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ABSTRACT

Stockwell, D.R., Bentley, K.W. & Kerr, C.B. (1991). *In vitro* dissolution of uranium mill products by the batch replacement method. Technical Memorandum 33, Supervising Scientist for the Alligator Rivers Region.

Sixteen post-calcination uranium concentrate samples from two Australian uranium mills were analysed using the batch replacement method of Dennis et al. The samples were selected to represent a range of sulphate content. Results indicate that the degree of *in vitro* dissolution varies between samples and between solvents, and that the proportion of the most soluble fraction (class D) is linearly related to the sulphate content of the material. Results using de-ionised water, physiological saline and a simulated lung fluid indicate that, following initial release of the soluble fraction, the extent and range of *in vitro* dissolution of the remaining material is comparable to values reported in the literature. Close examination of the *in vitro* dissolution results using the batch replacement method indicate, however, that the fraction of uranium released cannot be satisfactorily interpreted using a time-dependent exponential and a test appears to confirm this. Consequently, *in vitro* experimentation by the batch replacement method is of questionable value in prediction of *in vitro* behaviour in terms of dissolution half-times. Dissolution of concentrate in a gastric juice simulant (0.1 M HCl) was also measured as a potential indicator of gastrointestinal absorption following ingestion.

1 INTRODUCTION

The final stages of industrial uranium concentrate production involve the calcining, packaging, and sampling for analysis, of large amounts of material. Workers may be exposed to this material by inhalation or ingestion. The International Commission on Radiological Protection has formulated a model (ICRP 1966) to provide a basis for dosimetry. Predictions of material retention and excretion result in estimations of residual burden from which intake limits for radioactive materials may be derived. Critical to the model is a knowledge of the effective biological half-life of the material. For inhaled material the ICRP model uses three lung clearance classes: D (half-life 10 days or less), W (half-life 10-100 days), Y (half-life greater than 100 days). The dissolution characteristics and consequent human chemical and radioactive toxicity of uranium products are determined by the characteristics of the precipitation and calcination processes together with the efficiency of their operation. A knowledge of these dissolution characteristics is necessary to interpret incident exposures and routine bioassay measurements.

Previous studies by Steckel & West (1966), Cooke & Holt (1974), Kalkwarf (1978) Dennis et al. (1982), Eidson & Mewhinney (1980) and Eidson & Griffith (1984) have examined the dissolution of uranium from a number of uranic materials in simulated interstitial and surfactant lung fluids. (These fluids are designed to mimic human lung fluid in ionic composition, pH and buffering capacity [Moss 1979].) Dissolution half-time measurements on yellowcake from US uranium mill facilities (Dennis et al. 1982) had suggested changes to the accepted solubility of uranium in mill product from an insoluble (Y) classification to a mixture of soluble (D) class, less soluble (W) class and Y class components. The US Nuclear Regulatory Commission has incorporated these changes in Regulatory Guide 8.22, which outlines a bioassay program for US uranium mills (Nuclear Regulatory Commission 1988).

The objectives of the present study were: (1) to measure the *in vitro* dissolution behaviour of uranium product concentrate from Australian uranium facilities in de-ionised water, physiological saline and a similated lung fluid; (2) to assign lung clearance classes as defined by the ICRP to the materials examined; and (3) to measure the *in vitro* dissolution in simulated gastric juice. Our present study uses a modification of the batch method similar to that of Dennis et al. (1982).

The mill products examined in this report are from the Queensland Mines Ltd (QML) mill at Nabarlek and the Ranger Uranium Mines Pty Ltd (RUM) mill at Jabiru East, both in the Alligator Rivers Region of the Northern Territory. The mills, referred to here as mill A and mill B respectively, differ mainly in capacity rather than metallurgical practices: both employ acid leach, solvent extraction, ammonia neutralisation, slurry centrifugation and high temperature calcination of the precipitated ammonium diuranate. The mills are modern and employ current, best practicable technology. The principal difference in present metallurgy arises from the oxidation/acid leach: QML now uses Caros acid (H_2SO_4/H_2O_2) rather than manganese oxide/sulphuric acid. This process occurs early in the mill cycle and would be expected to have only a minimal effect on the composition of the final calcination product. The batches of QML product investigated included materials from the earlier manganese oxide and the current Caros acid leach, however batch selection constraints did not permit comparison between oxidation processes.

The final mill product is a high-temperature calcined non-homogenous uranium oxide concentrate which may vary between batches in physical, chemical and thermochemical composition and consequently in biological activity. Variability in composition may occur through contamination following co-precipitation (e.g. uranium sulphate), incomplete calcination resulting in a mixture of oxides, or through alterations in rates of precipitation following ammonia injection, giving different particle size distributions and surface area. Most of the materials examined in the previous studies referred to above were products containing ammonium diuranate, uranium trioxide and other thermochemical intermediates and final uranium octoxide product. Samples from the Ranger and Nabarlek mills are primarily uranium octoxide following calcination at elevated temperatures.

2 MATERIALS AND METHODS

2.1 Uranium concentrate samples

The uranium concentrates produced by the Ranger and Nabarlek mills are normally 98-99.5% U_3O_8, resulting from high temperature (800°C) calcination of ammonium diuranate. Sixteen samples of uranium concentrate were supplied by the two uranium mills (Table 1) from their archival collections and represented approximately 40 months (Mill A) and 4 months (Mill B) production. The sulphate content may be an indicator of the D solubility classification component and hence is possibly a useful parameter for evaluation of acute chemical toxicity. Samples were therefore chosen to cover the full range of observed batch sulphate content (0.4-8.2%). Most of these samples were atypical, derived either from early batches or following disruption of normal plant operation. Sulphate levels normally encountered in routine operation at both mills are between 0.2-1.5% w/w.

The archival samples were dried at 110°C for 24 hours before being allowed to cool in the presence of a strong desiccant. Unlike the products examined by Eidson & Mewhinney (1978) and by Kalkwarf (1978), our products were indistinguishable in colour both initially and during the course of dissolution, suggesting the absence of ammonium diuranate.

2.2 Composition of simulated lung fluid

The composition of the simulated lung fluid (SLF) used is shown in Table 2. It differs slightly in ionic composition from the SLF of Kalkwarf (1978), and contains a surfactant protein substitute (fraction 5 albumin BDH).

