE SNF Data Base shows that there are 7.71 MTHM and 585 disposal packages for UAlx+Al fuel in group 9 (See file Srfuelvol00.xls attached). The volume per disposal package could be calculated using equation 12. As noted earlier, no density correction was added to the equation.

$$\frac{volume}{pkg} = [\frac{7.71}{585}][\frac{MTHM}{pkg}][\frac{0.000763}{0.001156}][\frac{m^3/element}{MTHM/element}] \cong 8.7x10^{-3}\ \frac{m^3}{pkg} \quad (12)$$

For U_3Si_2 fuel, the SNF Data Base shows that there are 8.3 MTHM in 198 packages (See file Srfuelvol00.xls attached). The volume is estimated using the fuel information was based on a letter from James Snelgrove of Argonne National Laboratory (See Attachment B). The letter shows that for the U_3Si_2 fuel, the density of the fuel meat is 3.5 g(U)/cc (or 3.5 MT/m^3). The volume per disposal package could be calculated using equation 13. As noted earlier, no density correction was added to the equation.

$$\frac{volume}{pkg} = [\frac{8.3}{198}][\frac{MTHM}{pkg}][\frac{1}{3.5}][\frac{m^3}{MTHM}] \cong 1.2x10^{-2}\ \frac{m^3}{pkg} \quad (13)$$

It appears that the aluminum based fuel volume could vary significantly depending on fuel types. In addition, group 9 includes also a fuel with U_3O_8 dispersed in the aluminum matrix. Thus, an average volume per package of $1.0x10^{-2}$ m^3/pkg was calculated based on the quantities of the three types of fuel in the group. See the file Srfuelvol00.xls attached for the calculation.

U-ZrHx Fuel

For the U-Zr-Hx fuel, the TRIGA fuel specifications were used to determine both the volume of fuel meat and MTHM per element. Attachment B calculation shows that for the TRIGA fuel, each element contains 388 cm^3 of fuel meat and 195 g.U (or ~0.5 MTHM/m^3). The DOE SNF Data Base shows that there are 1.6 MTHM and 53 disposal packages for group 11 (See group 11 Table 1 above). The volume per disposal package could be calculated using equation 14. As noted earlier, a 10% non-ideal density correction (1.1 multiplier) was added to the equation for fuel swelling after irradiation, etc.

$$\frac{volume}{pkg} = [\frac{1.62}{53}][\frac{MTHM}{pkg}][\frac{1}{0.502577}][\frac{m^3}{MTHM}][1.1] \cong 6.7x10^{-2}\frac{m^3}{pkg} \qquad (14)$$

Other Volume Determination for Uranium Oxide Fuel

The value arrived at independently by RW for use with commercial oxide fuels of 1.1126 m^3/21 PWR element package equates to 0.115 m^3/MT. This agrees closely with the value of 0.113 m^3/MT by the above method for the uranium oxide fuel.

6.0 RESULTS

The Attachment C Computer Calculation spreadsheet was made with Microsoft® Excel 97 SR2 program load on a DELL OptiPlex GXMT 5166. Only simple arithmetic operations were used to calculate the results.

The EXCEL spreadsheets (File Srfuelvol00.xls) used to perform the inventory calculation are also included with this calculation on the CD-R.

7.0 REFERENCES

1. *SNF Data Base, Version 3.4.0 September 1998*
2. *CRC Handbook of Physic and Chemistry Ed 57*
3. *Technical Strategy for the Management of INEEL Spent Nuclear Fuel, March 1997*

8.0 ATTACHMENTS

Attachment A Densities and Fuel Composition Values

Attachment B Hand Calculation Results

Attachment C EXCEL Spreadsheet Results

Attachment D E-mail note from Vinod Vallikat

Attachment A Densities and Fuel Composition Values

Denny Fillmore provided the following densities and fuel composition values. The physical constants were taken from the CRC Handbook of Physic and Chemistry Ed 57. The fuel composition data were taken from the report of the INEEL Spent Nuclear Fuel task team "*Technical Strategy for the Management of INEEL Spent Nuclear Fuel*" which was in turn taken from fuel receipt criteria and from The Research, Training, Test and Production Reactor Directory, 3rd edition, 1988.

Note that not all of the information was used in the volume calculation.

