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United States Department of Energy

National Spent Nuclear Fuel Program

DOE Spent Nuclear Fuel Grouping In Support of Criticality, DBE, TSPA-LA

May 2000

U.S. Department of Energy
Assistant Secretary for Environmental Management
Office of Nuclear Material and Spent Fuel

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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 referenced in the report are not qualified.

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Rev. 0

DOE Spent Nuclear Fuel Grouping In Support of Criticality, DBE, TSPA-LA

May 2000

Idaho National Engineering and Environmental Laboratory
Idaho Falls, Idaho 83415

Prepared for the
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Assistant Secretary of Environmental Management
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DOE Spent Nuclear Fuel Grouping In Support of Criticality, DBE, TSPA-LA

May 2000

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ABSTRACT

This report presents the basis for grouping the over 250 Department of Energy (DOE) spent nuclear fuel (SNF) types in support of analyses for final repository disposal. For each of the required analyses, the parameters needed in conducting the analyses were identified and reviewed. The grouping proposed for the three types of analyses (criticality, design basis events, and total system performance assessment) are based on the similarities of DOE SNF as a function of these parameters. As necessary, further justifications are provided to further reduce the DOE SNF grouping in support of the Office of Civilian Radioactive Waste Management System's preclosure and postclosure safety cases.

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ACKNOWLEDGMENTS

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ACRONYMS

ARF	airborne release fraction
ATR	Advanced Test Reactor
BDBE	beyond design basis events
BOL	beginning-of-life
BWR	boiling water reactor
CFR	Code of Federal Regulations
DFA	driver fuel assembly
DBE	design basis events
DOE	Department of Energy
EDA	enhanced design alternative
EM	Office of Environmental Management
EOL	end-of-life
FERMI	Enrico Fermi Reactor
FFTF	Fast Flux Test Facility
FRR	foreign research reactor
FSV	Fort St. Vrain
HEU	high enriched uranium (>20% ^{235}U equivalent)
HLW	high-level waste
HWCTR	Heavy Water Components Test Reactor
INEEL	Idaho National Engineering and Environmental Laboratory
LA	license application
LEU	low enriched uranium (<5% ^{235}U equivalent)
LWBR	light water breeder reactor
M&O	Management and Operation (Contractor)
MAR	material at risk

MCO	multi-canister overpack
MEU	medium enriched uranium (>5% <20% ^{235}U equivalent)
MOX	mixed oxide fuel
MTHM	metric tons heavy metal
MWD/MTU	megawatt-day per metric ton of uranium
NNPP	Naval Nuclear Propulsion Program
NRC	Nuclear Regulatory Commission
NSNF	National Spent Nuclear Fuel (Program)
OCRWM	Office of Civilian Radioactive Waste Management
ORIGEN	Oak Ridge Isotope Generation
PB	Peach Bottom
PWR	pressurized water reactor
QARD	Quality Assurance Requirements and Description
RF	respirable fraction
RW	Office of Civilian Radioactive Waste Management
SiC	silicon carbide
SNF	spent nuclear fuel
SR/LA	site recommendation/license application
TDFA	test driver fuel assembly
TFA	test fuel assembly
ThC_2	thorium carbide
TMI	Three Mile Island
TRIGA	Training Research Isotopes — General Atomic
TSPA	total system performance assessment
U-Al	uranium aluminum
UC_2	uranium carbide

U-Mo	uranium molybdenum
U-Th	uranium thorium
U-Zr	uranium zirconium
U-ZrH_x	uranium zirconium hydride
VA	viability assessment

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DOE Spent Nuclear Fuel Grouping in Support of Criticality, DBE, and TSPA-LA

1. INTRODUCTION

Since the inception of the Department of Energy (DOE) spent nuclear fuel (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 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 purposes, such as repository disposition. However, none of the reports or the meetings were developed or conducted in a manner consistent with the Quality Assurance Requirements and Description (QARD) document.

The report titled *Grouping Method to Minimize Testing for Repository Emplacement of DOE SNF,* published in January 1997 is one such example.[1] It suggested 11 groups to represent the DOE-owned SNF for the purpose of performance assessment and gave reasons for that grouping selection. Since the publication of that report, through various analyses and sensitivity studies and discussions, the DOE SNF program has acquired a much better understanding of the DOE SNF performance in the proposed Yucca Mountain repository. These analyses and discussions provided the basis for further refinement of the original grouping, as well as expanding the grouping of DOE SNF for other analyses needs in support of the repository license. The November 17–18, 1998, DOE SNF Grouping Meeting[2] minutes documented some of these detailed discussions used to support the grouping efforts.

Volume 4 of the viability assessment (VA) contained the Office of Civilian Radioactive Waste Management's (RW's or OCRWM's) plan and cost estimate for the remaining work required to complete and submit a license application (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) preclosure safety case, (2) postclosure safety case, and (3) additional work needed to complete design decisions. Figure 1 shows a summary representation of the technical work needed to complete the LA.

This report presents the DOE SNF grouping used in support of criticality, design basis events (DBE), and total system performance assessment-license application (TSPA-LA) and the basis for each of the groupings.

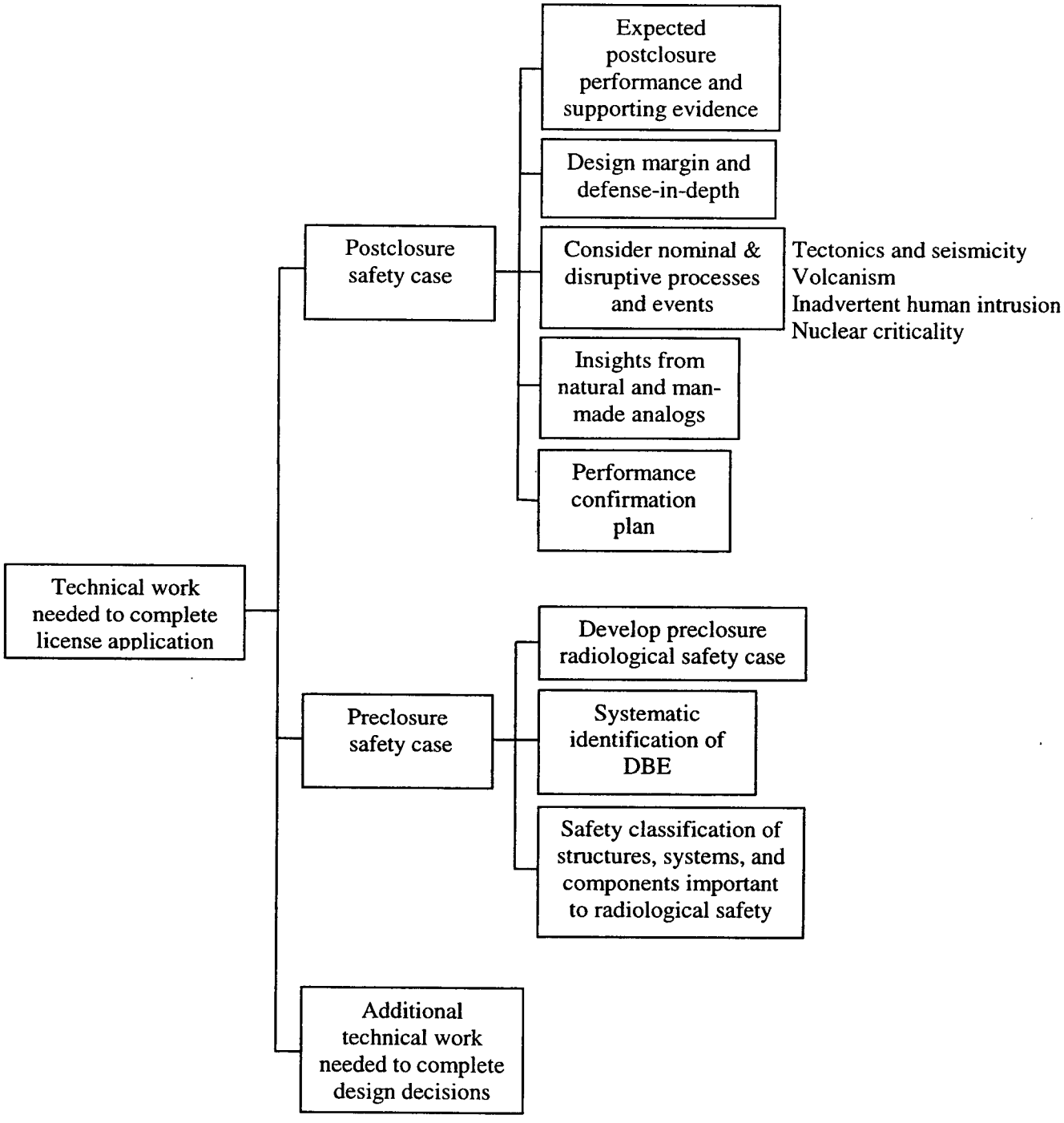

Figure 1. Repository technical work needed to complete site recommendation/license application—From Viability Assessment of a Repository at Yucca Mountain, DOE/RW-0508, Volume 4 with minor adjustment to reflect current needs.[3]

2. RESPONSIBILITIES/INTERFACES

The National Spent Nuclear Fuel (NSNF) Program will be responsible for interfacing with all the DOE sites to ensure that the proposed groupings in this report are consistent with each site's SNF management and final disposal plans.