The SLF was stabilised against bacterial contamination with pentachlorophenol, and adjusted to pH 7.3 immediately before use. The presence of phosphate ion in our SLF did not result in observable precipitation of uranyl phosphate in any of the experiments.

2.3 Other solvents used

Other solvents used in this study included de-ionised water, physiological saline, and 0.1 M HCl. The relative efficacy of the de-ionised water and saline as solvents was investigated to determine the possible cationic promotion of dissolution rates for ammonium diuranate thermochemical intermediate contaminants present in the final commercial product.

Dissolution in 0.1 M HCl (simulating gastric juice) was measured as an indicator of the potential contribution of gastrointestinal absorption of ingested uranium product.

2.4 Extraction and analysis

Our methodology was similar to that of Dennis et al. (1982) although we used shorter sampling times in the earlier stages to examine the Class D fraction. Uranium analysis in this study was by the delayed neutron technique (Amiel 1962) using the Australian Nuclear Science and Technology Organisation's 100 kW reactor Moata.

A weighed quantity (50 ± 5 mg) of each uranium concentrate sample was placed into a separate polyethylene centrifuge tube. Five millilitres of one of the solvents was added to each tube. Tubes were capped and shaken for 2 min to disperse material and initiate dissolution before being transferred to an environmental cabinet maintained at 37°C and shaken at 300 rpm continuously to maintain solids in suspension. The pH of the simulated lung fluid was maintained at 7.3 by addition of dilute hydrochloric acid as the pH drifted slowly upwards.

After two minutes agitation each sample was centrifuged for 5 min at 2000 g, the supernatant decanted for analysis, fresh solvent added and the procedure repeated. A total of 13 extracts of each sample were obtained; period of extraction varied as shown below:

Extract:	1	2	3	4	5	6	7	8	9	10	11	12	13
Extraction time:	2 min	2 min	60 min	56 m	1 h	17 h	26 h	23 h	35 h	83 h	47 h	117 h	119 h
Cumulative extraction time:	2 min	4 min	64 min	2 h	3 h	20 h	46 h	69 h	104 h	187 h	234 h	351 h	470 h

Five replicates of sample B4 were extracted is simulated lung fluid and analysed. The mass of uranium present in each extract was calculated as a percentage of the original 50 mg sample and subtracted cumulatively from 100% to give the result in terms of per cent uranium remaining undissolved.

3 RESULTS AND DISCUSSION

Tables 3-6 sumarise the results in terms of per cent uranium remaining undissolved after each extraction using simulated lung fluid, physiological saline, de-ionised water and 0.1 M hydrochloric acid respectively. Figures 1a-c present the per cent uranium remaining undissolved as a function of dissolution time (time in contact with the solvent with constant agitation) for simulated lung fluid, physiological saline and de-ionised water, respectively.

To check experimental consistency, dissolution of sample B4 in simulated lung fluid was repeated five times. Figure 2 illustrates the results of these measurements indicating four out of the five were virtually identical; with the fifth, showing a similar trend to the other four but displaced downwards by ≈ 10%.

For presentation purposes, the per cent uranium remaining undissolved is given in Figs 1a-c using a linear scale. When a log scale is used, as in Fig. 2, each curve suggests a composite of three exponentials (a short-term component, a medium-term component and a longer-term component). The simulated lung fluid results are quite similar to those obtained by other authors (Dennis et al. 1982). However using a computerised mathematical iteration Vardavas (1988, 1989, pers. comm.) concludes that there are no unique solutions (using the data from these experiments) for values of f_1, f_2, f_3, $t_{\frac{1}{2}}(1)$, $t_{\frac{1}{2}}(2)$ and $t_{\frac{1}{2}}(3)$ in the function:

$$F = f_1 \exp\left[-0.693t/t_{\frac{1}{2}}(1)\right] + f_2 \exp\left[-0.693t/t_{\frac{1}{2}}(2)\right] + f_3 \exp\left[-0.693t/t_{\frac{1}{2}}(3)\right]$$

where $f_1 + f_2 + f_3 = 1$ and $t_{\frac{1}{2}}(1)$, $t_{\frac{1}{2}}(2)$, and $t_{\frac{1}{2}}(3)$ correspond to lung model inhalation Class D, Class W and Class Y time frames, respectively. This computer iteration suggests that experiments would need to run for a longer time to separate Class Y and Class W components. More data points are needed between 50-300 days.

Closer examination of Fig. 2 indicates that changes in slope corresponding to each assumed exponential component occurred at points where experimental time scales changed (minutes→hours→days). This raised the question: was the apparent three-component exponential curve an artefact of the experimental technique?

This is partly confirmed in Figs 3a-c, where the per cent uranium remaining undissolved when plotted against extract number (rather than against time) indicates a linear correlation, with the exception of results for samples A9 and A10 (which were abnormally high in sulphate content) using simulated lung fluid as the solvent. This suggests that after the initial removal of the readily soluble fraction during the first or second extraction, the proportion removed with each subsequent extraction is, within experimental error, constant irrespective of the dissolution time.

To test this, an experiment was run using sample B4 and simulated lung fluid as the solvent: a constant dissolution time of two minutes was used and 10 successive extracts were analysed. Figure 4 compares the results of this experiment with the mean ± 2 standard deviations curve for the 5 replicates of the original work using sample B4. The results of the successive two-minute extractions fall within the ± 2 standard deviations envelope of the earlier measurements using longer dissolution times. Thus, for example, the result obtained after 16 minutes (8 changes of solvent) is identical with the results previously obtained after 69 hours which was also 8 changes of solvent. Thus the number of changes of solvent appears to be the significant parameter rather than the time. This, in our opinion, appears to invalidate the batch replacement technique.

Figure 5a presents the per cent uranium remaining undissolved as a function of dissolution time for 0.1 M HCl and Fig. 5b presents the corresponding per cent uranium remaining undissolved as a funciton of extract number. All products examined dissolved significantly in HCl.

Figure 6 illustrates the variation of the per cent uranium dissolved during the first extraction in each of the solvents used with per cent sulphate content in each of the 16 samples examined. For each solvent, the amount of uranium dissolved in the first extract appears to be linearly related to sulphate content. Linear regressions yield the following equations:

Simulated lung fluid	$y = 2.12x + 3$	$r = 0.88$
Physiological saline	$y = 3.83x + 0.58$	$r = 0.95$
0.1 M hydrochloric acid	$y = 3.54x + 6.56$	$r = 0.89$
Deionised water	$y = 4.67x - 1.54$	$r = 0.89$

where: y = % uranium dissolved in first extract,
x = % sulphate content of the uranium concentrate, and
r = linear correlation coefficient.

