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N Reactor (U-metal) Fuel Characteristics for Disposal Criticality Analysis

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ABSTRACT

DOE-owned spent nuclear fuels encompass many fuel types. In an effort to facilitate criticality analysis for these various fuel types, they were categorized into nine characteristic fuel groups with emphasis on fuel matrix composition. Out of each fuel group, a representative fuel type was chosen for analysis as a bounding case within that fuel group. Generally, burnup data, fissile enrichments, and total fuel and fissile mass govern the selection of the representative or candidate fuel within that group. Additionally, the criticality analysis will also require data to support design of the canister internals, thermal, and radiation shielding. The purpose of this report is to consolidate and provide in a concise format, material and information/data needed to perform supporting analyses to qualify N-Reactor fuels for acceptance into the designated repository.

The N Reactor fuels incorporate zirconium cladding and uranium metal with unique fabrication details in terms of physical size, and method of construction. The fuel construction and post-irradiation handling have created attendant issues relative to cladding failure in the underwater storage environment. These fuels were comprised of low-enriched metal (0.947 to 1.25 wt% ^{235}U) that were originally intended to generate weapons-grade plutonium for national defense. Modifications in subsequent fuel design and changes in the mode of reactor operation in later years were focused more toward power production.

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TERMS

burnup	-	is a measure of the amount of fissile material consumed before the fuel element is removed from the reactor
fertile	-	material, that after neutron capture(s) and decay, becomes fissile
fissile	-	materials which will undergo fission with neutrons of any energy
fissile loading	-	the amount of fissionable material per unit volume within a fuel pin, assembly, or reactor core
fuel grade	-	fuel irradiation allowed higher ^{240}Pu content to maximize ^{239}Pu production for use in reactor fuels
fuel handling unit	-	a method of accounting for distinct parts and pieces of spent nuclear fuels, whether rods, plates, assemblies, targets, etc.
moderator	-	material/medium in which neutrons from fission reactions are slowed or 'thermalized' to improve neutron capture by fissile material in the reactor
weapons grade	-	fuel irradiation was controlled to minimize ^{240}Pu production relative to ^{239}Pu (< 6% of total Pu)
Zircaloy-2	-	an improved zirconium alloy (ASTM grade R60802) with a low nickel composition which avoids hydriding and loss of ductility [Ref. 1, p. 324]

ACRONYMS

BOL	beginning-of-life
CSB	Canister Storage Building (Hanford)
CSER	criticality safety evaluation report
CVD	Cold Vacuum Drying
EOL	end-of-life
DOE	Department of Energy
FHU	fuel handling unit
INEEL	Idaho National Engineering and Environmental Laboratory
LEU	low-enriched uranium (always below 5 wt% ^{235}U, but sometimes defined as <2 wt% ^{235}U)
MCO	multi-canister overpack

MTHM	metric ton heavy metal
MWd/MTHM	megawatt-days/metric ton heavy metal
PIE	post-irradiation examination
SFD	Spent Fuel Database
SNF	spent nuclear fuel
TBV	to be verified
WP	waste package

1. INTRODUCTION

1.1 Hanford Site

The N Reactor is one of several graphite moderated reactors built at the Department of Energy's (DOE) Hanford Site. Over the years beginning with the Manhattan Project, various reactors were operated mainly to produce plutonium for national defense, and to a lesser degree both research and power production. The predominant fuel type in the fuels stored at Hanford consist of those discharged from the N Reactor and other miscellaneous reactors.

Table 1-1. N-Reactor Fuel Inventories. [Ref. 2, p. 1]

Fuel Type	Quantity (MTHM)	Storage Location
N Reactor	2100.0	105 K-Basins and PUREX[a]

a. 0.3 MTHM of N Reactor fuel in PUREX canyons; operations have since consolidated these fuels into the K-Basins.

2. REACTOR INFORMATION

The N Reactor core was operated successfully from 1963 to 1987 when the reactor was placed in standby status. In 1965, the reactor had been modified to produce steam for electrical generation as well as its original mission - plutonium production. After two years in standby status, the final core was discharged in April 1989. At full power, the thermal power output of the reactor was 4000 MW. The reactor was also capable of producing 13 million pounds/hr of low-pressure steam to produce 860 MW of electricity. [Ref. 2, p. 3]

The reactor vessel consisted of an 1800-ton graphite block 10 m (33 ft) high, 10 m (33 ft) wide, and 12 meters (39 ft) long. Within this block were an array of 1003 horizontal, Zircaloy-2 process tubes that held the 366 metric tons of uranium fuel and contained the cooling water. Reactor controls were provided by 84 horizontal, water-cooled rods containing boron. These rods provided not only the reactivity control, but (neutron) flux shaping and emergency shutdown control. An independent backup emergency control shutdown system was provided by 107 vertical channels penetrating the core that could be gravity-filled with neutron absorbing material. [Ref. 2, p. 3]

The N Reactor was designed as a graphite moderated, pressurized-water cooled 'thermal' reactor that utilized an LEU fuel cycle to produce Pu-239 for national defense. The total fuel handling unit (FHU) count is ~ 104,000 items. The total mass of the fuels reported in the Integrated Spent Nuclear Fuel Data Base (INSFDB) is shown in Table 2-1.

Table 2-1. N Reactor Fuel Inventories. [Ref. 2, p. 9]

Fuel Location and Type	Uranium MT	Plutonium MT	Total MT
Weapon grade , Mark IA	39.30	0.02	39.32
, Mark IV	291.90	0.20	292.10
Fuel grade , Mark IA	588.00	1.22	589.22
, Mark IV	1176.30	2.58	1178.88
PUREX	0.30	0.00	0.30
All N Reactor SNF	2095.80	4.02	2099.82*

* This number represents a slight discrepancy between this total and the 2100.2 MTHM reported for N-reactor fuels in the SFD (Appendix A, Table A-1). Work is underway to resolve these differences.

3. DOE N REACTOR FUEL INFORMATION

3.1 Reactor Design Parameters

3.1.1 N Reactor SNF

The N Reactor core was fueled with slightly enriched (0.947 wt%, and 0.947 to 1.25 wt% ^{235}U in Mark IV or Mark1A fuels respectively [Ref. 2, p. 3]) uranium metal clad in Zircaloy-2. Differences in the enrichment were selected based on the intended mode of reactor operation, i.e. plutonium or power production.

The N Reactor fuel elements consist of the two basic design variants, both of which use two concentric tubes of uranium metal co-extruded with Zircaloy-2 cladding. Lengths and diameters vary slightly by fuel type; these differences are described in the following text and shown in Table 3-1. There is a special case of twelve Mark IA fuel assemblies that are the same maximum length as the longest Mark IV fuels, i.e. 66.3 cm (26.1-in.) [Ref. 12, pg.1-2]

The data contained in Table 3-1 is representative of the fuel 'envelope' as it may affect packaging strategies. [Ref. 2, p. 6 & Ref. 12, pg. 2-2].

3.1.1.1 Mark IV Fuel Details [Ref. 2, p.3]

Mark IV fuel elements (Figure 3-1) used two concentric tubes of uranium metal co-extruded into Zircaloy cladding. The uranium enrichment for both layers was specified to be 0.947 wt% ^{235}U, yielding an average uranium weight of 22.7 kg (50 lb.) per element. These fuels had an outer diameter of 6.1 cm (2.42-in.) and lengths varied (to facilitate reactor fuel loading configurations) as follows: 44, 59, 62, and 66 cm (17.4, 23.2, 24.6, and 26.1-in.). The Mark IV fuel element inner and outer assemblies have Zircaloy-2 end caps with an axial length of 0.48 cm (0.19-in.) on each end.

Table 3-1. 105-N Reactor Fuel Element Description. [Ref. 2, p. 6]

		Mark IV				Mark IA		
Pre-irradiation enrichment of ^{235}U		0.947% enriched				0.947-1.25% enriched		
						(inner)	(outer)	
Type-Length Code[a]		E	S	A	C	M	T	F
Length, cm (in.)		66 (26.1)	62 (24.6)	59 (23.2)	44 (17.4)	53[b] (20.9)	50 (19.6)	38 (14.9)
Element Diameter, mm (in.)								
1. Outer of outer			61.47 (2.42)				60.96 (2.40)	
2. Inner of outer			43.18 (1.70)				44.96 (1.77)	
3. Outer of inner			32.51 (1.28)				31.75 (1.25)	
4. Inner of inner			12.19 (0.48)				11.18 (0.44)	
Cladding weight, kg (lb)								
1. Outer element		1.094 (2.41)	1.041 (2.29)	0.991 (2.18)	0.791 (1,74)	0.882 (1.94)	0.832 (1.83)	0.659 (1.45)
2. Inner element		0.550 (1.21)	0.523 (1.15)	0.500 (1.10)	0.400 (0.88)	0.536 (1.18)	0.509 (1.12)	0.405 (0.89)
Weight of uranium in outer								
1. 0.947 % ^{235}U,	kg (lb.)	16.0 (35.2)	15.0 (33.1)	14.2 (31.2)	10.5 (23.1)	- - -	- - -	- - -
2. 1.25 % ^{235}U,	kg (lb.)	- - -	- - -	- - -	- - -	11.1 (24.4)	10.4 (22.9)	7.9 (17.3)
Uranium isotopics [Ref. 12, pg. 2-2]		(0.947 wt%)				(1.25 wt%)		
^{235}U		0.9470				1.2500		
^{236}U		0.0392				0.0392		
^{238}U		99.0138				98.7108		
Weight of uranium in inner @ 0.947 % ^{235}U,	kg (lb.)	7.5 (16.5)	7.0 (15.5)	6.6 (14.6)	5.0 (10.9)	5.5 (12.1)	5.1 (11.3)	3.9 (8.6)
Maximum weight of an element,	kg (lb.)	25.15 (55.32)	23.65 (52.04)	22.31 (49.08)	16.65 (36.62)	18.01 (39.62)	16.89 (37.15)	12.8 4 (28.24)
Weighted average of uranium in element, kg (lb)			22.73 (50.0)				16.32 (35.9)	
Ratio of Zircaloy-2 to uranium, kg/MT (lb/ton)		140	141.6	143.2	154.1	171.0	172.5	180.7

a. Letter code differentiates the various lengths of Mark IV or Mark1A fuel elements, i.e. a type “E” element is 26.1 inches long.

b. There are twelve Mark IA assemblies that have an overall length of 66.3 cm; they will be dealt with as a special case fuel loading in a Mark IV fuel basket.

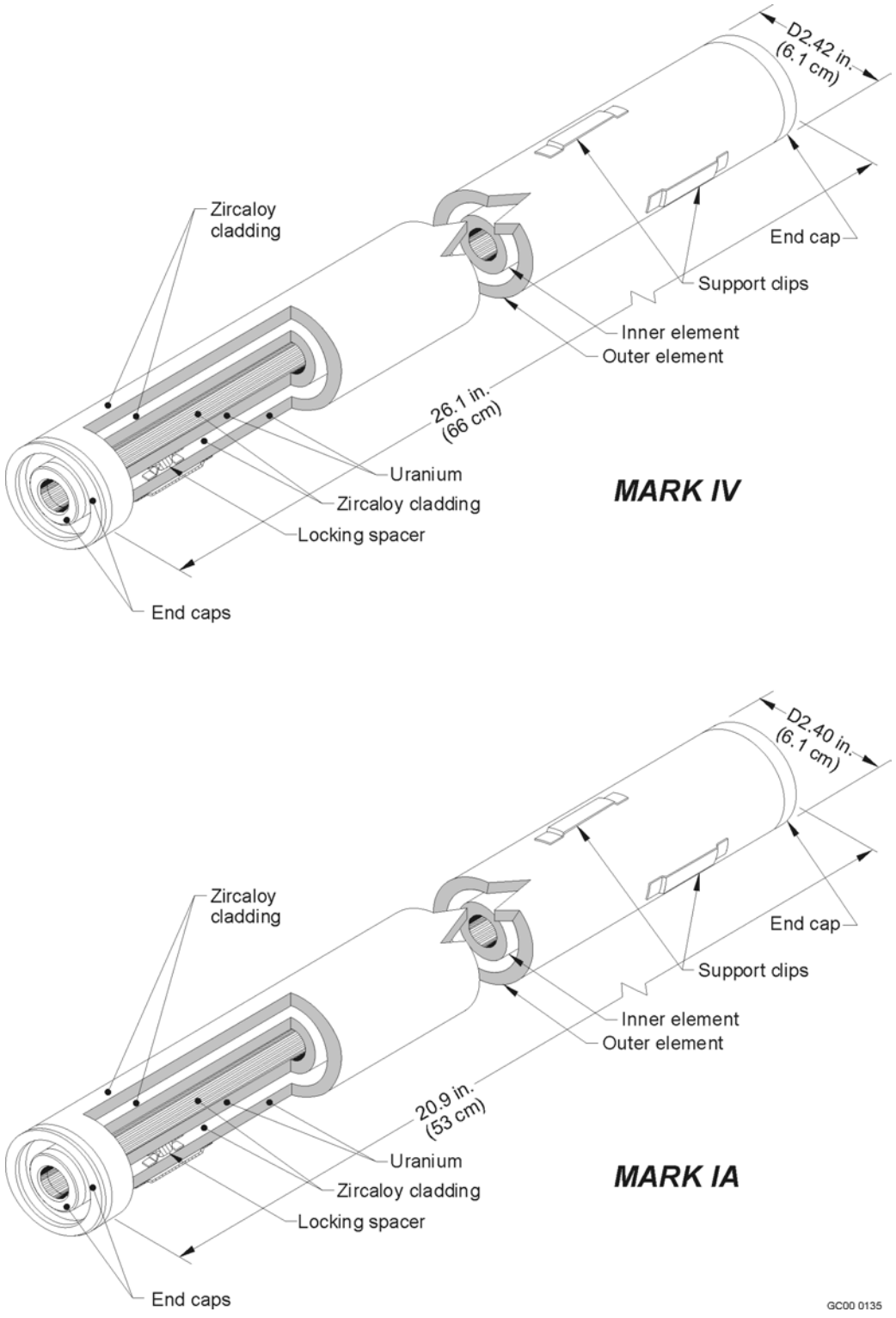


Figure 3-1. Characteristic N Reactor Fuel Types. (see Table 3-1 for other lengths) [Ref. 2, p. 4]

The construction the N-reactor fuel assemblies includes the use of annular cylinders of co-extruded uranium with Zircaloy cladding. This method of construction ultimately lead to concentric tubes filled with uranium metal alloy and an inner and outer cladding with attendant end caps. There is variability in the cladding thickness that is dependent not only on which tube is being examined, but which fuel type. The accompanying table provides a summary of calculated thickness based on specified dimensions of the fuel assemblies.

Table 3-2. Calculated Cladding Dimensions [Ref. 5, p. 5]

	Mk IV		Mk IA	
	cm	(inches)	cm	(inches)
Outer layer - outer tube	0.0640	(0.0252)	0.0635	(0.0250)
Inner layer - outer tube	0.0505	(0.0199)	0.0555	(0.0219)
Outer layer - inner tube	0.0765	(0.0301)	0.1015	(0.0400)
Inner layer - inner tube	0.0510	(0.0201)	0.0635	(0.0250)
End cap thickness	0.4830	(0.1900)	0.4830	(0.1900)

3.1.1.2 Mark IA Fuel Details [Ref. 2, p.3]

The Mark IA fuels (Figure 3-1) are differentiated from the Mark IV fuel elements in that the outer concentric tube of uranium metal consists of 1.25 wt% enriched in ^{235}U; the inner concentric tube still consists of a 0.947 wt% U enrichment. These fuels have a slightly smaller diameter of 6.1 cm (2.40-in.) than the Mark IV fuels, and their U metal weight of 16.3 kg (35.9 lb.) is somewhat less than that found in the average Mark IV elements. Fuel lengths varied by the following values: 38, 50, or 53 cm (14.9, 19.6, and 20.9-in.). An exception to these quoted lengths must include twelve Mark IA assemblies that measure 66.3 cm (26.1-in.) long. [Ref. 12]; a separate analysis has analyzed loading these special case fuels in a Mark IV basket configuration because of the added length. [Ref. 13] The Mark IA fuel element inner and outer assemblies have Zircaloy-2 end caps with an axial length of 0.483 cm (0.190-in.) on each end.

Both Mark IV and Mark IA fuels were co-located in the N reactor during operation, with the more highly-enriched Mark IA assemblies functioning as the seed or driver fuel.

3.1.1.3 Fuel Fabrication [Ref. 2, p. 5]

The following construction details are common to both types of fuels, regardless of the fuel type. Materials selection and fabrication processes were a deciding factor in cladding the elements in Zircaloy-2 to protect the uranium metal from the steam environment in the reactor, as well as supporting the efficient production of plutonium.

