RECONSTRUCTION OF INTERNAL DOSES FOR THE ALPHA-RISK CASE-CONTROL STUDY OF LUNG CANCER AND LEUKAEMIA AMONG EUROPEAN NUCLEAR WORKERS

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ABSTRACT

The alpha-risk study required the reconstruction of doses to lung and red bone marrow for lung cancer and leukaemia cases and their matched controls from cohorts of nuclear workers in the UK, France and Belgium. The dosimetrists and epidemiologists agreed requirements regarding the bioassay data, biokinetic and dosimetric models and dose assessment software to be used and doses to be reported. The best values to use for uncertainties on the monitoring data, setting of exposure regimes and characteristics of the exposure material, including lung solubility, were the responsibility of the dosimetrist responsible for each cohort. Among 1721 subjects, the median absorbed dose to the lung from alpha radiations was 2.1 mGy, with a maximum dose of 316 mGy. The lung doses calculated reflect the higher levels of exposure seen among workers in the early years of the nuclear industry compared to today.

INTRODUCTION

Epidemiological studies investigating the risk of cancers following radiation exposure require the estimation of the individual doses to specific organs/tissues(s) for all study subjects in order to be able to better quantify the dose-risk relationship. While dose to organs from sources external to the body can be relatively easily and accurately estimated from personal dosimeter results (e.g. film badges, thermoluminescent dosemeters), it is much more difficult to ascertain organ doses from internal emitters for large groups of workers: they cannot be measured directly and have to be reconstructed retrospectively using indirect measurements (e.g. bioassay or air sampling) and mathematical models of the intake, distribution, excretion and retention of materials. For this reason, the majority of large scale cohort studies of nuclear industry workers published to date [(1-3)] have focused on estimating health impact of external photons exclusively.

However, workers employed in some facilities – particularly facilities involved in the fuel cycle – are potentially exposed to internal radiation from a number of radionuclides, in particular uranium and plutonium. These groups of nuclear industry workers are of interest because they allow the *direct* study of health effects of internal exposure to radiation, a priority topic in radiation research today [(4, 5)]. At present, little information is available on the long-term health effects in populations exposed to plutonium (Pu) and uranium (U) isotopes.

For this reason, and to overcome the difficulties of estimating individual doses for tens of thousands of workers, a matched case-control study design was chosen, nested within much larger cohorts of workers (Table 1) to assess the risk of lung cancer and of leukaemia following exposure to Pu and U [(6)]. Cases of lung cancer and leukaemia were identified and controls selected, matched to individual cases on the basis of similar characteristics (age, sex, facility of work). Lung cancer and leukaemia were chosen as the cancers of interest as they arise in the tissues (lung and red bone marrow) with among the highest absorbed doses following intakes of Pu and U. The choice of a nested case-control study design, compared to a full cohort study design considerably reduces the number of

subjects for whom detailed historical monitoring and occupational data must be collected and doses reconstructed and allows the collection of individual data on potential confounders, in particular smoking and, where available, other occupational carcinogens. This study was part of the European Commission funded Alpha-Risk project looking at risks from internal emitters [7], which also included epidemiological studies of uranium miners.

In order to provide the realistic best estimates of dose that were required for the epidemiological study and also to ensure that the assessments for workers at different establishments were comparable, it was necessary to develop a common dosimetry protocol for the project. The protocol was developed by the dosimetrists responsible for each cohort along with some of the epidemiologists involved in the project. A novel method for assessing uncertainties in doses was also trialled as part of the project, although this is the subject of a separate publication [8] and hence is not reported in detail here.

DATA PREPARATION

Collection of subject-specific bioassay and work history data – bioassay

At the outset it was decided that the dose assessments would be based on urine data as such measurements had been used as a form of individual monitoring for plutonium and uranium exposures since the start of work in the facilities. The number of urine measurements available per subject was very variable, reflecting differences in radiation employment durations, involvement in incidents between subjects and monitoring procedures at different sites (Table 2). A small number of faecal and, in the case of the CEA-COGEMA cohort, in-vivo (lung) monitoring results were also used; these measurement data were generally only available post-incident although at AWE a campaign of faecal monitoring was conducted for reassurance purposes among workers in the late 1970s. Air sampling data were not directly used in the dose assessments, however they could provide information to identify incidents or periods of chronic exposure.

