

**SRTC CRITICALITY TECHNICAL REVIEW OF
SRT-CMA-930039**

by
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SAVANNAH RIVER TECHNOLOGY CENTER
Applied Technology Section
Applied Physics Group

SRT-CMA-930069

SRTC CRITICALITY SAFETY TECHNICAL REVIEW:

NUCLEAR CRITICALITY SAFETY EVALUATION; (SRT-CMA-930039)
DWPF MELTER-BATCH 1 (U)

December 3, 1993

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SUMMARY

Review of SRT-CMA-930039, "Nuclear Criticality Safety Evaluation; DWPF Melter-Batch 1 (U)," December 1, 1993, (Reference 1) has been performed by the SRTC Applied Physics Group. The NCSE is a criticality assessment of the Melt Cell in the DWPF. Additionally, this pertains only to Batch 1 operation, which differs from batches to follow. Plans for subsequent batch operations call for fissile material in the Salt Cell feed-stream, which necessitates a separate criticality evaluation in the future.

The NCSE under review concludes that the process is safe from criticality events, even in the event that all lithium and boron neutron poisons are lost, provided uranium enrichments are less than 40%. Furthermore, if all the lithium and as much as 98% of the boron would be lost, uranium enrichments of 100% would be allowable. After a thorough review of the NCSE, this reviewer agrees with that conclusion.

SCOPE OF TECHNICAL REVIEW

This technical review consisted of:

- an independent check of the methods and models employed,
- independent calculations
- application of ANSI/ANS 8.1,
- verification of WSRC Nuclear Criticality Safety Manual⁽²⁾ procedures.

DOCUMENTATION

Issuance of this memorandum transmits this technical review as critical data.

METHOD AND MODEL REVIEW

Method:

Cross-Sections

The 16-Group Hansen-Roach cross-section data library was employed. The Hansen-Roach Library is an extensively utilized database for criticality safety analysis.

Computer Codes

Cross-section preparation and processing was performed with the Joshua 70 version of HRXN⁽³⁾. This is a validated code developed at SRS. HRXN is capable of calculating k-infinity, and was also employed in this capacity.

One-dimensional criticality calculations were performed with the discrete-ordinates transport code, ANISN⁽³⁾, an SRS validated criticality calculational code developed at Oak Ridge National Laboratory and included in the Joshua 70 system at SRS. This code is widely used throughout the industry.

Model:

The materials used in the HRXN calculations were provided to this reviewer in microfiche form. The same applies to the ANISN model calculations. These output listings were thoroughly checked and verified.

EVALUATION

Data to perform this evaluation were derived from the NCSE under review, from references listed in the NCSE⁽¹⁾, from a previous technical review⁽⁴⁾ by this reviewer and from private communications with the author.

NCSE CONTENT EVALUATION

Bias Applied, Subcritical Margin and K-Safe:

Bias: The bias applied is 0.03, the maximum for uranium systems predicted with the HRXN/ANISN criticality method. Since the dominant fissile material under consideration is U-235, this is appropriate and conservative. Although some plutonium exists in the streams, it is only a small fraction of the uranium, and furthermore, plutonium has a smaller bias (0.011 maximum).

Subcritical Margin: The subcritical margin imposed for both normal and accident conditions is $0.05 \Delta k$, which is a commonly used value for criticality evaluations.

K-safe: The K-safe used for the NCSE is $1.000 - 0.030(\text{bias}) - 0.050(\text{margin}) = 0.920$. Thus, the k-eff predicted for acceptable configurations will be less than or equal to this value.

Review of NCSE Conclusions:

The NCSE under review concludes that the process is safe from criticality events. This includes the melter feed and subsequent melter glass product. A large array of calculations were performed in order to make this statement. Although the uranium

enrichment is not expected to exceed 1%, it was demonstrated that for many situations, the U-235 enrichment could be as high as 40% and k-safe would not be exceeded. This conclusion is dependent upon the assurance that materials remain well mixed, i. e., the fissile materials do not separate out. Under normal operation this is not expected to occur. However, significant loss of neutron poison in the form of boron and lithium could occur without jeopardizing criticality safety.

