Environmental Effects

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Spent Nuclear Fuel Characterization for a Bounding Reference Assembly for the Receiving Basin for Off-Site Fuel

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Abstract*: The Basis for Interim Operation (BIO) for the Receiving Basin for Off-Site Fuel (RBOF) facility at the Department of Energy (DOE) Savannah River Site (SRS) nuclear materials production complex, developed in accordance with draft DOE-STD-0019-93, required a hazard categorization for the safety analysis section as outlined in DOE-STD-1027-92. The RBOF facility was thus established as a Category-2 facility (having potential for significant on-site consequences from a radiological release) as defined in DOE 5480.23. Given the wide diversity of spent nuclear fuel stored in the RBOF facility, which made a detailed assessment of the total nuclear inventory virtually impossible, the categorization required a conservative calculation based on the concept of a hypothetical, bounding reference fuel assembly integrated over the total capacity of the facility. This scheme not only was simple but also precluded a potential delay in the completion of the BIO.*

The Receiving Basin for Off-Site Fuel (RBOF) at the Department of Energy (DOE) Savannah River Site (SRS) nuclear materials production complex near Aiken, South Carolina, is a facility designed for the receipt, storage, and conditioning of spent nuclear fuel (SNF) from off-site reactors.[1] The facility has been operational since 1963 and is now managed by the Excess Facilities and Reactor Fuel Storage Program Division of the Westinghouse Savannah River Company (WSRC), the prime operations contractor for SRS.

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OVERVIEW OF THE RECEIVING BASIN FOR OFF-SITE FUEL

The RBOF features a dry cask-wash pit, a cask unloading basin, two fuel storage basins, a fuel inspection basin, a fuel disassembly basin, and a fuel repackaging basin. All the basins are filled with water to variable depths and interconnected by canals. The water in the basins is continuously purified by circulation through a filter–deionizer system.

The main section (22 ft deep) of the large storage basin (27 ft wide by 40 ft long overall) and the small storage basin (13 ft wide by 27 ft long by 29 ft deep) are reserved for intact fuel bundles, which are arranged in rows defined by a vertical framework of racks made of aluminum I-beams rising from the bottom of the basins. Gratings, guide plates, and spacers (collectively referred to as “hardware”) are installed between the racks to define the individual storage slots along the rows. Up to four tiers of hardware per row can be installed, depending on the fuel type. The large basin contains 42 rows, each 18 ft long, and the small basin contains 11 rows, each 9 ft long. Twenty-one rows in the large basin and all the rows in the small basin are 9 inches wide; 18 other rows in the large basin are spaced at 12 inches. The remaining 3 rows in the large basin have spacings of 11, 16, and 25.5 inches, respectively.

Fuel assemblies in RBOF are largely bundled in locally fabricated, elongated, aluminum cans (tubes) with either a circular or a squared cross section. The

cylindrical cans are commonly referred to as general-purpose (GP) tubes. Assemblies of a specific fuel type are packed in cans as constrained by rigorous nuclear criticality safety criteria. Similar criteria dictate the distribution of up-standing cans throughout the storage basins.

The large basin also has special storage racks known as bucket storage and test tube storage. The first section provides space for 70 buckets in 5 rows; the "buckets" are special containers designed to hold reactor slugs. The second section has a capacity of 13 "test tubes," special containers for damaged (confirmed or suspected) fuel elements. At 29 ft, these sections are somewhat deeper than the main section.

RBOF AUTHORIZATION BASIS UPGRADE

The RBOF Authorization Basis is undergoing extensive revision to conform to current DOE Orders and Standards. The initial task, development of the Basis for Interim Operation (BIO), was originally completed in late 1994 and revised once in early 1995.[2] The BIO is based on DOE-STD-0019-93,[3] although this Standard was later superseded by draft DOE-STD-3011-94.[4] Pending the release of the Safety Analysis Report (SAR), tentatively scheduled for mid-1996, the BIO authorizes operation of the facility within an acceptable safety envelope. The safety analysis portion of the BIO required a hazard categorization as outlined in DOE-STD-1027-92.[5] A chemical–nuclear facility falls into one of three classes defined in DOE 5480.23[6] according to the potential consequences of radiological releases.

THE FUEL DIVERSITY DILEMMA

The RBOF is now loaded to about 85% capacity with a wide variety of SNF, mostly domestic research reactor (DRR) fuel received over the past 30 years. In the foreseeable future, the facility likely will reach full capacity with the increasing influx of foreign research reactor (FRR) fuel from various countries. The SNF inventory (current and expected) is traceable to a wide variety of original designs (chemical–nuclear composition and physical configuration) with an ample diversity of irradiation–cooling history. This variability, coupled with the continuous impact of radioactive decay, complicated the assessment of the nuclear inventory in support of the BIO. A detailed accounting by fuel type and individual assemblies therein required a monumental effort incompatible with the BIO task schedule.

THE RFA CONCEPT AS A PRACTICAL SOLUTION

The aforementioned difficulty with the evaluation of the nuclear inventory in RBOF was circumvented by a conservative calculation based on the concept of a hypothetical, bounding reference fuel assembly (RFA) integrated over the total capacity of the facility. Thus RBOF was conservatively established as a Category-2 facility (having potential for significant on-site consequences from a radiological release). The development of the RFA is herein described as derived from a systematic ranking of the real assemblies according to a maximum burnup criterion. This article focuses on illustrating the technique rather than on providing a detailed quantitative account.