In the co-extrusion fabrication process, each uranium tube was clad in both an inner and outer sleeve of Zircaloy-2 with an additional outer sheath of copper. The assemblies were evacuated and sealed to prevent oxidation during preheating and extrusion. A solid state diffusion bond formed between the uranium core and the Zircaloy-2 cladding at the elevated temperatures and pressure experienced during the extrusion process. The extruded fuel was then cut to the desired lengths and a recess was machined into the uranium at each end of the fuel section. Subsequent acid stripping of the copper sheath and acid etching to remove residual uranium from the cladding prepared the element for final closure. The tube ends were closed by placing a braze ring made of Zircaloy-2 and 5% beryllium, and a Zircaloy-2 end cap in each end of the fuel. The assembly was then induction heated to brazing temperature (approximately 1050 °C) in a vacuum. The junction of the end cap, braze, and cladding was fusion welded to alloy the braze material with the cladding to improve corrosion resistance and provide a hermetic seal. The fuel was then heat treated and cleaned with an abrasive grit blast and acid baths.

Six spacers were welded to the outside of the inner tube to ensure proper alignment and locking with the outer tube when the two tubes were assembled together. Eight Zircaloy-2 support clips were welded to the outside of the outer tube to ensure proper alignment of the fuel when inserted into the N Reactor core. Low-carbon steel 'shoes' were crimped onto these Zircaloy-2 clips to reduce the effect of their rubbing against the reactor tubes during fuel element insertion and removal. The shoes varied in thickness from 10 to 18 mils, and added 0.42 to 0.74 grams of iron. The composition of the components associated with the fuel elements and the major/minor chemical constituents are shown in Table 3-3.

Table 3-3. Chemical Composition of N Reactor Fuel Assembly Components.[a] [Ref. 2, p. 7]

Element	Uranium Alloy 601	Zircaloy-2	Braze Filler
Aluminum	700–900	75	145
Beryllium	10	- - -	4.75–5.25 wt%
Boron	0.25	0.5	0.5
Cadmium	0.25	0.5	0.5
Carbon	365–735	275	500
Chromium	65	0.05–0.15 wt%	0.05–0.15 wt%
Cobalt	- - -	10	20
Copper	75	50	60
Hafnium	- - -	200	200
Hydrogen	2	25	50
Iron	300–400	0.07–0.20 wt%	0.06–0.21 wt%
Lead	- - -	100	130
Magnesium	25	20	60
Manganese	25	50	60
Molybdenum	- - -	50	50
Nickel	100	0.03–0.08 wt%	0.03–0.08 wt%
Nitrogen	75	80	200
Oxygen	- - -	- - -	2300
Silicon	124	100	250
Sodium	- - -	20	20
Tin	- - -	1.20–1.70 wt%	1.14–1.70 wt%
Titanium	- - -	50	50
Tungsten	- - -	50	100
Uranium	Balance	3.5	4
Vanadium	- - -	50	50
Zirconium	65	Balance	Balance

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

3.1.2 Thermal [Ref. 10, p. 20]

Fuel Type	Safety Basis (see Note 1)	Design Basis
Mark IV	776 watts / MCO (maximum) 1420 W / 11.6 MTU = 122.4 W/MTU Fuel decayed to 5/31/98 270 assembly/MCO x 23.48 kgU/assembly = 6339.6 kgU/MCO 122.4 W/MTU * 6339.6 kgU/MCO = 775.96 W/MCO	403 watts Average (includes Mark IV & Mark IA)
Mark IA	585 watts / MCO (maximum) 1420 W / 11.6 MTU = 122.4 W/MTU Fuel decayed to 12/31/97 = 131.75 W/MTU 288 assembly/MCO x 16.59 kgU/assembly = 4777.92 kgU/MCO 122.4 W/MTU * 4777.92 kgU/MCO = 585 W/MCO	[Total fuel heat generation in the combined basins = 161,000 W fuel decayed to 5/31/98] 161,000 W total heat in KE&KW basins = 403 W/MCO 400 MCO's (nominal) to be processed

Notes: 1. 5 fuel baskets per MCO represents bounding case for Mark IV fuels, and 6 fuel baskets for Mark IA fuels.

3.1.3 Burnup Data

Irradiation times (and burnup) for both Mark IV and Mark IA fuels were determined by whether the reactor was being operated for weapons quality Pu (< 6% ^{240}Pu content after irradiation) or predominantly for Pu production for power reactors.

The concentration of radionuclides present in the N Reactor fuel is a function of the original fuel composition, irradiation history, and decay time. Short-lived fission products have decayed to insignificant concentrations during the time since the fuel was irradiated. The radionuclide activity and decay heat present in the N Reactor fuel were determined by ORIGEN2 analysis [Ref. 2, pgs.11-13]. The activity of radionuclides in Mark IV and 1A fuel with 12% ^{240}Pu content in the plutonium, which exceeds five curies per metric ton of initial uranium, is shown in Table 3-5 at 10, 20, and 30 years since irradiation. The activity of radionuclides in Mark IV and Mark IA fuel with 6% ^{240}Pu content is shown in Table 3-6 at 5 and 10 years since irradiation. However, the activity for the fuel with 6% ^{240}Pu content in

Table 3-4. Exposure and Time Since Irradiation of 105-N Reactor Fuel. [Ref. 2, p.10]

	Fuel	Mark IV	Mark IA
Exposure:	12% ^{240}Pu	2268 MWD/MTU[a]	2730 MWD/MTU
	6% ^{240}Pu	907 MWD/MTU	1089 MWD/MTU
Decay Time:[b]	Fuel grade[c]	13–23 years	13–23 years
	Weapon grade[c]	7–8 years	7–8 years

a. MWD/MTU stands for megawatt days per metric ton of initial uranium.

b. Time since last irradiation as of 12/93.

c. Weapon-grade fuel contains plutonium having approximately 6% ^{240}Pu isotope content or less; fuel-grade fuel contains plutonium having a higher ^{240}Pu isotope content.

Table 3-5. Activity of Selected Radionuclides in Mark IA and Mark IV Fuel with 12% ^{240}Pu in the Plutonium.[a] [Ref. 2, p.11]

	(Time Since Reactor Discharge)					
	10 Years		20 Years		30 Years	
Nuclide	Mark IA	Mark IV	Mark IA	Mark 1V	Mark IA	Mark IV
3 H	3.08E+01	2.64E+01	1.76E+01	1.51E+01	1.00E+01	8.59E-00
55 Fe	8.27E-00	7.64E-00	5.75E-01	5.31 E-01	4.00 E-02	3.69E-02
60 Co	6.14E-00	4.92E-00	1.65E-00	1.32E-00	4.42E-01	3.54E-01
85 Kr	5.89E+02	4.81E+02	3.09E+02	2.52E+02	1.62E+02	1.32E+02
90 Sr	6.80E+03	5.53E+03	5.36E+03	4.35E+03	4.23E+03	3.43E+03
90 Y	6.80E+03	5.53E+03	5.36E+03	4.35E+03	4.23E+03	3.43E+03
106 Ru	5.56E+01	5.15E+01	5.74E-02	5.31E-02	5.92E-05	5.48E-05
106 Rh	5.56E+01	5.15E+01	5.74E-02	5.31E-02	5.92E-05	5.48E-05
125 Sb	1.39E+02	1.27E+02	1.13E+01	1.04E+01	9.30E-01	8.50E-01
125m Te	3.38E+01	3.90E+01	2.77E-00	2.53E-00	2.27E-01	2.08E-01
134 Cs	1.49E+02	1.22E+02	5.17E-00	4.22E-00	1.79E-01	1.46E-01
137 Cs	8.39E+03	7.01E+03	6.66E+03	5.57E+03	5.29E+03	4.42E+03
137m Ba	7.94E+03	6.64E+03	6.30E+03	5.27E+03	5.00E+03	4.18E+03
144 Ce	3.24E+01	2.67E+01	4.40E-03	3.62E-03	5.96E-07	4.91E-07
144 Pr	3.24E+01	2.67E+01	4.40E-03	3.62E-03	5.96E-07	4.91E-07
147 Pm	2.24E+03	1.88E+03	1.59E+02	1.34E+02	1.14E+01	9.55E-00
151 Sm	9.77E+01	8.74E+01	9.05E+01	6.09E+01	8.38E+01	7.49E+01
154 Eu	1.00E+02	8.49E+01	4.47E+01	3.79E+01	2.00E+01	1.69E+01
155 Eu	3.73E+01	3.42E+01	9.21E-00	8.45E-00	2.28E-00	2.09E-00
238 Pu	5.05E+01	4.82E+01	4.66E+01	4.46E+01	4.31E+01	4.12E+01
239 Pu	1.10E+02	1.10E+02	1.10E+02	1.10E+02	1.10E+02	1.10E+02
240 Pu	5.97E+01	5.77E+01	5.97E+01	5.77E+01	5.96E+01	5.76E+01
241 Pu	4.47E+03	4.43E+03	2.76E+03	2.74E+03	1.71E+03	1.69E+03
241 Am	9.33E+01	9.26E+01	1.48E+02	1.47E+02	1.81E+02	1.79E+02
244 Cm	4.64E-00	4.54E-00	3.17E-00	3.10E-00	2.16E-00	2.11E-00
Total[b]	3.83E+04	3.25E+04	2.75E+04	2.32E+04	2.11E+04	1.78E+04
Decay Heat[b]	1.02E+02	8.57E+01	8.11E+01	6.87E+01	6.71E+01	5.73E+01

a. Units are curies per MTU, except decay heat is watts per MTU. Nuclides with activity below 5 curies per assembly are not listed.

b. Totals include effect of all radionuclides present in one MTU

Table 3-6. Activity of Selected Radionuclides in Mark IV Fuel with 6% ^{240}Pu in the Plutonium and Mark IA Fuel with 6% and 16% ^{240}Pu in the Plutonium.[a] [Ref. 2, p. 12]

	(Time Since Reactor Discharge)				
	5 Years		10 Years		
	6.00% ^{240}Pu		6.00% ^{240}Pu		16% ^{240}Pu
Nuclide	Mark IA	Mark IV	Mark IA	Mark IV	Mark IA
3 H	1.58E+01	1.34E+01	1.19E+01	1.01E+01	4.59E+01
55 Fe	1.38E+01	1.27E+01	3.63E-00	3.36E-00	1.22E+01
60 Co	4.86E-00	3.89E-00	2.52E-00	2.02E-00	8.78E-00
85 Kr	3.50E+02	2.87E+02	2.53E+02	2.08E+02	8.07E+02
90 Sr	3.30E+03	2.70E+03	2.93E+03	2.39E+03	9.32E+03
90 Y	3.30E+03	2.70E+03	2.93E+03	2.39E+03	9.32E+03
106 Ru	5.56E+02	5.12E+02	1.79E+01	1.64E+01	8.52E+01
106 Rh	5.56E+02	5.12E+02	1.79E+01	1.64E+01	8.52E+01
125 Sb	1.83E+02	1.67E+02	5.25E+01	4.76E+01	2.02E+02
125m Te	4.48E+01	4.06E+01	1.28E+01	1.16E+01	4.94E+01
134 Cs	1.19E+02	9.63E+01	2.22E+01	1.79E+01	3.01E+02
137 Cs	3.76E+03	3.13E+03	3.35E+03	2.79E+03	1.20E+04
137m Ba	3.56E+03	2.96E+03	3.17E+03	2.64E+03	1.14E+04
144 Ce	1.40E+03	1.16E+03	1.63E+01	1.34E+01	3.97E+01
144 Pr	1.40E+03	1.16E+03	1.63E+01	1.35E+01	3.97E+01
144m Pr	1.68E+01	1.39E+01	1.96E-01	1.61E-01	4.77E-01
147 Pm	4.26E+03	3.55E+03	1.14E+03	9.47E+02	2.72E+03
151 Sm	6.52E+01	5.70E+01	6.28E+01	5.49E+01	1.10E+02
154 Eu	2.18E+01	1.82E+01	1.46E+01	1.22E+01	2.17E+02
155 Eu	4.35E+01	4.02E+01	2.16E+01	2.00E+01	5.14E+01
238 Pu	7.49E-00	7.41E-00	7.20E-00	7.12E-00	1.22E+02
239 Pu	5.60E+01	5.58E+01	5.60E+01	5.58E+01	1.37E+02
240 Pu	1.42E+01	1.42E+01	1.42E+01	1.46E+01	9.99E+01
241 Pu	6.54E+02	6.64E+02	5.14E+02	5.22E+02	8.72E+03
241 Am	6.00E-00	6.10E-00	1.06E+01	1.08E+01	1.84E+02
244 Cm	4.28E-02	4.37E-02	3.54E-02	3.61E-02	2.62E+01
Total[b]	2.37E+04	1.99E+04	1.47E+04	1.22E+04	5.61E+04
Decay Heatb[b]	6.37E+01	5.35E+01	4.03E+01	3.37E+01	1.48E+02

a. Units are curies per MTU, except decay heat is watts per MTU. Nuclides with activity below 5 curies per assembly are not listed.

b. Totals include effect of all radionuclides present in one MTU.

Table 3-7. Mass of Selected Actinides in Mark IA and Mark IV Fuel with 12% ^{240}Pu in the Plutonium.[a] [Ref. 2, p. 13]

	Time Since Reactor Discharge					
	10 Years		20 Years		30 Years	
Nuclide	Mark IA	Mark IV	Mark IA	Mark IV	Mark IA	Mark IV
234 U	7.46E+01	7.45E+01	7.48E+01	7.47E+01	7.50E+01	7.49E+01
235 U	8.41E+03	7.04E+03	8.41E+03	7.04E+03	8.41E+03	7.04E+03
236 U	9.22E+02	8.14E+02	9.22E+02	8.14E+02	9.22E+02	8.15E+02
238 U	9.85E+05	9.87E+05	9.85E+05	9.87E+05	9.85E+05	9.87E+05
237 Np	4.11E+01	3.94E+01	4.17E+01	4.00E+01	4.25E+01	4.07E+01
239 Pu	1.76E+03	1.77E+03	1.76E+03	1.77E+03	1.76E+03	1.77E+03
240 Pu	2.62E+02	2.53E+02	2.62E+02	2.53E+02	2.62E+02	2.53E+02
241 Pu	4.33E+01	4.30E+01	2.68E+01	2.66E+01	1.66E+01	1.64E+01
241 Am	2.72E+01	2.70E+01	4.32E+01	4.28E+01	5.26E+01	5.22E+01
Total[b]	9.97E+05	9.97E+05	9.97E+05	9.97E+05	9.97E+05	9.97E+05

a. Units are grams per metric ton of unirradiated uranium. Actinides with mass below 10 grams per assembly are not listed.
b. Totals include all actinides from one metric ton of unirradiated uranium.

Table 3-8. Mass of Actinides in Mark IV Fuel with 6% ^{240}Pu in the Plutonium and Mark IA with 6% and 16% ^{240}Pu in the Plutonium.[a] [Ref. 2, p. 13]

	5 Years Since Discharge		10 Years Since Discharge		
	6% ^{240}Pu		6% ^{240}Pu		16% ^{240}Pu
Nuclide	Mark IA	Mark IV	Mark IA	Mark IV	Mark IA
234 U	7.45E+01	6.80E+01	7.45E+01	6.80E+01	6.94E+01
235 U	1.01E+04	8.35E+03	1.01E+04	8.35E+03	7.39E+03
236 U	7.36E+02	6.90E+02	7.36E+02	6.90E+02	1.18E+03
238 U	9.87E+05	9.89E+05	9.87E+05	9.89E+05	9.84E+05
237 Np	1.59E+01	1.56E+01	1.59E+01	1.56E+01	6.74E+01
239 Pu	9.01E+02	8.97E+02	9.01E+02	8.97E+02	2.20E+03
240 Pu	6.23E+01	6.25E+01	6.23E+01	6.25E+01	4.38E+02
241 Pu	6.34E+00	6.45E+00	4.99E+00	5.07E+00	8.46E+01
242 Pu	2.46E-01	2.50E-01	2.46E-01	2.50E-01	1.64E+01
241 Am	1.75E+00	1.78E+00	3.09E+00	3.14E+00	5.36E+01
Total[b]	9.99E+05	9.99E+05	9.99E+05	9.99E+05	9.95E+05

a. Units are grams per metric ton of unirradiated uranium. Actinides with mass below 10 grams per assembly are not listed.
b. Totals include all actinides from one metric ton of unirradiated uranium.

the plutonium is not bounding, as it can be seen to be less than that of the fuel with 12% ^{240}Pu content in the plutonium on any desired date, i.e. the 10 and 20 year decayed fuel with 12% ^{240}Pu content in the plutonium have greater activity than fuel with 6% ^{240}Pu content in the plutonium decayed for 5 and 10 years respectively. Tables 3-7 and 3-8 provide a more detailed breakdown of the N fuel actinide contents.

3.1.4 Chemical/Physical Properties

Uranium used in the fabrication of the N Reactor fuels was delivered in billets with specified/allowable alloys/impurities in what is otherwise a pure, uranium metal system. The standard alloy was specified as an FEDC Alloy 601 and had the following alloy materials incorporated.

Table 3-9. Fuel Billet Alloying Materials. [Ref. 8, p. 2 & 3]

Element	Concentration (ppm)
Iron	280–440[a]
Silicon	60–130
Aluminum	650–945[a]

a. Ingots containing both 280 to 300 ppm Fe and 650 to 700 ppm Al shall not be acceptable.