Normal conditions:

Under normal conditions, k-infinity is well below k-safe. The k-infinity calculated for any optimally moderated process is 0.1074, even for 100% enriched U-235, although the expected enrichment will be below 1% enriched. ANSI/ANS 8.1 states that any concentration is allowed, if enrichments less than 0.96% U-235 are assured. This application of the ANSI Standard only applies if the plutonium is shown to be the stated small fraction (less than 1%) of the uranium level.

During this review, it was discovered that material flowsheets for the Salt Cell to SRAT stream indicate that fissile materials will be in that feed-line. However, for Batch 1 operation this stream is to be simulated without fissile and thus does not require a criticality evaluation. *Therefore, it is cautioned that the flowsheet data indicating the Salt Cell to SRAT stream contains fissile is to be ignored.* The flow stream exiting the SRAT vessel, and subsequently enters the melter feed stream, did include fissile added by this stream and is therefore conservative.

Accident conditions:

Double contingency was not addressed, per se, due to the large margin between the expected 1% and the analyzed 40% uranium enrichments, and since all lithium and boron poisons were allowed to separate out. Optimum moderation calculations predict that a 40% enrichment is allowable.

☺☺☺ INDEPENDENT HRXN CALCULATIONS ☺☺☺

The basic conclusion derived in the NCSE was that the process remains criticality safe if the enrichment does not exceed 40% U-235. This conclusion was based on a 1/100 water fraction, which yields a 151 H/U-235 ratio, stated to be near optimum moderation. Independent HRXN calculations, that varied the melter feed water content associated with 40% enriched fuel, were performed to validate the optimum H/U-235 ratio conclusion. These results are given in Table 1. The first calculation in the table is a repeat of the HRXN problem on which the basic NCSE conclusion is made.

Table 1
HRXN Analysis Varying Melter Feed Water Content

Water Fraction	H/U-235 Ratio	K-Infinity	Comments
1/100	151	0.853	Repeat of NCSE case which yielded 0.853
1/500	61	0.834	Decrease in K
1/25	481	0.836	Decrease in K

Conclusion from independent calculations: Optimum water content is modeled. I agree with NCSE conclusions.

SAFETY MANUAL FORMAT AND PROCEDURES

The WSRC Nuclear Criticality Safety Manual spells out certain requirements that are to be included in a specifically formatted NCSE. This section reviews the compliance with that document.

SECTION	REMARKS
1.0 Introduction:	Included with appropriate contents
2.0 Description:	Included with appropriate contents
3.0 Requirements Documentation	Included with appropriate contents
4.0 Methodology	Included with appropriate contents
5.0 Discussion of Contingencies	Included with appropriate contents
6.0 Evaluation of Results	Included with appropriate contents
7.0 Design Features (entered as Administratively Controlled Limits and Requirements)	Included with appropriate contents
8.0 Summary and Conclusions	Included with appropriate contents
9.0 References	Included with appropriate contents

REFERENCES

1. SRT-CMA-930039, T. G. Williamson, "Nuclear Criticality Safety Evaluation; DWPF Melter-Batch 1 (U)," December 1, 1993.
2. WSRC Nuclear Criticality Safety Manual (U), WSRC-IM-93-13, Rev.1, 7/1/93.
3. DPSTM-86-700-3; H. K. Clark, "JOSHUA Nuclear Criticality Safety Modules," March 1987.
4. SRT-CMA-930070; R. W. Rathbun, "SRTC Criticality Safety Technical Review, Nuclear Criticality Safety Evaluation; (SRT-CMA-930034), DWPF Chemical Processing Cell-Batch 1 (U)," November 10, 1993.

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NUCLEAR CRITICALITY SAFETY EVALUATION

DWPf MELTER - BATCH 1(U)

December 1, 1993

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1.0 INTRODUCTION

The Savannah River Site (SRS) High Level Nuclear Waste will be vitrified in the Defense Waste Processing Facility (DWPf) for long term storage and disposal. This is a preliminary safety evaluation for the Melt Cell of the DWPf vitrification process for Batch 1 waste.

This evaluation demonstrates that the material in the Melt Cell remains subcritical for the contents of Batch 1 which contains uranium with less than 1% by weight U-235 (1).