FUEL DESCRIPTION

The RBOF harbors or is expected to receive a wide diversity of DRR and FRR fuel. The facility also contains some production assemblies from SRS reactors. As previously noted, the fuel differs significantly in chemical–nuclear composition, physical configuration, burnup, and cooling. The fuel is stored primarily as bundles of intact assemblies, but some cans contain partial assemblies, assembly elements (such as plates or rods), and fragments.

Table 1 is a representative listing of SNF by element composition and configuration. Fuel in any particular combination of the indicated parameters further differs in nominal (preirradiation) isotopic composition, cladding, element dimensions, and assembly design; Table 2 illustrates such variabilities for the Experimental Boiling-Water Reactor (EBWR) fuel. Moreover, individual assemblies of a given fuel also show significant variations in original composition (relative to a nominal value), burnup, and cooling. Table 3 provides an example based on a shipment of R-2 fuel from Studsvik Nuclear AB (Sweden).

FUEL SCREENING

Summary

The first phase of the work was to rank SNF by the amount of fissile material burned. The results (fuel

Table 1 Partial Listing of Fuel Types in RBOF

Fuel[a]	Composition[b]	Elements[b]
Ames Laboratory Research Reactor	U–Al	Plates
ANL–MXOX (Argonne National Laboratory West)	PuO_2–UO_2	Rods
ASTRA (Austrian Research Centre Seibersdorf)	U–Al	Plates
B&W (Babcock & Wilcox)	U–Al, PuO_2–UO_2	Plates, rods
BMI (Battelle Memorial Institute)	U–Al	Plates
Carolinas Virginia Tube Reactor	UO_2	Rods
Dresden	UO_2, UO_2–ThO_2	Rods
EBR-II	U–Pu	Tubes
EBWR (Experimental Boiling-Water Reactor)	UO_2, U–Zr–Nb	Plates, rods
Elk River	UO_2–ThO_2	Rods
Fermi	U–Mo	Rods
GCRE (Gas-Cooled Reactor Experiment)	UO_2, UO_2–BeO	Pellets
HFIR (High-Flux Isotope Reactor)	U_3O_8–Al	Plates
HTRE (Heat Transfer Experimental Reactor)	UO_2, UO_2–BeO	Capsules
HWCTR (Heavy-Water Components Test Reactor)	Th, U, UO_2, U–Zr	Tubes
Mark-18	Pu–Al, PuO_2–Al	Tubes
Mark-22	U–Al	Tubes
Mark-42	PuO_2–Al	Tubes
MURR (Missouri University Research Reactor)	U–Al	Plates
ORR (Oak Ridge Reactor)	U–Al, U–Al–Si, U_3O_8–Al	Plates
PCA (Pool Critical Assembly)	U–Al, U_3O_8–Al	Plates
R-2 (Studsvik Nuclear, Sweden)	U–Al	Plates
RHF (Reactor á Haut Flux, France)	U–Al	Plates
Saxton	UO_2, PuO_2–UO_2	Rods
SFF/SFO (Sterling Forest Research Center Reactor)	U–Al, U_3O_8–Al	Plates
SPERT-III/SPERT C	UO_2	Pellets
SRE (Sodium Reactor Experiment)	U, U–Mo, UO_2	Slugs
TRR (Taiwan Research Reactor)	U–Al	Tubes
UVA (University of Virginia)	U–Al	Plates
VBWR (Vallecitos Boiling-Water Reactor)	UO_2	Pellets

[a]Some fuels are no longer stored in the RBOF but are cited for completeness.
[b]Most FRR fuel types consist of U–Al plates.

Table 2 Summary of Characteristics of EBWR Fuel

Bundle ID[a]	Total bundles[b]	Bundle configuration	Fuel form	Element dimensions, inches
ET-i	54	1 assembly (6 plates)	U–Zr–Nb 1.44% ^{235}U	0.212 × 3.5/8 × 54
EH-i	50			0.280 × 3.5/8 × 54
EH-17A-II, III	2			
EHS-58	1			
T-i	7		U–Zr–Nb 0.71% ^{235}U	0.212 × 3.5/8 × 54
H-i	11			0.280 × 3.5/8 × 54
ET-11	1	1 assembly (47 plates)	U–Zr–Nb 1.44% ^{235}U	0.280 × 3.5/8 × various lengths
S-i	32	1 assembly (49 rods)	UO_2–ZrO_2–CaO 93.2% ^{235}U	3/8 × 51
E-2-i	59	1 assembly (36 rods)	UO_2 6% ^{235}U	0.430 × 54.21/32
N-i	51		UO_2 0.71% ^{235}U	

[a]Index i stands for sequential numbering.
[b]Inventory as of late 1993.

Table 3 Variations in R-2 Fuel Assemblies[a]

Unit	^{235}U, g Preirradiation	^{235}U, g Postirradiation	Exposure, MWh	Decay heat, W	Discharge date
1	157.82	38.46	2274.8	2.6	12 Jun 1986
2	248.88	111.67	2615.0	3.3	19 Sep 1987
3	157.98	39.45	2259.0	3.2	10 Oct 1987
4	251.01	119.19	2512.3	2.7	14 May 1986
5	247.44	113.88	2545.5	2.8	05 Feb 1987
6	248.27	111.18	2612.7	2.8	28 May 1986
7	250.46	109.74	2681.9	2.7	10 Oct 1986
8	251.29	111.03	2673.1	2.8	12 Aug 1986
9	157.66	32.16	2391.8	3.5	29 Jan 1988
10	250.57	112.18	2637.5	3.0	08 May 1987
11	250.64	112.72	2628.5	3.1	05 Feb 1987
12	250.64	110.55	2669.9	3.0	29 May 1987
13	248.29	107.72	2679.1	3.0	20 Oct 1986
14	251.92	112.02	2666.3	3.1	29 May 1987
15	157.93	33.63	2369.0	4.2	22 Oct 1988
16	157.69	32.93	2377.7	3.8	19 Jun 1988

[a]Representative sample based on one shipment.

names only, values not shown) are shown in Table 4, limited to fuels with the highest *burnup per storage slot* (this parameter is defined in the following section). These fuels are tentatively the worst in terms of the potential consequences from a radiological release.

