Appendix | Revised April 2015

Smith Ranch Analysis of Day, Week, and Year Classification Smith Ranch Radionuclide Mixture Assessment



1.0 SUMMARY

Based on discussions with the NRC and internal discussions within Cameco, an analysis of the solubility characteristics of Smith Ranch Highland yellowcake was performed and the material was classified according to the days, weeks, and years (D/W/Y) classification scheme of 10CFR20. This report provides a summary and dosimetry interpretation of the experiment performed by the Cameco Technology and Innovation Research Centre (CTI-RC)[1] to determine these solubility characteristics. This submission supersedes any previous reports submitted on this topic as it incorporates the most recent comments from the NRC and internal discussions.

The experimental results were interpreted in the context of the ICRP 30[2] solubility classifications of days, weeks, and years (D/W/Y) used by 10 CFR20. In addition, the annual limit on intake (ALI) and derived air concentrations (DAC) were calculated for each sample. The individual results were then averaged to establish the typical solubility characteristics, ALI and DAC for Smith Ranch - Highland material. In terms of the D/W/Y classification, the majority of the samples were classified as type D with a small component of type W material. The resulting ALI and DAC values are the same as the default type D material at 1 μ Ci and 5E-10 μ Ci/ml, respectively.

2.0 EXPERIMENT

The experimental circuit used in this experiment was designed specifically to study the kinetics of uranium dissolution in simulated lung fluids. Details of the experimental design and set up are found in the report Solubility of Radionuclides in Simulated Lung Fluid [1]. In brief, each site submitted several samples of their uranium concentrate for analysis. The samples were passed through a 20 μ m filter and the filtered uranium powder placed between two glass fiber filters in 47-mm polypropylene filter holders that were used as extraction cells. A 20 μ m filter was used to ensure that only particles of a size range, which may reach lungs and react with lung fluid, were included in the analysis; a more detailed analysis of the particulate size analysis conducted as part of this experiment is presented a CTI-RC Memo[3]. The simulated lung fluid was then passed over the sample at a predetermined flow rate. After passing over the product samples, the simulated lung fluid was collected and analyzed in a lab for uranium content. This uranium dissolution experiment was continued for 100 days. At the end of the experiment the residual uranium on a filter was analyzed using an X-Ray fluorescence method.

3.0 SOLUBILITY CLASSIFICATION

Current United States regulations in relation to internal dosimetry and solubility classifications are based on ICRP 30. ICRP 30 divides the respiratory tract into three regions and within each region are a number of compartments, each associated with a clearance pathway (absorption or particle transport). To describe the clearance of radioactive materials from the lungs, the materials are classified as D (day), W (week), and Y (year), referring to retention time (corresponding to dissolution half-time of the material) in the pulmonary region. The retention times for the classifications are as follows:

	Retention Time
	(Corresponds to Dissolution Half-Time)
Type D	up to 10
Type W	10 - 100
Type Y	greater than 100

4.0 **RESULTS**

The simulated lung fluid study conducted on the Smith Ranch – Highland uranium product by the Cameco Technology and Innovation Research Centre followed published methods in performing this experiment [Ansoborlo et al. 1999][4]. The study was run for 100 days and was followed up with an X-ray fluorescence (XRF) measurement of the residual material left on the glass fiber filter.

Table 1 shows the fraction of uranium remaining on the filter (retention fraction) at each sampling time based on ICP Mass Spectrometry analysis. Standard errors on these results vary from 5% to 10%, with the higher errors occurring later in the testing, near the 100 day mark. Table 2 shows the summary of the XRF results taken at the end of the experiment. It is important to note that though the ICP-MS indicated that between approximately 0.2% and 6% (\pm 10%) of the uranium remained undissolved at 100 days (Table 1), it was later shown through the XRF analysis that the majority of that material was dissolved uranium that had been reabsorbed by the filter. To determine if there had been reabsorption of uranium on the filter, first the filter was inspected visually. There were no visual traces of undissolved uranium remaining on the filter. It is relatively easy to see undissolved material (if any is present) on the filter. Following the visual inspection, the filter was dissolved and analyzed. The amount of uranium found was lower than or close to the detection limit of the method. This means that within experimental error there is essentially no material remaining undissolved at 100 days.



