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Criticality Safety Evaluation Report for MKIA Fuel Pertaining to Consolidating Fuel Storage

Prepared for the U.S. Department of Energy
Office of Environmental Restoration and
Waste Management

Westinghouse
Hanford Company Richland, Washington

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7. Abstract Irradiated fuel criticality calculations are performed for MKIA fuel, 1.25 wt% ²³⁵U, and 1.15 wt% ²³⁵U fuel pieces and solutions. Comparisons are made between WIMS and MCNP. WIMS and MCNP calculations are documented.		
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EXECUTIVE SUMMARY

N Reactor spent fuel is presently stored in canisters in the 105-KE and 105-KW fuel storage-basins. It had been proposed that all N Reactor spent fuel be consolidated into only one fuel-storage basin by stacking canisters and changing the spacing of the 105-KE/105-KW storage-basin array. The criticality concerns of spent fuel consolidation form the basis for the MCNP and WIMS-E criticality calculations in this supporting document.

Double or triple stacking of stainless steel canisters containing irradiated fuel with the canister spacing currently existing in either the 105-KE or 105-KW Basin or with the proposed consolidated fuel spacing is within criticality safety limits.

Because of a hypothesized fuel transfer train accident involving the west bay, enrichment is limited to 0.95-wt% ^{235}U (MKIV fuel), and no fuel above a single layer, such as hanging fuel, is allowed in the west bay. This document did not perform analyses to relax either of these two requirements in the west bay.

Under normal conditions with aluminum canisters (no fissionable material between canisters), calculated k_{∞} remained below the criticality safety limit. The only criticality limitation involves introduction of fissile material between canisters in the storage array. Fuel spilled out of canisters into the space between canisters or an unrestrained accumulation of fissile material (sludge) between canisters requires administrative control.

The limit for sludge accumulation depth between canisters is set at 9.525 cm (3.75 in.). This sludge accumulation limit allows for a layer of optimum reactive sludge to be doubled before the slab geometry limiting value $k_{\text{eff}} = 0.97$ is approached.

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LIST OF TERMS

CSER	criticality safety evaluation report
H/U	hydrogen-to-uranium atom ratio
ID	inner diameter
OD	outer diameter

GLOSSARY

k is the neutron multiplication factor and equals the neutron production rate divided by the neutron loss rate.

mk is a unit of nuclear reactivity. The reactivity difference between two values of k_{∞} is equal to

$$\Delta\rho = 1,000 \frac{(k_2 - k_1)}{k_2 k_1}$$

in units of mk .

MKIA is a low-enriched uranium metal fuel used in N Reactor. It is arranged in a tube-in-tube geometry with an inner tube composed of 0.947-wt% ²³⁵U-enriched uranium metal and an outer tube composed of 1.25-wt% ²³⁵U-enriched uranium metal.

MKIV is a low-enriched uranium metal fuel that was the base loading for N Reactor. It is arranged in a tube-in-tube geometry; both the inner tube and the outer tube are enriched to 0.947-wt% ²³⁵U.

Mark 0 is an open aluminum fuel-storage canister.

Mark I is a closed fuel-encapsulation canister, which may be either aluminum or stainless steel. (The Mark I canisters had graphite gaskets for sealing, which were not compatible with the plutonium-uranium extraction processes and could not be used for fuel shipments.)

Mark II is a closed stainless steel canister.

CRITICALITY SAFETY EVALUATION REPORT FOR MKIA FUEL PERTAINING TO CONSOLIDATING FUEL STORAGE

1.0 INTRODUCTION

It has been proposed that irradiated N Reactor fuel from the 105-KE and 105-KW Basins be consolidated in the 105-KW Basin only (ICF 1994). This represents a change in the storage of fissionable material, and according to the *Nuclear Criticality Safety Manual* (WHC-CM-4-29), a criticality safety evaluation report is required when a change in method of storage or transport is planned. The proposed change for the canister-array spacing and triple stacking of irradiated N Reactor fuel-storage canisters does represent such a change in the method of storage. This document fulfills the requirement for a criticality safety evaluation report for this change.

The scope of this evaluation is irradiated N Reactor fuel elements stored in canisters under water. This scope also considers changes in N Reactor irradiated fuel elements during an indefinite fuel-storage time: that is, the dissolution of the fuel elements themselves into uranium oxide or breakage of the elements into more reactive uranium rods that may be inside or between fuel-storage canisters.

N Reactor spent fuel is presently (August 1994) stored in canisters in the 105-KE and 105-KW fuel-storage basins. Consolidation of all N Reactor spent fuel into only one fuel-storage basin would require stacking canisters, thereby increasing the amount of fuel stored per unit floor area. The criticality concerns associated with compacted storage of spent fuel and with long-term basin storage in canisters forms the basis for the criticality calculations discussed in this report. This work extends previous calculations regarding K Basin criticality.

N Reactor fuel is composed of low-enriched uranium metal, arranged in a tube-in-tube geometry. The base loading for N Reactor was MKIV fuel, which contained an inner tube and an outer tube, both enriched to 0.947-wt% ^{235}U . The MKIA fuel also had a 0.947-wt% ^{235}U inner tube but had an outer tube composed of 1.25-wt% ^{235}U -enriched uranium metal. K Basin fuel-canister repackaging operations require that a criticality safety evaluation report be prepared to address criticality issues regarding 1.25-wt% ^{235}U -enriched N Reactor fuel. The repackaging operations will involve handling of both MKIV and MKIA fuel assemblies inside both aluminum and stainless steel canisters. Many of these spent-fuel assemblies suffered mechanical damage during reactor discharge operations; cracks allowed uranium metal to be exposed to basin water and much of this fuel is now badly corroded and in an advanced state of dissolution. Parametric analyses reported in this document characterize the criticality effects of partially dissolved fuel assemblies.

Extensive criticality calculations have been performed to produce critical and safe masses for N Reactor MKIV and MKIA fuel, billets, solution, scrap, and components. These results have been documented in *Criticality Safety Evaluation Report for 300 Area Ne Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-CSER-010 (Schwinkendorf 1994a).

This document summarizes some of the previous results and reports more recent results for the dependence of critical mass on exposure (fuel burnup). Previous analyses have been performed using the MCNP code (Briesmeister 1986), specifically on canisters containing MKIV and MKIA fuel, under a variety of normal and accident conditions (Wittekind 1991b, 1992, 1993). *N Reactor Spent Fuel Compacted Storage Criticality Calculations* (Wittekind 1991b) also established that the axial (vertical) separation of fissionable material by storage racks provides sufficient neutronic isolation that interaction between stacked layers of canisters does not occur. This document reports results of criticality analyses on models of progressive dissolution of fuel assemblies and interspersed moderation. The WIMS-E code (Gubbins et al. 1982) was used for scoping calculations. MCNP was used to analyze the model determined to be worst-case by WIMS-E as well as a sampling of several other scenarios to verify the trends predicted by WIMS-E.

Because of the hypothesized fuel transfer train accident involving the west bay, enrichment is limited to 0.95-wt% ^{235}U (MKIV fuel) and no fuel above a single layer, such as hanging fuel, is allowed in the west bay. This document did not perform analyses to relax either of these two requirements in the west bay.

2.0 FACILITY DESCRIPTION

N Reactor fuel dimensions are listed in Table 2-1. These elements are stored in canisters with seven assemblies per canister tube and two canister tubes per canister. K Basin canister dimensions are listed in Table 2-2. An engineered railing system on the floor limits fuel-storage canisters to one canister per canister-storage location in the fuel-storage array. The present irradiated-fuel canister-storage location is 50.24 cm (19.78 in.) by 26.99 cm (10.63 in.) (Drawings H-1-34762, H-1-34763, and H-1-34764). The present storage configuration is single stacked. The present fuel-storage canister array is shown in Figure 2-1.

The proposed compacted fuel-storage array would increase the demensions of each canister-storage location to 53.34 cm (21 in.) by 31.75 cm (12.5 in.) (ICF 1994). The proposed configuration would be triple stacked with no (0.0-cm [0.0-in.]) vertical spacing between canisters for option A or with 4.445-cm (1.75-in.) vertical spacing for option B. The proposed fuel-storage stacked-canister array is shown in Figure 2-2.

The number of irradiated fuel-storage canisters that can be lifted out of position and into motion is three per basin bay. There is no proposed change to this limitation.

The total number of canisters stored in basin 105-KE is 3,668, and the total number of canisters stored in basin 105-KW is 3,821 (Wittekind 1991a). The total number of possible floor area canister locations in basin 105-KE is 5,012; the total number of possible floor area canister locations in basin 105-KW is 5,186. The difference reflects 174 locations available in the 105-KW discharge chute. The 105-KE discharge chute was used for a fuel segregation program, so these canister storage racks were removed (Drawing H-1-34771). These discharge chute locations are being isolated from the main basin because of potential problems with seismically-induced leaks, so they will not be available as canister storage locations.

There are three types of fuel-storage canisters in use: Mark 0 open aluminum canisters (Drawing H-1-36935), Mark I fuel-encapsulation (closed) canisters (Drawing H-1-42793), which may be either aluminum or stainless steel, and Mark II closed stainless steel canisters (Drawing H-1-46215). The Mark I canisters had graphite gaskets for sealing, which were not compatible with the plutonium-uranium extraction processes and could not be used for fuel shipments. The total number of canisters of different types are listed below. The inventory values were identified in *Irradiated N Reactor Fuel Consolidated Storage Shielding Calculations*, Appendix D (Wittekind 1991a).

Basin	Mark 0	Mark I	Mark II	Total
105-KE	1,405 (est.)	1,406 (est.)	857 (open)	3,668
105-KW	0	777 Al, 996 SS	2,048 (closed)	3,821
Total	1,405	3,179	2,905	7,489

The total number of spent-fuel canisters exceeds the capacity of any single basin in their present single-layer canister-storage configuration.

Table 2-1. N Reactor Fuel-element Dimensions and Densities
Used in the Calculations.

Outer tube	MKIV fuel element		MKIA fuel element	
	(cm)	(in.)	(cm)	(in.)
Zirconium clad outer diameter	6.160	2.425	6.106	2.404
Uranium outer diameter	6.032	2.375	5.979	2.354
Uranium inner diameter	4.422	1.741	4.592	1.808
Zirconium clad inner diameter	4.321	1.701	4.481	1.764
Enrichment (wt%)				
²³⁵ U	0.947		1.25	
²³⁶ U	0.0392		0.0392	
²³⁸ U	99.0138		98.7108	

Inner tube	(cm)	(in.)	(cm)	(in.)
Zirconium clad outer diameter	3.249	1.279	3.165	1.246
Uranium outer diameter	3.096	1.219	2.962	1.166
Uranium inner diameter	1.321	0.520	1.245	0.490
Zirconium clad inner diameter	1.219	0.480	1.118	0.440
Enrichment (wt%)				
²³⁵ U	0.947		0.947	
²³⁶ U	0.0392		0.0392	
²³⁸ U	99.0138		99.0138	

	(cm)	(in.)	(cm)	(in.)
Assembly length	66.294	26.1	53.035	20.88
End cap thickness (allowance)	0.483	0.190	0.483	0.190

	(kg)	(lb)	(kg)	(lb)
Maximum assembly (weight U)	23.4	51.6	16.58	36.55

	Densities (g/cc)
Uranium fuel	18.82
Zirconium clad	6.55
Canister water	1.00
Nitrogen gas	0.00114
Stainless steel	8.03
Aluminum	2.71
Basin water	1.00
Concrete	2.37

Table 2-2. N Reactor Spent-fuel Canister Dimensions and Densities
Used in MCNP Calculations.

	Mark I canister type	Mark II canister type
Material	Al 6061	SS 304
Density (gm/cc)	2.71	8.03

	(cm)	(in.)	(cm)	(in.)
Base plate thickness	0.952	0.375	0.635	0.25
Canister outer diameter	20.937	8.243	20.813	8.194
Canister inner diameter	20.477	8.062	20.508	8.074
Canister body thickness	0.229	0.090	0.152	0.060
Canister cylinder separation (center-to-center)	24.765	9.75	24.765	9.75
Canister lid thickness	0.965	0.380	1.270	0.50
Vertical distances from canister bottom				
To MKIA top end	53.988	21.255	53.670	21.13
To MKIV top end	67.246	26.475	66.929	26.35
To nitrogen level	65.075	25.62	66.472	26.17
To lid bottom	70.155	27.62	67.488	26.57
Stainless steel 304 composition (wt%)				
Iron	68.71			
Chromium	19.00			
Nickel	9.25			
Manganese	2.00			
Silicon	1.00			
Carbon	0.04			
Aluminum 6061 composition (wt%)				
Aluminum	97.90			
Magnesium	1.00			
Silicon	0.60			
Chromium	0.25			
Copper	0.25			

NOTE: Dimensions refer to one canister cylinder, which is actually one half of a canister (see Figures 2-1 and 2-2).

Figure 2-1. Present Fuel-storage Canister Array Showing Contents
(Canisters are not Stacked).

WDW080c

Figure 2-2. Proposed Fuel-storage Canister Array Showing Contents
(Canisters may be Stacked Three High).

WDW080b

The maximum fissionable quantities are defined by the spacing of the irradiated-fuel canisters, the stacking of the canisters, and the number of fuel elements that can be contained within a canister.

- The present irradiated-fuel canister storage location is 50.24 cm (19.78 in.) by 26.99 cm (10.63 in.).
- The present storage configuration is single stacked.
- The present fuel-storage canister contains seven elements per canister half.

The most highly enriched N Reactor fuel element is the MKIA (1.25-wt% ²³⁵U outer element, 0.947-wt% ²³⁵U inner element, and 1.15-wt% ²³⁵U effective average), which has a maximum standard length of 53.035 cm (20.88 in.). The largest fuel element is the MKIV (0.947-wt% ²³⁵U), a 66.294-cm- (26.1-in.-) long assembly. There is also an MKIA fuel type (A12N09E) that is 66.294 cm (26.1 in.) long. The total inventory of these long elements is 12 assemblies and 2 inner elements in one single canister (see Appendix C), and because of this very limited number, they do not change any conclusions based on the standard 53.035-cm- (20.88-in.-) long MKIA fuel assemblies.

The chemical form of the fissile material is uranium metal although the metal is slowly oxidizing in the water storage medium into uranium oxide, more commonly referred to as "sludge," which may include other nonfissile material also.

The canisters are typically the basic unit for transfer of the fissile material. Movement of individual fuel elements is a nonstandard procedure and would probably take place in localized areas outside the main basin-storage array.

Presently there are two methods of handling water cleanup. One method is to pump water from skimmer pumps through a sand filter, then through ion-exchange modules, and back to the basin. The sand filter is backwashed into the sandfilter backwash pit by reversing the normal water flow. Another, separate method, is to pump water from recirculation pumps through cartridge filters to ion-exchange columns and then back to the basin. The ion-exchange modules used for the skimmer pumps are separate from and different than the ion-exchange columns used by the recirculation pumps. The ion-exchange modules and ion-exchange columns are taken out and buried when their dose rates become too high for practical operation.

3.0 REQUIREMENTS AND EXEMPTIONS

The requirements unique to this evaluation are in three categories:

- Spacing change for canisters from the present 50.24 cm (19.78 in.) by 26.99 cm (10.63 in.) to the proposed 53.34 cm (21 in.) by 31.75 cm (12.5 in.)
- Stacking change from the present single stacking to the proposed triple stacking
- Physical form change from metallic to oxide in water with continued indefinite storage, including the 1.15-wt% ^{235}U and 1.25-wt% ^{235}U (previous evaluations considered only the 0.947-wt% ^{235}U dissolution in water).

The west bay of each storage basin is presently limited to MKIV (0.95-wt% ^{235}U) and no hanging fuel storage. No analyses were performed to relax these requirements in the west bays.

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4.0 METHOD OF ANALYSIS AND RESULTS

Basic assumptions included the following.

1. All spent fuel had zero exposure (no fuel burnup), the most reactive situation.
2. Fuel was stored in Mark I aluminum encapsulation canisters.
3. Comparisons were to be made to Mark II stainless steel fuel-encapsulation canisters.
4. Canister spacing was typical of the 105-KE and 105-KW Basins.
5. Canisters were double-stacked with no stacking plates (0.00 cm [0.00 in.]) between layers.
6. All MKIV fuel elements were 66.294 cm (26.1 in.) long, and all MKIA fuel elements were 53.035 cm (20.88 in.) long.
7. Uranium density was 18.82 grams/cc with explicit end caps.
8. All temperatures were 300 K (26.85 °C [80.33 °F]).

All canister dimensions and densities are listed in Table 2-2. Mark 0 open canisters were not considered.

The following off-normal situations were considered, both separately and in combination.

1. Uranium metal in water may dissolve into a $\text{UO}_2 + \text{H}_2\text{O}$ solution (1.25-wt% ^{235}U for MKIA outer tubes, 1.15-wt% ^{235}U for MKIA assemblies, and 0.947-wt% ^{235}U for MKIA inner tubes and for MKIV outer and inner tubes).
2. Additional MKIA or MKIV fuel may be stored in the canisters; because of space limitations, only an inner element may be inserted in the canisters.
3. Uranium metal elements may break apart into pieces of a more reactive size (1.25-wt% ^{235}U for MKIA outer tubes, 1.15-wt% ^{235}U for MKIA assemblies, and 0.947-wt% ^{235}U for MKIA inner tubes or MKIV inner and outer tubes).
4. Spills of fuel elements, assemblies, or broken pieces may introduce fissile material into the basin water between canisters.
5. Basin leaks may reduce the amount of water remaining between canisters.