Table 4 Highest Burnup Fuel Types in RBOF[a]

Fissile material		
^{233}U	^{235}U	^{239}Pu
Dresden	Dresden MURR RHF Saxton	Mark-42 Saxton

[a]Ranking by fuel type only. Burnup values not shown.

Methodology

The burnup was calculated by subtracting the actinide material content (plutonium, thorium, and uranium—essentially unchanged because the fuel was removed from the source reactors) from the original (preirradiation) amount. The amount of depleted fissile material followed as the difference between the beginning-of-life (BOL) and the end-of-life (EOL) actinide amounts. Ideally, the burnup should have been computed for each individual assembly of each fuel type. In practice, though, this would have proven a monumental task because of the vast diversity of SNF in RBOF. Moreover, the fuel receipt records (especially for the older stock) were either incomplete, missing, or not readily retrievable. In many cases the inventories of interest were available for whole bundles of assemblies only.

Table 5 summarizes the burnup calculations for selected fuel types as extracted from a master spreadsheet. For convenience, SNF was classified as (1) uranium–aluminum alloy (high enrichment), (2) mixed-oxide containing plutonium, (3) uranium-based, (4) plutonium-based, and (5) mixed-oxide containing thorium. In each class the burnup per fuel unit (herein defined as the *specific burnup*) was calculated as shown for a specified inventory of a fuel type. A *fuel unit* stands primarily for a single assembly, but the concept extends to other identifiable fuel entities as constrained by available data or convenience. Differences in fuel units are reconciled by the normalization process described below.

Generally, monthly reports issued by RBOF consolidate the fuel inventory as configured for storage. Such reports provided the basis for ready calculations of *average* specific burnups for different batches of particular fuel types; however, some calculations were extended to

Table 5 Burnup Calculations for Selected DRR Fuel

Category	Fuel	Stock,[a] unit	Material	Initial loading,[b] kg/unit	Stock,[c] kg	Stock,[d] kg/unit	Burnup,[e] kg/unit	Storage capacity,[f] unit/slot	Burnup,[g] kg/slot
U–Al high enrichment	MURR	96 assemblies	^{235}U	7.850×10^{-1}	5.652×10^{1}	5.887×10^{-1}	1.963×10^{-1}	6	1.178
	RHF	4 assemblies		8.654	2.080×10^{1}	5.200	3.454	1	3.454
Mixed oxide (plutonium)	Saxton	1 can	^{239}Pu	6.950×10^{-1}	3.370×10^{-1}	3.370×10^{-1}	3.580×10^{-1}	3	1.074
	EBR-II	30 cans	^{235}U	1.450×10^{-1}	2.550	8.500×10^{-2}	6.000×10^{-2}	5	3.000×10^{-1}
			^{239}Pu	4.870×10^{-2}	1.135	3.783×10^{-2}	1.087×10^{-2}		5.433×10^{-2}
Uranium	Saxton	1 can	^{235}U	2.150	1.071	1.071	1.080	3	3.239
	SPERT-3	2 tubes		6.900×10^{-1}	6.030×10^{-1}	3.015×10^{-1}	3.885×10^{-1}	4	1.554
	SFO	678 cans		2.000×10^{-1}	1.025×10^{2}	1.511×10^{-1}	4.890×10^{-2}	9	4.401×10^{-1}
Plutonium[h]	Mark-i								
Mixed oxide (thorium)	Elk River	189 rods	^{233}U	0.000	1.472×10^{1}	6.680×10^{-3}[i]	-6.680×10^{-3}[j]	128	-8.55×10^{-1}
			^{235}U	4.162×10^{-2}	1.862×10^{2}	2.984×10^{-2}[i]	1.628×10^{-2}		2.084
	Dresden	87.139 rods	^{233}U	0.000		1.183[i]	–1.183[j]	1	–1.183
			^{235}U	3.386		2.255[i]	1.131		1.131

[a]Assemblies, bundles, cans, rods, or tubes.
[b]Nominal loadings from Appendix A in the original fuel receipt agreements.
[c]Data from RBOF nuclear accountability records.
[d]Stock (kg/unit) = stock (kg)/stock (units).
[e]Burnup (kg/unit) = initial loading (kg/unit) – stock (kg/unit).
[f]Storage capacity (units/slot) = GP tube length/unit length. A GP tube is 162.2 inches long. Assembly lengths (not shown) are obtained from Appendix A in the original fuel receipt agreements. The results are rounded down to the nearest integer.
[g]Burnup (kg/slot) = burnup (kg/unit) ∗ storage capacity (units/slot).
[h]These are primarily Mark-i (various designations) assemblies from SRS production reactors. Data for these assemblies have Unclassified Controlled Nuclear Information (UCNI) status or higher and cannot be published. Non-SRS units such as EPR-1 have low ^{239}Pu and ^{235}U burnups bounded by the Mark-i assemblies.
[i]Based on individual data.
[j]^{233}U burnup = initial ^{232}Th converted to ^{233}U minus ^{233}U remaining after irradiation. The reported values are negative because there is no ^{233}U initially.

generate specific burnup distributions on the basis of partitioned inventory data (down to single assemblies in some cases); this approach was appropriate for fuels with substantial burnup variability. The preliminary values derived from input data were next transformed to a storage slot basis via multiplication by the number of fuel units per slot. The different fuel types were then compared on this common (*normalized*) basis.

