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Kennecott Uranium Company's Comments on DRAFT REGULATORY GUIDE DG-8051 – BIOASSAY AT URANIUM MILLS (Federal Register Volume 77, Number 49 / Tuesday, March 13, 2012 / Notices)

Yours faithfully,

Oscar A Paulson
Oscar Paulson
Facility Supervisor

#3

3/13/2012
77 FR 14837

SUNSI REVIEW COMPLETE
TEMPLATE = ADM -013

E-RIDS= ADM-03
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M. Case (MJC)

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May 8, 2012

Ms. Cindy Bladey, Chief
Rules, Announcements, and Directives Branch (RADB)
Office of Administration
Mail Stop: TWB-05-B01M
U.S. Nuclear Regulatory Commission,
Washington, DC 20555-0001

Subject: Kennecott Uranium Company Comments on DRAFT REGULATORY GUIDE DG-8051- BIOSSAY AT URANIUM MILLS (Federal Register Volume 77, Number 49 /Tuesday, March 13, 2012 /Notices) Docket ID NRC-2012-0057

Dear Ms. Bladey:

Kennecott Uranium Company is a uranium recovery licensee and the owner of the Sweetwater Uranium Project, a conventional uranium mill located in Sweetwater County, Wyoming. Kennecott Uranium Company has reviewed ***DRAFT REGULATORY GUIDE DG-8051- BIOASSAY AT URANIUM MILLS*** and has the following comments:

Workers Requiring Bioassay

A clear and concise definition of workers requiring bioassay should be incorporated in the glossary. Licensees should have no doubt as to who must be bioassayed at their sites. It is clear, based on how the draft guidance is written for example, that any worker that must don a respirator must be bioassayed following completion of work; however the other requirements are not that clear. The document states that "*...workers who handle and work with uranium substances, or are sufficiently close to the process so that intake is possible*" must be bioassayed. The document does not discuss worker exposure time. Workers spending a very limited time with uranium substances are at very low risk of intake and thus bioassaying of them should not be required. An example would be a worker who spends an hour or less per week in proximity to uranium processing. In addition, infrequent visitors to a uranium recovery operation may be, for a very short time (minutes), *sufficiently close to the process so that intake is possible*, if they are touring the grinding, leaching or counter-current decantation areas of a conventional mill. There should be no requirement that such infrequent visitors be bioassayed.

Post Operational and Termination Bioassays

The guide should recognize that it is not always possible to collect a final bioassay sample from a worker upon termination of work involving exposure to uranium. In some cases, workers have failed to report to work and subsequently have been impossible to locate or contact making it impossible to obtain a final/post-operational bioassay. This is especially true in the case of contract workers. Contract employees work for the contractor and the licensee may be unable to obtain a post-operational bioassay of a contract worker who fails to report for work and cannot be found. This contingency should be recognized in the document.

Bioassays following Use of Respiratory Protection Devices

The draft guide states:

Bioassay specimens should be collected and evaluated after a respiratory protection device is used to reduce intake of radionuclides.

In some cases respiratory protection must be donned for a very short period of time (often less than fifteen (15) minutes) to take a reading or perform some minor tasks in an area designated for respirator use. This very short duration exposure with its low associated risk, does not warrant a special bioassay. This is especially true for facilities on standby where very brief respirator use is required on an infrequent basis. In any event if a worker were required to don a respirator even infrequently and for brief periods of time that worker would still be subject to bioassaying on a monthly basis. The Regulatory Guide should be revised to recognize such infrequent short periods of respirator use.

Additional Information Regarding Uranium and Bioassays Related to Uranium Recovery

Uranium recovery facilities in Wyoming dry their uranium product using either a rotary vacuum dryer or a multiple hearth roaster. The current version of Regulatory Guide 8.22 defines two (2) types of product produced at uranium recovery facilities as follows:

> *a. Low-fired yellowcake is defined as yellowcake dried at temperatures less than 400° C.*
> *b. High-fired (calcined) yellowcake is defined as yellowcake dried at temperatures of 400° C or more.*

REGULATORY GUIDE 8.22 BIOASSAY AT URANIUM MILLS

Product produced with rotary vacuum dryers is considered low-fired yellowcake and product produced from multiple hearth roasters is considered high-fired (calcined) yellowcake.

Attached in Appendix 1 is a paper entitled ***SOLUBILITY OF RADIONUCLIDES IN SIMULATED LUNG FLUID.*** This paper addresses the solubility of low-fired yellowcake from a rotary vacuum dryer. A review of this paper by Steve Brown of SENES is included in Appendix 1 as well. He concludes in his review that:

> *All ISR samples appear to exhibit solubility characteristics that meet the definition of absorption Type F as defined in ICRP 71 (i.e. the most soluble category) - see paper Figure 5. ICRP 71 considers Type F as "generally equivalent" to solubility Class D from the older ICRP 26/30. ICRP26/30 is the basis of 10 CFR 20 dosimetry (Uranium Derived Air Concentrations {DACs} and Annual Limits on Intake {ALIs} in 10 CFR 20, Appendix B, Table 1 .*
>
> and that
>
> *These products meet the definition of "low fired yellowcake" as used in NRC Regulatory Guide 8.22.*
>
> The Regulatory Guide should be revised to reflect this information in our Appendix 1.