| Table_1: Smith Ranch – Highland Material Dissolution Kinetics

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Sample No	SRH 482- 1	SRH 482- 6	SRH 482- 13	SRH 482- 21	SRH 482- 29	SRH 482- 39	SRH 482- 42	SRH 483- 1	SRH 483- 9	SRH 483- 16
U Total in 50 mg Sample (%)	50	50	50	50	50	50	50	50	50	50
U Total (mg)	77.4	79.4	78.2	78.3	78.9	78.7	79.0	79.0	79.4	79.5
Elapsed Time (d)										
0	100	100	100	100	100	100	100	100	100	100
0.02	95.24	98.13	93.24	99.04	97.19	81.57	97.48	98.72	74.18	99.47
0.13	81.22	97.37	83.13	95.77	89.16	54.77	87.97	96.99	43.50	98.72
0.31	63.60	96.02	62.18	87.24	74.07	41.45	75.91	95.57	33.86	96.20
1.00	48.67	81.73	28.25	53.69	33.19	29.28	52.10	86.32	17.50	83.94
1.31	44.02	67.05	22.38	46.55	26.27	27.43	47.58	82.84	14.70	76.09
2.00	28.02	45.36	13.35	31.60	18.10	24.15	34.01	70.53	10.96	43.09
2.31	21.24	40.82	12.38	27.90	16.72	22.56	29.77	65.78	9.29	34.33
3.00	15.14	32.05	10.89	22.21	14.22	20.32	22.23	55.69	7.85	20.86
4.00	11.38	27.58	9.49	18.38	11.97	17.06	19.02	48.29	6.25	16.75
7.00	7.38	20.12	4.48	11.94	8.96	14.12	14.30	35.43	2.59	10.68
9.00	4.71	16.48	4.26	9.39	7.32	12.99	11.04	26.22	1.48	7.85
11.00	3.24	12.04	4,24	7.14	6.37	11.74	8.18	18.87	1.26	5.82
14.00	3.05	9.03	4.23	5.26	5.86	10.66	6.74	13.55	1.03	4.55
21	3.01	4.22	4.22	4.66	5.60	9.47	6.20	2.79	0.71	4.01
28.00	2.99	2.87	4.21	4.61	5.58	8.33	6.08	2.42	0.53	3.84
36.00	2.98	2.76	4.20	4.59	5.57	7.72	6.01	2.02	0.48	3.74
42.00	2.97	2.73	4.19	4.58	5.56	7.05	5.99	2.01	0.43	3.69
49.00	2.97	2.72	4.19	4.57	5.55	6.80	5.98	2.00	0.40	3.66
56.00	2.96	2.70	4.18	4.56	5.54	6.61	5.97	1.99	0.37	3.62
63.00	2.95	2.69	4.17	4.55	5.53	6.51	5.96	1.98	0.34	3.59
71.00	2.94	2.68	4.15	4.54	5.52	6.43	5.95	1.97	0.32	3.56
84.00	2.92	2.66	4.14	4.52	5.51	6.23	5.93	1.96	0.30	3.53
100.00	2.92	2.64	4.12	4.51	5.49	5.96	5.91	1.94	0.26	3.51





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Sample No.	SRH 482-1	SRH 482-6	SRH 482-13	SRH 482-21	SRH 482-29	SRH 482-39	SRH 482-42	SRH 483-1	SRH 483-9	SRH 483-16
U Total in 50 mg Sample, (%)	77.4	79.4	78.2	78.3	78.9	78.7	79.0	79.0	79.4	79.5
U in a Sample Initial (mg)	38.71	39.68	39.08	39.16	39.44	39.33	39.495	39.48	39.7	39.77
U in a Sample Calculated from Extracted U (mg)	37.6	38.6	37.5	37.4	37.3	37	37.2	38.7	39.6	38.4
U Residual on a Filter After the Completion Calculated (mg)	1.1	1.1	1.6	1.8	2.1	2.3	2.3	0.8	0.1	1.4
U Residual on a Filter XRF Analyses After the Completion (mg)	0.5	0.4	0.3	0.8	0.6	0.5	1.2	0.6	0.1	0.5
Difference	0.6	0.7	1.3	1.0	1.5	1.8	1.1	0.2	0.0	0.9
Standard Error (%)	1.6	1.7	3.3	2.5	3.9	4.7	2.8	0.5	0.0	2.2





4.1 Absorption parameters and Dosimetric Quantities

This equation and the methods used to determine the parameters are described in detail in the CTI-RC report [1].