A *slot* is the minimum amount of physical space required to place a GP tube without regard to the constraints of nuclear criticality safety. On this basis, the overall capacity of RBOF storage basins is about 1700 cans. As was indicated, the number of fuel units in a GP tube depends on the fuel type and is administratively restricted to ensure nuclear criticality safety. For the current purpose, however, the individual capacity was assumed to be the number of fuel units (rounded down to the nearest unit) that could fit lengthwise in a GP tube (162.2 inches) regardless of safety limitations. This approach compensates for dimensional differences among the fuel units.

Exclusions

Certain fuels for which data were insufficient were fully or partially excluded from the master calculation spreadsheet on the premise that they had a low burnup or were unquestionably bounded by other types; for instance, the H. B. Robinson fuel had an initial enrichment of only 0.72% ^{235}U and a total uranium content of only 0.51 kg after exposure. In another case, buckets containing fragments (slugs) of Heavy-Water Components Test Reactor (HWCTR) fuel were bounded by driver assemblies in terms of ^{235}U burned.

Foreign Fuel

The FRR fuel yet to be delivered to RBOF could not be formally screened. To ensure that the RFA (defined in the section on "Reference Fuel Assembly") bounds any FRR fuel, a burnup ceiling of 3000 MWd per storage slot was imposed as an acceptance condition. This value is based on historical data for the highest exposure (about 2600 MWd) plus a 15% margin. On the basis of this criterion and unofficial fuel data, the hypothetical limit was calculated for the number of assemblies per storage slot allowable for particular fuel types. Table 6 summarizes the computations for selected fuels. In each case the required limit far exceeds the physical capacity of a storage slot. Nonetheless, the computational basis will be rigorously verified against the official fuel data as they become available.

Special Cases

Several fuels lack data on residual fissile materials, but their exposure histories are known. In these cases, the ^{235}U burnup was calculated as

$$\text{Burnup (g)} = 1.24\ \text{(g/MWd)} * \text{Exposure (MWd)}$$

The first factor on the right-hand side is based on 200 MeV per fission:

$$\text{Factor} = \frac{1\,\text{fission}}{200\,\text{MeV}} \times \frac{1\,\text{MeV}}{1.60\times10^{-13}\,\text{J}} \times \frac{1\times10^{6}\,\text{J}}{\text{MWs}} \times \frac{86\ 400\,\text{s}}{\text{day}}$$

$$\times \frac{1\,\text{mol}}{6.023\times10^{23}\,\text{atoms}} \times \frac{235\,\text{g}}{\text{mol}} = 1.05\ \text{g / MWd}$$

in turn adjusted for nonfission absorptions by the multiplier 1.169 (ratio of the absorption and fission cross sections for a thermal reaction).

RADIONUCLIDE SPECTRA OF WORST FUEL TYPES

Summary

The second phase encompassed the generation of the radionuclide distribution for the worst fuel types (highest specific burnup) identified in Phase 1. Tables 7 to 9 summarize the results for full cores of the Missouri University Research Reactor (MURR), Reactor á Haut Flux (RHF), and Dresden fuels, respectively. The Saxton fuel is not shown because it is

Table 6 Burnup Calculations for Selected FRR Fuel

Fuel	Stock, assembly	Nominal power, MW	Initial loading,[a] g/assembly	Burnup, %	Burnup,[b] g/assembly	Irradiation, d	Exposure,[c] MWd	Permissible storage,[d] assembly/slot
BER-2 (Germany)	34.5	10	180	0.56	100.8	281.65	2816.50	36
DR-3 (Denmark)	26	10	147	0.50	73.5	154.70	1547.00	50
GRR-1 (Greece)	33.3	5	180	0.30	54.0	291.71	1458.60	68
HIFAR (Australia)	25	10	150	0.39	58.5	118.51	1185.10	63
KUR (Japan)	22.7	5	180	0.24	43.2	159.12	795.60	85
Ljubijana (Slovenia)	83.3	0.25	133	0.15	20.0	5388.60	1347.10	185
MAPLE-X (Canada)	29	10	213	0.55	117.2	275.15	2751.50	31
Orphee (France)	7	14	840	0.30	252.0	102.10	1429.40	14
PARR (Pakistan)	25.4	5	196	0.35	68.6	282.17	1410.80	53
Salazar (Mexico)	100	1	133	0.15	20.0	1616.60	1616.60	185
Seoul-1 (Korea)	12.5	0.25	38	0.15	5.7	230.94	57.73	649

[a] ^{235}U.
[b] Burnup (g/assembly) = initial loading(g/assembly) $*$ burnup (%).
[c] Exposure (MWd) = irradiation (d) $*$ nominal power (MW).
[d] Permissible storage (assembly/slot) = stock $*$ 3000/exposure. Results are rounded down to nearest whole assembly.