4 CONCLUSIONS

1. In comparing the results of this work with that of previous authors using a similar technique, there appears to be good agreement. For the readily soluble (Class D) fraction this work shows that this fraction is linearly related to the sulphate content in the uranium

concentrate samples examined. In comparing solvents, similar results for a particular batch were obtained for simulated lung fluid, saline and de-ionised water.

2. From study of the data, however, it can be seen that changes of slope occur at changes of dissolution time. This dependence on an arbitrary experimental factor, the dissolution time interval, appears to invalidate the conclusions, based on this experimental procedure, on product solubility classifications previously made by other authors.

3. An experiment was run using constant short dissolution time intervals. This indicated that the rate of dissolution was in fact related to the number of times that fresh solvent was presented to the product material rather than to the total dissolution time. This strongly suggests that the commonly used batch technique is inappropriate, and that the results obtained using it are dependant on the choice of time interval. Further work needs to be done to confirm this conclusion.

4. All the products examined dissolved significantly in HCl (simulated gastric juice).

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Table 1. Details of samples of uranium concentrate

Mill A				Mill B			
Sample	Lot no.	% SO_4	% U_3O_8	Sample	Lot no.	% SO_4	% U_3O_8
A1	381	0.4	98.8	B1	3183	0.4	-
A2	379	0.7	98.4	B2	3164	1.5	-
A3	218	1.0	98.6	B3	3156	3.7	-
A4	246	1.8	97.5	B4	3154	5.5	-
A5	46	2.4	96.8	B5	3080	7.8	-
A6	133	3.7	95.8	B6	R15	-	93.7
A7	235	4.7	95.6				
A8	28	5.8	94.3				
A9	27	7.8	91.9				
A10	78	8.2	92.3				

- data not provided

Table 2. Composition of simulated lung fluid

Pentachorophenol was added as a preservative

Salt	conc. mM	Equiv. wt	Mass (g/L)
Sodium chloride	114	58.5	6.66
Sodium bicarbonate	31	84.0	2.60
Calcium acetate	5	97.2	0.49
Magnesium acetate	2	107.2	0.21
Dipotassium hydrogen phosphate	2	58.1	0.12
Dipotassium sulphate	1	87.1	0.09
Citric acid	1	70.0	0.07
Albumin (Fraction 5)	-	-	0.20

Table 3. Per cent uranium remaining undissolved after each extraction using simulated lung fluid

	Extract												
	1	2	3	4	5	6	7	8	9	10	11	12	13
MILL A													
1	96.4	95.7	95.3	94.9	94.6	93.8	92.6	90.5	89.1	87.3	80.9	78.7	76.8
2	98.0	96.6	94.9	93.9	93.5	92.6	91.3	89.3	87.0	85.3	84.1	81.9	79.5
3	93.5	92.2	89.4	87.4	85.4	80.1	78.8	77.9	76.6	75.0	74.2	72.6	69.4
4	94.8	93.6	91.7	91.1	90.8	90.2	87.8	86.0	84.1	82.9	82.5	81.1	80.6
5	95.5	94.6	92.5	91.2	88.7	87.3	86.0	85.5	84.5	83.4	82.4	80.4	78.6
6	84.0	81.1	78.7	77.9	77.2	76.1	75.0	72.4	69.8	67.9	58.5	55.7	53.5
7	88.5	86.4	83.5	83.0	82.5	81.5	79.5	78.8	71.2	68.9	68.0	66.3	65.1
8	76.9	73.2	67.2	63.1	60.4	54.3	51.2	50.1	48.1	45.6	44.1	42.1	41.0
9	81.0	72.2	56.6	52.4	49.3	37.7	29.3	27.5	25.6	24.0	22.8	21.4	19.6
10	80.4	74.3	70.7	70.1	69.3	68.8	68.0	65.6	64.2	62.7	61.1	58.6	56.5
MILL B													
1	96.1	95.1	93.6	92.5	92.0	90.2	88.7	86.7	79.9	77.9	74.5	73.2	70.5
2	93.3	92.0	90.7	89.4	88.7	87.2	85.1	83.2	80.3	77.7	75.5	73.3	70.4
3	89.2	86.9	84.9	83.2	82.5	81.5	80.2	76.6	71.1	69.1	67.2	65.9	63.9
4.1	84.1	80.7	78.8	78.0	76.7	75.3	73.2	69.9	65.9	64.9	62.7	61.1	58.1
4.2	83.2	80.2	79.8	78.3	77.5	76.1	74.7	72.4	68.2	65.8	63.5	62.3	59.9
4.3	81.1	77.2	74.7	72.3	70.4	69.4	67.8	63.8	59.6	57.1	54.4	52.8	50.0
4.4	86.3	83.1	80.8	79.6	77.3	76.0	74.5	72.6	68.1	65.3	62.7	60.9	57.9
4.5	85.4	82.0	80.0	78.5	76.7	75.2	73.6	71.4	67.8	65.3	63.1	61.4	58.7
5	86.1	77.3	73.8	71.7	70.6	69.7	68.5	67.6	65.4	64.1	63.1	61.0	55.5
6	96.5	95.4	91.8	88.8	87.2	85.4	81.8	79.6	75.8	73.6	71.1	69.6	66.6

Table 4. Per cent uranium remaining after each extraction using physiological saline (0.9% w/v)

