INL/EXT-07-12326

Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

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operated by Battelle Energy Alliance

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Prepared for the
U.S. Department of Energy
Assistant Secretary for Environmental Management
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

NATIONAL SPENT NUCLEAR FUEL PROGRAM ENGINEERING DESIGN FILE

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

1. Activity Title: National Spent Nuclear Fuel Program 2. WBS No. C.B.30.03.02.02.B2			
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5. Purpose:

This document represents a summary version of the criticality analysis done to support loading SNF in a Type 1a basket/standard canister combination. Specifically, this engineering design file (EDF) captures the information pertinent to the intact condition of four fuel types with different fissile loads and their calculated reactivities. These fuels are then degraded into various configurations inside a canister without the presence of significant moderation. The important aspect of this study is the portrayal of the fuel degradation and its effect on the reactivity of a single canister given the supposition there will be continued moderation exclusion from the canister. Subsequent analyses also investigate the most reactive 'dry' canister in a nine canister array inside a hypothetical transport cask, both dry and partial to complete flooding inside the transport cask. The analyses also includes a comparison of the most reactive configuration to other benchmarked fuels using a software package called TSUNAMI, which is part of the SCALE 5.0 suite of software.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

CONTENTS

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

FIGURES

TABLES

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

ACRONYMS

AENCF	average energy (of) neutrons causing fission
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
BFS	big physical stand
BOL	beginning of life
BSC	Bechtel SAIC, LLC
CRWMS	Civilian Radioactive Waste Management System
DOE	U.S. Department of Energy
DOE/RW	DOE Office of Radioactive Waste
DWG	drawing (document type)
EDF	engineering design file
ENDF	evaluated nuclear data file
FHU	fuel-handling unit
HEU	highly enriched uranium
HP	Hewlett-Packard
H/X	hydrogen to fissile ratio
ICSBE	International Handbook of Evaluated Criticality Safety Benchmark Experiments
INEEL	Idaho National Engineering and Environmental Laboratory
INL	Idaho National Laboratory
IPPE	Institute of Physics and Power Engineering (Russia)
k_{eff}	Effective neutron multiplication factor
M&O	management and operations

MCNP	Monte Carlo N-Particle Transport Code
MEU	medium-enriched uranium
MIT	Massachusetts Institute of Technology
MURR	Missouri University Research Reactor
NSNFP	National Spent Nuclear Fuel Program
OCRWM	Office of Civilian Radioactive Waste Management
ORR	Oak Ridge Research Reactor
PC	personal computer
SCM	software configuration management
SNF	Spent Nuclear Fuel
UAl_x	uranium aluminide
UNS	Unified Numbering System
YMP	Yucca Mountain Project

1. PURPOSE

The objective of these analyses are to demonstrate that DOE SNF canisters loaded for disposal in the national repository with Type 1a baskets can also meet criticality safety under transportation scenarios prescribed by 10CFR71.55. Several DOE SNFs (Appendix B) show which SNFs will be packaged using a Type 1a basket configuration. Of these SNFs, the following four fuel types were considered among the most reactive, either because of their individual fissile loads (per assembly) or what their collective fissile content contributes to a loaded canister:

- Advanced Test Reactor (ATR)
- Oak Ridge Research Reactor (ORR)
- Missouri University Research Reactor (MURR)
- Massachusetts Institute of Technology (MIT).

These four fuel types are evaluated to identify the parameters they represent in terms of criticality safety for a loaded SNF canister, e.g. total fissile, enrichment, and linear fissile loading in a canister. The analyses demonstrate the technical viability of the DOE Standardized SNF canister and associated packaging for addressing transportation issues relative to criticality safety.

The preliminary analyses provide a baseline comparison between the individual fuel types for their proposed loading in single DOE SNF canisters. Analyses address reconfigured fuel (i.e. fuel debris) in both horizontal and vertical orientations of the canister and the debris contained within each basket compartment. Special cases also analyze bounding configurations in which the fuel debris from stacked baskets is consolidated within one set of basket compartments, meaning basket base plates do not keep the fuel debris from consolidating in one lumped mass. Special cases also examine a non-mechanistic case in which fissile material is consolidated in the bottom of the canister and away from any poisoned basket plates. The remaining analyses examine the transport cask cavity relative to an array of canisters with their content in their most reactive configuration and subject to various degrees of flooding.

2. QUALITY ASSURANCE

This document was developed and is controlled in accordance with NSNFP procedures. Unless noted otherwise, information must be evaluated for adequacy relative to its specific use if relied on to support design or decisions important to safety or waste isolation.

The NSNFP procedures applied to this activity implement DOE/RW-0333P, *Quality Assurance Requirements and Description* [DOE 2004a], and are part of the NSNFP Quality Assurance Program. The NSNFP Quality Assurance Program has been assessed and accepted by representatives of the Office of Quality Assurance within the Office of Civilian Radioactive Waste Management for the work scope of the NSNFP. The NSNFP work scope extends to the work represented in this report.

The current, principal NSNFP procedures applied to this activity include the following:

- NSNFP 3.03, “Engineering Analysis”
- NSNFP Procedure 3.04, “Engineering Documentation”
- NSNFP Procedure 6.01, “Review and Approval of NSNFP Internal Documents”
- NSNFP Procedure 6.03, “Managing Document Control and Distribution”.

2.1 METHOD

The method to perform the criticality calculations consists of using the Monte Carlo N-Particle Transport Code (MCNP) Version 4B2 [CRWMS 1998a, 1998b] to calculate the effective neutron multiplication factor of the waste package. The calculations are performed using the continuous-energy, cross-section libraries, which are part of the qualified code system MCNP 4B2 [CRWMS 1998a, 1998b].

3. ASSUMPTIONS

3.1 Neutron Interaction Cross Sections for ^{137}Ba

Assumption: ^{138}Ba cross sections are assumed to adequately represent ^{137}Ba cross sections in the MCNP input..

Rationale: The cross sections of ^{137}Ba are not available in either evaluated nuclear data file ENDF/B-V or ENDF/B-VI cross-section libraries

Confirmation Status: This assumption is conservative, because the thermal neutron capture cross section and the resonance integral of ^{137}Ba (5.1 and 4 barn, respectively) are greater than the thermal neutron capture cross section and the resonance integral of ^{138}Ba (0.43 and 0.3 barn, respectively) [Parrington et al. 1996, p. 34] and thus does not require further confirmation.

Use in the Calculation: This assumption is used in the analyses used to produce the results given in Section 7.

3.2 Neutron Interaction Cross Sections for Zinc

Assumption: Al cross sections are assumed to adequately represent zinc cross sections in the MCNP input.

Rationale: The cross sections of zinc are not available in the MCNP 4B2LV cross-section libraries.

Confirmation Status: This assumption is conservative because the thermal neutron capture cross section and the resonance integral of zinc [Parrington et al., 1996, p. 24] are greater than the thermal neutron capture cross section and the resonance integral of Al [Parrington et al., 1996, p. 21] and thus does not require further confirmation.

Use in the Calculation: This assumption is used in the analyses used to produce the results given in Section 7.

3.3 Spent Nuclear Fuel Composition

Assumption: Beginning-of-life composition is conservatively assumed in the calculations, and no credit is taken for the initial boron neutron absorber present in the fuel.

Rationale: This assumption is conservative, because it results in a higher k_{eff} for the system.

Confirmation Status: This assumption does not require further confirmation.

Use in the Calculation: This assumption is used in the analyses used to produce the results given in Section 7.

3.4 Fissile Content

Assumption: Because the fuels analyzed are manufactured with various BOL fissile loadings, the highest fissile content is conservatively assumed.

Rationale: This assumption is conservative, because it maximizes the fissile isotope (^{235}U) content while minimizing the effect of neutron absorption by ^{238}U.

Confirmation Status: This assumption does not require further confirmation.

Use in the Calculation: This assumption is used in the analyses used to produce the results given in Section 7.

3.5 Moderator Saturation of Fissile Matrix Material

Assumption: Voids within the fuel matrix are assumed to have the following values and conservatively assumed to be completely waterlogged.

- ATR 11% (resultant H/X ratio of ~2.1; 89.18 g H_20/assembly)
- MURR 11% (resultant H/X ratio of ~2.0; 58.94 g H_20/assembly)
- MIT 19.75% (resultant H/X ratio of ~3.4; 70.04 g H_20/assembly)
- ORR ~30% (resultant H/X ratio of ~8.8; 116.83 g H_20/assembly)

Rationale: Fuel matrix materials have a production density that is less than theoretical density for the given composition [Knight, pg. 21, Fig. 17] and/or may have undergone fracturing due to radiation exposure. Water may have become entrained in the fuel matrix interstices during time of immersion in reactor operations and/or wet pool storage because of cladding defects. It is not likely that any fuel drying process will be able to remove this trapped water if present.

Confirmation Status: This assumption is conservative because the assumed water promotes thermal neutron behavior, and because not all of the fuel matrix interstices are water saturated due to drying and the fact that they may not be interconnected. This assumption does not require further confirmation.

Use in the Calculation: This assumption is used in the analyses used to produce the results given in Section 7.

4. USE OF COMPUTER SOFTWARE

4.1 Software

Microsoft EXCEL Version 2003 Service Pack-2, was used for performing tabular representations and arithmetical manipulations in a spreadsheet environment. KaleidaGraph Version 4.03 was used to create the X-Y plots associated with the tabular data presented in the various sections of this report. These two commercial software packages were installed on a personal computer (PC) Dell Optiplex GX270 operating under Microsoft Windows XP operating system and are exempt from software QA and CM requirements in accordance with NSNFP 19.01, Software Control.

4.1.1 MCNP

The MCNP code [CRWMS 1998b] is used to calculate the k_{eff} of the waste package. The software specifications are as follows:

- Status: Qualified
- Software name: MCNP
- Software version/revision number: Version 4B2
- Software tracking number (computer software configuration item): 30033 V4B2LV
- Computer type: Hewlett-Packard (HP) 9000 Series workstations
- Operating system: HP-UX 10.20 and 11.00
- Computer processing unit number: Software is installed on the Idaho National Laboratory (INL) workstation '*homer*', with property number 352667.

The MCNP software used is (a) appropriate for the application of k_{eff} calculations, (b) used only within the range of validation as documented in Civilian Radioactive Waste Management System management and operations [CRWMS 1998a] and Briesmeister [1997], and (c) obtained from the Software Configuration Management in accordance with LP-SI.11 Q-BSC.

4.1.2 SCALE 5.0

At the time the calculations were performed using KENO V.a/TSUNAMI code from SCALE 5.0, it was recognized that the code had not been validated and verified at the INL for these scoping calculations. The INL subsequently qualified SCALE 5.0 [SQAP] and evaluated the impacts on this work. Because no changes occurred to the SCALE software, e.g. 'patches' or revisions as a result of the validation and qualification process, the NSNFP concluded that the TSUNAMI cases reported in this EDF are valid. However, the initial model analyses were processed on the PC compatible version of SCALE 5.0. Subsequent case verifications reran the models on Sun workstations after the SCALE 5.0 software qualification efforts for those computers were completed.

5. Description of Canister and Canister Contents

The canister configurations analyzed will contain either two or three Type 1a baskets containing ATR, MURR, MIT, or ORR SNF. Section 6.1 describes the canister. The Type 1a basket is described in Section 6.2, and each of the four fuel types are described in section 6.3.

Fuel packaging is based on fissile content proposed for repository disposal under fully degraded/moderated conditions in a post-closure scenario that is much more severe than those expected for loading, storage, or transportation. [Ref. CRWMS 2004] Gd poisoning in the basket material is installed to address these post-closure repository scenarios.

5.1 DOE Standardized SNF Canister

The conceptual design for the 18" standardized DOE SNF canister is taken from DOE [1999, p. 5 and A-2]. The DOE SNF canister is a right circular cylinder made from stainless steel pipe (Type 316L, Unified Numbering System [UNS] S31603) with an outside diameter of 18 in. (457.2 mm) and a wall thickness of 0.375 in. (9.525 mm). The minimum internal length of the short canister is 100 in. (2,540.0 mm), and the nominal overall length is 10 ft (approximately 3,000.0 mm). There is a curved carbon steel (American Society for Testing and Materials [ASTM] A 516 Grade 70) impact plate, 50.8 mm (2.0 in.) thick, at the top and bottom boundaries of the canister. Dished heads seal the ends of the DOE SNF canister. The maximum loaded mass is 2,270 kg for the short canister [DOE 1999, Table 3.2]. A sketch of the canister is shown in Figure 6.1.

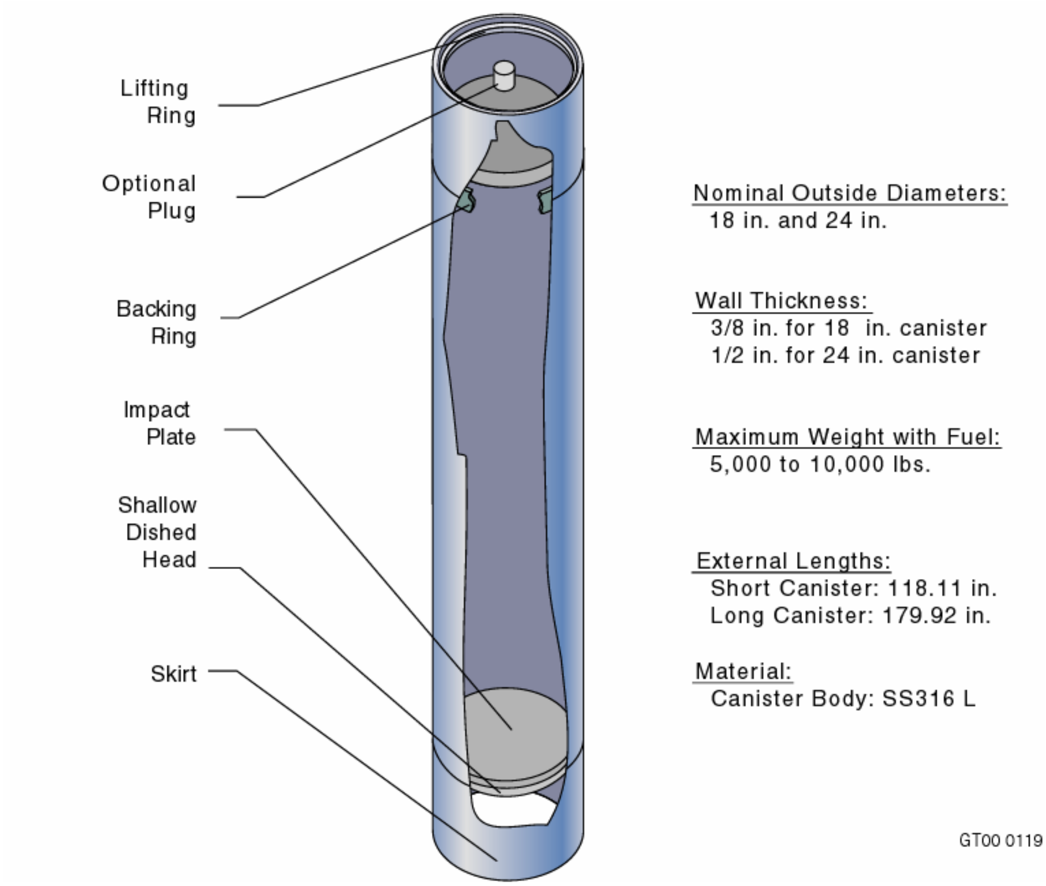

Figure 6.1. Plan view of a typical DOE standardized SNF canister.

At the present time, use of the Type 1a basket is intended for use only in the 18-in. diameter canister. Stacking of baskets within a canister will be dictated by the individual fuel lengths and thereby determine the use of 10-ft. or 15-ft long canisters.

5.2 Type 1a Basket

The DOE SNF canister typically contains a basket structure to hold the spent fuel. The basket is not a part of the DOE SNF canister. The basket structure provides material for controlling criticality, provides structural support, and acts as a guide for assemblies during loading. The basket structure to be used for packaging ATR SNF is designated as a Type 1a basket and is shown in Figures 6.2-1 and 6.2-2. Its length is specified to accommodate the length of the specific fuel to be loaded. The basket compartment dividers are made of low-C-Ni-Cr-Mo-Gd alloy (UNS N06464) with a Gd content of 2.0 wt% [DOE 2004b, pp. 53–55]. The basket structure contains two axial identical sections (layers) each with a circular base plate. The length of each ATR basket is 51-3/4 in. (1,341.45 mm). All Gd alloy plates and the 304L base plate used a thickness of 0.375 in. (9.525 mm). A cross-sectional view is shown in Figure 6.2-1 [DOE 2004b, p. 53]. The basket grid structure is surrounded by a type 304L stainless steel sleeve with an outer diameter of 16.90 in. (429.26 mm) and a thickness of 0.0625 in. (1.587 mm). The Type 1a basket contains 10 compartments. The basket compartments are defined by horizontal and vertical plates as shown in Figures 6.2-1 and 6.2.-2.

Baskets will be stacked on one another in the canister (see Figure 6.3-1). Canisters will then be seal-welded per American Society of Mechanical Engineers (ASME) Division 3 code qualifications. A similar approach will be taken with other fuels in terms of loading in a Type 1a basket. In the case of the shorter fuels, such as the MURR, MIT, and ORR fuels, the basket length will be commensurately shorter but will also be stacked three-high inside an 18-in. diameter, 10-ft long SNF canister.

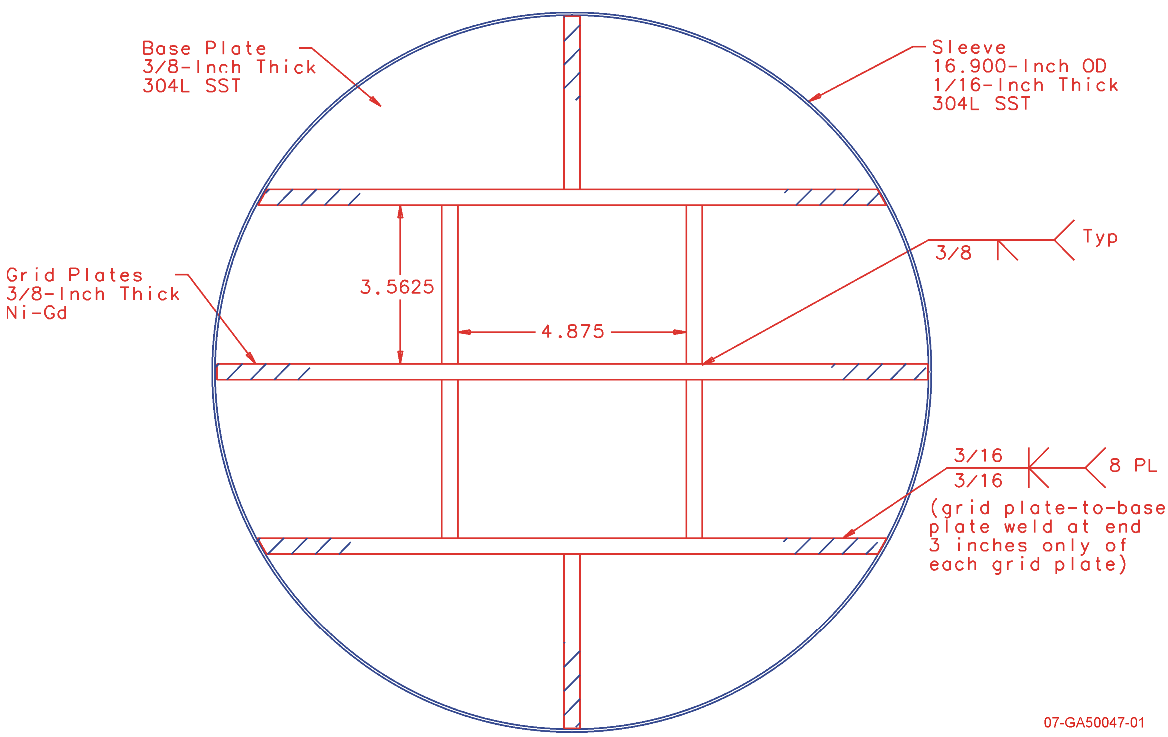

Figure 6.2-1. Cross-sectional schematic of the Type 1a basket structure and sleeve.

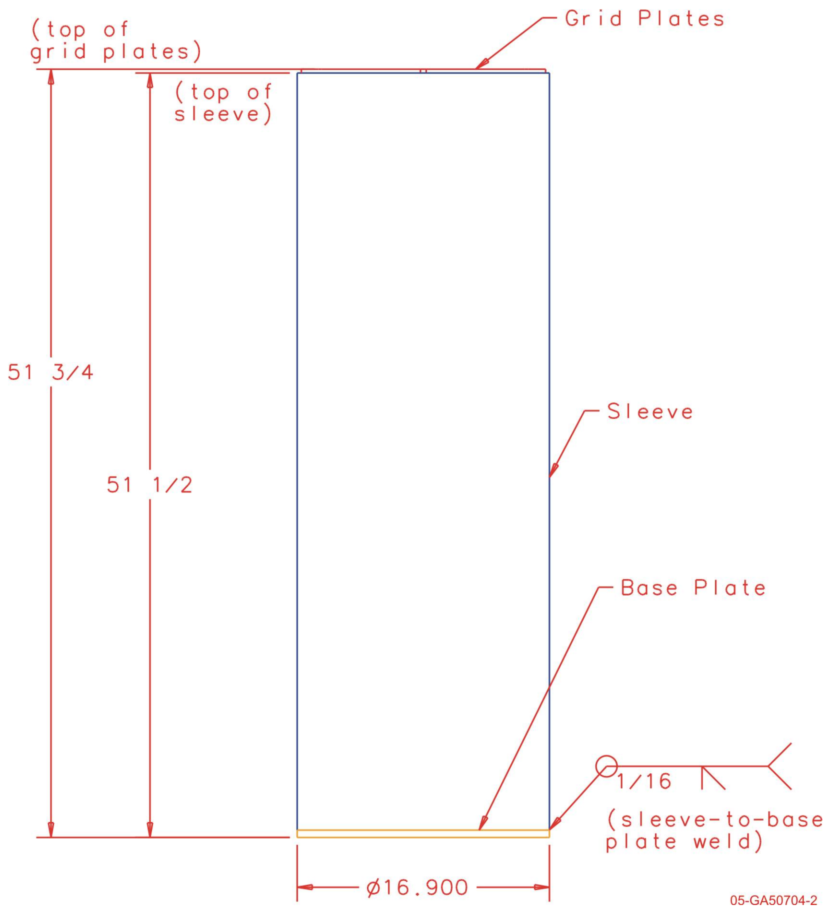

Figure 6.2-2. Type 1a SNF basket (for ATR SNF)

5.3 Spent Nuclear Fuel

The U.S. Department of Energy (DOE) SNF inventory is comprised of more than 250 types of fuels with differing shapes, fuel matrix compositions, fissile isotopes, cladding, and enrichments. In order to streamline the criticality analyses needed to qualify these fuels for acceptance into the national repository, the fuels were grouped by the primary common parameter of the fuel matrix material and the secondary parameter of enrichment. Packaging concepts were developed to satisfy handling requirements in the surface facility (including drop accidents) and post-closure repository conditions that included criticality analyses for waste package breaches. Eventually, the criticality analyses evolved into development of a limited number of generic basket designs, each capable of accommodating a variety of fuels based on their physical size. Regardless of the fuel type, the canister loadings, and any included poisons, are predicated on remaining below the subcritical limit for a degraded canister inside a flooded waste package in a horizontal orientation within the repository.

In the case of the fuel grouping that will be packaged into a canister using a Type 1a basket, four fuels were selected for these criticality analyses based on their fissile content and linear loading. The intent was to identify the bounding criticality analyses for these four fuel types as the basis for demonstrating criticality safety under prescribed transportation scenarios. The analyses would also identify any controls that may be necessary during canister loading to ensure that these analyses bound all loaded configurations. Prior to loading, all the different and disparate fuels identified for loading in a Type 1a basket will have to undergo their own criticality safety evaluation (CSE) based on their as-loaded condition, both dry and flooded. Specific parameters such as total fissile, linear fissile loading, calculated k_{eff}, and thermal output will have to demonstrate lesser values than those of the as-loaded, baseline ATR fuel.

A summary of the fuel loadings for these four fuels in a standardized SNF canister is shown in Table 6.3-1.

Table 6.3-1. Aluminum plate-fuel comparisons.

Fuel identifier →	ATR (HEU/ UAl_x) [10' canister]	ATR (HEU/ UAl_x) [15' canister]	ORR (MEU/ U-Al-Si)	ORR (HEU /U_3O_8)	MIT (HEU/ UAl_x)	MURR (HEU/ UAl_x)
BOL % enrichment	93.15	93.15	20.56	93.15	93.15	93.15
Assemblies/canister	20	30	30	30	30	24
Fissile/assembly (kg)	1.085	1.085	0.347	0.300	0.525	0.783
Fissile/canister (kg)	21.70	32.55	10.41	9.00	15.75	18.79
Canister fissile linear loading (g/cm)	78.619	78.619	41.001	35.447	62.003	73.990
'Homogenized' fissile atom-density/ canister (atom/b-cm)	1.45E-04	1.45E-04	6.79E-05	6.02E-05	1.05E-04	1.26E-04

5.3.1 Advance Test Reactor Fuel

The ATR fuel assemblies will be inserted in each one of 10 compartments in a Type 1a basket, and the baskets will be stacked three high in a 15' canister. Previous criticality analyses had examined ATR assemblies 'two-stacked' in a 10-ft. canister when their cropped length was quoted as <49.5-in. (1257.3 mm). Subsequent information [FRC-0022] revealed a longer assembly length was possible, thereby necessitating longer baskets stacked three-deep in a 15-ft. canister. This resulted in an attendant 50% increase in the fissile load per canister, but no change in the linear fissile load. The 'three-stack' analyses examined the fuel assemblies as though they met the earlier maximum dimension of 49.5-in.; this represents the more conservative (most reactive) case with respect to neutron interaction between assemblies.

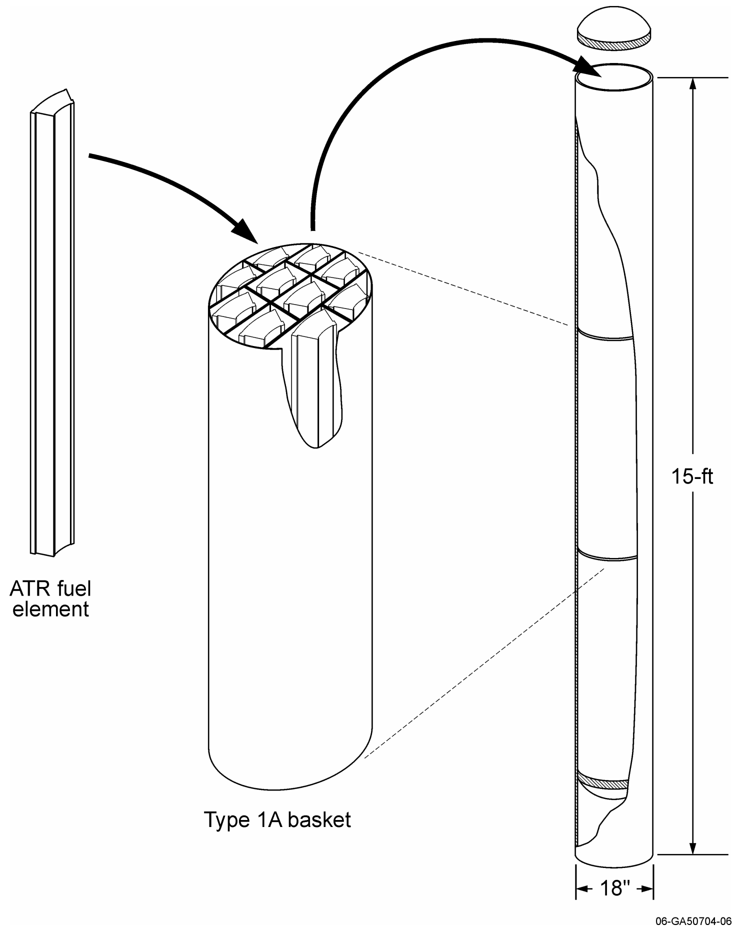

Figure 6.3-1. Conceptual canister.