Table 7 Actinide and Fission-Product Activities for MURR Core[a,b]

Nuclide	Type of irradiation: Uniform[c]	Type of irradiation: Cycled[d]	Nuclide	Type of irradiation: Uniform	Type of irradiation: Cycled
^{3}H	1.66×10^{1}	1.63×10^{1}	^{144}Ce	2.13×10^{4}	1.76×10^{4}
^{85}Kr	4.61×10^{2}	4.54×10^{2}	^{144}Pr	2.13×10^{4}	1.76×10^{4}
^{89}Sr	1.57×10^{1}	8.06	^{144m}Pr	2.56×10^{2}	2.11×10^{2}
^{90}Sr	3.91×10^{3}	3.89×10^{3}	^{147}Pm	8.52×10^{3}	7.86×10^{3}
^{90}Y	3.92×10^{3}	3.89×10^{3}	^{148m}Pm	1.19×10^{-2}	1.13×10^{-2}
^{91}Y	7.15×10^{1}	3.83×10^{1}	^{151}Sm	2.49×10^{1}	2.95×10^{1}
^{95}Zr	1.57×10^{2}	8.64×10^{1}	^{154}Eu	5.20×10^{1}	5.28×10^{1}
^{95}Nb	3.48×10^{2}	1.92×10^{2}	^{155}Eu	6.18×10^{1}	6.08×10^{1}
^{95m}Nb	1.16	6.41×10^{-1}	^{231}Th	9.90×10^{-3}	9.90×10^{-3}
^{99}Tc	5.90×10^{-1}	5.95×10^{-1}	^{234}Th	1.54×10^{-4}	1.54×10^{-4}
^{103}Ru	6.42×10^{-1}	3.11×10^{-1}	^{233}Pa	4.81×10^{-3}	4.82×10^{-3}
^{103m}Rh	5.78×10^{-1}	2.80×10^{-1}	^{234m}Pa	1.54×10^{-4}	1.54×10^{-4}
^{106}Ru	1.97×10^{3}	1.70×10^{3}	^{234}U	4.28×10^{-4}	4.35×10^{-4}
^{106}Rh	1.97×10^{3}	1.70×10^{3}	^{235}U	9.90×10^{-3}	9.90×10^{-3}
^{110m}Ag	8.25×10^{-1}	7.21×10^{-1}	^{236}U	1.96×10^{-2}	1.96×10^{-2}
^{119m}Sn	8.66×10^{-1}	6.98×10^{-1}	^{237}U	6.52×10^{-5}	Nil
^{123}Sn	4.93	3.39	^{238}U	1.54×10^{-4}	1.54×10^{-4}
^{125}Sb	1.76×10^{2}	1.66×10^{2}	^{237}Np	4.81×10^{-3}	4.82×10^{-3}
^{125m}Te	4.29×10^{1}	4.05×10^{1}	^{236}Pu	8.32×10^{-4}	9.15×10^{-4}
^{127}Te	1.06×10^{1}	6.97	^{238}Pu	5.29	6.12
^{127m}Te	1.09×10^{1}	7.11	^{239}Pu	9.75×10^{-2}	9.68×10^{-2}
^{129}Te	1.60×10^{-3}	7.53×10^{-4}	^{240}Pu	4.58×10^{-2}	4.66×10^{-2}
^{129m}Te	2.46×10^{-3}	1.16×10^{-3}	^{241}Pu	2.66	2.71
^{134}Cs	9.41×10^{2}	9.49×10^{2}	^{241}Am	9.28×10^{-3}	9.93×10^{-3}
^{137}Cs	4.07×10^{3}	4.05×10^{3}	^{242}Cm	1.09×10^{-3}	2.49×10^{-3}
^{137m}Ba	3.85×10^{3}	3.83×10^{3}	^{244}Cm	1.74×10^{-4}	1.80×10^{-4}
^{141}Ce	8.61×10^{-2}	4.04×10^{-2}			

[a]MURR core is made up of 8 assemblies.

[b]Computations using ORIGEN 2.1 code with pwrus cross-section library.

[c]Uniform exposure of 120 d at 11 MW (1320 MWd). All figures are in curies (Ci) for a cooling period of 2 years. Total activity = 7.35×10^{4} Ci. Omitted values are <0.001% of the overall activity.

[d]Twenty-four cycles of 5 d at 11 MW and 7.5 d at zero power. All figures are in curies (Ci) for a cooling period of 2 years. Total activity = 6.45×10^{4} Ci. Values labeled negligible are <0.001% of the overall activity.

bounded by the other types. The Mark-42 results have Unclassified Controlled Nuclear Information (UCNI) status and are therefore purposely omitted.

Preliminary Analysis

Because of the lack of exact data (power, irradiation length, and neutron spectrum) needed to simulate properly the burnup and depletion of fuel assemblies, a parametric study was first effected to compare the activities of actinide and fission products at 2 years following the irradiation of 10-kg masses of ^{233}U, ^{235}U, and ^{239}Pu. This cooling period is a lower bound for all fuels currently stored in RBOF and expected in the foreseeable future; DOE now requires RBOF to ensure the robustness of the BIO and, eventually, the SAR. For each material, the study evaluated the resulting activities for various combinations of power, irradiation length, and cross sections at a constant exposure of 3000 MWd. In turn, the results were weighted per assumed release fractions (0.1% for actinides/daughters and 100% for fission products) and inhalation committed dose-equivalent values.[7] Computations were performed with the Oak Ridge Isotope Generation and Depletion Code (ORIGEN).[8]

Table 8 Actinide and Fission-Product Activities for RHF Core[a,b]