4.1 SUMMARY OF PREVIOUS CRITICALITY WORK

Table 4-1 contains a summary of minimum critical-mass calculations for N Reactor fuel; masses were taken directly from WHC-SD-NR-CSER-010, Appendix B (Schwinkendorf 1994a). Broken N Reactor fuel was modeled with optimally sized and moderated rods. The optimum conditions (e.g., spacings, rod diameters) were taken from the original spreadsheet files in which these calculations were finalized. Four idealized geometries are included (spherical, hemispherical, cylindrical, and slab.) For each geometry, the minimum critical mass is shown for that fuel assembly or component along with the optimal conditions for each minimum. The center-to-center hexagonal spacing is adjusted until the critical mass is minimized. In the case of intact fuel assemblies, this spacing is 7.87 cm (3.1 in.) for both MKIV and MKIA fuel assemblies. The 1.15-wt% ^{235}U enriched-solution values, taken from Section 5.1 of this document, are included in Table 4-1 for completeness. These results were produced by first using WIMS-E to generate homogeneous, two-group cross sections for a lattice cell. These cross sections were then used in the neutron diffusion theory code GOLF (Schwinkendorf 1994b), which calculated the dimensions required for $k_{\text{eff}} = 1.00$. The masses were then calculated from the dimensions using spreadsheet software.

Consider the spherical critical masses from Table 4-1. The most reactive situations are the ones involving the highest ^{235}U enrichment, which is no surprise. The surprising effect is that the spherical critical masses for the solutions are greater than the geometries involving heterogeneous pieces. The following are arranged in terms of increasing spherical critical mass.

- 671-kg, 1.25-wt% ^{235}U rods (outer diameter [OD] = 0.762 cm, spacing = 1.619 cm)
- 1,087-kg, MKIA outer tubes (spacing = 7.37 cm, 1.15-wt% ^{235}U)
- 1,893-kg, MKIA assemblies (spacing = 7.87 cm, 1.15-wt% ^{235}U)
- 2,070-kg, 1.25-wt% ^{235}U solution (optimum hydrogen-to-uranium atom ratio [H/U] = 6)
- 2,886-kg, 0.947-wt% ^{235}U rods (OD = 1.27 cm, spacing = 2.286 cm)
- 4,220-kg, MKIV inner tubes (spacing = 4.83 cm, 0.95-wt% ^{235}U)
- 4,272-kg, 1.15-wt% ^{235}U solution (optimum H/U = 5)
- 4,315-kg, MKIA inner tubes (spacing = 4.83 cm, 0.95-wt% ^{235}U)
- 5,272-kg, MKIV outer tubes (spacing = 7.37 cm, 0.95-wt% ^{235}U)
- 7,794-kg, MKIV assemblies (spacing = 7.87 cm, 0.95-wt% ^{235}U)
- Unlimited, 0.947-wt% ^{235}U solution (optimum H/U = 5, maximum $k_{\infty} < 1$).

Table 4-1. Summary of Critical Masses for Various Idealized Geometries.

	Spherical (kg)	Hemispherical (kg)	Infinite cylinder (kg/m)	Infinite slab (kg/m ²)
MKIA assemblies	1,893	3,211	1,770	2,055
	(Spacing = 7.87 cm; 1.15-wt% ²³⁵ U)			
MKIA outer tubes	1,087	1,864	1,102	1,388
	(Spacing = 7.37 cm; 1.25-wt% ²³⁵ U)			
MKIA inner tubes	4,315	7,068	3,123	2,731
	(Spacing = 4.83 cm; 0.95-wt% ²³⁵ U)			
1.25-wt% rods	671	1,172	740	966
	(Spacing = 1.619 cm; OD = 0.762 cm)			
1.25-wt% solution	2,070	3,433	1,610	1,534
	(Optimum H/U = 6)			
1.15-wt% solution	4,272	6,944	2,780	2,162
	(Optimum H/U = 5)			
MKIV assemblies	7,794	12,568	4,909	3,841
	(Spacing = 7.87 cm; 0.95-wt% ²³⁵ U)			
MKIV outer tubes	5,272	8,521	3,476	2,862
	(Spacing = 7.37 cm; 0.95-wt% ²³⁵ U)			
MKIV inner tubes	4,220	6,898	3,054	2,786
	(Spacing = 4.83 cm; 0.95-wt% ²³⁵ U)			
0.947-wt% rods	2,886	4,774	2,247	2,124
	(Spacing = 2.286 cm; OD = 1.27 cm)			
0.947-wt% solution	Optimum H/U = 5; maximum $k_{\infty} < 1$ (therefore unlimited)			

Conclusion — These spherical critical mass results indicate that breaking fuel assemblies into fuel elements (outer or inner tubes) or into pieces (modeled as optimally sized and moderated uranium rods) will increase the k_{eff} . These spherical critical mass results also indicate that dissolution of fuel assemblies or fuel elements into a homogeneous solution will reduce the k_{eff} .

Calculations of the reactivity of canister spacing were reported in Section 5.1.3 of Wittekind (1991). The results demonstrated that closer canister spacing increased the reactivity of the canister array. A logical continuation is that wider separation of the canisters would not increase the reactivity of the canister array.

Conclusion — Increasing the spacing between canisters in the canister storage array would not increase reactivity.

4.2 WIMS-E VALIDATION AND APPLICATION

The WIMS-E code (Gubbins et al. 1982) has been extensively validated against critical experimental data. Previous validation efforts have covered low-enriched uranium metal billets (Erickson 1992; Schwinkendorf 1985a, and 1985b), MKIA fuel assemblies and uranium metal rods (Schwinkendorf 1992a), and low-enriched uranium solutions (Schwinkendorf 1992b).

The uranium metal rods validation included critical experimental data from two sources. The first (Hellens and Honeck 1962) was for 1.0-wt% ^{235}U -enriched rods and included measured boron poison effects. The published results were in the form of bucklings, not critical masses or k_{eff} . In order to compare WIMS-E results to the published results, WIMS-E results were output as two-group lattice-averaged cross sections for each experiment, and an analytical formula was used to calculate buckling for each case. WIMS-E results compared very well with experimental results, both as a function of water-to-uranium volume ratio and as a function of amount of poison added. The second source (Kupinski and Toffer 1970) contained data over a range of rod ODs and ^{235}U enrichments (0.444-cm [0.175-in.] OD to 7.62-cm [3.0-in.] OD and 3.0-wt% ^{235}U to 4.89-wt% ^{235}U). These results were in the form of critical masses in spherical and cylindrical geometry.

Validations also exist for the earlier WIMS-D version of the code in Schwinkendorf (1985b), which documents comparisons with annular uranium metal tubes over a range of enrichments from 0.947-wt% ^{235}U to 2.1-wt% ^{235}U . The WIMS-D code was shown to consistently predict criticality in a conservative manner, i.e., systems were predicted to be more reactive than experiments showed. Predicted critical masses were consistently less than measured values although the degree of conservatism would vary depending on the system being modeled. Uranium metal systems with large amounts of spatial self-shielding (e.g., billets) tended to be more conservative (less accurate) than "smaller" systems such as tubes and small-diameter rods. Trends always tended to be very good, such as the variation in k_{∞} , buckling, or critical mass as the moderator-to-fuel ratio varied. In other words, the shapes of these curves paralleled the data very nicely, even when a conservative bias was shown.

The WIMS-E code has been shown to be less conservative than the WIMS-D code, although this is believed to be more a function of the newer cross-section library than changes in the numerical methods employed in the code. WIMS-E tends to follow critical experimental data more accurately than WIMS-D but is still conservative. It has been the practice to neglect the imposition of a bias when using WIMS-E to calculate k_{∞} , buckling, or cross sections that are input to a diffusion theory code (to calculate idealized, finite dimensions).

WIMS-E is used in this document for scoping calculations and to illustrate trends while the criticality calculations are performed using MCNP.

4.3 MCNP VALIDATION AND APPLICATION

The MCNP code (Briesmeister 1986; Briesmeister and Hendricks 1990; Carter 1994) has been compared to critical experimental data for both fuel elements and solutions. Previous validation efforts have covered N Reactor fuel elements in water (Wittekind 1991b; Wittekind 1992; Wittekind 1993) and low enrichment uranium solutions (Wittekind 1994).

The MCNP N Reactor fuel bias was determined to be about -5 mk (Wittekind 1993). This means that MCNP would calculate k_{eff} about 5 mk less than experimental measurements. The MCNP low enrichment solution bias was determined to be about -3 mk (Wittekind 1994). This means that MCNP would calculate k_{eff} about 3 mk less than experimental measurements.

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5.0 CONTINGENCY ANALYSIS

The fissile material configurations considered MKIA and MKIV fuel assemblies, fuel dissolved into solution, and fuel broken apart into pieces (assumed to be rods of the most reactive size). These fissile material configurations were considered both inside and between canisters.

5.1 CRITICALITY OF 1.15 WEIGHT-PERCENT ^{235}U SOLUTION

Both 0.947-wt% ^{235}U - and 1.25-wt% ^{235}U -enriched uranium/water solutions were analyzed in Schwinkendorf (1994a). This same analysis was done for this document for 1.15-wt% ^{235}U solution, the mass-averaged enrichment of MKIA fuel. As expected, the k_{∞} of the solution lies between the curves for 0.947-wt% ^{235}U and 1.25-wt% ^{235}U solutions over the entire H/U range. This is shown graphically in Figure 5-1. The k_{∞} curve for 0.947-wt% ^{235}U solution never exceeds 1.0, so there is no critical mass. Both 1.15-wt% ^{235}U and 1.25-wt% ^{235}U solutions exceed 1.0 but only over a small range of H/U.

Figure 5-1. k -infinity Versus H/U Ratio — WIMS-E Calculations.

Figure 5-2 illustrates the calculated spherical critical mass over that range of H/U for which k_{∞} exceeds 1.0. Appendix A contains a listing of the FORTRAN pre-processor code used to generate the WIMS-E input files for these calculations. The GOLF code used the two-group cross sections for the uranium/water solutions that were output by WIMS-E for each H/U value. These uranium/water solutions are fully reflected with light water.

Figure 5-2. Critical Mass Versus H/U Ratio — WIMS-E/GOLF Calculations.

Conclusions — Minimum critical mass for 1.25-wt% ^{235}U solution, at optimal moderation of $\text{H/U} = 6$, is greater than for MKIA assemblies that have a weighted 1.15-wt% ^{235}U enrichment. Minimum critical mass for 1.15-wt% ^{235}U is not significantly different from that for the MKIA inners at 0.95-wt% ^{235}U or for the MKIV inners at 0.947-wt% ^{235}U . Therefore, dissolution of MKIA or MKIV fuel assemblies or fuel elements into homogeneous solution is not a criticality hazard because the reactivity is lower and the critical mass is greater in the homogeneous solution.

5.2 URANIUM IN MODERATOR EFFECT

The presence of fissionable material in the moderator tends generally to reduce the k_{∞} of the system. This has been explained in terms of decreasing the resonance escape probability (Schwinkendorf 1994a).

For low-enriched uranium metal systems, lattice heterogeneity allows neutrons to be born from fission in the fuel, escape to the moderator, where they can thermalize in the absence of resonance absorbers (i.e., ^{238}U), and then diffuse back into the lumped fuel region where they can cause additional fission. The introduction of uranium into the moderator region places additional absorption into this region; the neutrons thermalize, but additional neutrons are absorbed before they can reenter the fuel region.

This effect was quantified for 0.947-wt% ^{235}U metal rods immersed in a "sludge" solution of $\text{U}(0.947)\text{O}_2$ in water, for 1.25-wt% ^{235}U metal rods immersed

in a solution of $U(1.25)O_2$ in water, and for both MKIV and MKIA fuel assemblies immersed in a solution of $U(0.947)O_2$ in water. This analysis was originally performed to address a fire scenario in which uranium metal fuel would supposedly break up into pieces of scrap, with oxidized (i.e., burned) fuel material going into solution in the water used to combat the fire. For these rod calculations the concentration of UO_2 in water was varied from 2.0 g/cm^3 to 4.0 g/cm^3 . For the intact fuel assemblies, only 3.0 g/cm^3 UO_2 in the sludge solution was analyzed.

In all cases analyzed, peak reactivity was decreased when uranium went into the moderator region, and the spacing that maximized reactivity between either fuel or scrap pieces was increased. Reactivity of the system was also increased by the presence of solid uranium metal pieces, as compared to a homogeneous solution of just the UO_2 in water, as long as the UO_2 content of the solution was below 5.0 g/cm^3 . The minimum critical-mass type of system analyzed turned out to be optimally sized uranium metal scrap pieces optimally spaced in pure water.

Conclusions — As uranium is added to the moderator between broken fuel pieces, modeled as low enrichment uranium rods, the reactivity of the whole system decreases even though fissile material is being added to the system. Thus we may expect the most reactive situation to be the one with pure water as the moderator and with the fuel in heterogeneous lumps as rods, separate elements, or entire intact assemblies.

5.3 PARAMETRIC CALCULATIONS FOR FUEL CANISTERS

The W-PIJ module in the WIMS-E computer code was used to perform a large set of two-dimensional (r, θ) transport calculations on MKIV and MKIA fuel in fuel-storage canisters. Table 5-1 contains a matrix of the various combinations of intact and dissolved fuel that were analyzed. The rows of Table 5-1 contain variations of fuel and dissolved sludge inside canisters, and the columns contain different situations outside the fuel canisters.

Internal configurations modeled include the following:

- Clusters of intact MKIV or MKIA fuel assemblies
- Outer elements dissolved with inner elements remaining intact
- Inner elements dissolved with outer elements remaining intact
- Both outer and inner elements dissolved
- Extra inner element inside canister
- Assemblies broken into optimally sized and moderated rods.

Configurations between canisters, by column, include the following:

- Light water
- "Actual" sludge
- Uniform UO_2 -water solution
- Spilled intact fuel assemblies in water
- Spilled fuel fragments in water
- Spilled fuel fragments in 2.0 g/cm^3 UO_2 -water solution.

Table 5-1. WIMS-E Consolidated Fuel Criticality Calculation Matrix.

Configuration inside canisters	Configuration outside canisters										
	Light-water	"Actual" sludge	Uniform uranium-oxide solution			Fuel assemblies spilled in water		Fuel fragments spilled			
			0.95-wt% ²³⁵ U	1.15-wt% ²³⁵ U	1.25-wt% ²³⁵ U	MKIV	MKIA	In water		In UO ₂ solution	
								0.95-wt% ²³⁵ U	1.25-wt% ²³⁵ U	0.95-wt% ²³⁵ U	1.25-wt% ²³⁵ U
MKIA intact	0.891233	0.740972	1.000073	1.030175	1.043302	1.037297	1.070669	1.053685	1.100853	1.004463	1.048779
MKIA outer dissolved	0.882886	0.650576	0.952918	0.981246	0.993644	0.991289	1.025137	1.010503	1.056039	0.957769	1.003238
MKIA inner dissolved	0.883545	0.692557	0.966360	0.994400	1.006658	1.003448	1.036168	1.021313	1.066164	0.971081	1.014887
MKIA assembly dissolved	0.885622	0.613507	0.940187	0.968121	0.980368	0.979401	1.013863	0.999878	1.045207	0.944970	0.991491
MKIA intact + extras	0.885732	0.734847	0.994480	1.024661	1.037831	1.031651	1.065241	1.048366	1.095797	0.998968	1.043523
MKIA fragments + water	0.918705	0.771091	1.017266	1.045859	1.058374	1.053456	1.084552	1.068061	1.112764	1.021779	1.063080
MKIV intact	0.838566	0.689176	0.962509	0.993741	1.007361	1.001085	1.035234	1.019146	1.067907	0.929121	1.014122
MKIV outer dissolved	0.841557	0.524030	0.892279	0.920911	0.933467	0.932805	0.969452	0.957179	1.004246	0.898531	0.947830
MKIV inner dissolved	0.840932	0.608966	0.917783	0.946214	0.958657	0.956474	0.990769	0.977570	1.023559	0.924030	0.969910
MKIV assembly dissolved	0.849845	0.439877	0.873762	0.902328	0.914871	0.915459	0.954310	0.943017	0.990820	0.879729	0.932250
MKIV intact + extras	0.832231	0.682757	0.957626	0.989104	1.002839	0.996208	1.030721	1.014643	1.063881	0.964524	1.009912
MKIV fragments + water	0.855996	0.721460	0.981157	1.011669	1.025009	1.018987	1.050974	1.034911	1.082043	0.988449	1.030406

The WIMS-E (r, θ) calculations explicitly model a seven-assembly cluster of fuel (center assembly with a hexagonal ring of six assemblies around it) in a canister, surrounded by a lattice cell radius that corresponds (by conserving moderator volume) to the array spacing. The outer boundary is modeled as reflective, so that an infinite array of these canisters is modeled. If the cluster of seven intact fuel assemblies is offset from the center of the canister, it is possible to fit another inner element into the available space. This is represented in Table 5-1 as the "MKIA(V) intact + extras" row. The "MKIA(V) fragments + water" row represents a mixture of optimally reactive uranium rods in water.

Input filenames, along with the k_{∞} results, for all WIMS-E calculations may be found in Appendix B.

5.3.1 Parametric Calculations for Rods in Solution

For a homogeneous solution, a UO_2 concentration equal to 4.0 g/cm^3 has been found to be the optimum for maximizing k_{∞} (Schwinkendorf 1994a), but Figures 5-3 through 5-6 illustrate that when metal pieces are introduced, the UO_2 concentration that maximizes k_{∞} for the lattice is less than 4.0 g/cm^3 . The value 2.0 g/cm^3 was selected for the UO_2 concentration when scrap is imbedded in the solution simply to quantify how much reactivity is lost when UO_2 is added to the solution. The most reactive case occurs when there is no dissolved UO_2 in the moderator region (Table 5-1, "Fuel fragments spilled in water").

Figures 5-3 through 5-6 show how cell k_{∞} varies for 1.25-wt% ^{235}U -rods immersed in UO_2 solution, which ranges from 2.0 to 4.0 g/cm^3 UO_2 . The maximum reactive configuration was for rods immersed in 2.0 g/cm^3 UO_2 solution. For rods that are too large to represent broken pieces of N Reactor fuel elements, 3.0 g/cm^3 UO_2 solution appears to yield a higher k_{∞} than the 2.0 g/cm^3 UO_2 solution. Fuel scrap comes from fuel assembly components, and it is difficult to postulate scrap material with a characteristic size larger than a fuel element's radial thickness. For smaller rods that are more characteristic of the radial thickness of fuel elements and more realistically model broken fuel pieces, adding more UO_2 to the moderator results in a lower peak k_{∞} . As rod spacing becomes larger, the k_{∞} of the lattice asymptotically approaches the k_{∞} of the pure UO_2 solution as a limit.