3. SPENT NUCLEAR FUEL INFORMATION

All the information in this report 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 stated in the RW Quality Assurance Requirements and Descriptions DOE/RW-0333P Revision 9 Supplement III.2.4, "Unqualified DOE SNF data may be used for scientific investigation and design activities, provided traceability to its status as unqualified data is maintained. Unqualified data directly relied on to address safety and waste isolation issues shall be qualified in accordance with III.2.4.C at appropriate times."

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 GX1p. The computer has been certified to be year 2000 (Y2K) ready according to Bechtel BWXT Idaho, LLC's (the Idaho National Engineering and Environmental Laboratory [INEEL] management and operation contractor) Y2K desktop ready plan.

Computer software:	Microsoft® WORD and Excel 97 SR2
Computer hardware:	DELL OptiPlex GX1p

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5. BACKGROUND ON DOE SNF GROUPING

The main goal of grouping the DOE SNF is to supply data in a cost-effective manner to support DOE SNF management and disposal without increased risk to the public, environment, or worker safety. For the TSPA-VA, the DOE SNF inventory was first reduced to 34 DOE SNF groups based on fuel matrix, cladding, cladding condition, and enrichment. These parameters are the fuel characteristics that were determined to have the major impacts on the release of radionuclides from the DOE SNF and contributed to nuclear criticality scenarios.

From these 34 DOE SNF groups, it was determined, based on the results of the TSPA and associated sensitivity studies, that some groups may be combined to support either TSPA or 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 Volume 3 page 5-33.*[4] The condensed DOE SNF groups, the TSPA groups, and criticality analyses groups are shown in Figure 2 and Table 1. The typical 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. To further simplify the TSPA analysis, U-metal fuel was used as the design basis DOE fuel that bounds the entire DOE SNF inventory, because various sensitivity studies conducted as part of the TSPA-VA indicated that U-metal chemical dissolution properties bounded all of the other DOE SNF materials.

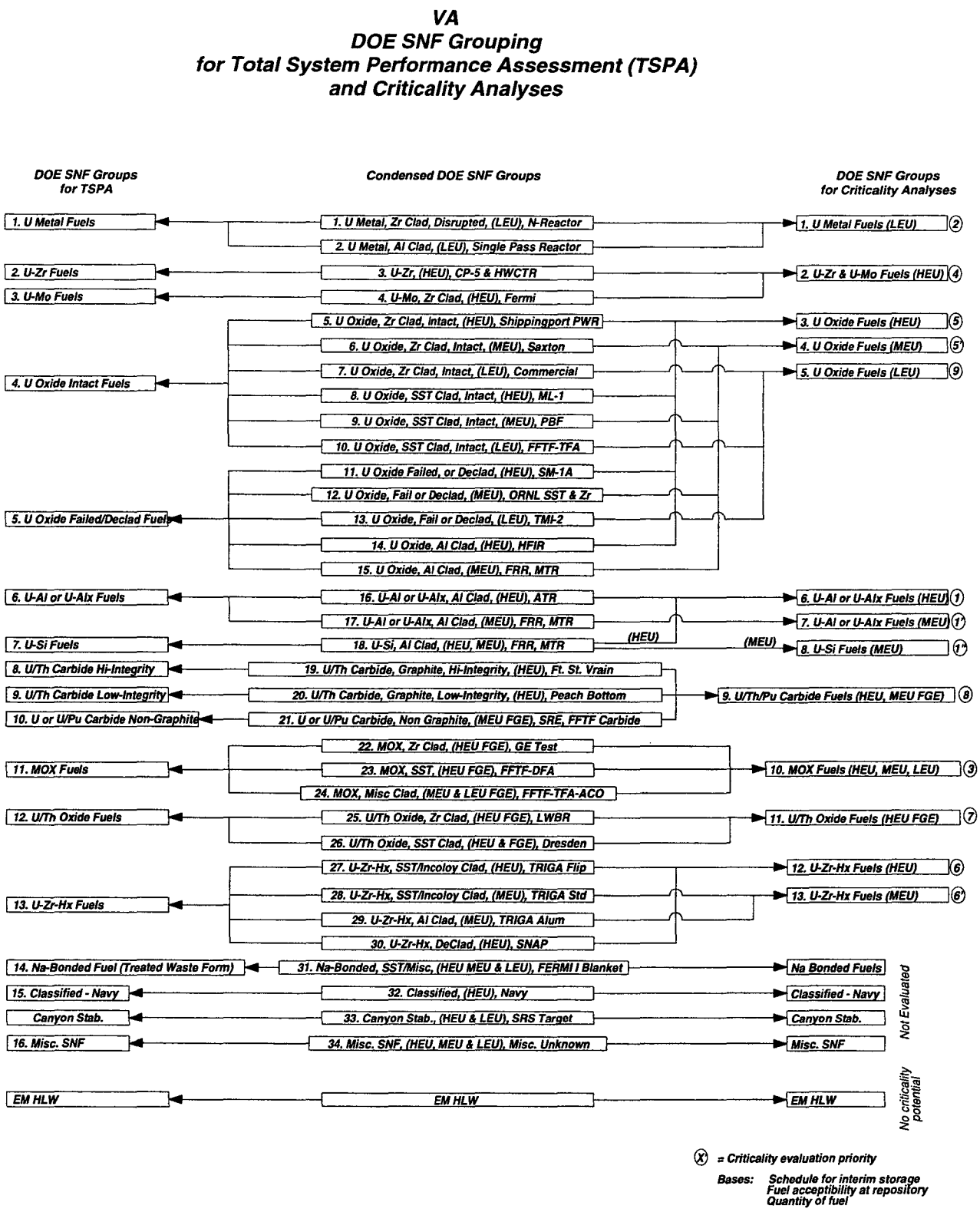

Figure 2. DOE SNF condensed groups, groups for TSPA-VA, and criticality analyses in FY 1998.

Table 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	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 non-graphite	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	

5.1 Group Selection for the DOE SNF

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. On November 17–18, 1998, a follow-up DOE SNF grouping meeting was held by the NSNF Program to further refine and document the grouping process based on RW's plan and cost estimate for the remaining work required to complete and submit an LA to the NRC. After RW selected the EDA II as the SR/LA design, the DOE SNF grouping team considered the impacts of the enhanced design to DOE SNF grouping activities. The following sections describe the parameters used for the grouping selection.

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 3. Figure 3 suggested that fuel matrix, fuel cladding, fuel condition, fuel enrichment, and burnup should be considered in the fuel grouping purpose. Sections 5.2 through 5.7 contain a brief discussion of these DOE SNF parameters.

Naval fuel 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.[5]

5.2 DOE SNF Fuel Matrix

The fuel matrices employed in the various DOE fuels over time were selected for various reasons associated with material production/reprocessing considerations, test materials to different reactor concepts, or some other specialized application. Selection of any fuel matrix in a production or test reactor certainly had to consider the reactor environment and operating conditions relative to fuel stability. However, many of the fuels ended up being reprocessed for fissile material recovery or recycling of valuable fissile material. The original fuel matrix must now be evaluated for its behavior in a different set of environmental conditions to include interaction with materials and under conditions different from those found in an operating reactor.

Fuel matrix formulations were based on some criteria associated with either reactor operation/material recovery or test material in support of advanced reactor concepts. None of the materials selected for the reactor fuels were ever considered for their longevity in a repository environment. Indeed, the materials may have been selected based on the ability to recover some portion of the spent fuel material in a chemical recovery process. The DOE SNF fuel matrix could be uranium metal fuels, various kinds of alloy fuels, or various forms of oxide and carbide fuels. These wide variations of DOE SNF are indicated on Figure 3.