The specified minimum density of the Alloy 601 fuel ingot material was listed as 18.77 grams per cubic centimeter. [Ref. 6, p. I-2] A 'production' density used in MCNP calculations by Hanford used a value of 18.82 grams per cubic centimeter. [Ref. 4, p. 5]

The specified density of the zirconium cladding is 6.55 grams per cubic centimeter. [Ref. 5, p. 5]

Table 3-10. Fuel Billet Allowable Impurities. [Ref. 8, p. 2 & 3]

Element	Maximum Concentration (ppm)
Beryllium	10
Boron	0.25
Cadmium	0.25
Carbon	735 (330 minimum)
Chromium	65
Copper	75
Hydrogen	2
Manganese	25
Magnesium	25
Nickel	100
Nitrogen[a]	75
Zirconium[b]	65

a. May be determined by sampling selected ingots on a MIL-STD-414 sampling plan.

b. May be determined on a composite of n ingots. The zirconium content of the composite sample shall not exceed 65/n ppm.

3.1.5 Fuel Storage

Extended storage of the N reactor fuels in an underwater environment has resulted in a degraded or damaged condition for much of the fuel over time (detailed in Appendix C). The sludge at the bottom of the fuel storage pools contains some small fraction of the fuel degradation products, but is not included for consideration in repository disposal.

Table 3-11. Chemical Inventory for Both KE and KW Basins [Ref. 10, p. 19]

Element	Uranium Alloy 601 (kg)	Zircaloy-2 Cladding (kg)	Braze Filler (kg)	Totals (kg)2
Al	1,480-1,900	11.1	0.411	1,700
B	0.53	0.074	0.00142	0.605
Be	21	- - -	142	163
C	769 - 1,550	40.7	1.42	1,200
Cd	0.53	0.074	0.00142	0.605
Co		1.48	0.0567	1.54
Cr	137	74-222	1.42 - 4.26	288
Cu	158	7.4	0.17	166
Fe	632-843	104 - 296	1.70 - 5.96	941
H	4.22	3.7	0.142	8.06
Hf		29.6	0.567	30.2
Hg	52.7	2.96	0.17	55.8
Mn	52.7	7.4	0.17	60.3
Mo	- - -	7.4	0.142	7.54
N	158	11.8	0.567	170
Na	- - -	2.96	0.0567	3.02
Ni	211	44.4 -118	0.851 - 2.27	294
O	- - -	- - -	6.53	6.53
Pb	- - -	14.8	0.369	15.2
Si	261	14.8	0.709	277
Sn	- - -	1,780 - 2,520	32.3 - 48.2	2190
Ti	- - -	7.4	0.142	7.54

Table 3-11. Chemical Inventory for Both KE and KW Basins (cont'd) [Ref. 10, p. 19]

Actinides	Uranium Alloy 601 (kg)	Zircaloy-2 Cladding (kg)	Braze Filler (kg)	Totals (kg)2
V	- - -	7.4	0.142	7.54
W	- - -	7.4	0.284	7.68
Zr	753	145,000	2,780	148,000
U	2,100,000	0.518	0.0113	2,100,000
Np	81.1	- - -	- - -	81.1
Pu	4,120	- - -	- - -	4,120
Am	109	- - -	- - -	109
Cm	0.018	- - -	- - -	0.018
Fission Products				
Se	12	- - -	- - -	12
Sr	152	- - -	- - -	152
Tc	170	- - -	- - -	170
Pd	133	- - -	- - -	133
Kr	77.1	- - -	- - -	77.1
I	46.1	- - -	- - -	46.1
Cs	4.65	- - -	- - -	4.65
Pm	0.50	- - -	- - -	0.50
Sm	176	- - -	- - -	176
Xe	1040	- - -	- - -	1040

1. For values with a range, the midpoint of the range is used

For purposes of estimating chemical behavior of a breached package, the various chemical species can be apportioned among the number of MCOs that are eventually generated. The number of MCOs to be generated with packaging operations ranges from ~394 (assumes no scrap baskets) to 440 (based on an estimation of the number of scrap baskets needed to accommodate damaged fuels).

4. MULTI-CANISTER OVERPACK (MCO)

Development of the MCO grew out of the Tri-Party Agreement between the State of Washington - Department of Ecology, U.S. Environmental Protection Agency (EPA), Region 10, and the Department of Energy (DOE). Part of the agreement mandated removal of spent reactor fuels from the wet-storage environment in the K-Basins because of the potential environmental concerns posed with continued wet-storage.

The MCO was initially developed as an interim dry-storage container for the various N-Reactor fuels. With an evolutionary design, it may be also end up being used as the package for transport and disposal in the repository.

Current wet-storage of the N Reactor fuels required a package design that must allow for both underwater loading of the fuel elements, and *in situ* drying of the MCO and its contents after loading.

One canister design has been proposed for use in the packaging, transport, and disposal of the N Reactor fuels [Ref. 8, # Dwg. H-2-828041, Shts 1, 2, & 3, Rev. 0]. The canister design (Figure 4-1) includes a nominal length of 4198.37 mm (165.29 -in.) and a maximum outer diameter of 642.9 mm (25.31-in.) Beyond these basic dimensions, fuel-specific internals have been designed for each canister based on the known maximum lengths of the fuels (Mark IV or IA) contained therein.

4.1 Multi-Canister Overpack Design Features

The MCO's are constructed out of 304L stainless steel having an outside diameter 60.92 cm (23.985-in.) and a wall thickness of 1.27 cm (0.5-in.) (Figure 4-1). The top portion of the MCO has a slightly larger diameter of 64.29 cm (25.31-in.) than the overall tube body in order to accommodate the top mechanical closure device. [Ref. 9, Dwg. # H-2-828042, Sht 1 of 3, Rev. 2] The overall length of the MCO is 422.707 cm (166.42-in.) with an inner cavity height to the top of the stacked baskets of 356.545 cm (140.372-in.) [Ref. 9, Dwg. # H-2-828041, Sht 1, Rev. 2]. The bottom plate has a thickness of 5.11 cm (2.01-in.) [Ref. 9, Dwg. # H-2-828044, Sht 1, Rev. 2] There is a metal 'structure' that adds another 57.91 cm (22.80-in.) to the top of the MCO above the basket that might best be approximated as a solid, 304L stainless steel shield plug. [Ref. 9, Dwg. # H-2-828041, Sht 1 of 3, Rev. 2].

In addition a central process post constructed out of 304L stainless steel is present in the MCO's. This central post is associated with the stacked baskets, and each post is drilled to facilitate water removal from the bottom of the MCO after underwater loading. In the case of the Mark IV fuel baskets, the post O.D. is 7.20 cm (2.835-in.) with a 1.37 cm (0.54-in.) thick wall. The Mark IA fuel and scrap baskets use a 16.83 cm (6.625-in.) post diameter and a 4.458 cm (1.755-in.[max.]) drilled hole in the center for a 6.18 cm (2.435-in.) wall thickness.

It is important to distinguish between what constitutes intact versus scrap material when discussing MCO basket loading. The following generic guidelines are to be used for differentiating material segregation between baskets.

Scrap can ". . .consist of material with a maximum dimension as small as 1/4 inch, but can also consist of pieces as large as entire fuel elements which do not fit in bottom plate sockets of the fuel basket." [Ref. 14, p.12]

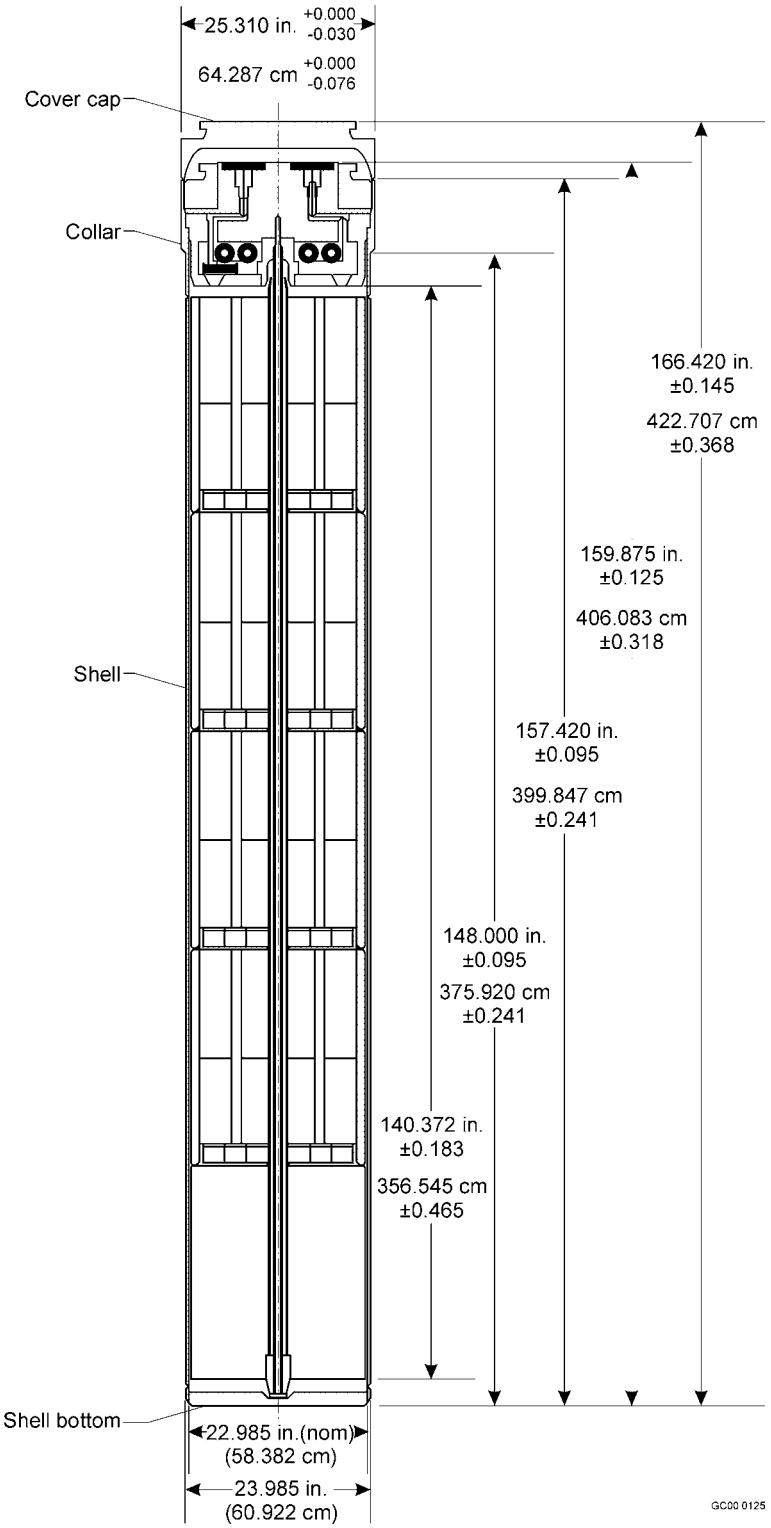
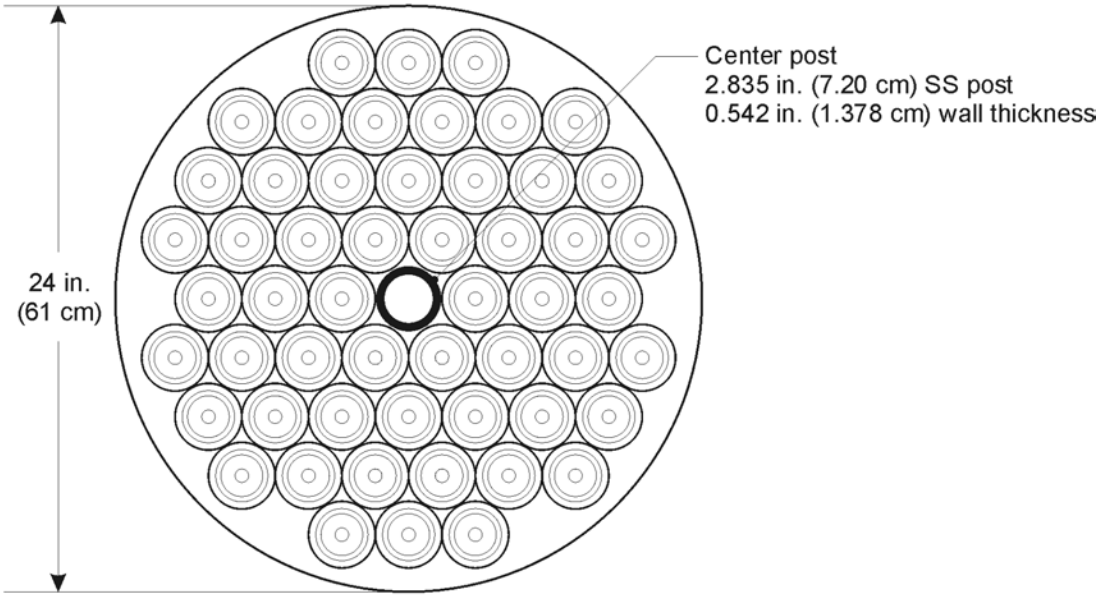


Figure 4-1. Multi-Canister Overpack. (depicted w/ 4-intact and 1 scrap Mk IV baskets) [Ref. 9, Dwg. H-2-828041, Sht. 1]

Loading Arrangement for Mark IV Fuel in MCO Container



Loading Arrangement for Mark IA Fuel in MCO Container

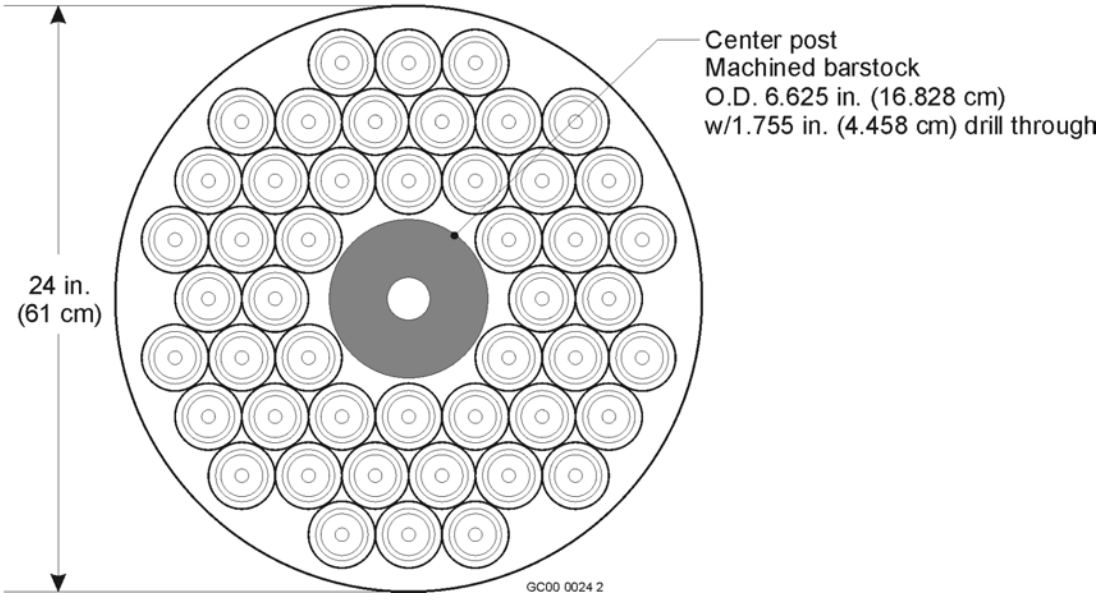


Figure 4-2. Example of Loading Arrangements in MCO's [Ref. 9].

'Intact' fuels, suitable for loading in MCO fuel baskets, are defined as material that is ". . . loaded to form fuel element pairs, at least one end of the outer element fits within the hole machined in the plate of the fuel basket, and the inner element fits within the outer element. Both elements must seat within the fuel basket holes such that the top of either element does not exceed the fuel basket height. Fuel element segments may be stacked (outer segments on intact inner or inner segments in intact outer) to form element pairs in a fuel basket position. The height of stacked segments can not exceed the length of intact element supporting the segment stack." [Ref. 14, p.14]

4.2 MCO Basket for Mark IV

The Mark IV fuel consists of an inner assembly with a pre-irradiation enrichment of 0.947% and an outer assembly with a pre-irradiation enrichment of 0.947%. The length of the Mark IV fuel ranges from 44 cm (17.4-in) to 66 cm (26.1-in.). Analyses should be based on the 66 cm (26.1-in.) long elements. The pertinent dimensions and weights comprising the Mark IV elements are given in Table 3-1 [Ref. 2].

The proposed configuration consists of fifty-four Mark IV elements per basket (Figure 4-2) loaded in an upright position. Five baskets containing Mark IV SNF are then placed into an MCO. Previous criticality safety calculations [Ref. 12 & 14] allow two of the five baskets within the MCO to contain scrap or degraded Mark IV fuel. Heat transfer considerations indicate that a maximum of two scrap baskets may be loaded in an MCO and that they must be placed into the MCO as top and bottom baskets. The Mark IV scrap baskets are limited to a maximum of 980 kg [Ref. 12, p. 4-4], and used a 0.95% enrichment value [Ref. 13, p. 33]. Scrap basket design does not limit the mass of scrap in a basket; the scrap limits are based on spills of the baskets in the K Basins where there is sludge containing fissile material on the floor [Ref. 12, pg. 4-4].