A typical ATR fuel element consists of 19 curved Al-clad uranium aluminide (UAl_x) plates containing highly enriched uranium (HEU) (93 ± 1 wt% ^{235}U) [Reed et al. 1992]. The nominal fissile loading (^{235}U) of the fresh fuel element is 1,075 g [Paige 1969]. The allowable uncertainty in the fuel loading is 1% or 10.75 g [INEEL 2003]. The highest fissile loading of 1,085.75 g was conservatively used in the present analysis.

Figure 6.3-2 presents a simplified view of a typical ATR fuel element. The fuel elements are cropped to the length of the fuel plates by removing the upper and lower end boxes. The fuel plates are 49.5 in. (1,257.3 mm) long with a fuel zone that is 48.76 in. (1,238.504 mm) long. The cropped length of a fuel assembly can range from 49.5 in.(1,257,30 mm) up to 51.0 in (1,295.40 mm). [FRC-0022]

The following data are characteristics for the ATR 7F fuel elements [Paige 1969]. The thickness of each plate is 0.05 in. (1.27 mm) except plates 1 and 19, which are 0.08 in. (2.03 mm) and 0.1 in. (2.54 mm), respectively. The fuel matrix section in each plate is 0.02 in. (0.51 mm) thick. The cladding is made of aluminum (T-6061). The plates are held in place by aluminum side plates that are 2.55 in. (64.77 mm) wide (radial thickness of the fuel assembly), 0.187 in. (4.75 mm) thick, and 49.5 in. (1,257.3 mm) long. When assembled, the angle of curvature of the fuel elements is 45º with an inner radius of 2.964 in. (75.29 mm) and an outer radius of 5.513 in. (140.03 mm). The detailed dimensions of each fuel plate and fuel matrix are presented in Table 6.1-1.

Analysis of the ATR fuels is based on the compositions provided for the ATR-7F elements from contract C-285, which requires 1,075 g ^{235}U nominal and 1,085.75 g ^{235}U [max] loading. The uranium loading varies by fuel plate by the following amounts:

- Plates 1, 2, 18, and 19 at 32.5 to 33.5 wt% U
- Plates 3, 4, 16, and 17 at 38.1 to 38.8 wt% U
- Plates 10, 11, 12, 13, 14, and 15 at 44.4 to 44.6 wt% U
- Plates 5, 6, 7, 8, and 9 at 44.8 to 45.2 wt% U.

Element weight is comprised of type 6061 Al side plates (1.17 kg), type 6061 Al (0.6 wt% Si) clad and type 1100 Al (0.1 wt% Si) frame (4.42 kg), and fuel matrix (3.02 kg) for a total assembly weight of 8.61 kg. The density of the T6061 Al side plate is 2.7 g/cm^3.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

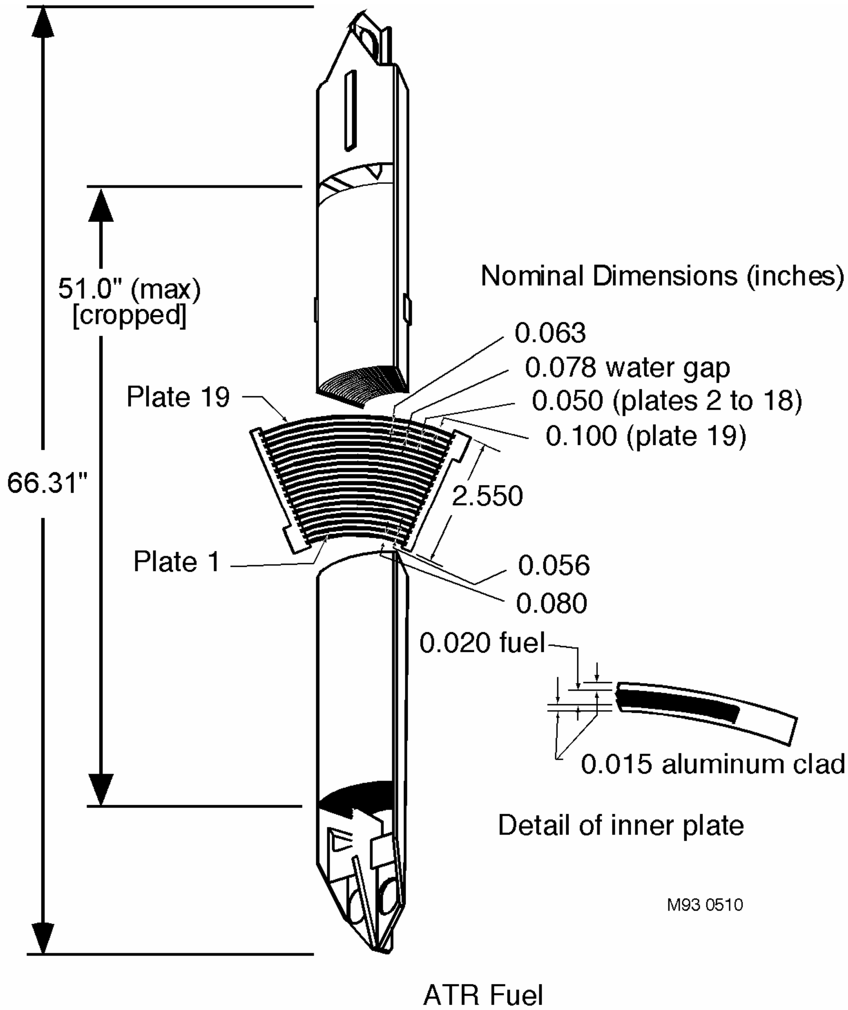

Figure 6.3-2. Simplified view of the ATR fuel element.

Table 6.3-2. Dimensions and fissile loading for individual plates in ATR fuel element.

Plate Number	Inner Radius (mm)	Outer Radius (mm)	Plate Arc Length (mm)	Fuel Meat Arc Length (mm)	^{235}U content, gms (max)
1	76.5810	78.6130	54.1020	41.3258	24.543
2	80.5942	81.8642	55.4228	49.2506	29.391
3	83.8454	85.1154	57.9882	51.8160	39.087
4	87.0966	88.3666	60.5028	54.3306	40.804
5	90.3478	91.6178	63.0936	56.9214	52.621
6	93.5990	94.8690	65.6336	59.4614	55.146

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

Plate Number	Inner Radius (mm)	Outer Radius (mm)	Plate Arc Length (mm)	Fuel Meat Arc Length (mm)	^{235}U content, gms (max)
7	96.8502	98.1202	68.1990	62.0268	57.570
8	100.1014	101.3714	70.7390	64.5668	59.994
9	103.3526	104.6226	73.3044	67.1322	62.418
10	106.6038	107.8738	75.8444	69.6722	64.842
11	109.8550	111.1250	78.4098	72.2376	67.266
12	113.1062	114.3762	80.9752	74.8030	69.690
13	116.3574	117.6274	83.5152	77.3430	72.114
14	119.6086	120.8786	86.0806	79.9084	74.538
15	122.8598	124.1298	88.6206	82.4484	77.063
16	126.1110	127.3810	91.1860	85.0138	64.640
17	129.3622	130.6322	93.7260	87.5538	66.559
18	132.6134	133.8834	96.2914	88.8492	54.338
19	135.8646	138.4046	100.8634	88.0872	53.126
				Total ^{235}U:	1085.75 gms

Source: Paige (1969), ATR 7F fuel element

5.3.2 Missouri University Research Reactor Fuel

The details of the MURR fuel were obtained from the MURR fuel specification [DWG 237]. The MURR fuel assembly is constructed from 24 fuel plates, two side plates, two combs, and several smaller pieces of hardware (screws, nuts, pins, and rivets). The fuel plates are attached to the fuel sections by riveting. Combs are attached to the fuel plates by pinning.

The overall length of a new fuel assembly is 32.5 in. (825.50 mm) [DWG 409] [CRWMS 1997a] with a cropped length of approximately 26.5 in. (673.10 mm) [OBU 2003] after removing the top and bottom ends of the assembly, which do not contain uranium materials. Figure 6.3-3 depicts a simplified view for both length and cross section of a cropped fuel element.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

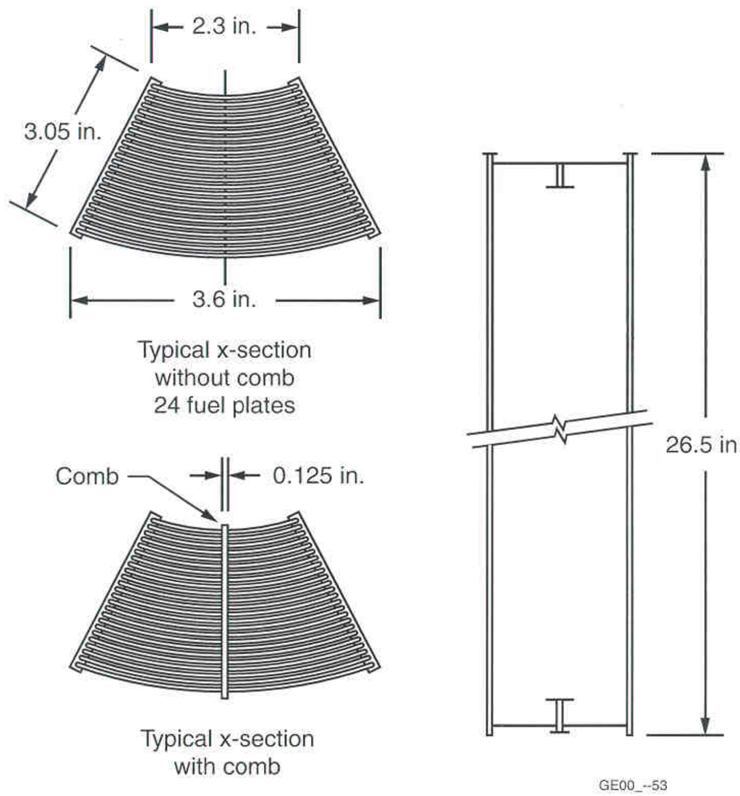

Figure 6.3-3. Simplified view of the MURR fuel element (cropped length).

The fueled portion of each plate can range from 23.25 to 24.75 in. (590.55 to 628.65 mm) [DWG 409]. The fuel matrix alloy is 0.020 in. (0.51 mm) thick, with an aluminum cladding thickness of 0.015 in. (0.38 mm), for total plate thicknesses of 0.050 in. (1.27 mm) [DWG 237].

The width of the individual plates is variable between 1.9929 to 4.3421 in. (50.62 to 110.29 mm), because the fuel element employs 45º wedge shaped construction with curved plates using an outer radius of 5.885 in. (149.48 mm) and an inner radius of 2.675 in. (67.95 mm) [OBU 2003].

Density of the aluminum used for the fuel plate cladding is 2.7 g/cm^3 [CRWMS 1997a]. Table 6.3-3 provides the fissile loading for each plate (element) in the assembly. This fuel matrix material consists of 93 $\pm$ 1.0% enriched U as a UAl_x powder which is in turn dispersed in Al powder and sintered together. Density of this Al powder used in the fuel matrix is 2.7 g/cm^3 [CRWMS 1997a] and density of the UAl_x matrix is 6.4 g/cm^3 [Knight 1993].

Table 6.3-3. MURR fuel assembly plate details.

Plate/ Element No.	UAl_x, Wt. (g) $\pm$3%	Al Matrix Wt. (g) [excluding Al in UAl_x] $\pm$3%	Al Clad Wt (g) $\pm$5%	^{235}U Content (g) $\pm$1%
1	29.0	21.0	80.0	19.260
2	30.9	22.0	86.0	20.393
3	32.7	23.3	89.0	21.526
4	34.6	24.9	93.0	22.659
5	36.2	26.3	100.0	23.793
6	37.9	27.5	106.0	24.926
7	39.5	28.8	108.0	26.059
8	41.2	29.7	113.0	27.192
9	43.0	31.4	116.0	28.325
10	44.6	32.8	120.0	29.459
11	46.6	33.9	123.0	30.592
12	48.1	35.3	128.0	31.725
13	49.9	36.6	131.0	32.858
14	51.5	38.0	131.0	33.992
15	53.3	38.6	136.0	35.125
16	55.0	40.5	138.0	36.258
17	56.8	41.0	138.0	37.391
18	58.5	42.3	140.0	38.524
19	60.2	43.5	143.0	39.658
20	61.9	44.8	147.0	40.791
21	63.6	46.0	150.0	41.924
22	65.6	47.3	154.0	43.057
23	67.3	48.5	170.0	44.191
24	69.0	49.7	172.0	45.324
Totals	1176.9	853.7	3012.0	775.0

5.3.3 Massachusetts Institute of Technology Fuel

The details of the MIT fuel were obtained from CRWMS [1997b]. The MIT SNF plate/assembly drawing is found in drawing (DWG) 419486 [DWG 419]. The MIT fuel assembly is constructed from 15 flat plates tilted at a 60º angle so that the resulting assembly has a rhomboidal (equilateral parallelogram with 60º acute angles) cross section, instead of the more common square or partial-arc cross section. Figure 6.3-4 depicts the MIT fuel assembly. The MIT fuel length values used in these analyses are shorter than the original as-built length of the MIT assembly, because the top and bottom ends of the assembly, which do not contain uranium materials, have been removed by cutting.

The flat plates are 2.552 +0.000, -0.002 in. (64.82 +0.0, -0.05 mm) wide and 23 in. (584.20 mm) long. All 15 plates are the same and have a finned cladding surface with a total thickness of 0.80 ± 0.003 in. (20.32 ± 0.08 mm) including a fin height of 0.010 ± 0.002 in. (0.25 ± 0.05 mm) on both faces. The fuel alloy is 0.030 +0.000, -0.002 in. (0.76 +0.0, -0.05 mm) thick, 2.177 +0.000, -0.1875 in. (55.30 +0.0, - 4.76 mm) wide, and 23.75 ± 0.233 in. (603.25 ± 5.92 mm) long [DWG 419].

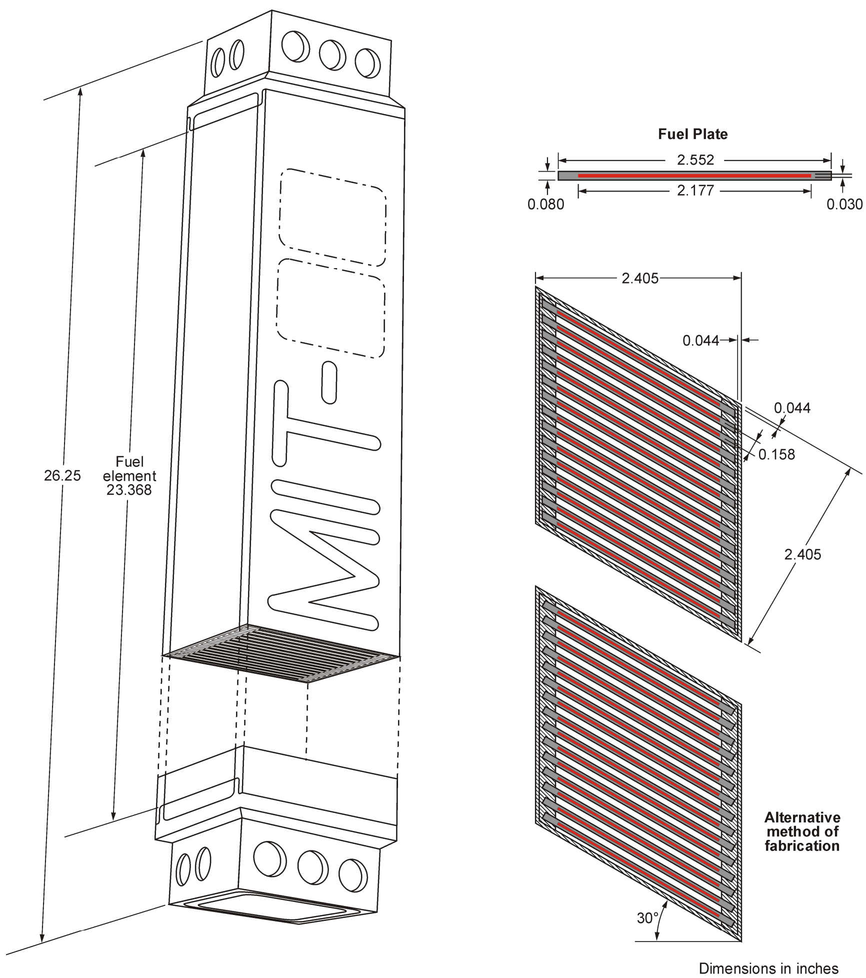

Figure 6.3-4. Simplified view of the MIT fuel element (cropped length).

The aluminum outer shroud that encloses the 15 fuel plates on 4 sides is a 2.405 in. (61.09 mm) outside dimension rhomboid with a 0.044 in. (1.12 mm) thick wall parallel with the fuel plates and a 0.188 in. (4.78 mm) thick comb plate at 60˚ to the fuel plates, and a nominal length (after cutting) of 23.368 in. (593.55 mm). The fuel plates are centered within this rhomboid angled 60˚ off the comb plate. The plates are fixed relative to each other by comb plates along two sides and the lip of the end fittings across the top and bottom. Drawing 419486 [DWG 419] shows a fuel plate center-to-center spacing of 0.158 in. (4.01 mm), which is the spacing of the notches on the comb plates [CRWMS 1997b].

The fuel plates consist of an aluminum cladding over UAl_x alloy. The maximum fuel mass for the MIT assembly is 514.25 g of ^{235}U with an enrichment of 93.5 wt%, and 1 wt% of U^{234}. The fissile atom-densities are not tabulated in this case, as all the fissile material is uniformly distributed over each plate in the assembly. The Al present in the UAl_x alloy is 30.5 wt%. The UAl_x alloy has a significant void volume if distributed over the maximum dimensions of the assembly, and thus can become waterlogged with a resultant increase in reactivity. The maximum void volume fraction in the fuel assembly, including space between the plates, is 0.6353 [CRWMS 1997b].

5.3.4 Oak Ridge Research Reactor Fuel

Details for the construction of the ORR fuel element are contained in drawings [OR-001], [OR-003], and [OR-004]; a simplified depiction of a fuel assembly appears in Figure 6.3-5. The element is constructed from 19 curved fuel plates that are held within two opposing aluminum comb plates. The ORR fuel length values used in these analyses are shorter than the original as-built length of the ORR assembly, because the top and bottom ends of the assembly, which do not contain uranium materials, have been removed by cutting. The ORR fuel description [CRWMS 1997b] contains the material information. The maximum fuel mass used for the ORR analysis is 347 g (see clarification following Table 6.3-4) of ^{235}U at an enrichment of 20.56 wt%. The uranium present in the U-Si-Al alloy is 77.5 wt%. There are two atoms of Si per three atoms of U, and Al completes the bulk of the fuel material. The U-Si-Al has a significant void volume if distributed over the assembly dimensions; the maximum void volume fraction in the assembly (including plate spacing) is 0.4064 [CRWMS 1997b].

The curved plates are. 2.776 +/- .01 in. (70.32 +/- .07 mm) maximum wide with a 5.5 in. (139.70 mm) inner radius of curvature. Seventeen of the plates comprise the inner plates, with a thickness of 0.0494 to 0.0510 in. (1.25 to 1.30 mm) and a 0.0105 in. (0.27 mm) minimum aluminum cladding on both sides of a 0.020 in. (0.51 mm) nominal fuel foil. A plate tolerance of 0.005 in. (0.13 mm) is the default value shown on the drawing. The two outer plates have a thickness of 0.063 to 0.066 in. (1.60 to 1.68 mm) with a 0.018 in. (0.46 mm) minimum cladding on both sides of a 0.020 in. (0.51 mm) nominal fuel foil. The inner and outer fuel plates are manufactured as flat laminated sheets with a 2.7925 in. (70.93 mm) minimum and 2.7955 in. (71.01 mm) maximum width that are formed to the 5.5 in. (139.70 mm) radius curvature. The fuel foil is not as wide as the aluminum cladding, and an aluminum strip is used to close each side of the finished fuel plate. For the inner fuel plates, the width of the fuel foil allows a 0.126 to 0.200 in. (3.20 to 5.08 mm) inset from the edge of the plate on both sides. The overall length of the outer plate is 27.120 to 27.130 in. (688.85 to 689.10 mm), and the fuel foil is centered within the plate longitudinally, with an inset at each end of 1.574 to 2.011 in. (39.98 to 51.08 mm). The top and bottom ends of the inner and outer fuel foils are chamfered, but this trimming of the fuel material was neglected [CRWMS 1997b].

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

The aluminum comb plates enclose the 19 fuel plates on two sides, creating an approximate 3.25 in. (82.55 mm) by 3.00 in. (76.20 mm) outside dimension rectangle and a nominal length (after cutting) of 27.125 in. (688.98 mm). The fuel plates are centered within this box and form a square fuel/water region with a 3.169 in. (80.49 mm) reference dimension (the longitudinal comb plate width). The plates are fixed relative to each other by comb plates along two sides and by a comb strap across the top and bottom. Drawing M-11495-OR-003, "Misc. Details for ORR Fuel Element", shows a fuel plate edge-to-edge spacing of 0.166 in. (4.22 mm), which is the spacing of the notches on the comb plates [CRWMS 1997b].

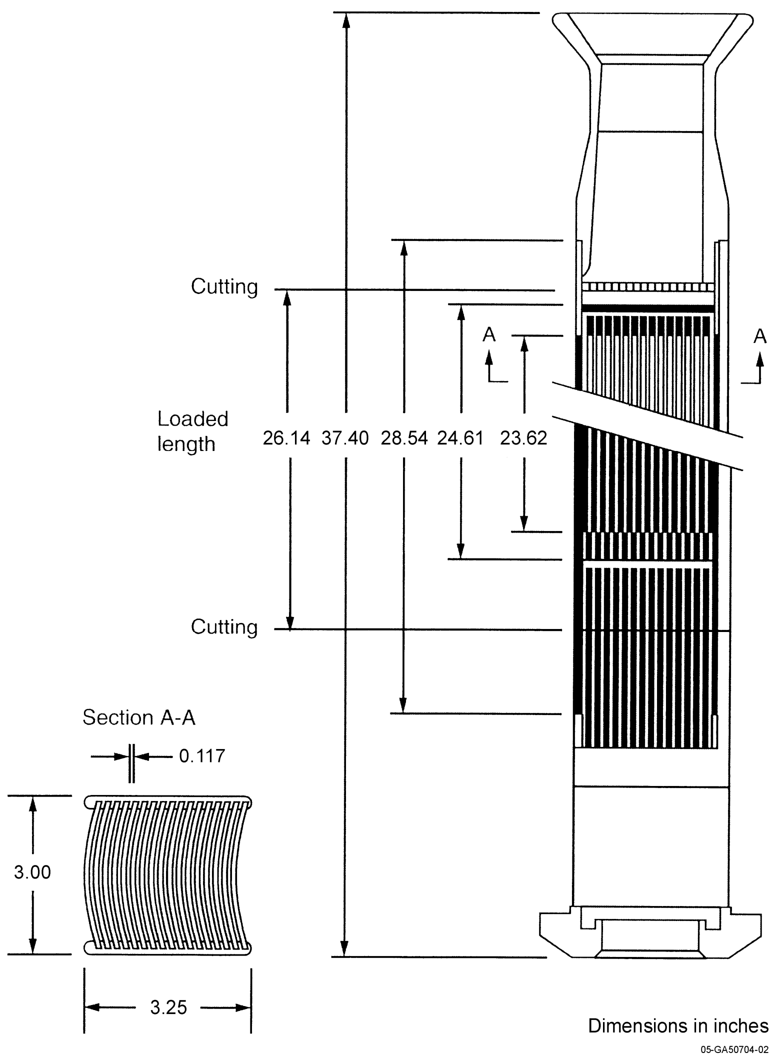

Figure 6.3-5. Simplified view of the ORR fuel element.

The ORR fuel Appendix A [OBU 2003] contains the material information associated with these fuels. The ORR fuel is an example of the variability that can exist across a fuel group. A summary of these variables (plate count per assembly, matrix composition, or enrichment) is reflected in Table 6.3-4 [OBU 2003].

Table 6.3-4. Variability in ORR fuel types.

		*BOL (per plate)				BOL (per assembly)	
No.	Type	^{235}U (g)	Total U (g)	Enrichment (%)	Plates (#)	^{235}U (g)	Total U (g)
1	19 plate U_3O_8 MEU**	17.9	90.63	19.75%	19	340	1721.5
2	19 plate U_3O_8 HEU***	15	16.13	92.99%	19	285	306.47
3	15 plate U_3O_8 HEU	11.13	11.97	92.98%	15	166.95	179.55
4	19 plate UAl_x HEU	21.85	48.65	44.91%	19	415.15	924.35
5	13 plate UAl_x MEU	24	121.21	19.80%	13	312	1575.73
6	13 plate U_3O_8 MEU	26.17	133.38	19.62%	13	340.21	1733.94
7	19 plate U_3Si_2 MEU	17.89	90.58	19.75%	19	339.91	1721.02
8	17 plate U_3Si_2 MEU	17.89	90.58	19.75%	17	304.13	1539.86
9	15 plate U_3Si_2 MEU	13.33	67.49	19.75%	15	199.95	1012.35
10	11 plate U_3Si_2 MEU	18	90.9	19.80%	11	198	999.9

*BOL = beginning of life
** MEU = medium-enriched uranium, which ranges from >5% and <20% enrichment
***HEU = highly enriched uranium (usually refers to fuel containing at least 93% ^{235}U).

The analyses done for the ORR fuels used two different fuels from this list in Table 6-17 (No. 2 and No. 6). Fuel No. 2 lists a ^{235}U content of 285 g assembly with a fissile enrichment of 92.99%. The analysis applied an increased fissile loading of 300 g ^{235}U per assembly at the same enrichment; use of this fuel in the analysis will be denoted as ORR HEU. Fuel No. 6 used 347 g ^{235}U rather than the listed 340.21 g with its 19.62% enrichment; use of this fuel in the analysis will be denoted as ORR medium-enriched uranium (MEU). Where all of these fuels were irradiated in the same reactor using the same control rods, reactivity should be considered relatively consistent between assembly types.

5.4 Materials Description

Tables 6.4-1 through 6.4-6 show the composition of the material used in the models. Material nomenclature for the materials used throughout this document includes: UNS S31603 stainless steel (referred to as 316L stainless steel); UNS S31600 stainless steel (referred to as 316 stainless steel); UNS S30403 stainless steel (referred to as 304L stainless steel); UNS K02700 carbon steel (referred to as A516 Grade 70 carbon steel); UNS N06464 low-C-Ni-Cr-Mo-Gd alloy (also referred to as Ni-Gd alloy); and UNS A96061 Al (referred to as Aluminum 6061).