Qualitative predictions made from past experience (lattice reactivity reduced by uranium in the moderator region Schwinkendorf [1994a]) are supported by the W-PIJ results. In general, the systems with the highest reactivity are those in which the water moderator is free of resonance-absorber material (i.e., ^{238}U). When there is only pure water between the canisters, the highest neutron multiplication factor is generally with all assemblies intact. In Table 5-1, these k_{∞} values are equal to 0.891233 for MKIA and 0.838566 for MKIV. When either outer tubes or inner tubes, or both, are dissolved, reactivity goes down for MKIA fuel (and up slightly for MKIV fuel), although these effects are not strong (see other k_{∞} values in the first column of Table 5-1.)

When an additional inner tube is placed into the canister, k_{∞} decreases somewhat (from 0.891233 to 0.885732 for MKIA; from 0.838566 to 0.832231 for

Figure 5-3. k -infinity Versus Spacing — 1.25-Wt%, ^{235}U ,
0.38-cm (0.15-in.) Outer-diameter Rods in $\text{UO}_2 + \text{H}_2\text{O}$.

Figure 5-4. k -infinity Versus Spacing — 1.25-Wt%, ^{235}U ,
0.635-cm (0.25-in.) Outer-diameter Rods in $\text{UO}_2 + \text{H}_2\text{O}$.

Figure 5-5. k -infinity Versus Spacing — 1.25-Wt%, ^{235}U , 1.27-cm (0.50-in.) Outer-diameter Rods in $\text{UO}_2 + \text{H}_2\text{O}$.

Figure 5-6. k -infinity Versus Spacing — 1.25 Wt%, ^{235}U , 3.81-cm (1.50-in.) Outer-diameter Rods in $\text{UO}_2 + \text{H}_2\text{O}$.

MKIV). The lattice spacing inside the canisters is slightly less than optimum and is therefore undermoderated; introducing another inner element causes the lattice to be even more undermoderated.

The last entry in the first column of Table 5-1, optimally moderated uranium rods in water, increased (from 0.891233 to 0.918705 for MKIA; from 0.838566 to 0.855996 for MKIV fuel) because optimum scrap is more reactive than intact fuel assemblies.

When heterogeneous fissionable material pieces are introduced into the moderator between canisters, reactivity goes up, most significantly when the physical form of the spilled material is heterogeneous metal pieces (debris) with no dissolved UO_2 in the moderator (e.g., compare the "Fuel fragments spilled in water" column with values in the "Light water" column of Table 5-1).

Actual sludge of Table 5-2 represents a more physically realistic sludge, characteristic of measured sludge samples (Wittekind 1994). When a typical measured sludge from Table 5-2 is added between canisters, reactivity goes down significantly. "Actual" sludge contains less fissionable material and significantly more neutron absorbers.

When 4.0 g/cm^3 UO_2 solution is introduced between canisters (i.e., see the "Uniform uranium oxide solution" columns), array reactivity goes up. The k_{∞} of the optimum UO_2 solution (which by itself is equal to 1.077) drives the array to higher values.

Conclusions — As more UO_2 is added to the solution in the moderator region, the maximum k_{eff} is lower than for the same size rods in pure water. Figure 5-6, contrary to this trend, was calculated using a coarse mesh, and this result is believed to be a mathematical artifact.

5.3.2 Parametric Calculations for Fuel Spills

When intact N Reactor fuel was spilled between canisters and immersed in a pure water moderator, this heterogeneity increased array reactivity further. Regardless of enrichment, the k_{∞} values in the "Uniform uranium oxide solution" columns are higher than the corresponding values in the first column of Table 5-1. For equal enrichment (e.g., MKIV fuel compared to $\text{U}[0.95]\text{O}_2$ solution), intact fuel pieces are more reactive than the solution. The intact MKIA fuel spill (at an average enrichment equal to 1.15-wt% ^{235}U) is even more reactive than the 1.25-wt%- ^{235}U UO_2 solution. Intact fuel immersed in UO_2 solution is less reactive than when immersed in pure water (Schwinkendorf 1994a).

Intact fuel does not have optimal heterogeneity for water lattices, so when optimal scrap was introduced into the array between canisters, array reactivity went up still further (see the "Fuel fragments spilled in water" columns in Table 5-1.) As the columns labeled "Fuel fragments spilled in UO_2 solution" shows, adding 2.0 g/cm^3 UO_2 solution to the scrap lattice reduces k_{∞} in all cases when compared with the corresponding entries under the "Fuel fragments spilled in water" columns.

Table 5-2. Isotopic Composition of Actual Sludge (wt%).

²³⁸ U	30.3602	²⁴¹ Am	0.00063	Mg	0.36897	V	0.1002
²³⁵ U	0.29027	²³⁷ Np	0.00275	Na	0.21478	Au	0.08697
²³⁶ U	0.02182	Si	0.0153	Ti	0.38903	Al	28.1511
²³⁴ U	0.00212	Cu	0.05657	Cd	0.03092	Fe	38.2198
²³⁹ Pu	0.04809	Ga	None	K	None	H	0.12988
²³⁸ Pu	2.8 E-5	Ca	None	Mn	0.34115	O	1.16936

NOTE: "None" means this nuclide was not available in the WIMS-E library (the remaining wt% were normalized to 100%).

Source: Wittekind, W. D., 1994, *MCNP Calculations of K-Infinity for 100KE Basin Sludge Samples*, Table 3-2, WHC-SD-NR-CSER-008, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

Conclusions — When fissile material is added to the moderator region between canisters, the fuel rods (used to model broken fuel pieces) in water have the highest k_{eff} , then intact fuel elements in water, followed by rods in UO_2 solution. All cases of adding heterogeneous fuel pieces between canisters raises the calculated k_{eff} .

5.3.3 Parametric Calculations for Fuel Pieces

When introducing metal scrap between canisters, care was given to the definition of "optimum rods." It is known that the optimum-geometry lattice for a minimum critical mass is close to, but not the same as, the lattice configuration for maximum reactivity. Optimum rods, as a model for scrap, have been identified previously as 1.27-cm- (0.50-in.-) diameter rods for a minimum critical mass of 0.95-wt% ^{235}U -enriched material; 0.76-cm- (0.3-in.-) diameter rods were identified as optimum for 1.25-wt% ^{235}U scrap (Schwinkendorf 1992a). For this analysis, however, it was desired to see how high reactivity in the array could be driven. For 0.95-wt% ^{235}U scrap in pure water, maximum reactivity was calculated to be achieved with 1.778-cm- (0.70-in.-) diameter rods (at a center-to-center hexagonal spacing equal to 2.92 cm [1.15 in.]), and 1.52-cm- (0.60-in.-) diameter rods at a lattice spacing of 2.565 cm (1.01 in.) for both 1.15-wt% ^{235}U - and 1.25-wt% ^{235}U -scrap. The parametric calculations were performed at a 0.254-cm (0.10-in.) resolution in rod diameter; the optimum rod diameters for 1.15-wt% ^{235}U and 1.25-wt% ^{235}U rods were equal only to within this resolution.

Figures 5-7 through 5-9 illustrate that the heterogeneous lattice geometry that yields a maximum k_{∞} does not minimize critical mass. Scrap, represented by maximum k_{∞} rod lattices, was used to drive the fuel canisters for the "Fuel fragments spilled" model. As uranium becomes oxidized, and UO_2 goes into solution, the trend is for cell reactivity to decrease. As shown in Schwinkendorf (1994a), UO_2 solution k_{∞} reaches maximum when the UO_2 content of the solution is 4.0 g/cm³. However, this k_{∞} is less than the k_{∞} of a uranium metal rod lattice immersed in pure water.

For the "Fuel fragments spilled in UO_2 solution" columns of Table 5-1, the scrap lattice was assumed to be composed of 1.27-cm- (0.50-in.-) OD rods, optimally spaced in 2.0 g/cm³ UO_2 solution. Optimal spacing for the 0.95-wt% ^{235}U -enriched case was 2.67 cm (1.05 in.) center-to-center, and 3.15 cm (1.24 in.) center-to-center for 1.25-wt% ^{235}U . The single most reactive geometry turned out to be, not surprisingly, the fragmentized MKIA fuel (rods at 1.15-wt% ^{235}U enrichment) inside the canisters, with 1.25-wt% ^{235}U -enriched scrap between the canisters. WIMS-E predicted that the base case of $k_{\infty} = 0.918705$ for no scrap outside the canisters was increased to $k_{\infty} = 1.112764$.

5.3.4 Parametric Calculations for Water Density

Interspersed moderation was also analyzed. Figure 5-10 shows the effect of removing the water moderator from the space between fuel canisters. WIMS-E predicted that the moderator density required for maximum reactivity was equal to 0.40 g/cm³. For densities greater than this, the water between canisters was acting to isolate the canisters from each other. The maximum WIMS-E

Figure 5-7. k-infinity, Critical Mass Versus Outer Rod Diameter — 0.95-Wt% ²³⁵U.

Figure 5-8. k-infinity, Critical Mass Versus Outer Rod Diameter — 1.15-Wt% ²³⁵U.

Figure 5-9. k -infinity, Critical Mass Versus Outer Rod Diameter — 1.25-Wt% ^{235}U .

Figure 5-10. k -infinity Versus Basin Water Density — MKIA Fuel in Canisters.

calculated reactivity for unexposed, intact MKIA fuel in aluminum canisters caused by loss of interspersed moderation is 0.974174 (from case mkia_can.011 in Appendix B).

Assuming this optimum density for water between unexposed, intact MKIA fuel in aluminum canisters (i.e., 0.40 g/cm^3), the density of water inside the fuel canisters was also varied. Figure 5-11 shows that full density moderation inside the unexposed, intact MKIA fuel in aluminum canisters is the most reactive configuration. This makes sense because the spacing of fuel assemblies inside the fuel canisters is somewhat less than optimum, which is a hexagonal center-to-center spacing equal to 7.87 cm (3.1 in).

Figure 5-11. k-infinity Versus Canister Water Density
— MKIA Fuel in Canisters.

Figure 5-12 shows interspersed moderation between exposed MKIA fuel fragments in aluminum canisters, both with and without fission products, as compared to unexposed, intact MKIA assemblies. The bottom curve is the unexposed, intact MKIA fuel in aluminum canisters as shown in Figure 5-10. The burnup effect of using isotopes characteristic of 6.0-wt% ^{240}Pu is approximately 20 mk all through the range of interspersed moderation. These isotopes were generated by WIMS-E executed in a burnup mode. The calculations allowed for both fragmentized MKIA fuel inside the aluminum canisters and loss of basin water to achieve optimum moderation. The optimum moderation for maximum reactivity resulted in 0.762-cm- (0.30-in.-) radius uranium rods in a 1.35-cm- (0.53-in.-) radius cell. Packing these optimum reactivity rods into the 10.254-cm radius, 66.8528-cm inside canister height gives 291.9 lb/canister. There is a limit on scrap in canisters of 180 lb so this maximum canister reactivity for MKIA fuel scrap would not be achieved in practice.

Figure 5-12. k -infinity Versus Basin Water Density
— Exposed MKIA Scrap in Canisters.

Conclusions — Loss of moderation from between canisters raises the k_{eff} , but not enough to challenge the criticality limit of $k_{eff} = 0.98$. There is a limitation in the two-dimensional WIMS-E calculation in that loss of moderation was modeled by water at less than full density instead of full density water at a level part way up the canisters.

5.3.5 Parametric Calculations for Temperature

Figure 5-13 shows calculated k_{∞} versus temperature. The reduction in k_{∞} with increasing temperature is caused by Doppler broadening where the ^{238}U resonance more effectively absorbs neutrons.

Conclusion — An increase in the fuel temperature decreases the reactivity so there is no criticality problem associated with higher fuel temperatures.

5.3.6 Parametric Calculations for Fuel Dissolution

The effects of partial dissolution were also analyzed. For example, scenarios in which only one, two, or three MKIA outers were dissolved, in different relative positions within the cluster of seven assemblies, were analyzed. None of these combinations produced any k_{∞} values that were larger than the intact value $k_{\infty} = 0.891233$ and most were at or below $k_{\infty} = 0.88$. Having two assemblies with dissolved outer tubes next to each other, versus at opposite points in the hexagonal ring, was found to produce only a small

Figure 5-13. k -infinity Versus Fuel Temperature
— MKIA Fuel in Canisters.

reactivity effect. The initial effect of dissolving fuel appears to be a reduction of k_{∞} to values around 0.869 for two outer tubes dissolved (there is some variation at the sub-mk level depending upon configuration). As the number of dissolved outer tubes is increased, k_{∞} increases to the value cited in Table 5-1 for all outer tubes dissolved, 0.882886. The existence of minima or maxima in physical systems arises because of competing effects. The initial reduction of k_{∞} when one or two outer tubes is dissolved is because of the addition of resonance absorbers into the moderator region while the subsequent k_{∞} increase is because of the increasing UO_2 concentration in the moderator. In other words, the initial dissolution reduces reactivity from the pure, clean lattice configuration, but as UO_2 concentration continues to increase, the moderator region itself begins to contribute to the system k_{∞} . However, as more fuel components dissolve, including the inner tubes, k_{∞} never reaches the initial value for the clean lattice configuration.

Figure 5-14 illustrates this behavior as more MKIA outer tubes are dissolved and go into solution inside the fuel canister.

The stainless steel canisters were typically 70 mk less reactive than the aluminum canisters for the fuel-canister array containing intact MKIA fuel and surrounded by pure water. This is consistent with earlier documentation suggesting a 50 mk difference for MKIV fuel (Wittekind 1992). All results shown in Table 5-1 were produced for aluminum fuel canisters with no fission products. Burnup isotopes characteristic of 6-wt% ^{240}Pu , including fission products, provide an additional reactivity reduction of 20 mk.

Figure 5-14. Dissolution of MKIA Outer Elements
— MKIA Fuel in Canisters.

Conclusion — Dissolution of intact MKIA assemblies decreases the system reactivity and does not pose a criticality problem.

5.4 MCNP CALCULATIONS FOR MKIA FUEL IN CANISTERS

The MKIA fuel is considered in the the calculations involving the limiting case for criticality because it is more reactive than the MKIV fuel. The previous survey calculations using WIMS-E have demonstrated that temperature, loss of basin water, and fuel-element dissolution is not a problem for criticality safety. The unlimited introduction of fissile material between canisters, however, is a situation that is of concern for criticality safety. The most limiting case involves breaking up assemblies into more reactively sized pieces in pure water rather than dissolving them into a homogeneous solution. MCNP was used to make final calculations because of its capacity to explicitly model three dimensions (Briesmeister 1986; Briesmeister and Hendricks 1990; Carter 1994).

5.4.1 MCNP Results Compared to WIMS-E Results

MCNP calculations were performed on several of the WIMS-E scenarios reported in Section 5.3 as worst cases. The WIMS-E W-PIJ calculations are ideal for scoping calculations because of their rapid execution times and the ease of creating input. The calculations, however, do not model the details of

three-dimensional geometries; they model a single, stacked layer of canisters by incorporating an axial buckling leakage correction. This approximate approach to modeling axial leakage assumes that there is no spatial variation in materials or geometry in the axial (vertical) direction. MCNP models directly the explicit three-dimensional geometry of the physical system. It should therefore not be surprising that the MCNP results are somewhat different from the corresponding WIMS-E results. The WIMS-E results are to be interpreted as indicating which configurations are more reactive than others.

The first MCNP calculations were for a single layer of intact MKIA fuel-storage canisters and represented a horizontally infinite array. The k_{eff} calculated by MCNP was equal to 0.70208 ± 0.00206 . When applying the three-dimensional MCNP approach to the worst-case and several other scenarios, it was found that k_{eff} for the WIMS-E worst case was equal to 1.09514 ± 0.00097 . This value, shown in Table 5-3, was smaller than the corresponding WIMS-E value.

Table 5-3. Comparison of WIMS-E with MCNP k_{eff} for Canister Arrays.

Case number	Calculation description	WIMS-E k_{eff}	MCNP $k_{\text{eff}} \pm 1\sigma/k_{\text{eff}}$
1	MKIA loading, all intact, pure water moderator, infinite lattice	0.891233	0.70208 ± 0.00206
2	Same as case 1, but homogeneous (optimum) UO_2 in water solution in between canisters; $E = 1.25\%$	1.043302	0.99855 ± 0.00186
3	Same as case 1, but homogeneous (optimum) UO_2 in water solution in between canisters; $E = 0.95\%$	1.000073	0.94960 ± 0.00175
4	MKIV loading, all intact, pure water moderator infinite lattice	0.838566	0.66874 ± 0.00151
5	MKIA canisters, but heterogeneous (optimum) 1.25-wt% ^{235}U rods + water between canisters	1.100853	1.06256 ± 0.00103
6	MKIA fragments inside canisters, and (optimum) 1.25-wt% ^{235}U rods + water between canisters	1.112764	1.09514 ± 0.00097
7	Same as case 6, but with proposed lattice spacing, no aluminum spacer plate	N/A	1.10463 ± 0.00098
8	Same as case 6, but optimum MKIA fuel assemblies surrounding the canister	1.084552	1.04518 ± 0.00102
9	Same as case 7, but sludge height only 30 cm deep	N/A	1.03302 ± 0.00105
10	Same as case 7, but no sludge layer	0.918705	0.73231 ± 0.00113

Conclusion — WIMS-E results are more conservative (higher k_{eff} values) than MCNP.

5.4.2 Stacked-canister Vertical-spacing Reactivity

An additional MCNP calculation was performed to consider two options:

- Option A - stacked canisters touching
- Option B - stacked canisters separated by 4.44 cm (1.75 in.).

Many previous MCNP calculations included a 0.635-cm (0.25-in.) aluminum stacking plate to separate the double-stacked canisters in the vertical direction (Drawing H-1-4344). Elimination of this plate brings the two canister layers closer together. Because neutronic isolation was established by previous analyses using this stacking plate, a comparison was made using the case of no stacking plate, no vertical spacing between canister layers. Table 5-4 shows the MCNP results for the cases with and without the aluminum stacking plate present between canister layers.