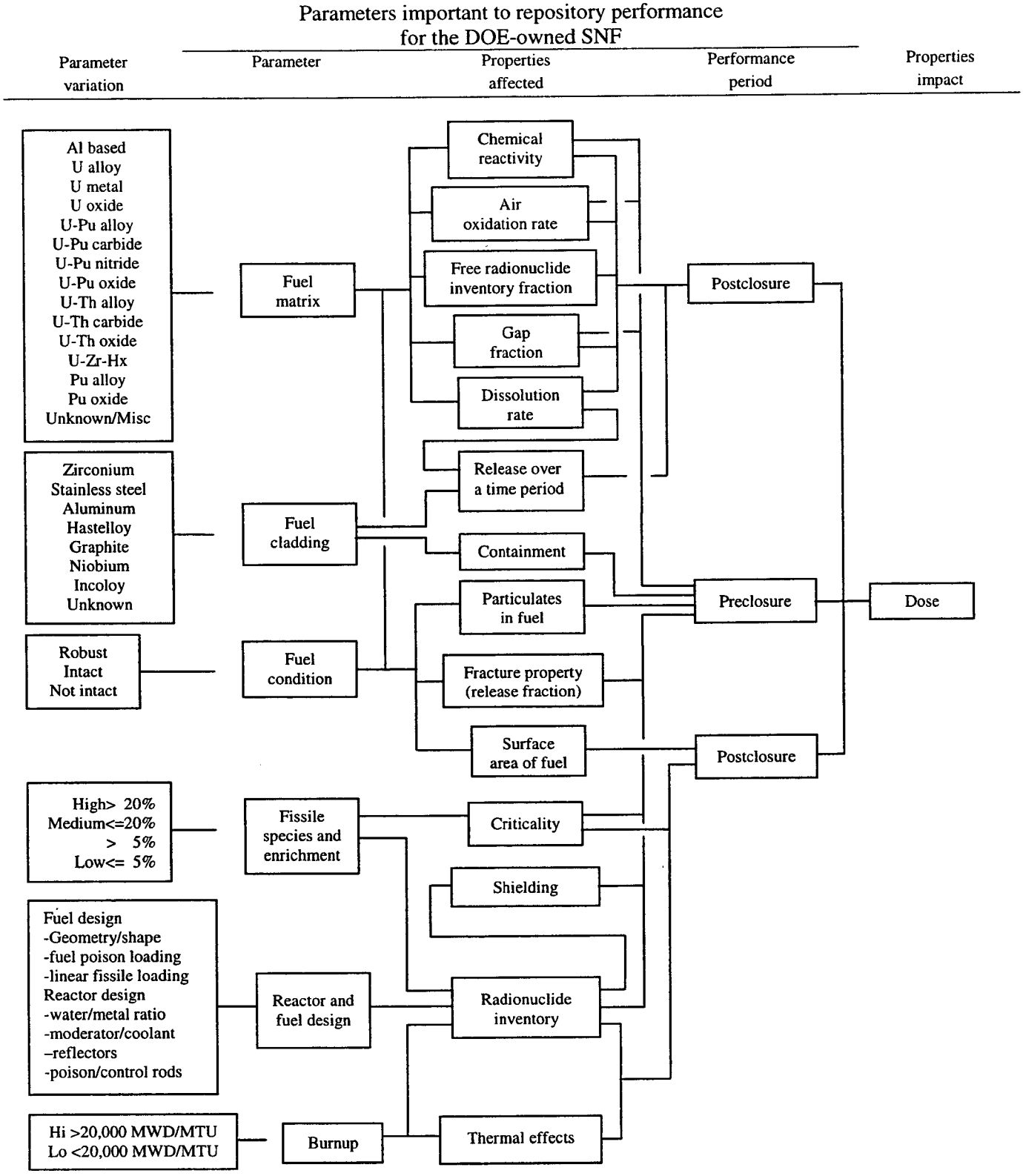

Figure 3. DOE SNF parameters and properties for criticality, DBE, and TSPA-SR analyses.

5.3 DOE SNF Cladding

Similarly to the fuel matrix, there is a wide variety of cladding materials employed in the fabrication of DOE SNF. The durable cladding materials include specifically the zirconium (similar to commercial clad) and the special cases silicon carbide coating on some of the graphite fuel particles. There will be a variety of zirconium alloys associated with the DOE fuels, because many were developmental tests of zirconium alloys leading to the various Zircaloy materials used in the commercial industry.

Cladding for most of the DOE SNF can be considered either durable or nondurable with respect to eventual waste package breach for the postulated repository conditions. The nondurable cladding would generally include both aluminum and stainless steel. The different types of DOE SNF cladding materials are shown on Figure 3.

5.4 DOE SNF Fuel Condition

The condition of the DOE SNF inventory also varies significantly. The fuel condition ranges from fuels removed from commercial reactors similar to RW's commercial fuel to the damaged fuels that come from commercial reactors such as Three Mile Island (TMI). While some DOE SNF may be in excellent condition such as the fuels from the Shippingport Pressurized Water Reactor (PWR) and Light Water Breeder Reactor (LWBR) Program, other DOE SNF inventory are not in such great shape. They included fuels such as the crushed TORY-IIA fuels and the General Electric Test Reactor fuel "element" that is stored at the INEEL and consists of a uranium sludge that has formed a semi-ceramic UO_2-U_3O_8 "cake." This uranium cake is the result of irradiated U-Al alloy capsules being dissolved and passed through a stainless steel filter.[6]

The DOE SNF conditions are described in Figure 3 as "Robust, Intact, or Not intact." In general, the robust fuel referred to the DOE fuel that is in excellent condition in terms of the fuel's physical condition as well as fuel that will tend to stay intact even if it is dropped. The intact fuel refers to the DOE fuel that is generally in good condition without significant cladding breached (using similar definition as the commercial SNF in terms of breached cladding). The not intact DOE fuel refers to fuel such as the fuel pieces from fuel examination and testing as well as the disrupted fuel such as TMI.

5.5 DOE SNF Enrichment

Enrichment varies widely among the DOE SNF. The DOE SNF in general contained enriched ^{233}U, ^{235}U and ^{239}Pu. The ATR fuel ^{235}U enrichment may be as high as 93%. On the other extreme, the DOE SNF inventory also contains reactor blankets with practically no fissile materials (see Reference 6). For the purpose of the grouping considerations, the fuel's fissile species and enrichment may be defined as high-enriched uranium (HEU), medium-enriched uranium (MEU), and low-enriched uranium (LEU) (or fissile material equivalent). HEU fuel has been defined as fuels with greater than 20% ^{235}U. MEU fuel has been defined as fuels with greater than 5% but less than or equal to 20% ^{235}U. LEU fuel has been defined as fuels with less than or equal to 5% ^{235}U. Figure 3 summarizes the fissile species and enrichment in term of high, medium, and low.

5.6 Burnup

Like all the parameters with the DOE SNF, burnup also varies widely within the DOE SNF inventory. From a very high burnup such as Shippingport PWR (with burnup as high as ~426,000 MWD/MTU) (see Reference 6) to very low burn fuel such as Power Burst Facility (with burnup as low as

~500 MWD/MTU) (see Reference 6). Figure 3 indicates two levels of burnup—high, being over or equal to 20,000 MWD/MTU and low, being less than 20,000 MWD/MTU.

5.7 Other Parameters

In addition, the fuel and reactor design variations were included as a parameter for use in the radionuclide inventory determination. Fuel design covers fuel geometry, shape, and fuel poison loading, etc. The reactor design covers variations such as moderator/coolant, reflectors, poison/control rods, and water/metal ratio. These reactor and fuel factors influence the neutron cross sections used in the depletion calculations. Thus, it is always prudent to assess the magnitude of effects these variations have on the neutron cross sections. However, these variations have no direct influence on the criticality, DBE, and performance assessment analyses; thus, they are not used for the grouping considerations.

Similarly, although the DOE SNF geometry and shape varies significantly from fuel to fuel, the variations mainly affect the way the fuels are packaged for disposal after the fissile loading has been considered. As an example, the length of the fuels in the DOE inventory varies from several inches to several feet. Ultimately, the length will determine the selection of either the nominal 10-ft or 15-ft SNF canister. Any decision relative to optimizing packaging of shorter fuels in the 15-ft canisters must be balanced against the number of 10-ft high-level waste (HLW) glass canisters (already being manufactured at West Valley and Savannah River Plant, and proposed for the INEEL) versus the number of 15-ft HLW canisters proposed for the Hanford site. The fuel's configuration is included in the criticality analysis as part of the fuel degradation scenario.

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6. CRITICALITY GROUPING SELECTION

While providing a relatively small contribution to the total metric tons of heavy metal (MTHM) within the repository (2,233 MTHM of the 7,000 MTHM allowed for defense-related materials, and 63,000 MTHM for commercial), the DOE SNF has many unique characteristics not found in commercial fuels that impact nuclear criticality evaluations.

Commercial fuels are evaluated based on whether they are either PWR or boiling water reactor (BWR) type fuels. These fuels have a very narrow range of enrichment ($3.75 \pm 0.5\%$ enrichment), well defined lengths and cross-sectional dimensions, relatively uniform cladding and fuel matrix composition, and closely controlled burnup values. Conversely, SNF within the DOE inventory exhibits a high degree of variability in terms of:

- Fuel matrix (metal, graphite, oxide, ceramic)
- Fissile material composition (metal, carbide, oxide, metal alloy)
- Cladding (aluminum, stainless steel, zirconium, other)
- Fissile species (^{235}U, ^{233}U, ^{239}Pu)
- Enrichment (0.7 to 90+% ^{235}U)
- Burnup value (several days for test elements to several years in demonstration reactors)
- Physical condition (intact, disrupted, particulate)
- Cross-sectional dimension (a few inches to a couple of feet)
- Length dimension (a few inches to several feet).