Nuclide	D_2O neutron spectrum[c] Natural U	D_2O neutron spectrum[c] Enriched U	Nuclide	D_2O neutron spectrum Natural U	D_2O neutron spectrum Enriched U
^{3}H	3.41×10^{1}	3.41×10^{1}	^{144}Pr	4.78×10^{4}	4.78×10^{4}
^{85}Kr	9.48×10^{2}	9.49×10^{2}	^{144m}Pr	5.73×10^{2}	5.74×10^{2}
^{89}Sr	4.92×10^{1}	4.92×10^{1}	^{147}Pm	1.88×10^{4}	1.87×10^{4}
^{90}Sr	8.08×10^{3}	8.06×10^{3}	^{151}Sm	1.97×10^{1}	1.99×10^{1}
^{90}Y	8.08×10^{3}	8.07×10^{3}	^{154}Eu	1.58×10^{2}	1.61×10^{2}
^{91}Y	2.13×10^{2}	2.13×10^{2}	^{155}Eu	9.34×10^{1}	9.47×10^{1}
^{95}Zr	4.54×10^{2}	4.54×10^{2}	^{231}Th	1.14×10^{-2}	1.14×10^{-2}
^{95}Nb	1.01×10^{3}	1.01×10^{3}	^{234}Th	2.16×10^{-4}	2.16×10^{-4}
^{95m}Nb	3.37	3.37	^{233}Pa	2.71×10^{-3}	2.76×10^{-3}
^{103}Ru	2.17	2.17	^{234m}Pa	2.16×10^{-4}	2.16×10^{-4}
^{103m}Rh	1.96	1.96	^{234}U	1.50×10^{-4}	1.58×10^{-4}
^{106}Ru	4.13×10^{3}	4.15×10^{3}	^{235}U	1.14×10^{-2}	1.14×10^{-2}
^{106}Rh	4.13×10^{3}	4.15×10^{3}	^{236}U	3.28×10^{-2}	3.29×10^{-2}
^{119m}Sn	1.42	1.46	^{237}U	4.40×10^{-5}	4.17×10^{-5}
^{123}Sn	1.01×10^{1}	1.02×10^{1}	^{238}U	2.16×10^{-4}	2.16×10^{-4}
^{125}Sb	3.03×10^{2}	3.07×10^{2}	^{237}Np	2.71×10^{-3}	2.76×10^{-3}
^{125m}Te	7.39×10^{1}	7.49×10^{1}	^{236}Pu	3.63×10^{-5}	4.00×10^{-5}
^{127}Te	2.38×10^{1}	2.40×10^{1}	^{238}Pu	1.56	1.62
^{127m}Te	2.43×10^{1}	2.45×10^{1}	^{239}Pu	8.53×10^{-2}	8.72×10^{-2}
^{134}Cs	1.43×10^{3}	1.48×10^{3}	^{240}Pu	4.32×10^{-2}	4.61×10^{-2}
^{137}Cs	8.34×10^{3}	8.34×10^{3}	^{241}Pu	1.79	1.70
^{134}Cs	9.41×10^{2}	7.89×10^{3}	^{241}Am	6.10×10^{-3}	5.79×10^{-3}
^{137m}Ba	7.89×10^{3}	4.78×10^{4}	^{242}Cm	3.22×10^{-4}	3.14×10^{-4}
^{144}Ce	4.78×10^{4}	4.78×10^{4}	^{244}Cm	4.32×10^{-5}	4.68×10^{-5}

[a]RHF core consists of 2 fuel tubes with a total of 280 plates.

[b]Computations using ORIGEN 2.1 code with candunau (natural uranium) and canduseu (enriched uranium) cross-section libraries.

[c]Uniform exposure of 50 d at 62.7 MW (3135 MWd) for both cross-section libraries. All figures are in curies (Ci) for a cooling period of 2 years. Total activity = 1.60×10^{5} Ci for either natural or enriched uranium.

The study provided the following insights:

- The activities of specific actinides and fission products are highly sensitive (>50%) to the neutron spectra during irradiation, but the spreads are largely damped (<3%) upon weighting the results per radiological hazards.
- For ^{233}U and ^{235}U, the actinide levels increase and the fission-product levels decrease with increasing irradiation length (and a corresponding power decrease per the imposed constant exposure). For ^{239}Pu, the activities of both groups decrease. In all cases, however, the overall radiological hazards decrease with increasing irradiation period.
- Plutonium-239 has a higher potential radiological hazard than ^{233}U and ^{235}U. The difference narrows with increasing irradiation length.

Methodology

Depletion–burnup calculations were effected with the ORIGEN 2.1 code;[8] the specific cross-section libraries are cited in the tabulated results. This code employs the matrix exponential method to compute the buildup, decay, and processing of radioactive materials. In each case the code accepts input data in the form of BOL composition, EOL composition, irradiation history (power level and irradiation length), and reactor-specific parameters (maximum power level, number of fuel assemblies, etc.). These variables are not all independent; the analyst prescribes the appropriate set according to the information on hand. The code also provides multiple options for calculation management and output.

It is not practical to elaborate on the specifics of each fuel type here; therefore this section is limited

Table 9 Actinide and Fission-Product Activities for Dresden Core[a,b]