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

Table 6.4-1. Composition and density of stainless steel 304L. (basket material)

Element	Composition (wt%)	Value Used (wt%)	Number Density [atom/(cm·b)]
Carbon (C)	0.03 (max.)	0.03	1.1943E-04
Manganese (Mn)	2.00 (max.)	2.00	1.7407E-03
Phosphorus (P)	0.045 (max.)	0.045	6.9467E-05
Sulfur (S)	0.03 (max.)	0.03	4.4869E-05
Silicon (Si)	0.75 (max.)	0.75	1.2769E-03
Chromium (Cr)	18.00–20.00	19.00	1.7472E-02
Nickel (Ni)	8.00–12.00	10.00	8.1467E-03
Nitrogen (N)	0.1	0.10	3.4138E-04
Iron (Fe)	Balance	68.045	5.8262E-02

Source: ASTM A 240/a 240M-97a, p.2. (UNS S30403)
Density = 7.94 g/cm^3.

Table 6.4-2. Composition and density of stainless steel 316L. (canister material)

Element	Composition (wt%)	Value Used (wt%)	Number Density [atom/(cm·b)]
Carbon (C)	0.03 (max.)	0.03	1.2003E-04
Manganese (Mn)	2.00 (max.)	2.00	1.7495E-03
Phosphorus (P)	0.045(max.)	0.045	6.9819E-05
Sulfur (S)	0.03 (max.)	0.03	4.5092E-05
Silicon (Si)	1.00 (max.)	1.0	1.7111E-03
Chromium (Cr)	16.00–18.00	17.00	1.5712E-02
Nickel (Ni)	10.00–14.00	12.00	9.8253E-03
Molybdenum (Mo)	2.00–3.00	2.50	1.2523E-03
Nitrogen (N)	0.10 (max.)	0.10	3.4310E-04
Iron (Fe)	Balance	65.295	5.6189E-02

Source: ASTM A 276-91a, p.2 (UNS S31603)

Density = 7.98 g/cm^3.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

Table 6.4-3. Composition and density of carbon steel A516 Grade 70 (canister impact plates)

Element	Composition (wt %)	Value Used (wt%)	Number Density [atom/(cm·b)]
Carbon (C)	0.30 (max.)	0.30	1.1021E-03
Manganese (Mn)	0.85–1.20	1.025	8.9921E-04
Phosphorous (P)	0.035 max.)	0.035	5.3419E-05
Sulfur (s)	0.035 (max.)	0.035	5.1751E-05
Silicon (Si)	0.15–0.40	0.275	4.8813E-04
Iron (Fe)	Balance	98.33	8.3225E-02

Source: ASTM A 516/A 516M-90, Table 1 (UNS K02700); >2” to 4” thick plate
Density = 7.85 g/cm^3.

Table 6.4-4. Composition and density of Ni-Gd alloy. (poison material in basket grid)

Element	Composition (Value Used) (wt%)	Number Density [atom/(cm·b)]
Gd	2	6.7096E-04
Mo	14.55	8.0005E-03
Cr	15.8	1.6030E-02
Fe	1	9.4465E-04
Co	1	8.9515E-04
Ni	65.65	5.9007E-02

Density = 8.73 g/cm^3
Source: ASTM B 932 - 04, Table 1 (UNS N06464).

Table 6.4-5. Composition and density of Aluminum 6061. (cladding material)

Element	Composition[a] (wt%)	Value Used (wt%)
Mg	0.8–1.2	1
Si	0.4–0.8	0.6
Fe	0.7 (max)	0.7
Cu	0.15–0.4	0.275
Cr	0.04–0.35	0.195
Mn	0.15 (max)	0.15
Zn [Note a]	0.25 (max)	0.25
Ti	0.15 (max)	0.15
Al	Balance	96.68

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

Element	Composition[a] (wt%)	Value Used (wt%)

Density[b] = 2.7065 g/cm^3
Source: a. ASM International 1990, p. 102; b. ASTM G 1-90, Table X1 indicates 2.7 g/cm^3; ASME 2001, Section II, Table NF-2 indicates a converted value from 0.098 lb/in.^{3} of 2.713 g/cm^3; therefore, the midpoint was used.

NOTE: See Assumption 4.4.

Table 6.4-6. ATR fuel number densities used in this report.

	Number densities used in this evaluation [atom/(cm·b)]			
Plate No.	^{235}U	^{238}U	^{234}U	Al
1	2.5677E-03	1.0917E-04	3.2902E-05	5.0781E-02
2	2.4883E-03	1.0580E-04	3.1885E-05	5.0875E-02
3	3.1422E-03	1.3360E-04	4.0264E-05	5.0098E-02
4	3.1228E-03	1.3277E-04	4.0015E-05	5.0121E-02
5	3.8426E-03	1.6338E-04	4.9239E-05	4.9266E-02
6	3.8506E-03	1.6372E-04	4.9341E-05	4.9256E-02
7	3.8512E-03	1.6374E-04	4.9348E-05	4.9256E-02
8	3.8517E-03	1.6377E-04	4.9355E-05	4.9255E-02
9	3.8522E-03	1.6379E-04	4.9361E-05	4.9255E-02
10	3.8526E-03	1.6381E-04	4.9367E-05	4.9254E-02
11	3.8531E-03	1.6383E-04	4.9372E-05	4.9253E-02
12	3.8535E-03	1.6384E-04	4.9378E-05	4.9253E-02
13	3.8538E-03	1.6386E-04	4.9383E-05	4.9253E-02
14	3.8542E-03	1.6387E-04	4.9387E-05	4.9252E-02
15	3.8596E-03	1.6410E-04	4.9456E-05	4.9246E-02
16	3.1388E-03	1.3346E-04	4.0220E-05	5.0102E-02
17	3.1365E-03	1.3336E-04	4.0190E-05	5.0105E-02
18	2.5232E-03	1.0728E-04	3.2331E-05	5.0834E-02
19	2.5234E-03	1.0729E-04	3.2334E-05	5.0834E-02

Fuel constituents such as boron are not included. The fissile number densities represent a slightly increased value for incorporating added conservatism into the model, e.g. 94% enrichment with 1085.75 g ^{235}U per assembly.

6. Calculations

This section describes the calculations performed to evaluate the k_{eff} of a sealed SNF canister containing one or more Type 1a baskets containing either ATR, MURR, MIT, or ORR fuels. Section 6.1 and 6.2 provide the details of the SNF canister and canister basket respectively. Section 6.3 covered the various fuels and their characteristics. Section 6.4 gives the composition of the materials used in the following calculations. The basic formulas used in these calculations are listed in Section 7.1. The intact, loaded configurations of the SNF canisters are outlined in Section 7.2, and Section 7.3 describes the various degraded conditions of the SNF canister internals that were evaluated.

Avogadro's number and atomic weights are from the chart of the radionuclides in Parrington et al. [1996]. The number of digits in the values cited here may be the result of a calculation or may reflect the input from another source; consequently, the number of digits should not be interpreted as an indication of accuracy. The metric units used in this document are calculated using the English units given in the cited references. The differences that might exist between the metric units calculated and any metric units cited in references have negligible effect on the calculations, and should not be interpreted as an indication of accuracy.

6.1 Formulas

The basic equation used to calculate the number density values for materials composed of one or more elements/isotopes is shown below. It is used in the calculations used to produce the results summarized in the section 7:

$$N_i = (m_i/m) * \rho * N_a / M_i = (V_i/V) * \rho_i * N_a / M_i$$

Where:

N_i = the number density in atoms/cm^3 of the i^{th} element/isotope

m_i = the mass in grams of the i^{th} element/isotope in the material

m = the mass in grams of the material; note that $m = \sum m_i$

N_a = the Avogadro's number ($6.022*10^{23}$ atoms/mole [Parrington et al. 1996, p. 59])

M_i = the atomic mass in g/mole of the i^{th} element/isotope

M = the atomic mass in g/mole of the material

V_i = the volume in cm^3 of the i^{th} element/isotope in the material

V = the volume in cm^3; note that $V = \sum V_i$

ρ_i = the density of the i^{th} element/isotope (g/cm^3)

ρ = the density of the material (g/cm^3); note that $\rho = \sum \rho_i{*}(V_i/V)$.

Volumes of cylinder segments (volume = area of circle segment x length of the cylinder) are also calculated, based on the equation for the area of a segment of a circle shown below [Beyer 1987, p. 125]:

$$\text{Area of a segment of a circle} = \left(R^2 \cos^{-1}\left(\frac{R-h}{R}\right) - (R-h)\sqrt{2Rh - h^2} \right)$$

Where:

R = the cylinder radius

h = the height of the segment.

6.2 "As-Loaded" Configurations

In the "as-loaded" configuration, the fuel, the canister, and its internal components are considered to be in their intended configuration. Modeling of the end structure of the DOE SNF canister treats both the impact plate and the dished head as a single piece that serves as an end reflector. The curved gap between the impact plate and the dished head is conservatively modeled as filled with carbon steel. Unless noted otherwise, the unoccupied spaces inside the DOE SNF canister are dry with the exception of an assumed water-logging of the fuel matrices – see Assumption 4.5.

Figure 7.2-1 presents a cross-sectional view of the baseline intact ATR (20) configuration modeled with MCNP. Figures 7.2-2 through 7.2-4 reflect the baseline model configuration for the other aluminum plate fuels in their as-loaded configurations inside a Type 1a basket. For conservatism, the fuels are placed artificially close to one another in the basket compartments for calculating maximum reactivity. Reality suggests a less optimal positioning of the fuel assemblies within the basket compartments at time of loading.

Analyses address the as-loaded configuration for single-canister scenarios, and three different scenarios involving an array of canisters in a hypothetical transportation cask (see section 7.4).

The results presented in Table 7.2-1 and Figures 7.2-1 through 7.2-4 show the relatively insignificant reactivities for single canisters, even when water-reflected. Both fuel and canister baskets are intact in these baseline analyses. The table and figures also show the small increase to single canister reactivity with the omission of Gd inside a water reflected canister.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

The loading of the MURR fuels will utilize a Type 1a basket configuration that precludes loading the assemblies in the two center compartments (the MURR (24) center void). The results of these analyses are plotted in Figure 7.2-5.

Table 7.2-1. Water-reflected, loaded SNF canister with dry/intact internals.

Fuel Type	Code Case with Gd	k_{eff} +2σ	Code Case without Gd	k_{eff} +2σ
ATR (20*)	atr20.o	0.1341	atr20-nogd.o	0.1825
ATR (30**)	atr30.o	0.1367	atr30-nogd.o	0.1857
MURR (24*)	mu24c.o	0.1583	mu24c-nogd.o	0.2016
MIT (30*)	mit30.o	0.1277	mit30-nogd.o	0.1742
ORR-HEU (30*)	orr30.o	0.1115	orr30-nogd.o	0.1549

* SNF count in a 10-ft canister
** SNF count in a 15-ft canister

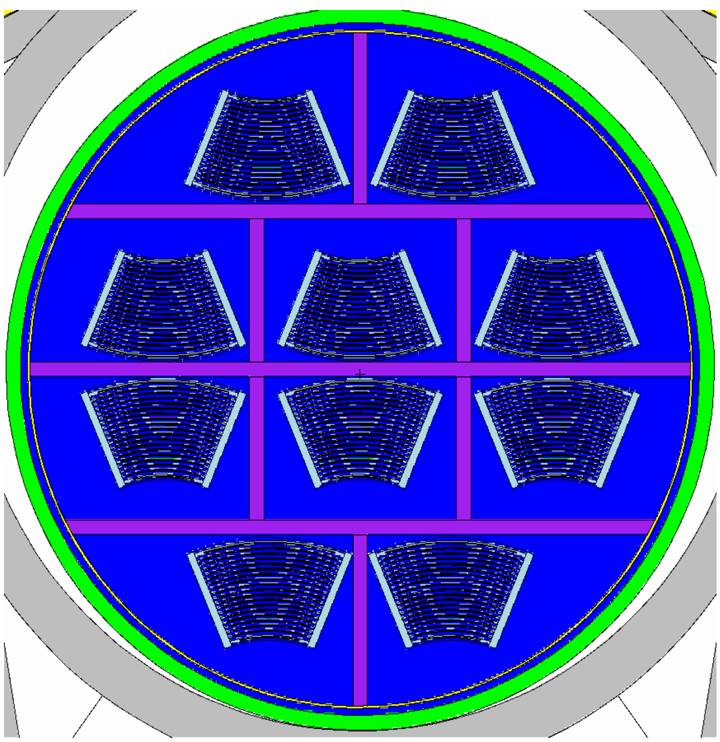

Figure 7.2-1. ATR fuel assemblies in a Type 1a basket.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

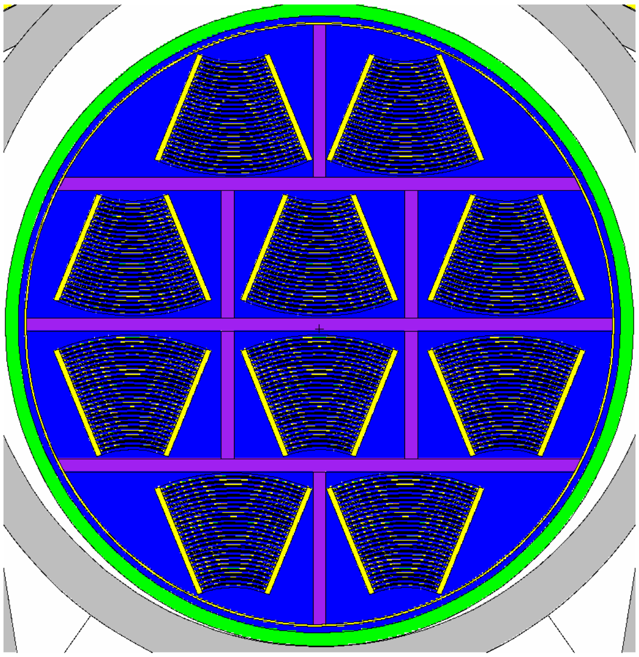

Figure 7.2-2. MURR fuel assemblies in a Type 1a basket.

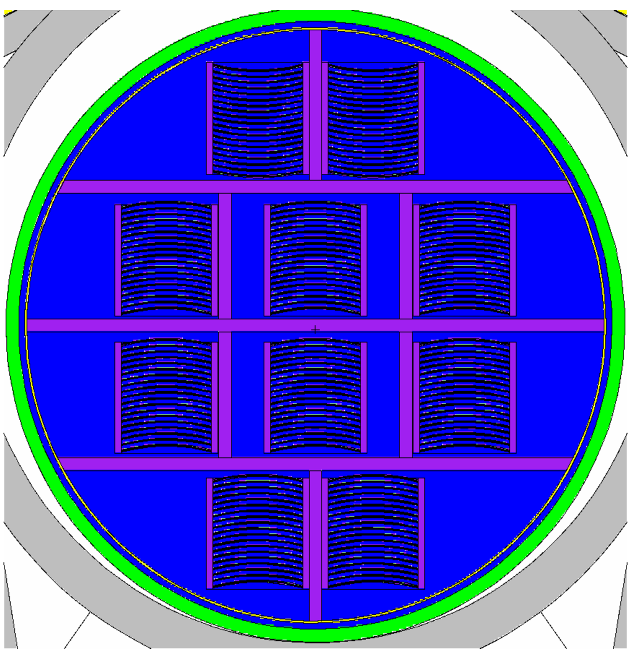

Figure 7.2-3. ORR fuel assemblies in a Type 1a basket.

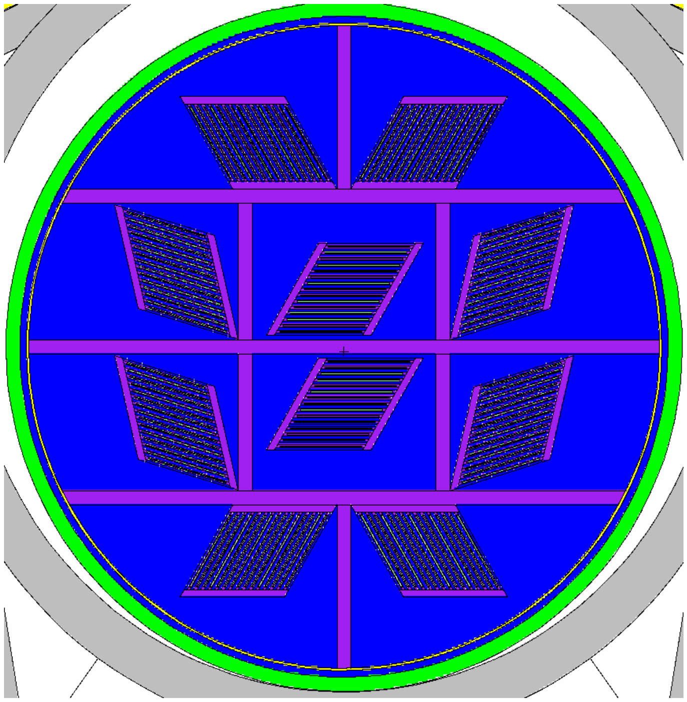

Figure 7.2-4. MIT fuel assemblies in a Type 1a basket.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

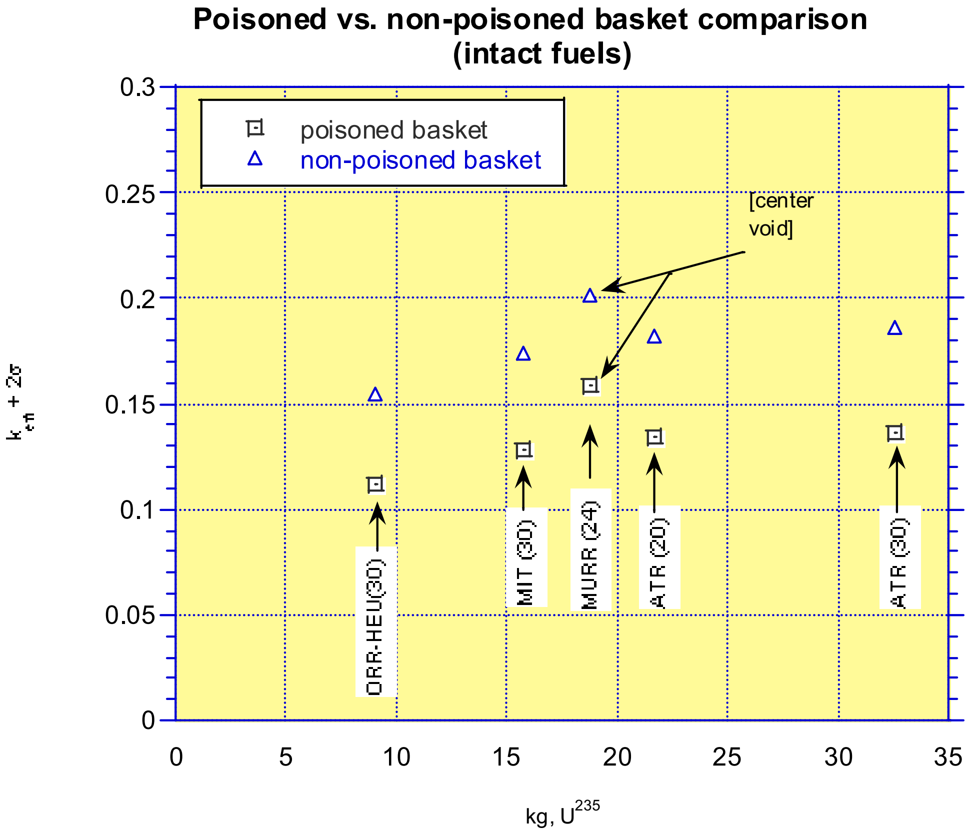

Figure 7.2-5. Gd effect on system reactivity for intact internals in a single, water reflected canister with and without poisoning.

Any baseline comparison case warrants an understanding of what the concentration of gadolinium (Gd) in the basket plate plays in meeting criticality safety. At issue is the credit allowed for any installed poisons. The Gd concentration in the basket plate material (UNS N06464; see Table 6.2-4) is installed to meet issues relative to total degradation of the fissile material inside a breached waste package within the post-closure repository environment. The presence of Gd is inconsequential to assuring criticality safety in a leak-tight canister with Type 1a basket and non-degraded fuels. No credit is being sought for the presence of Gd in the basket plates for any single canister configuration prior to the post-closure repository. Its presence is included in the model, because it is integral to any sealed canister utilizing a Type 1a basket configuration destined for repository disposal.

For comparison purposes, it is important to demonstrate the margin of safety for a single canister that could potentially be loaded underwater. Such a condition (both filled and reflected with water) is not likely to occur because of the operational difficulties associated with drying and sealing such a configuration. The calculated reactivities shown in Table 7.2-2 and Figure 7.2-6 portray that the proposed canister loadings can be enveloped with the ATR analyses (as fissile loads per canister decrease, so do the calculated reactivities). The comparative cases between the ATR (20) and ATR (30) loads demonstrate

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

the "infinite cylinder" principal, where, after a certain length and given the same linear fissile distribution, the calculated reactivity remains relatively constant.

Table 7.2-2. Results of intact fuels in a fully-flooded canister.

Fuel Type	Code Case	k_{eff} +2σ
ATR (20)	d1cnba.o	0.6805
ATR (30)*	d1cnbb+L.o	0.6810
MURR (24)	m1a24c.o	0.6397
MIT (30)	mit1d.o	0.6135
ORR-MEU (30)	o1bleu.o	0.6439

* in 15-ft long canister; all others in 10-ft canisters

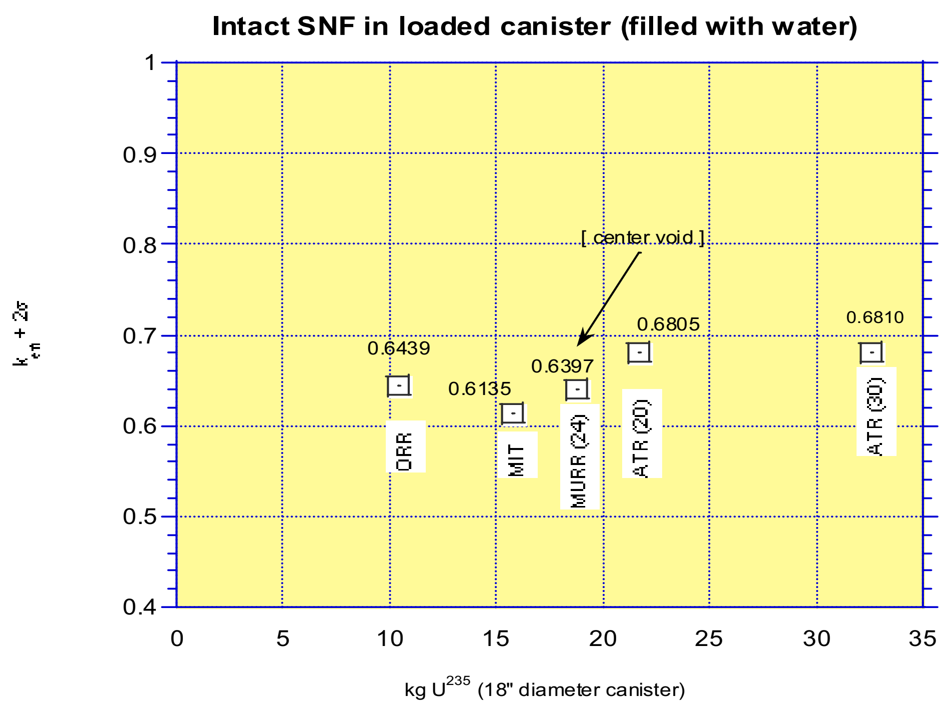

Figure 7.2-6. Calculated reactivities for intact SNF loaded in a fully-flooded canister.

6.3 Degraded Configurations

Various forms of both fuel and basket degradation might occur depending on the particular accident scenario. The variables in terms of degree of degradation (partial versus total collapse) and redistribution of canister internals (both fuel and basket) are too numerous to model in any practical manner for the most credible condition. Rather, the models selected focus on the most reactive configurations, and demonstrate with supplemental models how those most reactive cases satisfy that contention. This places some reliance on the use of non-mechanistic configurations to demonstrate maximum reactivities.

In general, highly-enriched fissile material in dry systems needs to attain the highest (most concentrated) fissile atom-density possible for maximum reactivity. Conversely, those same systems become more reactive in moderated conditions as they become more homogenous at an optimum H/X ratio and then less reactive as the fissile concentration becomes more dilute. The baseline cases addressed in these degraded case models demonstrate both these contentions.

Degraded mode calculations consider both horizontal and vertical orientation of the canisters, for a single canister as well as a multi-canister arrays inside a hypothetical transport cask. Difficulties arise when trying to apportion and fixate the location of potentially reconfigured fissile material after any drop accident. Hence, degraded scenarios for these analyses conservatively assumed complete separation of the UAl_x/Al fissile matrix from any of the cladding and assembly end fittings. Thereafter, the fuel matrix material is distributed within the basket compartments anywhere from a full-density matrix layer to uniform (homogeneous) distribution with a calculated void fraction.

Specialized cases evaluated “what if” scenarios with axial reconfiguration of the fissile mass at the bottom of a vertically oriented canister and away from any of the poisoned basket material; the basket material becomes a reflecting surface at this point. For the worst case (non-mechanistic) scenario, all the fissile mass in a canister, i.e. from all the assemblies in all the baskets, is reconfigured at the fissile matrix production density either as a sphere or a cylindrical shape limited to the internal diameter of the canister. The more reactive of these two configurations was then used as the basis for analyzing the nine-pack canister array inside the hypothetical transport cask.

Scenarios for fuel degradation in transportation are confined to ‘dry’ reconfiguration as a result of some transportation accident. For ‘dry’ fissile systems, greater compactness of the fissile material generally promotes greater system reactivity. The following degradation scenarios portray what are progressively more reactive dry configurations. For fissile materials separated from the cladding, the cladding ‘disappears’ from the basket compartment as the most conservative approach. Any inclusion would beg the question as to what orientation or position should be assumed, e.g. submerged in the matrix debris or floating on top.

6.3.1 Fuel Debris Separated and Horizontally Reconfigured in Basket Compartments

With horizontal orientation of a canister, either after a drop or during reorientation after a drop, analyses evaluated reactivity with the fissile matrix material separated from between the clad plates of the ATR fuel and then reconsolidated on the side wall of each basket compartment. A void fraction of varying percentages was then assigned to the fuel rubble to determine the effect on calculated reactivity.

Figure 7.3.1-1 depicts the configuration analyzed. All compartments were modeled with similar fuel rubblization, although the outer perimeter compartments result in slightly varying heights because of the canister radius influence on compartment area cross section.