Each of the above parameters in some way affects the predicted behavior of the SNF relative to the potential releasable quantity of radionuclide material per packaging and behavior of the fissile material in what is eventually predicted to become a degraded waste package.

6.1 Fissile Species and Enrichment

The fissile species tested or used in various DOE reactors includes both ^{233}U and ^{235}U as well as ^{239}Pu. The related parameter designating the amount of the fissile species compared to the rest of the actinide mix, referred to as enrichment, is a primary driver and determines whether a system can achieve criticality. The ability to assemble a critical mass of fissile material is very dependent on the overall enrichment of the fuel matrix. Enrichment is perhaps the most important parameter used to determine fissile atom-density used as a basic input into various criticality analyses codes (KENO, MCNP, DANT, etc.)

Fuels are typically segregated by enrichment, but generally not just because of criticality safety concerns. The HEU/MEU break at 20% enrichment is based on reporting and protection requirements associated with special nuclear materials. The LEU designation is intended to encompass fresh or spent commercial fuels and generally fall within the range of $> 0.7\%$ and $<5.0\%$ ^{235}U enrichment. MEU fuels are categorized in the range between 5% and 20% values for this report. For actual criticality analysis, MEU and HEU categories can and should be lumped together, as it is not until enrichment levels drop

below approximately 10% that significant increases in fissile quantities are needed to assemble a critical mass. From a criticality modeling standpoint, the criticality analysts maintain there is a 'cross over' point that occurs above the 6.5% enrichment where the homogeneous assembly will use less material to achieve criticality than a corresponding heterogeneous mass. Conversely, lower enrichments (<6.5%) are favored in heterogeneous assemblies.

Enrichment plays a significant role in the packaging approaches proposed for the highly enriched fuels. The unique nature of HEU fuels is that a much smaller fuel volume and fissile mass is required to achieve criticality.

6.2 Fuel Compound / Matrix

The components of the fuel compound/matrix that are important to grouping fuels are the mix of actinides and the compound. In many cases, these fissile materials are blended in hybrid actinide systems that include combinations of materials such as $^{239}Pu/^{238}U$ mixed oxides, $^{235}U/^{232}Th$ carbides, and $^{233}U/^{232}Th$ oxides. These material mixes are further complicated in terms of composition and form employed in the fabrication of the original fuels.

Solubility issues based on actinide species may affect fissile material movement both within a breached SNF canister and waste package, and transport once outside a waste package on a long-term basis. Actual solubility differences between all chemical forms of uranium and plutonium in a common system are separated by at least two orders of magnitude, with uranium being the more soluble of the species. Similar differences in solubility are also noted between uranium and thorium, with the thorium species being the less soluble. In a closed system such as a breached SNF canister that retains water and soluble materials at equilibrium, new issues are raised with respect to fissile material transport, both within and outside/away from a package. For the less robust fuels, e.g., $U\text{-}Al_x$, the fissile material may separate from the aluminum in the matrix and fuel cladding and move away from the basket structure to collect in the bottom of the SNF canister. In other cases, the fuel basket structure inside the canister may degrade before the fuels, leading to intact fuel consolidation inside the SNF canister. Both of these cases must be addressed in criticality analyses when neutron absorbers are an integral part of the package.

The various fuel matrix properties, e.g., carbides, oxides, metals, ceramics, selected for the various fuels in the DOE inventory are of interest when segregating fuels by matrix. The storage conditions hypothesized for the waste packages must examine the introduction of water into the waste package through a breach. Given scenarios that allow for the formation of a chemical equilibrium (bathtub) versus a flow-through model, the various fuel matrices (in conjunction with the cladding and the internal basket structure) are expected to behave differently. What happens to the fissile material preferentially to the cladding and SNF canister internals necessarily affect the decisions relative to neutron absorber installation in a SNF package.

6.3 Cladding

Cladding for most of the DOE SNF can be considered either durable or nondurable with respect to eventual waste package breach for the postulated repository conditions. The nondurable cladding would generally include both aluminum and stainless steel (includes the high nickel alloys such as Inconel).

The durable cladding materials include specifically the zirconium (similar to commercial clad) and the special case silicon carbide coating on some of the carbide fuel particles in graphite. There will be a variety of zirconium alloys associated with the DOE fuels, because many were developmental tests of zirconium alloys leading to the various Zircaloy materials used in the commercial industry. Because the condition of even the durable cladding may be in question for many fuels, both extremes of intact and of

degraded cladding must be considered in developing degraded configurations. For this reason, cladding is not an important parameter in grouping.

The bulk carbon associated with the graphite fuels is an inert material and is expected to retain much of its structural shape even after package breach. While the graphite offers some moderator exclusion, it does not preclude water contact with the coated particles while also providing some degree of moderation.

Cladding material does play a part in the analysis of fuel in a degraded SNF package. Within the DOE SNF inventory, many of the fuels have unique dimensions that will require a fuel-specific basket or structure within the SNF canister. For the nondurable cladding material or that with unknown condition, early exposure of the fuel matrix to water could facilitate movement and/or redistribution of fissile material inside the SNF canister prior to gross failure of the internal SNF canister structure or the canister itself. For the more durable cladding, the canister internals might degrade prior to the cladding, thereby resulting in a different set of assumptions relative to movement of fissile material within a breached SNF canister. Both of these cases have added implications if neutron absorbers are included in the basket material to provide for criticality safety.

6.4 Burnup

A fresh fuel assumption with no credit for burnable absorbers is made for DOE fuels so burnup analysis becomes a relatively unimportant component for grouping. In the special cases of breeder or converter reactors, adjustments may need to be made to the beginning-of-life (BOL) fissile loading to account for the production of other fissile species because they generally affect the fuel reactivity calculations. In this case, burnup data are needed to support criticality analyses.

NOTE: End-of-life (EOL) fissile values may not be the most reactive, but may occur sometime during the burn or even after reactor irradiation as in the case of Pu-239 systems.

Burnup data are needed for purposes other than grouping for criticality analyses. Burnup values also determine the curie inventory needed for input to (a) performance assessment (PA) calculations, (b) determination of self-protection (safeguards issues), (c) shielding calculations, and (d) thermal analysis.

Accurate burnup data for many of the DOE fuels are not readily available. Generally, the fuels within the DOE inventory originated from test reactors, either as operating fuels or test assemblies. As such, the test fuels were either evaluating some other parameter besides burnup, or burnup was of very little concern. Fuels used to operate the reactor would be changed out when their reactivity decreased to unacceptable levels. Historically, when the fuels were discharged from their respective reactor(s), the expectation was the fuels would be reprocessed for fissile material recovery. This approach to fuel management did not create a need for detailed burnup information because almost all fuels had a mandatory (minimum) fuel basin storage time of 5 years prior to reprocessing.

Within the realm of reported burnup for various fuels, several parameters of reactor operation can affect these calculations. Included in these calculations are both a determination of the fission products generated and the formation of heavy metals that experience parasitic neutron capture leading to their transmutation. For the special case of a breeder reactor, Oak Ridge Isotope Generation (ORIGEN) calculations can provide fissile quantities (and perhaps of differing fissile species) as a function of time where fissile material increases can be documented. In those specialized cases, the maximum fissile concentrations would be used as the more "reactive" fuel form for the input of fissile atom-densities in the MCNP code cases to be analyzed. BOL values are generally easier to verify; no significant advantage is

gained in applying for EOL values in criticality safety calculations with initial enrichment above about 20%. For SNF disposal scenarios dealing with the highly enriched fuels (>20% fissile), taking credit for burnup provides only a marginal decrease in the fuel reactivity. It is easier to provide documentation relative to BOL enrichments and fissile mass than to try and prove the extent of burnup and the uncertainties associated with reactor fluxes, position of the individual fuels in the reactor, and cumulative power information.

Nuclear reactors are generally categorized as either burners, converters, or breeders. It is only within the DOE SNF inventories that fuels associated with the latter two types of reactors are being evaluated for inclusion within the proposed repository. The converter reactor typically transforms fertile to fissile material at a rate approximately equal to the rate at which fissile material is consumed. In the case of the graphite reactors demonstrated at a commercial scale (Peach Bottom and Ft. St. Vrain), ^{232}Th converted to ^{233}U. The changes in both the quantity and type of fissile material (^{235}U consumed / ^{233}U created) may affect overall reactivity of the fuel element(s) at time of withdrawal from the reactor and in geological time when the poisoning effect of the fission products has decayed away. Therefore, burnup data and analyses need to be considered in the criticality analysis, but not in grouping for criticality.