Nuclide	Activity[c]	Nuclide	Activity	Nuclide	Activity
^{3}H	4.39×10^{1}	^{147}Pm	1.52×10^{4}	^{229}Th	8.74×10^{-3}
^{85}Kr	1.05×10^{3}	^{151}Sm	6.94×10^{1}	^{232}Th	1.72×10^{-2}
^{89}Sr	1.54×10^{1}	^{154}Eu	2.51×10^{2}	^{231}Pa	2.28×10^{-1}
^{90}Sr	7.68×10^{3}	^{155}Eu	2.66×10^{2}	^{233}Pa	8.59×10^{-2}
^{90}Y	7.68×10^{3}	^{208}Tl	8.46	^{232}U	4.09×10^{1}
^{91}Y	7.05×10^{1}	^{209}Pb	8.47×10^{-3}	^{233}U	3.93×10^{1}
^{95}Zr	1.74×10^{2}	^{211}Pb	1.66×10^{-2}	^{234}U	1.92
^{95}Nb	3.86×10^{2}	^{212}Pb	2.36×10^{1}	^{235}U	8.87×10^{-3}
^{99}Tc	Nil	^{211}Bi	1.66×10^{-2}	^{237}U	2.59×10^{-1}
^{106}Ru	1.32×10^{4}	^{212}Bi	2.36×10^{1}	^{238}U	8.42×10^{-2}
^{106}Rh	1.32×10^{4}	^{213}B	8.74×10^{-3}	^{237}Np	8.81×10^{-3}
^{110}Ag	Nil	^{212}Po	1.51×10^{1}	^{239}Np	1.94×10^{-1}
^{110m}Ag	1.88×10^{1}	^{213}Po	8.55×10^{-3}	^{236}Pu	Nil
^{113m}Cd	3.35	^{215}Po	1.66×10^{-2}	^{238}Pu	5.19×10^{1}
^{119m}Sn	3.93	^{216}Po	2.36×10^{1}	^{239}Pu	5.80×10^{1}
^{123}Sn	1.45×10^{1}	^{217}At	8.74×10^{-3}	^{240}Pu	4.60×10^{1}
^{125}Sb	8.70×10^{2}	^{219}Rn	1.66×10^{-2}	^{241}Pu	1.06×10^{4}
^{125m}Te	2.12×10^{2}	^{220}Rn	2.36×10^{1}	^{241}Am	3.97×10^{1}
^{127}Te	3.47×10^{1}	^{221}Fr	8.74×10^{-3}	^{242m}Am	3.41×10^{-1}
^{127m}Te	3.54×10^{1}	^{223}Ra	1.66×10^{-2}	^{242}Am	3.40×10^{-1}
^{134}Cs	3.50×10^{3}	^{224}Ra	2.36×10^{1}	^{243}Am	1.94×10^{-1}
^{137}Cs	9.28×10^{3}	^{225}Ra	8.74×10^{-3}	^{242}Cm	2.69×10^{1}
^{137m}Ba	8.78×10^{3}	^{225}Ac	8.74×10^{-3}	^{243}Cm	Nil
^{144}Ce	3.63×10^{4}	^{227}Ac	1.71×10^{-2}	^{244}Cm	6.74
^{144}Pr	3.63×10^{4}	^{227}Th	1.64×10^{-2}	^{246}Cm	Nil
^{144m}Pr	4.35×10^{2}	^{228}Th	2.35×10^{1}		

[a]Hypothetical, composite core based on the highest activities for the individual nuclides out of computational runs for three actual Dresden containers stored in RBOF.

[b]Computations using ORIGEN 2.1 code with bwrus cross-section library.

[c]All figures are in curies (Ci). Total activity = 1.66×10^{5} Ci. Values labeled as negligible are <0.001% of the overall activity.

to the MURR fuel for illustration. MURR is a pressurized-water vessel contained in an open pool. Light water acts as both moderator and coolant. The reactor has a core of eight assemblies, each of which consists of 24 curved, U–Al plates.[9] Table 10 lists the BOL assembly and core compositions. The assemblies are recycled in and out of the core about 24 times during their lifetime; each cycle consists of 5 days at power and 7.5 days cooling for an overall cycle of 300 days. The reactor operates at a normal power level of 10 MW with a specific power of 1.613 kW/kg ^{235}U. Table 11 shows the EOL compositions and burnups of the highest exposed assemblies out of seven fuel shipments to RBOF, each shipment of which consisted of eight assemblies (full

Table 10 BOL MURR Fuel Composition[a]

Material	Assembly Nominal	Assembly Maximum	Core maximum
^{235}U	775.0	782.8	6 262.0
U	832.0	840.3	6 723.0
^{238}U[b]	57.0	57.6	461.0
^{239}Pu	0.0	0.0	0.0
Th	0.0	0.0	0.0
Al (fuel)	866.2	866.2	6 930.0
Al (clad)	3 012.1	3 012.1	24 097.0

[a]All figures are in grams (g).

[b]Assumed as balance upon subtraction of ^{235}U.

Table 11 EOL MURR Fuel Composition

	Shipment						
Material[a]	**1**	**2**	**3**	**4**	**5**	**6**	**7**
^{237}Np	5.42	5.420	5.420	5.420	5.420	5.420	5.420
^{235}U	582.82	584.470	587.260	584.100	584.180	585.590	581.700
^{236}U	29.99	29.860	29.440	29.970	29.970	29.610	29.960
^{238}U	56.70	56.690	56.710	56.710	56.710	56.710	56.630
Pu	0.20	0.200	0.200	0.200	0.200	0.200	0.200
Exposure[b]	149.98	149.760	147.240	147.450	149.890	148.040	149.800
Irradiation[c]	120.00	120.000	120.000	120.000	120.000	120.000	120.000
Power[d]	1.25	1.248	1.227	1.229	1.249	1.234	1.248

[a]All figures are in grams (g).
[b]Exposure (MWd) = irradiation (d) × power (MW).
[c]Irradiation length in days.
[d]Power in megawatts (MW).