The Mark IV basket is an annular type basket constructed of 304L stainless steel (Figure 4-3). The fifty-four elements are housed in the annular section. The center post is comprised of a 7.201 cm (2.835-in.) outside diameter stainless steel, with a wall thickness of 1.378 cm (0.54-in.) [Ref. 4, Dwg. # H-2-828070, Sht. 2, Rev. 2]. There are six, 304L SS, round-bar support rods of 3.33 cm (1.3125-in.) diameter equally spaced around the outer periphery of the basket. These rods aid in distributing the axial load of the fuel/scrap baskets when the MCO is in the vertical position.

The basket used to house intact Mark IV fuel assemblies has an outer shell that extends approximately 35.56 cm (~14-in.) the height of the bucket. The outside shell is constructed of 18-gauge (0.048-in.) 304L stainless steel [Ref. 4, Dwg. # H-2-828070, Sht 1, Rev. 2]. The outer diameter of the outer shell is 57.468 cm (22.625-in. diameter). Each of the baskets, for intact assemblies, contains an aluminum element spacer guide at the bottom of the basket. This spacer is approximately 6.35 cm (2.5-in.) thick axially and arranges the Mark IV elements in the triangular pitch configuration at a typical center-to-center pitch of 6.99 cm (2.75-in.) [Ref. 4, Dwg. # H-2-828070, Sht. 1, Rev. 1] shown in Figure 4-2. The inside height of the basket is 67.30 cm (26.496-in.), with an overall outer height of 70.86 cm (27.897-in.). The perforated plate at the base of the basket is 3.05 cm (1.20-in.) thick.

The Mark IV scrap basket is also an annular type basket constructed primarily of 304L stainless steel (Figure 4-4). The Mark IV scrap material is housed in the annular section. The center post is comprised of a 7.201 cm (2.835-in.) outside diameter stainless steel pipe, with a wall thickness of 1.378 cm (0.54-

in.) with an outer shell comprised of 18-gauge (0.048-in.) stainless steel sheet metal [Ref. 4, Dwg. # H-2-828070, Sht 1 to 4, various revisions]. The outer diameter of the outer shell is equal to 57.468 cm (22.625-in.). The basket bottom plate is constructed from 3.05 cm (1.20-in.) thick stainless steel with 1.27 cm (0.5-in.) drain holes drilled through [Ref. 4, Dwg. # H-2-828075, Sht 5, Rev. 4]. The scrap baskets do not contain the aluminum element spacer guide at the bottom of the basket. The scrap baskets are divided equally into six compartments separated by six, full-height copper plates (ASTM B152) that are 0.318 cm (0.125-in.) thick. [Ref. 4, Dwg. # H-2-828075, Sht 3, Rev. 3]. The center of these assembled, truncated arcs form a hexagon with a 17.78 cm (7.0-in.) dimension across the flats (see Figure 4-7 for example of the scrap basket structure). The outer dimensional 'height' of the shroud (in cross section) is 19.84 cm (7.81-in.). The outer shroud height is 69.09 cm (27.20-in.).

4.3 MCO Basket for Mark IA Fuels

The basket is an annular type basket constructed of 304L stainless steel (Figure 4-5). The center post is machined stainless steel barstock (actual dimension of 16.828 cm (6.625-in.)) and the wall thickness is 6.19 cm (2.4375-in). The outer wall of the basket is comprised of 18-gauge stainless steel sheet metal, with the inner diameter of the outer shell equal to 57.468 cm (22.625-in.). The forty-eight elements are housed in the annular section. The overall outer height basket is 58.882 cm (23.182-in.) [Ref. 4, Dwg # H-2-828060, Sht 1, Rev 2]. The basket bottom plate is constructed from 3.05 cm (1.20-in.) thick stainless steel with drain holes drilled through. Each of the baskets contains an aluminum element spacer guide at the bottom of the basket. This spacer is approximately 6.35 cm (2.5-in.) thick and arranges the elements in the triangular pitch configuration at a typical center-to-center pitch of 6.99 cm (2.75-in.) as shown in Figure 4-2 [Ref. 4, Dwg # H-2-828060, Sht 4, Rev. 2].

The scrap baskets do not have the aluminum spacer element guide located at the bottom (Figure 4-6). Scrap consists of various sized pieces and sections from Mark IA elements that have structurally failed. The scrap baskets (Figure 4-7) are divided into six individual compartments consisting of 0.318 cm (0.125-in.) thick copper plate (ASTM B152) material [Ref. 4, Dwg # H-2-828065, Shts 1 thru 5, Rev. 1]. Each compartment occupies a 60° arc (six segments) within the canister. Six trapezoidal posts (equally spaced radially on the outer periphery of the basket) provide structural support and a small degree of standoff (58.882 cm [23.182-in.]) height from the basket shroud (56.642 cm [22.30-in.]). The basket shroud assembly is shaped like a truncated (on the pointed end) piece of pie. The 60° arc is formed on a 28.410 cm (11.185-in.) outer radius and an overall height from the outside of the radius to the outside of the flat truncation of 17.475 cm (6.88-in.). The flat-to-flat inside dimension of the hex shape formed by the joined surfaces of the scrap shrouds measures 22.54 cm (8.875-in.). A trapezoid piece of 304L SS bar stock is centered within each compartment on the outer radius. The following trapezoid cross-section dimensions (maximums) include a 3.734 cm (1.47-in.) height, 3.175 cm (1.25-in.) short base, a projected 7.960 cm (3.134-in.) long base w/ 0.635 cm (0.25-in.) radius on those corners. The thickness of the bottom plate of the basket is 3.05 cm (1.20-in.). The overall outer diameter of each basket is 57.468 cm (22.625-in.). The Mark IA scrap baskets are limited to a maximum loading to 575 kg [Ref. 12, p. 4-4], and used a 1.25% enrichment value [Ref. 13, p. 33].

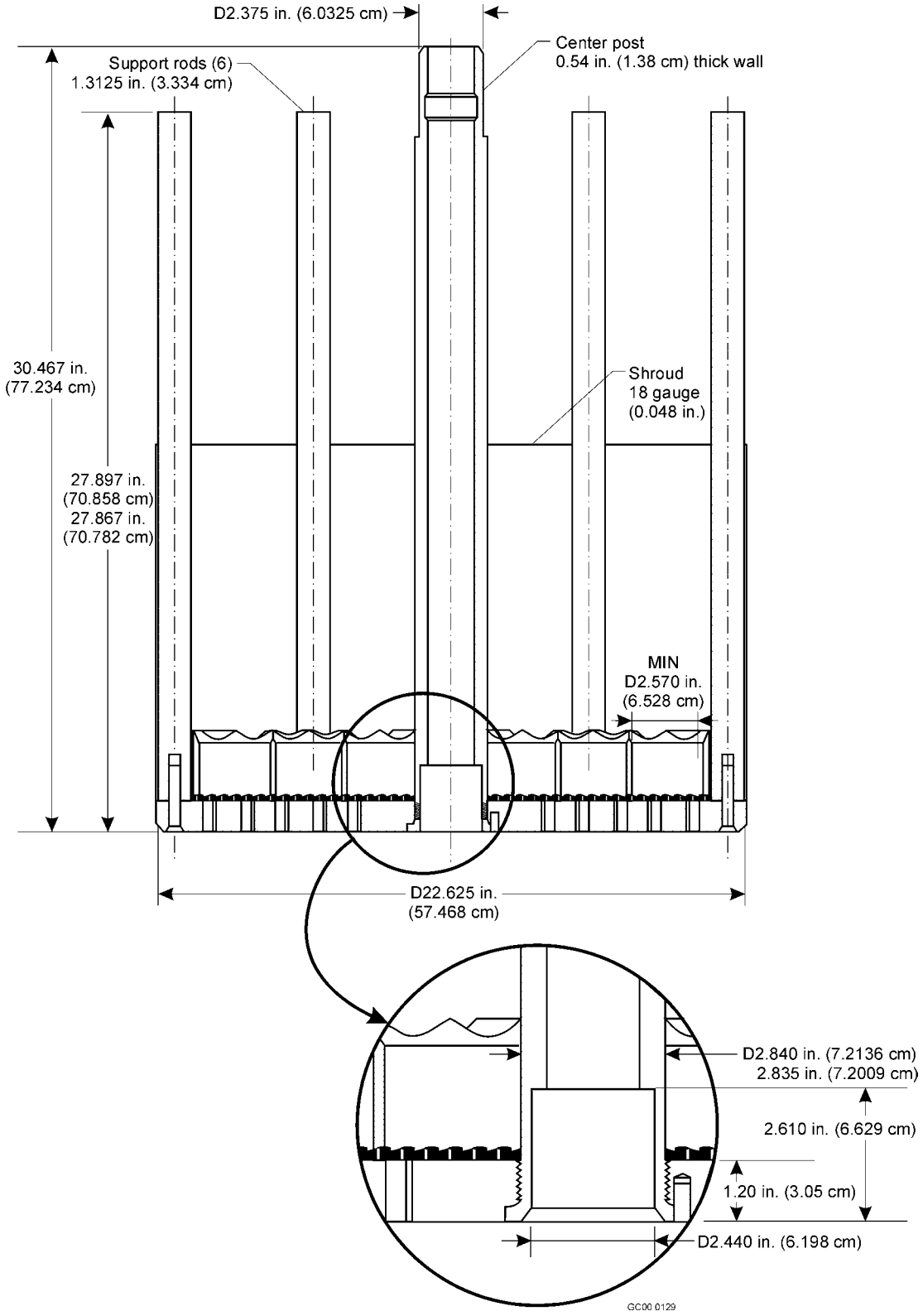


Figure 4-3. Mark IV SNF Intact Element Storage Basket.

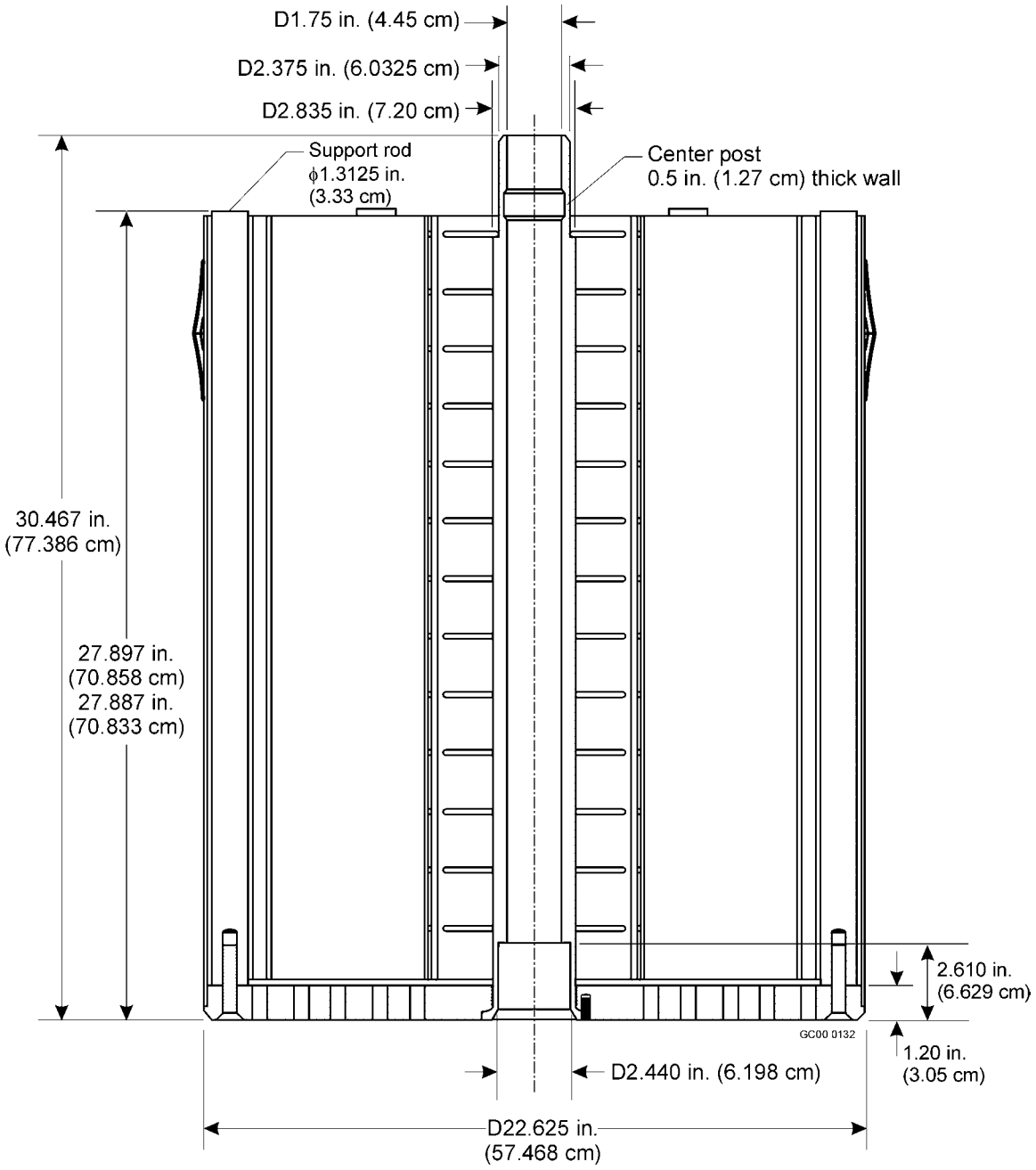


Figure 4-4. Mark IV SNF Scrap Material Storage Basket.

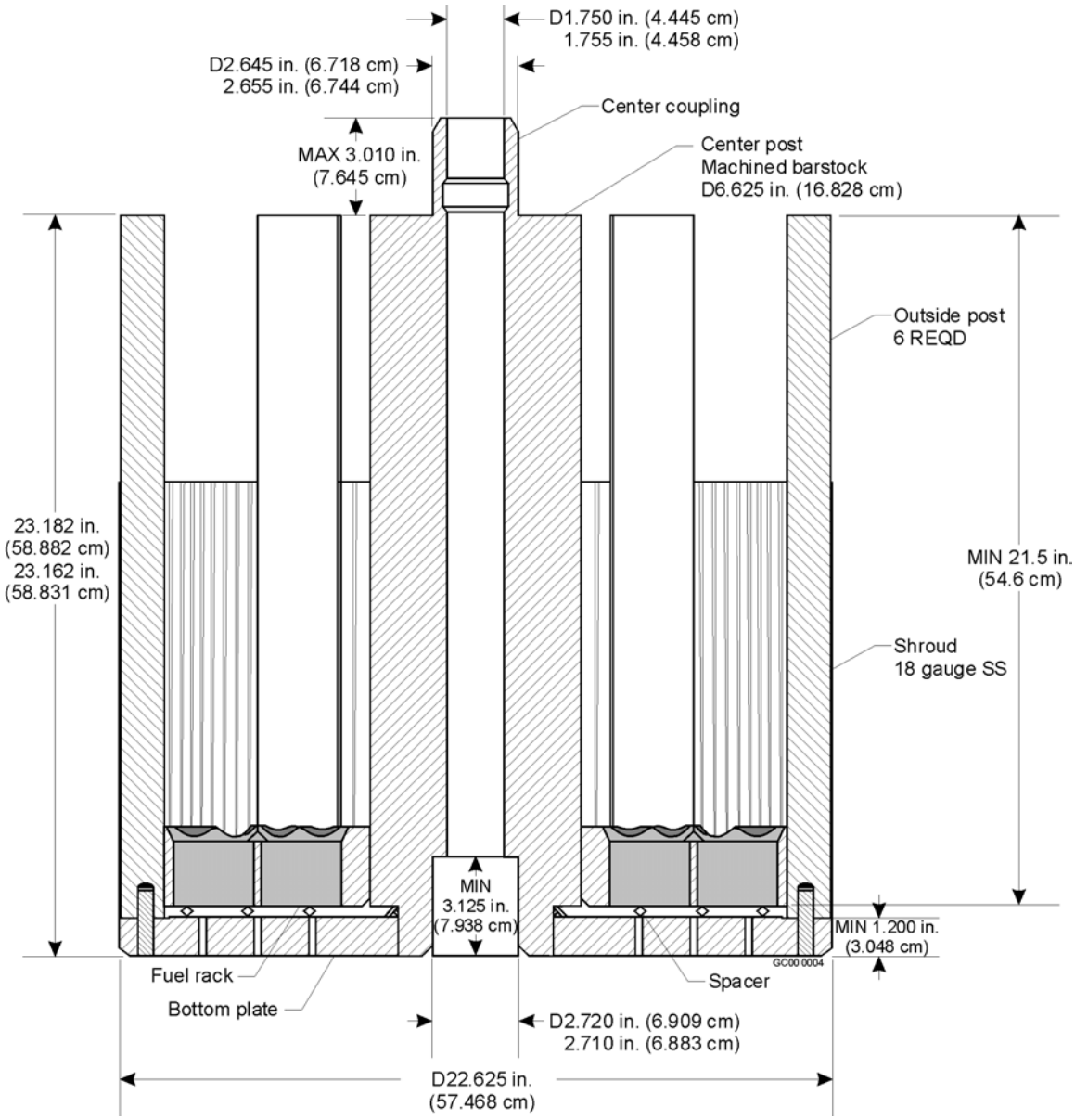


Figure 4-5. Mark IA SNF Intact Element Storage Basket.