The amount of work required to prepare the bioassay and potential exposure history data (work periods, incidents) for use in the dose assessment software varied widely between the cohorts. At AWE and BNFL, the bioassay data were available in electronic form. Information on potential exposure history at AWE had to be assembled by other means (e.g. extracted from paper records). Exposure information for the most complex plutonium exposure cases (e.g. those involving acute exposures) within the BNFL cohort was available in electronic form, other exposures histories were reconstructed from electronic records of monitoring and employment dates [9]. At UKAEA bioassay data from 1986 onward were available in electronic form but earlier measurement data had to be extracted from paper files; incident data, where available, were in paper format. For SCK•CEN/BN, the data were extracted from electronic and paper files and for CEA-COGEMA they were extracted from paper medical files.

There was monitoring data available for alpha-emitting nuclides other than Pu and U (^{210}Po, ^{226}Ra, ^{228}Th, ^{227}Ac, ^{242}Cm) among a small proportion of the UKAEA, AWE and SCK•CEN/BN cohorts and this was included in the study. In addition to data pertaining to exposure from internal exposure to radionuclides, doses from external exposure to radiation were assembled from the radiation records for these workers. External occupational radiation doses were generally readily available as they had been used in previous epidemiological studies [1]. Data on smoking and on X-ray examinations made

as part of medical examinations at work were extracted from occupational medical records. Non-occupational (environmental) radiation exposures, including radon which is of particular interest for lung cancer, were not included in the study. However, except for CEA-COGEMA where cases and controls were matched on geographical location (North or South of France), cases and controls were matched on working at the same facility and so there are unlikely to be large systematic differences in environmental exposures between cases and controls.

Reliability of the monitoring data

At all establishments, early techniques for plutonium in urine analysis involved chemical separation of the plutonium followed by gross alpha counting. Alpha and mass spectrometry was introduced at some facilities in later years. Fluorimetry has been used as a measurement for depleted and natural uranium in urine for many years and was still being used at the Springfields site at the end of the study period under consideration. Exposures to enriched uranium at AWE have been monitored using radiochemistry and gross alpha counting, delayed neutron activation and radiochemistry and alpha spectrometry techniques. The changes in techniques used for the analysis of plutonium and uranium in urine have generally resulted in improvements in specificity, sensitivity and accuracy for the isotopes of interest [(10)]. This can be demonstrated by looking at the measurements for an individual who has been monitored for a nuclide for many years: noticeable step changes in the profile are observed that are linked to changes in the analytical techniques used rather than the level of exposure. In addition, there is other evidence to suggest that measurements made in the early years may be unreliable. For example, at BNFL cross-contamination of plutonium samples occurred through the reuse of sample containers prior to 1971 [(9)]. At AWE the dry ashing of samples and the reuse of the platinum trays used in activity measurements in the 1950s may also have caused some samples to be cross-contaminated. Therefore, the dosimetrists defined a cut-off date, t_1, for each cohort and for Pu and U nuclides before which the monitoring data were deemed too unreliable to be used in the absence of any data after this date (Table 3). If a subject only had pre-t_1 data the epidemiologists were informed so that the subject could be excluded from the analysis.

Uncertainties in the monitoring data

Uncertainties in measurement data are an important input into the dose assessment process. In general it was decided to follow the approach recommended in the IDEAS guidelines [(11)], with a scattering factor (SF) representing the overall Type A (due to counting statistics) and Type B (all other sources) uncertainties on a measurement. With this approach the overall uncertainty on a measurement is often dominated by Type B uncertainties (mainly biological variability in excretion for urine monitoring) which are log-normally distributed and the SF is the geometric standard deviation of the distribution. However, AWE had actual values for some results below the detection limit: the errors on these are not adequately represented by a lognormal distribution as counting statistics, which are best represented by a Poisson distribution but which can often be approximated using a normal distribution, dominate at these levels. They thus took the approach of determining the value up to which normal errors would be likely to dominate over lognormal errors, then applying a SF if the result was above this value or assigning a Type A (normal) error below this.