Kennecott Uranium Company appreciates the opportunity to comment on this draft regulatory guide. If you have any questions please do not hesitate to contact me.

Sincerely yours,

Oscar A Paulson
Oscar Paulson
Facility Supervisor

cc: Katie Sweeney – National Mining Association (NMA)

Appendix 1

SOLUBILITY OF RADIONUCLIDES IN SIMULATED LUNG FLUID

*G. Tairova[1], M. Boucher[1], K. Toews[2], M. Ioffe[1], J. Takala[2], M. Murchie[1] and Engin Ozberk[2]

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ABSTRACT

The objective of this work was to assign FMS (fast, moderate and slow) absorption types to site-specific radionuclides found at various Cameco facilities. Kinetics studies of dissolution of various uranium-bearing samples in simulated lung fluid were carried out. Dissolution parameters were determined and the samples were assigned to the FMS absorption types. A software program was developed for the determination of dissolution parameters. The assignments were based on criteria established in the International Commission on Radiological Protection (ICRP) Publication 71. The assignments were used to determine the internal radiation dose for workers.

MET SOC
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Uranium 2010 - "The future is U"
Proceedings of the 3rd International Conference on Uranium
40th Annual Hydrometallurgy Meeting
Saskatoon, Saskatchewan, Canada
Edited by E.K. Lam, J.W. Rowson, E. Özberk

INTRODUCTION

One of the regulations of the Canadian Nuclear Safety Commission (CNSC) requires that an internal dose component be assigned for personnel working at Cameco sites in Canada. This approach was expanded to Cameco's US operations. Simulated lung fluid (SLF) solubility experiments are carried out for the various types of uranium-bearing materials present at the sites in order to produce site-specific solubility data. Without the site-specific solubility data, more conservative default values for the various uranium-bearing materials would have to be used.

A critical review of techniques for studies of solubility of uranium compounds in simulated lung fluid was described in a previous report [1]. The method used in this study was selected after a thorough analysis of the advantages and disadvantages of the existing techniques described in the literature.

For the purpose of calculating the radiation doses from radionuclides deposited in the lung by inhalation, uranium compounds were first divided into three solubility classes based on retention times in the lungs, according to ICRP Publication 30 (1979) [2]:

- Class D compounds, less than 10 days
- Class W, 10-100 days
- Class Y, greater than 100 days.

Further conditions were added, and, in the ICRP Publication 71 (1995) [3], the clearance of contaminating material from the lungs was treated as a result of two processes, namely, particle transport to the gastrointestinal tract and absorption into body fluids. For simplicity, lung retention (the amount of material that was not dissolved) for *in vitro* dissolution experiments was considered in the absence of particle transport simulating respiratory tract absorption. Subsequently, three new absorption types F, M or S (fast, moderate and slow) were created. Assignments to types F, M and S were carried out using the following criteria. If the retention was below 13% in 30 days, the material was assigned to type F. If the retention was in the range 13-87% between 30 and 180 days the material was assigned to type M. If the retention was above 87% in 180 days, or later, the material was assigned to type S. These categories correspond approximately to classes D, W and Y described in ICRP Publication 30 (1979).

During 2000-2005, an in-house method was developed to measure solubility parameters for uranium concentrate samples and the uranium site-specific, process-stream samples and aerosol samples collected at the Port Hope conversion facility (PHCF) and Blind River refinery (BRR). An additional series of SLF solubility experiments were carried out for uranium and other radionuclides-bearing materials present at the Key Lake (KLO), McArthur River (MCA) and Rabbit Lake (RLO) uranium facilities in order to produce site-specific solubility data. The data obtained were used to assign to the solubility types according to the International Committee on Radiological Protection (ICRP) Publication 71 requirements and to assign an internal dose component for personnel working at these stations. In 2006, experiments were carried out to study the solubility of 30 additional samples collected at the PHCF, Zircatec Precision Industries (ZPI; since renamed Cameco Fuel Manufacturing), Cameco technology development (CTD; since renamed Innovation & Technology Development – Research Centre) and Blind River refinery (BRR).

In 2009, a request was received from Cameco's safety, health, environment and quality (SHEQ) department to carry out similar lung fluid dissolution studies for uranium concentrate samples collected at the Crow Butte (CBO), Smith Ranch-Highland (SRH) and Rabbit Lake (RLO) operations.

The objective of this work was to assign the site-specific uranium compounds to absorption types according to the International Committee on Radiological Protection (ICRP) Publication 71 criteria.

For the sake of brevity, only a small part of the SLF studies of radionuclides are presented in this publication.

