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CRITICALITY SAFETY
EVALUATION REPORT FOR 300 AREA N REACTOR FUEL
FABRICATION AND STORAGE FACILITY

K. N. SCHWINKENDORF
TECHNOLOGY SUPPORT PROGRAMS

DECEMBER 1993

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1.0 INTRODUCTION

Criticality safety support calculations for the 300 Area Fuels Manufacturing Facility have been performed to update values currently found in the criticality prevention specifications, (Appendix B). The old safe masses and associated dimensions table has been updated with new values, which were produced using more modern computer codes that comply with Software Quality Assurance (SQA) requirements. Monte Carlo calculations, using MCNP (Carter 1991), were performed to spot check these results, obtained using the WIMS-E code (Gubbins et. al. 1982).

In addition, the criticality considerations associated with certain accident, or upset, conditions were analyzed. These scenarios include fire, the bringing together of multiple safe masses into one neutronically coupled system, misstacking, and accidental interspersed moderation.

2.0 REVISED LIMITS

2.1 SAFE MASSES

The WIMS-E lattice transport code was used to generate infinite multiplication factors (k_{∞}) and two-group cross sections for 300 Area fissile materials. These cross sections were used by the diffusion theory computer code GOLF (Schwinkendorf 1994) to predict finite dimensions required for different geometries, yielding a $k_{eff} = 0.98$ (Rogers 1991). "Safe mass" was established by dividing the mass corresponding to $k_{eff} = 0.98$ by two, in accordance with WHC-CM-4-29, *Nuclear Criticality Safety Manual*.

For each fissile material that has fixed dimensions, such as fuel inners, outers, and assemblies, as well as billets, these masses were calculated as a function of lattice spacing in water, to determine the minimum values. For fissile material scrap, the assumption is made that heterogeneous chunks of metal are present, with no control over the characteristic size of these pieces. These chunks are modeled as metal rods surrounded by moderator. A parametric variation is made, where the minimum safe mass (as a function of lattice spacing) is found for each of several rod outer diameters (OD). The "global" minimum is found from this family of curves, and is used as the safe mass for uranium scrap of that particular enrichment. For homogeneous systems (i.e., solutions) the minimum safe mass is determined parametrically as the hydrogen to uranium metal atom ratio (H/U) is varied. These safe mass results are contained in Table 1. These values are intended to replace the corresponding values in the table cited in Section 1.

The associated dimensions in Table 1 are the dimensions corresponding to the safe mass, in each of their respective geometries. These dimensions are not equal to the $k_{eff} = 0.98$ dimensions, but are related to those dimensions according to the scaling relationship:

$$D_A = 2R_{0.98} / \sqrt[3]{2}$$

The parameter G is a geometry factor that equals 1 for slab, 2 for cylinder, and 3 for either spherical or hemispherical. The D_A variable is what appears in Table 1, and $R_{0.98}$ is the $k_{eff} = 0.98$ radius ($R_{0.98}$ is interpreted as the

Table 1. Safe Masses and Associated Dimensions.

	Sphere		Hemisphere		Infinite Cylinder		Infinite Slab	
	Mass(lb)	Dia.(in)	Mass(lb)	Dia.(in)	Mass(lb/ft)	Dia.(in)	Mass(lb/ft ²)	Thick(in)
MKIV (0.95 wt% inner and outer)								
Billets	-----unlimited-----							
Assemblies	5228	34.5	8575	51.3	1170	22.6	322	9.8
Outers	3605	33.2	5923	49.2	839	21.7	241	9.5
Inners	3084	30.6	5122	45.7	771	19.9	239	8.9
Scrap	2150	28.6	3603	42.8	567	19.4	180	9.1
Solution	-----unlimited-----							
MKIA (0.95 wt% inner, 1.25 wt% outer)								
Billets	9859	35.9	15967	53.2	2134	23.6	584	10.0
Assemblies	1579	24.1	2716	36.4	486	15.9	187	6.8
Outers	932	22.1	1619	33.5	308	14.5	127	6.1
Inners	3176	30.7	5284	45.7	775	20.8	237	8.7
Scrap	586	20.1	1037	30.7	208	13.4	88	5.9
Solution	1503	28.1	2539	42.2	404	18.2	130	8.9
	<u>kg</u>	<u>cm</u>	<u>kg</u>	<u>cm</u>	<u>kg/m</u>	<u>cm</u>	<u>kg/m²</u>	<u>cm</u>
MKIV (0.95 wt% inner and outer)								
Billets	-----unlimited-----							
Assemblies	2371	87.6	3890	130.3	1741	57.4	1572	24.9
Outers	1635	84.3	2687	125.0	1249	55.1	1177	24.1
Inners	1399	77.7	2323	116.1	1147	50.5	1167	22.6
Scrap	975	72.6	1634	108.7	844	49.3	879	23.1
Solution	-----unlimited-----							
MKIA (0.95 wt% inner, 1.25 wt% outer)								
Billets	4472	91.2	7243	135.1	3176	59.9	2851	25.4
Assemblies	716	61.2	1232	92.5	723	40.4	913	17.3
Outers	423	56.1	734	85.1	458	36.8	620	15.5
Inners	1441	78.0	2397	116.1	1153	52.8	1157	22.1
Scrap	266	51.1	470	78.0	310	34.0	430	15.0
Solution	682	71.4	1152	107.2	601	46.2	635	22.6

Note that MKIA billets, scrap, and solution results are based on 1.25 wt% enrichment.

half-thickness for slab geometry).

Appendix B contains Table 1 information for both $k_{eff} = 0.98$ and critical masses. Figure 1 illustrates the variation in critical mass as a function of lattice spacing for MKIV fuel assemblies and components, in spherical geometry. Figure 2 shows these results for hemispherical geometry. These results are shown for MKIA fuel and components in Figures 3 and 4.

Figures 5 and 6 contain minimum critical and safe mass results for 0.95 wt% and 1.25 wt% uranium metal scrap (modeled as rods surrounded by water moderator). Each point plotted in these two figures represents the minimum from a lattice spacing variation, for each of the rod diameters.