Data for DOE SNF by Fuel Type

Uranium Metal
Density of uranium metal

$$\rho = 19.05 \quad g/cc$$

Uranium Zirconium Alloy
Uranium Zirconium mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Zr mass ratio 9.3% U 90.7% Zr Density $\rho_U = 19.05$ $\rho_{Zr} = 6.49$

$$\frac{1}{\rho} = \frac{0.093}{19.05} + \frac{.907}{6.49} \qquad \rho = 6.91 \quad g/cc$$

Uranium Molybdenum Alloy
Uranium Molybdenum mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Mo mass ratio 90% U 10% Mo Density $\rho_U = 19.05$ $\rho_{Mo} = 10.2$

$$\frac{1}{\rho} = \frac{0.9}{19.05} + \frac{0.1}{10.2} \qquad \rho = 17.5 \quad g/cc$$

Intact Uranium Oxide
Uranium Plutonium mass ratio, density of the fuel matrix

U-Pu mass ratio 99% U 1% Pu

$\rho = 10.96 \ g/cc$

Failed Uranium Oxide
Density of the fuel matrix

$\rho = 10.96 \ g/cc$

Uranium Alloy or Uranium Oxide or Uranium Aluminide in an aluminum matrix
Uranium Alloy mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Al mass ratio 71% U 29% Al Density $\rho_{UAlx} = 6.3$ $\rho_{Al} = 2.7$ $\rho_{U3O8} = 8.3$

Use average density of oxide and UAlx for U $\rho_{Ave} = (6.3+8.3)/2 = 7.3$

$$\frac{1}{\rho} = \frac{.71}{7.3} + \frac{.29}{2.7} \qquad \rho = 4.89 \ g/cc$$

Uranium Silicide
Density of the fuel matrix from fuel fabrication information

$\rho = 3.5 \ g/cc$

High integrity Uranium Thorium Carbide
Uranium Thorium mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Th mass ratio 5.2% U 94.8% Th Density $\rho_{UC2} = 11.8$ $\rho_{ThC} = 8.96$

$$\frac{1}{\rho} = \frac{0.052}{11.8} + \frac{.948}{8.96} \qquad \rho = 9.07 \ g/cc$$

Low integrity Uranium Thorium Carbide
Uranium Thorium mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Th mass ratio 17.4% U 82.6% Th Density $\rho_{UC2} = 11.8$ $\rho_{ThC} = 8.96$

$$\frac{1}{\rho} = \frac{.174}{11.8} + \frac{.826}{8.96} \qquad \rho = 9.35 \ g/cc$$

Non graphite Uranium and Plutonium Carbide
Density of the fuel matrix

$$\rho = 11.28 \ g/cc$$

MOX
Uranium Plutonium mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Pu mass ratio 85% U 15% Pu Density $\rho_{UO2} = 10.96$ $\rho_{PuO2} = 11.46$

$$\frac{1}{\rho} = \frac{.85}{10.96} + \frac{.15}{11.46} \qquad \rho = 11.03 \ g/cc$$

Uranium Thorium Oxide
Uranium Thorium mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Th mass ratio 1.3% U 98.7% Th Density $\rho_{UO2} = 10.96$ $\rho_{ThO2} = 9.86$

$$\frac{1}{\rho} = \frac{0.013}{10.96} + \frac{.987}{9.86} \qquad \rho = 9.87 \ g/cc$$

Uranium Zirconium Hydride

Uranium Zirconium mass ratio, density of the fuel matrix using a weighted density based on the ratio

U-Zr mass ratio 8.54% U 91.46% Zr Density $\rho_U = 19.05$ $\rho_{Zr} = 6.49$

$$\frac{1}{\rho} = \frac{.0854}{19.05} + \frac{.9146}{6.49} \qquad \rho = 6.89 \ g/cc$$

Attachment B Hand Calculation Results

The following sheets provide the hand calculations that determine the specific fuel meat volume values for UAlx, U_3Si_2, and $UZrH_x$. The data for UAlx are taken from ATR fuel specifications. For U_3Si_2 the data are taken from a letter reference which is enclosed. TRIGA fuel specifications are used for $UZrH_x$.

Calculations by Volumetric Methods

As indicated earlier, in some cases, the fuel composition was uncertain, and it was more straightforward to calculate a volume of fuel meat based on fuel drawings or specifications and attribute it to the Uranium content of the fuel. This was true of groups 9 (UAlx+Al, and U_3Si_2), and 11 (U-Zr-Hx). The following are these calculations.

UAlx+Al

For the UAlx+Al, the specific fuel volume for this fuel is based on the ATR fuel, which was selected as a typical representative for which data were available. The following values were taken from ATR fuel specifications, Fuel receipt criteria, letter JAH-195-85, and letter HJR-11-78.