Table 5-4. MCNP Double-stacked Canisters with and without Stacking Plates.

Case number	Calculation description	MCNP $k_{\text{eff}} \pm 1\sigma/k_{\text{eff}}$ (0.635 cm aluminum plate)	MCNP $k_{\text{eff}} \pm 1\sigma/k_{\text{eff}}$ (no vertical spacing)
1	Initial neutron source not vertically centered	0.67253 ± 0.00170	0.66666 ± 0.00164
2	Same as above, but initial neutron source vertically centered in both canisters	0.66872 ± 0.00190	0.66864 ± 0.00172

Note: Aluminum canisters, intact MKIV fuel, light water moderator, infinite lattice.

The MCNP results for the vertical-stacking comparison showed no reactivity effect caused by vertical spacing for MKIV fuel. The MKIA case was not calculated; because of the shorter MKIA length, there is increased distance between the fuel resulting in less neutronic interaction than the MKIV case.

One surprising effect was that the MCNP initial neutron-source location caused so much k_{eff} change for the MKIV fuel in aluminum canisters with a 0.635-cm (0.25-in.) aluminum stacking plate. The difference is

$$(0.66872 - 0.67253) / (0.66872 \times 0.67253) \times 1,000 \text{ mk} = -8.47 \text{ mk} \pm 3.80 \text{ mk}.$$

In the nonvertically centered initial source location, the initial neutron source is 3.3147-cm (1.3-in.) below the midpoint in the bottom canister and 3.3147 cm (1.3-in.) below the midpoint in the upper canister. A change of 2.2 MCNP-calculated standard deviations is not necessarily

statistically significant. The initial neutron source locations have not been shown to have a significant effect on MCNP-calculated results.

Conclusion — There is no increase of reactivity because of stacking fuel-storage canisters on top of each other. The only effect on criticality safety would be related to increased possibility of spilling fuel out of the canisters in the upper layers and inserting this fissile material between the canisters in a lower layer.

5.4.3 Results for Sludge Buildup

The final set of MCNP calculations addresses finite sludge buildup of 1.25-wt% ^{235}U -enriched uranium scrap in water outside and around the bottom of a canister array. Previous calculations addressed this sludge buildup in great detail, but for only 0.95-wt% ^{235}U -enriched material (Wittekind 1992). There is no possibility of a criticality with UO_2 in solution (by itself) for uranium enriched to less than 1.0-wt% ^{235}U . It was shown that any depth of sludge (or heterogeneous scrap optimally sized and spaced) was acceptable from a criticality perspective as long as only 0.95-wt% ^{235}U material was present. MCNP was therefore used to model a conservative sludge buildup around the outside of the canisters to various depths. Table 5-5 contains the results of several MCNP calculations modeling the variation in k_{eff} with sludge depth. This sludge was modeled as 1.25-wt% ^{235}U -enriched uranium metal rods, optimally sized and spaced in a lattice in pure water and accumulated outside and near the bottom of the canisters. These calculations show that as long as the sludge depth remains below 21.54 cm (8.48 in.), the calculated k_{eff} will remain below the limiting value of 0.98. A safe sludge depth, based on maximum reactive sludge composed of 1.25-wt% ^{235}U -enriched scrap material, is half this depth, or 10.77 cm (4.24 in.). This satisfies the double-contingency criterion by allowing for twice the sludge depth limit before the criticality limit of $k_{\text{eff}} = 0.98$ could be challenged.

Table 5-5. MCNP-calculated k_{eff} Versus Sludge Depth.

Sludge depth		MCNP $k_{\text{eff}} \pm 1\sigma/k_{\text{eff}}$
(cm)	(in.)	
0.00	0.00	0.73231 $\pm$ 0.00113
20.00	7.87	0.96938 $\pm$ 0.00104
21.54	8.48	0.98079 $\pm$ 0.00111
24.56	9.67	1.00022 $\pm$ 0.00112
30.00	11.81	1.03302 $\pm$ 0.00105

The use of the limiting value $k_{\text{eff}} = 0.98$ is consistent with established practice at the Hanford Site when dealing with low-enriched uranium material. Use of this limiting value is based on the *Nuclear Criticality Safety Manual* (WHC-CM-4-29). The criticality behavior of low-enriched uranium metal systems

is such that a large incremental mass is required to raise k_{eff} from 0.98 to 1.00. This is because the maximum k_{∞} for these systems exceeds 1.0 by only a very small amount, and k_{eff} increases on a steep asymptotic curve as mass is added to the system.

The MCNP model used to produce Table 5-5 assumed a worst-case scenario for the canister contents, completely fragmentized MKIA scrap at 1.15-wt% ^{235}U -enrichment. This assumption does not contribute to the occurrence of a contingency because there is no way of verifying the physical configuration or status of fuel assemblies inside canisters without opening the canisters; visual inspection cannot verify whether the fuel is intact or is partially dissolved. A large measure of conservatism is present in this calculation because the fissile material is modeled as pieces in pure water instead of the more likely pieces in solution or even a homogeneous solution.

According to WHC-CM-4-29 (Section 2.0, Paragraphs 5.1.2 and 5.1.3), the maximum calculated k_{eff} for cylinders is 0.98, while for slabs it is 0.97. The accumulation of fissile material between canisters will use the 0.97 limit. The acceptable margin of subcriticality should also accommodate code bias and statistical uncertainty (2σ or 95% confidence level). This criticality k_{eff} limit calculates to

$$0.97 - 0.005 - 2 \times (0.00111 \times 0.98079) = 0.96282.$$

The sludge limit for this k_{eff} can be calculated according to the Lagrange interpolating polynomial:

$$\begin{aligned} \text{depth } (k_{\text{eff}} = 0.96282) &= \frac{(0.96282 - 0.96938)(0.96282 - 0.98079)}{(0.73231 - 0.96938)(0.73231 - 0.98079)} * 0.00 \text{ cm} + \\ &\quad \frac{(0.96282 - 0.73231)(0.96282 - 0.98079)}{(0.96938 - 0.73231)(0.96938 - 0.98079)} * 20.00 \text{ cm} + \\ &\quad \frac{(0.96282 - 0.73231)(0.96282 - 0.96938)}{(0.98079 - 0.73231)(0.98079 - 0.96938)} * 21.54 \text{ cm} \\ &= 0.002 \times 0.0 \text{ cm} + 1.531 \times 20.00 \text{ cm} - \\ &\quad 0.533 \times 21.54 \text{ cm} \\ &= 19.14 \text{ cm (7.53 in.)}. \end{aligned}$$

Conclusions — As long as the sludge depth remains below 19.1 cm (7.53 in.), the calculated k_{eff} will remain below the criticality limit of 0.97 for slab geometry (including MCNP code bias of 5 mk and a 95% confidence level). A safe sludge depth, based on maximum reactive sludge composed of

1.25-wt% ^{235}U -enriched scrap material, is half this depth, or 9.525 cm (3.75 in.). This satisfies the double-contingency criterion by allowing for twice the sludge depth limit before the criticality limit of $k_{\text{eff}} = 0.97$ could be challenged.

5.5 EFFECT OF EXPOSURE ON CRITICAL MASS

One factor not addressed in Schwinkendorf (1994a) was the effect of fuel burnup, or exposure, on criticality. For this analysis burnup of MKIA fuel was simulated using WIMS-E, assuming an N Reactor graphite lattice and hot operating conditions calculated by the DCODE computer code (Thierer 1968). The WIMS-E code tracks over 30 specific fission products with the balance of the reactivity effects being lumped into a single "pseudo" fission product. The exposure-dependent fuel isotopes were then used to calculate maximum reactivity and minimum critical mass in water as a function of exposure. After the fission-product data were brought in from the WIMS-E burnup calculation, ^{135}Xe and ^{135}I were zeroed out because of the comparatively short half-lives of these isotopes. Table 5-6 contains the results of these exposure calculations. The increase in critical mass is shown for 2 days and for 92 days. There is a very slight reduction in minimum critical mass as the fuel decays further because of changing isotopic compositions.

Table 5-6. Variation in MKIA Minimum Critical Mass with Exposure.

Fuel exposure	Minimum critical mass (lb)
Green	4,173
2 days	4,372
92 days	5,340
10-year decay (from 92-day exposure)	5,300

Figure 5-15 shows the infinite neutron multiplication factor k_{∞} for MKIA fuel as a function of exposure in the hot operating reactor environment. Unlike MKIV fuel, MKIA fuel, because of its higher overall enrichment, does not show an increase in reactivity after an initial drop. MKIA fuel produces plutonium (which is responsible for the reactivity increase in MKIV fuel), but not to a sufficient degree to overcome the loss in ^{235}U . It may be observed, however, that MKIA fuel does indicate flattening of the reactivity curve as ^{239}Pu production partially mitigates ^{235}U depletion. WIMS-E isotopic data were generated at 10-day increments beginning at 2 days and extending to 92 days. This burnup calculation produced macroscopic cross sections that could then be used by other codes. The GOLF code (Schwinkendorf 1994b) was used to generate the finite spherical radii required for $k_{\text{eff}} = 1.0$. These two codes were thus used to predict MKIA k_{∞} and critical mass in an optimal water lattice.

Figure 5-15. k -infinity Versus Exposure
— N Reactor Hot Operating Conditions.

A set of lattice spacings in cold water was considered. Figures 5-16 and 5-17 illustrate these results. WIMS-E exposed the MKIA fuel at full power (12.931 MW/MT) out to 92 days followed by a ten-year "exposure" at 1.0 W/MT. The WIMS-E code thus "decayed" the fuel at essentially zero power for this 10-year period.

For each lattice-spacing value, the GOLF code used the two-group cross sections for the MKIA fuel that were output by WIMS-E. These calculations also assumed full reflection with light water. The water cross sections were generic light-water cross sections.

The cross-section library accessed by the WIMS-E code is the 69-group "1986" WIMS library. In the WIMS-E calculations, the W-PRES and W-RES modules were used to evaluate resonance integrals for the annular geometry. The W-THES module was used to calculate the collision probabilities, and the W-PIP module was used to evaluate the resulting fluxes. The purpose for including these burnup results has been to illustrate the degree of conservatism in place when basing limits on zero exposure. No burnup credit is taken in this document.

Conclusion — Actual exposure (or fuel burnup) of irradiated fuel reduces reactivity and provides a measure of conservatism that increases criticality safety.

Figure 5-16. k -infinity Versus Lattice Spacing
— MKIA Fuel, 92-day Burn, 10-year Decay.

Figure 5-17. Critical Mass Versus Spacing
— MKIA Fuel, 92-day Burn, 10-year Decay.

The cross-section library accessed by the WIMS-E code is the 69-group "1986" WIMS library. In the WIMS-E calculations, the W-PRES and W-RES modules were used to evaluate resonance integrals for the annular geometry. The W-THES module was used to calculate the collision probabilities, and the W-PIP module was used to evaluate the resulting fluxes. The purpose for including these burnup results has been to illustrate the degree of conservatism in place when basing limits on zero exposure. No burnup credit is taken in this document.

Conclusion — Actual exposure (or fuel burnup) of irradiated fuel reduces reactivity and provides a measure of conservatism that increases criticality safety.

6.0 CONCLUSIONS

From a criticality perspective, the storage array of fuel canisters is safe under normal conditions. For this analysis, normal conditions were assumed to be intact MKIA or MKIV assemblies inside vertical canisters without fissile material between the canisters. Several combinations of fuel degradation were analyzed. In general, fuel dissolution will tend to decrease reactivity from the intact configuration. Additional conservatism included modeling the irradiated fuel as unexposed uranium; no credit was taken for fuel burnup. These calculations were all modeled with aluminum canisters. Stainless steel canisters provide an additional safety margin.

The proposed increased spacing of fuel-storage canisters in the storage array (ICF 1994) will not increase array activity, because it was the reduction of spacing between canisters that Wittekind (1993) found increased array reactivity.

Vertical stacking of fuel-storage canisters two or three high is proposed by the fuel consolidation engineering study (ICF 1994). This proposal for stacking was shown in Wittekind (1993) to not be limiting for criticality safety nor to require any administrative controls.

Because of a hypothetical fuel-transfer train accident involving the west bay, enrichment is limited to 0.95-wt% ^{235}U (MKIV fuel), and no fuel above a single layer, such as hanging fuel, is allowed in the west bay. This document did not perform analyses to relax either of these two requirements in the west bay.

The off-normal conditions considered in this analysis included the following:

- Dissolving of fuel assemblies and/or elements into $\text{UO}_2 + \text{H}_2\text{O}$ solution
- An additional inner element resulting in eight assemblies in each canister barrel
- Fuel assemblies and/or elements breaking into more reactive pieces
- Reduction of water between canisters because of basin drain accidents
- Introduction of fissile material between canisters in the storage array.

Introduction of fissile material between canisters in the storage array proved to be the worst contingency of the off-normal conditions considered. Fuel spill out of canisters into the space between canisters or an unrestrained accumulation of fissile material between canisters could result in unacceptably high reactivities. The accumulation of sludge formed on the basin bottom by the corrosion of fuel elements with breached cladding also needs to be controlled. An administrative control on sludge depth between canisters would limit the unacceptably high reactivities that result when fissile material accumulates in the space between canisters. The allowable

safe limit for sludge depth was found to be 9.525 cm (3.75 in.). This limit allows for the layer of optimally reactive sludge to be doubled (to 19.05 cm [7.5 in.]) before the slab-geometry limiting value $k_{eff} = 0.97$ (including code bias and 95% confidence level) is approached.

7.0 REFERENCES

- Briesmeister, J. F., 1986, *MCNP - A General Monte Carlo Code for Neutron and Photon Transport, Version 3A*, LA-7396-M, Rev. 2, Los Alamos National Laboratory, Los Alamos, New Mexico.
- Briesmeister, J. F., and J. Hendricks, 1990, *MCNP4 Newsletter*, X-6:JFB-90-368, Los Alamos National Laboratory, Los Alamos, New Mexico.
- Carter, L. L., 1994, *Certification of MCNP Version 4A for WHC Computer Platforms*, WHC-SD-MP-SWD-30001, Rev. 6, Westinghouse Hanford Company, Richland, Washington.
- Drawing H-1-34762, *Structural Storage Basin Canister Supports Plan & Sections*, March 7, 1979.
- Drawing H-1-34763, *Structural Storage Basin Canister Supports Enlarged Plans & Details*, March 7, 1979.
- Drawing H-1-34764, *Structural Chute Area Canister Storage Plan & Details*, March 7, 1979.
- Drawing H-1-34771, *Storage Basin Canister Layout Full Storage Arrangement*, October 10, 1977.
- Drawing H-1-36935, *Irradiated Fuel Storage Canister*, sheets 1 and 2, April 18, 1980.
- Drawing H-1-42793, *Fuel Encapsulation Program Storage Canister*, sheets 1 and 2, July 15, 1981.
- Drawing H-1-43444, *Fuel Encapsulation MKII Canister Stacking Plate*, sheets 1 and 2, August 12, 1982.
- Drawing H-1-46215, *Fuel Encapsulation MKII Canister Assembly*, sheets 1 through 7, March 14, 1985.
- Erickson, D. G., 1992, *WIMS-E Critical Mass Validation for 1.95 wt% and 3.85 wt% Uranium Billets*, WHC-SD-NR-CSER-004, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Gubbins, M. E., M. J. Roth, and C. J. Taubman, 1982, *A General Introduction to the Use of the WIMS-E Modular Program*, AEEW-R 1329, Reactor Physics Division, AEE Winfrith, England.
- ICF, 1994, *Spent Fuel Consolidation in the 105KW Fuel Storage Basin, Project N-040, N040ES*, ICF Kaiser Hanford Company, Richland, Washington.
- Hellens, R. L., and H. C. Honeck, 1962, "A Summary and Preliminary Analysis of the BNL Slightly Enriched Uranium, Water Moderated Lattice Measurements," *Light Water Lattices*, Brookhaven National Laboratory, International Atomic Energy Agency, Vienna.