Figure 7 contains results for 1.25 wt% MKIA billets. The geometry for these masses is spherical. Figure 8 contains the corresponding lattice k_{∞} values. Figure 9 contains the lattice k_{∞} values for 0.95 wt% MKIV billets; at no point does k_{∞} exceed 1.0 for 0.95 wt% MKIV billets.

2.2 ACCIDENT CALCULATIONS

2.2.1 Effect of Volcanic Ash

One of the accident scenarios concerns a building roof collapse, with water flooding in conjunction with an ashfall from another volcanic eruption. Mt. St. Helens ash from the May 1980 eruption was high in silicon content. Calculations were performed to evaluate the effect of inclusion of silicon into the water moderator, for 0.95 wt% enriched uranium metal rods, used to represent scrap. The effect is small, and in the direction of decreased reactivity. A worst-case system was chosen for which to add silicon. If this worst-case is chosen to be the most reactive system, adding poison to the moderator will always result in lower reactivity. However, because the minimum critical mass occurs at a slightly higher value of lattice spacing than does the spacing required for maximum reactivity, this slightly larger lattice spacing constitutes a slightly over-moderated system. If silicon (or any other impurity) is added to an over-moderated system, water will be displaced, which will tend to increase reactivity (since the H/U is driven closer to the optimum). The other effect is simply the added absorption from the impurity. Depending upon how these two effects compete will determine the net reactivity perturbation. If crude input data are used in the lattice physics codes, one can get incorrect results. Even if the addition of silicon is less than 2.0 wt%, keeping the moderator density constant at 1.0 g/cm³ results in too much displacement of water. The actual mixture density may be calculated from the formula:

$$\rho_{mix} = \left[\frac{wf_{Si}}{2.33} + \frac{(1-wf_{Si})}{1.00} \right]^{-1}, \quad wf_{Si} = \frac{wt\%_{Si}}{100}$$

The above formula for calculating silicon in water mixture density provides progressively larger values as silicon content is increased, which are multiplied by the input weight-percents of the mixture components to predict the absolute quantity of each nuclide present. This results in a smaller reduction in hydrogen number density as a function of the addition of impurities.

Figure 1. Critical Mass vs Spacing.
MKIV Fuel and Components, spherical

Figure 2. Critical Mass vs Spacing.
MKIV Fuel and Components, hemispherical

Figure 3. Critical Mass vs Spacing.
MKIA Fuel and Components, spherical

Figure 4. Critical Mass vs Spacing.
MKIA Fuel and Components, hemispherical

Figure 5. Minimum Critical, Safe Mass
0.95 wt% Enriched Uranium Rods

Figure 6. Minimum Critical/Safe Mass.
1.25 wt% Enriched Uranium Rods

Figure 7. Critical Mass vs Spacing.
1.25 wt% Enriched Uranium Billets

Figure 8. Multiplication Factor.
MK-I-A Billets

Figure 9. Multiplication Factor.
MK-I-N Billets

Figure 10 demonstrates this difference. If both of these sets of compositions (the adjusted density being exact, and the constant density being approximate) are input to WIMS-E, the result is that the adjusted set of compositions produces a slightly negative reactivity perturbation, and the approximate approach results in a positive effect. The increased (and artificial) water displacement that results from making the constant density approximation is apparently sufficient to overcome the small negative reactivity effect of the increased parasitic absorption. Figure 11 shows the calculated reactivity as silicon is added to the system, for both the constant density and for the adjusted densities. The curve labeled "Min Crit Mass" represents the adjusted densities, and the curve labeled "Min CM -- dens=1" represents the approximate set. This initial increase in k_{∞} with silicon addition would eventually decrease, as more silicon displaces more water (i.e., the system will eventually become undermoderated). The curve labeled "Max Reactivity" shows silicon addition to a lattice where the spacing corresponds to maximum reactivity. The other two curves show silicon addition to a slightly larger lattice spacing, corresponding to minimum critical mass. Since this lattice spacing is slightly over-moderated, the initial reactivity is slightly less.

2.2.2 Effect of Fire

In the event of a fire, uranium metal may be oxidized to form UO_2 . This UO_2 would then be free to disperse homogeneously in water. Criticality analyses were performed to model homogeneous distributions of both 0.95 wt% and 1.25 wt% enriched uranium in water, as a function of H/U. The 0.95 wt% solution cannot go critical in this form; the k_{∞} for 0.95 wt% solution is always less than 1.0. This is not the case for 1.25 wt% solution, as shown in Figure 12. For H/U values for which criticality is possible, both critical and safe masses are plotted for 1.25 wt% solution in Figure 13.

Figure 10. Hydrogen Number Density.
Adjusted Density vs Simple Constant =1

Figure 11. k-infinity vs. Si Content.
0.947 wt% Uranium Metal Rods.

Figure 12. k -Infinity vs H/U.

Homogeneous U-235 Solution

Figure 13. Critical Mass vs. H/U.

Homogeneous (1.25 wt%) U-235 Solution

In a fire, uranium metal components may fragment as they burn, resulting in a distribution of debris sizes. This partial oxidation of uranium metal to homogeneous UO_2 was another effect considered. In this case, uranium scrap (modeled as rods) was immersed in a moderator composed of a homogeneous mixture of UO_2 and water. Because of the uncertainty in the characteristic size of the resulting fuel debris, a parametric study was performed to analyze the reactivity of the heterogeneous system, as rod size, spacing, and amount of UO_2 in the water is varied. First, Figure 14 shows k_∞ as a function of amount of UO_2 added for both 0.95 and 1.25 wt% UO_2 in water.

Figure 14. k_∞ vs UO_2 Content.
 UO_2 in Water Solution.