ATR Fuel Properties				
Matrix Material Per Assembly, kg	Total Uranium, kg	UAlx Density	UAlx Composition	Al Density
3.02 kg	1.156 kg	6.3	69% U	2.7

Based on the above data, the total kg of UAlx could be calculated below:

$$kg\ UAlx = [1.156][kg\ U][\frac{1}{0.69}][\frac{UAlx}{U}] \cong 1.675\ kg\ UAlx$$

Thus, the total Al kg could be calculated to be,

$$kg\ Al = [3.02][kg\ total] - [1.675][kg\ UAlx] \cong 1.345\ kg\ Al$$

Volume of the fuel meat per assembly could be calculated using the equation below,

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

or

$$assembly\ volume = [1675][\frac{g}{UAlx}][\frac{1}{6.3}][\frac{cm^3}{g}] + [1345][\frac{g}{UAlx}][\frac{1}{2.7}][\frac{cm^3}{g}] \cong 764\ cm^3$$

Assume that uranium makes up all the heavy metal in each assembly, the MTHM in each assembly would be (1.156 kg X 0.001 MT/kg) = .001156 MTHM.

U_3Si_2

For the U_3Si_2, the specific fuel volume for this fuel is based on the attached ANL reference letter from James Snelgrove dated 2-21-96.

Uranium Silicide Fuel Properties			
U_3Si_2 Density	U Loading gU/cc	Al Density	Comment
11.28	3.5	2.7	

$UZrH_x$

For the $UZrH_x$, the specific fuel data were taken from the report " Uranium-Zirconium Hydride Fuels for TRIGA Reactors" and "Characterization of TRIGA Fuel."

Uranium Zirconium Hydride Fuel Properties				
U Loading %/g	Fuel Diameter	Fuel Length	Center Zr Rod Diameter	Comment
8.5%/195 g	1.435 inch	15 inch	.225 inch	

Volume for this fuel could be calculated as follow,

$$assembly\ volume = [\frac{\pi}{4}][D^2 - d^2][h] \cong [\frac{\pi}{4}][1.435^2 - 0.225^2][15][2.54^3][in^3][\frac{cm^3}{in^3}]$$
$$\cong 388\ cm^3$$

Thus, volume per 195 g U loading (per fuel) is,

$$volume\ per\ fuel \cong [\frac{388}{195}][\frac{cm^3}{g\,U}][\frac{1}{10^6}][\frac{m^3}{cm^3}][10^6][\frac{g}{MT}]$$
$$\cong 1.99\,\frac{m^3}{MT\,U}$$

Attachment C EXCEL Spreadsheet Results

The following EXCEL spreadsheet shows the results of the volume calculation.

TSPA Cat	Fuel Type	MTHM	Group Disposal pkg total	Non-Ideal Density Correction	Fuel Vol/pkg, m3/pkg	molecular wt (compound)/ atomic wt (heavy metal) **uranium**	molecular wt (compound)/ atomic wt (heavy metal) **thorium**	molecular wt (compound)/ atomic wt (heavy metal) **plutonium**	Notes
1	Classified-Navy	65.00	NA	NA	NA	NA	NA	NA	
2	Pu/U Alloy	9.11	213	1.1	3.6E-02	NA	NA	NA	
3	Pu/U Carbide	0.11	3	1.1	3.8E-03	1.1	NA	1.1	Carbide mass correction, density=11.28
4	MOX	12.42	677	1.15	2.2E-03	1.1	NA	1.1	Oxide mass correction, 15% Pu, density=11.46
5	U/Th Carbide	26.28	561	1.1	6.3E-03	1.1	1.1	NA	94.8% Th, carbide mass correction
6	U/Th Oxide	50.35	67	1.1	9.5E-02	1.1	1.1	NA	Oxide mass correction, 98.7% Th
7	U Metal	2127.24	120	1.1	1.0E+00	NA	NA	NA	U density = 19
8	U Oxide	178.15	760	1.1	2.7E-02	1.1	NA	NA	
9	Aluminium Based Fuel (UAlx, U3Si2, U Oxide in Al)	20.94	1081	1.0	1.0E-02	NA	NA	NA	
10	Unknown	4.54	9	1.1	5.5E-02	1.1	NA	NA	Assume w/U02 fuel meat
11	U-Zr-Hx	1.62	53	1.1	6.7E-02	NA	NA	NA	
Total		2495.75	3544						

Attachment D E-mail Note From Vinod Vallikat

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

Subject: Re: SNF volume

Hi Henry,

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

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

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

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

Vrind <= Vrod

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

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

Hope this helps clear some of your questions.