- Kupinski, A. F., and H. Toffer, 1970, *Use of the HAMMER System for Evaluating Light-Water Moderated, Critical Assemblies*, DUN-7286, Douglas United Nuclear, Richland, Washington.
- Schwinkendorf, K. N., 1985, *WIMS Critical Mass Validation for 1.95 wt% and 3.85 wt% Uranium Billets*, UNI-3486, UNC Nuclear Industries, Incorporated, Richland, Washington.
- Schwinkendorf, K. N., 1992a, *Criticality Safety Evaluation Report, 105-KE Basin Fuel Encapsulation*, WHC-SD-NR-CSER-007, Rev. 1, Westinghouse Hanford Company, Richland, Washington.
- Schwinkendorf, K. N., 1992b, *Validation of WIMS-E for Prediction of Uranium, Plutonium Nitrate Solution Critical Masses*, WHC-IP-0840-FMEF, Westinghouse Hanford Company, Richland, Washington.
- Schwinkendorf, K. N., 1994a, *Criticality Safety Evaluation Report for 300 Area N Reactor Fuel Fabrication and Storage Facility*, WHC-SD-NR-CSER-010, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Schwinkendorf, K. N., 1994b, *Software Certification Package for the GOLF Code*, WHC-SD-NR-SWD-024, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Thierer, I., 1968, *DCODE, A Fuel Design/Analysis Code For the UNIVAC 1108*, DUN-5330, Douglas United Nuclear, Richland, Washington.
- WHC-CM-4-29, *Nuclear Criticality Safety Manual*, Westinghouse Hanford Company, Richland, Washington.
- Wittekind, W. D., 1991a, *Irradiated N Reactor Fuel Consolidated Storage Shielding Calculations*, WHC-SD-NR-ER-090, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Wittekind, W. D., 1991b, *N Reactor Spent Fuel Compacted Storage Criticality Calculations*, WHC-SD-NR-CSA-004, Westinghouse Hanford Company, Richland, Washington.
- Wittekind, W. D., 1992, *K Basin Criticality Evaluation for Irradiated Fuel Canisters in Sludge*, WHC-SD-NR-CSER-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Wittekind, W. D., 1993, *Criticality Safety Evaluation K Basin Storage Rack Seismic Qualification*, WHC-SD-NR-CSER-009, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Wittekind, W. D., 1994, *MCNP Calculations of K-Infinity for 100KE Basin Sludge Samples*, WHC-SD-NR-CSER-008, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

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APPENDIX A

SAMPLE WIMS-E W-PIJ AND MCNP INPUT FILES

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SAMPLE WIMS-E W-PIJ AND MCNP INPUT FILES

The following is a listing of the WIMS-E input file for the intact MKIA fuel-cluster model for the canister interior, with pure water moderator between fuel canisters. This input file produced $k_{eff} = 0.891233$:

* MKIA IN AL CANISTER MODEL

* SIX ASSEMBLIES, NO DAMAGE

*

XHEAD 1

BURN

NMESH 32

NREGION 10

NMATERIAL 5

ENDP

PRINT 0

CENT

MESH 4 1 4 1 4 1 4 1 11 1

MATERIAL 1 1.0 296.0 3 3001 11.1 6016 88.9

MATERIAL 2 6.55 296.0 2 91 100.0

MATERIAL 3 18.580 296.0 1 2235 0.9471 8238 99.0529

MATERIAL 4 2.70 296.0 2 27 100.0

MATERIAL 5 18.580 296.0 1 2235 1.2491 8238 98.7509

ANNULUS 1 0.5588 1 *INNER WATER REGION

ANNULUS 2 0.6223 2 *INNER CLAD

ANNULUS 3 1.4808 3 *INNER FUEL

ANNULUS 4 1.5824 2 *CLAD

ANNULUS 5 2.2403 1 *WATER

ANNULUS 6 2.2962 2 *CLAD

ANNULUS 7 2.9896 5 *OUTER FUEL

ANNULUS 8 3.0531 2 *OUTER CLAD

ANNULUS 9 3.876 1 *CELL RADIUS

ANNULUS 10 3.877 4 *CANISTER

BEGIN

*

* PRES CALCULATION

*

WPRES 1 2

GROUPS 15 27

NUCLIDE 2235 296.

NUCLIDE 8238 296.

PRINT 0

BEGIN

*

* PERSEUS CALCULATIONS

*

WTHES 2

PRINT 0

BEGIN

*

WRES 2 3

GROUPS 15 27

NUCLIDE 2235 296.

NUCLIDE 8238 296.

PRINT 0

BEGIN

*

* PERSEUS CALCULATIONS

*

WTHES 3

PRINT 0

BEGIN

*

WPIP 3

PRINT 0

TOLER 1.0E-5

OMEGA 1.25

BEGIN

```

*
* COLLAPSING TO 18 GROUPS TO DO CLUSTER PIJ
*
WCOND 3 4
GROUP 18
ENDP
PART 2 5 10 14 23 24 26 27 36 39 45 48 53 56 60 63 66 69
BEGIN
*
* MERGE
*
WMERGE 4 5
NOIF 1
ENDP
BEGIN
*
* PIJ CALCULATIONS FOR FUEL STORAGE CANISTER
*
WPIJ 5
CELLDIM 59 11 1 1 1 1
RODDIM 6 6 8 1
CONTROL 1 46 84 1 0 3
ENDP
CENT
ADEC
PRINT 1
ANNULUS -1 0.5588 1 *INNER WATER REGION
ANNULUS -2 0.6223 2 *INNER CLAD
ANNULUS -3 1.4808 3 *INNER FUEL
ANNULUS -4 1.5824 2 *CLAD
ANNULUS -5 2.2403 1 *WATER
ANNULUS -6 2.2962 2 *CLAD
ANNULUS -7 2.9896 5 *OUTER FUEL
ANNULUS -8 3.0531 2 *OUTER CLAD
ANNULUS -9 10.239 1 *WATER INSIDE CANISTER
ANNULUS -10 10.469 4 *ALUMINUM CANISTER
ANNULUS -11 14.644 1 *WATER MODERATOR
SURFACE 0.0
ARRAY 1 1 1 6.667 0.0
RODSUB 1 1 0.5588 1 *INNER WATER REGION
RODSUB 1 2 0.6223 2 *INNER CLAD
RODSUB 1 3 1.4808 3 *INNER FUEL
RODSUB 1 4 1.5824 2 *CLAD
RODSUB 1 5 2.2403 1 *WATER
RODSUB 1 6 2.2962 2 *CLAD
RODSUB 1 7 2.9896 5 *OUTER FUEL
RODSUB 1 8 3.0531 2 *OUTER CLAD
ARRAY 2 1 1 6.667 60.0
RODSUB 2 1 0.5588 1 *INNER WATER REGION
RODSUB 2 2 0.6223 2 *INNER CLAD
RODSUB 2 3 1.4808 3 *INNER FUEL
RODSUB 2 4 1.5824 2 *CLAD
RODSUB 2 5 2.2403 1 *WATER
RODSUB 2 6 2.2962 2 *CLAD
RODSUB 2 7 2.9896 5 *OUTER FUEL
RODSUB 2 8 3.0531 2 *OUTER CLAD
ARRAY 3 1 1 6.667 120.0
RODSUB 3 1 0.5588 1 *INNER WATER REGION
RODSUB 3 2 0.6223 2 *INNER CLAD
RODSUB 3 3 1.4808 3 *INNER FUEL
RODSUB 3 4 1.5824 2 *CLAD
RODSUB 3 5 2.2403 1 *WATER
RODSUB 3 6 2.2962 2 *CLAD
RODSUB 3 7 2.9896 5 *OUTER FUEL
RODSUB 3 8 3.0531 2 *OUTER CLAD
ARRAY 4 1 1 6.667 180.0
RODSUB 4 1 0.5588 1 *INNER WATER REGION
RODSUB 4 2 0.6223 2 *INNER CLAD
RODSUB 4 3 1.4808 3 *INNER FUEL
RODSUB 4 4 1.5824 2 *CLAD
RODSUB 4 5 2.2403 1 *WATER
RODSUB 4 6 2.2962 2 *CLAD
RODSUB 4 7 2.9896 5 *OUTER FUEL
RODSUB 4 8 3.0531 2 *OUTER CLAD
ARRAY 5 1 1 6.667 240.0
RODSUB 5 1 0.5588 1 *INNER WATER REGION
RODSUB 5 2 0.6223 2 *INNER CLAD

```

```

RODSUB 5 3 1.4808 3 *INNER FUEL
RODSUB 5 4 1.5824 2 *CLAD
RODSUB 5 5 2.2403 1 *WATER
RODSUB 5 6 2.2962 2 *CLAD
RODSUB 5 7 2.9896 5 *OUTER FUEL
RODSUB 5 8 3.0531 2 *OUTER CLAD
ARRAY 6 1 1 6.667 300.0
RODSUB 6 1 0.5588 1 *INNER WATER REGION
RODSUB 6 2 0.6223 2 *INNER CLAD
RODSUB 6 3 1.4808 3 *INNER FUEL
RODSUB 6 4 1.5824 2 *CLAD
RODSUB 6 5 2.2403 1 *WATER
RODSUB 6 6 2.2962 2 *CLAD
RODSUB 6 7 2.9896 5 *OUTER FUEL
RODSUB 6 8 3.0531 2 *OUTER CLAD
*
LINES 4 1 4 1 4 1 4 1 20 1 5
BVECTOR 0 1.00
BEGIN
WPIP 5
*
* PERFORM K-EFF CALCULATION -- AXIAL BUCKLING CORRECTION
*
BUCKLING 1.62E-03 1
BEGIN
STOP

```

The following WIMS-E input shows how intact MKIA assemblies, containing the same seven-assembly cluster arrangement as the input file above, were smeared into the space between fuel canisters. This input produced $k_{eff} = 1.070669$:

```

*****
* MKIA IN AL CANISTER MODEL
* SIX ASSEMBLIES, NO DAMAGE
*****
*
XHEAD 1
BURN
NMESH 32
NREGION 10
NMATERIAL 5
ENDP
PRINT 0
CENT
MESH 4 1 4 1 4 1 4 1 11 1
MATERIAL 1 1.0 296.0 3 3001 11.1 6016 88.9
MATERIAL 2 6.55 296.0 2 91 100.0
MATERIAL 3 18.580 296.0 1 2235 0.9471 8238 99.0529
MATERIAL 4 2.70 296.0 2 27 100.0
MATERIAL 5 18.580 296.0 1 2235 1.2491 8238 98.7509
ANNULUS 1 0.5588 1 *INNER WATER REGION
ANNULUS 2 0.6223 2 *INNER CLAD
ANNULUS 3 1.4808 3 *INNER FUEL
ANNULUS 4 1.5824 2 *CLAD
ANNULUS 5 2.2403 1 *WATER
ANNULUS 6 2.2962 2 *CLAD
ANNULUS 7 2.9896 5 *OUTER FUEL
ANNULUS 8 3.0531 2 *OUTER CLAD
ANNULUS 9 3.876 1 *CELL RADIUS
ANNULUS 10 3.877 4 *CANISTER
BEGIN
*
* PRES CALCULATION
*
WPRES 1 2
GROUPS 15 27
NUCLIDE 2235 296.
NUCLIDE 8238 296.
PRINT 0
BEGIN
*
* PERSEUS CALCULATIONS
*
WTHES 2

```

```

PRINT      0
BEGIN
*
WRES      2  3
GROUPS    15 27
NUCLIDE   2235 296.
NUCLIDE   8238 296.
PRINT      0
BEGIN
*
* PERSEUS CALCULATIONS
*
WTHES      3
PRINT      0
BEGIN
*
WPIP      3
PRINT      0
TOLER     1.0E-5
OMEGA     1.25
BEGIN
*
* COLLAPSING TO 18 GROUPS TO DO CLUSTER PIJ
*
WCOND      3  4
GROUP      18
ENDP
PART      2 5 10 14 23 24 26 27 36 39 45 48 53 56 60 63 66 69
BEGIN
*****
* MKIA FUEL ASSEMBLY MODEL
*****
*
XHEAD      5
BURN
NMESH      31
NREGION    9
NMATERIAL   4
ENDP
PRINT      0
CENT
MESH      4 1 4 1 4 1 4 1 11
MATERIAL 1 1.0      296.0 3 3001 11.1 6016 88.9
MATERIAL 2 6.55     296.0 2 91 100.0
MATERIAL 3 18.580    296.0 1 2235 0.9471 8238 99.0529
MATERIAL 4 18.580    296.0 1 2235 1.2491 8238 98.7509
ANNULUS 1 0.5588     1 *INNER WATER REGION
ANNULUS 2 0.6223     2 *INNER CLAD
ANNULUS 3 1.4808     3 *INNER FUEL
ANNULUS 4 1.5824     2 *CLAD
ANNULUS 5 2.2403     1 *WATER
ANNULUS 6 2.2962     2 *CLAD
ANNULUS 7 2.9896     4 *OUTER FUEL
ANNULUS 8 3.0531     2 *OUTER CLAD
ANNULUS 9 4.13415    1 *OPTIMUM CELL RADIUS
BEGIN
*
* PRES CALCULATION
*
WPRES      5  6
GROUPS    15 27
NUCLIDE   2235 296.
NUCLIDE   8238 296.
PRINT      0
BEGIN

```

```

*
* PERSEUS CALCULATIONS
*
WTHES      6
PRINT      0
BEGIN
*
WRES      6 7
GROUPS    15 27
NUCLIDE   2235 296.
NUCLIDE   8238 296.
PRINT      0
BEGIN
*
* PERSEUS CALCULATIONS
*
WTHES      7
PRINT      0
BEGIN
*
WPIP      7
PRINT      0
TOLER     1.0E-5
OMEGA     1.25
BEGIN
*
* COLLAPSING TO 18 GROUPS TO DO CLUSTER PIJ
*
WCOND     7 8
GROUP     18
ENDP
PART      2 5 10 14 23 24 26 27 36 39 45 48 53 56 60 63 66 69
BEGIN
*
* SMEAR THE CELL INTO ONE REGION
*
WSMEAR    8 9
MATE      1
MESH      1
ENDP
NEWMAT    1 1 2 3 4 5 6 7 8 9 10
          11 12 13 14 15 16 17 18 19 20
          21 22 23 24 25 26 27 28 29 30
          31
MCODE     1
BEGIN
*
* MERGE
*
WMERGE    4 9 10
NOIF      2
ENDP
BEGIN
*
* PIJ CALCULATIONS FOR FUEL STORAGE CANISTER
*
WPIJ      10
CELDDIM   59 11 1 1 1 1
RODDIM    6 6 8 1
CONTROL   1 46 84 1 0 3
ENDP
CENT
ADEC
PRINT     1
ANNULUS   -1 0.5588 1 *INNER WATER REGION
ANNULUS   -2 0.6223 2 *INNER CLAD
ANNULUS   -3 1.4808 3 *INNER FUEL
ANNULUS   -4 1.5824 2 *CLAD
ANNULUS   -5 2.2403 1 *WATER
ANNULUS   -6 2.2962 2 *CLAD
ANNULUS   -7 2.9896 5 *OUTER FUEL
ANNULUS   -8 3.0531 2 *OUTER CLAD
ANNULUS   -9 10.239 1 *WATER INSIDE CANISTER
ANNULUS  -10 10.469 4 *ALUMINUM CANISTER
ANNULUS  -11 14.644 6 *MKIA ASSEMBLIES IN WATER
SURFACE   0.0
ARRAY     1 1 1 6.667 0.0

```

```

RODSUB 1 1 0.5588 1 *INNER WATER REGION
RODSUB 1 2 0.6223 2 *INNER CLAD
RODSUB 1 3 1.4808 3 *INNER FUEL
RODSUB 1 4 1.5824 2 *CLAD
RODSUB 1 5 2.2403 1 *WATER
RODSUB 1 6 2.2962 2 *CLAD
RODSUB 1 7 2.9896 5 *OUTER FUEL
RODSUB 1 8 3.0531 2 *OUTER CLAD
ARRAY 2 1 1 6.667 60.0
RODSUB 2 1 0.5588 1 *INNER WATER REGION
RODSUB 2 2 0.6223 2 *INNER CLAD
RODSUB 2 3 1.4808 3 *INNER FUEL
RODSUB 2 4 1.5824 2 *CLAD
RODSUB 2 5 2.2403 1 *WATER
RODSUB 2 6 2.2962 2 *CLAD
RODSUB 2 7 2.9896 5 *OUTER FUEL
RODSUB 2 8 3.0531 2 *OUTER CLAD
ARRAY 3 1 1 6.667 120.0
RODSUB 3 1 0.5588 1 *INNER WATER REGION
RODSUB 3 2 0.6223 2 *INNER CLAD
RODSUB 3 3 1.4808 3 *INNER FUEL
RODSUB 3 4 1.5824 2 *CLAD
RODSUB 3 5 2.2403 1 *WATER
RODSUB 3 6 2.2962 2 *CLAD
RODSUB 3 7 2.9896 5 *OUTER FUEL
RODSUB 3 8 3.0531 2 *OUTER CLAD
ARRAY 4 1 1 6.667 180.0
RODSUB 4 1 0.5588 1 *INNER WATER REGION
RODSUB 4 2 0.6223 2 *INNER CLAD
RODSUB 4 3 1.4808 3 *INNER FUEL
RODSUB 4 4 1.5824 2 *CLAD
RODSUB 4 5 2.2403 1 *WATER
RODSUB 4 6 2.2962 2 *CLAD
RODSUB 4 7 2.9896 5 *OUTER FUEL
RODSUB 4 8 3.0531 2 *OUTER CLAD
ARRAY 5 1 1 6.667 240.0
RODSUB 5 1 0.5588 1 *INNER WATER REGION
RODSUB 5 2 0.6223 2 *INNER CLAD
RODSUB 5 3 1.4808 3 *INNER FUEL
RODSUB 5 4 1.5824 2 *CLAD
RODSUB 5 5 2.2403 1 *WATER
RODSUB 5 6 2.2962 2 *CLAD
RODSUB 5 7 2.9896 5 *OUTER FUEL
RODSUB 5 8 3.0531 2 *OUTER CLAD
ARRAY 6 1 1 6.667 300.0
RODSUB 6 1 0.5588 1 *INNER WATER REGION
RODSUB 6 2 0.6223 2 *INNER CLAD
RODSUB 6 3 1.4808 3 *INNER FUEL
RODSUB 6 4 1.5824 2 *CLAD
RODSUB 6 5 2.2403 1 *WATER
RODSUB 6 6 2.2962 2 *CLAD
RODSUB 6 7 2.9896 5 *OUTER FUEL
RODSUB 6 8 3.0531 2 *OUTER CLAD