A maximum in reactivity may be seen around 3.0, 4.0, and 5.0 g/cm^3 UO_2 in UO_2 /water solution. Second, Figures 15 through 20 show the reactivity of uranium metal rods immersed into 2.0, 3.0, and 4.0 g/cm^3 UO_2 in water moderator. These results include both 0.95 and 1.25 wt% enrichments. It is assumed that uranium metal does not melt and therefore coalesce into larger metal chunks. The largest debris sizes plausible are therefore those associated with the annular wall thickness. These figures show that the calculated k_∞ never exceeds 1.0 for 0.95 wt% enriched scrap. However, k_∞ for 1.25 wt% scrap does exceed 1.0. Figure 21 illustrates k_∞ as a function of the lattice spacing for 0.95 wt%, 0.25-in. OD rods immersed into $\text{UO}_2/\text{H}_2\text{O}$ moderator. This way of rendering the data makes it easier to see that as more UO_2 is added to the moderator, the spacing that maximizes reactivity shifts to greater values, but the peak in k_∞ decreases. Also, for large lattice spacings, the k_∞ approaches the k_∞ of the $\text{UO}_2/\text{H}_2\text{O}$ mixture, as would be expected. Figures 22 and 23 show k_∞ for MKIV and MKIA fuel assemblies immersed into 3.0 g/cm^3 UO_2 in water, compared to the reactivity when pure water is the moderator. Addition of uranium to the moderator causes a reduction in the resonance escape probability; neutrons thermalizing in the moderator can no longer do so in the absence of ^{238}U . Figure 24 shows both MKIV and MKIA k_{eff} in 3.0 g/cm^3 $\text{UO}_2/\text{H}_2\text{O}$ mixture as a function of the weight fraction of uranium in fuel to total uranium.

Figure 15. k -inf vs Mass Fraction.
0.95 wt% Rods in 2 g/cc UO₂ in water.

Figure 16. k -inf vs Mass Fraction.
0.95 wt% Rods in 3 g/cc UO₂ in water.

Figure 17. k -inf vs Mass Fraction.
0.95 wt% Rods in 4 g/cc UO₂ in water.

Figure 18. k -inf vs Mass Fraction.
1.25 wt% Rods in 2 g/cc UO₂ in water.

Figure 19. k -inf vs Mass Fraction.
1.25 wt% Rods in 3 g/cc UO₂ in water.

Figure 20. k -inf vs Mass Fraction.
1.25 wt% Rods in 4 g/cc UO₂ in water.

Figure 21. k -inf vs Lattice Spacing.
0.95 wt% 0.25" OD Uranium Metal Rods.

Figure 22. MKIV k -Infinity.
"Sludge" = 3.0 g/cc UO2(95) in Water.

Figure 23. MKIA k-Infinity.

"Sludge" = 3.0 g/cc UO₂(95) in Water.

Figure 24. MKIV and MKIA k-Infinity.

"Sludge" = 3.0 g/cc UO₂(95) in Water.

A fire analysis has provided estimates of the fraction of metal oxidized in fires of varying durations. This analysis predicts that up to 10% of the mass of uranium may be oxidized in the worst-case 12-hour fire.

In the event of a fire, it is anticipated that firefighting operations would be carried out using standard firefighting equipment. The hoses currently in use include both 2.5 in. and 1.5 in. hoses, which operate at approximately 100 psi. One scenario considered was the possibility of water pressure from fire hoses forcing fissile material into a geometrically unsafe configuration (i.e., a hemisphere). For this to happen, there would have to be sufficient force to raise uranium metal against gravity. If the weight of a uranium metal rod is divided by the cross-sectional area (length times diameter) this results in a measure of the gravity force per unit projected area. For 0.5-in. OD rods, this number is equal to 0.27 psi (in general, this value scales as $0.27(\text{rod OD}/0.50)$). This is significantly less than the 100 psi pressure from the fire hose; therefore the fire hose water pressure would be sufficient to move uranium scrap material around (even vertically). However, to approach the safe limits established in Table 1, not only is an unsafe geometry required, but also optimum moderation (i.e., spacing) and reflection.

Existing storage configurations consist of wooden boxes, each containing a 4x9 square array of fuel assemblies, spaced at 4.75 in. These storage boxes are limited to MKIA being stacked two high. This spacing is very much greater than the optimum spacing in full density water; water flooding of this array would present no criticality concern. In the fire scenario, however, the wooden boxes would be consumed and are considered removed. If the remaining fuel stack collapses into a slab, the slab values in Table 1 are applicable; these values correspond to one-half of a $k_{\text{eff}} = 0.98$ mass.

Two situations were considered: hexagonal touching and optimal spacing. If an 8x9 array of fuel assemblies collapses into a close-packed hexagonal array of the same floor area (as that occupied by a single storage box), the result is a slab layer 7.6-in. in height. Such a configuration is highly subcritical, because 1) the lattice is undermoderated and 2) there is high geometric leakage. Optimal spacing is a bounding result. No other spacing can result in a smaller areal critical mass. Addition of parasitic absorbers (such as ash) can only increase the mass required for criticality.

If the 8x9 array of MKIA fuel assemblies collapses into a slab with optimal lattice spacing, the slab height becomes 11.7 in. This may seem to violate the slab thickness limits in Table 1. However, the intent is to satisfy $k_{\text{eff}} < 0.98$. Slab heights required for $k_{\text{eff}} = 0.98$ are reported in Table 2, for both infinitely reflected and bare slabs. In the fire accident, removal of the boxes and subsequent loss of geometry control (resulting in optimal spacing) produces the 11.7-in. slab height (with k_{eff} equal to 0.9507). A second contingency could be a stacking error so that the array is three boxes high. In this case, the fuel assembly array that collapses into the slab is a 12x9 array. The resulting slab height in this case, assuming optimal spacing, is equal to 17.5 in. If an infinite slab has no reflector, the height required for $k_{\text{eff}} = 0.98$ becomes nearly this value (17.3 in.). One can argue that the double contingency criterion is satisfied. As long as physical controls are placed on the amount of water that the building can contain, advantage can be taken of bare slab values (i.e., if water can be prevented from accumulating above the surface of the spilled fuel layer).

Introduction of the water reflector to bring the system to critical would be considered as a third contingency, and therefore something that does not have to be protected against. In addition, the infinite slab still requires optimum water moderation up to the top of the collapsed fuel layer to attain $k_{eff} = 0.98$ for the 12x9 array case.