	Extract												
	1	2	3	4	5	6	7	8	9	10	11	12	13
MILL A													
1	97.3	96.8	96.4	95.6	95.4	94.4	92.7	91.3	88.6	85.9	84.5	82.4	80.2
2	98.2	97.8	97.5	97.3	97.2	96.4	95.5	93.8	92.1	92.0	90.8	88.9	86.9
3	94.2	93.7	93.2	92.6	92.5	91.6	90.6	88.3	86.3	85.0	83.3	81.5	81.0
4	94.3	94.0	93.9	93.8	93.7	93.5	93.4	92.7	92.2	91.7	91.2	90.2	88.9
5	90.6	89.5	88.3	87.9	87.7	87.4	86.8	85.4	83.8	82.3	81.9	80.3	78.5
6	87.7	86.1	85.1	84.6	84.3	83.5	82.0	81.2	79.7	77.8	75.7	73.6	72.0
7	83.8	83.0	82.1	82.0	81.2	80.8	80.5	79.2	78.8	78.3	77.8	77.0	76.5
8	68.8	66.3	63.5	62.6	62.3	61.4	60.0	59.1	57.2	55.7	54.4	53.2	52.1
9	63.4	59.5	55.8	55.0	54.6	53.9	52.4	51.0	49.6	48.6	47.9	47.1	46.3
10	72.4	71.5	71.3	71.1	70.7	69.7	68.5	66.7	64.1	61.2	58.3	56.0	54.3
MILL B													
1	97.5	97.2	96.9	96.4	95.9	95.1	93.9	92.5	90.7	89.1	87.6	86.1	84.0
2	92.9	91.6	91.2	90.7	88.4	87.4	85.9	84.6	82.5	80.9	79.5	78.2	76.7
3	85.1	84.6	84.2	84.0	83.7	83.2	81.6	80.1	77.4	75.6	74.3	72.8	70.9
4	78.8	78.3	78.0	77.7	77.5	76.7	74.7	73.0	70.4	68.1	66.7	65.4	63.8
5	74.3	73.1	72.7	72.4	72.1	71.3	69.6	67.9	65.2	63.5	61.9	60.1	59.4
6	96.8	96.4	96.0	95.7	95.4	95.0	93.5	91.9	88.5	86.4	84.2	82.7	80.6

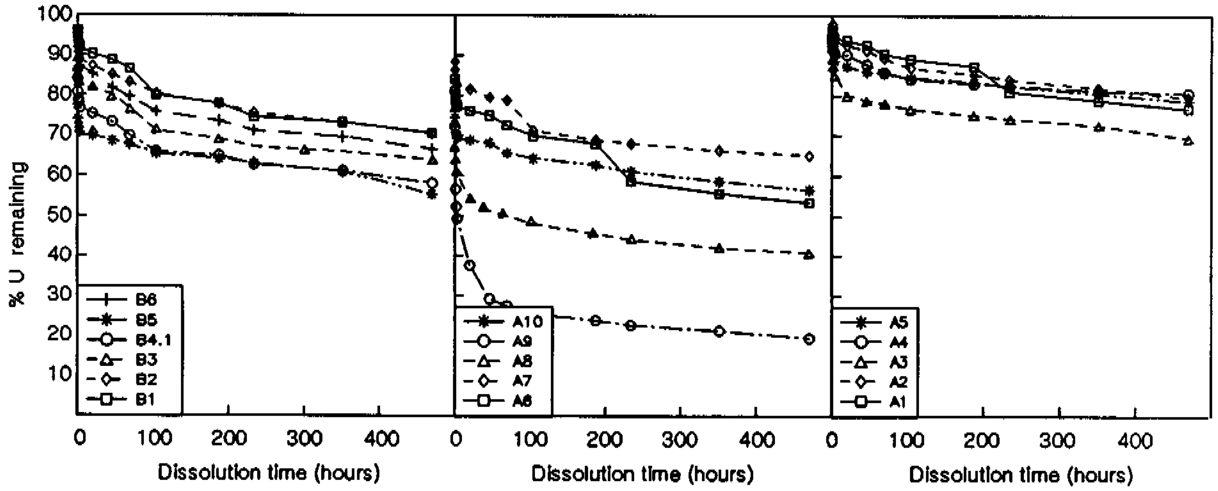
Table 5. Per cent uranium remaining after each extraction using de-ionised water

	Extract												
	1	2	3	4	5	6	7	8	9	10	11	12	13
MILL A													
1	97.2	96.9	96.6	96.3	96.1	95.3	94.7	92.2	90.5	89.4	88.5	87.9	86.3
2	98.7	98.3	97.7	97.4	97.2	96.9	96.3	95.9	95.6	95.2	94.7	94.4	93.9
3	97.0	96.4	96.2	96.0	95.4	95.2	94.9	94.5	94.2	93.9	93.5	93.1	92.6
4	94.1	93.9	93.8	92.8	92.7	92.3	91.7	91.3	90.9	90.4	90.0	89.3	88.4
5	94.1	93.7	93.4	92.7	92.6	92.4	92.0	91.7	91.3	90.9	90.5	90.0	89.0
6	85.5	84.6	84.2	83.9	83.5	82.4	81.0	80.0	78.2	76.9	76.1	75.4	74.2
7	83.5	83.0	82.5	82.3	82.0	81.2	80.4	78.9	78.0	77.2	76.0	75.1	73.9
8	68.1	67.7	67.1	66.8	66.4	65.7	64.6	63.6	64.4	61.1	59.4	58.6	57.3
9	45.6	44.1	43.8	43.6	43.5	43.1	42.9	42.5	41.7	41.3	40.9	40.5	39.9
10	72.4	71.8	71.6	71.7	71.0	70.3	69.4	68.7	67.8	66.3	65.6	64.7	62.3
MILL B													
1	97.3	97.0	96.7	96.4	96.0	95.7	95.2	94.9	94.4	93.9	93.5	92.7	91.1
2	93.3	92.8	92.5	92.1	91.7	91.3	90.7	90.3	89.8	89.4	89.1	88.4	88.0
3	85.4	83.9	83.6	83.0	82.6	82.1	81.6	81.1	80.5	80.0	79.5	78.8	78.3
4	79.2	78.1	77.5	77.0	76.7	76.1	75.4	74.8	74.1	73.3	72.7	71.4	70.7
5	73.1	72.6	72.4	72.2	72.0	71.5	71.0	70.5	70.0	69.6	69.1	68.6	68.4
6	96.6	96.3	96.0	95.7	95.5	95.1	94.5	94.0	93.5	92.9	92.3	91.2	90.4