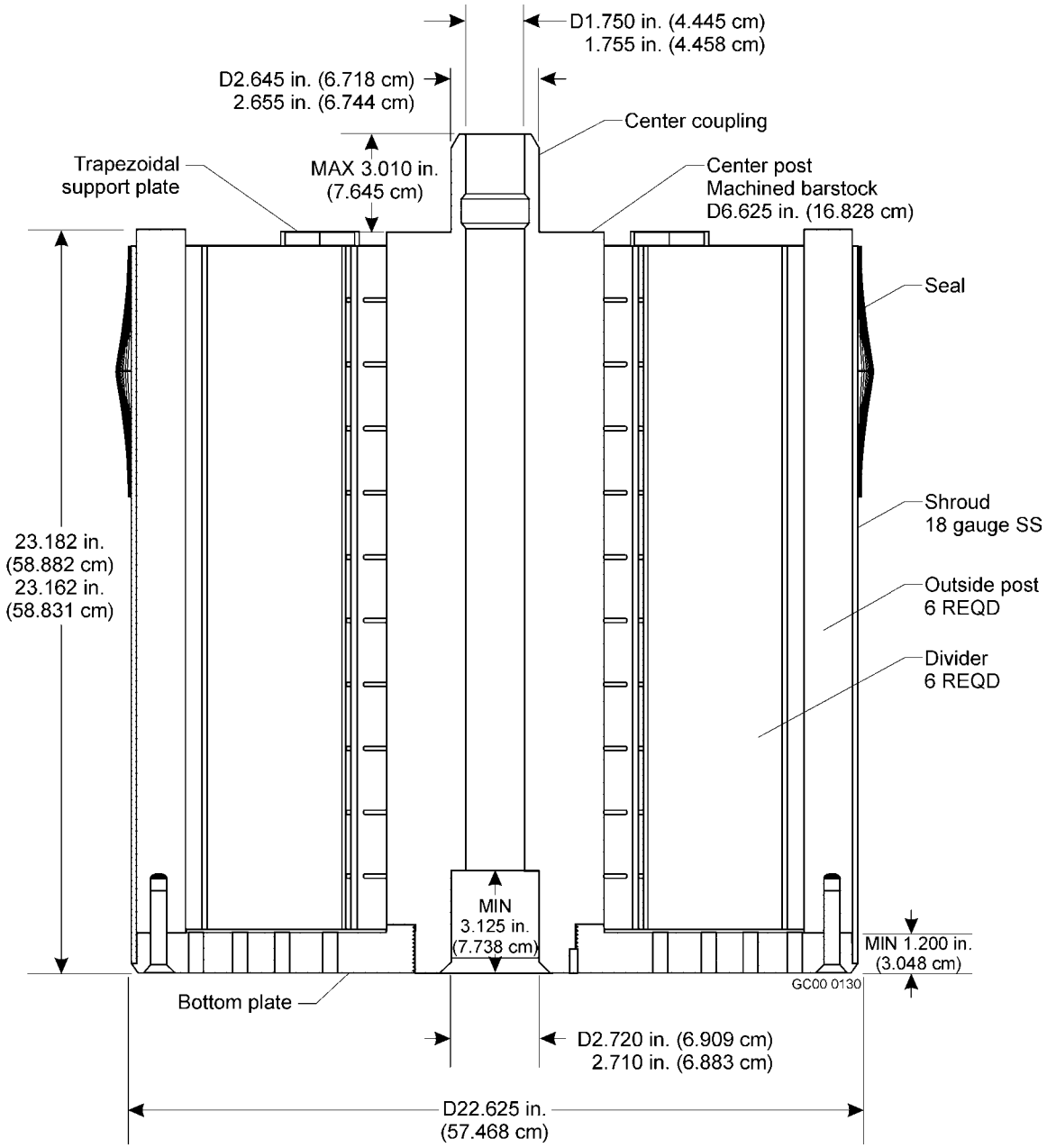


Figure 4-6. Mark IA SNF Scrap Material Storage Basket.

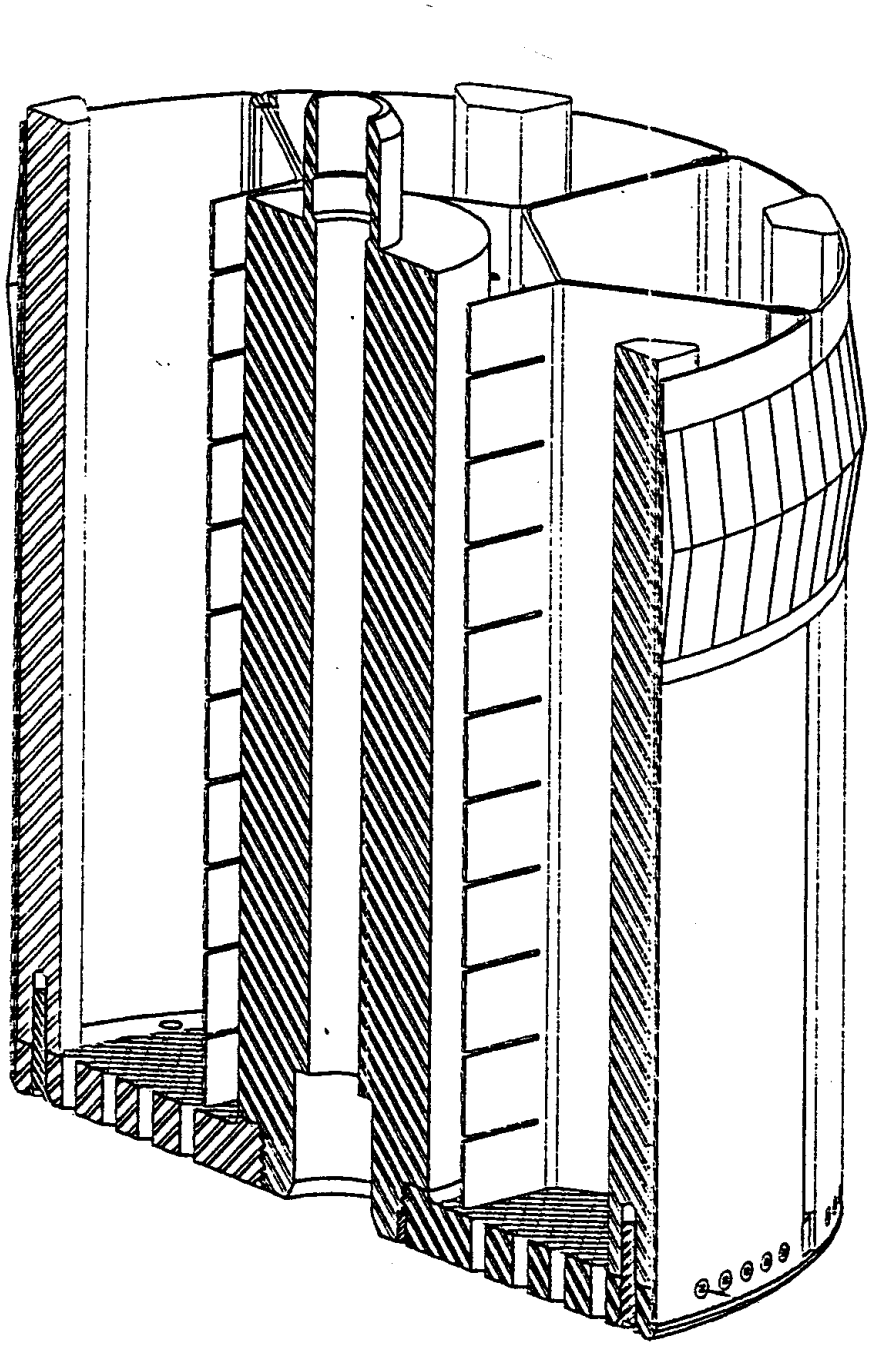


Figure 4-7. Isometric Mark IA SNF Scrap Material Storage Basket. [Ref 5., Dwg # H-2-828065, Sht 1 of 6]

4.4 MCO Assembly Weights

The weight of an assembled MCO is design dependent based on the type of fuel being loaded. Table 4-1 provides a comparison of the various masses that can be encountered based on empty, loaded (maximum), and loaded(maximum)/flooded conditions.

Table 4-1: MCO Assembly Weights [Ref. 3, p. 13]

	Mark IA fuels (maximum: 48/basket * 6 baskets = 288 elements) [240 elements & 1 scrap basket] [192 elements & 2 scrap baskets]	Mark IV fuels (maximum: 54/basket * 5 baskets = 270 elements) [216 elements & 1 scrap basket] [162 elements & 2 scrap baskets]
Empty	1990.5 kg (4379 lbs)	1905.5 kg (4192 lbs)
Loaded	7310.0 kg (16,082 lbs)	8746.4 kg (19,242 lbs)
Loaded / flooded	7906.4 kg (17,394 lbs)	9298.7 kg (20,457 lbs)

4.4.1 Scrap Baskets

There were some basic assumptions made relative to both (intact fuel) basket and scrap basket loading. All these assumptions are detailed in Section D-4 of Appendix D. For scrap baskets in particular, a brief summary of these assumptions follows:

- Only two scrap baskets may be loaded in any MCO (on either or both ends)*
- Scrap in Mark IV (scrap) baskets shall not exceed 26.5 inches in depth, nor 21 inches in depth for Mark IA fuels in their respective scrap baskets*

- Mark IA fuels are not allowed in any basket intended for Mark IV fuels, either scrap or intact*

**Note: some of the bases identified in earlier studies have been superceded by subsequent analyses. Scrap baskets are installed on the ends of the MCO due to heat transfer considerations; ultimately, for the degraded package condition, all baskets and their contents will be analyzed in a degraded condition and they also must meet criticality safety requirements. Similarly, while 'heights' are specified for scrap basket loadings, it will be mass loadings that govern acceptability. And as with all generalizations, there will be the exceptions caused as in the case of twelve specific Mark 1A fuel assemblies that need to be loaded into a Mark IV basket(s) to accommodate their overlength dimensions.*

The original specification identified a design feature common to both scrap baskets that provides 'six equally spaced 1/4 inch thick copper divider plates to segment the scrap into six equal area compartments'. Detailed design drawings now indicate those compartments will be constructed of 0.318 cm (0.125-in.) thick copper plate material (ASTM B-152 UNS 12200-060). Furthermore, the specification states: 'A partitioned area within the basket shall be designed for scrap fines loading. Scrap fines will vary in size from 1/4 inch pieces to approximately 1-inch pieces. The total volume of the partitioned area shall not exceed 10 percent of the basket area.' [Ref. 3, p. 17 & 18] The redesign of the center post in the Mark IA scrap basket virtually precludes intentional addition of any scrap between the center post and the inner wall of the scrap basket channels.

Details associated with assumptions used in sensitivity studies of loaded MCOs are found in [Ref. 12, Appendix B]

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Appendix A

U-metal Fuel Inventories

Table A-1. U-metal Fuel Inventories.

Compound	Clad	Site	Fuel Name	No. of Fuel Unit (yr 2035)	Fuel Unit	BOL U235 (kg)	Fissile Mass (kg)	Effect Enrich %	EOL U235 %	U Mass (kg)	Total Mass (kg)	Vol (M^3)	MTHM
U Metal	Zirconium	SRS	EBWR (U-METAL) ENRICHED HEAVY [64]	53	ASSEMBLY	42.93	37.82	1.27	1.14	2978.5	3922.0	0.76	2.98
U Metal	Zirconium	HANFORD	N REACTOR [147]	50694	ASSEMBLY		12991.86	1.13	0.95	1143793.0	1723596.0	99.87	1145.94
U Metal	Zirconium	HANFORD	N REACTOR [148]	52986	ASSEMBLY		12175.12	1.28	1.08	952385.2	1801524.0	104.38	954.26
U Metal	Zirconium	SRS	HWCTR SMT-1-2 (U-METAL) LEU [767]	2	ELEMENT		0.19	0.66	0.52	29.0	6.2	0.00	0.03
U Metal	Zirconium	SRS	HWCTR RMT (U-METAL) LEU [790]	5	ELEMENT		0.14	0.62	0.62	22.5	147.6	0.03	0.02
U Metal	Zirconium	SRS	HWCTR TWNT (U-METAL) LEU [791]	17	ELEMENT		3.01	0.65	0.58	461.7	771.8	0.42	0.46
U Metal	Zirconium	SRS	HWCTR ETWO (U-METAL) LEU [867]	6	ELEMENT		0.82	1.69	1.69	48.5	177.1	0.04	0.05
U Metal	Zirconium	SRS	HWCTR SMT-1-3 (U-METAL) LEU [868]	5	ELEMENT		0.18	0.52	0.39	34.9	15.4	0.00	0.03
U Metal	Zirconium	SRS	HWCTR IMT (U-METAL/ZR) DU [869]	65	ELEMENT		0.65	0.82	0.51	78.7	201.5	0.03	0.08
U Metal	Zirconium	SRS	HWCTR SPR (U-METAL) LEU [879]	4	ELEMENT		0.44	0.69	0.58	64.0	224.8	0.03	0.06
U Metal	Zirconium	SRS	HWCTR TFEN (U-METAL) LEU [880]	13	ELEMENT		2.33	1.31	0.93	176.8	590.2	0.22	0.18
U Metal	Zirconium	SRS	EBWR (U-METAL) ENRICHED THIN [887]	54	ASSEMBLY		28.68	1.31	1.11	2189.0	6548.9	0.77	2.19
U Metal	Zirconium	SRS	EBWR (U-METAL) ET-11 [888]	1	ASSEMBLY	0.58	0.50	1.29	1.10	38.3	60.0	0.01	0.04
U Metal	Zirconium	SRS	EBWR (U-METAL) NORMAL HEAVY [889]	11	ASSEMBLY	4.42	3.97	0.70	0.66	565.8	814.0	0.16	0.57
U Metal	Zirconium	SRS	EBWR (U-METAL) NORMAL THIN [890]	7	ASSEMBLY	2.02	2.00	0.72	0.69	278.9	385.0	0.10	0.28
												Total	2107.18

* Effective Enrichment % = Σ ($^{233}U + ^{235}U + ^{239}Pu$)/ U_{TOT} (in mass units)

Data from INEEL Spent Fuel Database as of December 22, 1999 (Rev. 3.60). The information contained in this database is neither 'qualified' nor has it been validated/verified. It can be used in a qualitative manner for selection of fuels to be evaluated in conceptualized packaging. Specific fuel assemblies information must be gathered from specific, referenced documents for any detailed analyses.

All the N Reactor fuel assemblies and the scrap materials employ an identical MCO canister configuration but with variations for the internal basket structures.

Appendix B

Source Term Data

Appendix B

Source Term Data

Time Since Last Irradiation

Based on Mark IV fuel assemblies with a 16% Pu-240 content discharged from the reactor in 1984, and decayed to 1/1/95.

Shielding Source Terms

The gamma sources for the N Reactor fuel assemblies are given below in Table B-1 through B-4. These values represent the highest and average source exhibited by any MCO containing N Reactor assemblies. An ORIGEN2 computer code was used to calculate the gamma ray source. [Ref. 7, p. 3]

Table B-1. Maximum Photon Source Term per MCO. [Ref. 7, p. 4]

Upper Energy Boundaries (MeV)	Average Energy (MeV)	Photons/s/MCO
0.02	1.50E-02	1.75E+15
0.03	2.50E-02	3.87E+14
0.05	3.75E-02	4.21E+14
0.07	5.75E-02	3.46E+14
0.10	8.50E-02	1.95E+14
0.15	1.25E-01	1.48E+14
0.30	2.25E-01	1.66E+14
0.45	3.75E-01	8.64E+13
0.70	6.62E-01[a]	2.81E+15
1.00	8.50E-01	1.04E+14
1.50	1.25E-00	4.33E+13
2.00	1.75E-00	1.29E+12
2.50	2.25E-00	9.42E+10
3.00	2.75E-00	4.67E+09
4.00	3.50E-00	6.04E+08
6.00	5.00E-00	3.71E+05
8.00	7.00E-00	4.23E+04
14.00	1.10E+01	4.84E+03
	Total	6.45E+15

a. Changed from 0.575 MeV to 0.662 MeV to accurately reflect ^{137m}Ba gamma ray.

Table B-2. Average Photon Source Term per MCO. [Ref. 7, p.5]

Upper Energy Boundaries (MeV)	Average Energy (MeV)	Photons/s/MCO
0.02	1.50E-02	1.56E+14
0.03	2.50E-02	3.25E+13
0.05	3.75E-02	3.60E+13
0.07	5.75E-02	3.12E+13
0.10	8.50E-02	1.74E+13
0.15	1.25E-01	1.21E+13
0.30	2.25E-01	1.46E+13
0.45	3.75E-01	6.60E+12
0.70	6.62E-01*	2.06E+14**
1.00	8.50E-01	2.79E+12
1.50	1.25E-00	1.79E+12
2.00	1.75E-00	6.24E+10
2.50	2.25E-00	2.84E+09
3.00	2.75E-00	6.81E+07
4.00	3.50E-00	8.67E+06
6.00	5.00E-00	2.03E+04
8.00	7.00E-00	2.31E+03
14.00	1.10E+01	2.63E+02
	Total	5.17E+14

* Changed from 0.575 MeV to 0.662 MeV to accurately reflect ^{137m}Ba gamma ray.