```

*
 LINES 4 1 4 1 4 1 4 1 20 1 5

BVECTOR 0 1.00

BEGIN

WPIP 10

*

* PERFORM K-EFF CALCULATION -- AXIAL BUCKLING CORRECTION

*

BUCKLING 1.62E-03 1

BEGIN

STOP

The following is a listing of the FORTRAN code used to produce the WIMS-E input files that produced k_{∞} as a function of the hydrogen-to-uranium atomic ratio for any user-input uranium enrichment. There were sixteen points plotted on these curves, so a pre-processor code was written to expedite the creation of input files for these cases.

```

program doit
dimension aw(4)
real mwwt
data aw / 235.0439, 238.0508, 15.9994, 1.0079 /
open(unit=4,file='nhud',status='old',err=99)
open(unit=7,file='soln',status='unknown',err=99)
mwwt = aw(3) + 2.0*aw(4)
write (6,*) ' What is the uranium enrichment? (wt% u-235)'
read (5,*) en
awu = 1./((en/100.)/aw(1) + (1-en/100.)/aw(2))
write (6,*) ' Atomic weight of U = ',awu
iu = 7
c write (6,*) ' How many H/U ratios do you want'
read (4,*) nhu
call front(iu,nhu,en)
c*****loop for H/U ratio ...
do 40 icycle = 1,nhu
  read (4,*) htu
  wtu = awu/(awu + 0.5*mwwt*htu)
  dmix = 1./(wtu/18.82 + (1.-wtu))
  w235 = wtu*en
  w238 = wtu*(100.-en)
  whyd = (1. - wtu)*11.19
  woxy = (1. - wtu)*88.81
  write(iu,101) icycle, htu
  write(iu,102) dmix
  write(iu,103) w235, w238
  write(iu,104) whyd, woxy
101  format(1x,'qual 3 ',i2,' * h/u = ',f7.1)
102  format(1x,'material 1 ',f8.5,' 296.0 1 ')
103  format(20x,' 2235 ',f10.6,' 8238 ',f9.5)
104  format(20x,' 3001 ',f10.6,' 6016 ',f9.5)
  40  continue
  call end(iu)
99  stop
end

subroutine front(i,nhu,en)
write(i,*)'*****'
write(i,10) en
10  format(1x,'* ',f7.4,' wt% u solutions: u and water')
write(i,*)'*****'
write(i,9) nhu
9  format(1x,'ncycle 1',i3)
write(i,*)'*'
write(i,*)'whead      1'
write(i,*)'nmesh      7'
write(i,*)'nregion    1'
write(i,*)'nmaterial  1'
write(i,*)'endp'
write(i,*)'print      0'
write(i,*)'cent'
write(i,*)'mesh      7'
return
end

subroutine end(i)
write(i,*)'annulus 1 4.0000 1 * infinite region of solution'
write(i,*)'begin'
write(i,*)'*'
write(i,*)'* perseus calculations'
write(i,*)'*'
write(i,*)'wthes      1'
write(i,*)'print      0'
write(i,*)'begin'
write(i,*)'*'
write(i,*)'wpip      1'
write(i,*)'print      0'
write(i,*)'toler 1.0e-5'

```

```

write(i,*)'omega 1.25'
write(i,*)'begin'
write(i,*)''
write(i,*)'* smear all meshes into one material'
write(i,*)''
write(i,*)'wsmeas 1 2'
write(i,*)'mate 1'
write(i,*)'mesh 1'
write(i,*)'endp'
write(i,*)'newmat 1 1 2 3 4 5 6 7'
write(i,*)'mcode 1'
write(i,*)'begin'
write(i,*)''
write(i,*)'* collapse to 2 groups'
write(i,*)''
write(i,*)'wcond 2 3'
write(i,*)'group 2'
write(i,*)'endp'
write(i,*)'part 27 69'
write(i,*)'begin'
write(i,*)''
write(i,*)'* print 2-group cross sections save it on unit 7'
write(i,*)''
write(i,*)'winter 3'
write(i,*)'print'
write(i,*)'file 3'
write(i,*)'matrix'
write(i,*)'snap'
write(i,*)'begin'
write(i,*)'finish'
write(i,*)'stop'
return
end

```


The following is a listing of the MCNP input file for the case with intact MKIA fuel inside an array of single-stacked canisters with pure water between the canisters. This input file produced $k_{eff} = 0.70208 \pm 0.00206$:

message:

```

3D N Reactor Mark IA MkIA in Canister double layer, 4 side reflection
c      across canister major axis x, to next canister y, vertical z
1 1 -2.37 1 -2 3 -4 5 -6 $ concrete basin floor
2 2 -8.030 6 -7 -30 $ canister base plate
3 4 -1.000 -25 7 -10 31 $ water-canister MkIA #1
4 5 -6.550 7 -10 32 -31 $ zirc-2 clad MkIA #1
5 5 -6.550 7 -8 33 -32 $ end cap lower MkIA #1
6 6 -18.820 8 -9 33 -32 $ fuel-outer MkIA #1
7 5 -6.550 9 -10 33 -32 $ end cap upper MkIA #1
8 5 -6.550 7 -10 34 -33 $ zirc-2 clad MkIA #1
9 4 -1.000 7 -10 35 -34 $ inner annulus MkIA #1
10 5 -6.550 7 -10 36 -35 $ zirc-2 clad MkIA #1
11 5 -6.550 7 -8 37 -36 $ end cap lower MkIA #1
12 7 -18.820 8 -9 37 -36 $ fuel-inner MkIA #1
13 5 -6.550 9 -10 37 -36 $ end cap upper MkIA #1
14 5 -6.550 7 -10 38 -37 $ zirc-2 clad MkIA #1
15 4 -1.000 7 -10 -38 $ center hole MkIA #1
16 4 -1.000 27 -26 25 -30 7 -10 39 $ water-canister MkIA #2
17 5 -6.550 7 -10 40 -39 $ zirc-2 clad MkIA #2
18 5 -6.550 7 -8 41 -40 $ end cap lower MkIA #2
19 6 -18.820 8 -9 41 -40 $ fuel-outer MkIA #2
20 5 -6.550 9 -10 41 -40 $ end cap upper MkIA #2
21 5 -6.550 7 -10 42 -41 $ zirc-2 clad MkIA #2
22 4 -1.000 7 -10 43 -42 $ inner annulus MkIA #2
23 5 -6.550 7 -10 44 -43 $ zirc-2 clad MkIA #2
24 5 -6.550 7 -8 45 -44 $ end cap lower MkIA #2
25 7 -18.820 8 -9 45 -44 $ fuel-inner MkIA #2
26 5 -6.550 9 -10 45 -44 $ end cap upper MkIA #2
27 5 -6.550 7 -10 46 -45 $ zirc-2 clad MkIA #2
28 4 -1.000 7 -10 -46 $ center hole MkIA #2
29 4 -1.000 28 -27 25 -30 7 -10 47 $ water-canister MkIA #3
30 5 -6.550 7 -10 48 -47 $ zirc-2 clad MkIA #3
31 5 -6.550 7 -8 49 -48 $ end cap lower MkIA #3
32 6 -18.820 8 -9 49 -48 $ fuel-outer MkIA #3
33 5 -6.550 9 -10 49 -48 $ end cap upper MkIA #3
34 5 -6.550 7 -10 50 -49 $ zirc-2 clad MkIA #3
35 4 -1.000 7 -10 51 -50 $ inner annulus MkIA #3
36 5 -6.550 7 -10 52 -51 $ zirc-2 clad MkIA #3
37 5 -6.550 7 -8 53 -52 $ end cap lower MkIA #3
38 7 -18.820 8 -9 53 -52 $ fuel-inner MkIA #3
39 5 -6.550 9 -10 53 -52 $ end cap upper MkIA #3
40 5 -6.550 7 -10 54 -53 $ zirc-2 clad MkIA #3
41 4 -1.000 7 -10 -54 $ center hole MkIA #3
42 4 -1.000 -26 -28 25 -30 7 -10 55 $ water-canister MkIA #4
43 5 -6.550 7 -10 56 -55 $ zirc-2 clad MkIA #4
44 5 -6.550 7 -8 57 -56 $ end cap lower MkIA #4
45 6 -18.820 8 -9 57 -56 $ fuel-outer MkIA #4
46 5 -6.550 9 -10 57 -56 $ end cap upper MkIA #4
47 5 -6.550 7 -10 58 -57 $ zirc-2 clad MkIA #4
48 4 -1.000 7 -10 59 -58 $ inner annulus MkIA #4
49 5 -6.550 7 -10 60 -59 $ zirc-2 clad MkIA #4
50 5 -6.550 7 -8 61 -60 $ end cap lower MkIA #4
51 7 -18.820 8 -9 61 -60 $ fuel-inner MkIA #4
52 5 -6.550 9 -10 61 -60 $ end cap upper MkIA #4
53 5 -6.550 7 -10 62 -61 $ zirc-2 clad MkIA #4
54 4 -1.000 7 -10 -62 $ center hole MkIA #4
55 4 -1.000 26 -27 25 -30 7 -10 63 $ water-canister MkIA #5
56 5 -6.550 7 -10 64 -63 $ zirc-2 clad MkIA #5
57 5 -6.550 7 -8 65 -64 $ end cap lower MkIA #5
58 6 -18.820 8 -9 65 -64 $ fuel-outer MkIA #5
59 5 -6.550 9 -10 65 -64 $ end cap upper MkIA #5
60 5 -6.550 7 -10 66 -65 $ zirc-2 clad MkIA #5
61 4 -1.000 7 -10 67 -66 $ inner annulus MkIA #5
62 5 -6.550 7 -10 68 -67 $ zirc-2 clad MkIA #5
63 5 -6.550 7 -8 69 -68 $ end cap lower MkIA #5
64 7 -18.820 8 -9 69 -68 $ fuel-inner MkIA #5
65 5 -6.550 9 -10 69 -68 $ end cap upper MkIA #5
66 5 -6.550 7 -10 70 -69 $ zirc-2 clad MkIA #5
67 4 -1.000 7 -10 -70 $ center hole MkIA #5

```

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68	4	-1.000	27 -28 25 -30	7 -10 71	\$ water-canister	MkIA #6
69	5	-6.550		7 -10 72 -71	\$ zirc-2 clad	MkIA #6
70	5	-6.550		7 -8 73 -72	\$ end cap lower	MkIA #6
71	6	-18.820		8 -9 73 -72	\$ fuel-outer	MkIA #6
72	5	-6.550		9 -10 73 -72	\$ end cap upper	MkIA #6
73	5	-6.550		7 -10 74 -73	\$ zirc-2 clad	MkIA #6
74	4	-1.000		7 -10 75 -74	\$ inner annulus	MkIA #6
75	5	-6.550		7 -10 76 -75	\$ zirc-2 clad	MkIA #6
76	5	-6.550		7 -8 77 -76	\$ end cap lower	MkIA #6
77	7	-18.820		8 -9 77 -76	\$ fuel-inner	MkIA #6
78	5	-6.550		9 -10 77 -76	\$ end cap upper	MkIA #6
79	5	-6.550		7 -10 78 -77	\$ zirc-2 clad	MkIA #6
80	4	-1.000		7 -10 -78	\$ center hole	MkIA #6
81	4	-1.000	28 26 25 -30	7 -10 79	\$ water-canister	MkIA #7
82	5	-6.550		7 -10 80 -79	\$ zirc-2 clad	MkIA #7
83	5	-6.550		7 -8 81 -80	\$ end cap lower	MkIA #7
84	6	-18.820		8 -9 81 -80	\$ fuel-outer	MkIA #7
85	5	-6.550		9 -10 81 -80	\$ end cap upper	MkIA #7
86	5	-6.550		7 -10 82 -81	\$ zirc-2 clad	MkIA #7
87	4	-1.000		7 -10 83 -82	\$ inner annulus	MkIA #7
88	5	-6.550		7 -10 84 -83	\$ zirc-2 clad	MkIA #7
89	5	-6.550		7 -8 85 -84	\$ end cap lower	MkIA #7
90	7	-18.820		8 -9 85 -84	\$ fuel-inner	MkIA #7
91	5	-6.550		9 -10 85 -84	\$ end cap upper	MkIA #7
92	5	-6.550		7 -10 86 -85	\$ zirc-2 clad	MkIA #7
93	4	-1.000		7 -10 -86	\$ center hole	MkIA #7
94	4	-1.000		10 -87	-30 \$ canister water	
95	9	-1.14e-3		87 -11	-30 \$ nitrogen gas	
96	2	-8.030		11 -12	-30 \$ canister lid	
97	2	-8.030		6 -12 30 -29	\$ canister tube	
98	3	-1.000	1 -2 3 -4 6 -12 29		\$ water-basin	
99	3	-1.000	1 -2 3 -4 12 -13		\$ water-basin	
100	3	-1.000	1 -2 3 -4 13 -14		\$ water-basin	
101	3	-1.000	1 -2 3 -4 14 -20		\$ water-basin	
102	3	-1.000	1 -2 3 -4 20 -21		\$ water-basin +100 cm	
103	3	-1.000	1 -2 3 -4 21 -22		\$ water-basin +200 cm	
104	3	-1.000	1 -2 3 -4 22 -23		\$ water-basin +300 cm	
105	3	-1.000	1 -2 3 -4 23 -24		\$ water-basin to surface	
106	0		-1: 2: -3: 4: -5: 24		\$ outside world	
*1	px	-12.6546			\$ side adjacent canister rack	
*2	px	12.3825			\$ midline canister (between barrels)	
*3	py	-13.4541			\$ back adjacent canister rack	
*4	py	13.4541			\$ front adjacent canister rack	
5	pz	-60.96			\$ bottom of basin concrete floor	
6	pz	0.0			\$ canister bottom = basin floor	
7	pz	0.635			\$ canister bottom plate top	
8	pz	1.1176			\$ top of lower MkIA end cap	
9	pz	53.1876			\$ bottom of upper MkIA end cap	
10	pz	53.6702			\$ top of upper MkIA end cap	
11	pz	67.4878			\$ bottom of canister top cover	
12	pz	68.7578			\$ top of canister top cover	
13	pz	73.66			\$ canister top = stacking plate bottom	
14	pz	74.295			\$ stacking plate top = canister bottom	
15	pz	74.930			\$ canister bottom plate top	
16	pz	75.4126			\$ top of lower MkIA end cap	
17	pz	127.330			\$ bottom of upper MkIA end cap	
18	pz	127.965			\$ top of upper MkIA end cap	
19	pz	141.7828			\$ bottom of canister top cover	
20	pz	143.0528			\$ top of canister top cover	
21	pz	243.0528			\$ +100 cm of basin water	
22	pz	343.0528			\$ +200 cm of basin water	
23	pz	443.0528			\$ +300 cm of basin water	
24	pz	487.68			\$ surface of basin water	
25	cz		3.876		\$ cell definition useful surface	
26	p	-0.57735	1.0 0.0 0.0		\$ cell definition useful surface	
27	p	0.57735	1.0 0.0 0.0		\$ cell definition useful surface	
28	px	0.0			\$ cell definition useful surface	

29	cz		10.406	\$ Canister Outside radius
30	cz		10.254	\$ Canister Inside radius
31	cz		3.0531	\$ MkIA #1 zirc-2 clad
32	cz		2.9896	\$ MkIA #1 fuel-outer
33	cz		2.2962	\$ MkIA #1 zirc-2 clad
34	cz		2.2403	\$ MkIA #1 water inner annulus
35	cz		1.5824	\$ MkIA #1 zirc-2 clad
36	cz		1.4808	\$ MkIA #1 fuel-inner
37	cz		0.6223	\$ MkIA #1 zirc-2 cla
38	cz		0.5588	\$ MkIA #1 water center hole
39	c/z	7.06	0.0	\$ MkIA #2 zirc-2 clad
40	c/z	7.06	0.0	\$ MkIA #2 fuel-outer
41	c/z	7.06	0.0	\$ MkIA #2 zirc-2 clad
42	c/z	7.06	0.0	\$ MkIA #2 water inner annulus
43	c/z	7.06	0.0	\$ MkIA #2 zirc-2 clad
44	c/z	7.06	0.0	\$ MkIA #2 fuel-inner
45	c/z	7.06	0.0	\$ MkIA #2 zirc-2 clad
46	c/z	7.06	0.0	\$ MkIA #2 water center hole
47	c/z	3.53	-6.12	\$ MkIA #3 zirc-2 clad
48	c/z	3.53	-6.12	\$ MkIA #3 fuel-outer
49	c/z	3.53	-6.12	\$ MkIA #3 zirc-2 clad
50	c/z	3.53	-6.12	\$ MkIA #3 water inner annulus
51	c/z	3.53	-6.12	\$ MkIA #3 zirc-2 clad
52	c/z	3.53	-6.12	\$ MkIA #3 fuel-inner
53	c/z	3.53	-6.12	\$ MkIA #3 zirc-2 clad
54	c/z	3.53	-6.12	\$ MkIA #3 water center hole
55	c/z	-3.53	-6.12	\$ MkIA #4 zirc-2 clad
56	c/z	-3.53	-6.12	\$ MkIA #4 fuel-outer
57	c/z	-3.53	-6.12	\$ MkIA #4 zirc-2 clad
58	c/z	-3.53	-6.12	\$ MkIA #4 water inner annulus
59	c/z	-3.53	-6.12	\$ MkIA #4 zirc-2 clad
60	c/z	-3.53	-6.12	\$ MkIA #4 fuel-inner
61	c/z	-3.53	-6.12	\$ MkIA #4 zirc-2 clad
62	c/z	-3.53	-6.12	\$ MkIA #4 water center hole
63	c/z	-7.06	0.0	\$ MkIA #5 zirc-2 clad
64	c/z	-7.06	0.0	\$ MkIA #5 fuel-outer
65	c/z	-7.06	0.0	\$ MkIA #5 zirc-2 clad
66	c/z	-7.06	0.0	\$ MkIA #5 water inner annulus
67	c/z	-7.06	0.0	\$ MkIA #5 zirc-2 clad
68	c/z	-7.06	0.0	\$ MkIA #5 fuel-inner
69	c/z	-7.06	0.0	\$ MkIA #5 zirc-2 clad
70	c/z	-7.06	0.0	\$ MkIA #5 water center hole
71	c/z	-3.53	6.12	\$ MkIA #6 zirc-2 clad
72	c/z	-3.53	6.12	\$ MkIA #6 fuel-outer
73	c/z	-3.53	6.12	\$ MkIA #6 zirc-2 clad
74	c/z	-3.53	6.12	\$ MkIA #6 water inner annulus
75	c/z	-3.53	6.12	\$ MkIA #6 zirc-2 clad
76	c/z	-3.53	6.12	\$ MkIA #6 fuel-inner
77	c/z	-3.53	6.12	\$ MkIA #6 zirc-2 clad
78	c/z	-3.53	6.12	\$ MkIA #6 water center hole
79	c/z	3.53	6.12	\$ MkIA #7 zirc-2 clad
80	c/z	3.53	6.12	\$ MkIA #7 fuel-outer
81	c/z	3.53	6.12	\$ MkIA #7 zirc-2 clad
82	c/z	3.53	6.12	\$ MkIA #7 water inner annulus
83	c/z	3.53	6.12	\$ MkIA #7 zirc-2 clad
84	c/z	3.53	6.12	\$ MkIA #7 fuel-inner
85	c/z	3.53	6.12	\$ MkIA #7 zirc-2 clad
86	c/z	3.53	6.12	\$ MkIA #7 water center hole
87	pz	66.4718		\$ Canister water surface (0.4 inches Nitrogen)

mode n			
imp:n	1	104r	0.0
vol	105j		
	1j		
tmp	2.585e-8	104r	
	1j		
kcode	1000	1.0	10 100