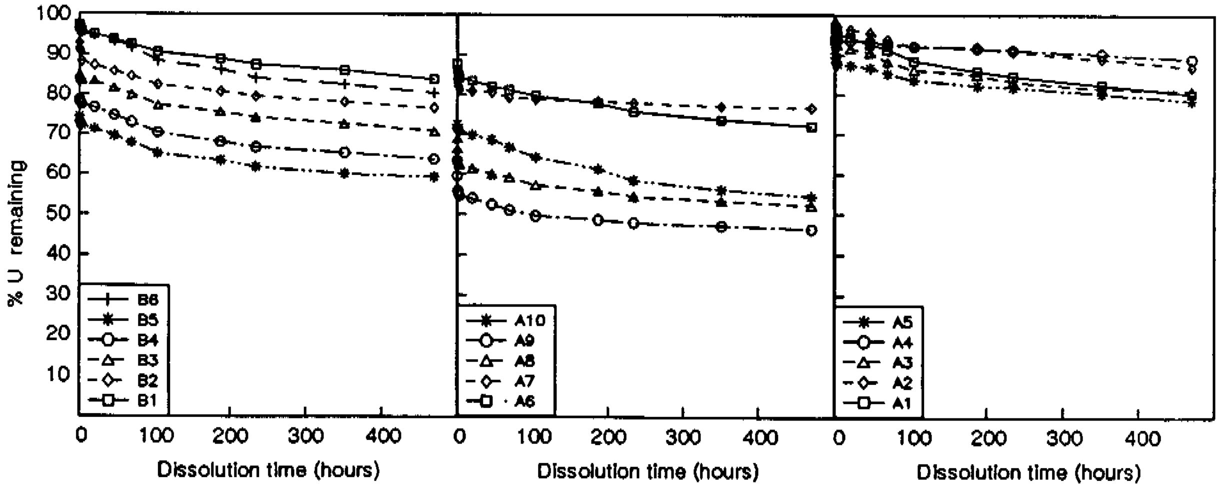
Table 6. Per cent uranium remaining after each extraction using 0.1 M hydrochloric acid

	Extract												
	1	2	3	4	5	6	7	8	9	10	11	12	13
MILL A													
1	90.3	86.5	83.1	79.7	77.1	54.4	29.6	16.0	1.7	1.0	1.0	1.0	1.0
2	95.7	93.7	87.7	84.0	82.4	56.0	32.7	18.6	5.8	3.8	2.8	2.0	1.7
3	83.9	80.6	76.8	74.0	71.6	50.7	31.4	17.7	2.3	1.0	1.0	1.0	1.0
4	91.9	89.6	86.5	84.1	82.5	62.6	46.3	35.8	18.1	13.7	10.6	7.6	5.8
5	91.6	89.3	86.4	83.5	81.3	60.8	44.6	33.1	15.3	9.8	6.0	2.8	1.0
6	78.3	74.5	71.4	68.6	66.5	51.7	34.4	25.0	12.8	6.1	3.8	2.4	1.0
7	78.2	75.9	73.7	71.7	69.8	53.9	39.3	40.6	16.5	6.2	4.7	3.5	1.0
8	59.2	56.4	54.0	51.6	49.9	36.8	25.7	19.3	18.4	11.9	9.6	7.7	5.0
9	64.8	61.4	59.1	58.0	57.3	48.3	45.9	41.5	37.3	36.4	35.8	35.4	32.0
10	67.5	64.8	62.5	60.2	58.7	45.0	29.6	21.5	10.1	3.5	1.1	1.0	1.0
MILL B													
1	91.1	88.6	82.3	78.0	74.7	51.4	21.4	6.3	1.0	1.0	1.0	1.0	1.0
2	86.0	82.1	74.6	70.5	70.1	44.1	15.2	9.5	4.1	3.1	2.8	2.6	2.4
3	81.9	78.6	73.8	70.3	67.5	45.7	20.1	7.8	1.0	1.0	1.0	1.0	1.0
4	74.6	70.2	69.2	66.1	63.5	40.2	15.5	5.4	1.0	1.0	1.0	1.0	1.0
5	70.3	67.2	62.0	59.4	57.5	32.8	16.9	7.4	1.0	1.0	1.0	1.0	1.0
6	87.9	83.9	77.7	73.5	70.0	43.4	14.2	3.2	1.0	1.0	1.0	1.0	1.0

(a) Simulated lung fluid



(b) Physiological saline



(c) De-ionised water

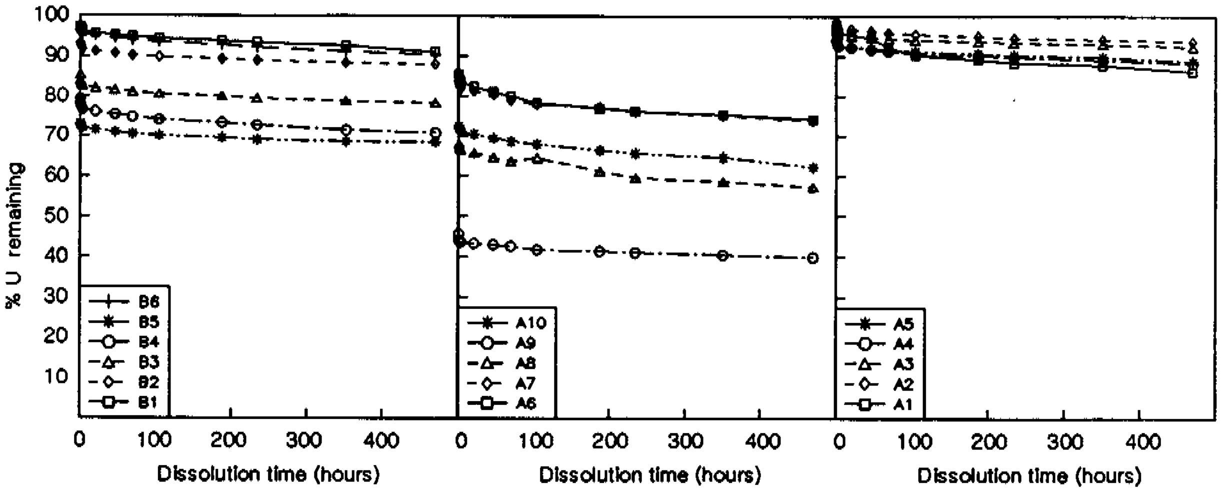


Figure 1. Per cent uranium remaining after each extraction versus dissolution time using: (a) simulated lung fluid; (b) physiological saline; and (c) de-ionised water