For breeder elements (limited to DOE research and demonstration reactors), ^{235}U is typically consumed to produce ^{239}Pu. Once again, determining the reactivity of the elements must include the quantity and type of fissile material bred into the system.

Burnup cannot be ignored, however, because the knowledge of that value is needed to calculate both thermal response of the package over time when loaded with the prescribed number of fuel elements and also the shielding calculations of a standard SNF canister. Coincidentally, the curie inventory in any given SNF package will be determined by the fissile loads allowed by criticality safety calculations. The curie quantities of the various isotopes will in turn be used to calculate the release values in the TSPA. Out of all packages studied, there will necessarily evolve a bounding case fuel canister with the highest fission product inventory and that will be determined by the allowable fissile load.

6.5 Other Considerations

The analyses of a specific fuel type within a fuel group is intended to bound the rest of the fuels within any defined group. Any fuel not already qualified with a detailed criticality analysis and its attendant thermal, shielding, and finite element analyses might qualify for use of the same package. If all the parameters (physical mass, fissile mass, dimensions, burnup, thermal) used in the design of the SNF canister and its fissile loading are less than those found in the detailed analyses, then qualification of the package for repository acceptance should be *pro forma*. Actual qualification of the individual packages with given fuel types at time of loading must still qualify those fuels for handling during packaging in an operational facility and for transportation.

One unique feature of fuels within the DOE inventory is the number of individual SNF assemblies within a group. These were generally test assemblies that were used to determine material properties/behavior or some other physical parameter in a reactor environment. None of these fuels justify individual storage/disposal inside one of the 18-in. diameter SNF canisters, so some selective combination of these disparate fuels might be considered in order to minimize the SNF canister count. It would be to some advantage to at least have fuel matrix material of similar properties packaged together in order to minimize the chemical interactions that might occur if dissimilar fuel matrices are comingled. This approach does recognize that dissimilar cladding material can be incorporated in the same SNF canister. TRIGA fuels are a prime example of mixed cladding in conjunction with a single SNF canister. Further removed from consideration would be any mixing/matching during packaging of fuels that,

because of dimensional considerations, allow both dissimilar cladding and fuel matrix material in the same package.

Some of the fuels in a given group, because of dimensional differences, might be better suited to disposal in an SNF canister with basket internals that were qualified for a different fuel type/group. Such a transposition would still require qualification of the fuel in its SNF canister. On a comparative basis, if all the parameters (fissile and physical mass, burnup, dimensions, etc.) are met for this new package configuration, e.g., Peach Bottom (graphite) fuels fit into a variant of the TRIGA or Fermi canister basket design, then only the storage and transportation need be addressed in detailed analysis.

The cross-sectional area of the fuel elements determines the actual loading configuration inside the nominal 18-in. diameter canister (for MEU or HEU fuels) or the 24-in. canister (for LEU fuels). This dimension, more than any other, dictates the internal basket structure/layout, physical spacing of the fuels inside the basket, and use of neutron poisons in the package. It may also limit the options for packaging other fuels, which in cross-section dimension exceed those for which the basket was designed.

Included in the variety of fuels considered for disposal are pins, plates, and rods, either singly or in assemblies. The nature of these configurations may either contribute to or detract from criticality safety depending on void volumes within and between the assemblies and the mechanisms for fissile material consolidation. The physical shape of other fuels in other groups may not preclude the use of a canister basket design in a different group. However, no current argument exists for mixing fuels across groups or even within groups because of the perceived complexities associated with a criticality analysis of a hybrid fuel package.

6.6 Fuel Groups

After examining all of the characteristics of SNF that affect nuclear criticality (see Reference 2), it was determined that the fuels could be grouped by a few important characteristics to simplify the nuclear criticality analysis; and by the judicious selection of the representative fuel within the group, the other characteristics could be addressed. The primary SNF characteristics that were used for the grouping were the fissile material enrichment, the fissile material and matrix material, and the cladding material. Figure 4 shows the methodology used in the development of the DOE SNF groups for use in the criticality analyses. The groups are discussed below.

6.6.1

Group	Type
U- metal	N-Reactor

The N-Reactor fuels constitute by far the greatest quantity (in MTHM) of SNF within the DOE inventory. Within this group of fuels, the Mark 1A and Mark IV fuels encompass over 90% of the fuel mass within this group. Current plans for packaging and disposal of these fuels include the use of a multi-canister overpack (MCO).

Two different enrichment values are represented with the Mark 1A (1.15% smeared enrichment) and Mark IV (0.947% enriched) fuels. For purposes of criticality analysis, the Mark IA fuel provided the most reactive fuel for loading within the MCO and would be considered to be the bounding case fuel. The Mark 1A fuel constitutes approximately 1/3 of the total N-Reactor fuel mass.[7]

6.6.2

Group	Type
MOX	FFTF

Mixed oxide (MOX) fuels were used exclusively as test fuels within the DOE complex. The Fast Flux Test Facility (FFTF) fuels, while not necessarily the most highly enriched within the

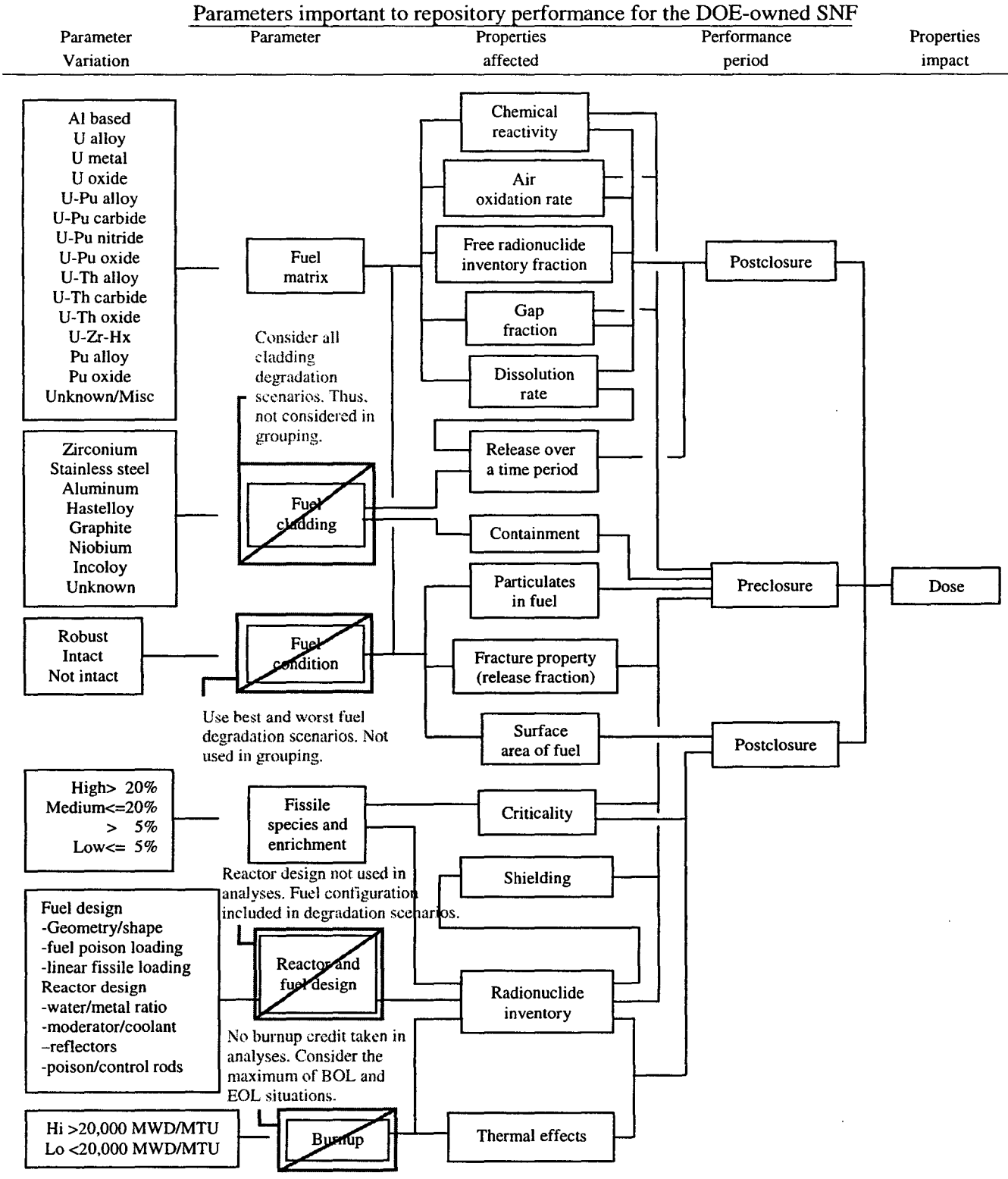

Figure 4. DOE SNF parameters used in grouping DOE SNF for criticality analyses.