The method that was used to determine the DWY classification is to fit a curve to the dissolution data in Table 1 and use the fit parameters from that curve to classify the material into DWY. To estimate the dissolution times and the fraction of material assigned to each category, an exponential model was used. The number of terms used in the model was based on statistical tests; the decision as to how many terms to use was based on the amount of error associated with each fit. For this experiment, the equation with the lowest error was chosen for each sample. In all but one case, for Smith Ranch – Highland samples, the equation with the lowest error. The three term equation used was:

$$\frac{M}{M_0} = f_1 \exp(-0.693\frac{t}{T_1}) + f_2 \exp(-0.693\frac{t}{T_2}) + f_3 \exp(-0.693\frac{t}{T_3})$$
(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_3 - fraction of total U with corresponding dissolution half-time T_3

 $f_1 + f_2 + f_3 = 100\%$

Note: the two term equation was of the same format, with the term involving f_3 and T_3 removed.

The results of the X-Ray diffraction analysis performed on three of the Smith Ranch – Highland samples indicated the presence of crystalline metaschoepite, $UO_3 \cdot 2H_2O$, and uranium trioxide hydrate, $UO_3 \cdot 0.8H_2O$, in all three samples [3]. Small quantities of unidentified amorphous material might also be present in almost all samples.

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

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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.

4.2 **Dosimetric Quantities**

The current ALI and DAC used at Smith Ranch – Highland are based on the type D Uranium Natural parameters in 10CFR20. The purpose of this experiment was to determine the site specific solubility for material at the site. Part of this assessment was to determine the breakdown of Smith Ranch – Highland uranium product into the DWY solubility types defined by ICRP 30 and 10CFR20. In addition, we needed to consider whether any mixture of isotopes existed. Isotopic analysis of the uranium product showed that only isotopes of uranium were present in fresh yellowcake. At the Cameco Crow Butte Operation, which is similar to Smith Ranch – Highland, smears from surfaces throughout the main plant were collected and analyzed for the presence of alpha (natural uranium) and beta/gamma (Th²³⁴ and Pa²³⁴m) emitters. A dual channel alpha and beta/gamma counter was used to measure the emissions from the smears. These smears characterize the material that has the potential to be made airborne throughout the plant and inhaled by workers. The smears indicated that natural uranium and the short-lived decay products are in approximate equilibrium. Though this analysis has not been performed at the Smith Ranch – Highland operation as of yet, the results are expected to be similar. To account for the presence of these short-lived decay products at the time of inhalation, a mixed ALI and DAC was calculated and presented Table 3; the calculation used was (C_{Unat}/ALI_{Unat}+C_{Th234}/ALI_{Th234}+C_{Pa234}/ALI_{Pa234})⁻¹ and similarly for the DAC, where C represents the concentration ratio of each material assuming equilibrium. As shown, the ALI and DAC for each solubility class are unchanged from the Uranium Natural values. Note, as per standard protocol, all final results are shown to one significant digit as this is the number of significant digits shown in 10CFR20 and therefore the maximum that can be used in the final output of the equations.

Isotope	Solubility	ALI (µCi)	DAC (µCi/ml)
Uranium Natural	D	1	5E-10
	W	0.8	3E-10
	Y	0.05	2E-11
²³⁴ Th	W*	200	8E-8
	Y	200	6E-8
²³⁴ Pa	W*	8000	3E-6
	Y	7000	3E-6
Final Default	D	1	5E-10
Mixture	W	0.8	3E-10
	Y	0.05	2E-11

Table 3: Mixed ALI and DAC Values

* The solubility class for 234 Th and 234 Pa was assumed to be W, because the isotopes are not expected to be in oxide or hydroxide forms.

As part of the proposed sampling plan to the NRC, additional sampling for isotopes such as Ra226 is ongoing. Should any subsequent analysis indicate the detectable presence of additional radionuclides, the mixed ALI and DAC will be adjusted accordingly.

Table 4 shows the solubility parameter fit results following the method outlined in the report Solubility of Radionuclides in Simulated Lung Fluid [1]. The F values represent the fraction of material with the given dissolution half-time (the T value). It is again important to note that this fit was performed based on the results of Table 1 and the fraction assigned to retention times (T values) greater than 100 days should be reassigned to the time from of 10-100 days as subsequent analysis showed no measureable material remaining after 100 days, within error. This is a conservative assignment because some of that material absorbed onto the filter could have been dissolved before day 10, however we are assuming all of the absorbed material was dissolved after day 10.

Along with the solubility parameters, Table 4 also shows the subsequent DWY classification. Each sample is assigned to DWY based on the dissolution half time values, T values, and the fraction assigned to each classification is based on the F values. For example, using sample SRH 482-6, T_1 is less than 10 days, so 84.5% (the F_1 value) is assigned to class D and 15.5% (F_2) is assigned to class W because T_2 is between 10 and 100. In cases where both T_1 and T_2 are below 10 days, the fraction considered Class D is the sum of F_1 and F_2 . As stated, where a T_3 exists, its associated F value is included in the Type W fraction based on XRF results. For all samples the fraction of material classed as Type D is greater than 80% and most are greater than 90%.