EXPERIMENTAL

The experimental work included:

1) obtaining samples from Cameco sites
2) sieving and separation of samples to obtain the inhalable particle size <20 µm
3) analysis of samples for total uranium (carried out at BRR analytical services)
4) studies of the dissolution kinetics in SLF
5) calculation of kinetics parameters and assignment of samples to FMS absorption types

The Samples Used in the Experiments

Cameco Port Hope conversion facility (PHCF) process stream samples and Cameco concentrate samples obtained from mine sites were used in experiments.

The PHCF process stream samples included:

(i) Uranium tetrafluoride
(ii) Uranium dioxide, UO_2 (ceramic)
(iii) Uranium trioxide, UO_3
(iv) Ammonium diuranate, ADU

The results of total uranium analyses in PHCF process stream samples are presented in Table 1.

Table 1 - Cameco Port Hope conversion facility process stream samples used in the experiments

Compound	Formula	Sample No. Comments	Total Uranium in 50 mg Sample (mg)
Uranium tetrafluoride calcined	UF_4	Calciner C #000311	38
Uranium dioxide ceramic	UO_2	Lot#200DL154	44
Uranium trioxide	UO_3	Autosampler Lot#275492	42
Ammonium diuranate	$(NH_4)_2U_2O_7{\cdot}XH_2O$		38

Description of the Cameco concentrate samples, used in the experiments, including the results of total uranium analyses, are presented in Table 2.

Table 2 - Cameco concentrate samples used in the experiments

Concentrate Sample / Location/ Experiment year	Total Uranium in 50 mg Sample (mg)
Crow 2009, Crow Butte Operation, 2009	38.4
KLO 2005, Key Lake Operation, Packaging Area, 2005	39.6
RLO 2009, Rabbit Lake Operation, 2009	34.0
SRH 2009, Smith Ranch Operation, 2009	39.5

Reagents

The reagents used are presented in Table 3.

Table 3 - The reagents used in the solubility experiments

Chemical/Description	Supplier
Nitric acid (double-distilled), HNO_3	Seastar Chemicals Inc.
Hydrochloric acid, concentrated (35-37% assay), HCl	Fisher Scientific
Magnesium chloride hexahydrate, $MgCl_2 \cdot 6H_2O$	Fisher Scientific
Sodium chloride, NaCl	Fisher Scientific
Potassium chloride, KCl	Fisher Scientific
Sodium hydrogen phosphate heptahydrate, $Na_2HPO_4 \cdot 7H_2O$	Fisher Scientific
Sodium sulphate, Na_2SO_4	Fisher Scientific
Calcium chloride dihydrate, $CaCl_2 \cdot 2H_2O$	Fisher Scientific
Sodium acetate trihydrate, $NaH_3C_2O_2 \cdot 3H_2O$	Fisher Scientific
Sodium hydrogen carbonate, $NaHCO_3$	Fisher Scientific
Sodium citrate dihydrate, $Na_3H_3C_6O_7 \cdot 2H_2O$	Fisher Scientific
Deionized water	Laboratory Supply

Simulated Lung Fluid

The composition of SLF solutions used in the experiments is illustrated in Table 4. This formulation was suggested by Moss in 1976 [4]. The selection of the lung fluid was based on the analysis of the information available in the literature [5].

The electrolyte compositions of actual human lung fluid and the selected simulant were almost identical. Moss suggested replacing the protein components of actual lung fluid by an ionically equivalent amount of citrate in the simulant. Actual lung fluid proteins are poorly characterized and generally not available in large quantities, and substitute proteins hinder filtration and promote bacterial growth in solutions. Phospholipids, also known to be present in trace amounts in actual lung fluid, were not included in the simulant for the same reasons. The previous studies did not indicate any effect of organic ingredients on the dissolution rate of uranium yellow cake samples.

Simulated lung fluid was prepared in ten-litre batches by addition of ten-times the amounts of the components listed in Table 3 into 1 L of distilled water in a 10-L polyethylene container. Each reagent dissolved by continuous stirring and only then the next was added. The aqueous solution obtained was diluted with distilled water up to the 10-L mark. The solution was filtered and saturated with air/5% CO_2. Finally, the pH of the solution was adjusted to 7.3-7.4 by addition of small volumes of 1N hydrochloric acid, HCl. The pH of the solution was checked every day and maintained at 7.4 by addition of HCl as needed.

Table 4 - Composition of the additives in 1 L of aqueous SLF solution (O.R. Moss, 1976)

Name	Formula	Mass (g)
Magnesium chloride hexahydrate	$MgCl_2 \cdot 6H_2O$	0.2033
Sodium chloride	NaCl	6.0193
Potassium chloride	KCl	0.2982
Sodium hydrogen phosphate heptahydrate	$Na_2HPO_4 \cdot 7H_2O$	0.2680
Sodium sulphate	Na_2SO_4	0.0710
Calcium chloride dehydrate	$CaCl_2 \cdot 2H_2O$	0.3676
Sodium acetate trihydrate	$NaH_3C_2O_2 \cdot 3H_2O$	0.9526
Sodium hydrogen carbonate	$NaHCO_3$	2.6043
Sodium citrate dihydrate	$Na_3H_3C_6O_7 \cdot 2H_2O$	0.0970

Instrumental

The extracted uranium concentration in simulated lung fluid was determined using inductively coupled plasma-mass spectrometry (ICP-MS).