-Vinod
To: Vinod Vallikat
cc: DFF@inel.gov, DCRESAP@inel.gov
From: HENRY@inel.gov

Date: 02/23/98 08:51:28 AM MST
Subject: SNF volume

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

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

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

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

DOE Spent Nuclear Fuel Information In Support of TSPA - SR

Appendix D
Radionuclide Inventory Summary of DOE SNF and HLW

Source of Table D-1: EDF-NSNF-003 [National[A]]

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

DOE COMPLEX	TSPA Group	TSPA Group	TSPA Group	TSPA Group	TSPA Group	TSPA Group
2,333 MTHM SNF, no HLW	1	1	2	2	3	3
	MTHM	Disposal packages	MTHM	Disposal packages	MTHM	Disposal packages
Hanford	0		0.01	14	0.07	2
INEEL	65		6.47	181	0.03	2
SRS	0		2.03	5	0.00	0
Total	65		8.50	200	0.10	4
	total curies	Average Ci/disposal pkg	Total curies	Average Ci/disposal pkg	total curies	Average Ci/disposal pkg
Isotopes						
AC227	N/A	N/A	1.6704E-03	8.2284E-06	2.8114E-04	7.0284E-05
AM241	N/A	N/A	2.9552E+04	1.4557E+02	1.1040E+03	2.7601E+02
AM242M	N/A	N/A	3.8527E+01	1.8979E-01	1.9216E+00	4.8039E-01
AM243	N/A	N/A	1.1080E+02	5.4582E-01	1.9218E-11	4.8044E-12
C14	N/A	N/A	7.4422E+00	3.6661E-02	4.9769E-02	1.2442E-02
CL36	N/A	N/A	9.4942E-02	4.6769E-04	1.2689E-04	3.1723E-05
CM244	N/A	N/A	1.6965E+03	8.3573E+00	4.8907E-13	1.2227E-13
CM245	N/A	N/A	8.2811E-01	4.0794E-03	4.8828E-18	1.2207E-18
CM246	N/A	N/A	1.6696E-01	8.2247E-04	1.5961E-21	3.9903E-22
CS135	N/A	N/A	8.8922E+00	4.3804E-02	1.0346E-01	2.5864E-02
CS137	N/A	N/A	1.0213E+06	5.0310E+03	1.1551E+04	2.8878E+03
I129	N/A	N/A	6.1474E-01	3.0283E-03	2.6272E-03	6.5679E-04
NB93M	N/A	N/A	1.6640E+01	8.1970E-02	1.3665E-01	3.4162E-02
NB94	N/A	N/A	3.1156E-01	1.5348E-03	1.9045E-02	4.7613E-03
NI 59	N/A	N/A	1.4169E+00	6.9798E-03	1.5609E-01	3.9021E-02
NI 63	N/A	N/A	8.2675E+01	4.0726E-01	2.5945E+01	6.4863E+00
NP237	N/A	N/A	5.1717E+00	2.5477E-02	1.6205E-02	4.0512E-03
PA231	N/A	N/A	3.7382E-03	1.8415E-05	5.0828E-04	1.2707E-04
PB210	N/A	N/A	1.4048E-06	6.9200E-09	3.1158E-08	7.7896E-09
PD107	N/A	N/A	1.0234E+00	5.0414E-03	2.9211E-03	7.3027E-04
PU238	N/A	N/A	9.0989E+03	4.4822E+01	1.6212E+02	4.0531E+01
PU239	N/A	N/A	1.9079E+03	9.3986E+00	8.8240E+02	2.2060E+02
PU240	N/A	N/A	1.8196E+03	8.9634E+00	7.6164E+02	1.9041E+02
PU241	N/A	N/A	4.4786E+04	2.2062E+02	6.3516E+03	1.5879E+03
PU242	N/A	N/A	4.5985E+00	2.2653E-02	2.0334E-05	5.0834E-06
RA226	N/A	N/A	4.5617E-06	2.2471E-08	8.6606E-08	2.1652E-08
RA228	N/A	N/A	1.4354E-06	7.0709E-09	5.4356E-08	1.3589E-08
SE79	N/A	N/A	1.3090E+01	6.4481E-02	3.7899E-02	9.4747E-03
SM151	N/A	N/A	1.1273E+04	5.5533E+01	7.6368E+02	1.9092E+02
SN126	N/A	N/A	9.5880E+00	4.7231E-02	8.6370E-02	2.1593E-02
SR90	N/A	N/A	9.1383E+05	4.5016E+03	4.9643E+03	1.2411E+03
TC99	N/A	N/A	3.0751E+02	1.5148E+00	1.0306E+00	2.5766E-01
TH229	N/A	N/A	8.8225E-06	4.3461E-08	5.8164E-09	1.4541E-09
TH230	N/A	N/A	7.2022E-04	3.5479E-06	2.5979E-06	6.4947E-07
TH232	N/A	N/A	7.1734E-07	3.5337E-09	2.4092E-09	6.0229E-10
U233	N/A	N/A	2.4779E-03	1.2206E-05	1.5386E-06	3.8465E-07
U234	N/A	N/A	1.0779E+01	5.3101E-02	1.6569E-02	4.1422E-03
U235	N/A	N/A	5.9292E+00	2.9208E-02	1.7752E-02	4.4381E-03
U236	N/A	N/A	9.0566E+00	4.4614E-02	6.0542E-03	1.5136E-03
U238	N/A	N/A	1.8724E+00	9.2237E-03	8.1088E-03	2.0272E-03
ZR93	N/A	N/A	3.5524E+01	1.7499E-01	1.5123E-01	3.7807E-02

Table D-1. (continued).