grouping, provided the basis for analyzing multiple fuel assemblies within a single canister. Within the MOX grouping, the FFTF fuels contain approximately 2,332 kg of fissile material; no other type fuel within the grouping contains more than 50.0 kg. Furthermore, of the several plutonium enrichments ranging from 22.43 to 29.28%, the driver fuel assembly (DFA) Type 4.1 w/ 29+% enrichment was used as the base case for analysis. Proposed fissile masses per SNF canister are as great for these fuels as for any combination of other fuel types within this group.[8]

6.6.3 Group — Type

U-Mo/U-Zr — Fermi

The U-metal alloy fuels are dominated by the Fermi (Core 1 and 2) fuels at nearly 1,000 kg of fissile material in 214 canisters; as such, they comprise 96.48% of the fissile mass in this group. The Fermi fuels also represent the best documentation of any fuel type within this group.[9]

6.6.4 Group — Type

HEU oxide — Shippingport PWR

Shippingport PWR fuels consist of two different cores, both with highly enriched plate fuels (93.2% @ BOL), but with slightly different fissile loads. Data from both cores will be used in the analysis; Core 1 data will use the calculated burnup values for thermal and shielding analysis, and the Core 2 fissile mass will be used in the criticality analysis.[10]

6.6.5 Group — Type

U/Th oxide — Shippingport LWBR

Shippingport LWBR fuels provide a uniquely distinctive fuel in many ways. All of the fuel types in this grouping originate from the same reactor. The nature of the reactor design dictated differences in fissile loadings and the size of the various assemblies. Some of the blanket assemblies contain a higher fissile mass than the seed assemblies, but an atom-density/unit volume equivalent to the seed assemblies. Their size (too large) precludes disposal in an 18-in. canister, and disposal in a 24-in. canister is questionable, even with significant neutron poisoning. In addition, all these fuels are composed of high ^{233}U enrichments (~98%) mixed in binary fuel pellets of UO_2 (~5.2 wt% U [max.]) with a thoria (ThO_2) matrix.[11]

6.6.6 Group — Type

Graphite/Carbide — Fort St. Vrain (FSV)

Two demonstration commercial reactors generated distinctive graphite fuel types, Peach Bottom and FSV. The fuels used a combination of UC_2 and ThC_2 fuel particles embedded in a graphite matrix. The two fuel types have many common properties; their greatest difference is the physical size of the fuel assembly and how they end up being packaged for criticality safety. Preliminary indications are that both fuel types are volume limited; i.e., it is not possible to put enough fuel in the canister to reach nuclear criticality concerns, because the fissile mass needed for criticality is greater than that contained in the volume of fuel loaded in an 18-in. standard canister.

6.6.7 Group — Type

$U\text{-}ZrH_x$ — TRIGA

The TRIGA fuel elements have been constructed in a variety of configurations (differing clad materials, a variety of enrichments). These fuels are unique in that they incorporate a moderator in the matrix as zirconium hydride. The selection of the fuel type for criticality analysis within this group represents a departure from previous criteria applied to the other fuel groups. Within the other fuel groups, the quantity of fuel along with available characterization data was a determining feature in selection. Within the $U\text{-}ZrH_x$ group, the TRIGA fuel life improvement program fuels with their 70% ^{235}U enrichments constituted the worst case for packaging because of the reactivity of the fuels.[12]

6.6.8 Group — Type

Aluminum — Al-based fuels and/or melt dilute form

Earlier studies with aluminum fuels evolved around the MIT fuel design. Subsequent decisions are now proposing the use of a melt/dilute (with depleted uranium) to reduce the enrichment levels below 20%. While there is some small effect on the expected allowable fissile mass per SNF canister, the greater effect is the standardized shape and composition of the new waste form created with the metal ingots generated in the melt/dilute process.

6.6.9 Group — Type

LEU oxide — TMI debris

Within the DOE SNF inventory are a number of fuel assemblies from commercial reactors. Among these fuels are a set of canisters (344) containing the core debris from the TMI-2 accident. These canisters consist of (a) damaged fuel assemblies, (b) knock-out drums, and (c) filter units. The 184 assemblies (initial core load) are distributed throughout the 344 canisters.

The initial core loading was approximately 10% into its planned burnup lifetime, so the fissile load is relatively closer to the BOL enrichments than any of the other commercial fuels in the DOE inventory. While the internal basket design within the TMI fuel canister allowed for installation of one intact fuel assembly, it would not preclude (however unlikely) the addition of debris pieces such that the equivalent of more than one fuel assembly could be packed into a canister (see Reference 6).

7. DBE GROUPING SELECTION

DOE SNF consists of numerous fuel types with a variety of histories. For analytical purposes, it is effective to consolidate DOE SNF into groups that behave similarly relative to the conditions of interest such that rigorous analysis need only be performed for a bounding fuel to represent each group. This approach will be applied to support repository performance assessment, criticality studies, DBE analysis, and other analyses needed to qualify DOE fuels for disposition in the Yucca Mountain Monitored Geologic Repository.

The objective of this DBE grouping is to develop a methodology to reduce the required analyses while assuring that all potential radiological consequences from handling DOE fuels are thoroughly and defensibly bounded. This is achieved by identifying a minimum number of DBE groups such that a bounding fuel from one of the groups will produce the limiting DBE dose consequence for any credible DBE scenario.

7.1 Basis for DBE Fuel Groups

DBEs represent accident scenarios that are considered credible. DBE analysis is performed in order to enable the repository systems to be designed to accommodate these postulated accidents without exceeding regulatory limits that protect worker and public health and safety. For purposes of demonstrating compliance with repository licensing requirements, the DBE consequence is measured in terms of the radiological dose. Regulations limit the maximum allowable dose based on the DBE category. DBE analysis consists first of identifying possible scenarios. Frequency screening is then performed to determine if it is considered credible (i.e., is estimated to occur one or more times in one million years) and, if so, to determine its DBE category. Consequence analysis is then performed to determine the consequences that could result in the event that the DBE occurs.

Because the intent of the DBE fuel grouping is to facilitate identification of the bounding fuel for any credible DBE, the types of credible DBEs were first grouped. The criteria for the DBE fuel groups were then selected based on fuel properties that contribute to DBE dose consequence for the types of credible DBEs. Nearly all of the DBEs evaluated and deemed credible were the result of some type of energetic impact (i.e., crane drop, runaway transporter, rock fall). Consequently, fuels that are susceptible to mechanical damage and/or likely to release significant radioactivity as a result of a mechanical impact were grouped together. Some DOE fuels also have chemically reactive properties that could increase the radioactive material available for release and could also provide a propellant force to augment dispersion if exposed to an environment that enables chemical energy to be released. These fuels were also grouped together. Several fuels are both mechanically robust and chemically stable such that their release is clearly bounded by those in the two groups already mentioned. Accordingly, three DBE fuel groups that share dominant release mechanisms were defined. Each of the DOE SNFs was placed into one of three DBE fuel groups based on its fuel matrix:

- Stable metal fuels that are mechanically and chemically stable such that their properties are not expected to be affected by a DBE
- Nonmetal fuels that may fracture and create additional releasable particulate as a result of a DBE-related impact
- Other fuels with chemical properties that could increase release fractions as a result of reactions that may occur in conjunction with a DBE (i.e., could react with air or water, contain flammable gases or solids).

Because the release mechanisms for degraded fuels may differ from those of intact fuels, each of these three groups is further subdivided into those with intact fuels and those with degraded fuels. A fuel was presumed to be intact if its cladding is intact. Otherwise, the fuel was conservatively presumed to be "not intact." For purposes of DBE analyses, if fuels that are not intact can be conservatively considered to be 100% particulate, the DBE dose calculation is simplified. Further, if using this conservative assumption, the analyses show that the dose potential of these fuels is acceptable, the need for demonstrating the particulate content and size distribution is eliminated.

The six resulting DBE groups are determined solely by the fuel matrix and condition (See Figure 5) and can be represented by a 3×2 matrix whose rows are determined by the fuel compound and whose columns are determined by the fuel condition. Each of the DOE spent fuel types can be placed into one of these six DBE groups. The DBE groups are shown in Table 2 along with the dominant release mechanism for each.

Table 2. DBE fuel groups.

	INTACT	NOT INTACT
Stable Metal Fuels U-Alx U and Pu alloys U-Th alloys Intact Carbides Dispersion Fuels (U-Si)	Negligible release of fuel fines or particulate. Release is predominantly the surface crud.	Dispersion of preexisting particulate.
Non-metal Fuels U Oxides Mixed Oxides Pu Oxides U-Zr-Hx	Dispersion of particulate created from fuel fractures as a result of the DBE.	Dispersion of preexisting particulate plus any particulate created by the DBE.
Other Fuels U-Metal U-Moly Nitrides Degraded Carbides Other Miscellaneous Fuels Unknown Fuels	Assumed to be chemically reactive and to be completely oxidized.	Dispersion of preexisting particulate plus any additional release due to oxidation.