An ATM sonic sifter was employed to obtain samples with particle sizes less than 20 μm.

Experimental Circuit

Experiments were carried out in a circuit previously employed for the studies of the kinetics of uranium dissolution in SLF (Figure 1). The experimental circuit was housed in a wooden box to prevent any contamination by spurious uranium contaminant that might have been present in the laboratory environment. The box was insulated and the temperature was maintained at 37°C using a thermostatically controlled heater-fan. Access to the different parts of the apparatus was through wooden and Plexiglas double doors. The SLF reservoir and the sample collection bottles were sealed with plastic caps. All other components in contact with the solution were plastic. Other materials, such as metal and glass, were avoided because of potential corrosion, contamination and adsorption problems.

The 50-mg concentrate samples (particle size <20 μm) were placed between two glass fiber filters in 47-mm polypropylene filter holders used as extraction cells. The outlet sides of the extraction cells were connected by Teflon tubing to 500-mL polypropylene sample collection bottles.

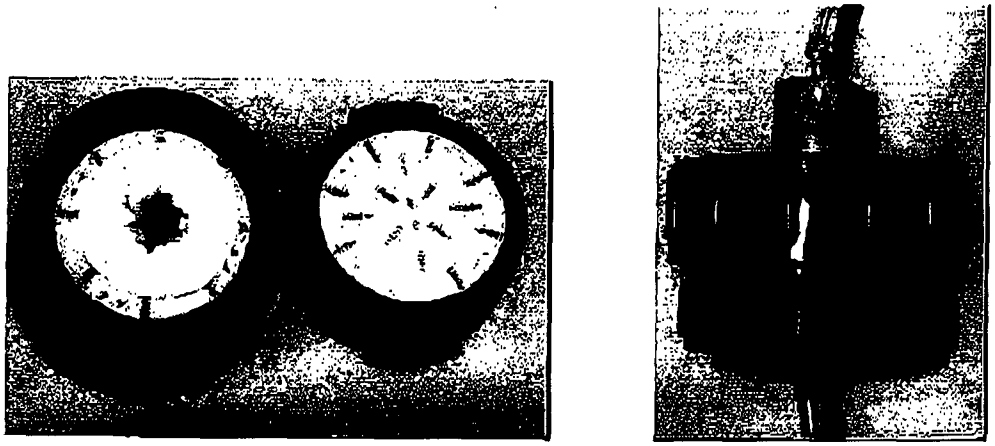

Figure 1 - A filter holder used as an extraction cell in experiments

The required quantities of the original samples were estimated based on the concentrations of uranium in the sample and the volumes of SLF used for the dissolution. The required volumes of SLF were estimated based on the flow rates, applied over 100 days of extraction with consideration of the detection limits for the uranium and with the assumption that the uranium contained in the samples had a low solubility in SLF. The SLF was passed through the sample at predetermined flow rates.

The SLF was supplied to the extraction cells from two 8-L vessels using a forty-channel peristaltic pump. A third 10-L vessel was used to continuously supply fresh SLF using a two-channel peristaltic pump. The solution was saturated with a 5% mixture of CO_2 in air to simulate (to a degree) the oxidizing conditions in the lung.

The starting flow rates were significantly higher to prevent oversaturation (and possible re-precipitation) of uranium at the beginning of the experiments, when the concentrations of the uranium was high. The flow rates were gradually decreased from 1 mL/min to 0.05 mL/min with the decrease of uranium concentrations in the sample in order to obtain the concentrations of dissolved elements in SLF, which would be sufficient for measurements. The volumes of simulated lung fluid samples with dissolved uranium gradually increased over the course of extraction.

The samples of SLF, after passing through the extraction cells, were collected, acidified with double-distilled nitric acid (to prevent precipitation of elements) and submitted to the PHCF analytical services for determination of the uranium concentration by ICP-MS.

An uninterrupted power supply line was installed to the circuit to ensure continuous operation.