Table 2. Slab Heights for Various Fuels and k_{eff} s.
All values are in cm (in.).

	Infinite reflection:		Bare slab:	
	MKIA	MKIV	MKIA	MKIV
h_{crit}	36.83 (14.5)	61.02 (24.0)	46.27 (18.2)	70.57 (27.7)
$h_{0.98}$	34.80 (13.7)	50.10 (19.7)	44.02 (17.3)	59.62 (23.4)
h_{safe}	17.40 (6.8)	25.05 (9.8)	22.01 (8.6)	29.81 (11.7)

Note: $h_{safe} = h_{0.98}/2$, by definition.

Values in Tables 1 and 2 were calculated assuming no geometric leakage in either perpendicular direction (i.e., both length and width of the array are assumed to be infinite). Finite perpendicular leakage would increase the height necessary to attain a given k_{eff} .

There are aisles between fuel storage boxes. If the fraction of the building floor area covered by fuel storage boxes is accounted for, then the collapsed slab height would be less. To conserve mass, the aisles would fill with fuel assemblies, reducing the collapsed height proportionately. For example, if the floor area fraction is equal to 2/3, then the collapsed slab height (for optimal lattice spacing) would be equal to $(2/3)11.7 = 7.8$ in. MCNP calculations for actual storage arrays are contained in Section 4.0 of this document.

3.0 STORAGE ARRAY CALCULATIONS

Storage box arrays of MKIA assemblies have been analyzed. MKIA fuel assemblies are the most reactive fuel because they contain 1.25 wt% enriched outers, and are therefore a bounding case. The MCNP Monte Carlo computer code was used to analyze the storage boxes, in arrays stacked two, three, four, and five high. The unit storage box model is shown in Figure 25. Explicit modeling of the plywood in the boxes was employed. WIMS-E was used to parametrically vary the moderator density to find the optimum value, 0.20 g/cm³ H₂O. Figure 26 illustrates the calculated k_{eff} for both MKIV and MKIA fuel as a function of interspersed moderator density. These WIMS-E calculations are in very good agreement with previous calculations performed using the HAMMER lattice transport code (Fowler et. al. 1973). Table 3 contains the MCNP results for arrays of MKIA storage boxes. MCNP was used to analyze both full density and interspersed water moderation (density equal to 0.20 g/cm³) in addition to the wooden storage box moderation. For comparison, Table 4 contains results from MCNP, WIMS-E, and HAMMER for the infinite MKIA lattice, for both full and interspersed moderator density (water only).

All results contained in Table 3 were produced assuming a plywood density equal to 0.60 g/cm^3 . The chemical composition was assumed to consist of cellulose ($\text{C}_6\text{H}_{10}\text{O}_5$). Since there is some uncertainty in the density of all wood products, this density was varied for case number 5. For plywood density equal to 0.70 g/cm^3 , k_{eff} was equal to 0.57513 ± 0.0055 . For plywood density equal to 0.80 g/cm^3 , the calculated k_{eff} was 0.59337 ± 0.0060 .

Table 3. MCNP Results for MKIA Fuel Storage Boxes.

filename	Case	Case Description	k_{eff}	1σ	$k+2\sigma$
boxr	1	1 storage box, reflected boundaries	1.04480	0.0041	1.05337
boxv	2	1 storage box, vacuum boundaries	0.17618	0.0099	0.17967
boxs	3	2x3x1 array, concrete floor	0.32184	0.0082	0.32712
box_inf	4	2x∞x1 array, concrete floor	0.34208	0.0076	0.34728
box_ism	5	like #4, but interspersed moderation	0.60731	0.0061	0.61472
box_fdm	6	like #4, but full density moderation	0.59166	0.0062	0.59900
box2_i	7	2x∞x2 array, concrete floor	0.54543	0.0072	0.55328
box2_ism	8	like #7, but interspersed moderation	0.69595	0.0067	0.70528
box_3i	9	3x∞x2 array, concrete floor	0.62108	0.0073	0.63015
box_3ism	10	like #9, but interspersed moderation	0.73229	0.0053	0.74005
box_4i	11	4x∞x2 array, concrete floor	0.65370	0.0056	0.66102
box_2ii	12	2x∞x∞ array, concrete floor	0.75543	0.0045	0.76223
box_2iim	13	like #12, but interspersed moderation	0.75086	0.0048	0.75807
box_3ii	14	3x∞x∞ array, concrete floor	0.86698	0.0052	0.87600
box_3iim	15	like #14, but interspersed moderation	0.79122	0.0050	0.79913
box_4ii	16	4x∞x∞ array, concrete floor	0.92847	0.0049	0.93757
box_5ii	17	5x∞x∞ array, concrete floor	0.95515	0.0042	0.96317

- Notes:
1. Array stacking units are (height x width x depth)
 2. Unless otherwise stated, all cases are unmoderated, except for the moderation associated with the plywood box.
 3. Case 1 above is actually a k_{∞} , resulting from an infinite array of storage boxes, stacked infinitely in all directions.
 4. Interspersed moderation corresponds to a water density of 0.20 gm/cm^3 introduced in addition to the moderator associated with the wooden storage boxes.

Table 4. Code Comparison for MKIA Infinite Lattice.
(Square Pitch = 4.75 in.).

Computer Code	Full Density Moderator	Moderator Density = 0.20 g/cm^3
WIMS-E	0.74146	1.13832
MCNP	0.68898 ± 0.0045	1.12254 ± 0.0038
HAMMER	0.72	1.13

Figure 25. MCNP Model of Fuel Assembly Storage Box.

Figure 26. Interspersed Moderation.
Fixed Square Lattice Spacing = 4.75 in

4.0 MONTE CARLO VERIFICATION AND VALIDATION

MCNP calculations have been performed as a spot-check on the parametric calculations performed by WIMS-E/GOLF. MCNP is a Monte Carlo computer code under WHC software QA control that uses random number generators to simulate neutral particle transport through complex 3D user-defined geometries. MCNP was used to verify that the optimum geometries predicted to be critical by WIMS-E and GOLF are, in fact, critical. MCNP was executed on a sampling of these geometries, producing results for k_{eff} as shown in Table 5 below. Some of these data appear in WHC-SD-NR-007 Rev 1, but additional data presented here include results for MKIA and MKIV inners and outers.