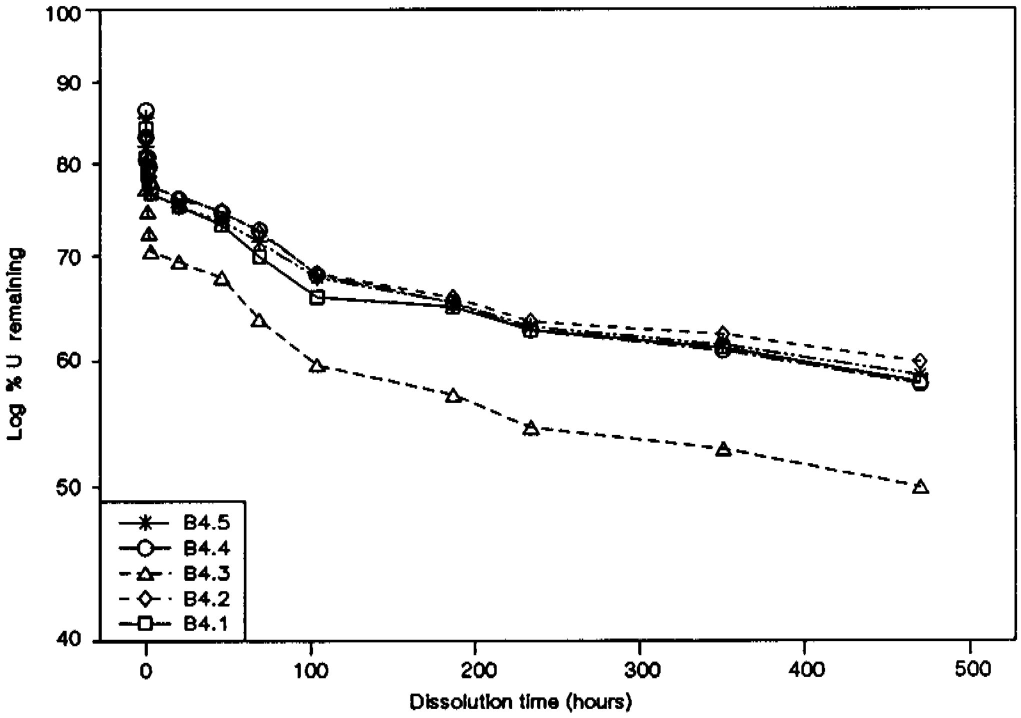
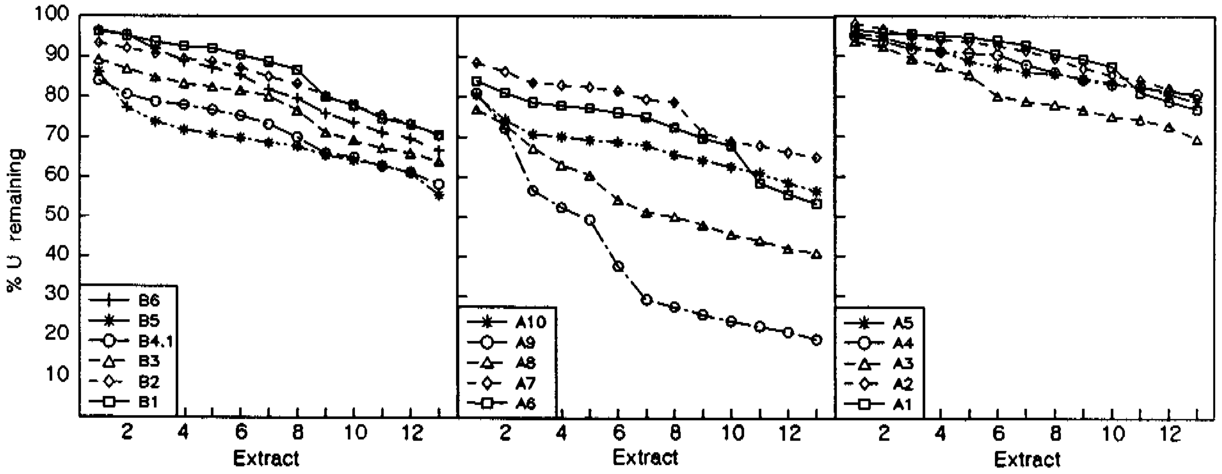
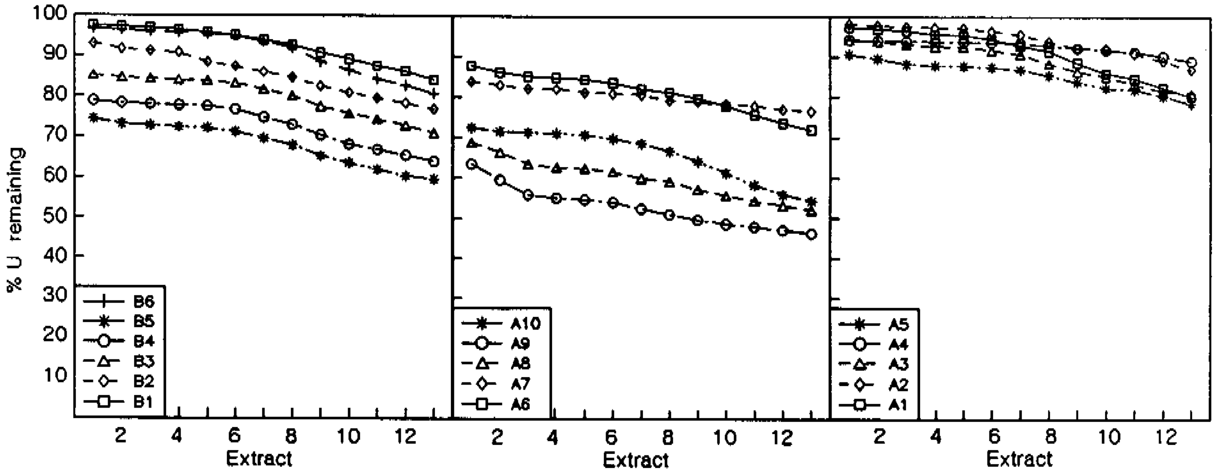