	TSPA Group	TSPA Group	TSPA Group	TSPA Group	TSPA Group	TSPA Group
2,333 MTHM SNF, no HLW	4	4	5	5	6	6
	MTHM	Disposal packages	MTHM	Disposal packages	MTHM	Disposal packages
Hanford	9.62	561	0.00	0	0.00	0
INEEL	1.96	71	24.52	524	46.98	64
SRS	0.00	2	0.00	0	0.00	0
Total	11.59	634	24.52	524	46.98	64
	total curies	Average Ci/disposal pkg	Total curies	Average Ci/disposal pkg	total curies	Average Ci/disposal pkg
Isotopes						
AC227	6.9072E-03	1.0878E-05	5.1112E-01	9.7541E-04	4.3321E+01	6.7689E-01
AM241	1.5629E+05	2.4612E+02	2.0185E+03	3.8522E+00	2.7958E+04	4.3684E+02
AM242M	2.6475E+02	4.1693E-01	4.9194E-01	9.3881E-04	3.6781E+01	5.7471E-01
AM243	7.7041E+01	1.2132E-01	1.3419E+01	2.5608E-02	1.0588E+02	1.6544E+00
C14	4.9983E+00	7.8713E-03	1.0433E+02	1.9910E-01	3.7884E+01	5.9194E-01
CL36	5.3427E-02	8.4137E-05	1.3493E+00	2.5750E-03	8.0320E-01	1.2550E-02
CM244	1.7359E+03	2.7336E+00	3.1810E+02	6.0706E-01	1.6199E+03	2.5312E+01
CM245	7.7855E-01	1.2261E-03	1.8394E-01	3.5104E-04	7.9319E-01	1.2394E-02
CM246	1.4553E-01	2.2918E-04	4.0735E-02	7.7739E-05	1.5984E-01	2.4974E-03
CS135	4.4635E+00	7.0291E-03	9.5306E+00	1.8188E-02	1.1730E+01	1.8328E-01
CS137	1.6169E+06	2.5462E+03	1.0198E+06	1.9462E+03	5.8495E+05	9.1399E+03
I129	2.7378E-01	4.3115E-04	1.0227E+00	1.9517E-03	7.3383E-01	1.1466E-02
NB93M	1.2207E+01	1.9223E-02	8.8762E+00	1.6939E-02	3.5161E+01	5.4939E-01
NB94	5.7380E-01	9.0362E-04	9.7512E-02	1.8609E-04	8.9039E-01	1.3912E-02
NI 59	4.6690E+00	7.3528E-03	8.2254E+00	1.5697E-02	2.6152E+00	4.0863E-02
NI 63	3.0735E+03	4.8402E+00	2.0054E+02	3.8272E-01	2.4444E+02	3.8193E+00
NP237	3.6013E+00	5.6714E-03	8.3048E+00	1.5849E-02	2.6207E+00	4.0948E-02
PA231	1.2528E-02	1.9729E-05	9.0387E+00	1.7249E-02	5.4367E+01	8.4949E-01
PB210	1.4545E-06	2.2905E-09	2.2649E-03	4.3224E-06	4.4784E-03	6.9975E-05
PD107	6.4574E-01	1.0169E-03	4.4181E-01	8.4314E-04	9.1091E-01	1.4233E-02
PU238	2.6818E+04	4.2233E+01	4.1157E+04	7.8544E+01	1.1786E+04	1.8416E+02
PU239	1.1025E+05	1.7362E+02	1.3180E+02	2.5153E-01	2.3830E+03	3.7235E+01
PU240	9.5763E+04	1.5081E+02	2.0044E+02	3.8252E-01	4.1108E+03	6.4232E+01
PU241	8.4919E+05	1.3373E+03	2.8099E+03	5.3624E+00	8.7454E+04	1.3665E+03
PU242	4.8136E+00	7.5805E-03	3.1696E-01	6.0489E-04	1.1730E+01	1.8329E-01
RA226	6.0163E-06	9.4745E-09	2.5465E-03	4.8598E-06	7.5340E-03	1.1772E-04
RA228	1.9067E-06	3.0027E-09	3.2199E+00	6.1448E-03	4.9389E+00	7.7170E-02
SE79	3.3024E+00	5.2006E-03	1.4770E+01	2.8187E-02	1.4137E+01	2.2089E-01
SM151	8.5699E+04	1.3496E+02	2.2234E+04	4.2432E+01	5.0959E+03	7.9623E+01
SN126	6.3618E+00	1.0019E-02	7.4159E+00	1.4152E-02	1.8171E+01	2.8392E-01
SR90	7.5551E+05	1.1898E+03	9.4373E+05	1.8010E+03	5.1003E+05	7.9692E+03
TC99	1.1628E+02	1.8312E-01	3.7315E+02	7.1213E-01	1.9916E+02	3.1119E+00
TH229	4.5919E-06	7.2313E-09	1.4364E+01	2.7411E-02	2.5266E+01	3.9478E-01
TH230	1.2134E-03	1.9109E-06	9.1664E-01	1.7493E-03	3.9824E-01	6.2225E-03
TH232	2.7550E-07	4.3386E-10	2.5463E+00	4.8593E-03	4.3174E+00	6.7460E-02
U233	1.2433E-03	1.9580E-06	3.4353E+03	6.5559E+00	5.1799E+03	8.0936E+01
U234	1.8852E+01	2.9688E-02	2.9527E+02	5.6350E-01	2.6888E+02	4.2012E+00
U235	5.3964E-01	8.4982E-04	1.3920E+00	2.6566E-03	1.4132E-01	2.2081E-03
U236	2.6743E+00	4.2115E-03	1.1173E+01	2.1323E-02	1.9287E+00	3.0136E-02
U238	8.9269E-01	1.4058E-03	2.9388E-02	5.6085E-05	2.3381E+00	3.6533E-02
ZR93	1.7205E+01	2.7095E-02	5.0930E+02	9.7194E-01	4.0033E+01	6.2551E-01