Table 5. MCNP Results for WIMS-E/GOLF Comparison.

<u>Fissile Material</u>	<u>Radius(cm)⁽¹⁾</u>	<u>Geometry</u>	<u>MCNP k_{eff}</u>	<u>1σ</u>
MKIA Assemblies	42.357	Spherical	1.0010	0.0033
	63.649	Hemispherical	1.0128	0.0021
MKIA Inners	56.145	Spherical	0.9948	0.0030
	83.382	Hemispherical	1.0010	0.0025
MKIA Outers	38.488	Spherical	0.9968	0.0028
	58.038	Hemispherical	1.0137	0.0016
MKIV Assemblies	65.218	Spherical	0.9975	0.0016
	96.355	Hemispherical	1.0050	0.0031
MKIV Inners	56.211	Spherical	0.9951	0.0027
	83.427	Hemispherical	1.0078	0.0027
MKIV Outers	62.201	Spherical	0.9972	0.0025
	91.968	Hemispherical	1.0200	0.0021
0.5 in. OD Rods (0.95 wt% Scrap)	50.755	Spherical	0.9985	0.0027
	75.629	Hemispherical	1.0100	0.0027

(1) Radius predicted to be critical by WIMS-E/GOLF, assuming infinite (i.e. 25.0 cm) water reflector thickness.

Monte Carlo verifications similar to these in Table 5 were performed with the MONK5 code (Schwinkendorf, Appendix B 1992). Agreement was very good. Additional MCNP calculations were performed for slab and hemisphere geometries. In these cases, $k_{eff} = 0.98$ dimensions were calculated. Table 6 contains the results. Direct comparison between MCNP and WIMS-E/GOLF results are shown in cases 1, 3, 5, and 7 (i.e., the fully water-reflected cases). The final two cases combine slab and hemisphere geometries into a single configuration. A hemisphere with $k_{eff} = 0.98$ was placed on top of an infinite slab (also with $k_{eff} = 0.98$). This represents a "quadruple contingency" accident; both the slab and the hemisphere each contain twice the safe mass for their respective geometries. Both MKIV and MKIA assemblies yield $k_{eff} > 1$ for this configuration. These MCNP calculations also consistently show that the k_{eff} that results from assuming a concrete reflector (floor) is somewhat higher than by assuming water reflection in all directions.

Table 6. MCNP Results for Hemispherical Pile above Slab.

<u>Fissile Material</u>	<u>Radius/Height⁽²⁾</u>	<u>Geometry</u>	<u>MCNP k_{eff}</u>	<u>1σ</u>
MKIV Assemblies	50.0978 cm	Infinite Slab (water below)	0.9770	0.0023
MKIV Assemblies	50.0978 cm	Infinite Slab (concrete floor)	0.9806	0.0012
MKIA Assemblies	34.7984 cm	Infinite Slab (water below)	0.9809	0.0015
MKIA Assemblies	34.7984 cm	Infinite Slab (concrete floor)	0.9916	0.0017
MKIV Assemblies	82.1163 cm	Hemisphere (water below)	0.9884	0.0018
MKIV Assemblies	82.1163 cm	Hemisphere (concrete floor)	0.9857	0.0020
MKIA Assemblies	58.2711 cm	Hemisphere (water below)	0.9968	0.0023
MKIA Assemblies	58.2711 cm	Hemisphere (concrete floor)	0.9986	0.0025
MKIV Assemblies	H=50.0978 cm R=82.1163 cm	Hemisphere on top of slab	1.0112	0.0019
MKIA Assemblies	H=34.7984 cm R=58.2711 cm	Hemisphere on top of slab	1.0528	0.0012

(2) Dimension predicted to yield $k_{eff} = 0.98$ by WIMS-E/GOLF, assuming infinite (i.e., 25.0 cm) water reflector thickness. These dimensions are the radii for hemispherical and heights for slab geometries.

In addition to these MCNP spot-checks, both WIMS-E/GOLF and MCNP have been previously validated against results from critical experiments. Both WIMS-E/GOLF and MCNP were used to predict critical experiments performed with MKIA fuel assemblies (Schwinkendorf, Appendix A 1992). These experiments measured the approach to critical for MKIA fuel assemblies in water, in cylindrical geometry, with a core height equal to two assemblies. Three different lattice spacings were used. WIMS-E/GOLF was used to predict critical radii for finite cylinder geometry, and thus critical mass. For a hexagonal lattice spacing equal to 3.1 in. (for minimum critical mass), MCNP was executed assuming the same cylinder radius predicted to be critical by WIMS-E/GOLF. The MCNP result was $k_{eff} = 0.99203 \pm 0.0017$. MCNP predicted that the WIMS-E/GOLF result was 8 mk below critical. This WIMS-E/GOLF-predicted critical mass was slightly less than the experimental critical mass. Finally, MCNP was executed using the cylinder radius corresponding to the experimental dimension. This k_{eff} result was equal to 1.00, to well within the 1σ error.

5.0 CONCLUSIONS

This document has presented the results of analyses that show that current fuel storage box stacking limits, as specified in WHC-NR-M-4, *Nuclear Reactor Facilities Criticality Specifications*, pp. 3-7, are safe from a nuclear criticality safety standpoint.

One of the accident scenarios concerns a building roof collapse, with water flooding in conjunction with an ashfall from another volcanic eruption. Mt. St. Helens ash from the May 1980 eruption was high in silicon content. Calculations were performed to evaluate the effect of inclusion of silicon into the water moderator, for 0.95 wt% enriched uranium metal rods, used to represent scrap. The effect is small, and in the direction of decreased reactivity.