The dosimetrist for each cohort determined appropriate SF values for their measurements using their knowledge of the relevant measurement techniques (Table 4). The SF for UKAEA

measurements was determined by examining the dispersion of measurement values about an average value from workers who were excreting activity at a more-or-less constant rate.

As discussed under reliability of the data above, in addition to the random uncertainties associated with the measurement results, it was clear that, in the UKAEA Harwell, BNFL and AWE cohorts, early urine measurements tended to consistently overestimate the excreted activity. To correct for this, the earliest measurement results for total alpha activity, total Pu activity and ^{239}Pu activity for UKAEA Harwell workers were divided by factors of 5, 6 and 2 respectively. These factors were obtained empirically by studying the urine records for workers who had been monitored over many years. In such cases the apparent change in urinary excretion rate that occurred when urine analysis methods changed gave clues as to the overestimates introduced by the earlier, less satisfactory methods. The intakes for BNFL plutonium workers were divided by a factor of 3, prior to the calculation of doses, if only pre-1971 routine measurement data were available [(9)]. At AWE, no systematic corrections were made to the data but high SF values were chosen to reflect the uncertainties in this early data.

Facility-specific exposure data (AMAD, material solubility, fingerprints)

For UKAEA Harwell, BNFL Sellafield and the CEA-COGEMA cohorts there was some information [(12-14)] to support the use of an activity median aerodynamic diameter (AMAD) of 5 µm. For the other cohorts and facilities there was no specific information available on particle sizes, so the ICRP default AMAD of 5 µm was chosen on the basis that this was the most likely particle size in workplaces [(15)]. Although dose per unit intakes values can vary significantly dependent upon AMAD, doses per unit urine excretion, as used in this study, are less affected by AMAD. This is because the effect of AMAD on assessed intake per unit urine excretion tends to counteract the effects of AMAD on dose per unit intake. For example, for plutonium oxide, the effective dose per unit intake for a 1 µm AMAD is about 80% higher than the value for a 5 µm AMAD, but effective and lung dose per unit bioassay values only increase by about 13 and 42% respectively. Compared to uncertainties arising from knowledge of lung solubility and intake regimes, AMAD was felt to be a relatively minor source of error [(16)].

The lung solubility of the exposure material is extremely important when assessing lung doses on the basis of urine data. As individual measurement records did not contain information on the exposure material, this had to be based on information available on the materials used or known to be present in the workplaces (buildings) in which the individual had worked. Solubility of materials in the lung is described using the parameters, methodology and terminology presented in the ICRP publication 66 Human Respiratory Tract Model (HRTM) [(17)]. The lung solubility parameter values used (Table 5) were derived by assigning the material found in the workplace to the appropriate HRTM default solubility 'Type' [(17)] or from experimental evidence [(18)] or by re-evaluating historical intake assessments to obtain a best fit mixture of default solubility Types which were then translated into specific HRTM absorption parameters. While some workplaces had a limited range of exposure materials from routine processes that could be assigned to a single solubility type, at other workplaces, for example research laboratories where a large range of materials had been used, this was not the case. Also, some workers, for example maintenance staff, can work in a range of buildings over a working period and the individual could have potentially been exposed to a range of materials with different solubility characteristics (Table 5). The bioassay assessment software used (see below) did not have the facility to directly handle exposures in a single period that were to a set

mixture of materials each with their own solubility parameters, so a single set of parameters had to be derived to represent these mixtures. Calculations on idealized datasets showed that an exposure to 1:1:1 mixture of Types F, M and S uranium could be reasonably represented by setting the HRTM solubility parameter values to 0.36, 100 and 1.7×10^{-3} for f_r, s_r and s_s respectively.