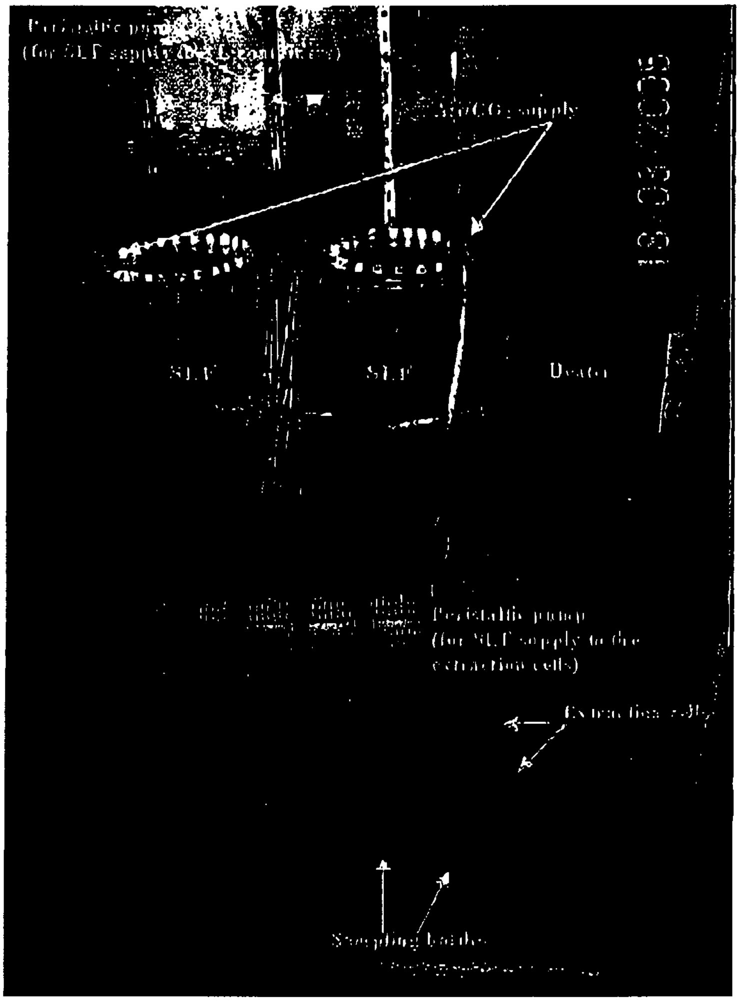

Figure 2 - The circuit for the studies of solubility of radionuclides in SLF

RESULTS AND DISCUSSIONS

Dissolution Kinetics and Dissolution Parameters

Dissolution half-times for a composite aerosol sample were calculated using the two-exponential model and the following Equation (1), described previously, where percentage of undissolved uranium is expressed as [6]:

$$M/M_0=f_1\exp(-0.693t/T_1) + f_2\exp(-0.693t/T_2) \quad (1)$$

where
M - mass of undissolved uranium at time t
M_0 – initial mass of uranium
t – elapsed time
f_1 - fraction of total U with corresponding dissolution half-time T_1
f_2 - fraction of total U with corresponding dissolution half-time T_2
$f_1 + f_2 = 100\%$

A software program, developed at the research centre, was employed for the determination of dissolution parameters.

The program uses a Simplex algorithm to minimize the functional F calculated as the sum of squared differences between the experimental and theoretical values characterizing the uranium content (fraction of dissolved or undissolved uranium). To ensure that the global minimum is found, the minimization calculations are repeated several hundred times using Monte Carlo technique to vary the initial conditions. It is assumed that the global minimum is the set of optimized parameters corresponding to the smallest functional found in the series of these 200 to 500 minimizations. The procedure was tested several times and it was confirmed that 200 repetitions are sufficient to locate the global minimum.

After the global minimum is found, the Hessian matrix consisting of the second derivatives of F with respect to all the free parameters involved in the model is calculated. The calculations are performed using analytical expressions for the second derivatives (as opposed to numerical differentiation in most of the programs such as MathCAD or Mathematica). It was checked for each derivative that the analytical expressions were consistent with the numerically differentiated. The standard deviations of parameters were calculated from the inverse Hessian matrix using the MSE (Mean Square Error) calculated as the square root of F/(N-v), where N is the number of data points and v is the number of parameters used in the model. (N - v is the number of degrees of freedom).

The program automatically tests several kinetics models applied to the same set of experimental data. For each model, the minimization is repeated 200 to 500 times (this number is specified by the user). Then, the program selects the model that has the smallest MSE, which is considered to be the best approximation to the experimental data. Some models with a large number of adjustable parameters that show lower F are not the best description for the experiment because they have higher MSE (lower denominator).

Absolute values of standard errors for parameters are given in parenthesis. In some cases, when calculated values of uncertainties are high, an additional sampling during first day and (or) after last day of extraction is required in order to obtain lower uncertainties.

The value of the Mean Square Deviation (MSD) characterizes an average error for the curve fitting, i.e., the difference between the experimental values of uranium extraction, $(1 - M/M_0)*100\%$, and the theoretical value.
The use of equation (7) allows one to determine the rapid (f_r) and slow (f_s) fractions and their corresponding half-times, T_r and T_s. The equation $S = \ln2/T$, allows the half-time T to be transformed into dissolution rate constant S (or S_r and S_s, respectively). Parameters f_r, S_r and f_s, S_s can be used in a LUDEP program to calculate the dose intake according to ICRP Publication 66.