Figure 2. Repeat measurements for sample B4 in simulated lung fluid

(a) Simulated lung fluid



(b) Physiological saline



(c) De-ionised water

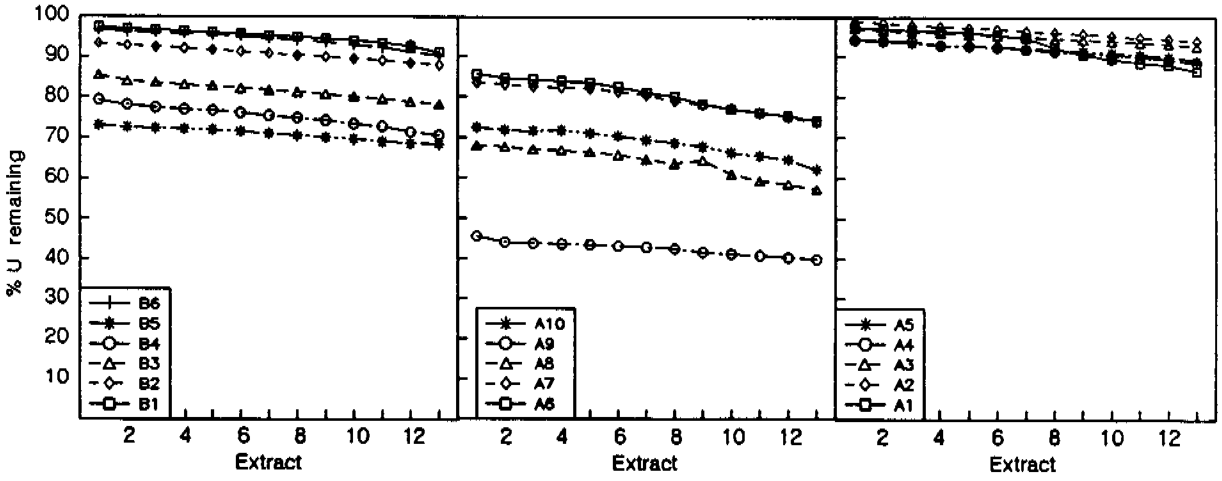


Figure 3. Per cent uranium remaining after each extration versus extract number using: (a) simulated lung fluid; (b) physiological saline; and (c) de-ionised water

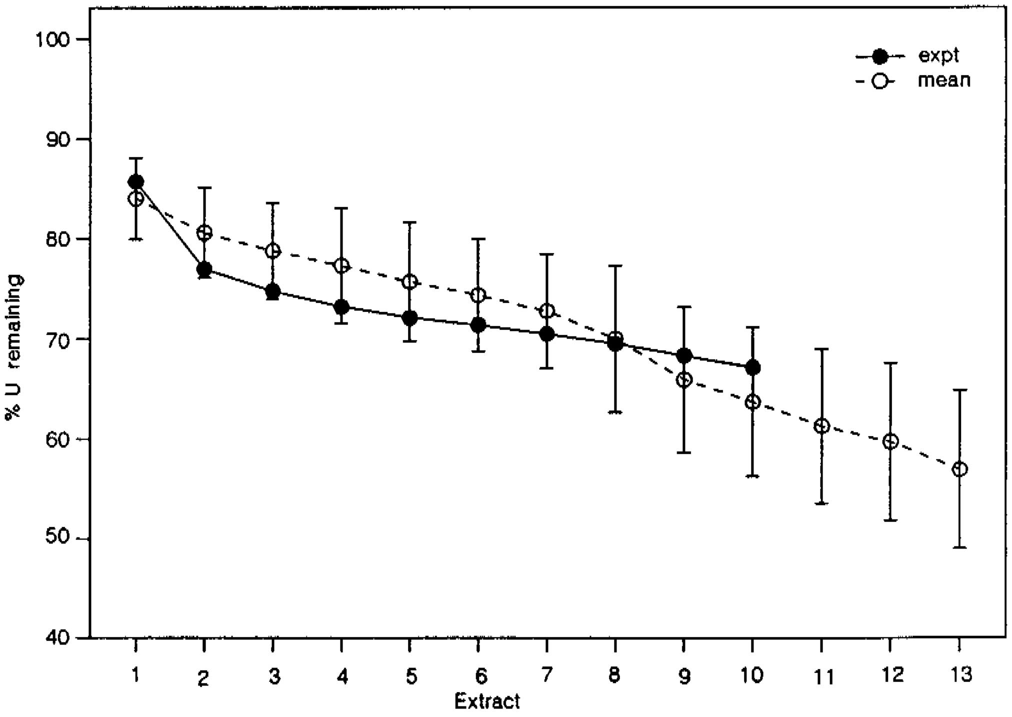


Figure 4. Comparison of short-term constant time interval experiment with mean results for sample B4 in simulated lung fluid