Table D-1. (continued).

	TSPA Group	TSPA Group	TSPA Group	TSPA Group	TSPA Group	TSPA Group
2,333 MTHM SNF, no HLW	7	7	8	8	9	9
	MTHM	Disposal packages	MTHM	Disposal packages	MTHM	Disposal packages
Hanford	1962.50	93	17.21	40	0.00	1
INEEL	6.46	15	148.41	514	0.00	0
SRS	15.85	3	0.60	158	19.54	1010
Total	1984.81	111	166.22	712	19.54	1011
	total curies	Average Ci/disposal pkg	Total curies	Average Ci/disposal pkg	total curies	Average Ci/disposal pkg
Isotopes						
AC227	9.4598E-03	8.3715E-05	9.5981E-03	1.3387E-05	5.0395E-04	4.9798E-07
AM241	4.9505E+05	4.3810E+03	3.3829E+05	4.7181E+02	6.9208E+03	6.8388E+00
AM242M	4.3493E+01	3.8490E-01	4.4713E+02	6.2362E-01	3.1168E+00	3.0799E-03
AM243	1.5954E+02	1.4119E+00	1.3034E+03	1.8178E+00	5.8299E+00	5.7608E-03
C14	6.2323E+02	5.5153E+00	8.8438E+01	1.2335E-01	4.7436E-03	4.6874E-06
CL36	6.5499E-02	5.7964E-04	1.0753E+00	1.4997E-03	0.0000E+00	0.0000E+00
CM244	3.2261E+03	2.8550E+01	2.8329E+04	3.9511E+01	4.7897E+01	4.7329E-02
CM245	1.4689E+00	1.2999E-02	1.2677E+01	1.7681E-02	3.4165E-03	3.3759E-06
CM246	2.3494E-01	2.0791E-03	2.3873E+00	3.3295E-03	2.1995E-04	2.1735E-07
CS135	7.5848E+01	6.7122E-01	4.1765E+01	5.8250E-02	3.7535E+01	3.7090E-02
CS137	8.4366E+06	7.4660E+04	4.0671E+06	5.6724E+03	4.3773E+06	4.3254E+03
I129	6.9468E+00	6.1476E-02	3.3379E+00	4.6553E-03	2.0779E+00	2.0533E-03
NB93M	3.4648E+02	3.0662E+00	1.7207E+02	2.3999E-01	1.8036E+01	1.7822E-02
NB94	1.4658E-01	1.2972E-03	3.8646E+00	5.3900E-03	3.6642E-03	3.6208E-06
NI59	3.4848E+01	3.0839E-01	5.2893E+01	7.3770E-02	0.0000E+00	0.0000E+00
NI63	3.2880E+03	2.9097E+01	6.6577E+04	9.2855E+01	1.4937E-19	1.4760E-22
NP237	7.3539E+01	6.5079E-01	3.6739E+01	5.1240E-02	1.0294E+01	1.0171E-02