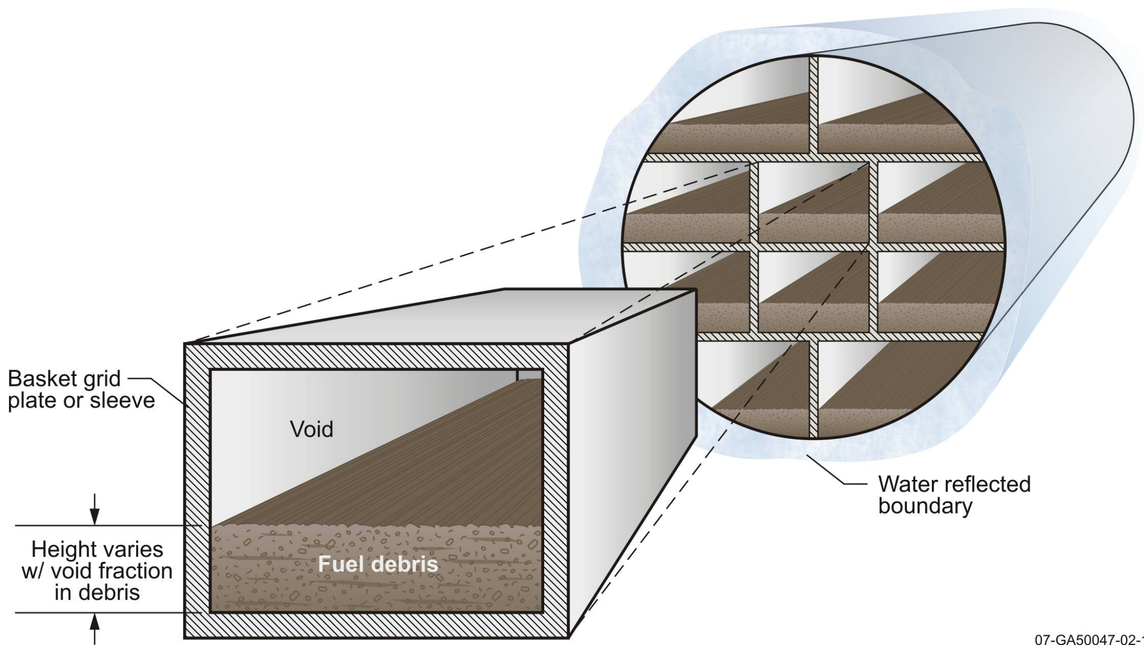

Figure 7.3.1-1 Cross section of canister with fuel debris separated and horizontally reconfigured in a Type 1a basket compartment

The first of the degraded cases examined rubblized fuel, with the fuel matrix material separating from between the aluminum cladding and depositing as a layer on the floor of each basket compartment in a horizontally oriented canister; see Figure 7.3.1-1. Table 7.3.1-1 itemizes the results of the criticality calculation for this configuration; Figure 7.3.1-2 depicts the results listed in Table 7.3.1-1. This analysis was done with a nominal 11 vol % water assumed trapped in the interstitial voids within the ATR fuel matrix; this equates to an H/X ratio of approximately 2.1; analysis of other fuels used varying 'trapped water' content per Assumption 4.5. These analyses were done with four different fuel types for non-flooded cases within the SNF canister. The individual canisters were externally water reflected in all cases.

Table 7.3.1-1. Effect of fuel rubblization in a horizontally oriented basket compartments.

Fuel Type	Code Case (with 11 vol% Water in SNF)	k_{eff} +2σ
ATR (20)	homa_0-w.o	0.0855
ATR (30)*	homa30-w.o	0.0894
MURR (24)	homa24c-w.o	0.0885
MIT (30)**	mthom_0-w.o	0.0844
ORR-HEU (30)***	ohom_0-w.o	0.0463

* in 15-ft long canister; all others in 10-ft canisters

** 19.75 vol% water in fuel matrix

*** ~30 vol% water in fuel matrix

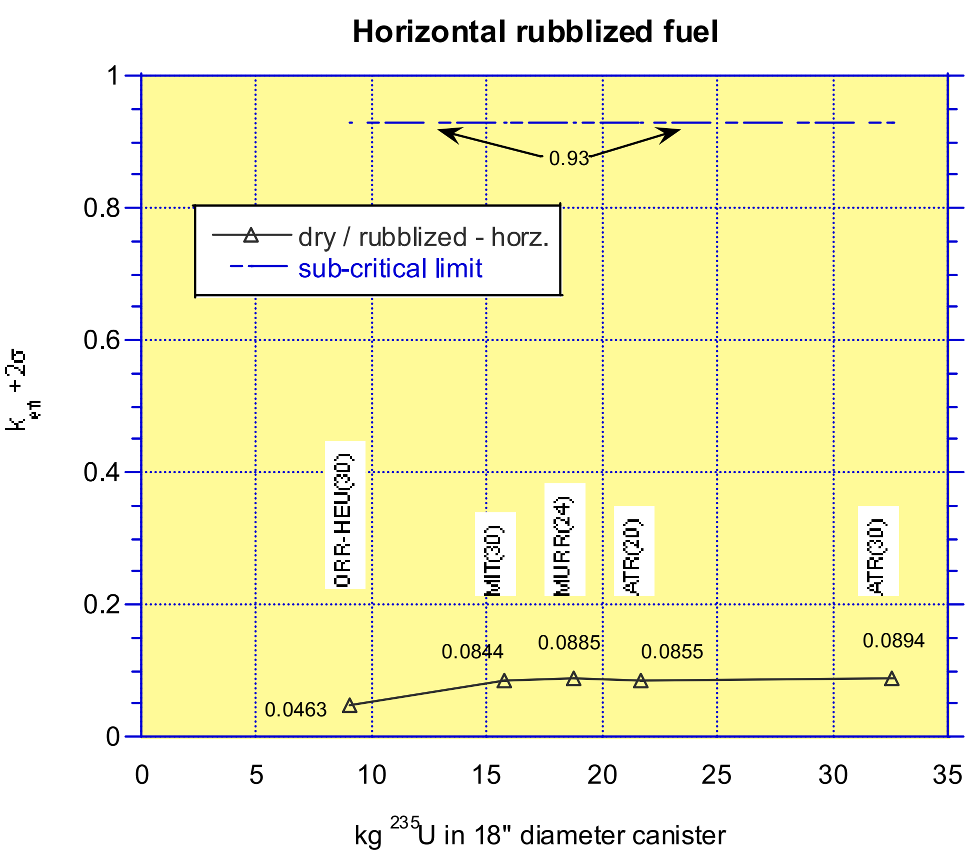

Figure 7.3.1-2. Rubblized fuel in a horizontal canister.

6.3.2 Fuel Debris Separated and Vertically Reconfigured in Basket Compartments

While SNF canisters are typically stored, shipped and emplaced in the repository with horizontal orientation, there are brief periods when the SNF canister experiences vertical orientation during loading and unloading operations from transport casks and/or waste packages. Such orientation with rubblized fuel presents a different set of neutronic interactions from that of a horizontal canister. Note that the vertical debris model increases the calculated reactivity by a factor of two over that of the horizontal case for the same fuel.

The following reactivities were calculated as a result of assuming minimum void fraction in each of the fissile matrix debris piles formed in the bottom of each canister compartment; see Table 7.3.2-1. The analysis retained the appropriate water content in the fuel matrix material (see Assumption 4.5). Whether for ATR (20) two-stack in a 10-ft canister or ATR (30) three-stack arrangement inside a 15-ft canister, the infinite cylinder length principle, i.e. reactivity of fissile content per unit length remains constant beyond a given length, is demonstrated with Figure 7.3.2-1.

Table 7.3.2-1. Reactivities for degraded fuel in a vertically oriented canister.

Fuel Type	Code Case (w/ 11 vol% water in SNF)	$k_{eff} + 2\sigma$
ATR (20)	homaV_0-w.o	0.1974
ATR (30)*	homaV30_0-w.o	0.1970
MURR (24)[(a)]	homV24_0-w.o	0.1412
MIT (30)**	mthomV_0-w.o	0.1361
ORR-HEU (30)***	ohomV_0-w.o	0.0740

* 15-ft long canister; others packaged in 10-ft canisters

** 19.75 vol% water in fuel matrix

*** ~30 vol% water in fuel matrix

(a) Six boundary FHUs removed from 30 FHUs; constitutes a misload.

Note that the cases for the ATR(20) [case: homaV_0-w.o] and MURR(24) [case: homV24_0-w.o] become the starting points for each of the two following void fraction calculations for those respective fuels.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

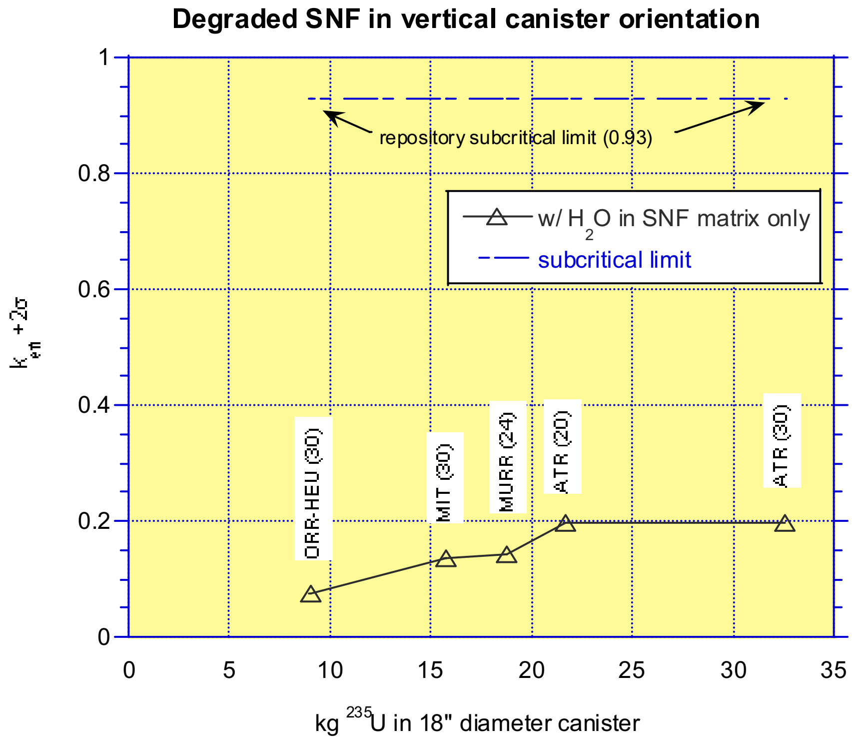

Figure 7.3.2-1. Reactivities of degraded fuel in vertically oriented 18" canister.

Assigning a void fraction of varying percentages to the fuel rubble determined the effect on calculated reactivity. Figure 7.3.2-2 depicts a side view of three stacked, vertically oriented basket compartments. All compartments with fuel were modeled with similar fuel rubblization, although the outer perimeter compartments result in slightly varying debris heights because of the canister radius influence on basket compartment area cross section.

The results shown in Table 7.3.2-2 represent the rubblization of the ATR fuel matrix material away from any cladding or side plates of the assemblies. All of fissile material from each fuel assembly is retained within its respective basket compartment in a vertical orientation. The fissile matrix also retains its 11 vol% water. The tabular results and Figure 7.3.2-3 demonstrate that as fissile atom densities decrease in a dry system with a fixed geometry, so too do the calculated reactivities.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

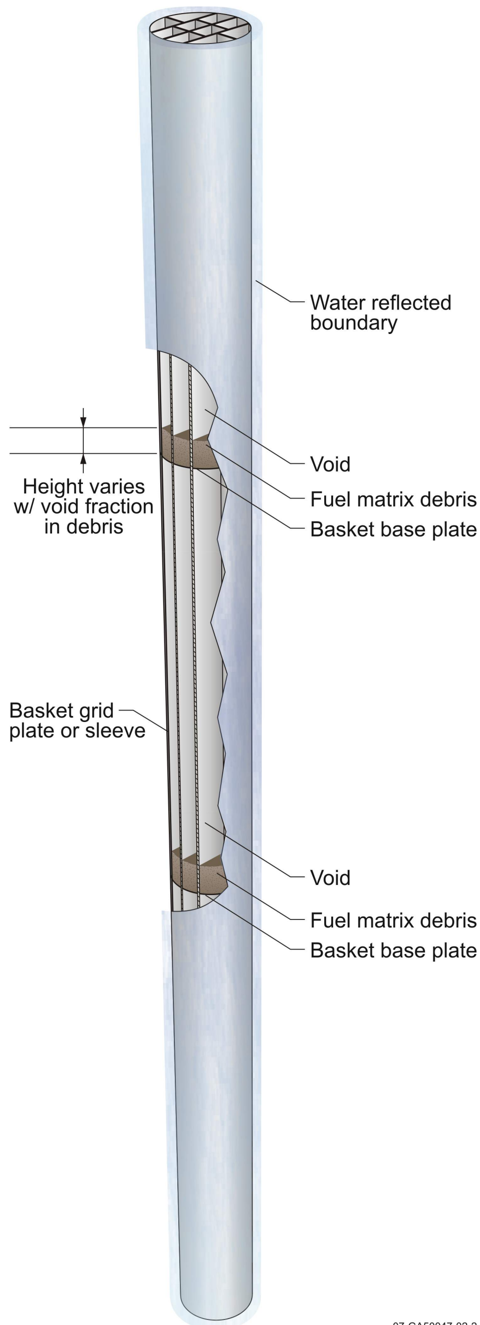

Figure 7.3.2-2 Side view cross-section of two stacked, vertically oriented Type 1a basket compartments.

Table 7.3.2-2. Results for degraded ATR (20) fuel inside vertically oriented basket compartments.

	ATR (20) Vertical Code Case	$k_{eff} + 2\sigma$
vf=0	homaV_0-w.o	0.1974
vf=0.2	homaV_20-w.o	0.1742
vf=0.4	homaV_40-w.o	0.1473
vf=0.5	homaV_50-w.o	0.1320
vf=0.6	homaV_60-w.o	0.1152
vf=0.75	homaV_75-w.o	0.0882
vf=0.817 (full)	homaV_fl-w.o	0.0811

*vf = void fraction (in any debris formation)

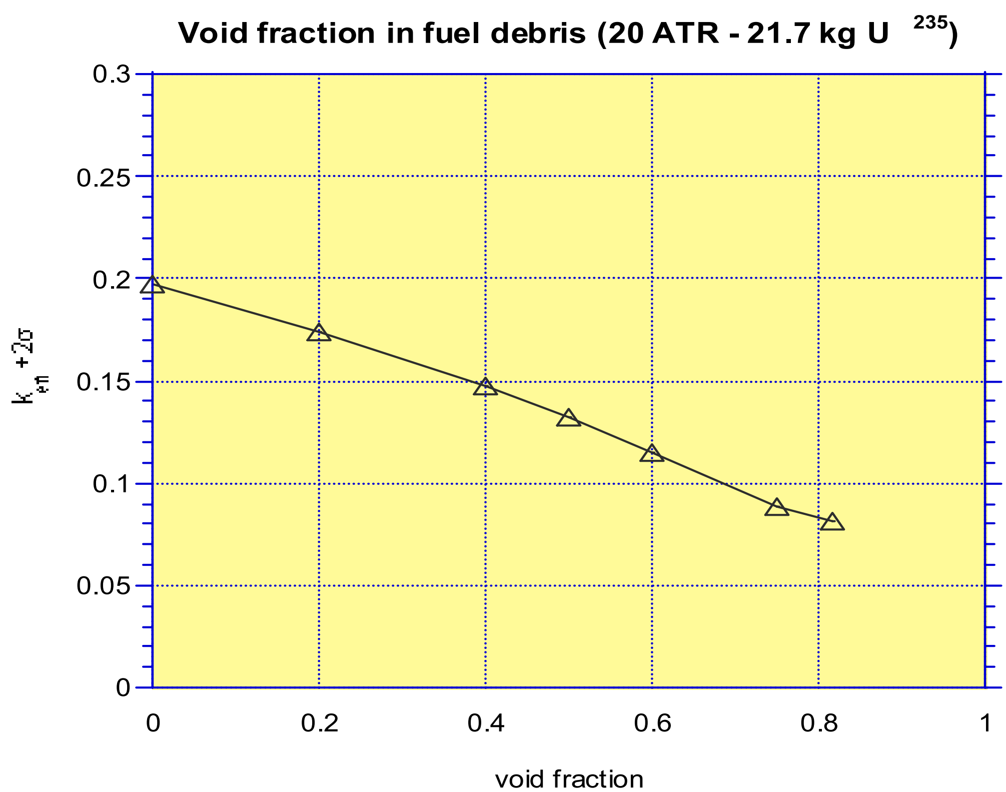

Figure 7.3.2-3. Reactivity as a function of void fraction in ATR fuel.

A similar calculation with the MURR (24) fuel packaging shows the same behavior as the ATR (20) fuel packaging, only with lower reactivities due to lower total fissile mass (18.79 vs. 21.7 kg ^{235}U). Table 7.3.2-3 and Figure 7.3.2-4 depict this trend.

Table 7.3.2-3. Results for degraded MURR (24) fuel inside vertically oriented canister compartments.

	MURR(24)[(a)] Vertical Code Case	k_{eff} +2σ
vf=0	homV24_0-w.o	0.1412
vf=0.2	homV24_20-w.o	0.1300
vf=0.4	homV24_40-w.o	0.1159
vf=0.6	homV24_60-w.o	0.0991
vf=0.7	homV24_75-w.o	0.0900
vf=0.78 (full)	homV24_fll-w.o	0.0866

*vf = void fraction (in any debris formation)
(a) Six boundary FHUs removed from 30 FHUs; constitutes a misload. The 'misload' was selected for this tabular presentation because it yielded a higher reactivity than the proposed center void configuration.

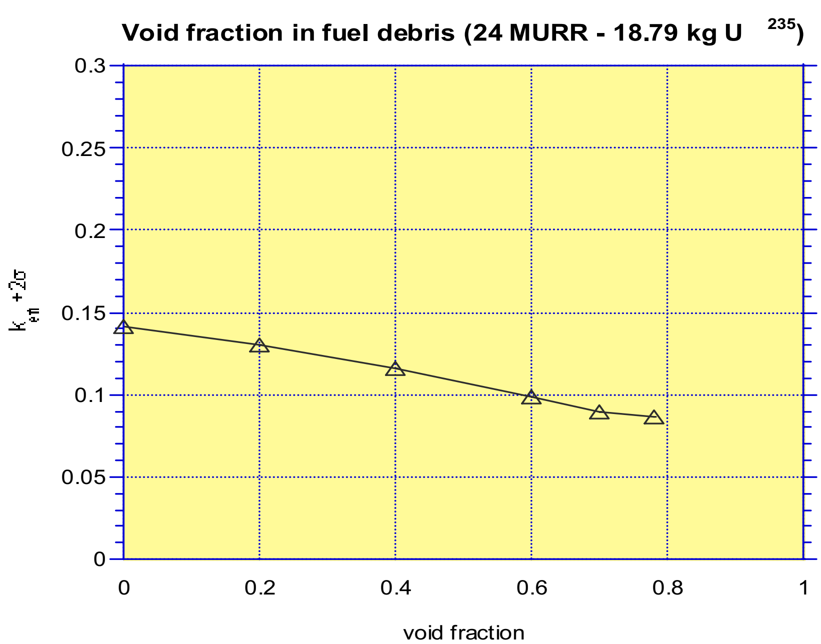

Figure 7.3.2-4. Reactivity as a function of void fraction in MURR fuel.

6.3.3 Basket Grid Plates Fail and Reconfigure Horizontally

The sensitivity of plate spacing was modeled by moving the horizontal plates closer together, but limited by the geometry of the fuel in the compartment. Figure 7.3.3-1a shows a conventional spacing of fuel assemblies within adjacent basket compartments. Figure 7.3.3-1b 'drops' the top horizontal plate such that it rests on the fuel assembly below. All fuel elements in each basket compartment were placed closer together (vertically) as a result of this reconfiguration. The drawings are not to scale; rather they depict relative positioning of basket plates and fuel shapes. The degraded plate/degraded fuel condition in the horizontal canister orientation was not analyzed because the distribution of the fuel debris over the length of the canister provides such a high surface to volume ratio for neutron leakage. A more reactive configuration than horizontal debris formation would be bounded by the vertical (or axial) configuration analyzed in Section 7.3.4.

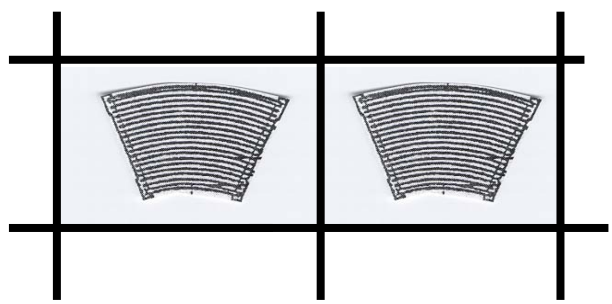

Figure 7.3.3-1a Conventional ATR spacing within Type 1a basket

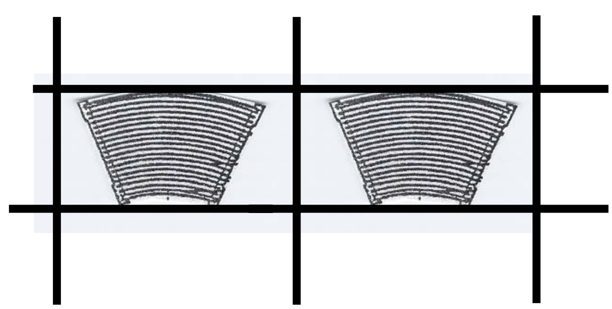

Figure 7.3.3-1b Collapsed basket plates constrained by fuel shape

To determine sensitivity to basket plate displacement, the analysis posed a canister side-drop where the plates would somehow move closer together until they were touching the ATR assemblies. All other parameters were retained (11 vol% water in ATR fuel matrix, water reflection of the canister, and no fuel or canister deformation). The results shown in Table 7.2-5 show a slight increase in reactivity (k_{eff} = 0.1407) versus the undamaged configuration for ATR (30) fuels in an intact basket of k_{eff} = 0.1367 (see Table 7.1-1).

Table 7.3.3-1. Result for ATR (30) fuel side drop.

Fuel Type	Code Case	k_{eff} +2σ
ATR (30)	atr30-side-dropr.o	0.1384

6.3.4 Basket Base Plates Fail and Fuel Debris Reconfigures Vertically

More reactive debris configuration than the horizontal orientation, this model analyzed the artificial positioning of the basket base plates at the bottom of the canister, and with the fuel matrix debris from all three baskets collected and consolidated as a single layer in the bottom of a vertically oriented SNF canister. This constitutes the axial reconfiguration scenario. This is strictly a non-mechanistic reconfiguration, as there are no accident scenarios that might allow the base plates to separate from their respective baskets while maintaining relative positioning of the basket side plates. This simplified model is depicted in Figure 7.3.4-1. The analysis modeled each of the compartments with the fuel matrix debris material from the above compartments in a three stack basket arrangement.

Although limited in nature because of the small amount of time any canister spends in a vertical orientation, axial reconfiguration of the fuel has always posed the greatest concern in terms of increased reactivity due to accident scenarios. Whether rubblization of the canister internals occurs with a vertical drop scenario, or a side drop and a vertical orientation upon recovery efforts, fissile material reconfiguration needed to be addressed for SNF canisters in transport. An axial reconfiguration scenario examined rubblization of the fuel in a three-stack ATR (30), with the fuel matrix material separating from between the aluminum cladding and collecting inside the bottom 10 basket compartments. Once again with the 11 vol% water in the fuel matrix and external water reflection, the most reactive condition is the consolidated debris with minimal void fraction. These results are shown in Table 7.3.4-1 and Figure 7.3.4-2.

Table 7.3.4-1. Result for ATR (30) fuel end-drop.

	ATR End Drop	k_{eff} +2σ
vf=0.0	homaE_0.o	0.4097
vf=0.20	homaE_20.o	0.3667
vf=0.50	homaE_50.o	0.2868

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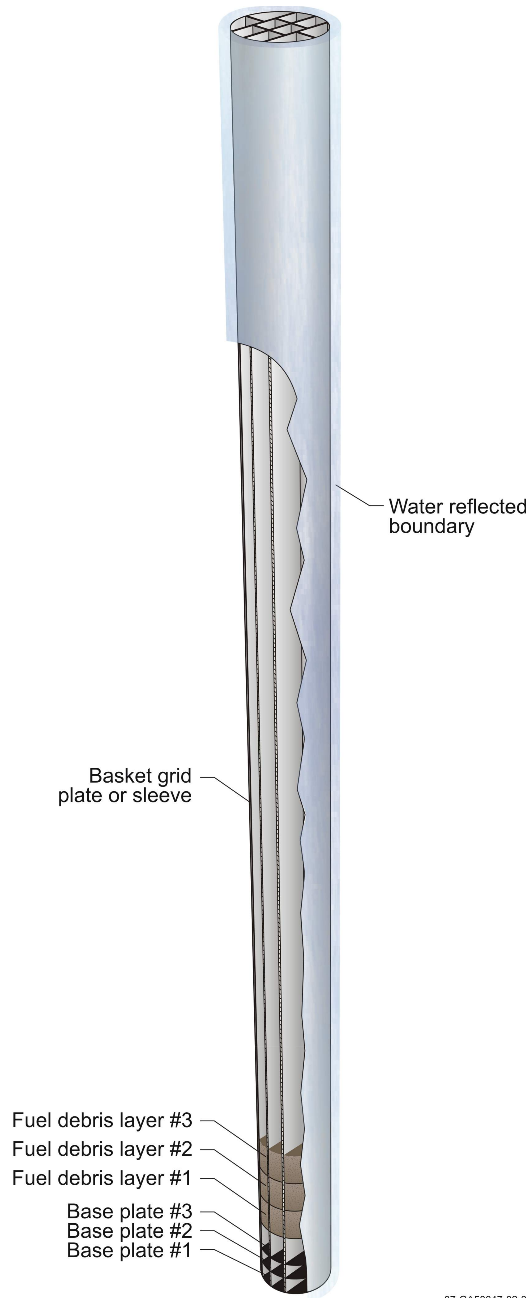

Figure 7.3.4-1 Collapsed basket plates and fuel debris constrained in basket compartments

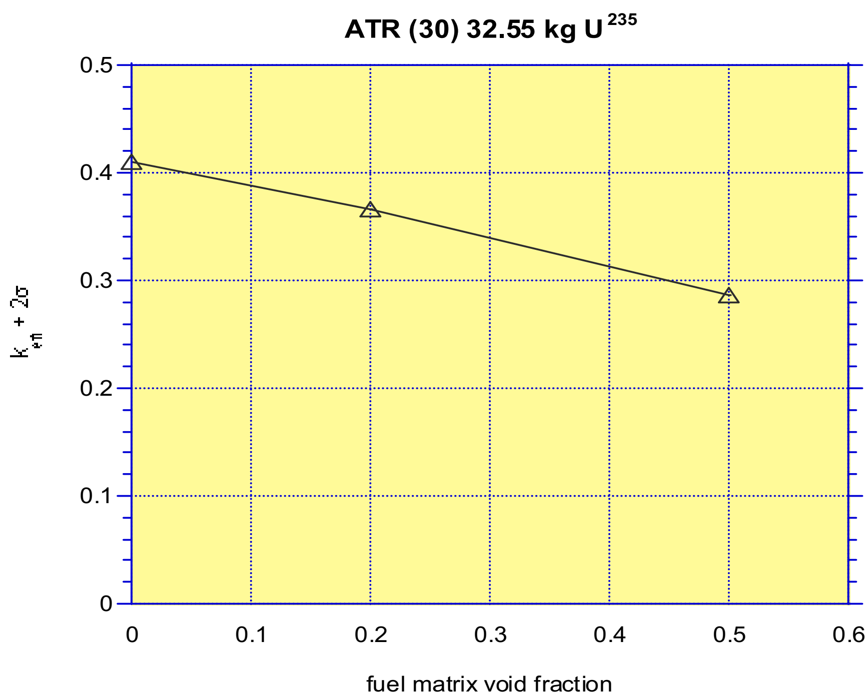

Figure 7.3.4-2. Reactivity in a vertically oriented canister with fuel degraded and reconfigured in one set of basket compartments.