Finally, Table 4 includes the ALI and DAC for each sample and an overall plant average. The final ALI and DAC values were calculated by taking a weighted sum of the percent contribution of each solubility class within a sample multiplied by the appropriate mixed ALI and DAC for that solubility class, from Table 3. One additional note is that the default ALI and DAC from 10CFR20 are based on a particle size of 1 μ m. The particulate size analysis for Smith Ranch – Highland product shows that particle sizes are significantly larger than the default assumption. This means that use of default values based on a 1 μ m assumption are conservative for our product; a larger particle size would be associated with a lower dose. However, the more conservative defaults have been used in determining the site specific ALI and DAC for Smith Ranch – Highland. Again, the final result is to one significant digit as this is the number of digits available in 10CFR20, with two significant digits have been carried through the intermediate calculations as per standard protocol.





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Sample	F ₁	T ₁	F ₂	T ₂	F ₃	T ₃	MSD	Type D	Type W	ALI	DAC
No. ^a	(%)	(d)	(%)	(d)	(%)	(d)	(%)	(%)	(%)	(µCi)	(µCi/ml)
SRH 482-1	21.8	0.1	74.8	1.3	3.5	288.4	1.9	96.5	3.5	0.99	4.9E-10
SRH 482-6	84.5	1.6	15.5	18.2			3.8	84.5	15.5	0.97	4.7E-10
SRH 482-13	76.2	0.4	19.7	1.8	4.1	8.94E+08	1.0	95.9	4.1	0.99	4.9E-10
SRH 482-21	80.8	0.9	14.8	5.0	4.4	5.34E+08	1.6	95.6	4.4	0.99	4.9E-10
SRH 482-29	84.7	0.5	9.9	4.3	5.4	1.11E+09	1.4	94.6	5.4	0.99	4.9E-10
SRH 482-39	53.4	0.0	33.4	1.2	13.3	60.1	1.7	86.7	13.3	0.97	4.7E-10
SRH 482-42	55.7	0.5	38.0	2.7	6.3	814.6	1.5	93.7	6.3	0.99	4.9E-10
SRH 483-1	18.5	1.8	79.9	4.9	1.6	3.02E+08	1.9	98.4	1.6	1.00	5.0E-10
SRH 483-9	56.3	0.0	42.7	0.9	1.0	2033157	1.6	99.0	1.0	1.00	5.0E-10
SRH 483-16	96.6	1.7	3.4	10000			5.9	96.6	3.4	0.99	4.9E-10
									Average	1	5E-10

Table 3: Smith Ranch – Highland Solubility Parameters and Dosimetric Quantities

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5.0 CONCLUSION

Cameco has completed a solubility study to analyze the solubility characteristics of the Smith Ranch – Highland yellowcake product. This study showed that the yellowcake produced at Smith Ranch – Highland was primarily of solubility type D with a relatively low type W component. The resulting ALI and DAC values are the same as the default type D material at 1 μ Ci and 5E-10 μ Ci/ml. As stated in the introduction, this analysis of the data incorporates the most recent comments from the NRC and Cameco and supersedes all previous submissions in this area.

6.0 **REFERENCE LIST**

- Cameco Innovation and Technology Development Research Centre, Solubility of Radionuclides in Simulated Lung Fluid, Report Number 200183, Report Number RC-09-05, 2009
- 2. ICRP, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30 Part 1, Ann ICRP 2(3-4), 1979.
- 3. Tairova, G. "Analysis of Smith Ranch Highlandand Smith Ranch-Highland Concentrate Samples", Cameco Technology and Innovation Research Centre Memo, September, 2011.
- 4. Ansoborlo, E.; Hengé-Napoli, M.H.; Chazel, V.; Gibert, R.; Guilmette, L.A. *Review and Critical Analysis of Available In Vitro Dissolution Tests*, Health Physics, 77(6): 638-645; 1999.

ADDENDUM 1 TO APPENDIX I

RADIOLOGICAL MONITORING EQUIPMENT

The following equipment will be used at the Smith Ranch site and its related satellite facilities. For personnel monitoring, either Ludlum model 3 or equivalent ratemeters and 43-5 or equivalent probes or Ludlum model 2241 scalar/ratemeters, or equivalent and 43-5 or equivalent probes. For equipment alpha scanning, a Ludlum 2241 scaler/ratemeter or equivalent and a 43-65 or equivalent probe is used. There are several probes that are compatible with these meters that may be used in the future. If these meters are used with alternate compatible probes, the minimum detectable concentration (MDC) will be verified to ensure it meets the requirements. To meet the requirement for measurement of beta radiation, where ambient background conditions permit, monitoring for both alpha and beta radiation will be performed using a Ludlum 2224 scaler/ratemeter or equivalent and a 43-89 or equivalent probe.