** Scaled to conserve energy as a result of changing energy bin from 0.575 MeV to 0.662 MeV.

Table B-3. Maximum Neutron Source Term for an MCO. [Ref. 7, p. 8]

Component of Source	Source Strength (neutrons/s/MCO)
(α,n)	3.95×10^6
Spontaneous fissions	7.79×10^6
Total	1.17×10^7

Table B-4. Average Neutron Source Term for an MCO. [Ref. 7, p. 8]

Component of Source	Source Strength (neutrons/s/MCO)
(α,n)	2.07×10^6
Spontaneous fissions	2.35×10^6
Total	4.42×10^6

Table B-5. N-Fuel Source Term - Shielding Design Basis [Ref. 10, p. 20]

(based on Mk IV fuel at 16% Pu-240 aged 13.5 years)

Safety Basis and Design Basis are the same.

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
H-3	4.03E+01	1.36E-03	Sn-123	4.08E-09	1.26E-11
C-14	5.27E-01	1.54E-04	Sn-126	1.22E-01	3.57E-05
Fe-55	5.23E-00	1.75E-04	Sb-124	0.00	0.00
Co-60	6.27E-00	9.64E-02	Sb-125	0.00	0.00
Ni-59	3.03E-02	1.21E-06	Sb-126	1.71E-02	3.10E-04
Ni-63	3.53E-00	3.57E-04	Sb-126m	1.22E-01	1.57E-03
Se-79	6.23E-02	1.93E-05	Te-123m	1.19E-13	1.72E-16
Kr-85	6.23E+02	9.33E-01	Te-125m	0.00	0.00
Sr-89	0.00	0.00	Te-127	1.16E-10	1.57E-13
Sr-90	8.19E+03	9.50E-00	Te-127m	1.19E-10	5.88E-14
Y-90	8.19E+03	4.53E+01	Te-129	0.00	0.00
Y-91	0.00	0.00	Te-129m	0.00	0.00
Zr-93	2.83E-01	3.25E-05	I-129	4.88E-03	2.29E-06
Zr-95	4.20E-18	2.12E-20	Cs-134	1.07E+02	1.10E-00
Nb-93m	1.38E-01	2.47E-05	Cs-135	5.77E-02	1.93E-05
Nb-95	9.33E-18	4.48E-20	Cs-137	1.13E+04	1.14E+01
Nb-95m	3.12E-20	4.12E-23	Ba-137m	1.07E+04	4.18E+01
Tc-99	2.08E-00	1.05E-03	Ce-141	0.00	0.00
Ru-103	0.00	0.00	Ce-144	1.75E-00	1.15E-03
Ru-106	9.38E-00	5.59E-04	Pr-143	0.00	0.00
Rb-103m	0.00	0.00	Pr-144	1.73E-00	1.26E-02
Rb-106	9.38E-00	8.97E-02	Pr-144m	2.10E-02	7.05E-06

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Pd-107	1.44E-02	7.94E-07	Pm-147	1.06E+03	3.88E-01
Ag-110	4.15E-06	3.11E-11	Pm-148	0.00	0.00
Ag-110m	3.12E-04	5.18E-06	Pm-148m	0.00	0.00
Cd-113m	3.96E-00	4.35E-03	Sm-151	1.08E+02	1.26E-02
Cd-115m	0.00	0.00	Eu-152	1.22E-00	5.50E-03
In-113m	2.59E-11	5.98E-14	Eu-154	2.02E+02	1.81E-00
Sn-113	2.59E-11	4.30E-15	Eu-155	3.42E+01	2.48E-02
Sn-119m	0.00	0.00	Gd-153	3.92E-06	3.53E-09
Sn-121m	0.00	0.00	Tb-160	2.22E-19	1.78E-21
	Fission and Activation Product Totals			4.06E+04	1.12E+02
		Actinides			
U-234	3.92E-01	1.11E-02	Pu-241	9.42E+03	2.92E-01
U-235	1.31E-02	3.54E-04	Pu-242	7.46E-02	2.17E-03
U-236	7.12E-02	1.89E-03	Am-241	2.92E+02	9.57E-00
U-238	3.35E-01	8.35E-03	Am-242	3.19E-01	2.66E-03
Np-237	4.42E-02	1.27E-03	Am-242m	3.21E-01	4.91E-04
Pu-238	1.28E+02	4.16E-00	Am-243	2.22E-01	7.02E-03
Pu-239	1.68E+02	5.14E-00	Cm-242	2.65E-01	9.59E-03
Pu-240	1.28E+02	3.90E-00	Cm-244	4.62E-00	1.59E-01
		Actinide Totals		1.01E+04	2.33E+01

06/02/1997 RADNCU2A run for shielding design basis (Mk IV fuel at 16% Pu-240). Results decayed to 5/31/1998. Total mass of fuel in this run 2.50e+00 MTU. Total activity of fuel in this run 1.32e+05 Ci. Total heat generation from the fuel in this run 3.55e+02 W.

Appendix C

Estimates of Particulate Mass in Multi-Canister Overpacks

[excerpted directly from Ref. 11; no graphics or pictures included]

ESTIMATES OF PARTICULATE MASS IN MULTI-CANISTER OVERPACKS [Ref. 11]

1.0 PURPOSE

The purpose of this report is to document the process and results of expert judgement estimates of the particulate mass loaded with K-basins fuel into a multi-canister overpack container (MCO). The purpose of the process is to create defensible low, best-estimate, and bounding values using the best available evidence and to elucidate the relationship between evidence and values.

2.0 SUMMARY

Low, best-estimate, and bounding values for the particulate mass loaded into a multi-canister overpack container with fuel and scrap are provided here. Due to inherent variability in the geometry of fuel (damage level, character of scrap, character of particulate, etc.) and the limited amount of characterization data presently and economically available, a statistically derived estimate cannot be calculated. Therefore, values here are based upon collective expert judgement with a clear traceable path from available evidence to the bottom-line results. The judgement process and body of evidence are documented here.

The judgement process was carried out by a group of persons with diverse backgrounds but familiarity with K-basins fuel and the issues associated with particulate loading. Participants were: Thierry Flament (Numatec), Lewis Muhlestein (ARES); Al Pitner (Duke Engineering & Services Hanford), Marty Plys (Fauske & Associates), and Jim Sloughter (Numatec). A brief statement of qualifications appears in Appendix A.

The motivation for the expert judgement was to consider new characterization data which was unavailable for previous estimates, and to; reconsider the estimation technique based upon the observations.

Basic steps in the process were to gather and review characterization data, identify and mathematically describe locations holding particulate, formulate appropriate technical bases and assumptions, list the required parameters, and provide parameter values supported by the evidence. Evidence used for the judgement includes videotapes and photographs of fuel assemblies and particulate, survey data on the damage state of K-basins fuel, and characterization data on particulate. Simple geometric relations were developed for the identified particulate locations. Reasons for selection of each parameter value are documented here.

The rounded best-estimate particulate loading of an MCO is 8 kg and the bounding particulate loading is 60 kg; a low value is 2 kg. The greatest contribution to these values, nearly 50%, is particulate generated in the scrap basket before and during CVD -- this is noteworthy because it can contain only a very limited amount of chemically bound water.

3.0 SOURCES

Sources of information and evidence used as basic references by the panel to formulate technical bases, create assumptions, and assign values to process and geometric parameters are described here.

3.1 K-Basins Fuel Survey Data

A full basin video survey of fuel stored in K East Basin was conducted in 1994 [Pitner 1995]. The fuel in this basin is stored in open canisters with the tops of the fuel elements visible in most of the canisters (some were covered with debris). The condition of each of the ~35,000 fuel assemblies visually examined was rated based on the top end views. The video survey was subsequently followed up by limited "lift and look" surveys where about 225 individual assemblies were extracted from selected canisters and examined over their full length [Pitner 1997a]. These extended examinations basically confirmed the results for fuel damage distributions established from the previous "top only" visual survey.

In K West Basin, the fuel is stored in sealed canisters with lids placed atop the canisters. Valve systems are incorporated in the lids to permit the addition of corrosion inhibitor (potassium nitrite) and cover gas (nitrogen). A limited "lift and look" survey was conducted in K West Basin where lids were removed from twenty selected canisters and about 250 fuel assemblies were extracted and examined [Pitner 1997b].

The fuel condition in these surveys was rated based on the following categories.

Intact - No evidence of cladding breach or deposited sludge. An example of an intact inner element from K East Basin is shown in Figure 1. The element is being held by a grapple used to extract it from the canister. A W spring and spacer shoe used to center the inner element inside the outer element are visible in the photo.

Breached - Minor cladding rupture with no reacted fuel or deposited sludge visibly present. Figure 2 presents an example of a breached outer fuel element from K West Basin. A top end rupture and a minor spiral crack can be seen. A portion of the top of the

element evidently protruded into the cover gas above the water in the canister, and ' does not exhibit the grayish coating seen on the lower portion.

Defected - Definite evidence of cladding breach with reacted fuel egressing from the element. The amount of exposed fuel may be significant, but there is no gross cladding splitting, element dilation, or fuel voiding. In Figure 3 (not included), the top end of an inner element from K East Basin is seen to be ruptured with reacted fuel present in the area of the rupture.

Bad - Gross failure is evident with substantial element dilation, cladding splitting, fuel "mushrooming," or fuel voiding. Figure 4 (not included) shows a canister in K East Basin that contains fuel assemblies that would all be rated in the bad category. Considerable cladding splitting and fuel voiding are prevalent in this canister.

The fuel damage distribution determined from these surveys is presented in the following table.

	K East	K West
Intact	49 %	50 %
Breached	9 %	39 %
Defected	38 %	0 %
Bad	4 %	11 %

About half the fuel assemblies in both basins were found to be intact. It is seen that no K West fuel assemblies fall in the defected category. This presumably reflects the effectiveness of the corrosion inhibitor added to the canisters. While the relative level of bad fuel assemblies in K West appears to be significantly greater than that in K East, the actual degree of damage on bad K West elements is generally less than that seen on bad K East elements. Figure 5 (not included) shows a K West element that has split cladding on the top end, and would accordingly be classified as bad. The overall damage level on this element, however, is significantly less than any of those shown in Figure 4 (not included) for K East bad elements.

When fuel elements were extracted from the canisters for examination or transfer to a shipping cask for subsequent characterization activities, sludge clouds often trailed the element, particularly in the K East Basin. This is sludge that is drawn from the bottom of the canister by the lifting action on the element. Figure 6 (not included) shows an intense sludge cloud trailing a K East outer element being transferred from the canister to a shipping container. The sludge clouds were very flocculent in nature, but generally settled relatively quickly.

Most of the fuel elements in the K Basins had a thin grayish colored coating on their surface. It was found that this light coating was easily removed by a wire brush apparatus. Figure 7 (not included) shows an outer element after wire brushing the portion below the top spacers shoes. As shown, the lower brushed portion appears clean, while the grey film can still be seen on the upper portion. The rust blisters on the carbon steel spacer shoes were also readily removed by wire brushing.

Some of the fuel elements in the sealed K West canisters were found to have a heavy white or translucent coating on them. This coating is illustrated in Figure 8 (not included), which shows the coating present on both the inner and outer elements of the fuel assembly. It was noted that the coating sometimes flaked off the surface of the element when struck by another object. About 15 % of the canisters opened in K West were found to have this heavy coating on the fuel assemblies, and always in both barrels of the canister. All of the fuel elements displaying this heavy white coating were contained in aluminum canisters.

3.2 Subsurface Examination

Two outer fuel elements from K West Basin and one outer element from K East Basin were transferred to a hot cell to examine the fuel condition underneath damaged cladding areas [Pitney . 1997c]. The elements were selected based on the apparent "peelability" of the damaged cladding and the potential for corroded fuel beneath the damaged area.

The in basin appearance of K West Element 6743U is shown in Figure 9 (not included). The fuel element had open cracks near its midsection and split open cladding with some fuel voiding at the bottom. Significant sludge trails out the bottom were noted when the fuel element was extracted from its canister for visual examination and again when it was extracted for transfer to the Single Fuel Element Canister (SFEC) for shipment to the hot cells. No sludge was seen to exit from the midsection breach during the in-basin handling.

The in-basin appearance of K East Element 5427E is shown in Figure 10 (not included). The inner element was found to be stuck under the displaced end cap of the outer element, so the full assembly was transferred to the SFEC. There was major damage apparent on both ends of the outer fuel element. Substantial sludge trailed from the assembly when it was removed from the canister, and particulate material was also released from the top end during the transfer operation. The outer element was subsequently broken in two near its cracked midsection when a flaring tool was used in the hot cell to open the top end so that a grapple could be inserted.

The in-basin appearance of K West Element 7913U is shown in Figure 11(not included). The element had an open crack near its midsection and also at the bottom. The fuel element actually had a hinged appearance at the midsection breach and appeared ready to break in two. Considerable sludge was released when the element was extracted from its canister. At the visual examination station in the basin, sludge was observed to be continuously suctioned from the bottom breach area by a nearby pump used to capture loose debris in the pit. When the fuel element was extracted a second time for transfer to the SFEC, an overhead camera showed that sludge was periodically expelled from the

midsection breach area as the element flexed during the extraction and transfer operation.

The loose cladding around damaged areas was peeled away and the underlying surface examined. Figure 12 (not included) shows the center breach area on the 6743U element after' peeling and particulate sampling. As much as possible particulate matter under the peeled region was collected using a combination of picks, scrapers, and brushes. However, it was found that for the most part the fuel under the peeled cladding was quite firm and it was difficult to remove particulate material. The largest amount obtained from any damage area (top of 5427E) is shown in Figure 13 (not included). The mass of this sample was 15.5 g.

The water from each SFEC was strained to capture any residue material that might have escaped from the fuel elements during shipping and handling. Typically. the quantity of material recovered from the SFEC was equal to or greater than that which could be scraped from damaged areas on the fuel elements after peeling away the cladding. This suggests that the cleaning procedure planned for the fuel elements prior to MCO loading should be quite effective in removing particulate material.

The total particulate material quantities recovered from all damaged areas on the fuel elements are given in the following table.

Element	Particulate Recovered (g)
KW 6743U	6.1
KE 5427E	20.3
KW 7913U	11.9

An additional element was transferred to the hot cells to recover the heavy white coating from a K West fuel element. Figure 14 (not included) shows a putty knife being used to remove the coating from the surface of the element. This technique was successful in removing the material, which typically spalled off as large flakes. Figure 15 (not included) shows the total sample recovered after scraping the full outer surface of the element. The weight of this sample was 8.5 g.

3.3 Fuel Handling, Loading, Transfer, and CVD Process

A maximum time of 30 days is used for basket queuing at the basins. Transport to CVD is assumed to require four hours; results are not sensitive to these values.

The following schedule for CVD allows for two cycles with these steps:

17.3 hours at 10 C .
25.6 hours at 50 C
14.5 hours at 75 C
4.5 hours at 25 C.

3.4 Fuel Oxidation

Fuel corrosion at K-East causes cesium release corresponding to a range from 200 to 980 grams. U per day, where the low value corresponds to well-controlled water chemistry and temperature. A mean value of 500 g U/day for the basin is chosen here, leading to an average corrosion rate for an MCO at the basin temperature of 2.5 g U/day.

The reaction of uranium with water or steam results in slightly hyperstoiciometric U02 when no oxygen is present, but also results in formation of U03 when oxygen is present. U03 is partly transformed into U03·2H2O at temperatures below about 80 C. The corrosion rate is about a factor of 20 lower when oxygen is present compared to the rate in a nearly oxygen-free environment, with the change occurring at about 10 ppm oxygen. No U03 or U03 hydrate can be formed in the absence of oxygen; the process is not thermodynamically possible. The extent of conversion of U02 to hydrated U03 can be 30% in two months at 80 C when the starting material is a finely divided powder.

Corrosion of uranium in water saturated with hydrogen and in steam occur at the same rate and are correlated by [Pearce, 1989]

$$\log K = 7.364 - 3016/T$$

for water without oxygen, and considering the factor of 20 reduction with oxygen results in

$$\log K = 6.063 - 3016/T$$

In mixture of steam and inert gases, the reaction rate decreases when the steam partial pressure is below saturation, and the applicable correlation is $\log K = 5.024 - 2401/T + 0.5 \log P$ where P is in kPa. In the above equations, T in is Kelvin and K is grams of weight gained per cm^2 per hour. Thus, the rate of formation of particulate is found by multiplying the value for K by the surface area and the molecular weight ratio of the oxide to oxygen, 270/32.

A reaction rate multiplier is typically applied to the correlations above to account for surface roughness, irradiation, and the possible presence of hydrides which have large specific surface area; and assumed values are discussed in Section 5 below.

3.5 Geometry and Miscellaneous Values

Fuel element dimensions and derived values used here are:

Outer Element: 2.42" OD, 1.70" ID, 26.1 " length, surface area 2210 cm^2

Inner Element: 1.28" OD, 0.48" ID, 26.0" length, surface area 942 cm^2 so

the total surface area of an assembly is 3152 cm^2.