6.3.5 Fuel non-mechanistically separates and assembles in a sphere below basket and cladding material

Figure 7.3.5-1 represents what was expected to be the most reactive configuration based on knowledge of the neutronics typically associated with bare spherical assembly of fissile material. A sphere represents the most optimal shaped and smallest surface/volume ratio for neutron economy.

The most extreme degradation case examines a non-mechanistic configuration. This involves axial reconfiguration of the fuel matrix material, with the mass of fuel matrix material in a single canister forming into a spherical geometry below and away from all poisoned basket plate material. The spherical shape is typically considered to be the most reactive geometry for any dry fissile system. The results for this scenario for both the ATR (30) and MURR (24) are provided in Table 7.3.5-1.

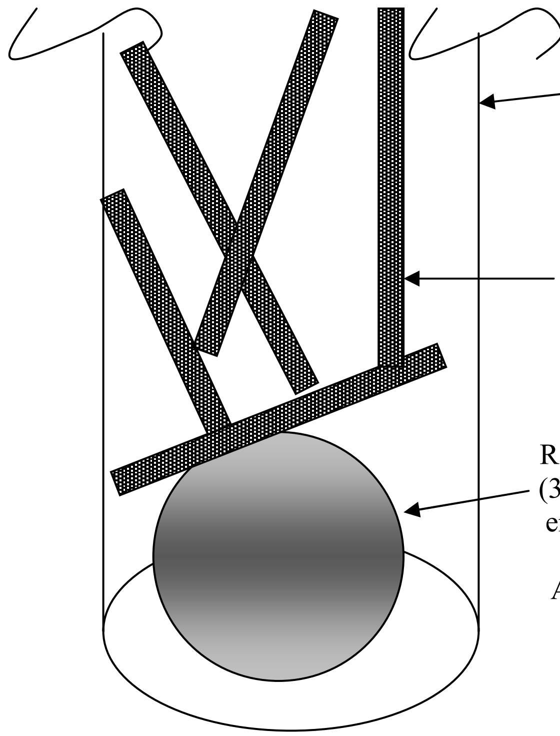

Figure 7.3.5-1. Spherical fuel debris without poison in sphere.

Table 7.3.5-1. Results of degraded spherical fuel matrix mass in SNF canister.

Fuel Type	Code Case	k_{eff} +2σ
ATR (30) (sphere)	sph_0.o	0.5479
MURR (24)	sph-mu2410.o	0.4160

6.3.6 Fuel non-mechanistically separates and assembles in a cylinder below basket and cladding material

An adjunct calculation examined cylindrical fuel debris formation in the bottom of a vertically oriented SNF canister. The cylindrical shape was limited by the internal diameter of the SNF canister, and the base plates and basket debris constituted reflective boundaries. This cylindrical model better represents the artificial reassembly of fuel matrix debris in the bottom of the canister. This scenario is depicted in Figure 7.3.6-1.

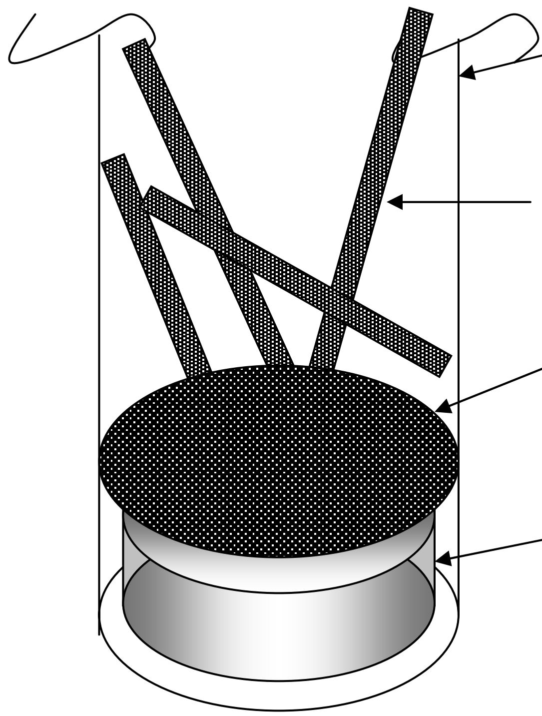

Figure 7.3.6-1. Cylindrical fuel debris without poison in cylinder.

The results shown in Table 7.3.6-1 for any of the aluminum plate fuels shows a slightly increased reactivity over that calculated for the spherical geometry in the preceding table. This increase is attributed to the reflective nature of the boundaries imposed by the canister wall, top and bottom reflection by the basket plates and bottom canister impact plate, respectively, and the external water reflection of the canister. This configuration of the ATR (30) canister was then used as the most reactive case for modeling in hypothetical transport cask analysis (see Section 7.4). No attempt was made to optimize the L/D ratio of the cylindrical shape at the bottom of the canister, since rubblizing the fuel to the extent necessary to migrate and reform the fuel debris in the bottom of the canister would fill all voids radially.

Table 7.3.6-1. Results of degraded cylindrical fuel matrix mass in SNF canister.

Fuel Type	Code Case	k_{eff} +2σ
ATR (30)	cyl_0+ca.o	0.6249
MURR (24)	cy_0-mu24.o	0.4625
MIT (30)	cy_0-mit.o	0.5026
ORR-MEU (30)	cy_0-orr.o	0.5440

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The calculated reactivity for the MIT and ORR fuels is attributed to the size difference in the cylindrical geometry because of the different matrix (oxide versus silicide versus aluminide) and the increased water volume % increase in the fuel matrix (see Assumption 4.5). The ATR (30) still represents the bounding case for maximum reactivity.

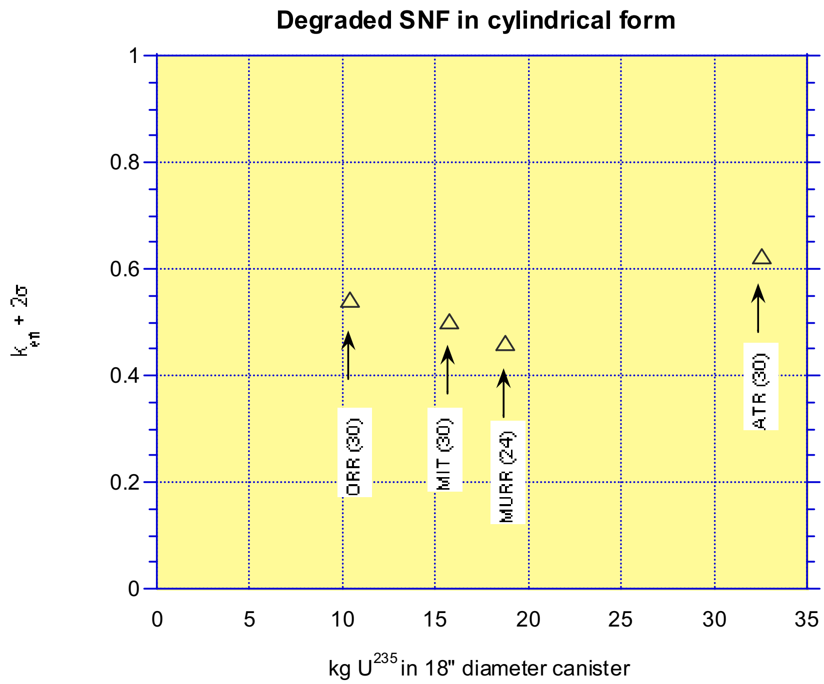

Figure 7.3.6-2. Degraded fuel accumulated in the bottom of a vertically oriented cask.

6.3.7 Fuel Debris in a Flooded Canister

As a confirmatory analysis for problems that might arise with breach and flooding of a loaded canister, the analyst examined non-mechanistically degraded ATR fuel assemblies still contained within their poisoned basket compartments, but with full flooding in the SNF canister. This model is a variant of Figure 7.3.2-2, only with water inserted into the void fraction within the fuel debris. The various water volume fractions (wvf) shown in the 4th column of Table 7.3.7-1 represent the differences in free volume within each of the three separate basket compartment volumes (created because of the way the basket compartment volumes are defined by the basket divider positions). Given the fully loaded 15' canister with ATR fuel, the typical subcritical limit of 0.95 could be exceeded, given maximal fuel rubblization <u>and</u> optimum moderation.

Table 7.3.7-1. Results of degraded and flooded ATR fuel matrix mass in SNF canister.

$k_{eff} \pm \sigma$	$k_{eff} + 2\sigma$	AENCF keV	Comment	File Name
0.9508 ± 0.0008	0.9523	10.0	wvf= 0.817, 0.792 and 0.775; fuel compartments are completely full; 3 baskets in a 15' SNF canister	homa_fl+L.o

6.4 Hypothetical Cask Configurations

While the future cask vendor will perform final criticality safety evaluations for the authorized cask configuration, it is of interest to know *a priori* the potential for being able to transport multiple SNF canisters while maintaining adequate margins of criticality safety. To this end, analyses were performed with a hypothetical transport cask/basket combination using both a nine-pack and seven-pack canister array.

Four cask configurations were evaluated. The first is based on a flooded cask with all of the canisters in their worst-case single canister reactivities ("basket base plates fail and reconfigure vertically" scenario). The next three are with the canisters in their "as-loaded" configuration but with three different scenarios relative to inleakage of water.

- canisters dry and the cask cavity flooded.
- cask and the canister cavities flooded.
- canisters flooded and the cask cavity dry. This case represents the differentially flooded scenario in which both cask and canisters were flooded, but in which the cask cavity drains at a more rapid rate than the canister cavities.

The 9-pack array was modeled as a centering pipe (20" XS pipe in 304L SS [0.500" wall thickness]) with eight canisters arrayed around the outside and separated by 0.500" thick plates; see Figure 7.4-2a. The fissile matrix inside the SNF canisters incorporates the vol% water appropriate to the fuel analyzed (see Assumption 4.5). Each case for the flooded cask configurations evaluated 0%, 10%, and 100% flooding outside the canisters in the array (to model the potential breach of the transport cask).

A similar criticality analysis was completed with a 7-pack array; see Figure 7.4-2b. This configuration was selected because of the concerns related to apparent dimensional constraints found in a 9-pack array, and also because a triangular pitch is known to produce a more reactive configuration. The basket model used the same 0.5-in. compartment thickness for each of the SNF canisters.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

All cases showed decreasing reactivities with both decreased fissile loads (as expected with decreased fissile atom densities) in their canisters, and with increased flooding inside the transport cask. These results are shown in Table 7.4-1 and Figure 7.4-1. Decreased reactivities with decreased fissile loads is an expected condition due to lower fissile masses, and the concurrent decrease in either fissile mass geometry of decreased fissile atom density as a function of the fissile matrix composition. Addition of water to the transport cask provides an effective neutronic isolation barrier. Thermalized neutrons are more subject to capture if reflected into the SNF canister with its Gd-poisoned basket, or ultimate capture or escape outside the canisters.

Table 7.4-1. Results of nine aluminum fueled canisters loaded in a cask.

Fuel Type	Water Volume Fraction Outside Canisters (%)	Code Case	k_{eff} +2σ
ATR (30)	0	m9cya_0.o	0.8656
	1	m9cya_1.o	0.8666
	2	m9cya_2.o	0.8625
	3	m9cya_3.o	0.8617
	4	m9cya_4.o	0.8580
	5	m9cya_5.o	0.8550
	6	m9cya_6.o	0.8523
	7	m9cya_7.o	0.8491
	8	m9cya_8.o	0.8450
	9	m9cya_9.o	0.8426
	10	m9cya_10.o	0.8386
	100	m9cya_100.o	0.7472
MURR (24)	0	m9cy_0.o	0.6568
	10	m9cy_10.o	0.6339
	100	m9cy_100.o	0.5721
MIT (30)	0	m9cymi_0.o	0.7013
	10	m9cymi_10.o	0.6893
	100	m9cymi_100.o	0.6261
ORR-MEU (30)	0	m9cyo_0.o	0.6960
	10	m9cyo_10.o	0.6729
	100	m9cyo_100.o	0.6281

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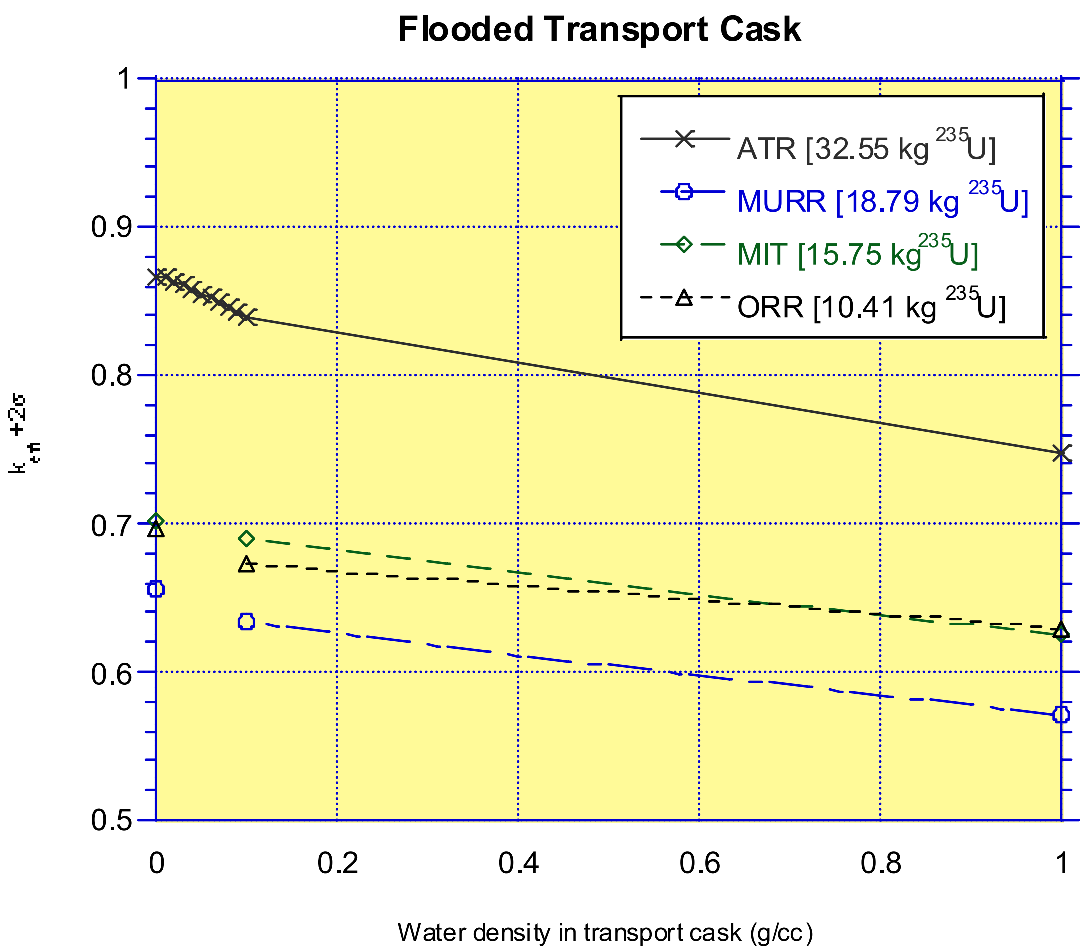

Figure 7.4-1. Flooded transport cask with degraded fuel in most reactive (cylindrical- vertical) SNF nine-pack canister.

Follow-up calculations addressed reactivities for a combination of flooded cask and/or canisters with intact internals. Using the ATR(30) model, configurations were examined in both nine-pack and seven-pack arrays as shown in Figures 7.4-2a and 7.4-2b respectively. The seven-pack array provides a better probability of being able to fit the canisters inside a transport cask. Result show that the hex array configuration modeled in the seven-pack proved just as reactive as the nine-pack array in spite of the 22% decrease in fissile load in the transport cask.

The intact analyses examined: (1) dry canisters / dry cask, (2) dry canisters / flooded cask, (3) flooded canisters / flooded cask, and (4) flooded canisters / dry cask [the differentially flooded case]. As displayed in Table 7.4-2, all conditions remained subcritical. Two special cases were analyzed without the benefit of Gd-poisoned basket materials inside the SNF canisters (300 series stainless steel substituted); while reactivity increased substantially for the loaded cask, the results remained below any expected subcritical limit. These specific results reinforce the concept that poisoning of the canisters is needed to support only the hypothetical degradation scenarios associated with post-closure conditions in the repository. Whether transportation requirements allow credit for only a portion of installed poisons or none at all in the SNF canister appears to be a moot point.

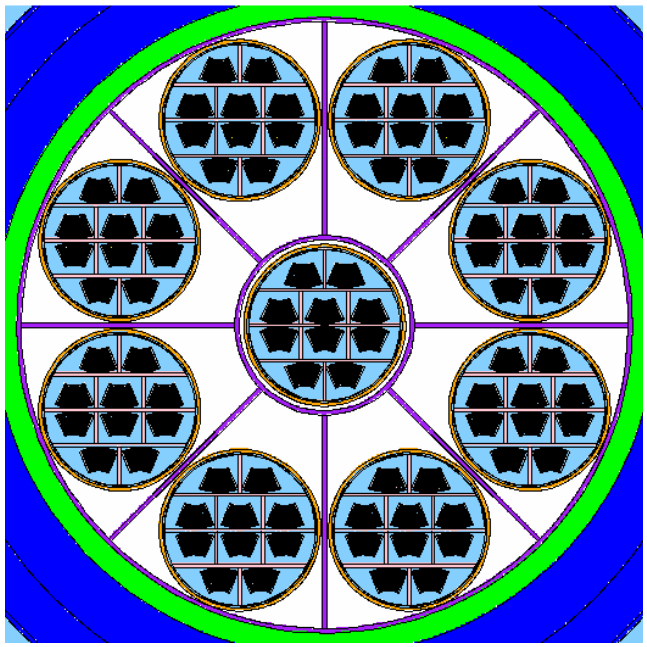

Figure 7.4-2a Nine-pack transport cask array

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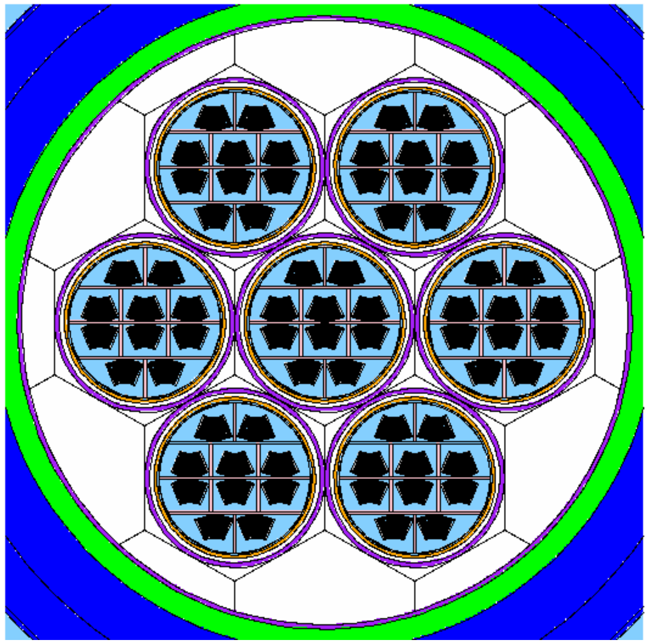

Figure 7.4-2b Seven-pack transport cask array

Table 7.4-2 Reactivities of ATR(30) loaded fuel canisters w/ intact internals inside a transport cask
9-pack canister array - fuel baskets (ANA w/ 2% Gd) and fuels intact

<-- Canisters -->

Cask -->	- dry		- flooded	
	($k_{eff} + 2\sigma$)	Code case	($k_{eff} + 2\sigma$)	Code case
-dry	0.2710	csk9int_dd.o	0.1876	csk9int_wd.o
-flooded	0.7076	csk9int_dw.o	0.6701	csk9int_ww.o
	0.8787*	csk9int_dwSS.o**	0.8364*	csk9int_wwSS.o

7-pack (hex) canister array- fuel baskets (ANA w/ 2% Gd) and fuels intact

-flooded	0.7036	csk7int_dwH.o	- - -	- - -
	0.8776*	csk7int_dwHSS.o**	- - -	- - -

* SS basket rather than ANA

** Differentially flooded 'worst case'

As shown in the dry canister/dry cask case, the reactivity of the loaded cask (whether a seven- or nine-pack) is relatively benign given any non-moderated neutrons in the package. Addition of water into the transport cask (but with dry canisters) drops the calculated reactivity from 0.2710 → 0.1876 due to the added neutronic isolation provided between canisters. Thermalization of neutrons under these circumstances makes their incidental capture more likely. Addition of water to the flooded canisters inside the flooded cask shows an appreciable but still acceptable increase in system reactivity (0.6701); the thermalization of neutrons also increases the effectiveness of the Gd poisoning present in the basket material. In the differentially flooded case (canisters flooded / cask dry), the reactivity increases (0.6701 → 0.7076) with the decrease in the neutronic isolation between canisters. Yet even without poisoning in the basket material as modeled with stainless steel in place of ANA material, the calculated reactivity remains subcritical (k_{eff}<0.90).

Dimensional constraints within a nine-pack loading in a transport cask suggested a smaller array might be required to assure an ability to actually load a transport cask. Such an array used the same wall compartment thickness between canisters (0.50" 300 series stainless steel). Even with removal of 22.2% of the fissile material in a transports cask, the resulting hex array proved as reactive as the nine-pack array in its most reactive (differentially flooded) case because of better neutronic coupling between the canisters. Yet this configuration can still remain subcritical, even with the removal of Gd poisoning from the canister baskets. If needed, the reactivity of the loaded cask can be further reduced by increased spacing between canisters or use of a poisoned cask insert.

6.5 Benchmark Evaluations

NSNFP has proposed use of the TSUNAMI computer code to identify applicable benchmark experiments to aid in determining an appropriate calculational bias. The TSUNAMI-3D sensitivity and uncertainty calculation sequence in SCALE 5 [SCALE 2005] was used to determine whether currently available benchmark experiments adequately cover the application cases.

TSUNAMI is a software code that allows for comparison of existing criticality benchmarks through a range-of-applicability analysis. The following is from a TSUNAMI training document [SCALE 2005, pg. 4]:

Integral Indices Assess Similarity

- Correlation coefficient, c_k, gives degree of shared variance in k_{eff} between design application and benchmark experiment. Requires cross-section covariance data.

Covariance between Experiment (e) and Application (a) due to all nuclides and reactions

$$c_k = \frac{\sigma_{ae}^2}{\sigma_a \sigma_e}$$

Standard deviations for Application (a) and Experiment (e) due to all nuclides and reactions

Acceptance Criteria for c_k

- Values of c_k relating a single experiment to a single application:
  - 1.0 systems are identical
  - 0.0 systems completely different
  - 0.9 - 1.0 systems are similar
  - 0.8 - 0.9 systems may be similar
  - (80-90% of uncertainty is common to both systems).

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

The criticality calculations for the spherical and cylindrical model used the MCNP 4b code. The use of TSUNAMI requires a KENO model when calculating both the application and benchmark cases, the two most reactive cases were selected and converted from MCNP into KENO input format. These cases are (1) a SNF canister model with spherical degraded ATR fuel with water in fuel (sph_0) and (2) a SNF canister model with cylindrical degraded ATR fuel with water in fuel (cyl_0+ca). To represent the dry system, water reflection outside the canister was removed. Table 7.5.1 shows a side-by-side comparison of the k_{eff}s for the MCNP and KENO codes for both the spherical and cylindrical models. The MCNP models used ENDF/B-V continuous cross sections with light water hydrogen scattering. The KENO model used 238 energy group ENDF/B-V cross sections.

Table 7.5-1. Comparison of MCNP and KENO models.

Model	MCNP	KENO V.a
Spherical Fuel Model	0.4189 ± 0.0005	0.4234 ± 0.0006
Cylindrical Fuel Model	0.4987 ± 0.0006	0.5081 ± 0.0003

The calculated difference between MCNP and KENO for the spherical model is:

$$k_{eff}\ \%\,Difference = \frac{0.4234 - 0.4189}{0.4189} \times 100 = 1.1\%$$

The calculated difference between MCNP and KENO for the cylindrical model is:

$$k_{eff}\ \%\,Difference = \frac{0.5078 - 0.4987}{0.4987} \times 100 = 1.8\%$$

The TSUNAMI-3D sequence calculates the forward and adjoint angular flux and then performs first order linear perturbation to determine cross-section sensitivity. Because the neutron spectrum is quite hard, the adjoint calculation requires significantly higher number of neutrons per generation (approximately 80,000) than in the forward case. This is necessary to achieve agreement between the forward k_{eff} and the adjoint k_{eff}. The model used the TSUNAMI default of S_{10} quadrature for calculating the angular flux. Also, it must be noted that the geometry models require subdivision using spherical or cylindrical shells. The geometry subdivision is needed to obtain adequate resolution of the forward and adjoint angular flux. Accurate forward and adjoint flux values are essential for the perturbation calculation, which is the key to TSUNAMI results.

Examination of the TSUNAMI-3D produced sensitivity data file associated with the spherical application model (see Figure 7.5-1) shows that the ^{235}U fission cross-section sensitivity dominates. However, the hydrogen scattering and aluminum scattering sensitivities are comparable to the ^{235}U fission sensitivity over much of the energy spectrum.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

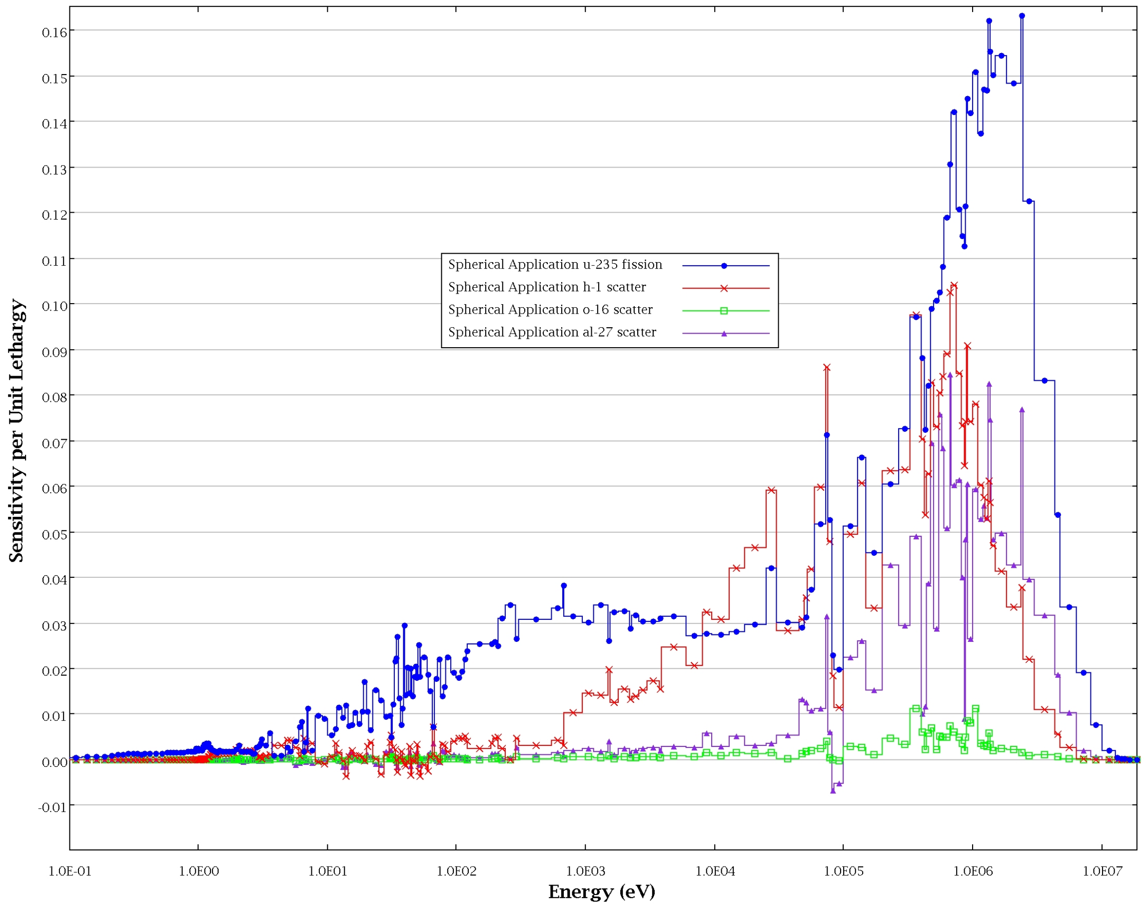

Figure 7.5-1. Scattering sensitivities.