The radionuclides detected in the bioassay sample will depend upon the preparative technique, analytical endpoints and measurement technique used. As well as the radionuclides measured in the sample, there may be other nuclides present in the exposure material that will need to be accounted for in the intake and dose assessment. For plutonium, while total plutonium α activity is generally measured and reported, activity from the associated β emitter ^{241}Pu and its more radiologically significant, α-emitting ^{241}Am decay product must also be taken into account. The relative abundance of ^{241}Pu in the exposure material was accounted for in the UKAEA, BNFL, CEA-COGEMA and SCK•CEN/BN cohorts. At AWE the radionuclide fingerprint of the exposure material could not be determined for the subjects so the relative abundance of ^{241}Pu was not included. However, initial calculations suggested that these isotopes would not make a major contribution to the dose (see also Reporting of internal doses below)

Uranium activity determined in urine samples could arise from both occupational and dietary intakes. Background levels of 4.77, 1.13, 0.26 and 0.86 $mBqd^{-1}$ were subtracted from results for the BNFL [19] and UKAEA Winfrith, Harwell and Dounreay sites respectively to account for this. Among the other cohorts no background subtraction was made because the background levels among workers at their establishments were low compared to the reporting level and/or the detection limits of the analytical techniques used.

CALCULATION OF INTAKES AND DOSES

Dose assessment software used

It was decided at an early stage to use software based on the Integrated Modules for Bioassay Analysis (IMBA) for this project [20]. All the dose calculations except those for the BNFL cohort used a version of IMBA Professional Plus that had been made available for the project. For the BNFL cohort bespoke software based on the IMBA modules called PUMASS and UMASS was used for plutonium and uranium assessments respectively [21]. This software had been specifically designed to enable the automated assessment of the large number of subjects in the BNFL cohort and had been previously used for this purpose [22].

Biokinetic and dosimetric models used

Following a review of the existing and proposed metabolic models for plutonium, it was judged that a revision of the ICRP publication 67 model [23] developed by Leggett [24] represented best current knowledge and thus was implemented in IMBA. A review of information on uranium metabolism concluded that the model recommended in ICRP publication 69 [25] still represented best current knowledge and should be used for this study.

It was recognised that as lung cancer was the primary outcome of interest in the epidemiological analysis, lung dosimetry would be particularly important. Consideration was given to updating the HRTM [17] to reflect the outcomes of the latest research in this area [26]. However, it was decided that, at that time, the evidence for change was limited and that it would be prudent to continue using the

standard HRTM to generate the point estimates of lung dose. Transport through the gastrointestinal tract was based on the ICRP 30 model [27]. Doses were calculated using ICRP publication 23 [28] reference organ/tissue masses and radionuclide transformation data from ICRP publication 38 [29].

Dose assessment protocol

The dosimetrists were initially blind to the case/control status of the study participants. The reference date for a case and his/her controls was set by the epidemiologists to the date of the cancer. However, as bioassay data after the reference date were excluded to ensure that the dose assessments for controls were not biased by the availability of more accurate bioassay results compared to the cases, the dosimetrists did know that some of the subjects were controls.

Chronic intakes were assigned to any period of a worker's career that involved a potential risk of internal exposure by plutonium and/or uranium. Just one chronic intake was assigned if a worker did the same job in the same facility throughout his career, but a sequence of chronic intakes would be assigned for a subject with a more complex work-history. The start and end dates of chronic intakes were determined from records of work history for the UKAEA and AWE cohorts and from exposure files for the CEA-COGEMA cohort. Where these data were not available or did not align with the monitoring data, for UKAEA the chronic intake was set to half a monitoring interval before or after the start and end dates of monitoring, respectively; for AWE start dates were based on start of employment and end dates were set to 1 month after the last sample. By default, for BNFL workers, chronic exposure periods were started 1 month and 6 months prior to the first sample for U and Pu respectively, as these were the usual monitoring intervals for these nuclides, however these dates were adjusted to take into account known periods of employment or other relevant information.