Regulatory Guide 8.30 indicates that the removable release limit is 1000 dpm/100cm² and the total activity release limit is 5000 dpm/100cm². The MDC for scaler measurements will be determined based on the method by Strom and Stansbury as referenced in NUREG 1507, shown in equation (1). Cameco has assumed a conservative value of 500 dpm/100cm² (10% of the total activity limit) as the maximum MDC in this calculation. The efficiency assumption for all equipment is 15%, which is somewhat below the typical equipment efficiency.

$$MDC \left(\frac{DPM}{100 cm^2}\right) = \frac{3+3.29\sqrt{R_b t_g(1+\frac{t_g}{t_b})}}{\varepsilon t_g(\frac{SA}{100 cm^2})}$$

where: $R_b =$ the background count rate

- t_g = the sample count time
- $t_b =$ the background count time
- ε = the counter efficiency

SA = probe surface area (cm²)

For scanning using a ratemeter, the MDC will be based on Regulatory Guide 1507. The scan MDC is calculated as follows:

$$Scan MDC \left(\frac{DPM}{100 cm^2}\right) = \frac{MDCR}{\sqrt{p}\epsilon_i\epsilon_s \frac{Probe Area}{100 cm^2}}$$

Where: MDCR = Scan Minimal Detectable Count Rate = $s_i^*(60/i)$

i = Scan Interval = 6 seconds

 $S_i = minimum$ number of net counts detectable in interval = $d'\sqrt{b_i}$

d' = level of performance (Table 6.1 from NUREG 1507) (false negative portion =0.6, true positive = 0.95)

 b_i = average number of bkg counts in interval = bkg cpm * Scan interal/60

p = surveyor efficiency; assumed 0.5

 ϵ_i = equipment efficiency (18%)

 ϵ_s = surface efficiency (0.54) from section 5 of NUREG 1507

(1)

(2)

Tables 1 and 2 below show the maximum background measurement for this type of equipment for varying count times and both detector types.

Meter/Detector	Active Surface Area (cm ²)	Count Time (min)	Background Count Time (min)	Maximum Background Count (in specified time)	MDC (dpm/100cm ²)
Model 2241/43-5	76	0.5	5	525	494
	76	0.5	1	75	488
	76	1	5	1100	498
	76	1	1	130	494
Model 2241/43-65	63	0.5	5	350	496
	63	0.5	1	50	490
	63	1	5	750	499
	63	1	1	$\begin{array}{c c c c c c c c c c c c c c c c c c c $	499
Model 2224/43-89	125	0.5	5	1600	498
	125	0.5	1	235	498
	125	1	5	3100	495
	125	1	1	375	497

Table 1: Maximum Background Count Rates to meet an MDC of 500 dpm/100cm² for Scalar Counting

Table 2: Maximum Background Count Rates to meet an MDC of 500 dpm/100cm² for Scanning

Meter/Detector	Bkg Counts (cpm)	Scan Inteval (sec)	d'	Si	MDCR	р	ϵ_i	ϵ_s	Probe Area (cm ²)	Scan MDC (dpm/100cm ²)
Model 3/43-5	10	6	1.38	1.69	17	0.5	0.15	0.54	63	482

Based on actual plant surveys, the alpha background conditions are typically less than 10 counts per minute (cpm), meaning for alpha measurements with a 30 second count time on a scaler meter, the actual MDC is 254 dpm/100cm² for a 63 cm² active area probe and 128 dpm/100cm² for a125 cm² probe for scaler meters and 382 dpm/100cm² for a ratemeter. This is less than 10% of the 5000 dpm/100cm² limit for total contamination. In cases where the beta background does not meet the conditions outlined in **Table 1**, the equipment may be moved to a lower background area where the criteria is met. Prior to beginning development of a survey program for beta contamination, knowledge of background conditions are required, as this will impact MDC and scanning methodology. Background alpha and beta measurements were made using a Ludlum dual channel alpha/beta 43-89 probe at the personal frisking stations. Using the scaler mode of the detector at these locations, background alpha measurements averaged approximately 4 cpm, and the beta measurements averaged approximately 1400 cpm. This average is above the maximum background counts in **Table 1**, and most individual areas were in excess of the maximum criteria. MDC calculations followed the method outlined in equation (1).