PA231	2.4539E-02	2.1716E-04	1.7635E-02	2.4596E-05	5.4102E-03	5.3460E-06
PB210	9.5000E-07	8.4071E-09	2.4393E-05	3.4020E-08	1.6200E-08	1.6008E-11
PD107	1.4412E+01	1.2754E-01	9.9026E+00	1.3811E-02	1.2460E+00	1.2313E-03
PU238	1.1215E+05	9.9244E+02	1.9136E+05	2.6689E+02	1.4920E+04	1.4743E+01
PU239	2.1901E+05	1.9382E+03	3.9381E+04	5.4925E+01	4.9690E+03	4.9101E+00
PU240	1.3111E+05	1.1603E+03	5.3174E+04	7.4162E+01	3.0245E+03	2.9886E+00
PU241	1.6380E+06	1.4496E+04	1.4303E+06	1.9948E+03	1.3222E+05	1.3065E+02
PU242	7.4580E+01	6.6000E-01	1.4848E+02	2.0708E-01	2.5466E+00	2.5164E-03
RA226	1.9576E-03	1.7324E-05	7.1187E-05	9.9284E-08	2.9513E-07	2.9163E-10
RA228	9.6464E-07	8.5366E-09	4.6377E-03	6.4682E-06	7.9154E-10	7.8216E-13
SE79	3.3980E+01	3.0071E-01	3.8336E+01	5.3467E-02	5.6819E+01	5.6145E-02
SM151	1.3717E+05	1.2139E+03	5.0350E+04	7.0223E+01	4.1608E+04	4.1115E+01
SN126	1.5030E+02	1.3301E+00	7.1777E+01	1.0011E-01	1.9697E+01	1.9463E-02
SR90	6.7240E+06	5.9505E+04	3.0112E+06	4.1997E+03	4.1863E+06	4.1367E+03
TC99	3.3040E+03	2.9239E+01	1.4305E+03	1.9951E+00	1.1882E+03	1.1741E+00
TH229	3.4521E-05	3.0550E-07	1.3988E-02	1.9509E-05	1.6994E-06	1.6793E-09
TH230	2.2551E-03	1.9957E-05	1.0141E-02	1.4144E-05	1.3974E-04	1.3808E-07
TH232	1.2362E-06	1.0940E-08	4.8958E-03	6.8282E-06	1.9998E-08	1.9760E-11
U233	1.4284E-02	1.2641E-04	5.3761E+00	7.4981E-03	2.0592E-03	2.0348E-06
U234	8.3064E+02	7.3508E+00	6.1785E+01	8.6172E-02	1.7136E+00	1.6933E-03
U235	3.5628E+01	3.1529E-01	8.4020E+00	1.1718E-02	2.0832E+01	2.0585E-02
U236	1.3980E+02	1.2372E+00	3.5304E+01	4.9239E-02	3.8077E+01	3.7626E-02
U238	6.5851E+02	5.8275E+00	5.3299E+01	7.4337E-02	3.1227E+00	3.0857E-03
ZR93	4.3359E+02	3.8371E+00	2.1293E+02	2.9697E-01	1.1659E+02	1.1521E-01

Table D-1. (continued).