2.0 DESCRIPTION

Waste salt solution is processed in the Tank Farm In-Tank Precipitation process to remove cesium-137 and to adsorb strontium on monosodium titanate (MST). The filtered tetraphenylborate slurry is then washed in the DWPf Late Wash Facility (LWF) before it is fed to the DWPf Salt Processing Cell (SPC). In the Salt Processing Cell the precipitate slurry is processed in the Precipitate Reactor (PR). The Precipitate Hydrolysis Aqueous (PHA) product from the PR is then combined with the sludge feed and the frit in the DWPf Chemical Process Cell (CPC) to produce a melter feed. In the Melt Cell the waste is immobilized in a glass matrix and transferred to sealed canisters.

3.0 REQUIREMENTS DOCUMENTATION

There are no specific documents which apply uniquely to this evaluation.

4.0 METHODOLOGY

The JOSHUA (J70) Nuclear Criticality Safety Module (2) HRXN was used for this analysis. HRXN computes the atom densities for mixtures and combines them with the Hansen-Roach 16-group nuclear cross sections to calculate infinite system multiplication factors and to prepare cross sections for transport codes. ANISN solves the one-dimensional radiation transport equation in the discrete ordinates approximation.

Calculation biases for these modules have been established by Clark for uranium and plutonium (3,4). The DWPf waste contains uranium, plutonium and traces of other actinides, however, the primary fissile material is uranium. For uranium systems with a hydrogen-to-fissile (H/F) of 100, HRXN-ANISN calculates a K_{eff} of 0.97 for a critical system. At this point on the bias curve, H/F = 100, the calculated multiplication factor of 0.97 is at a minimum so its use for other H/F ratios is conservative. With a safety factor of 0.05 a K_{eff} of 0.92 would be safe. For plutonium systems at the same H/F ratio, HRXN-ANISN calculates a K_{eff} of 1.00

Attachment 1: SRT-CMA-930039

for a critical system so a calculated multiplication factor of 0.92 would be safe for plutonium. Because of the lack of experimental data, there are no bias corrections for other materials, such as iron, manganese, aluminum and sodium. The safety margin of 0.05 is sufficient to cover the bias for these materials.

5.0 DISCUSSION OF CONTINGENCIES

This evaluation is for feed to the melter in which the only fissile materials are in Batch 1 sludge. Batch 1 sludge contains uranium enriched to less than 1% U-235 (1). The evaluation demonstrates that the melter is safe from a criticality standpoint as long as the U-235 enrichment remains below 40% and there are no mechanisms to separate the fissile material from other metal oxides. This calculation assumes that the melt mixture is near the optimum hydrogen-to-fissile ratio and that the neutron poisons, boron and lithium, are not present. Because of the conservatism of assuming optimum H/F ratios and the assumption that all of the boron and lithium poisons are removed, and, because of the large margin between U-235 enrichments of 40% and less than 1%, a double contingency analysis was not performed. Assurance must be given that the sludge feed is properly represented by reference 6 and that there is no credible mechanism for isolating the fissile materials.

This analysis is for Batch 1 feed in a melter which contains no fissile materials from previous runs. For different feed material compositions and for multiple batches, additional evaluations must be performed.

6.0 EVALUATION AND RESULTS

Clark (5) has analyzed the melter for the case in which the appropriate amount of frit has not been added and the melter inventory becomes saturated with waste oxides. He analyzed for uranium enriched to 2.66% U-235 and plutonium with 60.83% Pu-239. For this case HRXN calculates a safe infinite multiplication factor, K_{inf} , of 0.075. Clark also has shown that the system remains safe if the fissile isotope content is doubled.

Feed material for the melter comes from the Slurry Mix Evaporator (SME) as an aqueous slurry containing 58 weight percent water. This water is driven off in the melter. Contents of the feed and the glass are listed in Table 1 (6). For this analysis the manganese, magnesium and sodium formates have been converted to an equivalent quantity of metal oxides.