4.0 TOP-LEVEL TECHNICAL BASES AND ASSUMPTIONS

Top-Level technical bases and assumptions derived logically by the expert panel from the sources presented above are listed here as the first step in formulation of particulate estimates. Detailed technical bases and assumptions are provided in Section 5 for the selection of individual equations and parameter values.

Top-Level technical bases formulated from the evidence include:

1. Particulate Locations. Based upon visual evidence, particulate loaded into an MCO is assumed to predominantly be found in the following locations:

1.1 A surface layer on cladding. Some fuel elements in K West canisters have a transparent or milky white coating; other elements from both basins have a grey coat, with the exception of one canister in K-West whose elements have a red coating.

1.2 An oxide layer on exposed uranium. Ultimately uranium corrosion is responsible for creation of particulate.

1.3 Particles adhering to uranium at fuel damage locations, either exposed or beneath the surface of cladding at the damage location.

2. Negligible Loaded Particulate Contributors. Based upon visual evidence, and anticipated fuel handling, negligible particulate will be loaded into an MCO via:

2.1 Blockages in the annulus and center void of an element.

2.2 Accumulation of non-adherent beds of particles in voids at ballooned element ends or peeled cladding on element sides. This is in contrast to tenaciously adhering particles found under mildly displaced cladding at failure locations, and represents for example particulate observed falling out of assemblies and causing notable turbid plumes during fuel handling..

3. Generated Particulate. Particulate can be generated during queuing for transfer to CVD and during CVD itself from the simple mechanism of oxidation of exposed metallic surfaces. The method of [Pajunen, 1996] is adopted and modified for this contribution. This contribution is considered because the duration of queuing can, via literature correlations and best estimates of fuel reactive area, lead to non-negligible corrosion, and temperatures during processing are high compared to those of storage in the K-basins, again leading to non-negligible production of corrosion products.

4. Character of Fuel and Scrap. Observations of fuel damage and project definitions used to assign fuel assemblies to scrap and fuel baskets in an MCO are important for assignment of parameters such as reactive surface area. Scrap and fuel character are described by:

4.1 The effective number of assemblies loaded into a scrap basket is given by the scrap basket criticality weight limit, 930 kg, and the average weight of an element, 25.07 kg, or 37.1 assemblies. [By calculation, a Mark IA scrap basket loading of 575 kg and an average weight of a Mark IA assembly of 18kg yields the equivalent of 31.9 assemblies.]

4.2 Fuel elements described as "bad" or having split cladding and missing end caps are assumed to be placed in scrap baskets [Pajunen, 1997].

4.3 An MCO contains one scrap basket and four fuel baskets.

4.4 Fuel will be cleaned by tumbling.

4.5 Handling and transport of damaged fuel elements already shipped from the K-basins has resulted in removal of large amounts of sludge as described above.

Top-level assumptions formulated from the evidence to quantify the amount of particulate are:

1. Cladding Surface Layers. The white surface layer loading is 0.01 g/cm^2 based on [Pitner, 1997], and the grey surface layer loading is 0.0006 g/cm^2 based on [Marschman, 1997]. The white layer value is used for bounding estimate, and the grey value is used for a best-estimate. Incipient formation of white layers in local spots and the presence of local reddish spots are subsumed into the bounding estimate.

2. Adherent Oxide Layer Thicknesses. A bounding oxide layer thickness is 16 microns and a best-estimate is 3 microns, the largest value found in [Marschman,. Pyecha, and Abrefah, 1997].

3. Reaction Area: Area of Adherent Oxide Layers. A bounding uranium area for the scrap basket is 6 m^2 based upon [Duncan, 1997]. A nominal value of 3 m^2 was selected based upon judgement. A low value of 1 m^2 was chosen. The bounding uranium area of 7 m^2 and a best estimate of 3 m^2 were used for fuel in four fuel baskets [Duncan, 1997].

4. Tenacious Particulate. A best-estimate for particulate on a badly damaged fuel element placed in a scrap basket is 20 g based on [Pitner, 1997], and a bounding value is a factor of ten larger, 200 g. The best-estimate for scrap is used as the bounding value for an element placed in a scrap basket.

5. Fuel Cleaning Efficacy. Cleaning of fuel is very effective, and only the most tenacious particulate as seen by [Pitner, 1997] will remain. As noted above, no particulate is observed plugging flow channels after fuel is moved. All handling experience supports this assertion. Similarly, small fuel fragments collected after fuel tumbling will be cleaned so that negligible associated sludge will be loaded into an MCO.

5.0 CALCULATION

5.1 Scope

The calculation considers materials which may be loaded into an MCO or generated in an MCO that are literally not cladding and fuel in their original form. Particulate, corrosion product, and visible layers are included in this scope for two main reasons:

1. Such materials are potentially entrainable from an MCO either during normal processing or during accidents, and/or

2. Such materials may contain chemically bound or otherwise difficult to remove water which may through thermal decomposition or radiolysis affect MCO pressure.

Thus while some observed layers may not be entrainable, they could harbor water, and some particulate may have little water, but may be entrainable; in either case these material forms are of interest here.

The scope of time for particulate generation of interest here is between MCO loading and shipment to the canister storage building for staging. This is selected because only particulate generated in this time period has the potential to chemically bind free water or water vapor, and the water inventory after CVD determines the potential for over pressurization in shipping and staging.

5.2 Corrosion Product and Particulate Locations and Geometry

Corrosion product and particulate are considered in the following locations:

1. Visible Layers on Cladding. Two kinds of layers are considered here: A white or clear layer found in some K -West canisters'(Figure 14) and a grey layer found on all other fuel surfaces (Figures 2 and 7). The white layer appears to resist brush cleaning but is brittle and may fall off during tumbling; the grey layer is easily removed by brush cleaning but is also very thin. Since either of these layers are observed on all cladding surfaces, the associated mass is given by:

$$M_{layer} = A_{assy} N_{assy} \ (m/A_{layer})$$

where

M_{layer} = *Layer mass, kg*

A_{assy} = *Assembly area, cm²*

$N_{assy,}$ = *Number of assemblies, fuel plus scrap.*

(m/A_{layer}) = *Mass per unit area .of layer, kg / cm²*

The area of an assembly is given by geometric data presented in Section 4.5:

$A_{assy,} = 3152\ cm^2$

'The number of assemblies is the sum of fuel assemblies for an MCO with four fuel baskets

$N_{assy,fuel} = 4 * 54 = 216.$

The number of equivalent assemblies in a scrap basket is determined by the scrap basket weight limit divided by the average weight of an assembly:

$$N_{assy,scrap} = \frac{930\ kg/scrap\ basket}{25.07\ kg/assembly} = 37.1$$

The mass per unit area of the layer is selected by judgement.

2. Oxide Film on Scrap. All exposed uranium metal should have some oxide film because some corrosion must have occurred; an example is shown in Figure 16 (not included). Scrap is considered separately from fuel because the area for scrap is deduced differently than the area for fuel. The mass of an oxide layer is given by:

$$M_{ox,scrap} = A_{ox,scrap} (m/A_{ox})$$

where

$M_{ox,sasp}$ = *Oxide mass on scrap, kg . .*

$A_{ox,scrap}$ = *Oxidation area of scrap,* m^2

$(m/A_{ox,scrap})$ = *Mass per unit area of oxide layer,* kg/m^2

The area $A_{ox,scrap}$ is chosen by expert judgement as discussed below. The mass per unit area is further decomposed as the product of layer thickness and density:

$$(m/A_{ox}) = \rho_{layer} \delta_{layer}$$

ρ_{layer} = *Layer density, kg/m3*

δ_{layer} = *Layer thickness, m ..*

where both the layer density and thickness are chosen by judgement.

3. Oxide Layer on Fuel. Fuel in fuel baskets has areas with cladding failure and the exposed metal must have some oxide layer. The mass of this oxide is given by the same formula as for the scrap basket oxide: .

$$M_{ox,fuel} = A_{ox,fuel} (m/A_{ox})$$

Here the exposed area is chosen by expert judgement considering different factors than for the scrap basket, but the mass per unit area is chosen in the same way as for that in the scrap basket.

4. Particulate on Scrap. Particulate is found to be associated with failed areas, both on the failed area and locally underneath adjacent cladding that has become unbonded from the fuel (Figures 12 and 13 - not included). Particulate associated with fuel and scrap are considered separately because the damage of scrap is more extensive and thus separate judgements are appropriate. The mass of particulate associated with scrap is given by:

$$M_{par,scrap} = N_{assy,scrap}\, m_{par,assy}$$

where

$M_{par,scrap}$ = *Particulate mass on scrap, kg*

$m_{par,assy}$ = *Mass of particulate per assembly, kg*

The mass of particulate per assembly in a scrap basket is determined by judgement.

5. Particulate on Fuel. Similarly, particulate may be found on the failed area and locally underneath adjacent cladding on fuel assemblies in fuel baskets, and the total mass is:

$$M_{par,fuel} = N_{assy,fuel}\, m_{par,assy}$$

The mass of particulate per assembly in a fuel basket is determined by judgement.

6. Generated Particulate on Scrap. Particulate can be generated by oxidation after the MCO is loaded, and prior to shipment from CVD to staging, and it is possible that some portion of this particulate may chemically bind with free water or its vapor. The mass of generated particulate is found using the reaction rates and time information described above in Sections 4.3 and 4.4:

$$M_{gen,scrap} = F\, A_{ox,scrap}\, (270/32)\, \Sigma\, K_i\, t_i$$

where

$M_{gen,scrap}$ = *Mass of generated particulate, scrap*

F = *Rate law multiplier.*

K_i = *Reaction rate constant, mg U/cm^2 hr*

t_i = *Time interval, hr*

The rate law multiplier is chosen as described in Section 5.3.

7. Generated Particulate on Fuel. Particulate generated in fuel baskets is similarly found by:

$$M_{gen,fuel} = F\, A_{ox,fuel}\, (270/32)\, \mathrm{S}\, K_i\, t_i.$$

5.3 Expert Judgment of Parameter Values

Low, best-estimate, and bounding values are described here for the parameters listed in the preceding equations. Three values are chosen because: (1) By considering more than just a bounding value, experts are forced to think about realistic expectations and variability in the quantities and therefore the estimates are perceived to be more physically based; (2) The reason for selection of the bounding value is better understood in relation to choice of the best estimate: (3) It is easier for persons using this work to understand these estimates than to intuitively grasp the meaning of more complete probability density functions (pdf's); and (4) It is perceived that pdf's with greater detail are simple not justified by the data. Definition of these values are:

Low: The contents of an MCO could occasionally be characterized by this value.

Best-estimate: The contents of an MCO are well-represented by this value, although there is expected to be local variation within an MCO.

Bounding: The contents of an MCO are not reasonably expected to exceed a particulate quantity derived by using the bounding parameter value. Locally some fuel or scrap could have parameter values higher than given by the bounding value, but it would be unreasonable and inappropriate to assign such a value to calculate the contents of an entire MCO.

TABLE 5-1. SELECTION OF PARAMETER VALUES

Mass per unit area of layer,

	(m/A_{layer})	*Mg/cm2*	
Bounding	10 mg/cm^2		The clear or white layer found on cladding in some K-West canisters is about an order of magnitude thicker than that of the grey layer found on other cladding surfaces. The chosen value bounds the largest observation, 8 mg/cm^2 [Pitner, 1997].
Best-est. .	0.6 mg/cm^2		Most fuel elements have the grey layer with an estimated mass per area of 0.55 mg/cm^2 [Marschman and Abrefah, 1997].
Low	0.6 mg/cm2		No reason and little consequence to selecting different low value than the best-estimate.

Oxidation area of scrap,

	$A_{ox,scrap}$	cm^2	
Bounding	60,000	cm^2	From [Duncan, 1997]
Best-est.	30,000	cm^2	Selected as one half of bounding.
Low	10,000	cm^2	Scrap can consist of damaged assemblies with few "fine" fragments by its definition.

Oxidation area of fuel,

	$A_{ox,fuel}$	cm^2	
Bounding	70,000 cm^2		From [Duncan, 1997]
Best-est.	3000	cm^2	From [Duncan, 1997]
Low	1000	cm^2	Arbitrarily chosen to be lower than the best-estimate; the actual value used here is not significant:

Density of oxide layer,

	ρ_{layer}	g / cm^3
All	10 *g/cm^3*	Approximate density of uranium dioxide; these oxide layers have lower than theoretical density. Hydrated layers would have a theoretical density half this value, but this can be considered explicitly for water content evaluations.

Thickness of oxide layer, m

	δ_{layer}	
Bounding	16 *μm*	Equal to the largest observed thickness for an oxide layer in the process of-sloughing off [Abrefah et al, 1996] after simulated conditioning; greater than the largest reported value of 3 *μm* for fuel actually in the K basins [Abrefah et al, 1997]. It is well known that the layer tends to slough off due to the factor of two density ratio between oxide and metal, leading to the linear oxidation rate law.
Best-est.	3 *μm*	Reported observation [Abrefah et al, 1997].
Low	3 *μm*	No compelling reason to consider a lower value.

Mass of particulate per assembly in scrap basket, g

	$M_{par,assy}$	
Bounding	200 g	An order of magnitude greater than the largest observed particulate mass on an element, 20 g [Pitner, 1997c]; the particular assembly selected was among the worst observed. The value is about 1 % of the mass of an assembly and it is deemed incredible that such a mass of particles would survive cleaning. The value is equivalent to a particle layer of about 1 mm if 10% of the assembly area were damaged, indicating that cladding would have to be badly deformed to retain such particles, lending credence to the cleaning assumption. This value accounts for contributions of both inner and outer elements because it is so large, and the damage fraction of inner elements is small.

Best-est.	20 g	This is the worst measured value, taken from an element destined for a scrap basket, so it is at least representative of scrap. Most inners are undamaged, so using the worst measured value for an outer element therefore accounts for contributions from inners
Low	6 g	Lowest measured value.

Mass of particulate per assembly in fuel basket, g

$m_{par,assy}$

Bounding	20 g	Fuel cannot look worse than scrap, so a best-estimate value for scrap must bound that for fuel, and as above the value for an element is appropriately extended to cover an assembly:
Best-est.	7.6 g	38 % of fuel elements are damaged, hence 38 % is applied to the bounding figure to create a best estimate.
Low	1.3 g	Most fuel assemblies are undamaged. An MCO with one canister pair containing damaged fuel would have a value 141216, or 6.5% of the bounding figure. The value could well be zero, but is not considered important.

Reaction rate law multiplier (dimensionless)

Bounding	10	Value from SNF Databook [Duncan, 1997].
Best-est.	3	Based on surface roughness [Johnson and Pitner, 1995] .
Low	3	K-Basins fuel judged more reactive than fresh metal.

5.4 Resulting Particulate Mass Values

Generated particulate values are evaluated in Table 5-2, and other values given in Table 5-1 are employed in equations developed above as shown in Table 5-3. Results are rounded to yield the overall values summarized in Table 5-4. Note that in some cases, products of judged parameters are used. In

these cases, for simplicity, values at the same level are consistently carried through to yield the table result. This means that the frequency associated with the bounding value in the table is less than that associated with the input parameters, and in such cases the bounding values are more conservative than intended.

TABLE 5-2. CALCULATION OF GENERATED PARTICULATE PER UNIT AREA, UNIT RATE LAW FACTOR.