In the event of a fire and partial combustion of uranium metal, uranium oxide addition to the surrounding water only serves to reduce reactivity compared to the case with pure water moderator. This is because uranium in the water moderator causes neutrons to slow down to thermal energies in the presence of resonance absorbers (i.e., ^{238}U), which decreases the resonance escape probability, and therefore k_{∞} . New limits as contained in Table 1 of this document are all based on the most conservative assumptions of optimum moderation by pure water and infinite reflection. In the event of total combustion and removal of the wooden boxes, the remaining fuel would collapse into a slab of height greater than that in Table 1, but still substantially below the height required for $k_{\text{eff}} = 0.98$. This assumes that the array is stacked two boxes high. If there was a stacking error, such as having a full third layer of fuel boxes, the collapsed slab height would still not exceed the $k_{\text{eff}} = 0.98$ criterion, for a bare slab. If the array of boxes was only partially mis-stacked (e.g., one out of two, or four out of five columns), the resulting collapsed array height would be well within the $k_{\text{eff}} = 0.98$ criterion.

Another conservatism in the method used to calculate collapsed stack height was to divide the cross-sectional area occupied by a single fuel assembly (assuming optimum spacing), multiplied by the number of assemblies in a stack, by the length of the fuel box. Without taking credit for the fact that fuel lengths are less than the storage box widths, this results in a collapsed fuel slab that is too high, by this ratio (box width/fuel length). This approach was used because different fuel types have different lengths. However, MKIA fuel is the shortest length used. If the collapsed stack height were to be calculated by dividing the total volume occupied by fuel, by the floor area of the box, the $k_{\text{eff}} = 0.98$ criterion is met easily. Without this correction, the optimally-collapsed full third layer slab is just at this limit: $k_{\text{eff}} = 0.982$ for a collapsed optimally-spaced bare slab.

Storage box arrays of MKIA assemblies have been analyzed. MKIA fuel assemblies are the most reactive fuel because they contain 1.25 wt% enriched outers, and are therefore, a bounding case. Under normal conditions, calculated k_{eff} is substantially subcritical. Even under water flooding conditions (either full density or interspersed moderation of optimum density), storage array k_{eff} is less than 0.90, even for arrays that are infinite in length and width, and stacked three boxes high. The corresponding k_{eff} for two boxes high (the maximum allowed by current criticality safety

specifications) never exceeds 0.80. Boxes stacked to five high, without interspersed moderation, reached a k_{eff} of 0.955 and a k_{eff} of 0.963 at 2σ . Introduction of interspersed moderation would reduce the k_{eff} to less than 0.95 because of the effect of thermal versus absorption effects of water introduction in small versus large arrays.

6.0 REFERENCES

- Carter, L. L. 1991, *Certification of MCNP Version 3B for the Hanford Cray*, WHC-SD-MP-SWD-30001, Rev. 3, Westinghouse Hanford Company, Richland, Washington.
- Fowler, K. L., J. P. McNeece, and H. Toffer, 1973, *Criticality Safety Analysis Report Movement of Unirradiated Reactor Fuel Between 300 Area and N Reactor*, UNI-58, United Nuclear Industries, Inc., 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.
- Rogers, C. A., 1991, *Use of K_{eff} Subcritical Limit of 0.98*, WHC-SD-SQA-CSA-20330, Westinghouse Hanford Company, Richland, Washington.
- Schwinkendorf, K. N., 1992, *Criticality Safety Evaluation Report, 105-KE Basin Fuel Encapsulation*, WHC-SD-NR-CSER-007, Rev 1, Westinghouse Hanford Company, Richland, Washington.
- Schwinkendorf, K. N., 1994, *Software Certification Package for the GOLF Code*, WHC-SD-NR-SWD-024, Westinghouse Hanford Company, Richland, Washington.

APPENDIX A

Selected WIMS-E Input files for Uranium Metal Pieces

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The following are selected WIMS-E input files for 1.25 wt% enriched uranium metal billets, and for both MKIV and MKIA N Reactor fuel assemblies. These input cases represent lattice spacing searches, used for the determination of maximum reactivity and minimum mass required for a given effective multiplication factor. The following is a listing of the WIMS-E input file for 1.25 wt% enriched billets:

* 1.25 WT% URANIUM BILLETS -- MK-I-A BILLETS

```

NCYCLE 1 11
XHEAD 1
BURN
NMESH 45
NREGION 3
NMATERIAL 2
ENDP
PRINT 0
CENT
MESH 10 15 20
MATERIAL 1 18.770 293.0 1 2235 1.2491 8238 98.7509
MATERIAL 2 1.00 293. 3 3001 11.19 6016 88.81
ANNULUS 1 3.55600 2 *MODERATOR
ANNULUS 2 8.85190 1 *FUEL
QUAL 1 1
ANNULUS 3 9.50 2 *MODERATOR
QUAL 1 2
ANNULUS 3 9.75 2 *MODERATOR
QUAL 1 3
ANNULUS 3 10.00 2 *MODERATOR
QUAL 1 4
ANNULUS 3 10.25 2 *MODERATOR
QUAL 1 5
ANNULUS 3 10.50 2 *MODERATOR
QUAL 1 6
ANNULUS 3 10.75 2 *MODERATOR
QUAL 1 7
ANNULUS 3 11.00 2 *MODERATOR
QUAL 1 8
ANNULUS 3 11.25 2 *MODERATOR
QUAL 1 9
ANNULUS 3 11.50 2 *MODERATOR
QUAL 1 10
ANNULUS 3 11.75 2 *MODERATOR
QUAL 1 11
ANNULUS 3 12.00 2 *MODERATOR
BEGIN
*
* PRES CALCULATION
*
```

```

WPRES 1 2
GROUPS 15 27
NUCLIDE 2235 293.
NUCLIDE 8238 293.
PRINT 0
BEGIN
*
* PERSEUS CALCULATIONS
*
WTHES 2
PRINT 0
BEGIN
*
WRES 2 3
GROUPS 15 27
NUCLIDE 2235 293.
NUCLIDE 8238 293.
PRINT 0
BEGIN
*
* PERSEUS CALCULATIONS
*
WTHES 3
PRINT 0
BEGIN
*
WPIP 3
PRINT 0
TOLER 1.0E-5
OMEGA 1.25
BEGIN
*
* COLLAPSING TO 2 GROUPS TO OUTPUT TWO-GROUP CROSS SECTIONS
*
WCOND 3 4
GROUP 2
ENDP
PART 27 69
BEGIN
*
* SMEAR THE CELL INTO ONE REGION
*
WSMEAR 4 5
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 32 33 34 35 36 37 38 39 40
        41 42 43 44 45
MCODE 1
BEGIN