Evidence for acute intakes came from reports of incidents in which the worker was involved, from air-sampling data, from nose-blow results and from special monitoring (e.g. in-vivo, faecal results). When a single intake is assigned in IMBA, but is not supported by any data, then IMBA calculates a value of zero (or near zero) for that intake and no significant error arises. More serious is the error of failing to assign an acute intake date when one really occurred, because this can lead to systematic errors in the intake calculation. For this reason, precautionary assumptions toward the allocation of potential acute intakes were made (within the current limitation of 10 intake regimes allowed by IMBA). As discussed earlier, the lung solubility of the exposure material for each intake could be set independently based on facility information and/or incident files.

IMBA uses a maximum likelihood method to provide an estimate of the intake(s) based on the best fit between the observed bioassay data and that predicted from the entered intake regimes. However, this method cannot calculate a dose for a worker if all their measurement results are below the detection/reporting level. At AWE, subjects with only below reporting level data (where the reporting level is greater than zero) were rejected from the study. Very few subjects came into this category because, from about the mid-1960s onward, the actual measurement result was reported for all positive results. At BNFL, the last measurement result in the exposure period was set as positive at the daily excretion normalised limit of detection and chronic intake was assumed over the period. For UKAEA and CEA-COGEMA workers who had only less than detection/reporting level data, the Bayesian fitting tool in IMBA was used to provide a central estimate of the intake by extracting the median from the posterior probability distribution.

Reporting of internal doses

Because the objective of the epidemiological study was originally to assess the risks of lung cancer and leukaemia, the organs for which annual doses needed to be reconstructed were the lung and red bone marrow. The doses were reported as absorbed dose (in terms of Gy) so that no assumptions were made regarding the relative biological effectiveness of the radiation, as this was one of the issues the epidemiologists wished to study directly. These doses were calculated from the intake estimates using IMBA. The radiation weighting factors for α, β and X-ray/γ radiations were set to 1, 0 and 0 respectively, so that only absorbed dose from α radiations would be calculated. The dose from β/γ emitters such as ^{241}Pu and the β/γ component of α-emitting isotopes was not calculated as the relatively small contribution they would make to dose did not justify the significant increase in assessment run times that would be required for their calculation. The α radiation dose from ^{241}Am ingrowing from ^{241}Pu in the exposure material was included in the Pu dose for the UKAEA, BNFL, CEA-COGEMA and SCK•CEN/BN cohorts.

The individual alpha doses to the bronchial, bronchiolar, alveolar-interstitial regions and thoracic lymph nodes, and red bone marrow, were provided to the epidemiologists. These were reported as annual absorbed doses to each tissue/region for each nuclide. For the main epidemiological analysis, the dose to the lung was calculated as the arithmetic mean of the doses to the bronchial, bronchiolar and alveolar-interstitial regions – the contribution from thoracic lymph nodes was not included. Sensitivity analyses were also conducted calculating the lung dose as the sum of the doses to each region taking into account the proportion of the total lung mass each region represents. For this calculation, the doses to the bronchial, bronchiolar and alveolar-interstitial regions were multiplied by 0.0006, 0.0017 and 0.9977 respectively.

DISCUSSION

Sets of annual absorbed doses from alpha radiations to lung and red bone marrow were provided to the epidemiologists on the basis of the above protocol. The greatest number of assessments was required for the BNFL cohort (Table 6). For some of the cohorts, a considerable number of the subjects were monitored and had assessments for both plutonium and uranium. A small number of assessments were required for nuclides (^{210}Po, ^{226}Ra, ^{228}Th, ^{227}Ac, ^{242}Cm) other than those of plutonium, along with associated americium, and uranium. These assessments were carried out in a similar manner to those for plutonium and uranium.

During the course of the study it became apparent that there were insufficient leukemia cases within the overall study cohort to undertake a meaningful epidemiological analysis for this outcome, so the red bone marrow doses supplied were not used. The red bone marrow doses are not considered further here as they were not available for review.