Given these background conditions, the use of beta radiation measurements for personnel scans with a limit of 1000 dpm/100cm² is not achievable at some locations in the plant. As a result, Cameco proposes to either install shielding sufficient to reduce the gamma background to permit the beta MDC to be met; or 2) survey for alpha contamination and, if the survey meets acceptable limits, proceed to a beta monitoring location with background gamma radiation levels sufficient to meet the beta MDC.

For personnel scanning Cameco proposes using a ratemeter for a quick survey to determine any high areas and then use a 30 second scaler count to determine the level of any contamination. All frisking

stations at the Smith Ranch licensed facilities will be below conservative background assumption, or it will be adjusted for a specific area, which will be verified monthly by a Health Physics Technician (HPT). Additionally, frisking stations currently have a Ludlum 43-5 probe in use, which meets the active probe size assumption of 63 cm². There are currently two options being considered if a scaler meter is chosen: (1) purchase of a Ludlum scaler meter, likely a model 2241, or (2) using the pulse output from Ludlum 3 ratemeters to produce and report a 30 second integrated count. The feasibility of the second option is still under investigation.

For equipment releases, the current practice involves 1 minute scaler measurements of specified points. Cameco plans to continue with the use of this procedure. The associated MDC for an alpha measurement, using the assumptions of a 63 cm² probe, an efficiency of 15%, a background of 10 cpm and a 1 minute count time, is 187 dpm/100 cm². Using a 125 cm² probe would result in an MDC of 94 dpm/100 cm². In areas with elevated background beta radiation, where a beta MDC of 500 dpm/100cm² cannot be achieved, the equipment will be moved to a beta monitoring location with background gamma radiation levels sufficient to meet the beta MDC. In areas where the MDC guidance of 500 dpm/100cm² can be achieved, the beta contamination could be measured directly using the same methodology and a 43-89 probe or equivalent.

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ADDENDUM 2 TO APPENDIX I

SMITH RANCH RADIONUCLIDE MIXTURE ASSESSMENT

NRC staff have requested that Cameco provide additional airborne isotopic sampling and analysis for the life of the Smith Ranch-Highland site to ensure that compliance with 10 CFR 20.1204(g) is being maintained (RAI TR-38). Cameco collected airborne particulate samples from the SRH in-plant sampling stations over a 31 month period. The collected samples were analyzed for natural uranium, thorium-230, radium-226 and lead-210. Cameco then assessed the data for compliance with 10 CFR 20.1204(g). The results of the study are provided below.

10 CFR 20.1204 (g) states "when a mixture of radionuclides in air exists, licensees may disregard certain radionuclides within the mixture if:

1) The licensee uses the total activity of the mixture in demonstrating compliance with the dose limits in 10CFR20.1201 and in complying with the monitoring requirements in 10CFR20.1502(b), and

2) The concentration of any radionuclide disregarded is less than 10 percent of its DAC, and

3) The sum of these percentages for all the radionuclides disregarded in the mixture does not exceed 30%".

In accordance with the guidance provided in NRC Regulatory Guide 4.14, airborne particulate samples from sampling stations located at the CPP, satellites and wellfields were collected and analyzed for natural uranium, thorium-230, radium-226 and lead-210 over a 31 month period between July 2009 and February 2011. Samples were collected using a high-flow pump operating continuously for approximately one week, with particulates collected on a 47mm filter. The raw data, including sample LLDs, used in this analysis is included in **Table 3**. Polonium-210 was not analyzed directly, however it is a decay product of Pb-210 and because there is no source of Po-210 present other than the decay of Pb-210, it is assumed that there is full equilibrium between the two radionuclides.

The purpose of this sampling was to determine whether the concentrations of these radionuclides remain at less than 10% of their respective DACs in the facility, and whether the sum of these percentages for any disregarded radionuclides remains at less than 30%. The DAC for Class W Th230, which is lower than for Class Y, is $3 \times 10^{-12} \,\mu$ Ci/ml. To meet the DAC requirement, Th230 must be present at a concentration less than $3 \times 10^{-13} \,\mu$ Ci/ml. Similarly, Ra226 must be present at a concentration less than $3 \times 10^{-11} \,\mu$ Ci/ml, which is 10% of its DAC of $3 \times 10^{-10} \,\mu$ Ci/ml. Pb210 must be present at a concentration less than $1 \times 10^{-11} \,\mu$ Ci/ml, which is 10% of its DAC of $1 \times 10^{-10} \,\mu$ Ci/ml, and Po-210 must be present at concentration less than $3 \times 10^{-11} \,\mu$ Ci/ml, which is 10% of its DAC of $1 \times 10^{-10} \,\mu$ Ci/ml, and Po-210 must be present at concentration less than $4 \,\mu$ Ci/ml, which is 10% of its DAC of $3 \times 10^{-10} \,\mu$ Ci/ml. A DAC of $5 \times 10^{-10} \,\mu$ Ci/ml was used for natural uranium as solubility studies performed for both the Crow Butte and Smith Ranch-Highland operations have demonstrated that the uranium is of class D solubility.