```

```

*
* PRINT THE 2-GROUP CROSS SECTIONS AND SAVE IT ON UNIT 7 (FROM SNAP)
*
WINTER 5
PRINT
FILE 3
MATRIX
SNAP
BEGIN
FINISH
STOP

```

The following WIMS-E input file is for MKIV N Reactor fuel assemblies:

```

*****
* CRITICALITY CALCULATIONS FOR MKIV FUEL
*****
NCYCLE 1 11
XHEAD 1
NMESH 30
NREGION 9
NMATERIAL 3
ENDP
PRINT 0
CENT
MESH 4 1 4 1 4 1 4 1 10
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
ANNULUS 1 0.6095 1 *INNER WATER REGION
ANNULUS 2 0.6605 2 *INNER CLAD
ANNULUS 3 1.5480 3 *INNER FUEL
ANNULUS 4 1.6245 2 *CLAD
ANNULUS 5 2.1605 1 *WATER
ANNULUS 6 2.2110 2 *CLAD
ANNULUS 7 3.0165 3 *OUTER FUEL
ANNULUS 8 3.0800 2 *OUTER CLAD
QUAL 1 1
ANNULUS 9 3.60071 1 *CELL RADIUS
QUAL 1 2
ANNULUS 9 3.73407 1 *CELL RADIUS
QUAL 1 3
ANNULUS 9 3.86743 1 *CELL RADIUS
QUAL 1 4
ANNULUS 9 4.00079 1 *CELL RADIUS
QUAL 1 5
ANNULUS 9 4.13415 1 *CELL RADIUS
QUAL 1 6
ANNULUS 9 4.26751 1 *CELL RADIUS
QUAL 1 7
ANNULUS 9 4.40086 1 *CELL RADIUS
QUAL 1 8
ANNULUS 9 4.53422 1 *CELL RADIUS
QUAL 1 9

```

```

ANNULUS 9 4.66758 1 *CELL RADIUS
QUAL 1 10
ANNULUS 9 4.80094 1 *CELL RADIUS
QUAL 1 11
ANNULUS 9 4.93430 1 *CELL RADIUS
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 2 GROUPS TO OUTPUT TWO-GROUP CROSS SECTIONS
*
WCOND 3 4
GROUP 2
ENDP
PART 27 69
BEGIN
*
* SMEAR THE CELL INTO ONE REGION
*
WSMEAR 4 5
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

```

```

MCODE 1
BEGIN

```

```

*
```

```

* PRINT THE 2-GROUP CROSS SECTIONS AND SAVE IT ON UNIT 7 (FROM SNAP)

```

```

*
```

```

WINTER 5

```

```

PRINT

```

```

FILE 3

```

```

MATRIX

```

```

SNAP

```

```

BEGIN

```

```

FINISH

```

```

STOP

```

The following WIMS-E input file is for MKIA N Reactor fuel assemblies:

```

*****

```

```

* CRITICALITY CALCULATIONS FOR MKIA FUEL

```

```

*****

```

```

NCYCLE 1 11

```

```

XHEAD 1

```

```

NMESH 30

```

```

NREGION 9

```

```

NMATERIAL 4

```

```

ENDP

```

```

PRINT 0

```

```

CENT

```

```

MESH 4 1 4 1 4 1 4 1 10

```

```

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 99.0062

```

```

                234 0.000070 236 0.000397

```

```

MATERIAL 4 18.580 296.0 1 2235 1.2491 8238 98.7092

```

```

                234 0.000085 236 0.000332

```

```

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

```

```

QUAL 1 1

```

```

ANNULUS 9 3.60071 1 *CELL RADIUS

```

```

QUAL 1 2

```

```

ANNULUS 9 3.73407 1 *CELL RADIUS

```

```

QUAL 1 3

```

```

ANNULUS 9 3.86743 1 *CELL RADIUS

```

```

QUAL 1 4

```

```

ANNULUS 9 4.00079 1 *CELL RADIUS

```

QUAL 1 5
 ANNULUS 9 4.13415 1 *CELL RADIUS
 QUAL 1 6
 ANNULUS 9 4.26751 1 *CELL RADIUS
 QUAL 1 7
 ANNULUS 9 4.40086 1 *CELL RADIUS
 QUAL 1 8
 ANNULUS 9 4.53422 1 *CELL RADIUS
 QUAL 1 9
 ANNULUS 9 4.66758 1 *CELL RADIUS
 QUAL 1 10
 ANNULUS 9 4.80094 1 *CELL RADIUS
 QUAL 1 11
 ANNULUS 9 4.93430 1 *CELL RADIUS
 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 2 GROUPS TO OUTPUT TWO-GROUP CROSS SECTIONS

*

WCOND 3 4
 GROUP 2
 ENDP

PART 27 69

BEGIN

*

* SMEAR THE CELL INTO ONE REGION

*

WSMEAR 4 5

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

MCODE 1

BEGIN

*

* PRINT THE 2-GROUP CROSS SECTIONS AND SAVE IT ON UNIT 7 (FROM SNAP)

*

WINTER 5

PRINT

FILE 3

MATRIX

SNAP

BEGIN

FINISH

STOP

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APPENDIX B

Computed Masses and Dimensions for $k_{eff} = 0.98$ and $k_{eff} = 1.00$

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$k_{eff} = 0.98$ Masses and Dimensions

	Sphere		Hemisphere		Infinite Cylinder		Infinite Slab	
	Mass(lb)	Dia.(in)	Mass(lb)	Dia.(in)	Mass(lb/ft)	Dia.(in)	Mass(lb/ft ²)	Thick(in)
MKIV (0.95 wt% inner and outer)								
Billets	-----unlimited-----							
Assemblies	10458	43.5	17151	64.6	2340	32.0	645	19.7
Outers	7210	41.7	11847	62.0	1679	30.7	483	19.0
Inners	6169	38.5	10245	57.5	1543	28.2	479	17.9
Scrap	4301	36	7207	53.9	1134	27.5	361	18.3
Solution	-----unlimited-----							