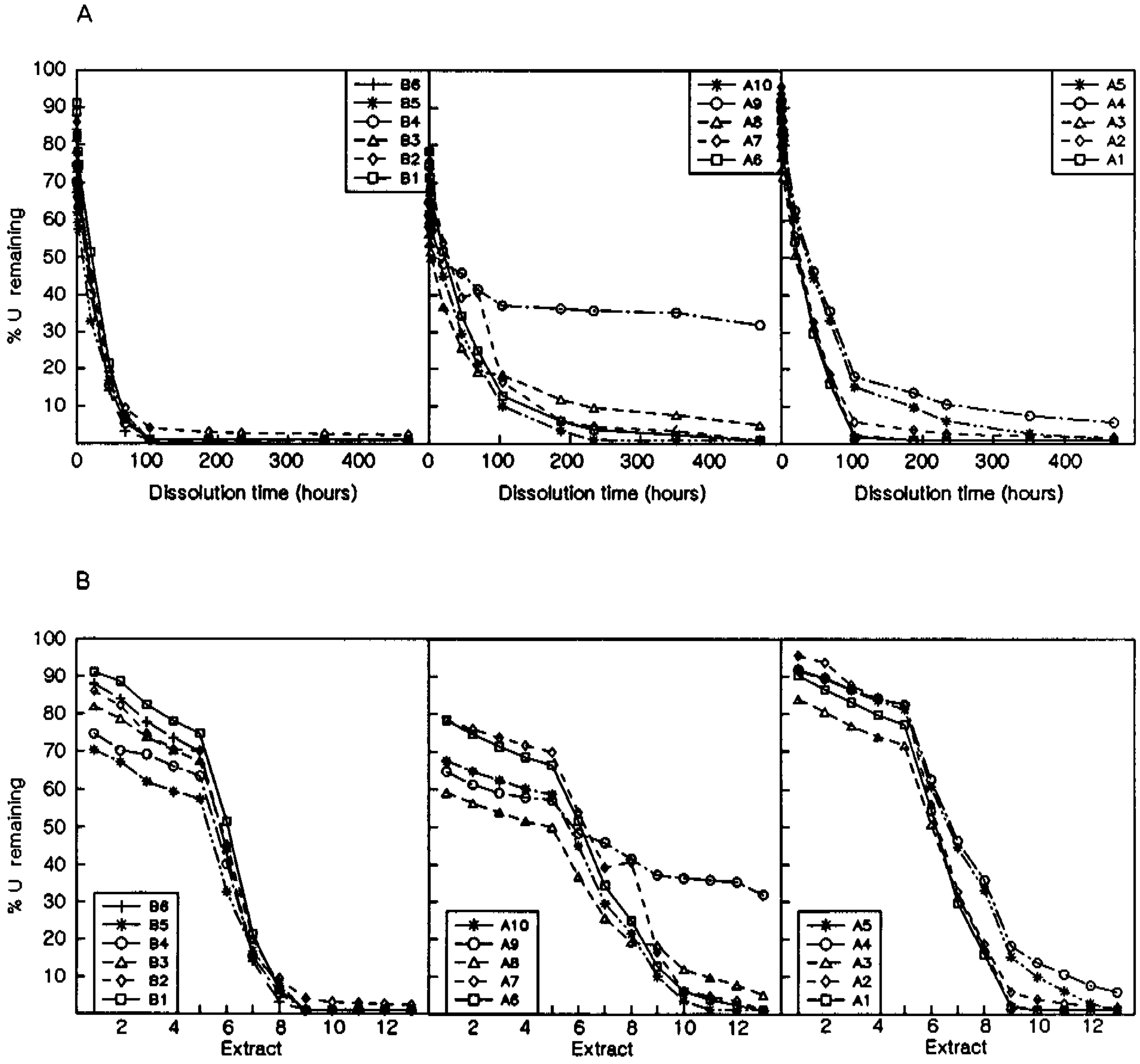


Figure 5. Per cent uranium remaining after each extraction in 0.1 M HCl verus: (a) dissolution time; and (b) extract number

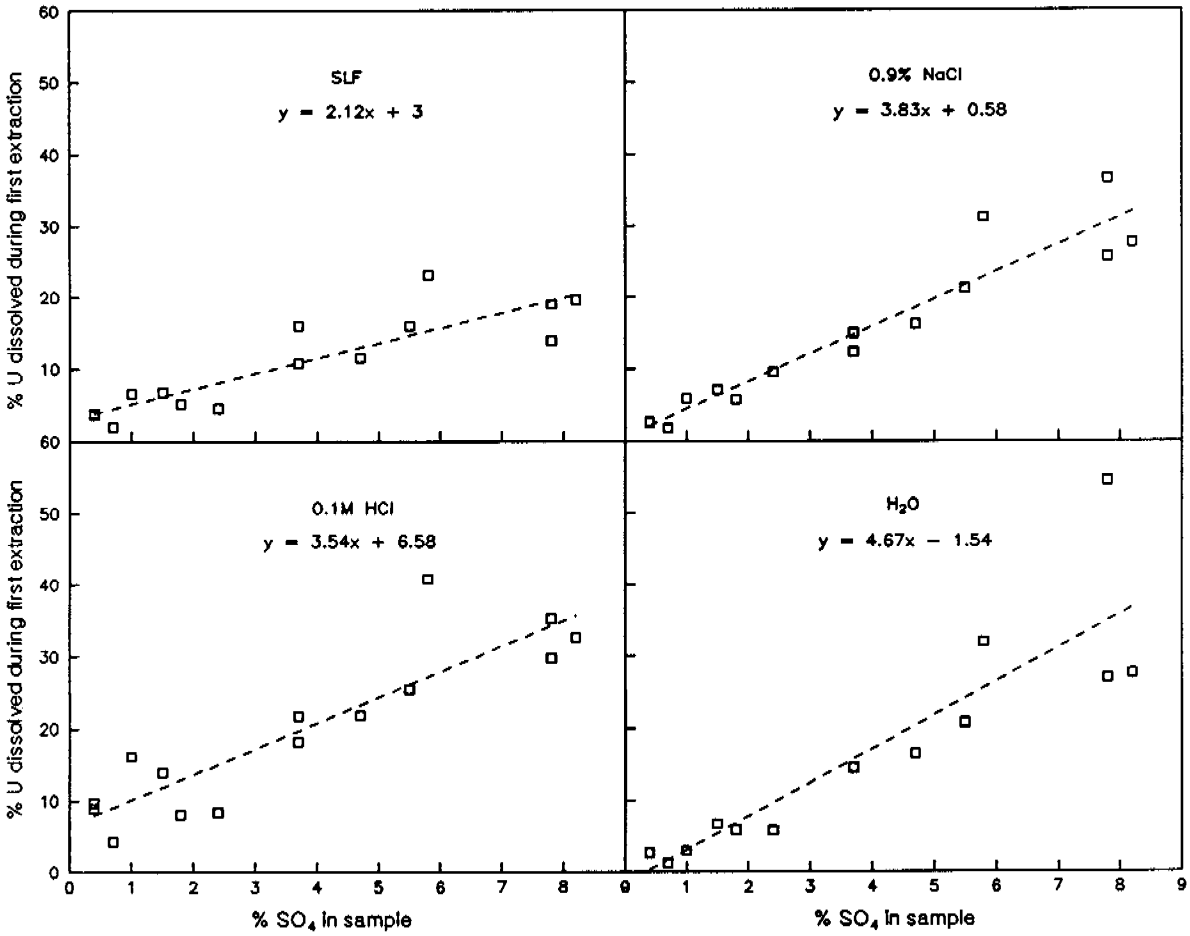


Figure 6. Linear correlation between per cent uranium dissolved during first extraction in each solvent with per cent sulphate content of each sample

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RESEARCH PUBLICATIONS

Alligator Rivers Region Research Institute Research Report 1983-84
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Alligator Rivers Region Research Institute Annual Research Summary 1987-88
Alligator Rivers Region Research Institute Annual Research Summary 1988-89 (in press)

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	*A limnological survey of the Alligator Rivers Region. II. Freshwater algae, exclusive of diatoms. 1986 (pb, mf - 176 pp.)	H.U. Ling & P.A. Tyler
RR 4	*Ecological studies on the freshwater fishes of the Alligator Rivers Region, Northern Territory. Volume I. Outline of the study, summary, conclusions and recommendations. 1986 (pb, mf - 63 pp.)	K.A. Bishop, S.A. Allen, D.A. Pollard & M.G. Cook
	Ecological studies on the freshwater fishes of the Alligator Rivers Region, Northern Territory. Volume II. Synecology. 1990	K.A. Bishop, S.A. Allen, D.A. Pollard & M.G. Cook
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