Utilizing the analytical data collected over the 31 month period, the average concentration for each isotope was calculated. The average value for each isotope was then compared with the DAC values from 10 CFR 20. **Table 1** presents the analysis of the in-plant isotopic samples collected at the Smith Ranch-Highland operation. For this analysis, background environmental concentrations of these isotopes have not been subtracted.

	Lab Result Unat (µCi/ml)	Lab Result Th-230 (µCi/ml)	Lab Result Ra-226 (µCi/ml)	Lab Result Pb-210 (µCi/ml)	Po-210 (μCi/ml)
Average Concentration	1.07E-12	0.00E+00	4.16E-15	3.73E-14	3.73E-14
DAC	5.00E-10	3.00E-12	3.00E-10	1.00E-10	3.00E-10
Average as % of DAC	0.2%	0.000%	0.001%	0.04%	0.001%

Table 1: Isotopic Analysis of Airborne Dust Samples at Smith Ranch-Highland

As shown in **Table 1**, all of the isotopes analyzed were present in concentrations significantly less than 10% of their respective DACs. In addition, the sum of the DAC percentages from Th230, Ra226, Pb210 and Po-210 combined is significantly less than 1%, thereby meeting the 10 CFR 20.1204(g) "disregard" criteria of being less than 30% of their respective DAC. Therefore, these three radionuclides can be disregarded from the determination of internal dose under 10 CFR 20.1204(g).

The NRC has, in the past (unrelated to this license renewal action), requested that Cameco address potential doses from the short lived decay products of U238, specifically Th234 and Pa234m. As stated in section 3.3.1 of ICRP 30, daughter products produced in the body after an intake of a specified radionuclide are taken into account in the ALI specified by ICRP 30. This means that the ALI for uranium in ICRP 30 and 10 CFR 20 Appendix B already include the dose from Th234 and Pa234m that is produced as a result of radioactive decay after the intake of U238. Therefore, the only additional dose that needs to be considered is from decay products present at the time of inhalation. Given that the half-lives of these short-lived daughter products, 24.1 days for Th234 and 1.17 minutes for Pa234m, are so short, less than 1% of the total activity potentially inhaled at the time of an intake remains after 6 months. To be conservative however, a comparison of concentrations with the annual DAC in 10 CRF 20 Appendix B for these radionuclides was performed, where available. The comparison considered that U238 contributes 49% of the specific activity of Unat and assumed both daughter products to be in equilibrium with the U238 at the time of inhalation. Table 2 shows the calculated U238 activity, based on the measured Unat average activity, and a comparison of the daughter product activities to DACs from 10 CFR 20 where available. It is worth noting that Pa234m does not have a DAC in 10 CFR 20, so a comparison could not be made. In addition, Pa234 was excluded from the analysis because only 0.13% of the Pa234m decays into Pa234 making it an extremely small contributor.

	Calculated U238 Activity (µCi/ml)	Th234 Activity (µCi/ml)	Pa234m Activity (μCi/ml)
Average	5.2e-13	5.2e-13	5.2e-13
DAC		6E-8	Not available
Average as % of DAC		0.0009%	

1 able 2: Comparison of Snort-Lived Decay Products of 0238 to DA	Гab	ole 2:	Compariso	n of Short-	Lived Decay	y Products of	' U238 to D A	AC
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As with the other radionuclides discussed previously, Th234 is less than the criteria in 10CFR 20.1204(g) and can be disregarded. It should be noted that neither ICRP 30 nor 10 CRF 20 have provide calculated ALIs or DACs for Pa234m. Therefore, within the bounds of 10 CFR 20 regulations, there does not appear to be a method or a mandate to calculate a dose for this radionuclide. Nevertheless, the low activities present from these radionuclides, their extremely short half-lives and the fact that the dose from both Th234 and Pa234m are already included in the ALI and dose for uranium for all but a potentially brief time period presents a strong argument that the risk to workers from these radionuclides is minimal.

The internal dose from airborne particulates will continue to be assessed based on the total alpha activity present compared with the DAC and ALI values for natural uranium. In addition, because Smith Ranch-Highland is a long running operation and, because the particulate concentrations are so far below the criteria in 10 CFR 20.1204(g), Cameco considers that, going forward, performing in-plant isotopic analysis once per license term at time of license renewal will be sufficient to confirm that no changes have occurred.