The DBE dose is calculated as a product of the material at risk (MAR) in the DBE, its damage ratio, its airborne release fraction (ARF), its respirable fraction (RF), and the product of any leak path factors such as canister retention, deposition in the facility, filtration, etc. All fuels within a DBE fuel group respond similarly to postulated DBE scenarios and will, thus, share the same ARF and RF. Other factors are not dependent upon the fuel but on the facility and site. Consequently, the radiological dose calculated by the DBE analyses for each fuel within the group is proportional to the dose potential of its MAR. The dose potential of all fuels within a DBE fuel group is, therefore, bounded by that of the fuel in the group with the most MAR. Because the MAR is a strong function of a fuel's burnup, burnup is important to DBE analyses. Though not specifically used in the DBE grouping, the bounding fuels for each group (i.e., highest MAR) are likely to be those with the highest burnup.

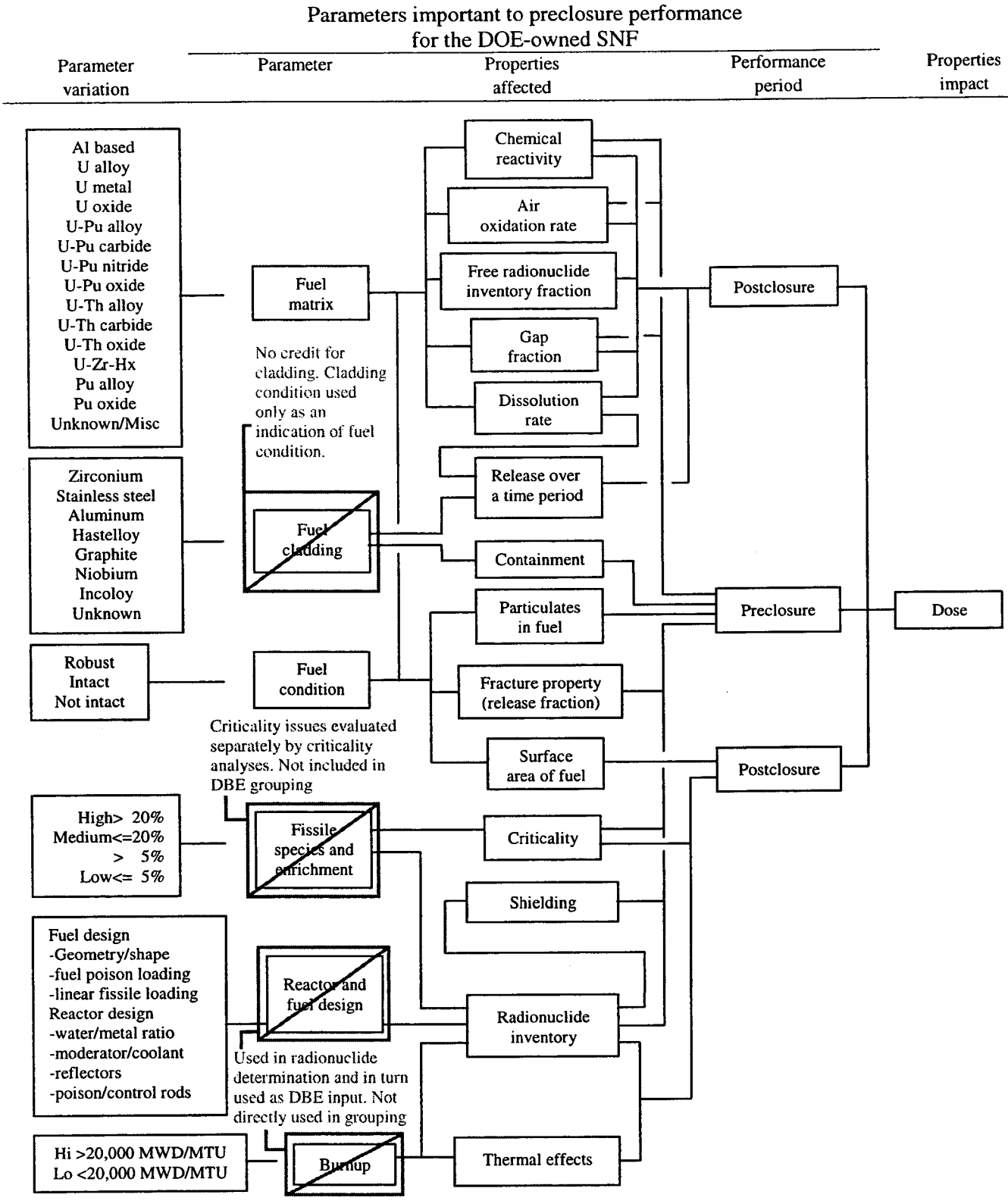

Figure 5. DOE SNF parameters used in grouping DOE SNF for DBE analyses.

Similarly, the fuel with the highest product of MAR×ARF×RF could be considered as bounding for all DOE SNF with respect to a potential DBE release. To the extent that the controls determined necessary for this fuel are applied to all DOE SNF, the DBE fuel groups could be collapsed into one fuel group with a single bounding DBE for all DOE SNF. The six groups however will be carried through the analyses in order to preserve the ability for group specific requirements definition and to facilitate beyond design basis events (BDBE) analyses.

Preliminary DBE analyses performed during 1999 used existing data and the DBE group release fractions to calculate DBE doses and identify the fuels that are likely to be bounding for each of the six DBE fuel groups identified. Conservative assumptions were applied such as 100% particulate, no credit for cladding or canister, and no credit for facility deposition or filtration. Even using these extremely conservative assumptions, all stable metal intact fuels were found to be well below regulatory limits as were the majority of the other fuels. However, more realistic source term and release fraction data and/or additional allocation of performance to natural and engineered systems will be necessary to demonstrate that all fuels will comply with regulatory requirements. The details of the analyses and resulting recommendations have been documented in a report entitled, *Preliminary Design Basis Event Analyses of DOE SNF*, ANL-WPS-SE-000001, Revision 0, MOL.19990720.0405.

8. TSPA-LA GROUPING SELECTION

After considering all of the parameters and properties for the postclosure performance period, the grouping team concluded that the fuel matrix would be the only parameter needed to group DOE SNF to support postclosure analyses. Figure 6 summarizes the reasons for not including the other parameters in the PA grouping consideration. In addition, analyses completed as part of the FY 1999 TSPA have demonstrated that DOE SNF can be represented by a surrogate spent fuel with properties like the U-metal fuel.[13] 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).

Using fuel matrix as the variable, all the DOE SNF could be placed into 11 groups as indicated in Table 3. Figure 7 shows the methodology used in development of the DOE SNF Groups for TSPA-LA. Discussion of each DOE SNF group is presented in Section 8.1.

Table 3. 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][a]	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 (U-Al_x, U_3Si_2, U oxide in Al)	FRR pin cluster U_3Si_2-LEU Canada [660]	
10	Unknown	Miscellaneous RSWF fuel [366]	
11	U-ZrH_x	TRIGA (Aluminum) Cornell Univ. [235]	

a. The number in [ ] is the fuel identification used in the DOE SNF Database, Version 3.7.0.