Table 1
Melter Composition

	Melter Feed		Glass	
	Lb/hr	Wt Frac.	Lb/hr	Wt Frac.
Water	340.03	5.79E-01		
Al ₂ O ₃	10.28	1.75E-02	10.17	4.56E-02
B ₂ O ₃	12.96	2.21E-02	17.25	7.74E-02
Ca(NO ₃) ₂	2.61	4.44E-03		
Cu(COOH) ₂	1.55	2.64E-03		
Fe ₂ O ₃	23.73	4.04E-02	21.14	9.49E-02
FeO			2.11	9.49E-03
H ₃ BO ₃	8.08	1.38E-02		
KOOH	9.48	1.61E-02		
K ₂ O	0.11	1.92E-04	5.38	2.41E-02
Li ₂ O	11.34	1.93E-02	11.23	5.04E-02
(Li ₆) ₂ O	0.85	1.45E-03	0.84	3.78E-03
(Li ₇) ₂ O	10.49	1.79E-02	10.39	4.66E-02
MgO	4.67	7.96E-03	4.63	2.08E-02
MnO ₂	3.18	5.41E-03		
MnO			2.77	1.24E-02
Na ₂ O	14.01	2.39E-02	19.57	8.78E-02
NaNO ₃	15.84	2.70E-02		
SiO ₂	126.00	2.15E-01	124.70	5.60E-01
U ₃ O ₈	2.33	3.97E-03	2.31	1.03E-02
PuO ₂	0.01	1.47E-05	0.01	3.84E-05
Sum	586.98	1.00	222.84	1.00
Flow	599.1 Lb/hr		228.0 Lb/hr	
Density	82.65 Lb/ft ³		156.6 Lb/ft ³	
	1.306 g/cc		2.475 g/cc	
% Flow	98.0%		97.7%	

The materials listed in Table 1, which constitute 98.0% and 97.7% of the mass of the feed and the glass respectively, were analyzed with the results shown in Table 2. For these cases the uranium was treated as U₃O₈ at 100% U-235 and plutonium as PuO₂ at 100% Pu-239.

Table 2
HRXN Analysis
Materials from Table 1
(JOB 1321)

	K_{inf}
Melter Feed	0.0285
Melter Glass	0.1074

(The notation JOB 1321 indicates that the values in the table are on microfiche DWPf6508 Job 1321)

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These materials contain boron with 19.9% boron-10 and lithium with 7.5% Lithium-6, both of which are good neutron poisons. An analysis was done for the extreme cases in which all of the boron and lithium were removed and for different water contents in the feed material. These results are in Table 3.

Table 3
HRXN Analysis
Reduced Boron and Water
(JOB 1321)

Melter Feed	K_{inf}
1/10 water, All B & Li, 100 % U-235	0.0429
1/100 water, All B & Li, 100 % U-235	0.0721
All water, No B & Li, 100 % U-235	0.571
1/100 water, No B & Li, 100 % U-235	1.298
1/100 water, No B & Li, 40% U-235	0.853
1/100 water, 2% B, No Li, 100% U-235	0.711
1/100 water, 1% B, No Li, 100% U-235	0.992
Glass	
No B & Li, 100 % U-235	1.329
No B & Li, 50 % U-235	0.956
No B & Li, 40 % U-235	0.854
2% B, No Li, 100 % U-235	0.848
1% B, No Li, 100 % U-235	1.01

The notation 1/10 water means that the water content was reduced to 1/10 of the value in Table 1 and the mixture density adjusted on the basis that all materials are in solution. The material contents for these cases are listed in Appendix 1. The notation no B & Li, 1% B or 2% B means that the B_2O_3 , H_3BO_3 and the Li_2O contents were reduced and the solution density adjusted accordingly. For the case of 1/100 water the hydrogen-to-fissile ratio, H/F, is 151, which is near the most reactive H/F combination. These data indicate that, even without corrections for finite geometry, and, in the extreme case for which no boron or lithium is included, the melter will always remain subcritical as long as the uranium enrichment remains below 40% U-235 and the metal oxides remain with the fissile oxides. If the boron content of the glass is greater than 2 % of the value in Table 1, the melter remains safe even for 100 % U-235.