MKIA (0.95 wt% inner, 1.25 wt% outer)								
Billets	19719	45.3	31934	67.0	4269	33.4	1168	20.0
Assemblies	3158	30.3	5433	45.8	972	22.6	374	13.7
Outers	1864	27.8	3239	42.2	617	20.6	254	12.2
Inners	6352	38.6	10568	57.6	1551	29.5	474	17.4
Scrap	1173	25.4	2074	38.7	417	19.0	177	11.8
Solution	3007	35.4	5078	53.1	809	25.8	260	17.8

	<u>kg</u>	<u>cm</u>	<u>kg</u>	<u>cm</u>	<u>kg/m</u>	<u>cm</u>	<u>kg/m²</u>	<u>cm</u>
MKIV (0.95 wt% inner and outer)								
Billets	-----unlimited-----							
Assemblies	4744	110.5	7780	164.1	3482	81.3	3149	50.0
Outers	3270	105.9	5374	157.5	2499	78.0	2358	48.3
Inners	2798	97.8	4647	146.1	2296	71.6	2339	45.5
Scrap	1951	91.4	3269	136.9	1688	69.9	1763	46.5
Solution	-----unlimited-----							

MKIA (0.95 wt% inner, 1.25 wt% outer)								
Billets	8944	115.1	14485	170.2	6353	84.8	5703	50.8
Assemblies	1432	77.0	2464	116.3	1446	57.4	1826	34.8
Outers	845	70.6	1469	107.2	918	52.3	1240	31.0
Inners	2881	98.0	4794	146.3	2308	74.9	2314	44.2
Scrap	532	64.5	941	98.3	621	48.3	864	30.0
Solution	1364	89.9	2303	134.9	1204	65.5	1269	45.2

Note that MKIA billets, scrap, and solution results are based on 1.25 wt% enrichment.

Critical Masses and Dimensions

	Sphere		Hemisphere		Infinite Cylinder		Infinite Slab	
	Mass(lb)	Dia.(in)	Mass(lb)	Dia.(in)	Mass(lb/ft)	Dia.(in)	Mass(lb/ft ²)	Thick(in)
MKIV (0.95 wt% inner and outer)								
Billets	-----unlimited-----							
Assemblies	17184	51.3	27709	75.8	3299	38.0	786	24.0
Outers	11624	48.9	18787	72.4	2336	36.2	586	23.0
Inners	9303	44.2	15207	65.6	2052	32.5	570	21.4
Scrap	6362	39.9	10525	59.5	1510	30.3	435	18.9
Solution	-----unlimited-----							
MKIA (0.95 wt% inner, 1.25 wt% outer)								
Billets	31392	52.9	50120	77.9	5883	39.3	1390	23.8
Assemblies	4173	33.3	7080	50.1	1189	24.2	421	14.5
Outers	2397	30.3	4109	45.6	741	22.6	284	13.7
Inners	9514	44.2	15583	65.6	2098	34.3	559	20.5
Scrap	1481	27.4	2584	41.6	497	20.8	198	13.2
Solution	4563	40.7	7570	60.7	1082	29.8	314	24.3
	<u>kg</u>	<u>cm</u>	<u>kg</u>	<u>cm</u>	<u>kg/m</u>	<u>cm</u>	<u>kg/m²</u>	<u>cm</u>
MKIV (0.95 wt% inner and outer)								
Billets	-----unlimited-----							
Assemblies	7795	130.3	12569	192.5	4909	96.5	3838	61.0
Outers	5273	124.2	8522	183.9	3476	91.9	2861	58.4
Inners	4220	112.3	6898	166.6	3054	82.6	2783	54.4
Scrap	2886	101.3	4774	151.1	2247	77.0	2124	48.0
Solution	-----unlimited-----							
MKIA (0.95 wt% inner, 1.25 wt% outer)								
Billets	14239	134.4	22734	197.9	8755	99.8	6787	60.5
Assemblies	1893	84.6	3211	127.3	1769	61.5	2056	36.8
Outers	1087	77.0	1864	115.8	1103	57.4	1387	34.8
Inners	4315	112.3	7068	166.6	3122	87.1	2729	52.1
Scrap	672	69.6	1172	105.7	740	52.8	967	33.5
Solution	2070	103.4	3434	154.2	1610	75.7	1533	61.7

Note that MKIA billets, scrap, and solution results are based on 1.25 wt% enrichment.

Old Safe Masses and Associated Dimensions

	Sphere		Hemisphere		Infinite Cylinder		Infinite Slab	
	<u>Mass(lb)</u>	<u>Dia.(in)</u>	<u>Mass(lb)</u>	<u>Dia.(in)</u>	<u>Mass(lb/ft)</u>	<u>Dia.(in)</u>	<u>Mass(lb/ft²)</u>	<u>Thick(in)</u>
MKIV (0.95 wt% inner and outer)								
Billets	-----unlimited-----							
Assemblies	4810	33.4	7720	49.7	970	20.8	240	7.2
Outers	4380	34.9	6960	51.4	850	21.6	205	7.5
Inners	2420	28.6	3920	42.3	560	17.6	165	6.0
Scrap	2180	19.7	2220	N/A	N/A	N/A	156	5.5
Solution	-----unlimited-----							
MKIA (0.95 wt% inner, 1.25 wt% outer)								
Billets	17500	N/A	27600	N/A	N/A	28.5	630	11.0
Assemblies	1490	23.7	2480	35.5	436	14.8	160	5.4
Outers	1100	23.0	1840	34.5	330	14.4	120	5.3
Inners	2630	29.1	4250	43.1	607	17.8	174	6.1
Scrap	560	N/A	1190	N/A	N/A	11.4	80	4.0
Solution	3340	N/A	N/A	N/A	N/A	24.7	N/A	9.4

N/A = Data Not Available