OPERATION	TEMPERATURE (Celsius)	DURATION MASS (hours)	GENERATED SPECIFIC micrograms/cm^2 With 0_2	Without 0_2
Queuing & Loading	10	720	153.6	3092.
Transfer to CVD	10	4	0.85	17.2
Connection	10	17.3	3.7	74.3
Drying + Test	50	25.6	n/a	2914.
Shipping Test	75	14.5	n/a	10047.
Loading	25	4.5	n/a	60.9
TOTAL - 2 Cycles			26140	29170

TABLE 5-3. CALCULATION OF PARTICULATE MASS VALUES

1. Cladding Surface Film, Scrap

Bounding: 3150 cm^2/assy * 37.1 assy * 10^{-5} kg/cm^2 = 1.2 kg
Best-Est: 3150 cm^2/assy * 37.1 assy * $6*10^{-7}$ kg/cm^2 = 0.1 kg
Low: 3150 cm^2/assy * 37.1 assy * $6* 10^{-7}$ kg/cm^2 = 0.1 kg

2. Cladding Surface Film, Fuel

Bounding: 3150 cm^2/assy * 216 assy * 10^{-5} kg/cm^2 = 6.4 kg
Best-Est: 3150 cm^2/assy * 216 assy * $6*10^{-7}$ kg/cm^2 = 0.5 kg
Low: 3150 cm^2/assy * 216 assy * $6*10^{-7}$ kg/cm^2 = 0.5 kg

3. Oxide Film, Scrap

Bounding: 60,000 cm^2 * 16* 10^{-4} cm * 0.01 kg/cm^3 = 0.96 kg
Best-Est: 30,000 cm^2 * 3*10^{-4} cm * 0.01 kg/cm^3 = 0.09 kg
Low: 10,000 cm^2 * 3*10^{-4} cm * 0.01 kg/cm^3 = 0.03 kg

4. Oxide Film, Fuel

Bounding: 70,000 cm^2 * 16*10^{-4} cm * 0.01 kg/cm^3 = 1.12 kg
Best-Est: 3000 cm^2 * 3 * 10^{-4} cm * 0.01 kg/cm^3 = 0.009 kg
Low: 1000 cm^2 * 3*10^{-4} cm * 0.01 kg/cm^3 = 0.003 kg

5. Particulate on Scrap

Bounding: 0.2 kg/assy * 37.1 assy = 7.4 kg
Best-Est.: 0.02 kg/assy * 37.1 assy = 0.74 kg
Low: 0.006 kg/assy * 37.1 assy = 0.22 kg

6. Particulate on Fuel

Bounding: 0.02 kg/assy * 216 assy = 4.32 kg
Best-Est: 0.0076 kg/assy * 216 assy = 1.64 kg
Low: 0.0013 kg/assy * 216 assy = 0.28 kg

7. Generated Particulate, Scrap

Bounding: 29.17*10^{-6} kg/cm^2 * 100,000 cm^2 * 10 multiplier = 29.17 kg
Best-Est.: 29.17*10^{-6} kg/cm^2 * 50,000 cm^2 * 3 multiplier = 4.37 kg
Low: 29.17*10^{-6} kg/cm^2 * 10,000 cm^2 * 3 multiplier = 0.875 kg

8. Generated Particulate, Fuel

Bounding: 29.17*10^{-6} kg/cm^2 * 30,000 cm^2 * 10 multiplier = 8.75 kg
Best-Est.: 29.17*10^{-6} kg/cm^2 * 3000 cm^2 * 3 multiplier = 0.263 kg
Low: 29.17*10^{-6} kg/cm^2 * 1000 cm^2 * 3 multiplier = 0.087 kg

TABLE 5-4. SUMMARY OF DERIVED PARTICULATE MASS VALUES (All values in kg)

SOURCE	BOUNDING	BEST-ESTIMATE	LOW
1. Cladding surface film scrap	6.4	0.4	0.4
2. Cladding surface film fuel	1.2	0.1.	0.1
3. Oxide film, scrap	0.96	0.09	0.03
4. Oxide film, fuel	1.12	0.009	< 0.003
5. Particulate on scrap	7.4	0.74	0.22
6. Particulate on fuel	4.32	1.64	< 0.28
7. Generated particulate, scrap	29.2	4.4	0.9
8. Generated particulate, fuel	8.75	0.263	< 0.087
TOTAL	59.75	7.7	2.02
ROUNDED TOTAL	60	8	2

5.5 Remarks on Use of Particulate Inventory Values

Values in the above table should be used with awareness of their source.

With respect to MCO pressurization after CVD, the water content of each table entry must be separately evaluated. The bounding cladding layer composition is likely to be aluminum hydroxide, while that of the grey layer is uranium peroxide tetrahydrate. The density used for the oxide layer is that of the oxide in the table above, but if this layer is assumed hydrated; then half the value should be used. The great majority of the generated particulate simply cannot be hydrated because it is generated in an oxygen-free environment.

With respect to particulate available for entrainment from an MCO, it is likely that little of the loaded inventory is entrainable, but generated particulate implies sloughing of the oxide layer and therefore this contribution is likely to be entrainable.

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Appendix D

Excerpted Abstracts and Summaries from Various Criticality Safety Analyses for N Reactor Fuels

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Excerpted Abstracts and Summaries from Various Criticality Safety Analyses for N Reactor Fuels

The following sections capture either abstract and/or summary information of previous criticality analyses done with N Reactor fuels in various storage and package configurations.

D-1 Criticality Safety Evaluation of Irradiated N Reactor MK IA and Mk IV Fuel in Unconstrained Mark I and Mark II Fuel Canisters (WHC-SD-NR-CSER-009, Rev 0), 1993

The analysis examined the technical details covering the safety classification of irradiated fuel canister storage racks in the 100K area irradiated fuel storage basins. Computer calculations showed that, for all credible off-normal situations, the criticality safety limit ($k_{eff} + 2\delta < 0.98$) is not violated due to the absence of fuel storage racks and the complete loss of spacing control alone. The following findings with MCNP code calculations were summarized as follows:

- If the racks are removed and canister stay in place, no reactivity change is observed.
- If canisters are moved together, reactivity increases.
  - For MKIV and MKIA assemblies in stainless steel canisters or aluminum canisters, with or without uranium oxide buildup, the criticality limit ($k_{eff} + 2\delta < 0.98$) is not exceeded
- If canisters are broken apart and close packed in a hexagonal geometry, the reactivity increases.
  - For MKIV and MKIA fuel assemblies in stainless steel canisters, and MKIV in aluminum canisters, this does not exceed the criticality limit ($k_{eff} + 2\delta < 0.98$)/
  - For MKIA fuel in aluminum canisters, this exceeds the criticality limit ($k_{eff} + 2\delta < 0.98$). Because loss of canister integrity is also necessary before the criticality limit is violated, the complete loss of spacing control by itself does not violate the criticality safety limit.
- If canisters are tipped, the k_{eff} is less than for standing canisters. The tipped configuration is critically safe for both types of fuel and both canister materials
- Single canister fuel spill off-normal assumed fuel elements around existing canisters. The calculations for spilling all the fuel out of a single canister indicate the following:
  - For MKIV and MKIA fuel in stainless steel or aluminum canisters, the criticality limit ($k_{eff} + 2\delta < 0.98$) is not exceeded under any condition.
- Multiple canister fuel spill off-normal assumed one canister of fuel elements around each existing canister. The calculations for a large number of canister drops indicate the following:
  - For MKIV fuel in stainless steel or aluminum canisters, the criticality limit ($k_{eff} + 2\delta < 0.98$) is not exceeded.

- MKIA fuel without storage racks in stainless steel or aluminum canisters will exceed the criticality limit ($k_{eff} + 2\delta < 0.98$). Because multiple fuel canister spills are also necessary before the criticality limit is violated, the complete loss of spacing control by itself does not violate the criticality safety limit.

D-2 Use of a K_{eff} Subcritical Limit of 0.98 (WHC-SD-SQA-CSA-20330, Rev. 0, 1991)

This report addressed Audit Finding No. CS.2-1 by Tiger Team for Westinghouse Hanford Facilities. A review is provided of the use of a subcritical limit on k_{eff} of 0.98 in operations with lowly enriched uranium. This limit is used when directly relatable experimental critical mass data is available which can be extrapolated to specific situations using a validated computational technique.

Use of a subcritical limit of 0.98 for fuel handling and storage limits, based upon individual, isolated batches of N Reactor fuel, is found to be justified. Although use of a subcritical limit of 0.98 was specifically justified in the analysis of the PUREX dissolve, based upon the criteria in effect at that time, this results in less conservative limits than those normally used for processing. It is recommended that PUREX dissolver limits be reestablished using a subcritical limit of 0.95.

D-3 K Basin Criticality Evaluation for Irradiated Fuel Canisters in Sludge (WHC-SD-NR-CSER-001, Rev. 0, 1992)

The criticality evaluations consider a series of accident scenarios in the KE basin (uranium enrichment of 0.95 Wt% or less) with various accumulations of sludge. The sludge is due to fuel oxidation of damaged fuel, fuel segregation activities, and proposed fuel repackaging.

Basin sludge or repackaging sludge was modeled as wet UO_2 powder with a density between 1 and about 6 g/cm^3 UO_2. Wet UO_2 powder was also imbedded with 3.175 cm (1.25-in.) diameter uranium metal rods to simulate broken fuel pieces in the most reactive geometry. Different scenarios considered the UO_2 around or underneath MKIV fuel encapsulation canisters, and the uranium metal rodded or unrodded UO_2 powder inside and/or outside fuel encapsulation canisters.

The analyses were performed with the WIMS-E and MCNP codes. Some results from these calculations are:

- The minimum slab (4 g/cm^3 with imbedded uranium metal rods) thickness is 125.36 cm (49.35-iches) for k_{eff} of 0.98 as calculated by WIMS-E.
- Approximately a 90 mk reduction in k_∞ is due to the use of stainless steel fuel encapsulation canisters as compared to the use of aluminum fuel encapsulation canisters.
- Canisters containing MKIV fuel with water moderation were more reactive than canister containing wet UO_2 powder.
- Canisters containing MKIV fuel with water moderation were only slightly less reactive (by 18.55 ± 5.71 mk for 3 g/cm^3 UO_2 with uranium rods) than canisters containing wet UO_2

powder with uranium metal rods. This canister containing rodded UO_2 would contain 242.55 kg (534.73 lbs) of uranium as compared to 333.56 kg (735.38 lbs) for MKIV fuel.

- UO_2 powder accumulation around canisters was more reactive than UO_2 accumulation underneath the fuel encapsulation canisters.

- UO_2 powder accumulation of 3 g/cm^3 UO_2, and depths to 90 cm outside the fuel encapsulation canisters had MCNP calculated k_∞ below 0.96 for both stainless steel and aluminum canisters.

- Uranium metal rods in UO_2 powder accumulation for 3 g/cm^3 UO_2 and depths to 90 cm outside the fuel encapsulation canisters had MCNP calculated k_∞ below 0.98 for both stainless steel and aluminum canisters.

In all cases with Mark II stainless steel canisters, the MCNP calculated k_∞ remained safely below the criticality safety limit of $k_{eff} = 0.98$, even for fuel storage configurations in the KE Basins considering the presence of basin sludge modeled as wet UO_2 powder with and without imbedded 3.175 cm (1.25-in.) diameter uranium rods. The potential for criticality is extremely remote.

D-4 Criticality Safety Evaluation Report for Spent Nuclear Fuel Processing and Storage Facilities (HNF-SD-SNF-CSER-005, Rev. 3, 1997) [Note: the following information has been superceded by information contained in Ref. 12 (HNF-SD-SNF-CSER-005, Rev. 5B)]

1.1 Introduction Spent N Reactor fuel will be unloaded out of the existing canisters in Both K-East and K-West Basins, and loaded into MCO containers with specially built baskets containing either 54 Mark IV or 48 Mark IA fuel assemblies. MCO containers holding Mark IV fuel material will contain five tiers , or baskets, of fuel. Because of the shorter length of all but eleven of the Mark IA fuel assemblies, MCO containers holding Mark IA fuel will contain six fuel baskets. The Mark IA basket will have a central stainless steel insert of 6-inch diameter, schedule 80 pipe to keep fuel material out of the central area of the basket for criticality control. [NOTE: this central pipe has since been changed to a machined barstock component with a small bored hole to better exclude uranium; see Figure 4-6 and Ref. 4 for new design details] This central insert excludes fuel elements and uranium scrap material from the central region of the basket, which is the region of highest neutron importance. Exclusion of scrap from this central region helps maintain k_{eff} less than the criticality criterion of 0.95. . .

A fraction of the fuel inventory is damaged or corroded and does not closely resemble fuel elements. This fissionable material will be referred to as scrap. Scrap is material that came from stored, spent fuel assemblies, but also may contain corrosion products. Scrap models used in this report are optimally configured and therefore bound all forms of N Reactor uranium scrap. Because of the heterogeneous lattice effects, optimal scrap material is more reactive than is homogeneous sludge material; optimal scrap therefore bounds sludge and will be used in the analysis rather than sludge. The older SPR with enrichments greater than natural will be loaded into scrap baskets in the top and bottom locations of the Mark IA MCOs. The SPR fuel has enrichments up to that of N Reactor fuel, 1.25 wt%, and can be bounded by N Reactor fuel analyses.

1.2 Summary of Analysis Results Criticality criteria require that the k_{eff} of the activities with (and storage of) the MCOs in the CSB, and activities with the MCOs in their transport casks outside the K basins, be less than 0.95 under both normal conditions and credible off-normal conditions, including uncertainties. Existing criticality criteria for K Basins pool operations allow an upper limitation k_{eff} of 0.98. Both these limits are nominal and actually lower when calculational uncertainties and code biases are considered. Analysis has shown that the k_{eff} of the MCO/cask package under normal conditions will be below 0.95 by a substantial degree. Dry MCO/cask packages will have k_{eff}s less than 0.4. MCOs flooded with water and loaded with intact N Reactor fuel assemblies will have k_{eff} less than 0.90. . . .

The MCO container, holding dry fuel material, cannot be made critical under any conditions. The only criticality concern is with water moderation internal to the MCO. Both Mark IV and Mark IA MCO containers are below $k_{eff} = 0.89$ for flooded intact fuel loadings. Loading scrap in the top and bottom tiers with intact fuel in the other tiers has a k_{eff} less than 0.90, which is the normally allowed arrangement of packing scrap in an MCO. However, if the second basket is misplaced from the top to the second from the bottom, so that two scrap baskets occupy the bottom two tiers, the k_{eff} increases to less that 0.91. The 0.95 wt% scrap, when loaded into Mark IV scrap baskets, tends to be more limiting because of the absence of the safety class structure that excluded scrap from the center of the basket, as is the case when loading Mark IA scrap into Mark IA baskets, which contain the 6-inch stainless steel insert.

The inside diameter of the MCO provides geometry control and ensures that k_{eff} is always less than 1.0. Additional constraints and limits ensure that the k_{eff} is less than 0.95. For accidents, two independent, concurrent, and highly unlikely incidents must occur before the k_{eff} is allowed to exceed 0.95. This is the two contingency principle of criticality safety.

The 100g drop is the limiting design basis accident (DBA). . . .A conservative packing fraction of 0.4 was used . . . a more realistic random packing fraction for uniform-size spheres has been shown to equal approximately 0.64 (Berryman 1983) . . . and would drive the system far less reactive than 0.40 . . . bottom plates of the baskets are assumed to lose their structural attachment to the central tubular insert, and the rubblized fuel and cladding filled baskets fall on top of one another . . . this limiting accident condition, k_{eff} at the upper 95% confidence level for the Mark IV loaded MCO is less the 0.94; this one result is the most limiting and requires closer attention to demonstrate compliance with criticality safety limits.

. . . The drop accident is assumed to completely rubblize the fuel. If this drop accident occurs during transport of the MCO container . . .when the MCO is fully flooded, the MCO could exceed the $k_{eff} < 0.95$ criterion for a transient phase during rebound . . .The drop accident must both agglomerate smaller fuel fragments, which result from the impact, into larger particles and disperse these particles into optimal spacing during the same rebound. This scenario is considered incredible. The calculated k_{eff} for this scenario does not exceed 0.98. The end-state does satisfy $k_{eff} < 0.95$ once the rubble has compacted down to a packing fraction of 0.40 due to gravity from fuel crushing . . . .

Normally fuel types are to be segregated by enrichment. However, placing Mark IV fuel assemblies, components, and scrap together in baskets designed for Mark IA fuel (with central criticality-control insert) is allowed, because of the lower unit reactivity of Mark IV fuel.

[Note: any limiting conditions identified in CSER-005, Rev. 3 have been supplanted by the following assumptions taken from HNF-SD-SNF-CSER-005, Rev 5B.]

Because of their length, the twelve 26.1-in. long Mark 1A assemblies in the K West Basis will be loaded into Mark IV baskets [Ref. 13]. Otherwise, fuel and scrap with an enrichment greater than 0.95 wt% shall only be loaded into Mark 1A fuel and scrap baskets.

Conservatisms in the Analysis

Many conservatisms have been built into this analysis.

- The safety limit is 0.95. The margin of safety provided by using the 0.95 limit for the low-enriched uranium metal fuel in the K Basins is far greater than the margin of safety provided by using the same limit for the greater enrichment of commercial fuels of which the limit is based.
- It is assumed that the baskets completely fail in an accident.
- The reduction of reactivity resulting from fuel burnup, fission products in the spent fuel, and other nonfissionable material introduced as contaminants in the K Basin sludge and fuel debris, is not included in the analyses (although burnup effects are discussed in Appendix D).
- The scrap is completely optimized (i.e. optimum particle size and optimum water-to-fuel ratio).
- The fuel rubblizes to optimized scrap in a drop accident even though this is judged to be incredible.
- The most reactive loading configuration of the MCO fuel baskets is used.

Analysis Assumptions

The analyses performed to evaluate the acceptability of the normal and accident conditions for the MCO were based on the assumptions defined below.

Assumption 1 The MCO is either always in a cask, loaded into the MCO handling machine (MHM), in the CSB storage tubes, or in the CSB sampling/weld station when flooded with water. Note that flooded MCOs in the MHM, CSB storage tubes, or CSB sampling/weld station were analyzed as a contingency.

Assumption 2 Mark IA fuel or scrap is normally loaded into Mark IA fuel or scrap baskets, which have the center post that serves as a criticality feature to exclude fissile material from the center of the basket. The only exceptions to this are for the 26.1-in. long Mark IA fuel assemblies, which will be loaded into Mark IV baskets.

Assumption 3 The credible misloading scenario for Mark IA fuel being inadvertently loaded into a Mark IV basket is 14 Mark IA fuel assemblies. For scrap this is equivalent to 155 kg, the mass of 14 Mark IA outer elements, with an enrichment of 1.25 wt% ^{235}U.

Assumption 4 Following the drop of a Mark IV MCO, the basket base plates remain intact. These plates are not safety class; however their mass is conserved in the models.

Assumption 5 The misload for Mark IV scrap baskets includes the center pipe.