The active volume of the melter is cylindrical, 6 feet (1.83 m) in diameter with a nominal height 34 inches (0.86 m) (7). The melter nominally contains 5.61×10^3 kg glass. For the weight fractions listed in Table 1, the glass contains 57.8 kg U_3O_8 and 216 g PuO_2 . At 40 % uranium enrichment and 100% Pu-239 this mass corresponds to 19.6 kg U-235 and 190 g Pu-239. A cylinder with radius 0.91 r and thickness 0.86 m is a

flat, "high leakage" geometry. An ANISN calculation with the glass material and no boron or lithium ($K_{inf} = 1.329$) in an infinite planar geometry 0.86 m thick with 30-cm water reflection on both sides has $K_{eff} = 0.85$ (JOB 867). A correction for radial leakage in this planar geometry reduces K_{eff} to 0.50. Thus, if all of the boron and lithium were lost and if 100% enriched uranium were in the melter and if the material in the melter were constrained to a thickness no greater than 0.86 m, the system would remain safe.

CONCLUSION

Compositions of the feed material and of the glass from reference 6 have been used to determine the criticality safety of the Melter under normal conditions. These materials remain safe from a criticality standpoint. For a case in which all of the boron and lithium is lost, the melter remains safe for feed material described in Table 1 with the uranium enriched to no more than 40% U-235. For the composition listed in Table 1 this value corresponds to 19.6 kg U-235 and 190 g Pu-239. If the boron (19.9% B-10) content of the glass is greater than 2% of the value in Table 1, the melter remains safe even for 100% U-235. The material in the DWPF sludge feed Batch 1 can be processed in the melter cell safely since it contains uranium with less than 1% by weight U-235 (1).

REFERENCES

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6. A. S. Choi, "Material Balance Tables for the DWPF Coupled Feed Flow Sheet with Batch 1 Sludge (U)", WSRC-TR-93-203, Rev. 0, (1993).
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APPENDIX A

COMPOSITIONS OF MATERIALS IN MELTER

The following tables were generated on a spread sheet with the assumption that the materials are in solution and the solution density is determined from

$$\frac{1}{\rho_s} = \sum_i \frac{(wt. frac)_i}{\rho_i}$$

where the subscript s refers to the solution and i to the individual components. Compound densities were taken from the Handbook of Physics and Chemistry, 34th edition.

Melter Feed to DWPF	Lb/hr	Wt. Frac.	Density	Wt. Frac/ Density
Water	340.03	5.80E-01	1.00	0.5802
Al ₂ O ₃	10.28	1.75E-02	3.50	0.0050
B ₂ O ₃	12.96	2.21E-02	1.84	0.0120
Ca(NO ₃) ₂	2.61	4.45E-03	2.36	0.0019
Cu(COOH) ₂	1.55	2.64E-03	1.83	0.0014
Fe ₂ O ₃	23.73	4.05E-02	5.24	0.0077
H ₃ BO ₃	8.08	1.38E-02	1.44	0.0096
KCOOH	9.48	1.62E-02	1.91	0.0085
Li ₂ O	11.34	1.93E-02	2.01	0.0096
(Li6) ₂ O	0.85	1.45E-03		
(Li7) ₂ O	10.49	1.79E-02		
MgO	4.67	7.97E-03	3.58	0.0022
MnO ₂	3.18	5.42E-03	5.03	0.0011
Na ₂ O	14.01	2.39E-02	2.27	0.0105
NaNO ₃	15.84	2.70E-02	2.26	0.0120
SiO ₂	126.00	2.15E-01	2.32	0.0927
U ₃ O ₈	2.33	3.97E-03	8.39	0.0005
PuO ₂	0.01	1.47E-05	11.46	1.287E-06
Sum	586.08	1.00		0.75
		Density g/cc		1.325