Area	Detailed Location	Result Unat	Result Th-230	Result Ra-226	Result Pb-210
	LID	1 0F-16	1 0F-16	1 0F-16	2 0F-15
CPP	Dryer Area	7 08F-13	<u>-2 07F_14</u>		_5 38F-14
CPP	Dryer Area	$\frac{7.08E-13}{2.41E-12}$	9 36F-15	8 85E-14	<u>1 74F-12</u>
CPP	Dryer Area	7 40E-13	-3 97F-14	-2 59E-14	_4.71E-12
CPP	Dryer Area	2 82E-13	-7 36E-14	-2.39E-14	-3 37F-14
CPP	Dryer Area	1 17E-11	_2 10E-15	1 18F-14	6 76E-14
CPP	Dryer Area	<u>1.17E-11</u>	-2 10E-15	1 18E-14	6.76E-14
CPP	Dryer Area	2 90E-12	7.70E-16	2 50E-15	5 20E-15
Wellfields	Header House 15-22	5 10E-15	1.10E-15	9.00E-16	2.20E-15
Wellfields	Hender House 15 20	6 70E 15	7.00E 16	2 30E 15	4 30E-14
Wellfields	Header House F 16	2.00E-15	3.50E-17	4.10E-15	1.40E-14
CPP	Lab	1.08E 13	6 76E 14	\$ 12E 15	
CPP	Lab	1.96E-13	4 72E 15	3.05E 14	-1.03D-12
CPP	Lau	1.99E-13	4.72E-15	-5.03E-14	-J.00E-13
CPP	Lab	1.41E-13	-4.71E-10	4.04E-15	1.21E-13
CPP		1.41E-13	-4./1E-10	4.04E-15	1.21E-13
CPP	Precipitation Area	3.54E-13	-1.4/E-14	-3.01E-14	2.19E-13
CPP	Precipitation Area	1.04E-13	-2.07E-14	2.30E-15	1.20E-12
	Precipitation Area	2.11E-13	-2.80E-15	4.50E-14	-8.88E-13
CPP	Precipitation Area	3.12E-13	1.01E-14	-2.53E-14	5.99E-13
	Precipitation Area	3.68E-14	-2.44E-16	2.45E-14	2.01E-14
CPP	Precipitation Area	2.68E-14	-2.44E-16	2.45E-14	2.01E-14
СРР	Precipitation Area	6.80E-14	4.30E-16	1.60E-15	0
СРР	Precipitation Area	1.70E-13	2.40E-16	5.40E-15	1.30E-14
CPP	Precipitation Area	3.40E-13	5.80E-16	2.00E-15	1.40E-14
CPP	RO Area	1.19E-13	-1.48E-15	1.02E-14	7.53E-14
CPP	RO Area	1.19E-13	-1.48E-15	1.02E-14	7.53E-14
CPP	RO Area	1.70E-14	2.80E-16	7.50E-16	1.10E-14
Satellites	S-1 Process Area	2.30E-15	-1.40E-15	1.90E-15	5.00E-14
Satellites	Sat 2 - Process Area	8.4E-16	-9.4E-16	8.7E-15	2.4E-14
Satellites	Sat 2 - RO Area	1.8E-15	-1.1E-15	3.8E-14	4.5E-14
Satellites	Sat 3 - Process Area	1.85E-15	-5.3E-16	9.7E-15	4.3E-14
Satellites	Sat 3 - Trailer Bay	2E-15	-6.9E-16	2.2E-14	3.9E-14
Satellites	SR-1 - RO Area	1.60E-14	-9.80E-16	1.20E-15	6.20E-14
Satellites	SR-2 - Process Area	1.90E-15	-1.05E-15	5.80E-15	4.30E-14
Satellites	SR-2 - RO Area	5.70E-16	-1.10E-15	2.86E-16	1.10E-14
CPP	YC Storage	1.70E-12	1.30E-15	1.40E-15	5.30E-15
CPP	YC Storage Area	6.20E-12	-2.19E-16	3.23E-15	7.56E-14
CPP	Yellowcake Storage	4.85E-13	-5.79E-15	-3.66E-14	-4.04E-14
CPP	Yellowcake Storage Area	2.58E-13	-3.61E-14	-1.87E-14	-4.25E-13
CPP	Yellowcake Storage Area	6.20E-15	-2.19E-16	3.23E-15	7.56E-14

Table 3: In-Plant Isotopic Data from Smith Ranch - Highland