The median individual lung dose for AWE is somewhat higher than those calculated for the other cohorts (Table 6). However, as the intake data were not centrally collected and reported as part of the study, it is not possible to tell to what degree this reflects differences in the level of exposure as opposed to assumptions made regarding the lung solubility of the material between cohorts. Applying a w_R value of 20 to the median doses gives equivalent doses to the lung from all alpha emissions in the range of about 5 to 230 mSv among the groups. As the doses for other organs are not known, the total effective dose cannot be calculated. However, the lung component of effective

dose varies from about 0.5 to 27 mSv. It should be noted that the doses have not been calculated over a 50 year period but to the reference date for the case or control and thus strictly speaking these are not equivalent or effective doses. However, these doses do not seem unrealistic given the levels of exposure that might have occurred in the early years (1950s to 1970s) covered by this study.

The assessments produced, being point estimates, provide no information on the uncertainties associated with them. The association between dose to lung produced by the dosimetrists and risk of mortality from lung cancer is analyzed in the associated epidemiology paper [6].

Some issues that have been noted in the course of preparing these assessments and that may warrant further attention in future dosimetric protocols for epidemiological studies are:

a) Many different sets of solubility parameter values were used by the dosimetrists. The justification for the use of such a large number of different values is probably not strong and it may be better to have a more restricted set.
b) Even though a common dosimetry protocol was used, there was scope for different approaches to be taken by the assessors that could impact on doses produced for an individual cohort. For example, the changes to hypotheses and parameter values made to obtain an acceptable fit between observed and expected monitoring results in the assessment process was not defined between different assessors.
c) Inputs to the individual dose assessments, such as material solubility, intakes, number and type of bioassay measurements, were not provided to the epidemiologists along with the dose information. With hindsight, if this data had been provided it could have been included in sensitivity analyses of the study results.

In addition, since the doses for this study were calculated, ICRP has now accepted the proposed modification to the lung model [30], which could affect the dose-risk relationships reported in the epidemiological analysis [6].

Some of these issues were addressed in a follow-up project to design a protocol for dose assessment for a putative European cohort study of uranium workers and miners [31].

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Table 1 Cohorts participating in the α-risk case-control study of lung cancer and leukaemia

Cohort	Organisation responsible for cohort dosimetry	Locations	Activities at location
UK Atomic Energy Authority (UKAEA)	Nuvia	Harwell, UK	Nuclear research
		Dounreay, UK	Fast breeder reactor
		Winfrith, UK	Heavy water reactor
Atomic Weapons Establishment (AWE)	AWE	Aldermaston and Burghfield, UK	Nuclear weapons research and production
British Nuclear Fuels Ltd (BNFL)	Public Health England (PHE) - took over responsibility from Westlakes Scientific Consulting (WSC) after it closed in 2010	Sellafield, UK	Nuclear fuel reprocessing
		Springfields, UK	Nuclear fuel production
Commissariat à l'Energie Atomique (CEA) - Compagnie de Gestion des Matière Nucléaires (COGEMA) now AREVA NC	Institut de Radioprotection et de Sûreté Nucléaire (IRSN)	Fontenay-aux-Roses and Saclay, France	Nuclear research
		La Hague, France	Fuel reprocessing
		Pierrelatte, Marcoule, Grenoble, Cadarache, France	Mixed activities (research, production and/or reprocessing)
Studiecentrum voor Kernenergie . Centre d'Etude de l'Energie Nucléaire (SCK•CEN), Belgonucléaire (BN)	SCK•CEN	Mol, Belgium	Nuclear research
		Dessel, Belgium	Nuclear fuel production(MOX)