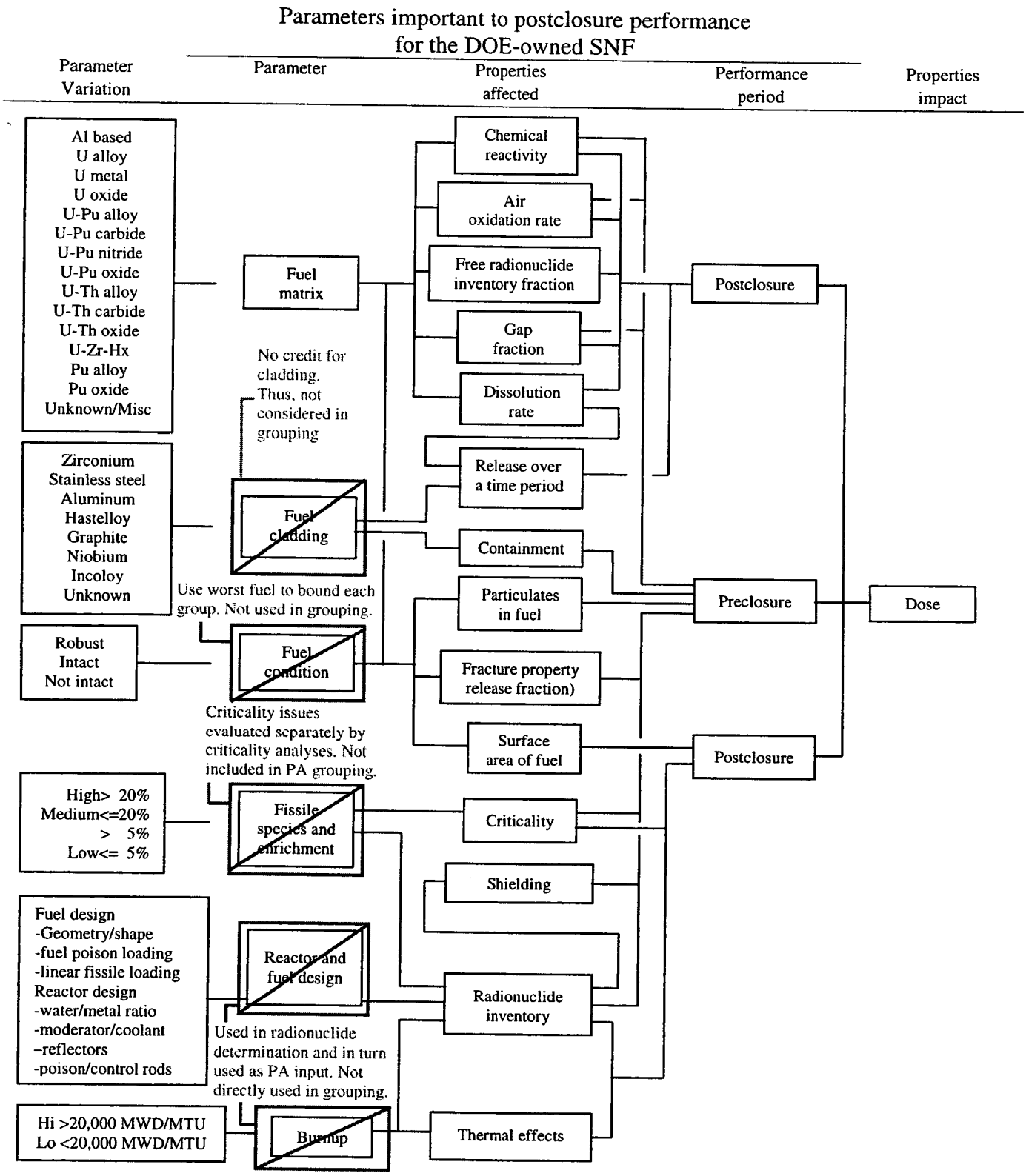

Figure 6. DOE SNF parameters used in grouping DOE SNF for TSPA-SR, LA analyses.

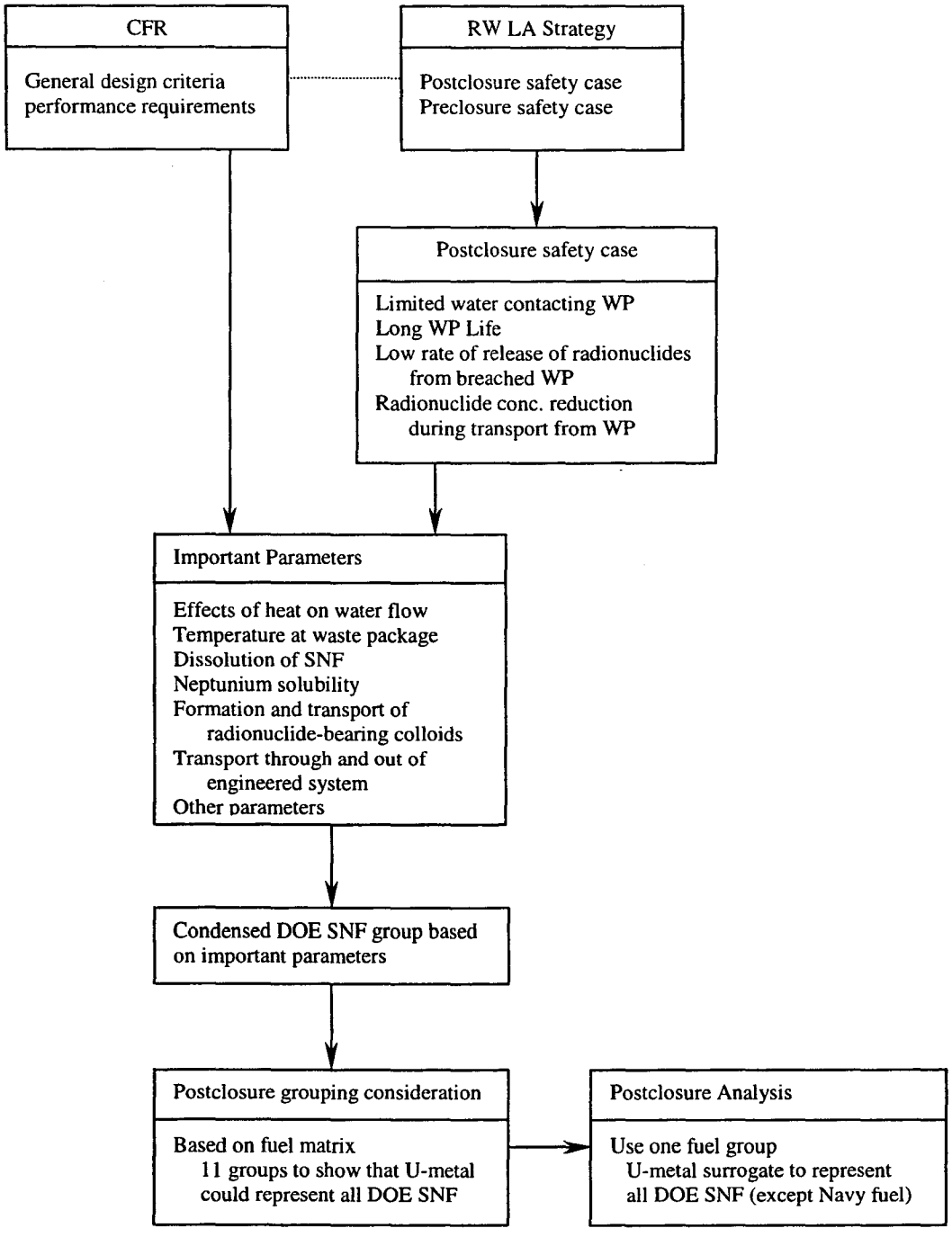

Figure 7. DOE SNF postclosure grouping methodology.

8.1 DOE SNF Grouping for Performance Assessment

The source term for analysis of the repository performance (i.e., release of the radionuclides to the environment) is the total curies of activity. This can be stated for the total repository or for a waste package. The release rate is the curies released per unit time that become available for transport to the environment. If one assumes that the major release of radionuclides is controlled by the corrosion of the fuel matrix containing the radionuclides (a valid assumption because all other release mechanisms such as solid transport are small under expected repository conditions), then the corrosion rate can be used to calculate the release rate. The corrosion rate for a material is expressed in grams per unit area per time. When the corrosion rate is multiplied by the surface area, then grams of material per time that corrode is obtained. When the total mass and source term of the fuel matrix is included in the calculation and neglecting any holdup of the radionuclides in the corrosion products, then the release rate can be determined. 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. The grain boundary effects are dependent on the microstructure of the fuel, which is dependent on the material and possibility the irradiation history. 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 Pacific Northwest National Laboratory 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 9, 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 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 approximately 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) make up approximately 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, uses a U-Zr alloy fuel, expects to generate approximately 26% of the MTHM of the Pu/U alloy fuel group by the year 2035. Enrichment is expected to be about 12% ^{235}U (see Reference 6).

Group 3—Plutonium/Uranium Carbide Fuel: This group consists primarily of fuels from the FFTF. The FFTF fuels are either UC_2 pellet or UC_2 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 (see Reference 6).

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 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 (see Reference 6).

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 Peach Bottom 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 approximately 93% enriched uranium. The fertile particles contain only thorium. One difference between the FSV and Peach Bottom fuels is that the Peach Bottom particles lack the silicon carbide coating. The fuel particles in the FSV and Peach Bottom Core 2 fuel assemblies are in excellent condition. However, the fuel particles in the Peach Bottom 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 (see Reference 6).

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 (see Reference 6).

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.[14]

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. Because enrichment should not alter the release rate for fuels with the same matrix, enrichments from the typical of approximately 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 (see Reference 6).

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 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 better than the standard U-oxide fuel (see Reference 6).

9. CONCLUSIONS

This report documents the method chosen to represent the DOE-EM SNF in criticality, DBE, and TSPA that will be used to support the Yucca Mountain SR and LA.

For the reader's understanding, the report gives the history of analysis of the DOE-EM fuel for repository acceptability, the progression of the grouping methods that were used, and how the grouping changed as repository analyses results indicated that some characteristics were not important to repository performance.

The DOE SNF grouping will facilitate analyses that show the assumptions used in the representation of DOE-EM SNF are conservative and justifiable.

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Revision History

Revision No.	Revision Description	Revision Date
0	New document (DAR #NSNF-149)	May 2000