Melter Feed 1/10 Water	Lb/hr	Wt. Frac.	Density	Wt. Frac/ Density
Water	34.03	1.22E-01	1.00	0.1215
Al ₂ O ₃	10.28	3.67E-02	3.50	0.0105
B ₂ O ₃	12.96	4.63E-02	1.84	0.0251
Ca(NO ₃) ₂	2.61	9.31E-03	2.36	0.0039
Cu(COOH) ₂	1.55	5.53E-03	1.83	0.0030
Fe ₂ O ₃	23.73	8.47E-02	5.24	0.0162
H ₃ BO ₃	8.08	2.88E-02	1.44	0.0200
KCOOH	9.48	3.38E-02	1.91	0.0177
Li ₂ O	11.34	4.05E-02	2.01	0.0201
(Li6) ₂ O	0.85	3.04E-03		
(Li7) ₂ O	10.49	3.75E-02		
MgO	4.67	1.67E-02	3.58	0.0047
MnO ₂	3.18	1.13E-02	5.03	0.0023
Na ₂ O	14.01	5.00E-02	2.27	0.0220
NaNO ₃	15.84	5.66E-02	2.26	0.0250
SiO ₂	126.00	4.50E-01	2.32	0.1939
U ₃ O ₈	2.33	8.31E-03	8.39	0.0010
PuO ₂	0.01	3.09E-05	11.46	2.692E-06
Sum	280.08	1.00		0.49
		Density g/cc		2.054

Melter Feed 1/100 Water	Lb/hr	Wt. Frac.	Density	Wt. Frac/ Density
Water	3.40	1.36E-02	1.00	0.0136
Al ₂ O ₃	10.28	4.12E-02	3.50	0.0118
B ₂ O ₃	12.96	5.20E-02	1.84	0.0282
Ca(NO ₃) ₂	2.61	1.05E-02	2.36	0.0044
Cu(COOH) ₂	1.55	6.21E-03	1.83	0.0034
Fe ₂ O ₃	23.73	9.51E-02	5.24	0.0182
H ₃ BO ₃	8.08	3.24E-02	1.44	0.0225
KCOOH	9.48	3.80E-02	1.91	0.0199
Li ₂ O	11.34	4.55E-02	2.01	0.0226
(Li6) ₂ O	0.85	3.41E-03		
(Li7) ₂ O	10.49	4.21E-02		
MgO	4.67	1.87E-02	3.58	0.0052
MnO ₂	3.18	1.27E-02	5.03	0.0025
Na ₂ O	14.01	5.62E-02	2.27	0.0247
NaNO ₃	15.84	6.35E-02	2.26	0.0281
SiO ₂	126.00	5.05E-01	2.32	0.2177
U ₃ O ₈	2.33	9.33E-03	8.39	0.0011
PuO ₂	0.01	3.46E-05	11.46	3.023E-06
Sum	249.45	1.00		0.42
		Density g/cc		2.359

Attachment 1: SRT-CMA-930039

Melter Feed No Boron or Lithium		Wt. Frac/ Density		Wt. Frac/ Density	
	Lb/hr	Wt. Frac.	Density		Density
Water	340.03	6.14E-01	1.00	0.6141	
Al ₂ O ₃	10.28	1.86E-02	3.50	0.0053	
B ₂ O ₃	0.00	0.00E+00	1.84	0.0000	
Ca(NO ₃) ₂	2.61	4.71E-03	2.36	0.0020	
Cu(COOH) ₂	1.55	2.80E-03	1.83	0.0015	
Fe ₂ O ₃	23.73	4.29E-02	5.24	0.0082	
H ₃ BO ₃	0.00	0.00E+00	1.44	0.0000	
KCOOH	9.48	1.71E-02	1.91	0.0090	
Li ₂ O	0.00	0.00E+00	2.01	0.0000	
(Li ₆) ₂ O	0.00	0.00E+00			
(Li ₇) ₂ O	0.00	0.00E+00			
MgO	4.67	8.44E-03	3.58	0.0024	
MnO ₂	3.18	5.73E-03	5.03	0.0011	
Na ₂ O	14.01	2.53E-02	2.27	0.0111	
NaNO ₃	15.84	2.86E-02	2.26	0.0127	
SiO ₂	126.00	2.28E-01	2.32	0.0981	
U ₃ O ₈	2.33	4.20E-03	8.39	0.0005	
PuO ₂	0.01	1.56E-05	11.46	1.362E-06	
Sum	553.70	1.00		0.77	
		Density	g/cc		1.306