A program was developed at Cameco research centre for calculation of dissolution parameters using non-linear regression analysis.

Dissolution kinetics for hypothetic two-component sample containing fraction f_1 with dissolution half-time T_1 and fraction f_2 with dissolution half-time T_2 is illustrated in Figure 3.

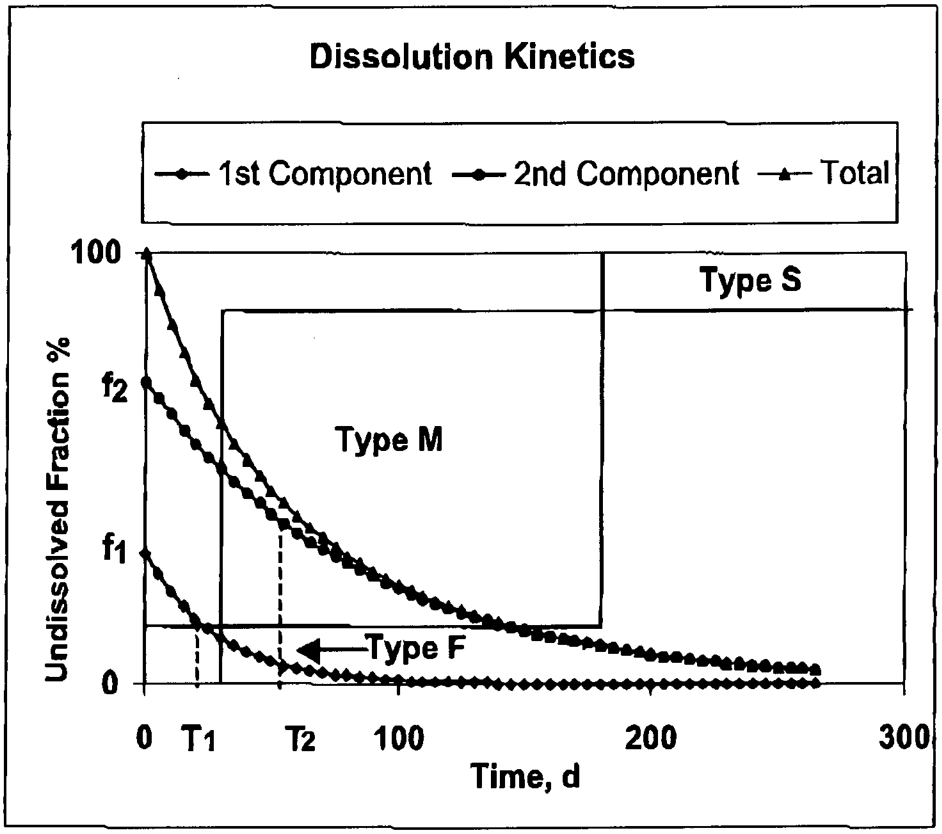

Figure 3 - Dissolution kinetics for two-component sample containing fraction f_1 with dissolution half-time T_1 and fraction f_2 with dissolution half-time T_2

In the ICRP 71 respiratory tract model (1995) [7], the clearance of contaminating material from the lungs is treated as a result from the sum of two processes, namely, the particle transport to the gastro-intestinal tract and the absorption into body fluids. For simplicity, for *in vitro* dissolution experiments, lung retention (the amount of material that has not dissolved) is considered in the absence of particle transport simulating respiratory tract absorption. The assignment of the compounds to absorption Types F, M or S (fast, moderate and slow), is carried out using the criteria illustrated in Figure 3. Retention below 13% in 30 days, or earlier, would assign the material to Type F (retention above 13% after 30 days would exclude Type F). Retention above 87% in 180 days, or later, would assign the material to Type S (retention below 87% before 180 days would exclude Type S). Retention in the range 13-87% between 30 and 180 days would positively assign the material to Type M.

Absolute values of standard errors for parameters are given in parenthesis. In some cases, when calculated values of uncertainties are high, an additional sampling during first day and (or) after last day of extraction is required in order to obtain lower uncertainties.

The value of the Mean Square Deviation (MSD) characterizes an average error for the curve fitting, i.e., the difference between the experimental values of uranium extraction, $(1 - M/M_0)*100\%$, and theoretical value.

Dissolution Kinetics and Dissolution Parameters of PHCF Samples

The dissolution of uranium process stream powder samples was carried out over 103 days. The dissolution kinetics of Cameco process stream samples are shown in Figure 4.