Table 12 Radionuclide Distribution of RBOF Reference Fuel Assembly

Nuclide	**Activity**[a]	**Nuclide**	**Activity**	**Nuclide**	**Activity**
^{3}H	5.16×10^{1}	^{144}Pr	4.78×10^{4}	^{231}Th	1.14×10^{-2}
^{85}Kr	1.05×10^{3}	^{144m}Pr	5.74×10^{2}	^{232}Th	1.72×10^{-2}
^{89}Sr	4.92×10^{1}	^{147}Pm	1.88×10^{4}	^{234}Th	2.16×10^{-4}
^{90}Sr	8.08×10^{3}	^{148m}Pm	8.93×10^{-3}	^{231}Pa	2.28×10^{-1}
^{90}Y	8.08×10^{3}	^{151}Sm	6.94×10^{1}	^{233}Pa	8.59×10^{-2}
^{91}Y	2.13×10^{2}	^{154}Eu	7.27×10^{2}	^{234m}Pa	2.16×10^{-4}
^{95}Zr	4.54×10^{2}	^{155}Eu	3.81×10^{2}	^{232}U	4.09×10^{1}
^{95}Nb	1.01×10^{3}	^{208}Tl	8.46	^{233}U	3.93×10^{1}
^{95m}Nb	3.37	^{209}Pb	8.74×10^{-3}	^{234}U	1.92
^{99}Tc	1.03	^{211}Pb	1.66×10^{-2}	^{235}U	1.14×10^{-2}
^{103}Ru	2.17	^{212}Pb	2.36×10^{1}	^{236}U	3.29×10^{-2}
^{103m}Rh	1.96	^{211}Bi	1.66×10^{-2}	^{237}U	2.59×10^{-1}
^{106}Ru	2.11×10^{4}	^{212}Bi	2.36×10^{1}	^{238}U	8.42×10^{-2}
^{106}Rh	2.11×10^{4}	^{213}B	8.74×10^{-3}	^{237}Np	8.81×10^{-3}
^{110}Ag	2.32	^{212}Po	1.51×10^{1}	^{239}Np	9.62
^{110m}Ag	1.74×10^{2}	^{213}Po	8.55×10^{-3}	^{236}Pu	1.12×10^{2}
^{113m}Cd	6.955	^{215}Po	1.66×10^{-2}	^{238}Pu	5.19×10^{1}
^{119m}Sn	3.93	^{216}Po	2.36×10^{1}	^{239}Pu	5.80×10^{1}
^{123}Sn	1.45×10^{1}	^{217}At	8.74×10^{-3}	^{240}Pu	9.78×10^{3}
^{125}Sb	8.70×10^{2}	^{219}Rn	1.66×10^{-2}	^{241}Pu	1.06×10^{4}
^{125m}Te	2.12×10^{2}	^{220}Rn	2.36×10^{1}	^{241}Am	5.17×10^{1}
^{127}Te	3.47×10^{1}	^{221}Fr	8.74×10^{-3}	^{242m}Am	3.41×10^{-1}
^{127m}Te	3.54×10^{1}	^{223}Ra	1.66×10^{-2}	^{242}Am	3.40×10^{-1}
^{129}Te	1.20×10^{-3}	^{224}Ra	2.36×10^{1}	^{243}Am	9.62
^{129m}Te	1.85×10^{-3}	^{225}Ra	8.74×10^{-3}	^{242}Cm	4.90×10^{2}
^{134}Cs	1.03×10^{4}	^{225}Ac	8.74×10^{-3}	^{243}Cm	4.90
^{137}Cs	9.28×10^{3}	^{227}Ac	1.71×10^{-2}	^{244}Cm	2.75×10^{3}
^{137m}Ba	8.78×10^{3}	^{227}Th	1.64×10^{-2}	^{246}Cm	2.15×10^{-1}
^{141}Ce	6.46×10^{-2}	^{228}Th	2.35×10^{1}		
^{144}Ce	4.78×10^{4}	^{229}Th	8.74×10^{-3}		

[a]All figures are in curies (Ci). Total activity = 2.31×10^{5} Ci.

cores). The maximum exposed assembly among the ones compared was used as the basis for calculations.

With the preceding information on hand, the radionuclide distribution for the MURR core (Table 7) was generated for the following scenarios: (1) normal cycled operation as described earlier but at a power level of 11 MW (10% increase over nominal) and (2) uniform irradiation at 11 MW for 120 days (same overall exposure of 1320 MWd).

REFERENCE FUEL ASSEMBLY

Summary

With the worst (highest specific burnup) fuels and their corresponding radionuclide distribution established, the last phase was the definition of the desired bounding RFA. Table 12 shows the radionuclide distribution for the RFA.

Methodology

The radionuclide distribution for the RFA was constructed from the highest activities of the individual radionuclides in the distributions of the worst fuels (MURR, RHF, Dresden, and Mark-42) as shown in Tables 7 to 9. (The Mark-42 fuel is properly accounted for but not explicitly shown because of its UCNI status.) The MURR distribution (based on a full core of eight assemblies) was first adjusted by the factor 6/8 throughout to compensate for the actual capacity (six assemblies) of a storage slot in RBOF.

CONCLUSIONS

As derived from the highest burnup fuels at minimum cooling, the RFA unquestionably bounds any single assembly, or bundle of assemblies in a single storage slot, of any fuel now stored in RBOF or expected in the foreseeable future. Moreover, the nuclear inventory arising from one RFA integrated over the hypothetical, conservative capacity of RBOF (1700 slots) is absolutely bounding. (An even more conservative capacity of 2200 slots was assumed to establish RBOF as a Category-2 facility.) In the context of the applicability of RFA, the safety documentation is extremely robust and likely will remain valid for the rest of the useful life of RBOF.

The RFA concept illustrates a simple, elegant, and cost-effective solution to a uniquely complex situation. As such, it is potentially applicable to analogous scenarios in both the nuclear and the nonnuclear sectors. In the course of the review and approval of the RBOF BIO, DOE not only endorsed the RFA notion but also imposed a minimum cooling requirement of 2 years for fuel received in RBOF.

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