6.5.1 Applicability of Benchmark Experiments

The analyses for the axial reconfiguration of ATR fuel matrix material away from the poisoned basket materials suggests benchmark experiments, such as GODIVA or other HEU fast systems, may be applicable. However, unlike GODIVA, the application cases are not devoid of hydrogen. For instance the spherical assembly model H/X ratio is 2.1. Furthermore, the aluminum content of the application cases is quite high. For instance, the Al/X ratio is 14.5 for the spherical assembly model. Thus, several other benchmark experiments were selected from the International Handbook of Evaluated Criticality Safety Benchmark Experiments (ICSBE) (ICSBE 2004).

The TSUNAMI-3D and the TSUNAMI-IP calculations were performed for the two SNF canister models to determine area of applicability. Because the canister models are fast systems, fast benchmark experiments from the ICSBE Handbook (ICSBE 2004) were examined. Initially, six evaluations from the Handbook were selected, and the TSUNAMI-3D and TSUNAMI-IP runs were made to calculate correlation coefficient, c_k. The models selected for evaluations are listed in Table 7.5-2.

Table 7.5-2. Initial comparison models.

Benchmark Experiment I.D.	Description
HEU-MET-FAST-001	Bare, HEU sphere experiment (GODIVA) performed at Los Alamos National Laboratory.
HEU-MET-FAST-007	U metal slabs moderated with polyethylene, Plexiglas®, and Teflon® experiments performed at Oak Ridge National Laboratory. Case 2 was evaluated.
HEU-MET-FAST-021	Steel reflected spherical assembly of U-235 (90%) experiment performed at VNIIEF.
HEU-MET-FAST-022	Duralumin reflected spherical assembly of U-235 (90%) experiment performed at VNIIEF.
HEU-MET-FAST-065	Unreflected cylinder of HEU experiment performed by VNIIEF.
HEU-MET-MIXED-005	Critical experiments with heterogeneous compositions of HEU, SiO_2, and polyethylene performed in the Institute of Physics and Power Engineering (IPPE) at the Big Physical Stand (BFS) facility. Five experiments were evaluated.

Fourteen cases were subsequently evaluated, as shown in Table 7.5-3. In some cases, the water in the fuel was removed to compare the c_k values; c_k ranged from 0.34 to 0.90. The most similar system was Case 5 of the BFS experiments (BFS-79-5 2000). In literally all cases, the MCNP calculated a higher reactivity when compared to the KENO model.

Table 7.5-3. c_k values and k_{eff}'s for application and benchmark experiments.

Benchmark Experiment	Application	MCNP 4b, $k_{eff} + 2\sigma$	KENO V.a, $k_{eff} + 2\sigma$	c_k
GODIVA (HEU-MET-FAST-001)	SNF canister model with cylindrical degraded ATR fuel with water in fuel	0.9986	0.9978	0.48
GODIVA (HEU-MET-FAST-001)	SNF canister model with cylindrical degraded ATR fuel without water in fuel	0.9986	0.9978	0.73
ORNL (HEU-MET-FAST-007)	SNF canister model with cylindrical degraded ATR fuel without water in fuel	0.9973	0.9944	0.67
VNIIEF (HEU-MET-021)	SNF canister model with cylindrical degraded ATR fuel with water in fuel	1.0082	1.0055	0.84
VNIIEF (HEU-MET-065)	SNF canister model with cylindrical degraded ATR fuel with water in fuel	0.9976	0.9968	0.85
BFS-79-1 (HEU-MET-MIXED-005)	SNF canister model with cylindrical degraded ATR fuel with water in fuel	1.0139	0.9994	0.79
BFS-79-2	SNF canister model with cylindrical	1.0239	1.0045	0.66

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

Benchmark Experiment	Application	MCNP 4b, $k_{eff} + 2\sigma$	KENO V.a, $k_{eff} + 2\sigma$	c_k
(HEU-MET-MIXED-005)	degraded ATR fuel with water in fuel			
BFS-79-3 (HEU-MET-MIXED-005)	SNF canister model with cylindrical degraded ATR fuel with water in fuel	1.0199	1.0008	0.34
BFS-79-4 (HEU-MET-MIXED-005)	SNF canister model with cylindrical degraded ATR fuel with water in fuel	1.0170	1.0067	0.85
BFS-79-5 (HEU-MET-MIXED-005)	SNF canister model with cylindrical degraded ATR fuel with water in fuel	1.0074	0.9953	0.91
GODIVA (HEU-MET-FAST-001)	SNF canister model with spherical degraded ATR fuel with water in fuel	0.9986	0.9978	0.62
VNIIEF (HEU-MET-065)	SNF canister model with spherical degraded ATR fuel with water in fuel	0.9976	0.9968	0.61
VNIIEF (HEU-MET-FAST-022)	SNF canister model with spherical degraded ATR fuel with water in fuel	0.9935	0.9934	0.64
BFS-79-5 (HEU-MET-MIXED-005)	SNF canister model with spherical degraded ATR fuel with water in fuel	1.0074	0.9953	0.86

Figure 7.5-2 shows the ^{235}U fission cross-section sensitivity for the spherical model application case compared with the duralumin reflected U sphere (HEU-MET-FAST-022) and the heterogeneous Institute of Physics and Power Engineering (IPPE) experiment (HEU-MET-MIXED-005). The figure shows that the duralumin reflected sphere and IPPE experiments envelop the ^{235}U fission cross-section sensitivity for the spherical application case.

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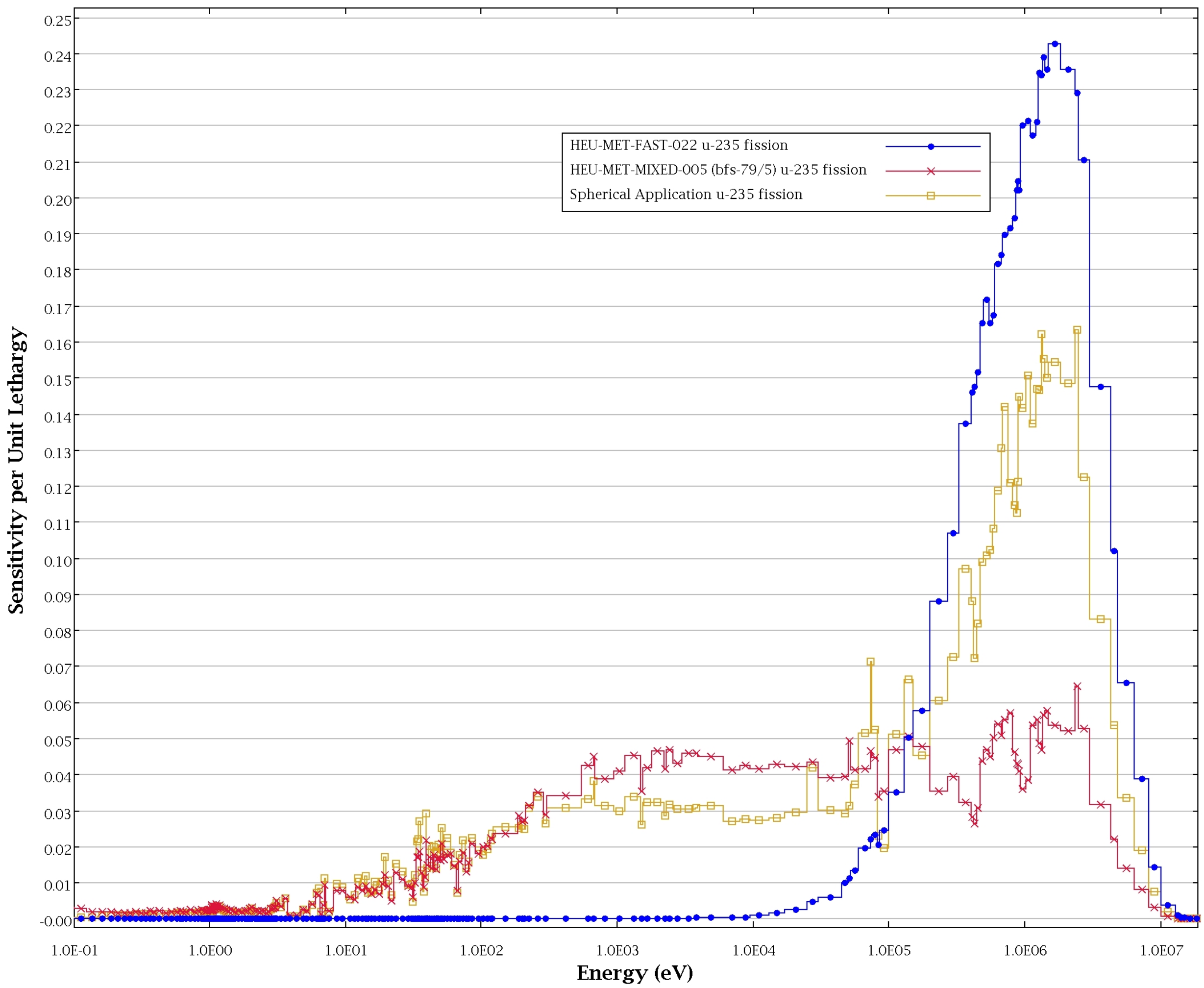

Figure 7.5-2. Cross-section sensitivity model comparisons.

Figure 7.5-3 shows the ^{27}Al scattering cross-section sensitivity for the spherical model application case compared with the duralumin reflected U sphere (HEU-MET-FAST-022) and the heterogeneous IPPE experiment (HEU-MET-MIXED-005). The figure shows that the IPPE experiment is relatively insensitive to the ^{27}Al scattering cross section. The duralumin reflected experiment shows comparatively significant sensitivity to the ^{27}Al scattering cross section, but it does not envelope the ^{27}Al scattering cross-section sensitivity for the spherical model application.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

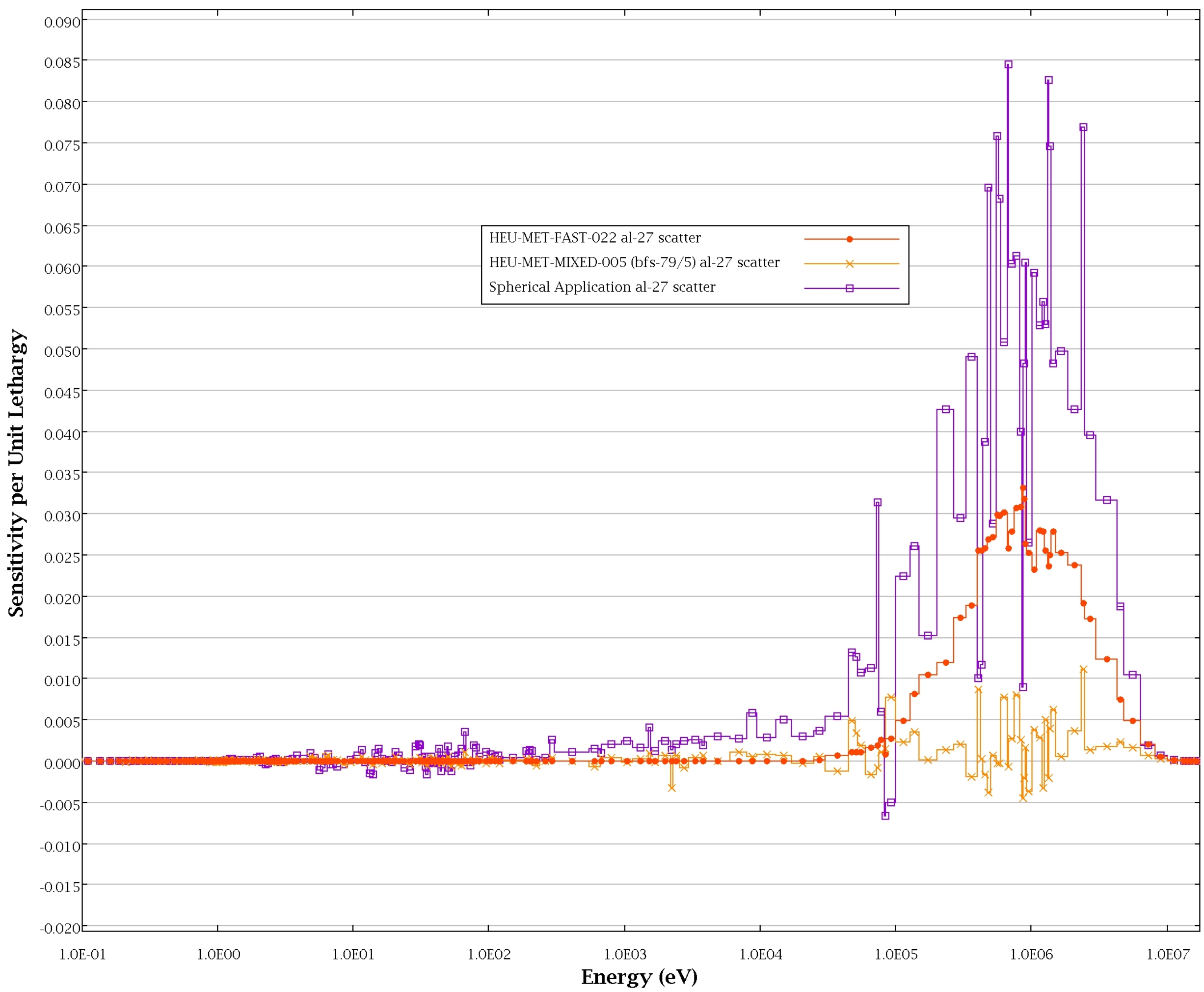

Figure 7.5-3. ^{27}Al scattering cross-section sensitivity.

Figure 7.5-4 shows the hydrogen scattering cross-section sensitivity for the spherical model application case compared with the heterogeneous IPPE experiment. The duralumin reflected experiment did not contain hydrogen. The figure shows that the IPPE experiment is relatively insensitive to the hydrogen scattering cross section and does not envelope the sensitivity for the spherical model application.

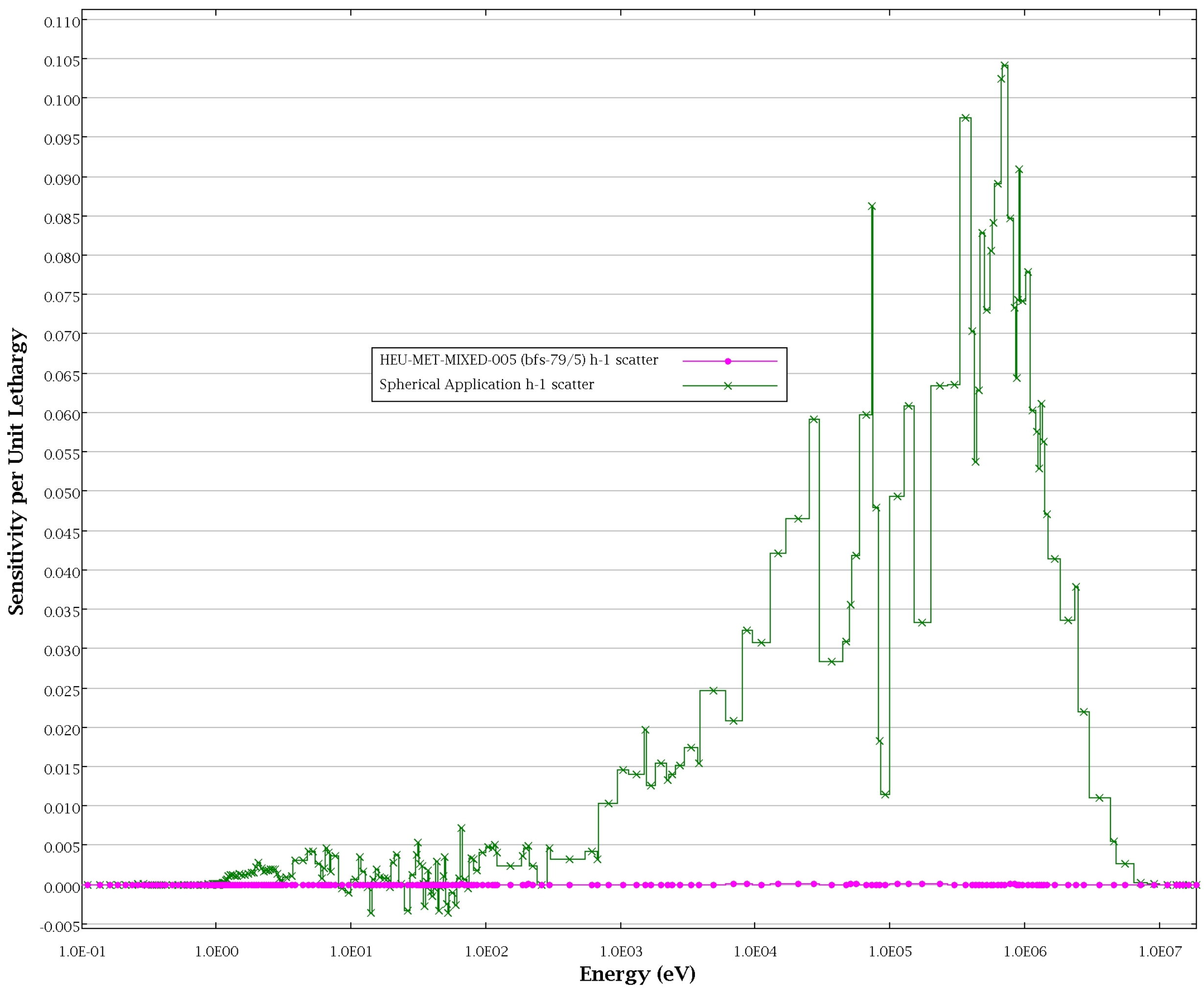

Figure 7.5-4. Hydrogen scatter sensitivity.

Directly out of the HEU-MET-MIXED-005 report, Section 4.0, Results of Sample Calculations states: “Many of the calculated results presented in Table 6 exceed the benchmark-model k_{eff} value by over 1%. The specific reasons for this are unknown.” So even this best specific benchmark has a degree of uncertainty associated with the reactivity measurements. The HEU-MET-MIXED-005 contains some Fe in the system, heterogeneous SiO_2 with Al-clad HEU pucks, and slightly higher moderation rather than homogeneous UAl_x models.

The spherical and cylindrical shapes of the debris modeled for the ATR fuels are represented with the UAl_x/Al fuel matrix from within the plates of the ATR assembly. As such, these models are represented by a homogenous reconfiguration of the matrix material. The HEU-MET-MIXED-005, which provides the highest c_k, is a heterogeneous, large-core reactor configuration. The canister model with degraded ATR fuel is basically a dry, fast system of rather small volume.

Based on the initial TSUNAMI calculation results, there appears to be insufficient availability of applicable benchmark critical experiments to quantitatively establish a calculation bias for the criticality scenarios under consideration. However, the TSUNAMI results confirm that applicable critical benchmark experiments are available for the dominant cross section of concern for the application cases, such as ^{235}U fission. The lack of identified applicable benchmark experiments for ^{27}Al and hydrogen suggests that a calculation bias should be applied. Furthermore, it appears that calculations for the application cases performed with MCNP may tend to be approximately 2% less than the calculation results obtained from KENO. Therefore, an added calculation bias of 5% is considered sufficient to ensure conservative results.

Based on the initial TSUNAMI calculations, no experiments in the ICSBE Handbook adequately represent the characteristics of this dry $^{235}U/Al$ system. Without meaningful comparative benchmarks, parameters such as pitch-to-rod diameter, assembly separation, and presence of neutron absorber material are inconsequential to establishing viable benchmarks for the dry $^{235}U/Al$ system. A more comprehensive TSUNAMI analysis may involve examination of over 100 benchmark experiments without yielding any other or more meaningful benchmarks given the relatively few 'dry' critical benchmarks.

The 5% bias and uncertainties typically assigned to established benchmarks cannot be supported with these analyses when dealing with this dry, critical system. However, given the conservatisms already incorporated in the $^{235}U/Al$ model, a negotiated administrative margin of perhaps 5% could be added to such a non-benchmarked system so that a subcritical limit with a $k_{eff} +2\sigma$ less than 0.90 is considered to provide a suitable margin of safety.

7. Summary

Four different aluminum plate fuel types were analyzed in a leak-tight standardized DOE SNF canister, with internal conditions ranging from intact to non-mechanistic worst case scenarios. Analyses were performed for single canisters as well as a nine-pack array using the most reactive single canister. The reactivity of the various dry-fuel configurations has been shown to be a direct function of the fissile mass per canister <u>and</u> the most consolidated fissile mass.

7.1.1 Intact Cases

A 15-ft. canister loading configuration containing three Type 1a baskets, each basket containing 10 ATR fuel assemblies, is considered the limiting case from a criticality safety perspective. This configuration will establish the canister loading limits in terms of total fissile (kg) and linear loading (g/cm) within the canister. As such, this configuration is expected to bound any other fuel type(s) that will be loaded into a canister using a Type 1a basket.

Analyses for intact fuels were completed for a single, dry as-loaded canister for four different aluminum plate fuels, such as ATR, MURR, ORR, and MIT utilizing a Type 1a basket. For comparison purposes, the same canister model was also analyzed without the presence of Gd in the basket plate material. In none of these cases did the calculated reactivity, as measured by $k_{eff} +2\sigma$ rise above 0.20. All analyses were conducted with water (see Assumption 4.5) in the fuel matrix and full water reflection of the canister. To further demonstrate the adequacy of the criticality safety for the single canister, the loaded canisters were also analyzed intact and fully-flooded. For the proposed canister loadings, the calculated reactivities remained below a $k_{eff} + 2\sigma \leq 0.6810$ when flooded.

Reactivity of the fuels has been shown to be a direct function of the fissile mass per canister. Accident conditions that might lead to axial reconfiguration of fissile material inside a vertically oriented canister are of primary concern for any canister handling prior to placement in the repository. In a moderator-excluded condition inside a loaded canister, invariably a more consolidated fissile mass produces the more reactive system.

7.1.2 Degraded Cases

Accident conditions that might lead to axial reconfiguration of fissile material inside a vertically oriented canister are of primary concern for transportation. Analyses for post-closure repository scenarios have already demonstrated criticality safety for horizontal configurations under fully degraded conditions. [CRWMS 2004] Analysis of canisters postulated to undergo various accident conditions during handling and transport operations considered vertical orientation with rubblization of the fuels. The accident scenarios were non-mechanistic, relying on postulated conditions to create most reactive configurations. Intermediate cases with baskets intact and fuels degraded, and basket deformation with fuels intact were also evaluated. Calculations employed a vertical canister orientation when in search of the most reactive configuration; in these vertical cases, all four fuel types were evaluated. As fuel matrix consolidation/concentration occurs, whether in the bottom of each basket or the bottom of the canister upon basket failure, subcritical limits are never exceeded under any proposed reconfiguration.

Results for degraded fuels with the vertical orientation also calculated the net effect of what void fraction in the rubble contributes or detracts from reactivity. Tables 7.2-2 and 7.2-3 (along with their accompanying figures) demonstrate that increased void fractions in the fuel rubble generate predictable decreases in reactivity due to the decreased fissile atom densities in the reactive zones in each basket. Consequently, configurations with all the fissile material concentrated into the bottom of a canister were evaluated to determine the bounding case.

Deflections of the various basket components were analyzed for increased reactivity for both side-drops and end-drops of a canister loaded with 30 ATR assemblies. In the case of the side drop, the assumption was the compartment plates somehow fractured and settled on the intact fuels in the compartment below. This resulted in a slight increase in reactivity from 0.1367 (see Table 7.1-1; case atr30.0) to 0.1407 (see Table 7.2-5). The more reactive increase occurred with an end-drop of the ATR (30), where all the fuel matrix material from 30 ATR assemblies reconfigured in the bottom basket; this configuration assumes loss of separation between baskets because of damage to the basket base plates. Once again, the vertical reconfiguration with minimum void fraction provided the maximum calculated reactivity for the enveloping fuel (see Table 7.2-6).

For extreme cases, all loss of basket and fuel geometry allowed for formation of a spherical shape in the bottom of the canister. The spherical shape was chosen for reasons known to promote minimum critical mass because it provides the smallest surface to volume ratio contributing to neutron leakage. Reactivity increased to approximately 0.55 for a 15-ft. canister loaded with ATR fuel (see Table 7.2-7; case sph_0.o). Analyses also examined a slightly more probable configuration with the fissile matrix debris forming a cylinder in the bottom of the canister, limited by the internal diameter of the SNF canister itself. Because this resulted in a more reactive configuration than the spherical shape (k_{eff} +2σ 0.6249), all four fuel types were analyzed in this configuration. The results for these single canister reactivities are shown in Table 7.2-8.

A confirmatory analysis examined non-mechanistically degraded ATR fuel assemblies still contained within their poisoned basket compartments, but with full flooding in the SNF canister. Given a fully loaded 15’ canister with ATR fuel, the typical subcritical limit of 0.95 would be exceeded, given maximal fuel rubblization and canister flooding regardless of canister orientation.

The last analyses also considered a hypothetical transport cask configuration to identify whether multiple canisters might be shipped safely. The nine-pack transportation arrays used the most reactive configuration as the input to the transport cask array calculations. The transport cask array values (see Table 7.3-1) evaluated the canisters with moderator exclusion, with 10% water density inside the transport cask and the transport cask fully flooded. Addition of water to a loaded transport cask appears to neutronically isolate the canisters. Although criticality calculations for the loaded transportation package will be done through the selected transportation cask vendor, these results indicate that with leak-tight canisters, criticality safety of the loaded cask can be demonstrated for all required transportation scenarios. Expansion of these transport cask analyses also included intact canister scenarios for all combinations of dry and flooded, both transport cask and SNF canisters. None of these calculated reactivities resulted in k_{eff} +2σ greater than 0.90.

7.1.3 Benchmarks

Benchmarks for dry systems, and more particularly systems with a variety of compositions for the fissile matrix material, are virtually nonexistent.

Greater biases and uncertainties are expected for transportation, but the ability to meet the subcritical limits for transportation can very likely be met given the criticality safety margins already calculated for the configurations analyzed in this report.