	TSPA Group	TSPA Group	TSPA Group	TSPA Group		
2,333 MTHM SNF, no HLW	10	10	11	11		
	MTHM	Disposal packages	MTHM	Disposal packages		
Hanford	0.07	2	0.03	1	sum MTHM Hanf	1989.51
INEEL	4.17	7	1.48	50	sum MTHM INEEL	305.47
SRS	0.00	0	0.00	0	sum MTHM SRS	38.02
Total	4.24	9	1.51	51	sum MTHM cmplx	2333.00
						complex
	total curies	Average Ci/disposal pkg	total curies	Average Ci/disposal pkg		total curies
Isotopes						
AC227	4.6356E-01	5.1506E-02	3.0139E-05	5.6866E-07	AC227	4.4324E+01
AM241	1.0369E+03	1.1521E+02	3.2794E+02	6.1876E+00	AM241	1.0585E+06
AM242M	1.7869E+00	1.9855E-01	1.2776E+00	2.4105E-02	AM242M	8.3928E+02
AM243	1.6780E-01	1.8644E-02	6.1095E-01	1.1527E-02	AM243	1.7767E+03
C14	8.2064E+01	9.1182E+00	7.8025E+00	1.4722E-01	C14	9.5624E+02
CL36	2.0925E-01	2.3250E-02	1.6690E-01	3.1491E-03	CL36	3.8179E+00
CM244	6.9330E+00	7.7034E-01	3.9970E+01	7.5414E-01	CM244	3.7020E+04
CM245	2.6340E-03	2.9266E-04	2.3274E-03	4.3914E-05	CM245	1.6738E+01
CM246	4.4327E-04	4.9252E-05	1.8749E-04	3.5375E-06	CM246	3.1361E+00
CS135	1.7063E+02	1.8959E+01	2.2818E+00	4.3053E-02	CS135	3.6278E+02
CS137	3.0735E+06	3.4150E+05	2.6798E+05	5.0563E+03	CS137	2.4477E+07
I129	4.3407E+00	4.8230E-01	7.1573E-02	1.3504E-03	I129	1.9423E+01
NB93M	2.2549E+02	2.5055E+01	1.9158E+00	3.6147E-02	NB93M	8.3702E+02
NB94	3.1404E+01	3.4893E+00	4.8933E-01	9.2327E-03	NB94	3.7800E+01
NI 59	2.5739E+02	2.8598E+01	3.1746E+01	5.9898E-01	NI 59	3.9396E+02
NI 63	4.2072E+03	4.6746E+02	3.8391E+03	7.2435E+01	NI 63	8.1538E+04
NP237	1.2667E+01	1.4074E+00	5.2306E-01	9.8691E-03	NP237	1.5348E+02
PA231	8.3808E-01	9.3120E-02	2.0880E-04	3.9396E-06	PA231	6.4309E+01
PB210	5.1383E-05	5.7092E-06	1.3639E-08	2.5734E-10	PB210	6.8230E-03
PD107	4.8230E+00	5.3589E-01	6.5034E-02	1.2271E-03	PD107	3.3474E+01
PU238	8.9736E+02	9.9706E+01	1.2135E+03	2.2897E+01	PU238	4.0956E+05
PU239	1.0616E+03	1.1795E+02	3.1591E+02	5.9606E+00	PU239	3.8029E+05
PU240	6.9853E+02	7.7614E+01	1.4739E+02	2.7810E+00	PU240	2.9081E+05
PU241	8.4377E+03	9.3753E+02	2.4763E+04	4.6722E+02	PU241	4.2243E+06
PU242	2.9571E-02	3.2857E-03	1.5073E-01	2.8440E-03	PU242	2.4724E+02
RA226	1.2542E-04	1.3935E-05	2.4265E-07	4.5784E-09	RA226	1.2246E-02
RA228	8.9623E-05	9.9581E-06	7.0705E-06	1.3341E-07	RA228	8.1635E+00
SE79	6.2637E+01	6.9596E+00	1.2230E+00	2.3075E-02	SE79	2.3833E+02
SM151	2.0225E+05	2.2473E+04	3.1307E+03	5.9069E+01	SM151	5.5958E+05
SN126	1.4255E+02	1.5839E+01	1.1855E+00	2.2369E-02	SN126	4.2714E+02
SR90	2.6437E+06	2.9374E+05	2.5209E+05	4.7564E+03	SR90	1.9945E+07
TC99	1.7043E+03	1.8937E+02	4.1382E+01	7.8078E-01	TC99	8.6655E+03
TH229	7.3962E-07	8.2180E-08	3.7542E-06	7.0834E-08	TH229	3.9643E+01
TH230	3.4884E-04	3.8760E-05	1.9164E-04	3.6158E-06	TH230	1.3299E+00
TH232	3.0472E-07	3.3858E-08	1.0641E-05	2.0078E-07	TH232	6.8686E+00
U233	1.9536E-04	2.1707E-05	5.7657E-03	1.0879E-04	U233	8.6206E+03
U234	6.6128E+00	7.3476E-01	3.9964E+00	7.5405E-02	U234	1.4985E+03
U235	2.3720E+00	2.6356E-01	7.4128E-01	1.3986E-02	U235	7.5995E+01
U236	7.2050E-01	8.0056E-02	1.2149E+00	2.2922E-02	U236	2.3996E+02
U238	1.0307E+00	1.1453E-01	3.8729E-01	7.3074E-03	U238	7.2149E+02
ZR93	2.5012E+02	2.7791E+01	7.1798E+00	1.3547E-01	ZR93	1.6226E+03

Table D-2. HLW Inventory.

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

a. Assumed 4 canisters per container.
Source: RW M&O 1995 TSPA x4 [Total[B]].

References:

A. National SNF Program engineering design file: EDF-NSNF-003 DOE SNF, Templates and Application

B. Total System Performance Assessment – 1995: An Evaluation of the Potential Yucca Mountain Repository, November 1995.