Melter Feed No B or Li 1/100 water		Wt. Frac/ Density		Wt. Frac/ Density	
	Lb/hr	Wt. Frac.	Density		Density
Water	3.40	1.57E-02	1.00	0.0157	
Al ₂ O ₃	10.28	4.74E-02	3.50	0.0135	
B ₂ O ₃	0.00	0.00E+00	1.84	0.0000	
Ca(NO ₃) ₂	2.61	1.20E-02	2.36	0.0051	
Cu(COOH) ₂	1.55	7.13E-03	1.83	0.0039	
Fe ₂ O ₃	23.73	1.09E-01	5.24	0.0209	
H ₃ BO ₃	0.00	0.00E+00	1.44	0.0000	
KCOOH	9.48	4.37E-02	1.91	0.0229	
Li ₂ O	0.00	0.00E+00	2.01	0.0000	
(Li ₆) ₂ O	0.00	0.00E+00			
(Li ₇) ₂ O	0.00	0.00E+00			
MgO	4.67	2.15E-02	3.58	0.0060	
MnO ₂	3.18	1.46E-02	5.03	0.0029	
Na ₂ O	14.01	6.45E-02	2.27	0.0284	
NaNO ₃	15.84	7.30E-02	2.26	0.0323	
SiO ₂	126.00	5.80E-01	2.32	0.2502	
U ₃ O ₈	2.33	1.07E-02	8.39	0.0013	
PuO ₂	0.01	3.98E-05	11.46	3.474E-06	
Sum	217.08	1.00		0.40	
		Density	g/cc		2.481

Glass		Wt Frac/ Density		Wt Frac/ Density	
	Lb/hr	Wt. Frac.	Density		Density
Water	0.00	0.00E+00	1.00	0.0000	
Al ₂ O ₃	10.17	4.60E-02	3.50	0.0131	
B ₂ O ₃	17.25	7.80E-02	1.84	0.0423	
Fe ₂ O ₃	21.14	9.55E-02	5.24	0.0182	
FeO	2.11	9.55E-03	5.70	0.0017	
K ₂ O	5.38	2.43E-02	2.32	0.0105	
Li ₂ O	11.23	5.08E-02	2.01	0.0252	
(Li ₆) ₂ O	0.84	3.81E-03			
(Li ₇) ₂ O	10.39	4.69E-02			
MgO	4.63	2.09E-02	3.58	0.0058	
MnO	2.77	1.25E-02	5.45	0.0023	
Na ₂ O	19.57	8.84E-02	2.27	0.0390	
SiO ₂	124.70	5.64E-01	2.32	0.2429	
U ₃ O ₈	2.31	1.04E-02	8.39	0.0012	
PuO ₂	0.01	3.87E-05	11.46	3.374E-06	
Sum	221.27	1.00		0.40	
		Density	g/cc		2.486

Glass		Wt Frac/ Density		Wt Frac/ Density	
	Lb/hr	Wt. Frac.	Density		Density
Water	0.00	0.00E+00	1.00	0.0000	
Al ₂ O ₃	10.17	5.28E-02	3.50	0.0151	
B ₂ O ₃	0.00	0.00E+00	1.84	0.0000	
Fe ₂ O ₃	21.14	1.10E-01	5.24	0.0209	
FeO	2.11	1.10E-02	5.70	0.0019	
K ₂ O	5.38	2.79E-02	2.32	0.0120	
Li ₂ O	0.00	0.00E+00	2.01	0.0000	
(Li ₆) ₂ O	0.00	0.00E+00			
(Li ₇) ₂ O	0.00	0.00E+00			
MgO	4.63	2.40E-02	3.58	0.0067	
MnO	2.77	1.44E-02	5.45	0.0026	
Na ₂ O	19.57	1.02E-01	2.27	0.0447	
SiO ₂	124.70	6.47E-01	2.32	0.2788	
U ₃ O ₈	2.31	1.20E-02	8.39	0.0014	
PuO ₂	0.01	4.44E-05	11.46	3.873E-06	
Sum	192.79	1.00		0.38	
		Density	g/cc		2.603

Attachment 1 SRT-CMA-930039

END

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