\$ 001-242	everywhere
\$ 243	outside world
\$ 001-242 300K	everywhere
\$ 243	outside world

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ksrc	-2.628	0.000	30.4673	\$ MkIA #1 point 1
	2.628	0.000	30.4673	\$ MkIA #1 point 2
	7.060	-2.628	30.4673	\$ MkIA #2 point 1
	7.060	2.628	30.4673	\$ MkIA #2 point 2
	3.530	-8.748	30.4673	\$ MkIA #3 point 1
	3.530	-3.492	30.4673	\$ MkIA #3 point 2
	-3.530	-8.748	30.4673	\$ MkIA #4 point 1
	-3.530	-3.492	30.4673	\$ MkIA #4 point 2
	-7.060	-2.628	30.4673	\$ MkIA #5 point 1
	-7.060	2.628	30.4673	\$ MkIA #5 point 2
	3.530	8.748	30.4673	\$ MkIA #6 point 1
	3.530	3.492	30.4673	\$ MkIA #6 point 2
	-3.530	8.748	30.4673	\$ MkIA #7 point 1
	-3.530	3.492	30.4673	\$ MkIA #7 point 2
m1	26000.55c	-0.00312		\$ normal concrete (2.37 g/cc)
	1001.50c	-0.00206		
	8016.50c	-0.49335		
	12000.50c	-0.00209		
	20000.50c	-0.24959		
	14000.50c	-0.18816		
	13027.50c	-0.00510		
	16032.50c	-0.00082		
	6000.50c	-0.05571		
mt1	poly.01t			
m2	6000.50c	-0.0004		\$ stainless steel 304 (8.03 g/cc)
	25055.50c	-0.0200		
	14000.50c	-0.0100		
	24000.50c	-0.1900		
	28000.50c	-0.0925		
	26000.55c	-0.6871		
m3	1001.50c	0.66667		\$ basin water
	8016.50c	0.33333		
mt3	lwtr.01t			\$ 300 K = 17 C = 62.6 F
m4	1001.50c	0.66667		\$ canister water
	8016.50c	0.33333		
mt4	lwtr.01t			\$ 300 K = 17 C = 62.6 F
m5	40000.50c	-1.00000		\$ zirconium
m6	92235.50c	-0.012491		\$ outer fuel (1.2491 wt% U-235)
	92236.50c	-0.000392		
	92238.50c	-0.987117		
m7	92235.50c	-0.009471		\$ inner fuel (0.9471 wt% U-235)
	92236.50c	-0.000392		
	92238.50c	-0.990137		
m8	29000.50c	-0.0025		\$ aluminum 6061 t6 (2.71 g/cc)
	14000.50c	-0.0060		
	12000.50c	-0.0100		
	24000.50c	-0.0025		
	13027.50c	-0.9790		
m9	7014.50c	-1.000		\$ nitrogen gas (1.138e-3 g/cc)
totnu				
ctme	480.			
print	30 40 60 110 126			

The following is a listing of the MCNP file that produced a $k_{eff} = 0.98$ for sludge accumulation outside and around the bottom of a single-stacked layer of fuel-storage canisters. This sludge was modeled as optimally sized and spaced 1.25-wt% ^{235}U -enriched uranium metal rods; the sludge level was adjusted to yield the limiting value $k_{eff} = 0.98$. The surface number 86 defines this sludge level.