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Appendix A

MCNP Code – Input files

Appendix A

MCNP Code – Input files

input file name	date created	size bytes
	Directory Table7.2-1	
atr20	2/15/2007	31,094
atr20-nogd	2/15/2007	31,094
atr30	2/15/2007	31,231
atr30-nogd	2/15/2007	31,229
mit30	2/15/2007	21,011
mit30-nogd	2/15/2007	21,011
mu24c	2/15/2007	28,409
mu24c-nogd	2/15/2007	28,409
orr30	2/15/2007	23,712
orr30-nogd	2/15/2007	23,712
	Directory Table7.2-2	
d1cnba	2/15/2007	34,839
d1cnbb+L	2/15/2007	35,022
m1a24c	2/15/2007	31,911
mit1d	2/15/2007	24,586
o1bleu	2/15/2007	27,483
	Directory Table7.3.1-1	
homa24c-w	2/15/2007	20,014
homa30-w	2/15/2007	20,223
homa_0-w	2/15/2007	20,090
mthom_0-w	2/15/2007	20,076
ohom_0-w	2/15/2007	20,080
	Directory Table7.3.2-1	
homaV30_0-w	2/15/2007	20,422

input file name	date created	size bytes
	Directory Table7.3.3-1	
atr30-side-dropr	2/15/2007	30,827
	Directory Table7.3.4-1	
homaE_0	2/15/2007	16,007
homaE_20	2/15/2007	16,014
homaE_50	2/15/2007	16,014
	Directory Table7.3.5-1	
sph-mu2410	2/15/2007	6,573
sph_0	2/15/2007	6,632
	Directory Table7.3.6-1	
cyl_0+ca	2/15/2007	7,028
cy_0-mit	2/15/2007	6,951
cy_0-mu24	2/15/2007	6,951
cy_0-orr	2/15/2007	6,977
	Directory Table7.3.7-1	
homa_fl+L	2/15/2007	20,958
	Directory Table7.4-1	
m9cya_0	2/15/2007	8,427
m9cya_10	2/15/2007	8,048
m9cya_100	2/15/2007	8,038
m9cymi_0	2/15/2007	8,348

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

homaV_0-w	2/15/2007	20,084
homV24_0-w	2/15/2007	19,625
mthomV_0-w	2/15/2007	20,267
ohomV_0-w	2/15/2007	20,272

Directory Table7.3.2-2

homaV_0-w	2/15/2007	20,084
homaV_20-w	2/15/2007	20,024
homaV_40-w	2/15/2007	20,024
homaV_50-w	2/15/2007	20,017
homaV_60-w	2/15/2007	20,024
homaV_75-w	2/15/2007	20,024
homaV_fl-w	2/15/2007	20,507

Directory Table7.3.2-3

homV24_0-w	2/15/2007	19,625
homV24_20-w	2/15/2007	19,988
homV24_40-w	2/15/2007	19,988
homV24_60-w	2/15/2007	19,988
homV24_75-w	2/15/2007	19,988
homV24_fll-w	2/15/2007	20,314

m9cymi_10	2/15/2007	8,438
m9cymi_100	2/15/2007	8,429
m9cyo_0	2/15/2007	8,375
m9cyo_10	2/15/2007	8,474
m9cyo_100	2/15/2007	8,465
m9cy_0	2/15/2007	8,348
m9cy_10	2/15/2007	8,447
m9cy_100	2/15/2007	8,438

Directory Table7.4-2

csk7int_dwH	2/15/2007	29,160
csk7int_dwHSS	2/15/2007	29,160
csk9int_dd	2/15/2007	29,378
csk9int_dw	2/15/2007	29,378
csk9int_dwSS	2/15/2007	29,378
csk9int_wd	2/15/2007	29,380
csk9int_ww	2/15/2007	29,573
csk9int_wwSS	2/15/2007	29,572

Appendix B

Fuels Identified for Loading in Type 1a Baskets

Appendix B

Fuels Identified for Loading in Type 1a Baskets

The concept of the Type 1a basket developed out of planned loading of Advanced Test Reactor (ATR) fuels. The dimensions of the basket compartments lend themselves to many other fuels, not only within the aluminum fuel group, but also for several other fuels within the other eight fuel groups identified for criticality analysis for the repository.

The basket design is flexible in terms of being able to vary the basket lengths to deal with longer or shorter fuels, and then stacking the baskets to load a canister. The physical incorporation of gadolinium poisoning will be constant to the basket compartment plates regardless of the decreased fissile loads for these various fuels.

The selection of ATR as the baseline fuel for loading in the Type 1a basket bounds the total fissile in a canister (32.55 kg). This approach also provides a basis for linear loading (78.62 g/cm) in a canister which should not be exceeded by any of the other fuels identified for loading in a Type 1a basket. A necessary condition for reconfiguring any fuel from existing storage into a disposal canister requires a criticality safety evaluation (CSE) prior to loading; this will generate an associated load map. Each CSE must evaluate both dry and flooded conditions for the as-loaded fuel, and demonstrate an acceptable reactivity that is less than the established subcritical limit. Preferably, the calculated reactivity should be shown to be lower than the baseline (ATR) fuel in a comparable configuration. This would allow foregoing any other criticality evaluations where the other parameters (total fissile mass, enrichment, canister linear loading, physical weight, and thermal output) are also below baseline values.

There is a small number of fuel types identified for installation in Type 1a baskets where at least one of the aforementioned parameters is in excess of the ATR fuel. As an example, the CP5 Converter Cylinders have a fissile linear loading per fuel assembly that is 106.4% of baseline, and a fissile atom-density in the fuel element itself that is 499.3% of baseline. Inserting this fuel type into a canister results in a total fissile mass of 16.791 kg, or 51.6% of baseline for a fully loaded canister. Similarly, the homogenous distribution of the fissile mass throughout the canister produces a calculated fissile atom-density of only 83.3% of baseline fuel concentrations. However, there are only two pieces of this fuel for a total of less than 1.2 kg ^{235}U. Similarly, a fully loaded canister with GA RERTR fuel might have raised a concern @ 97.8% of ATR baseline atom-density in a canister were it not there is only one piece of that fuel with 381 grams of ^{235}U.

Showing the calculated, homogenous distribution of fissile atom-densities in a loaded canister allows a relative comparison against the loaded canister with ATR fuel under near equivalent conditions, equating to maximum reactivity if the canister were to flood.

The fuels identified for loading in a Type 1a basket were all listed in the Table B-1 information as though there is enough of any one reactor specific fuel to fill a canister. Such is not always the case. However, of the calculated fully-loaded canisters, none show a fissile atom-density in the loaded canister greater than 50% of ATR baseline values except for the three fuels (MURR, ORR, and MIT) selected for these analyses, and the two noted exceptions. This fact is important because the fissile mass for the loaded canisters shown in Table B-1 is based on the end-of-life values reported on the DOE 741 forms associated with all fuel transfers. Even if 50% burnup were arbitrarily assigned to any of these other fuels, a fully-loaded canister at double the fissile mass would still have a fissile atom-density within the loaded canister that is less than the baseline ATR values.

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

Table B-1 List of DOE fuels identified for disposal in a Type 1a basket configuration.

Fuel Category: UAlx	Fissile/FHU	% of baseline fuel	Fissile/ FHU length	% of baseline fuel	SNF can. Intrnl. dia.	SNF can. Intrnl. lngth	Basket design	basket(s) / canister	FHUs per canister (max)	fissile/canister	% of baseline fuel	Linear loading	% of baseline fuel	Fissile/atom-density		Fissile/atom-density	
Fuel Name [Fuel ID #]	(g)	(%)	(g/cm)	(%)	(cm)	(cm)	(type)	(#)	(#)	(kg)	(%)	(g/cm)	(%)	(atom/b-cm) [per FHU]	% of baseline fuel	(atom/b-cm) [per canister]	% of baseline fuel
Baseline Fuel																	
ATR (w/ poisoned basket)	1085.00	100%	8.62960	100.0%	43.82	414.02	Type 1a-3	3	30	32.550	100%	78.6194	100%	3.989E-04	100%	1.336E-04	100%
ANLJ [5]	135.969	12.5%	1.0569	12.2%	43.82	256.50	Type 1a-1	1	10	1.360	4.2%	5.3009	6.7%	5.938E-05	14.9%	9.008E-06	6.7%
ARMF (PLATES) [8]	12.130	1.1%	0.1873	2.2%	43.82	256.50	Type 1a-3	3	30	0.364	1.1%	1.4187	1.8%	3.885E-04	97.4%	2.411E-06	1.8%
ARMF/CFRMF MARK I [9]	186.036	17.1%	1.8841	21.8%	43.82	256.50	Type 1a-2	2	20	3.721	11.4%	14.5057	18.5%	7.041E-05	17.7%	2.465E-05	18.5%
ARMF/CFRMF MARK I LL [10]	110.000	10.1%	1.1140	12.9%	43.82	256.50	Type 1a-2	2	20	2.200	6.8%	8.5770	10.9%	4.159E-05	10.4%	1.457E-05	10.9%
ARMF/CFRMF MARK II [11]	135.500	12.5%	1.3723	15.9%	43.82	256.50	Type 1a-2	2	20	2.710	8.3%	10.5653	13.4%	5.123E-05	12.8%	1.795E-05	13.4%
ARMF/CFRMF MARK III [12]	22.000	2.0%	0.2228	2.6%	43.82	256.50	Type 1a-2	2	20	0.440	1.4%	1.7154	2.2%	8.317E-06	2.1%	2.915E-06	2.2%
ATR [15]	683.479	63.0%	5.4361	63.0%	43.82	414.02	Type 1a-2	3	30	20.504	63.0%	49.5251	63.0%	2.022E-04	50.7%	8.416E-05	63.0%
ATR [16]	750.530	69.2%	4.4601	51.7%	43.82	414.02	Type 1a-3	3	30	22.516	69.2%	54.3836	69.2%	1.628E-04	40.8%	9.241E-05	69.2%
ATSR [17]	149.557	13.8%	2.3187	26.9%	43.82	256.50	Type 1a-3	3	30	4.487	13.8%	17.4920	22.2%	9.520E-05	23.9%	2.972E-05	22.2%
BNL MEDICAL RX (BMRR) [21]	111.911	10.3%	1.7906	20.7%	43.82	256.50	Type 1a-3	3	30	3.357	10.3%	13.0890	16.6%	7.882E-05	19.8%	2.224E-05	16.6%
GTRR [87]	161.408	14.9%	2.3108	26.8%	43.82	256.50	Type 1a-3	3	30	4.842	14.9%	18.8781	24.0%	1.118E-04	28.0%	3.208E-05	24.0%
GENTR [97]	230.284	21.2%	4.5332	52.5%	43.82	256.50	Type 1a-4	4	40	9.211	28.3%	35.9118	45.7%	3.667E-04	91.9%	6.102E-05	45.7%
JMTR (JAPAN) [123]	217.072	20.0%	2.7134	31.4%	43.82	256.50	Type 1a-2	2	20	4.341	13.3%	16.9257	21.5%	1.171E-04	29.4%	2.876E-05	21.5%
MIT [135]	346.483	31.9%	5.1962	60.2%	43.82	256.50	Type 1a-3	3	30	10.394	31.9%	40.5244	51.5%	3.566E-04	89.4%	6.886E-05	51.5%
MIT [136]	308.382	28.4%	4.6248	53.6%	43.82	256.50	Type 1a-3	3	30	9.251	28.4%	36.0681	45.9%	3.174E-04	79.6%	6.129E-05	45.9%
MURR (COLUMBIA) [142]	593.895	54.7%	7.1944	83.4%	43.82	256.50	Type 1a-3	3	24	14.253	43.8%	55.5691	70.7%	2.601E-04	65.2%	9.443E-05	70.7%
MURR (COLUMBIA) [143]	595.016	54.8%	6.8393	79.3%	43.82	256.50	Type 1a-3	3	24	14.280	43.9%	55.6740	70.8%	2.684E-04	67.3%	9.460E-05	70.8%
UMRR (ROLLA) [146]	186.000	17.1%	2.1379	24.8%	43.82	256.50	Type 1a-2	2	20	3.720	11.4%	14.5029	18.4%	8.279E-05	20.8%	2.464E-05	18.4%
OHIO STATE [157]	132.483	12.2%	2.0655	23.9%	43.82	256.50	Type 1a-3	3	30	3.975	12.2%	15.4951	19.7%	9.114E-05	22.9%	2.633E-05	19.7%
OHIO STATE [158]	172.290	15.9%	1.9380	22.5%	43.82	256.50	Type 1a-2	2	20	3.446	10.6%	13.4339	17.1%	8.552E-05	21.4%	2.283E-05	17.1%
ORR [165]	260.789	24.0%	4.0558	47.0%	43.82	256.50	Type 1a-3	3	30	7.824	24.0%	30.5016	38.8%	1.694E-04	42.5%	5.183E-05	38.8%
PURDUE UNIVERSITY [177]	16.500	1.5%	0.2014	2.3%	43.82	256.50	Type 1a-2	2	20	0.330	1.0%	1.2865	1.6%	4.459E-04	111.8%	2.186E-06	1.6%
PURDUE UNIVERSITY [178]	215.910	19.9%	2.6356	30.5%	43.82	256.50	Type 1a-2	2	20	4.318	13.3%	16.8351	21.4%	1.194E-04	29.9%	2.861E-05	21.4%
RINSC [180]	109.955	10.1%	1.7593	20.4%	43.82	256.50	Type 1a-3	3	30	3.299	10.1%	12.8603	16.4%	7.763E-05	19.5%	2.185E-05	16.4%
RINSC [181]	259.611	23.9%	2.5552	29.6%	43.82	256.50	Type 1a-2	2	20	5.192	16.0%	20.2426	25.7%	1.128E-04	28.3%	3.440E-05	25.7%

UNIV OF FLORIDA (ARGONAUT) [272]	14.707	1.4%	0.2135	2.5%	43.82	256.50	Type 1a-3	3	30	0.441	1.4%	1.7201	2.2%	1.390E-05	3.5%	2.923E-06	2.2%
UNIV OF FLORIDA (ARGONAUT) [273]	14.109	1.3%	0.2048	2.4%	43.82	256.50	Type 1a-3	3	30	0.423	1.3%	1.6502	2.1%	1.334E-05	3.3%	2.804E-06	2.1%
UNIV OF MASS-LOWELL [274]	122.677	11.3%	1.9319	22.4%	43.82	256.50	Type 1a-3	3	30	3.680	11.3%	14.3481	18.3%	8.525E-05	21.4%	2.438E-05	18.3%
UNIV OF MASS-LOWELL [275]	68.870	6.3%	1.0846	12.6%	43.82	256.50	Type 1a-3	3	30	2.066	6.3%	8.0550	10.2%	4.786E-05	12.0%	1.369E-05	10.2%
UNIV OF MICHIGAN [276]	93.925	8.7%	1.0749	12.5%	43.82	256.50	Type 1a-2	2	20	1.879	5.8%	7.3236	9.3%	4.463E-05	11.2%	1.244E-05	9.3%
UNIV OF MICHIGAN [277]	122.159	11.3%	1.2698	14.7%	43.82	256.50	Type 1a-2	2	20	2.443	7.5%	9.5250	12.1%	5.273E-05	13.2%	1.619E-05	12.1%
UNIV OF VIRGINIA [279]	137.895	12.7%	1.5794	18.3%	43.82	256.50	Type 1a-2	2	20	2.758	8.5%	10.7520	13.7%	6.441E-05	16.1%	1.827E-05	13.7%
WORCESTER POLY INSTITUTE [287]	173.000	15.9%	2.7244	31.6%	43.82	256.50	Type 1a-3	3	30	5.190	15.9%	20.2339	25.7%	1.390E-04	34.8%	3.438E-05	25.7%
FRR MTR-C (JAPAN) [289]	85.059	7.8%	0.9666	11.2%	43.82	256.50	Type 1a-2	2	20	1.701	5.2%	6.6323	8.4%	4.273E-05	10.7%	1.127E-05	8.4%
FRR TUBES (DENMARK) [298]	77.400	7.1%	1.2384	14.4%	43.82	256.50	Type 1a-3	3	30	2.322	7.1%	9.0526	11.5%	3.808E-05	9.5%	1.538E-05	11.5%
FRR TUBES (AUSTRALIA) [299]	100.000	9.2%	1.6000	18.5%	43.82	256.50	Type 1a-3	3	30	3.000	9.2%	11.6959	14.9%	4.920E-05	12.3%	1.987E-05	14.9%
FRR TUBES (AUSTRALIA) [300]	54.049	5.0%	0.8648	10.0%	43.82	256.50	Type 1a-3	3	30	1.621	5.0%	6.3215	8.0%	2.659E-05	6.7%	1.074E-05	8.0%
GRR (GREECE) [440]	114.823	10.6%	1.4759	17.1%	43.82	256.50	Type 1a-3	3	30	3.445	10.6%	13.4296	17.1%	9.872E-05	24.8%	2.282E-05	17.1%
SAPHIR (SWITZERLAND) [443]	218.743	20.2%	2.4999	29.0%	43.82	256.50	Type 1a-2	2	20	4.375	13.4%	17.0560	21.7%	1.044E-04	26.2%	2.898E-05	21.7%
SAPHIR (SWITZERLAND) [444]	103.504	9.5%	1.1829	13.7%	43.82	256.50	Type 1a-2	2	20	2.070	6.4%	8.0705	10.3%	4.661E-05	11.7%	1.371E-05	10.3%
JRR-4 (JAPAN) [505]	131.348	12.1%	1.9781	22.9%	43.82	256.50	Type 1a-3	3	30	3.940	12.1%	15.3624	19.5%	6.599E-05	16.5%	2.610E-05	19.5%
FRR MTR-S (JAPAN) [506]	165.900	15.3%	1.8852	21.8%	43.82	256.50	Type 1a-2	2	20	3.318	10.2%	12.9357	16.5%	8.334E-05	20.9%	2.198E-05	16.5%
JMTR (JAPAN) [507]	287.000	26.5%	3.2614	37.8%	43.82	256.50	Type 1a-2	2	20	5.740	17.6%	22.3782	28.5%	1.442E-04	36.1%	3.803E-05	28.5%
FRR MTR-S (JAPAN) [508]	193.200	17.8%	2.1955	25.4%	43.82	256.50	Type 1a-2	2	20	3.864	11.9%	15.0643	19.2%	9.706E-05	24.3%	2.560E-05	19.2%
FRR MTR-C (NETHERLANDS) [509]	63.200	5.8%	0.7182	8.3%	43.82	256.50	Type 1a-2	2	20	1.264	3.9%	4.9279	6.3%	3.175E-05	8.0%	8.374E-06	6.3%
FRR MTR-S (NETHERLANDS) [510]	120.000	11.1%	1.3636	15.8%	43.82	256.50	Type 1a-2	2	20	2.400	7.4%	9.3567	11.9%	6.029E-05	15.1%	1.590E-05	11.9%
FRR MTR-C (CANADA) [512]	81.500	7.5%	0.9261	10.7%	43.82	256.50	Type 1a-2	2	20	1.630	5.0%	6.3548	8.1%	4.094E-05	10.3%	1.080E-05	8.1%
FRR MTR-S (CANADA) [513]	145.000	13.4%	1.6477	19.1%	43.82	256.50	Type 1a-2	2	20	2.900	8.9%	11.3060	14.4%	7.284E-05	18.3%	1.921E-05	14.4%
FRR ASTRA (AUSTRIA) [515]	122.500	11.3%	1.4032	16.3%	43.82	256.50	Type 1a-2	2	20	2.450	7.5%	9.5517	12.1%	5.869E-05	14.7%	1.623E-05	12.1%
FRR MTR-C (GERMANY) [517]	52.360	4.8%	0.5950	6.9%	43.82	256.50	Type 1a-2	2	20	1.047	3.2%	4.0827	5.2%	2.630E-05	6.6%	6.937E-06	5.2%
FRR MTR-S (GERMANY) [519]	70.840	6.5%	0.8050	9.3%	43.82	256.50	Type 1a-2	2	20	1.417	4.4%	5.5236	7.0%	3.559E-05	8.9%	9.386E-06	7.0%
FRR MTR-C (SWEDEN) [523]	149.606	13.8%	1.7001	19.7%	43.82	256.50	Type 1a-2	2	20	2.992	9.2%	11.6652	14.8%	7.516E-05	18.8%	1.982E-05	14.8%
FRR MTR-C2 (TURKEY) [527]	124.000	11.4%	1.4091	16.3%	43.82	256.50	Type 1a-2	2	20	2.480	7.6%	9.6686	12.3%	6.229E-05	15.6%	1.643E-05	12.3%
FRR MTR-S (TURKEY) [528]	168.000	15.5%	1.9091	22.1%	43.82	256.50	Type 1a-2	2	20	3.360	10.3%	13.0994	16.7%	8.440E-05	21.2%	2.226E-05	16.7%
FRR MTR-C (GREECE) [531]	79.950	7.4%	0.9085	10.5%	43.82	256.50	Type 1a-2	2	20	1.599	4.9%	6.2339	7.9%	4.017E-05	10.1%	1.059E-05	7.9%
FRR MTR-S (GREECE) [532]	144.300	13.3%	1.6398	19.0%	43.82	256.50	Type 1a-2	2	20	2.886	8.9%	11.2515	14.3%	7.249E-05	18.2%	1.912E-05	14.3%
FRR MTR-C (PORTUGAL) [540]	74.700	6.9%	0.7863	9.1%	43.82	256.50	Type 1a-2	2	20	1.494	4.6%	5.8246	7.4%	1.077E-04	27.0%	9.897E-06	7.4%
FRR MTR-O (PORTUGAL) [541]	80.000	7.4%	0.8421	9.8%	43.82	256.50	Type 1a-2	2	20	1.600	4.9%	6.2378	7.9%	1.230E-04	30.8%	1.060E-05	7.9%
FRR MTR-S (PORTUGAL) [542]	138.600	12.8%	1.4589	16.9%	43.82	256.50	Type 1a-2	2	20	2.772	8.5%	10.8070	13.7%	1.065E-04	26.7%	1.836E-05	13.7%
IEA-R1 (BRAZIL) [545]	131.078	12.1%	1.5015	17.4%	43.82	256.50	Type 1a-2	2	20	2.622	8.1%	10.2205	13.0%	6.099E-05	15.3%	1.737E-05	13.0%
FRR MTR (ARGENTINA) [547]	123.750	11.4%	1.3026	15.1%	43.82	256.50	Type 1a-2	2	20	2.475	7.6%	9.6491	12.3%	5.748E-05	14.4%	1.640E-05	12.3%
FRR MTR (JAPAN) [551]	128.982	11.9%	1.3577	15.7%	43.82	256.50	Type 1a-2	2	20	2.580	7.9%	10.0571	12.8%	1.332E-04	33.4%	1.709E-05	12.8%
FRR MTR-C (JAPAN) [552]	95.000	8.8%	1.0000	11.6%	43.82	256.50	Type 1a-2	2	20	1.900	5.8%	7.4074	9.4%	6.944E-05	17.4%	1.259E-05	9.4%
FRR MTR-S (JAPAN) [553]	147.636	13.6%	1.5541	18.0%	43.82	256.50	Type 1a-2	2	20	2.953	9.1%	11.5116	14.6%	6.828E-05	17.1%	1.956E-05	14.6%
ZPRL (TAIWAN) [554]	121.312	11.2%	1.4582	16.9%	43.82	256.50	Type 1a-2	2	20	2.426	7.5%	9.4590	12.0%	6.253E-05	15.7%	1.607E-05	12.0%
FRR MTR (TAIWAN) [555]	300.000	27.6%	3.1579	36.6%	43.82	256.50	Type 1a-2	2	20	6.000	18.4%	23.3918	29.8%	1.392E-04	34.9%	3.975E-05	29.8%