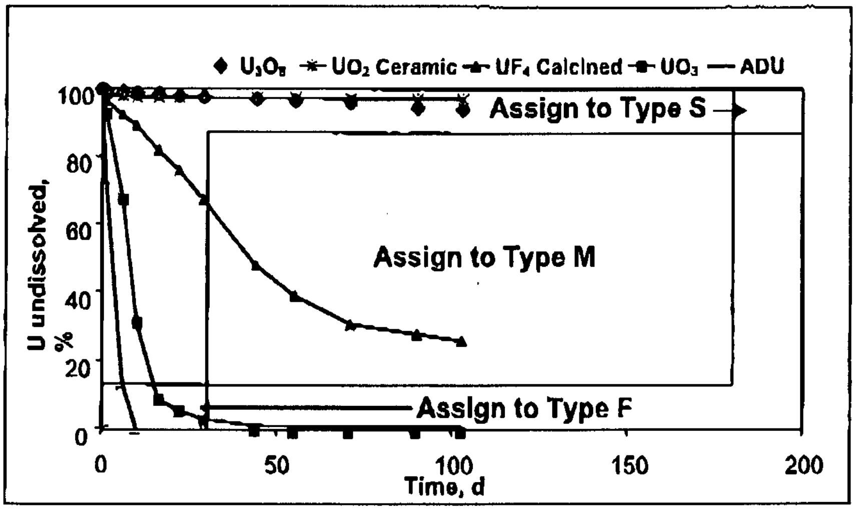

Figure 4 - Dissolution kinetics of Cameco process stream samples

The dissolution parameters for PHCF process stream samples, calculated using one of the equations described above, are given in Table 5.

Based on the ICRP Publication 71 criteria, the PHCF process stream samples could be assigned to the absorption types as shown in Figure 4 and Table 5.

Table 5 - Assignment of PHCF process stream samples to FMS Type

Compound	F_1 (%)	T_1 (d)	F_2 (%)	T_2 (d)	MSD (%)	Absorption Type
ADU	100	2(0.1)	-	-	2.4	F
UO_3	100	6.5(0.5)	-	-	5.6	F
UF_4 calcined	97(8)	44(6)	3(8)	∞	3.6	M
UO_2 ceramic	2.1(0.1)	0.4(0.1)	97.9(0.1)	∞	0.1	S
U_3O_8	0.56(0.1)	0.3(0.4)	99.4(0.1)	∞	0.1	S

Assignment of concentrate samples to absorption types is presented in Table 6.

Table 6 - Assignment of concentrate samples to absorption types

Concentrate Sample / Location/ Experiment year	F_1 (%)	T_1 (d)	F_2 (%)	T_2 (d)	MSD (%)	Absorption Type
Crow 2009, Crow Butte Operation, 2009	78.5(8.6)	0.2(0.1)	21.5(8.6)	2.2(0.3)	0.8	F
KLO 2005, Key Lake Operation Concentrate, Packaging Area, 2005	5.3(0.6)	0.11 (0.06)	94.7(0.6)	∞ (∞)	0.3	S
RLO 2009, Rabbit Lake Operation Concentrate, 2009	93.5(4.2)	7.0(1.9)	6.5(4.2)	∞ (∞)	2.3	F(M)
SRH 2009, Smith Ranch Operation Concentrate, 2009	88.8(4.6)	1.0(0.2)	11.2(4.6)	61.3 (14.6)	2.4	F

Dissolution Kinetics and Dissolution Parameters of Cameco Concentrate Samples

The dissolution kinetics of Cameco concentrate samples are shown in Figure 5.

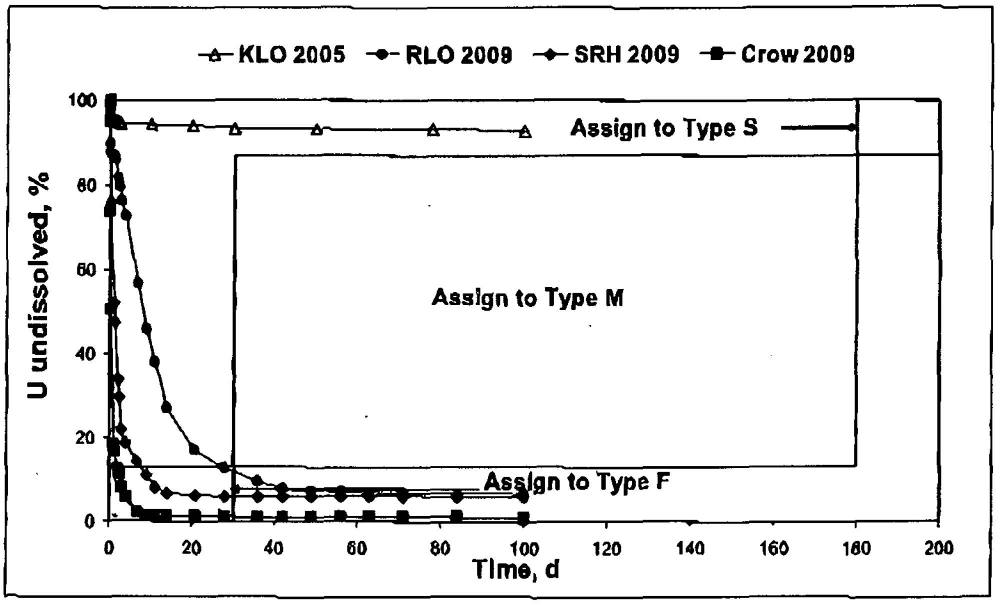

Figure 5 - Dissolution kinetics of concentrate samples

CONCLUSIONS

Dissolution studies of Cameco's site-specific radionuclides were carried out. A new software for calculations of solubility parameters and assignment to FMS absorption types, based on the ICRP Publication 71 requirements, was developed and applied.