message:

```

3D N Reactor Mark IA MkIA in Canister double layer, 4 side reflection
c across canister major axis x, to next canister y, vertical z
1 1 -2.37 1 -2 3 -4 5 -6 $ concrete basin floor
2 2 -8.030 6 -7 -30 $ canister base plate
c 4 4 -1.000 10 -87 -30 $ canister water
5 9 -1.14e-3 87 -11 -30 $ nitrogen gas
6 2 -8.030 11 -12 -30 $ canister lid
7 2 -8.030 6 -12 30 -29 $ canister tube
8 0 1 -2 3 -4 6 -86 29 fill=1 $ scrap sludge level
9 3 -1.00 1 -2 3 -4 86 -12 29 $ water-basin
10 3 -1.00 1 -2 3 -4 12 -13 $ above can. water-basin
11 3 -1.00 1 -2 3 -4 13 -20 $ water-basin
12 3 -1.00 1 -2 3 -4 20 -21 $ water-basin +100 cm
13 3 -1.00 1 -2 3 -4 21 -22 $ water-basin +200 cm
14 3 -1.00 1 -2 3 -4 22 -23 $ water-basin +300 cm
15 3 -1.00 1 -2 3 -4 23 -24 $ water-basin to surface
16 0 -110 113 -109 112 -114 111 u=1 lat=2 fill=2
17 0 -116 119 -115 118 -120 117 u=3 lat=2 fill=4
18 3 -1.00 90 u=2 $ fuel rod outer cylinder
19 11 -18.82 -90 u=2 $ fuel rod inner cylinder
20 3 -1.00 90 u=4 $ fuel rod outer cylinder
21 12 -18.82 -90 u=4 $ fuel rod inner cylinder
22 0 7 -87 -30 fill=3
23 0 -1: 2: -3: 4: -5: 24 $ outside world

*1 px -14.2875 $ side adjacent canister rack
*2 px 12.3825 $ midline canister (between barrels)
*3 py -15.8750 $ back adjacent canister rack
*4 py 15.8750 $ front adjacent canister rack
5 pz -60.96 $ bottom of basin concrete floor
6 pz 0.0 $ canister bottom = basin floor
7 pz 0.635 $ canister bottom plate top
c 9 pz 53.1876 $ bottom of upper MkIA end cap
c 10 pz 53.6702 $ top of upper MkIA end cap
11 pz 67.4878 $ bottom of canister top cover
12 pz 68.7578 $ top of canister top cover
13 pz 73.66 $ canister top = stacking plate bottom
20 pz 142.4178 $ top of canister top cover
21 pz 242.4178 $ +100 cm of basin water
22 pz 342.4178 $ +200 cm of basin water
23 pz 442.4178 $ +300 cm of basin water
24 pz 487.68 $ surface of basin water
29 cz 10.406 $ Canister Outside radius
30 cz 10.254 $ Canister Inside radius
86 pz 21.5325 $ outside canister sludge level
87 pz 66.4718 $ Canister water surface (0.4 inches Nitrogen)
90 cz 0.762 $ fuel rod to fill universe
109 p 1 1.73205 0 2.5712446
110 px 1.2856223
111 p -1 1.73205 0 -2.5712446
112 p 1 1.73205 0 -2.5712446
113 px -1.2856223
114 p -1 1.73205 0 2.5712446
115 p 1 1.73205 0 2.5712446
116 px 1.2856223
117 p -1 1.73205 0 -2.5712446
118 p 1 1.73205 0 -2.5712446
119 px -1.2856223
120 p -1 1.73205 0 2.5712446

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mode n			
imp:n	1 19r 0		
vol	20j	\$ 001-242	everywhere
	1j	\$ 243	outside world
tmp	2.585e-8 20r	\$ 001-242 300K	everywhere
	1j	\$ 243	outside world
kcode	1000 1.0 20 250		
ksrc	-2.628 0.000 30.4673	\$ MkIA #1 point 1	
	2.628 0.000 30.4673	\$ MkIA #1 point 2	
	7.060 -2.628 30.4673	\$ MkIA #2 point 1	
	7.060 2.628 30.4673	\$ MkIA #2 point 2	
	3.530 -8.748 30.4673	\$ MkIA #3 point 1	
	3.530 -3.492 30.4673	\$ MkIA #3 point 2	
	-3.530 -8.748 30.4673	\$ MkIA #4 point 1	
	-3.530 -3.492 30.4673	\$ MkIA #4 point 2	
	-7.060 -2.628 30.4673	\$ MkIA #5 point 1	
	-7.060 2.628 30.4673	\$ MkIA #5 point 2	
	3.530 8.748 30.4673	\$ MkIA #6 point 1	
	3.530 3.492 30.4673	\$ MkIA #6 point 2	
	-3.530 8.748 30.4673	\$ MkIA #7 point 1	
	-3.530 3.492 30.4673	\$ MkIA #7 point 2	
m1	26000.55c -0.00312	\$ normal concrete (2.37 g/cc)	
	1001.50c -0.00206		
	8016.50c -0.49335		
	12000.50c -0.00209		
	20000.50c -0.24959		
	14000.50c -0.18816		
	13027.50c -0.00510		
	16032.50c -0.00082		
	6000.50c -0.05571		
mt1	poly.01t		
m2	6000.50c -0.0004	\$ stainless steel 304 (8.03 g/cc)	
	25055.50c -0.0200		
	14000.50c -0.0100		
	24000.50c -0.1900		
	28000.50c -0.0925		
	26000.55c -0.6871		
m3	1001.50c 0.66667	\$ basin water	
	8016.50c 0.33333		
mt3	lwtr.01t	\$ 300 K = 17 C = 62.6 F	
m4	1001.50c 0.66667	\$ canister water	
	8016.50c 0.33333		
mt4	lwtr.01t	\$ 300 K = 17 C = 62.6 F	
m5	40000.50c -1.00000	\$ zirconium	
m6	92235.50c -0.012491	\$ outer fuel (1.2491 wt% U-235)	
	92236.50c -0.000392		
	92238.50c -0.987117		
m7	92235.50c -0.009471	\$ inner fuel (0.9471 wt% U-235)	
	92236.50c -0.000392		
	92238.50c -0.990137		
m8	29000.50c -0.0025	\$ aluminum 6061 t6 (2.71 g/cc)	
	14000.50c -0.0060		
	12000.50c -0.0100		
	24000.50c -0.0025		
	13027.50c -0.9790		
m9	7014.50c -1.000	\$ nitrogen gas (1.138e-3 g/cc)	
m11	92234.50c -8.5e-5	\$ 1.25wt% uranium enrichment	
	92235.50c -.012491	\$ with fuel spilled in water	
	92236.50c -3.32e-4		
	92238.50c -.987092		
m12	92235.50c -.0114932	\$ 1.15wt% uranium enrichment	
	92238.50c -.9885068	\$ with fuel spilled in water	
totnu			
ctme	480.		
print	30 40 60 110 126		

APPENDIX B

LISTING OF WIMS-E RESULTS WITH INPUT FILENAMES

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LISTING OF WIMS-E RESULTS WITH INPUT FILENAMES

These input filenames are stored on the SECC disk cluster under the directory /tmp_mnt/home/h56712/wimse/hp:

<u>Description</u>	<u>k_∞</u>	<u>input file</u>
Aluminum canister MKIA loading, all intact pure water moderator full water reflection outer albedo = 1.0 axially infinite	0.438720	mkia_can.001
Aluminum canister MKIA loading, all intact pure water moderator full water reflection outer albedo = 1.0 axially infinite cluster offset, with additional MKIA inner	0.445013	mkia_can.002
<p>The following cases assumed an axial buckling correction = 1.62×10^{-3}. This corresponds to a finite height = 30.7 in. (including extrapolation distance).</p>		
Aluminum canister MKIA loading, all intact pure water moderator full water reflection outer albedo = 1.0	0.419572	mkia_can.003
Aluminum canister MKIA loading, all intact pure water moderator full water reflection outer albedo = 1.0 cluster offset, with additional MKIA inner	0.425565	mkia_can.004
Aluminum canister MKIA loading, all intact pure water moderator full water reflection outer albedo = 0.0	0.360045	mkia_can.005
Aluminum canister MKIA loading, all intact pure water moderator full water reflection outer albedo = 0.0 cluster offset, with additional MKIA inner	0.368152	mkia_can.006

Aluminum canister
 MKIA loading, all intact
 pure water moderator 0.891233 mkia_can.007
 infinite lattice
 outer albedo = 1.0

Aluminum canister
 MKIA loading, all intact
 pure water moderator 0.885732 mkia_can.008
 infinite lattice
 outer albedo = 1.0
 cluster offset, with additional MKIA inner

The following two cases duplicate the previous two cases, but remove the axial buckling leakage correction, plus reduced water density outside canisters.

Aluminum canister
 MKIA loading, all intact
 pure water moderator 0.942756 mkia_can.009
 infinite lattice
 outer albedo = 1.0

<u>outside water density</u>	<u>k_∞</u>
1.00	0.942756
0.90	0.965522
0.80	0.989418
0.70	1.014334
0.60	1.040010
0.50	1.065976
0.40	1.091346
0.30	1.114586
0.20	1.132953
0.10	1.141899

Aluminum canister
 MKIA loading, all intact
 pure water moderator 0.937500 mkia_can.010
 infinite lattice
 outer albedo = 1.0
 cluster offset, with additional MKIA inner

<u>outside water density</u>	<u>k_∞</u>
1.00	0.937500
0.90	0.959910
0.80	0.983445
0.70	1.008000
0.60	1.033304
0.50	1.058904
0.40	1.083906
0.30	1.106767
0.20	1.124684
0.10	1.132953

The following two cases duplicate the previous two cases, except that the axial buckling leakage correction has been put back in.

Aluminum canister
 MKIA loading, all intact
 pure water moderator 0.890970 mkia_can.011
 infinite lattice
 outer albedo = 1.0 (same as case mkia_can.007, but ran at LANL)

<u>outside water density</u>	<u>k_∞</u>
1.00	0.890970
0.90	0.909006
0.80	0.926919
0.70	0.944054
0.60	0.959270
0.50	0.970563
0.40	0.974174
0.30	0.962534
0.20	0.917967
0.10	0.787002

Aluminum canister
 MKIA loading, all intact
 pure water moderator 0.885556 mkia_can.012
 infinite lattice
 outer albedo = 1.0
 cluster offset, with additional MKIA inner

<u>outside water density</u>	<u>k_∞</u>
1.00	0.885556
0.90	0.903282
0.80	0.920890
0.70	0.937740
0.60	0.952688
0.50	0.963764
0.40	0.967229
0.30	0.955572
0.20	0.911279
0.10	0.781753

The following two cases start at the most reactive configuration for interspersed moderation between canisters (i.e., density = 0.40 g/cm³), and then reduces moderator density inside the canister. This should find a global optimum for interspersed moderation. These cases are for an infinite array of single-stacked canisters.

Aluminum canister
MKIA loading, all intact
pure water moderator
infinite lattice
outer albedo = 1.0

0.974174

mkia_can.013

<u>inside water density</u>	<u>k_∞</u>
1.00	0.974174
0.90	0.965806
0.80	0.955419
0.70	0.942563
0.60	0.926559
0.50	0.906304
0.40	0.879727
0.30	0.842255
0.20	0.781270
0.10	0.646053

Aluminum canister
MKIA loading, all intact
pure water moderator
infinite lattice
outer albedo = 1.0

0.967232

mkia_can.014

cluster offset, with additional MKIA inner

<u>inside water density</u>	<u>k_∞</u>
1.00	0.967232
0.90	0.958054
0.80	0.947012
0.70	0.933707
0.60	0.917541
0.50	0.897531
0.40	0.871813
0.30	0.836235
0.20	0.779137
0.10	0.652611

Aluminum canister
MKIA loading, one MKIA
outer is dissolved
UO₂-water density = 1.77 g/cm³
infinite lattice
outer albedo = 1.0

0.872453

mkia_can.015

same as above, but two
MKIA outers dissolved
orientation is "para"
infinite lattice
outer albedo = 1.0

0.868583

mkia_can.016

UO₂-water density = 2.47 g/cm³

same as above, but two MKIA outers dissolved orientation is "meta" infinite lattice outer albedo = 1.0; UO_2 -water density = 2.47 g/cm^3	0.868602	mkia_can.017
same as above, but two MKIA outers dissolved orientation is "ortho" infinite lattice outer albedo = 1.0 UO_2 -water density = 2.47 g/cm^3	0.868629	mkia_can.018
same as above, but three MKIA outers dissolved orientation is "ortho", plus center is dissolved infinite lattice outer albedo = 1.0 UO_2 -water density = 3.11 g/cm^3	0.868391	mkia_can.019
same as above, but all MKIA outers are dissolved UO_2 -water density = 5.21 g/cm^3	0.882903	mkia_can.020
same as 018, but refined MKIA outers dissolved orientation is "ortho" infinite lattice outer albedo = 1.0 UO_2 -water density = 2.47 g/cm^3	0.868846	mkia_can.021
same as 017, but refined MKIA outers dissolved orientation is "meta" infinite lattice outer albedo = 1.0 UO_2 -water density = 2.47 g/cm^3	0.868833	mkia_can.022
same as 016, but refined MKIA outers dissolved orientation is "para" infinite lattice outer albedo = 1.0 UO_2 -water density = 2.47 g/cm^3	0.868817	mkia_can.023
same as 020, but refined MKIA outers are all dissolved UO_2 -water density = 5.21 g/cm^3	0.882886	mkia_can.024
same as 021, but center MKIA assembly is also dissolved UO_2 -water density = 3.11 g/cm^3	0.868881	mkia_can.025

same as 021, but another MKIA outer is dissolved, in the "far" position UO ₂ -water density = 3.11 g/cm ³	0.869060	mkia_can.026
same as 021, but another MKIA outer is dissolved, in the "near" position UO ₂ -water density = 3.11 g/cm ³	0.869000	mkia_can.027
same as 027, but another MKIA outer is dissolved, in the "far" position UO ₂ -water density = 3.70 g/cm ³	0.871455	mkia_can.028
same as 028, but another MKIA outer is dissolved, filling in one of the remaining positions in the ring UO ₂ -water density = 4.25 g/cm ³	0.874845	mkia_can.029
same as 029, but the last remaining MKIA in the ring now has a dissolved outer UO ₂ -water density = 4.75 g/cm ³	0.878706	mkia_can.030
same as 024, but now the MKIA inners are all dissolved, leaving all outers intact UO ₂ -water density = 3.37 g/cm ³	0.883545	mkia_can.031
same as 031, but now all fuel components are dissolved effective enrichment = 1.15 wt% UO ₂ -water density = 6.57 g/cm ³	0.885622	mkia_can.032
staggered dissolution pattern central assembly intact	0.876024	mkia_can.033
same as 007, but with one MKIA assembly missing, no spacing adjustment in the ring	0.865617	mkia_can.034
same as 007, but with one MKIA assembly missing, with spacing adjustment in the ring	0.866031	mkia_can.035
same as 007, but with MKIA outers only, no assemblies	0.890870	mkia_can.036
same as 036, but with one MKIA outer missing in the ring with spacing adjustment	0.854143 0.853619	mkia_can.037 mkia_can.038

Stainless steel canister MKIA loading, all intact pure water moderator infinite lattice outer albedo = 1.0	0.836925	mkia_can.039
like 007, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.053357 1.053685	mkia_can.040 .140
like 040, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.098128 1.100853	mkia_can.044 .144
like 007, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.25%	1.043302	mkia_can.041
like 007, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 0.95%	1.000073	mkia_can.042
like 007, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.15%	1.030175	mkia_can.043
like 024, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 0.95%	0.952918	mkia_can.045
like 024, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.15%	0.981246	mkia_can.046
like 024, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.25%	0.993644	mkia_can.047
like 031, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 0.95%	0.966360	mkia_can.048
like 031, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.15%	0.994400	mkia_can.049
like 031, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.25%	1.006658	mkia_can.050
same as 032, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 0.95%	0.940187	mkia_can.051

same as 032, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.15%	0.968121	mkia_can.052
same as 032, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.25%	0.980368	mkia_can.053
like 008, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 0.95%	0.994480	mkia_can.054
like 008, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.15%	1.024661	mkia_can.055
like 008, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.25%	1.037831	mkia_can.056
like 024, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.011133 1.010503	mkia_can.057 .157
like 024, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.055031 1.056039	mkia_can.058 .158
like 031, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.021620 1.021313	mkia_can.059 .159
like 031, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.064680 1.066164	mkia_can.060 .160
like 032, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.000909 0.999878	mkia_can.061 .161
like 032, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.044876 1.045207	mkia_can.062 .162
like 008, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.048122 1.048366	mkia_can.063 .163
like 008, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.093193 1.095797	mkia_can.064 .164
like 007, but optimum MKIV fuel assemblies now surround the canister	1.037297	mkia_can.065

like 007, but optimum MKIA fuel assemblies now surround the canister	1.070669	mkia_can.066
like 024, but optimum MKIV fuel assemblies now surround the canister	0.991289	mkia_can.067
like 024, but optimum MKIA fuel assemblies now surround the canister	1.025137	mkia_can.068
like 031, but optimum MKIV fuel assemblies now surround the canister	1.003448	mkia_can.069
like 031, but optimum MKIA fuel assemblies now surround the canister	1.036168	mkia_can.070
like 032, but optimum MKIV fuel assemblies now surround the canister	0.979401	mkia_can.071
like 032, but optimum MKIA fuel assemblies now surround the canister	1.013863	mkia_can.072
like 008, but optimum MKIV fuel assemblies now surround the canister	1.031651	mkia_can.073
like 008, but optimum MKIA fuel assemblies now surround the canister (plus, with no axial leakage)	1.065241 1.127889	mkia_can.074
optimum scrap inside canister, 1.15 wt% ²³⁵ U enrichment, with axial leakage correction	0.918705	mkia_can.075

This is case 075, but varying
interspersed water moderation
between canisters

mkia_can.076

<u>outside water density</u>	<u>k_∞</u>
1.00	0.918705
0.90	0.935649
0.80	0.952252
0.70	0.967853
0.60	0.981318
0.50	0.990679
0.40	0.992339
0.30	0.979032
0.20	0.933900
0.10	0.804777

This is case 075, but varying
interspersed water moderation
between canisters, with SS canister

mkia_can.077

<u>outside water density</u>	<u>k_∞</u>
1.00	0.867394
0.90	0.883680
0.80	0.899866
0.70	0.915388
0.60	0.929250
0.50	0.939696
0.40	0.943398
0.30	0.933559
0.20	0.894069
0.10	0.774966

This is case 075, but varying
interspersed water moderation
between canisters

mkia_can.078

mkia_can.099

<u>outside water density</u>	<u>6 wt% ²⁴⁰Pu FP k_∞</u>	<u>green fuel k_∞</u>
1.00	0.909586	0.918706
0.90	0.926080	0.935648
0.80	0.942207	0.952253
0.70	0.957309	0.967852
0.60	0.970254	0.981317
0.50	0.979110	0.990681
0.40	0.980295	0.992336
0.30	0.966693	0.979035
0.20	0.921713	0.933898
0.10	0.794229	0.804772

This is case 075, but varying
interspersed water moderation
between canisters, temperature effect

mkia_can.079

<u>fuel temperature (K)</u>	<u>k_∞</u>
293.	0.918705
325.	0.918224
350.	0.917866
375.	0.917517
400.	0.917181
425.	0.916857
450.	0.916542
475.	0.916235
500.	0.915935
525.	0.915642

coefficient = -1.44 pcm/K

same as 075, but homogeneous
(optimum) UO₂ in water solution
in between canisters; E = 0.95%

1.017266

mkia_can.080

same as 075, but homogeneous
(optimum) UO₂ in water solution
in between canisters; E = 1.15%

1.045859

mkia_can.081

same as 075, but homogeneous
(optimum) UO₂ in water solution
in between canisters; E = 1.25%

1.058374

mkia_can.082

like 075, but optimum MKIV
fuel assemblies now surround
the canister

1.053456

mkia_can.083

like 075, but optimum MKIA
fuel assemblies now surround
the canister

1.084552

mkia_can.084

like 075, but heterogeneous
(optimum) 0.95 rods + water
in between canisters

1.068061

mkia_can.085

like 075, but heterogeneous
(optimum) 1.25 rods + water
in between canisters

1.112764

mkia_can.086

The following MKIA calculations were performed using WIMS-E 5a:

like 007, but heterogeneous
(optimum) 0.95 rods + UO₂ solution
in between canisters

1.004463

mkia_can.087

like 007, but heterogeneous (optimum) 1.25 rods + UO ₂ solution in between canisters	1.048779	mkia_can.088
like 024, but heterogeneous (optimum) 0.95 rods + UO ₂ solution in between canisters	0.957769	mkia_can.089
like 024, but heterogeneous (optimum) 1.25 rods + UO ₂ solution in between canisters	1.003238	mkia_can.090
like 031, but heterogeneous (optimum) 0.95 rods + UO ₂ solution in between canisters	0.971081	mkia_can.091
like 031, but heterogeneous (optimum) 1.25 rods + UO ₂ solution in between canisters	1.014887	mkia_can.092
like 032, but heterogeneous (optimum) 0.95 rods + UO ₂ solution in between canisters	0.944970	mkia_can.093
like 032, but heterogeneous (optimum) 1.25 rods + UO ₂ solution in between canisters	0.991491	mkia_can.094
like 008, but heterogeneous (optimum) 0.95 rods + UO ₂ solution in between canisters	0.998968	mkia_can.095
like 008, but heterogeneous (optimum) 1.25 rods + UO ₂ solution in between canisters	1.043523	mkia_can.096
like 075, but heterogeneous (optimum) 0.95 rods + UO ₂ solution in between canisters	1.021779	mkia_can.097
like 075, but heterogeneous (optimum) 1.25 rods + UO ₂ solution in between canisters	1.063080	mkia_can.098
like 007, but typical sludge composition surrounds canisters	0.740972	mkia_can.100
MKIA outers are all dissolved typical sludge surrounds canisters UO ₂ -water density = 5.21 g/cm ³	0.650576	mkia_can.101

same as 101, but now the MKIA inners are all dissolved, leaving all outers intact typical sludge surrounds canisters UO ₂ -water density = 3.37 g/cm ³	0.692557	mkia_can.102
same as 102, but now all fuel components are dissolved effective enrichment = 1.15 wt% typical sludge surrounds canisters UO ₂ -water density = 6.57 g/cm ³	0.613507	mkia_can.103
MKIA loading, all intact pure water moderator infinite lattice, typical sludge surrounds canisters cluster offset, with additional MKIA inner	0.734847	mkia_can.104
optimum scrap inside canister, 1.15 wt% ²³⁵ U enrichment, with typical sludge surrounds canisters	0.771091	mkia_can.105

MKIV calculations:

Aluminum canister MKIV loading, all intact pure water moderator infinite lattice outer albedo = 1.0	0.838566	mkiv_can.001
same as 001, but with stainless steel canister	0.786849	mkiv_can.002
same as mkia_024, but refined MKIV outers are all dissolved, UO ₂ -water density = 7.32 g/cm ³	0.841557	mkiv_can.003
same as 003, but now all MKIV inners are all dissolved, UO ₂ -water density = 4.67 g/cm ³	0.840932	mkiv_can.004
same as 004, but now all MKIV assemblies are all dissolved UO ₂ -water density = 8.88 g/cm ³	0.849845	mkiv_can.005
same as mkia_008, but now do this for mkiv assemblies, with a mkiv inner	0.832231	mkiv_can.006
like 001, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 0.95%	0.962509	mkiv_can.007

like 001, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.15%	0.993741	mkiv_can.008
like 001, but homogeneous (optimum) UO_2 in water solution in between canisters; E = 1.25%	1.007361	mkiv_can.009
like 001, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.018956 1.019146	mkiv_can.010 .110
like 010, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.065300 1.067907	mkiv_can.011 .111
same as 003, but canister is now surrounded by 0.95 UO_2 solution MKIV outers are all dissolved inside UO_2 -water density = 7.32 g/cm ³	0.892279	mkiv_can.012
same as 003, but canister is now surrounded by 1.15 UO_2 solution MKIV inners are all dissolved inside UO_2 -water density = 7.32 g/cm ³	0.920911	mkiv_can.013
same as 003, but canister is now surrounded by 1.25 sludge MKIV inners are all dissolved inside UO_2 -water density = 7.32 g/cm ³	0.933467	mkiv_can.014
same as 004, but canister is now surrounded by 0.95 UO_2 solution MKIV inners are all dissolved UO_2 -water density = 4.67 g/cm ³	0.917783	mkiv_can.015
same as 004, but canister is now surrounded by 1.15 UO_2 solution MKIV inners are all dissolved UO_2 -water density = 4.67 g/cm ³	0.946214	mkiv_can.016
same as 004, but canister is now surrounded by 1.25 UO_2 solution MKIV inners are all dissolved UO_2 -water density = 4.67 g/cm ³	0.958657	mkiv_can.017
same as 005, but canister is now surrounded by 0.95 UO_2 solution UO_2 -water density = 8.88 g/cm ³	0.873762	mkiv_can.018
same as 005, but canister is now surrounded by 1.15 UO_2 solution UO_2 -water density = 8.88 g/cm ³	0.902328	mkiv_can.019

same as 005, but canister is now surrounded by 1.25 UO ₂ solution UO ₂ -water density = 8.88 g/cm ³	0.914871	mkiv_can.020
like 012, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.959150 0.957179	mkiv_can.021 .121
like 012, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.005408 1.004246	mkiv_can.022 .122
like 015, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.978623 0.977570	mkiv_can.023 .123
like 015, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.023264 1.023559	mkiv_can.024 .124
like 018, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.945963 0.943017	mkiv_can.025 .125
like 018, but heterogeneous (optimum) 1.25 rods + water in between canisters	0.993572 0.990820	mkiv_can.026 .126
like 006, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 0.95%	0.957626	mkia_can.027
like 006, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.15%	0.989104	mkia_can.028
like 006, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.25%	1.002839	mkia_can.029
like 006, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.014529 1.014643	mkiv_can.030 .130
like 006, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.061381 1.063881	mkiv_can.031 .131
like 001, but optimum MKIV fuel assemblies now surround the canister	1.001085	mkiv_can.032
like 001, but optimum MKIA fuel assemblies now surround the canister	1.035234	mkiv_can.033

like 012, but optimum MKIV fuel assemblies now surround the canister	0.932805	mkiv_can.034
like 012, but optimum MKIA fuel assemblies now surround the canister	0.969452	mkiv_can.035
like 015, but optimum MKIV fuel assemblies now surround the canister	0.956474	mkiv_can.036
like 015, but optimum MKIA fuel assemblies now surround the canister	0.990769	mkiv_can.037
like 018, but optimum MKIV fuel assemblies now surround the canister	0.915459	mkiv_can.038
like 018, but optimum MKIA fuel assemblies now surround the canister	0.954310	mkiv_can.039
like 006, but optimum MKIV fuel assemblies now surround the canister	0.996208	mkiv_can.040
like 006, but optimum MKIA fuel assemblies now surround the canister	1.030721	mkiv_can.041
optimum scrap inside canister, 0.95 wt% ²³⁵ U enrichment, with axial leakage correction	0.855996	mkiv_can.042
same as 042, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 0.95%	0.981157	mkiv_can.043
same as 042, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.15%	1.011669	mkiv_can.044
same as 042, but homogeneous (optimum) UO ₂ in water solution in between canisters; E = 1.25%	1.025009	mkiv_can.045
like 042, but optimum MKIV fuel assemblies now surround the canister	1.018987	mkiv_can.046
like 042, but optimum MKIA fuel assemblies now surround the canister	1.050974	mkiv_can.047

like 042, but heterogeneous (optimum) 0.95 rods + water in between canisters	1.034911	mkiv_can.048
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like 042, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.082043	mkiv_can.049
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The following MKIV calculations were performed using WIMS-E 5a:

like 001, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.969121	mkiv_can.050
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like 001, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.014122	mkiv_can.051
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like 012, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.898531	mkiv_can.052
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like 012, but heterogeneous (optimum) 1.25 rods + water in between canisters	0.947830	mkiv_can.053
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like 015, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.924030	mkiv_can.054
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like 015, but heterogeneous (optimum) 1.25 rods + water in between canisters	0.969910	mkiv_can.055
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like 018, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.879729	mkiv_can.056
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like 018, but heterogeneous (optimum) 1.25 rods + water in between canisters	0.932250	mkiv_can.057
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like 006, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.964524	mkiv_can.058
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like 006, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.009912	mkiv_can.059
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like 042, but heterogeneous (optimum) 0.95 rods + water in between canisters	0.988449	mkiv_can.060
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like 042, but heterogeneous (optimum) 1.25 rods + water in between canisters	1.030406	mkiv_can.061
Aluminum canister MKIV loading, all intact pure water moderator infinite lattice typical sludge between canisters	0.689176	mkiv_can.062
same as mkia_024, but refined MKIV outers are all dissolved UO ₂ -water density = 7.32 g/cm ³ typical sludge between canisters	0.524030	mkiv_can.063
same as 063, but now all MKIV inners are all dissolved UO ₂ -water density = 4.67 g/cm ³	0.608966	mkiv_can.064
same as 064, but now all MKIV assemblies are all dissolved UO ₂ -water density = 8.88 g/cm ³	0.439877	mkiv_can.065
same as 006, but now typical sludge surrounding canisters with a mkiv inner	0.682757	mkiv_can.066
optimum scrap inside canister, 0.95 wt% ²³⁵ U enrichment, with axial leakage correction	0.721460	mkiv_can.067

APPENDIX C

MKIA FUEL 26.1-INCH LENGTH ACCOUNTABILITY QUESTION

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MKIA FUEL 26.1-INCH LENGTH ACCOUNTABILITY QUESTION

[50] From: Karen R Morris at WHC146 9/20/94 1:17PM (1127 bytes: 20 ln)
To: Warren D Wittekind at WHC71
Receipt Requested
cc: Lorie L Blehm
Subject: Accountability Question-How many 26.1 in long MKIA in KW ?

----- Message Contents -----

Karen, when you have a moment, check to ascertain how many 26.1 inch long Mark IA (1.25 wt% outer, 0.947 wt% inner) are in 105-KW Basin. This is a valid fuel type, but I believe there may be none (zero) in 105 KWBasin.

This information is being used in a criticality calculation relative to basin consolidation.

Warren D. Wittekind H0-38 376-8175.

We have 12 assemblies and 2 inners of A12N09E (MK1A/26") currently in the 105 KW Basin. All of the assemblies are located in the same canister and the same key date. The inners are located in a separate canister, different key date.

Thank you

Karen Morris
3-1565

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APPENDIX D

INDEPENDENT REVIEWER COMMENTS

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INDEPENDENT REVIEWER COMMENTS

REVIEWERS COMMENTS FOR WHC-SD-NR-CSER-015

E. M. Miller of Criticality and Radiological Analyses carried out an independent, technical review of WHC-SD-NR-CSER-015, *Criticality Safety Evaluation Report for MARK 1A Fuel Pertaining to Consolidating Fuel Storage*, and provides the following comments.

The criticality safety evaluation was reviewed for technical adequacy for qualification of storing N Reactor fuel in several layers in K Basins. The references were confirmed and the computer runs cover adequate cases to establish the intended conclusion. The computer runs and contingency cases established the safety of the stack fuel in the basins. The validation of the MCNP criticality computer code covers the low enriched uranium metal and oxide analyses done for the evaluation. The bases for k_{eff} limits of 0.98 and 0.97 are addressed. Further review of using a limit of 0.98 is given in WHC-SD-SQA-CSA-20330, entitled "Use of k_{eff} Subcriticality Limit of 0.98." Editorial comments and suggested clarifications were incorporated into to the text.

One of the primary effects of this CSER is to require a limit on the depth of fuel, fuel fragments, and uranium oxide in water, all labelled sludge, that would be allowed to accumulate on the fuel storage basin floor among the fuel storage canisters. The depth limit is based on MCNP calculations and interpolation calculations. The thickness limit is set at 3.75 in. Figure III.B.5-6 in ARH-600 shows that an infinite slab must be thicker than 7 in. to be critical for full water reflection and a uranium enrichment of 2.5 wt%. An enrichment of 2.5 is twice the maximum conceivable in the K Basins, except for a small number of specially treated canisters. This comparison, the many previous analyses referenced, and the criticality tests on the reactivity of N Reactor fuel in the K Basins gives confidence in the analysis results.

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