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

RU-1 (URAGUAY) [557]	104.480	9.6%	1.2796	14.8%	43.82	256.50	Type 1a-2	2	20	2.090	6.4%	8.1466	10.4%	5.828E-05	14.6%	1.384E-05	10.4%
PRR-1 (PHILLIPPINES) [558]	114.902	10.6%	1.1456	13.3%	43.82	256.50	Type 1a-2	2	20	2.298	7.1%	8.9592	11.4%	4.907E-05	12.3%	1.522E-05	11.4%
FRR MTR (VENEZUELA) [559]	101.688	9.4%	1.0704	12.4%	43.82	256.50	Type 1a-2	2	20	2.034	6.2%	7.9288	10.1%	1.046E-04	26.2%	1.347E-05	10.1%
FRR MTR (JAPAN) [565]	322.677	29.7%	3.3966	39.4%	43.82	256.50	Type 1a-2	2	20	6.454	19.8%	25.1600	32.0%	1.386E-04	34.7%	4.275E-05	32.0%
ASTRA (AUSTRIA) [566]	121.590	11.2%	1.7712	20.5%	43.82	256.50	Type 1a-3	3	30	3.648	11.2%	14.2211	18.1%	7.398E-05	18.5%	2.417E-05	18.1%
ENEA SALUGGIA (ITALY) [574]	127.725	11.8%	1.9500	22.6%	43.82	256.50	Type 1a-3	3	30	3.832	11.8%	14.9386	19.0%	8.165E-05	20.5%	2.538E-05	19.0%
FMRB (GERMANY) [577]	111.339	10.3%	1.1720	13.6%	43.82	256.50	Type 1a-2	2	20	2.227	6.8%	8.6814	11.0%	2.173E-04	54.5%	1.475E-05	11.0%
FRR MTR-C (GERMANY) [579]	55.460	5.1%	0.5838	6.8%	43.82	256.50	Type 1a-2	2	20	1.109	3.4%	4.3244	5.5%	2.576E-05	6.5%	7.348E-06	5.5%
FRR MTR-S (GERMANY) [582]	113.160	10.4%	1.1912	13.8%	43.82	256.50	Type 1a-2	2	20	2.263	7.0%	8.8234	11.2%	5.256E-05	13.2%	1.499E-05	11.2%
FRR MTR-S (GERMANY) [584]	122.120	11.3%	1.2855	14.9%	43.82	256.50	Type 1a-2	2	20	2.442	7.5%	9.5220	12.1%	5.672E-05	14.2%	1.618E-05	12.1%
FRR MTR-S (GERMANY) [585]	79.200	7.3%	0.8337	9.7%	43.82	256.50	Type 1a-2	2	20	1.584	4.9%	6.1754	7.9%	3.679E-05	9.2%	1.049E-05	7.9%
FRR MTR-S (GERMANY) [588]	116.480	10.7%	1.2261	14.2%	43.82	256.50	Type 1a-2	2	20	2.330	7.2%	9.0823	11.6%	5.410E-05	13.6%	1.543E-05	11.6%
IAN-R1 (COLUMBIA) [596]	136.776	12.6%	1.4397	16.7%	43.82	256.50	Type 1a-2	2	20	2.736	8.4%	10.6648	13.6%	2.103E-04	52.7%	1.812E-05	13.6%
FRR MTR-C (JAPAN) [600]	70.228	6.5%	0.7392	8.6%	43.82	256.50	Type 1a-2	2	20	1.405	4.3%	5.4759	7.0%	1.080E-04	27.1%	9.305E-06	7.0%
KURR (JAPAN) [601]	122.201	11.3%	1.5974	18.5%	43.82	256.50	Type 1a-3	3	30	3.666	11.3%	14.2925	18.2%	6.668E-05	16.7%	2.429E-05	18.2%
FRR MTR-S (JAPAN) [602]	136.800	12.6%	1.4400	16.7%	43.82	256.50	Type 1a-2	2	20	2.736	8.4%	10.6667	13.6%	1.054E-04	26.4%	1.813E-05	13.6%
FRR MTR (JAPAN) [605]	284.715	26.2%	2.9970	34.7%	43.82	256.50	Type 1a-2	2	20	5.694	17.5%	22.2000	28.2%	1.385E-04	34.7%	3.772E-05	28.2%
JRR-2 (JAPAN) [606]	130.379	12.0%	1.9635	22.8%	43.82	256.50	Type 1a-3	3	30	3.911	12.0%	15.2490	19.4%	8.664E-05	21.7%	2.591E-05	19.4%
FRR MTR-S (NETHERLANDS) [607]	50.000	4.6%	0.5263	6.1%	43.82	256.50	Type 1a-2	2	20	1.000	3.1%	3.8986	5.0%	6.954E-05	17.4%	6.625E-06	5.0%
FRR MTR-S (NETHERLANDS) [608]	95.000	8.8%	1.0000	11.6%	43.82	256.50	Type 1a-2	2	20	1.900	5.8%	7.4074	9.4%	6.933E-05	17.4%	1.259E-05	9.4%
FRR MTR (NETHERLANDS) [609]	211.788	19.5%	2.2293	25.8%	43.82	256.50	Type 1a-2	2	20	4.236	13.0%	16.5137	21.0%	1.381E-04	34.6%	2.806E-05	21.0%
FRR MTR-C (CANADA) [612]	68.200	6.3%	0.7179	8.3%	43.82	256.50	Type 1a-2	2	20	1.364	4.2%	5.3177	6.8%	3.168E-05	7.9%	9.036E-06	6.8%
MACMASTER (CANADA) [614]	100.211	9.2%	1.4802	17.2%	43.82	256.50	Type 1a-3	3	30	3.006	9.2%	11.7206	14.9%	6.055E-05	15.2%	1.992E-05	14.9%
FRR MTR (TAIWAN) [628]	126.720	11.7%	1.3339	15.5%	43.82	256.50	Type 1a-2	2	20	2.534	7.8%	9.8807	12.6%	1.386E-04	34.7%	1.679E-05	12.6%
THOR (TAIWAN) [629]	102.477	9.4%	1.0214	11.8%	43.82	256.50	Type 1a-2	2	20	2.050	6.3%	7.9904	10.2%	4.391E-05	11.0%	1.358E-05	10.2%
FRR MTR-C (PORTUGAL) [631]	88.200	8.1%	0.9284	10.8%	43.82	256.50	Type 1a-2	2	20	1.764	5.4%	6.8772	8.7%	8.136E-05	20.4%	1.169E-05	8.7%
FRR MTR-S (PORTUGAL) [632]	158.400	14.6%	1.6674	19.3%	43.82	256.50	Type 1a-2	2	20	3.168	9.7%	12.3509	15.7%	8.344E-05	20.9%	2.099E-05	15.7%
TRR-1 (THAILAND) [633]	133.754	12.3%	1.5374	17.8%	43.82	256.50	Type 1a-2	2	20	2.675	8.2%	10.4292	13.3%	6.471E-05	16.2%	1.772E-05	13.3%
RA-3 (ARGENTINA) [634]	118.399	10.9%	1.3454	15.6%	43.82	256.50	Type 1a-2	2	20	2.368	7.3%	9.2319	11.7%	5.386E-05	13.5%	1.569E-05	11.7%
FRR MTR-C (ARGENTINA) [635]	107.800	9.9%	1.1347	13.1%	43.82	256.50	Type 1a-2	2	20	2.156	6.6%	8.4055	10.7%	5.007E-05	12.6%	1.428E-05	10.7%
RA-3 (ARGENTINA) [636]	119.103	11.0%	1.3534	15.7%	43.82	256.50	Type 1a-2	2	20	2.382	7.3%	9.2868	11.8%	5.418E-05	13.6%	1.578E-05	11.8%
PRR-1 (PHILIPPIINES) [638]	144.467	13.3%	1.4341	16.6%	43.82	256.50	Type 1a-2	2	20	2.889	8.9%	11.2645	14.3%	6.170E-05	15.5%	1.914E-05	14.3%
FRR MTR-O (TURKEY) [642]	85.000	7.8%	0.8947	10.4%	43.82	256.50	Type 1a-2	2	20	1.700	5.2%	6.6277	8.4%	6.222E-05	15.6%	1.126E-05	8.4%
FRR MTR-C (TURKEY) [643]	103.500	9.5%	1.0895	12.6%	43.82	256.50	Type 1a-2	2	20	2.070	6.4%	8.0702	10.3%	6.630E-05	16.6%	1.371E-05	10.3%
FRR MTR-S (TURKEY) [644]	140.000	12.9%	1.4737	17.1%	43.82	256.50	Type 1a-2	2	20	2.800	8.6%	10.9162	13.9%	6.943E-05	17.4%	1.855E-05	13.9%
ASTRA (AUSTRIA) [646]	88.783	8.2%	1.2933	15.0%	43.82	256.50	Type 1a-3	3	30	2.663	8.2%	10.3840	13.2%	5.409E-05	13.6%	1.764E-05	13.2%
FRR MTR (AUSTRALIA) [649]	248.751	22.9%	2.8267	32.8%	43.82	256.50	Type 1a-2	2	20	4.975	15.3%	19.3958	24.7%	1.250E-04	31.3%	3.296E-05	24.7%
FRR ASTRA (AUSTRIA) [654]	50.000	4.6%	0.5263	6.1%	43.82	256.50	Type 1a-2	2	20	1.000	3.1%	3.8986	5.0%	2.322E-05	5.8%	6.625E-06	5.0%
FRR MTR-C1 (SWITZERLAND) [656]	61.280	5.6%	0.6451	7.5%	43.82	256.50	Type 1a-2	2	20	1.226	3.8%	4.7782	6.1%	4.779E-05	12.0%	8.119E-06	6.1%
FRR MTR-C2 (SWITZERLAND) [657]	74.880	6.9%	0.7882	9.1%	43.82	256.50	Type 1a-2	2	20	1.498	4.6%	5.8386	7.4%	4.796E-05	12.0%	9.921E-06	7.4%
FRR MTR-S (SWITZERLAND) [658]	87.420	8.1%	0.9202	10.7%	43.82	256.50	Type 1a-2	2	20	1.748	5.4%	6.8164	8.7%	3.788E-05	9.5%	1.158E-05	8.7%
FRR PIN CLUSTER (CANADA) [663]	207.100	19.1%	0.6768	7.8%	43.82	414.02	Type 1a-1	1	10	2.071	6.4%	5.0022	6.4%	1.844E-04	46.2%	8.500E-06	6.4%
FRR TUBES (GERMANY) [673]	90.000	8.3%	1.4400	16.7%	43.82	256.50	Type 1a-3	3	30	2.700	8.3%	10.5263	13.4%	4.434E-05	11.1%	1.789E-05	13.4%
FRR TUBES (GERMANY) [674]	100.000	9.2%	1.6000	18.5%	43.82	256.50	Type 1a-3	3	30	3.000	9.2%	11.6959	14.9%	4.927E-05	12.4%	1.987E-05	14.9%

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

FRR TUBES (GERMANY) [675]	112.500	10.4%	1.8000	20.9%	43.82	256.50	Type 1a-3	3	30	3.375	10.4%	13.1579	16.7%	5.543E-05	13.9%	2.236E-05	16.7%
FRR TUBES (DENMARK) [676]	58.310	5.4%	0.9330	10.8%	43.82	256.50	Type 1a-3	3	30	1.749	5.4%	6.8199	8.7%	2.869E-05	7.2%	1.159E-05	8.7%
FRR TUBES (DENMARK) [678]	73.500	6.8%	1.1760	13.6%	43.82	256.50	Type 1a-3	3	30	2.205	6.8%	8.5965	10.9%	3.616E-05	9.1%	1.461E-05	10.9%
HIFAR (AUSTRALIA) [680]	94.809	8.7%	1.4856	17.2%	43.82	256.50	Type 1a-3	3	30	2.844	8.7%	11.0888	14.1%	4.695E-05	11.8%	1.884E-05	14.1%
FRR TUBES (GERMANY) [683]	90.000	8.3%	1.4400	16.7%	43.82	256.50	Type 1a-3	3	30	2.700	8.3%	10.5263	13.4%	4.434E-05	11.1%	1.789E-05	13.4%
FRR TUBES (AUSTRALIA) [684]	79.900	7.4%	1.2784	14.8%	43.82	256.50	Type 1a-3	3	30	2.397	7.4%	9.3450	11.9%	3.931E-05	9.9%	1.588E-05	11.9%
FRR TUBES (GERMANY) [685]	102.000	9.4%	1.6320	18.9%	43.82	256.50	Type 1a-3	3	30	3.060	9.4%	11.9298	15.2%	5.026E-05	12.6%	2.027E-05	15.2%
RECH-1 (CHILE) [708]	84.081	7.7%	0.8467	9.8%	43.82	256.50	Type 1a-2	2	20	1.682	5.2%	6.5560	8.3%	2.851E-04	71.5%	1.114E-05	8.3%
ASTRA (AUSTRIA) [712]	195.312	18.0%	2.8450	33.0%	43.82	256.50	Type 1a-3	3	30	5.859	18.0%	22.8435	29.1%	1.190E-04	29.8%	3.882E-05	29.1%
HOR (NETHERLANDS) [713]	94.194	8.7%	1.0704	12.4%	43.82	256.50	Type 1a-2	2	20	1.884	5.8%	7.3446	9.3%	4.732E-05	11.9%	1.248E-05	9.3%
DR-3 (DENMARK) [714]	69.564	6.4%	1.1130	12.9%	43.82	256.50	Type 1a-3	3	30	2.087	6.4%	8.1361	10.3%	4.167E-05	10.4%	1.383E-05	10.3%
FRR MTR-S (CANADA) [720]	121.520	11.2%	1.2792	14.8%	43.82	256.50	Type 1a-2	2	20	2.430	7.5%	9.4752	12.1%	5.644E-05	14.2%	1.610E-05	12.1%
FRR ASTRA (AUSTRIA) [738]	313.000	28.8%	3.2947	38.2%	43.82	256.50	Type 1a-2	2	20	6.260	19.2%	24.4055	31.0%	1.454E-04	36.4%	4.147E-05	31.0%
FRG-1 (GERMANY) [741]	184.905	17.0%	1.9464	22.6%	43.82	256.50	Type 1a-2	2	20	3.698	11.4%	14.4175	18.3%	1.830E-04	45.9%	2.450E-05	18.3%
FRG-1 (GERMANY) [742]	96.281	8.9%	1.0135	11.7%	43.82	256.50	Type 1a-2	2	20	1.926	5.9%	7.5073	9.5%	9.531E-05	23.9%	1.276E-05	9.5%
JEN-1 (SPAIN) [749]	123.789	11.4%	1.2018	13.9%	43.82	256.50	Type 1a-2	2	20	2.476	7.6%	9.6522	12.3%	5.153E-05	12.9%	1.640E-05	12.3%
NEREIDE (FRANCE) [751]	152.483	14.1%	1.7467	20.2%	43.82	256.50	Type 1a-2	2	20	3.050	9.4%	11.8895	15.1%	7.413E-05	18.6%	2.020E-05	15.1%
BER-II [HMI] (GERMANY) [758]	82.971	7.6%	0.9219	10.7%	43.82	256.50	Type 1a-2	2	20	1.659	5.1%	6.4695	8.2%	3.837E-05	9.6%	1.099E-05	8.2%
DR-3 (DENMARK) [759]	86.969	8.0%	1.3915	16.1%	43.82	256.50	Type 1a-3	3	30	2.609	8.0%	10.1718	12.9%	5.209E-05	13.1%	1.728E-05	12.9%
ENEA SALUGGIA (ITALY) [760]	130.836	12.1%	1.9975	23.1%	43.82	256.50	Type 1a-3	3	30	3.925	12.1%	15.3024	19.5%	8.364E-05	21.0%	2.600E-05	19.5%
ESSOR (ITALY) [762]	390.048	35.9%	2.3639	27.4%	43.82	256.50	Type 1a-1	1	10	3.900	12.0%	15.2066	19.3%	1.093E-04	27.4%	2.584E-05	19.3%
IOWA ST. UNIV. [792]	147.000	13.5%	2.2511	26.1%	43.82	256.50	Type 1a-3	3	30	4.410	13.5%	17.1930	21.9%	1.087E-04	27.2%	2.922E-05	21.9%
JEN-1 (SPAIN) [795]	124.038	11.4%	1.2042	14.0%	43.82	256.50	Type 1a-2	2	20	2.481	7.6%	9.6715	12.3%	5.164E-05	12.9%	1.643E-05	12.3%
R-2 SVTR (SWEDEN) [801]	95.367	8.8%	1.4560	16.9%	43.82	256.50	Type 1a-3	3	30	2.861	8.8%	11.1541	14.2%	6.113E-04	153.3%	1.895E-05	14.2%
IAN-R1 (COLUMBIA) [803]	127.362	11.7%	1.3407	15.5%	43.82	256.50	Type 1a-2	2	20	2.547	7.8%	9.9308	12.6%	1.958E-04	49.1%	1.687E-05	12.6%
FRM (GERMANY) [805]	139.404	12.8%	1.5968	18.5%	43.82	256.50	Type 1a-2	2	20	2.788	8.6%	10.8697	13.8%	6.679E-05	16.7%	1.847E-05	13.8%
FRM (GERMANY) [806]	72.762	6.7%	0.7970	9.2%	43.82	256.50	Type 1a-2	2	20	1.455	4.5%	5.6735	7.2%	3.333E-05	8.4%	9.641E-06	7.2%
RV-1 (VENEZUELA) [816]	114.343	10.5%	1.1397	13.2%	43.82	256.50	Type 1a-2	2	20	2.287	7.0%	8.9156	11.3%	4.887E-05	12.3%	1.515E-05	11.3%
ATR [843]	621.313	57.3%	4.9416	57.3%	43.82	256.50	Type 1a-1	1	10	6.213	19.1%	24.2227	30.8%	1.838E-04	46.1%	4.116E-05	30.8%
ORR [850]	139.209	12.8%	2.1650	25.1%	43.82	256.50	Type 1a-3	3	30	4.176	12.8%	16.2818	20.7%	9.043E-05	22.7%	2.767E-05	20.7%
UMRR (ROLLA) [881]	153.610	14.2%	1.7656	20.5%	43.82	256.50	Type 1a-2	2	20	3.072	9.4%	11.9773	15.2%	6.837E-05	17.1%	2.035E-05	15.2%
JRR-2 (JAPAN) [885]	156.999	14.5%	2.3644	27.4%	43.82	256.50	Type 1a-3	3	30	4.710	14.5%	18.3624	23.4%	7.271E-05	18.2%	3.120E-05	23.4%
JMTR (JAPAN) [886]	216.942	20.0%	2.7118	31.4%	43.82	256.50	Type 1a-2	2	20	4.339	13.3%	16.9156	21.5%	1.171E-04	29.3%	2.874E-05	21.5%
FRJ (GERMANY) [933]	37.564	3.5%	0.5963	6.9%	43.82	256.50	Type 1a-3	3	30	1.127	3.5%	4.3935	5.6%	2.239E-05	5.6%	7.466E-06	5.6%
R-2 SVTR (SWEDEN) [942]	139.438	12.9%	2.1452	24.9%	43.82	256.50	Type 1a-3	3	30	4.183	12.9%	16.3085	20.7%	9.008E-04	225.8%	2.771E-05	20.7%
RPI (PORTUGAL) [943]	122.664	11.3%	1.4049	16.3%	43.82	256.50	Type 1a-2	2	20	2.453	7.5%	9.5645	12.2%	5.790E-05	14.5%	1.625E-05	12.2%
ORR [944]	238.255	22.0%	2.4461	28.3%	43.82	256.50	Type 1a-2	2	20	4.765	14.6%	18.5773	23.6%	9.817E-05	24.6%	3.157E-05	23.6%
SAPHIR (SWITZERLAND) [945]	121.738	11.2%	1.3913	16.1%	43.82	256.50	Type 1a-2	2	20	2.435	7.5%	9.4922	12.1%	5.808E-05	14.6%	1.613E-05	12.1%
UNIV OF VIRGINIA [952]	220.367	20.3%	2.5240	29.2%	43.82	256.50	Type 1a-2	2	20	4.407	13.5%	17.1826	21.9%	1.066E-04	26.7%	2.920E-05	21.9%
IOWA ST. UNIV. [953]	158.093	14.6%	2.3709	27.5%	43.82	256.50	Type 1a-3	3	30	4.743	14.6%	18.4904	23.5%	7.816E-05	19.6%	3.142E-05	23.5%
IEA-R1 (BRAZIL) [954]	102.425	9.4%	1.2266	14.2%	43.82	256.50	Type 1a-2	2	20	2.049	6.3%	7.9864	10.2%	4.982E-05	12.5%	1.357E-05	10.2%
MURR (COLUMBIA) [962]	593.895	54.7%	7.1944	83.4%	43.82	256.50	Type 1a-2	2	20	11.878	36.5%	46.3076	58.9%	2.601E-04	65.2%	7.869E-05	58.9%
FRJ TUBES (GERMANY) [999]	116.307	10.7%	1.8461	21.4%	43.82	256.50	Type 1a-3	3	30	3.489	10.7%	13.6031	17.3%	6.933E-05	17.4%	2.312E-05	17.3%
FRJ (GERMANY) [1000]	106.681	9.8%	1.6934	19.6%	43.82	256.50	Type 1a-3	3	30	3.200	9.8%	12.4773	15.9%	6.360E-05	15.9%	2.120E-05	15.9%

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

UNIV OF MICHIGAN (CONTROL) [1005]	61.681	5.7%	0.6412	7.4%	43.82	256.50	Type 1a-2	2	20	1.234	3.8%	4.8094	6.1%	2.662E-05	6.7%	8.172E-06	6.1%
JRR-3M (JAPAN) [1056]	212.061	19.5%	2.6508	30.7%	43.82	256.50	Type 1a-2	2	20	4.241	13.0%	16.5350	21.0%	1.170E-04	29.3%	2.810E-05	21.0%
DR-3 (DENMARK) [1059]	82.187	7.6%	1.3150	15.2%	43.82	256.50	Type 1a-3	3	30	2.466	7.6%	9.6125	12.2%	4.923E-05	12.3%	1.633E-05	12.2%
MNR (CANADA) [1064]	100.211	9.2%	1.4802	17.2%	43.82	256.50	Type 1a-3	3	30	3.006	9.2%	11.7206	14.9%	6.055E-05	15.2%	1.992E-05	14.9%
FRR FMRB (GERMANY) [1066]	111.339	10.3%	1.1720	13.6%	43.82	256.50	Type 1a-2	2	20	2.227	6.8%	8.6814	11.0%	2.173E-04	54.5%	1.475E-05	11.0%
FRR MTR-S (GERMANY) [1067]	168.000	15.5%	1.9091	22.1%	43.82	256.50	Type 1a-2	2	20	3.360	10.3%	13.0994	16.7%	8.440E-05	21.2%	2.226E-05	16.7%
FRR MTR-S (GERMANY) [1068]	74.536	6.9%	0.7846	9.1%	43.82	256.50	Type 1a-2	2	20	1.491	4.6%	5.8118	7.4%	3.462E-05	8.7%	9.876E-06	7.4%
GRR (GREECE) [1069]	114.823	10.6%	1.4759	17.1%	43.82	256.50	Type 1a-3	3	30	3.445	10.6%	13.4296	17.1%	9.872E-05	24.8%	2.282E-05	17.1%
JRR-4 (JAPAN) [1070]	131.348	12.1%	1.9781	22.9%	43.82	256.50	Type 1a-3	3	30	3.940	12.1%	15.3624	19.5%	6.599E-05	16.5%	2.610E-05	19.5%
JRR-4 (JAPAN) [1071]	150.086	13.8%	2.2603	26.2%	43.82	256.50	Type 1a-3	3	30	4.503	13.8%	17.5539	22.3%	7.540E-05	18.9%	2.983E-05	22.3%
RU-1 (URAGUAY) [1073]	104.480	9.6%	1.2796	14.8%	43.82	256.50	Type 1a-2	2	20	2.090	6.4%	8.1466	10.4%	5.828E-05	14.6%	1.384E-05	10.4%
IEA-R1 (BRAZIL) [1076]	131.078	12.1%	1.5015	17.4%	43.82	256.50	Type 1a-2	2	20	2.622	8.1%	10.2205	13.0%	6.099E-05	15.3%	1.737E-05	13.0%
U metal																	
EBWR ENRICHED HEAVY [64]	713.737	65.8%	4.8448	56.1%	43.82	256.50	Type 1a-1	1	10	7.137	21.9%	27.8260	35.4%	1.37E-04	34.3%	4.73E-05	35.4%
HWCTR RMT & SMT [790]	32.603	3.0%	0.1088	1.3%	43.82	414.02	Type 1a-1	1	10	0.326	1.0%	0.7875	1.0%	1.28E-05	3.2%	1.34E-06	1.0%
HWCTR TWNT [791]	143.923	13.3%	0.4885	5.7%	43.82	414.02	Type 1a-1	1	10	1.439	4.4%	3.4762	4.4%	1.48E-05	3.7%	5.91E-06	4.4%
HWCTR ETWO [867]	124.791	11.5%	0.4164	4.8%	43.82	414.02	Type 1a-1	1	10	1.248	3.8%	3.0141	3.8%	4.97E-05	12.4%	5.12E-06	3.8%
EBWR ENRICHED THIN [887]	531.147	49.0%	3.6054	41.8%	43.82	256.50	Type 1a-1	1	10	5.311	16.3%	20.7075	26.3%	1.02E-04	25.5%	3.52E-05	26.3%
EBWR NORMAL HEAVY [889]	361.153	33.3%	2.4515	28.4%	43.82	256.50	Type 1a-1	1	10	3.612	11.1%	14.0800	17.9%	6.92E-05	17.3%	2.39E-05	17.9%
EBWR NORMAL THIN [890]	286.309	26.4%	1.9434	22.5%	43.82	256.50	Type 1a-1	1	10	2.863	8.8%	11.1621	14.2%	5.48E-05	13.7%	1.90E-05	14.2%
UZr / UMo																	
CP-5 CONVERTER CYLINDERS [36]	559.695	51.6%	9.181	106.4%	43.82	256.5	Type 1a-3	3	30	16.791	51.6%	65.461	83.3%	1.992E-03	499.3%	1.112E-04	83.3%
HWCTR DRIVER [117]	403.568	37.2%	1.358	15.7%	43.82	414.02	Type 1a-1	1	10	4.036	12.4%	9.748	12.4%	1.299E-04	32.6%	1.656E-05	12.4%
HWCTR 3EMT-2 [118]	28.595	2.6%	0.248	2.9%	43.82	414.02	Type 1a-2	2	20	0.572	1.8%	1.381	1.8%	2.959E-06	0.7%	2.347E-06	1.8%
SPEC (ORME) [208]	123.000	11.3%	3.725	43.2%	43.82	256.5	Type 1a-4	4	40	4.920	15.1%	19.181	24.4%	2.466E-04	61.8%	3.259E-05	24.4%
UZrHx																	
GA RERTR [90]	381.44	35.2%	4.031	46.7%	43.82	256.5	Type 1a-2	2	20	7.63	23.4%	29.7373	37.8%	3.26E-04	81.7%	1.31E-04	97.8%
BER-II TRIGA (GERMANY) [236]	192.50	17.7%	2.059	23.9%	43.82	256.5	Type 1a-2	2	20	3.85	11.8%	15.0072	19.1%	8.68E-05	21.8%	6.59E-05	49.4%
HEU oxide																	
HFBR [102]	211.187	19.5%	3.394	39.3%	43.82	256.5	Type 1a-3	3	30	6.336	19.5%	24.700	31%	1.457E-04	36.5%	4.197E-05	31.4%
ORR [461]	152.765	14.1%	2.217	25.7%	43.82	256.5	Type 1a-3	3	30	4.583	14.1%	17.867	23%	9.260E-05	23.2%	3.036E-05	22.7%
HFBR [706]	216.786	20.0%	3.480	40.3%	43.82	256.5	Type 1a-3	3	30	6.504	20.0%	25.355	32%	1.767E-04	44.3%	4.309E-05	32.3%
ORR [753]	44.725	4.1%	0.696	8.1%	43.82	256.5	Type 1a-3	3	30	1.342	4.1%	5.231	7%	2.883E-05	7.2%	8.889E-06	6.7%
ORR [903]	176.838	16.3%	2.750	31.9%	43.82	256.5	Type 1a-3	3	30	5.305	16.3%	20.683	26.3%	1.151E-04	28.9%	3.515E-05	26.3%
GCRE (1Z SERIES) [916]	315.764	29.1%	3.965	46.0%	43.82	256.5	Type 1a-3	3	30	9.473	29.1%	36.931	47.0%	4.143E-04	103.9%	6.276E-05	47.0%

Title: Criticality Analysis for Proposed Maximum Fuel Loading in a Standardized SNF Canister with Type 1a Baskets

HFBR [961]	211.19	19.5%	3.394	39.3%	43.82	256.5	Type 1a-3	3	30	6.336	19.5%	24.700	31.4%	1.457E-04	36.5%	4.197E-05	31.4%
ASTRA (AUSTRIA) [1058]	144.05	13.3%	2.098	24.3%	43.82	256.5	Type 1a-3	3	30	4.322	13.3%	16.848	21.4%	8.776E-05	22.0%	2.863E-05	21.4%
U/Th oxide																	
ERR [68]	1059.71	97.7%	6.570	76.1%	43.82	256.5	Type 1a-1	1	10	10.597	32.6%	41.3141	52.5%	1.30E-03	326.7%	7.02E-06	5.3%
Th-U carbide																	
PEACH BOTTOM UNIT I CORE I [169]	253.00	23.3%	0.6917	8.0%	43.815	414.02	Type 1a-1	1	10	2.530	7.8%	6.1108	7.8%	2.786E-05	7.0%	1.045E-05	7.8%
PEACH BOTTOM UNIT I CORE I [170]	218.05	20.1%	0.5962	6.9%	43.815	414.02	Type 1a-1	1	10	2.180	6.7%	5.2666	6.7%	2.401E-05	6.0%	9.005E-06	6.7%
PEACH BOTTOM UNIT I CORE II [171]	117.14	10.8%	0.3660	4.2%	43.815	414.02	Type 1a-1	1	10	1.171	3.6%	2.8292	3.6%	1.511E-05	3.8%	4.837E-06	3.6%
PEACH BOTTOM UNIT I CORE II (INTACT) [206]	103.39	9.5%	0.3230	3.7%	43.815	414.02	Type 1a-1	1	10	1.034	3.2%	2.4971	3.2%	1.333E-05	3.3%	4.269E-06	3.2%
LEU oxide																	
HWCTR SPRO [115]	17.67	1.6%	0.3710	4.3%	43.815	256.5	Type 1a-4	4	40	0.706868	2.2%	2.756	3.5%	4.391E-05	11.0%	2.919E-06	2.2%
HWCTR SOT [120]	20.91	1.9%	0.6097	7.1%	43.815	256.5	Type 1a-4	4	40	0.836212	2.6%	3.260	4.1%	7.216E-05	18.1%	3.453E-06	2.6%

B-1 References

DOE 2004b, *Packaging Strategies for Criticality Safety for "Other" DOE Fuels in a Repository*. DOE/SNF/REP-090, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office, ACC: MOL.20040708.0386.