ACKNOWLEDGEMENTS

Site personnel at PHCF, CBO, SRH and RLO collected the samples. The authors of this report would like to acknowledge the contributions of all team members. Support of this work by BRR, PHCF and research centre analytical services is gratefully acknowledged.

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3 "Age-Dependent Doses to Members of the Public from Intake of Radionuclides: Part 4. Inhalation Dose Coefficients", ICRP Publication 71", A report of a Task Group Committee 2 of the International Commission on Radiological Protection, Pergamon, 1995, 393-405.

4 O.R.Moss, "Simulants of Lung Interstitial Fluid", Health Physics, Vol. 36, 1976, 447-448.

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6 T.T.Mercer, "On the Role of Particle size in the Dissolution of Lung Fluid", Health Physics, Pergamon Press, Vol. 13, 1967, 1211-1221.

7 "Age-dependent Doses to Members of the Public from Intake of Radionuclides, Part 4: Inhalation Dose Coefficients", ICRP Publication 71. A report of a Task Group Committee 2 of the International Commission on Radiological Protection, Pergamon, 1995, 393-405.

TO: Oscar Paulson **9 September 2010**
FROM: Steve Brown
RE: Review of Paper "Solubility of Radionuclides in Simulated Lung Fluid", G. Tairova et al, 2010

Background:

Over the last year or two, multiple ISR source material license applicants have received from NRC "Requests for Additional Information" (RAIs) and or identified "Open Health Physics Issues" requesting justification for considering yellowcake products as "soluble" (absorption Class D, USNRC 10 CFR 20, *Standards for Protection Against Radiation*) and/or requesting justification that these products are "low fired yellowcake" per Regulatory Guide 8.22, *Bioassay at Uranium Mills*. If this is the case, per Regulatory Guide 8.22, monthly urinalysis is the proper routine bioassay technique; in vivo lung counting is not appropriate nor required on regular basis, etc. Find below some implications (my personal opinions) from the data presented in this paper.

1. All ISR samples appear to exhibit solubility characteristics that meet the definition of absorption Type F as defined in ICRP 71 (i.e. the most soluble category) - see paper Figure 5. ICRP 71 considers Type F as "generally equivalent" to solubility Class D from the older ICRP 26/30. ICRP26/30 is the basis of 10 CFR 20 dosimetry (Uranium Derived Air Concentrations {DACs} and Annual Limits on Intake {ALIs} in 10 CFR 20, Appendix B, Table 1).

2. Smith Ranch samples appear to exhibit smaller components typical of what was defined by ICRP 26/30 as Class W (moderate solubility). Crow Butte samples are exclusively Class D.

3. The Smith Ranch samples suggest a relatively small % of Class W components (the "slower" component = F_2 @11.2 %, T_2 @ 61 days – See Table 6 of paper), nonetheless, anywhere from 10 -100 days is still "Class W" (not " Class Y" {"insoluble"}, as I understand had been suggested recently to one or more applicants).

4. Accordingly, the vast majority of any intake would be translocated or systemically absorbed from the lung in time periods of less than a month or two.

5. These products meet the definition of "low fired yellowcake" as used in NRC Regulatory Guide 8.22. Accordingly, Table 1 of Regulatory Guide 8.22 can be used to establish action levels and required actions. Additionally Figure 2 of Regulatory Guide 8.22 can be used to estimate uranium concentration in urine following a single acute intake since these ISR products are clearly "low-fired yellowcake".

6. Accordingly, there would appear to be little, if any, dose impact from these products to the pulmonary system from long term retention (except perhaps in case of an acute intake) and any residual pulmonary retention from low level chronic exposure beyond a month or two would probably be to small to measure by the current state of art of in vivo lung counting techniques. That is, routine urinalysis, at frequencies $\leq$ 30 days, is the

appropriate bioassay method to assess intake for low-fired yellowcake products.

Conclusion:

At the risk of generalizing, one could conclude that the above should be true for any modern ISR yellowcake products precipitated with peroxide and dried at the low temperatures of modern vacuum dryers (i.e., producing UO_4 / UO_3 and associated hydrates).

References

ICRP 26 International Commission on Radiological Protection. *Recommendations of the International Commission on Radiological Protection,* 1977.

ICRP 30 International Commission on Radiological Protection. *Limits for Intakes of Radionuclides by Workers*, 1979.

ICRP 71 International Commission on Radiological Protection. *Age-Dependent Doses to Members of the Public from Intake of Radionuclides Part 4: Inhalation Dose Coefficients*, 1995