Table 2 Bioassay results available in the study

Cohort/facility		Descriptive statistics on urine monitoring of the cohorts									
		Plutonium					Uranium				
	Total no. of subjects[#]	No. of subjects monitored	Number of urine samples per subject			No. with all <LOD results	No. of subjects monitored	Number of samples per subject			No. with all <LOD results
			Mean	Median	Min, max			Mean	Median	Min, max	
UKAEA Harwell	91	76	32.6	22	1, 200	14	47	20.4	8	1, 200	12
Dounreay	106	89	7.8	5	1, 53	31	90	7.9	5	1, 48	45
Winfrith	44	33	10.6	8	1, 43	1	43	11.5	11	1, 36	2
Total	241	198					180				
AWE	582	298	30.7	22	0*, 189	0	284	45.2	35	0, 191	0
BNFL Sellafield	561	369	79.0	43	1, 572	69	162	21.7	8	1, 181	64
Springfields	1064	0	NA	NA	NA	NA	1064	127.1	54	1, 1007	83
Total	1625	369					1226				
CEA-COGEMA	63	43	18.6	14	1, 53	32	55	19.5	10	1, 120	41
SCK•CEN/BN Mol	Data not available										
Dessel	Data not available										

[#] = number of cases and controls; NA = not applicable; * = with a urine result that was subsequently found to be ineligible.

Table 3 Date before which measurement data are deemed to unreliable to be used on their own

Cohort/facility		Year of start of facility activities	Year from which the measurement techniques are deemed 'reliable'	
			Plutonium	Uranium
UKAEA	Harwell	1946	1970	1946
	Dounreay	1955	1970	1955
	Winfrith	1957	1970	1957
AWE		1952	1963	1960
BNFL	Sellafield	1948	1971	1948
	Springfields	1946	Not applicable	1948
CEA-COGEMA		1946	1967	1967
SCK•CEN/BN	Mol	1953	1960	1960
	Dessel	1957	1960	1960

Table 4 Scattering factors (SF) for measurement results

Cohort	Measurement type	Range of SF values used*	
		Plutonium	Uranium
UKAEA	Urine	1.2 – 2.2	1.7 – 2.2
AWE	Urine	2.1 – 3.0	1.8 – 3.0
BNFL	Urine	1.8	1.8
CEA-COGEMA	Urine	2.0	2.0
SCK•CEN	Urine	1.8	1.8
All	Faeces	3.0	3.0

*Where a range of SF values are given, this indicates that SF value varies according to the technique in use at the time.

Table 5 Lung absorption parameter values used in the assessments

			f_r	s_r	s_s (x 10^{-4})	No. of assessments[#]
ICRP66 Type F			1	100	-	-
ICRP66 Type M			0.1	100	50	-
ICRP66 Type S			0.001	100	1	-
1:1:1 Type S/M/F			0.36	100	17	-
Plutonium compounds						
UKAEA	Harwell (insoluble)*		0.005	100	1.5	7
	Harwell (intermediate)*		0.07	100	2.0	7
	Harwell (soluble)*		0.13	100	2.5	36
	Harwell site average		0.060	100	2.8	25
	Dounreay (insol)		ICRP66 Type S	See above		24
	Dounreay (more sol)		0.05	100	20	77
	Winfrith		0.007	100	1.5	32
AWE	(insoluble)		0.001	2	1.0	276
	(soluble)		0.2	2	20	81
BNFL	(nitrate[+])		0.28	49	58	368
	(oxide)		ICRP66 Type S			9
CEA-COGEMA		MOX material	ICRP66 Type S	See above		16
		Radiochem labs	0.05	100	20	27
SCK•CEN/BN		Mol (MOX)	ICRP66 Type S	See above		
		Dessel (MOX)	ICRP66 Type S	See above		
Uranium compounds						
UKAEA	Harwell 1 building		0.42	100	20	11
	Dounreay (insol)		0.05	100	20	15
	All other exposures		1:1:1 Type S/M/F	See above		157
AWE	All exposures		0.03	1	5	282
BNFL	All exposures		1:1:1 Type S,M,F	See above		1226
CEA-COGEMA		UF_6	ICRP66 Type F	See above		8
		Mixture/not known	ICRP66 Type M	See above		49
SCK•CEN	(insoluble)		ICRP66 Type S	See above		
	(soluble)		ICRP66 Type F	See above		

[#] The number of assessments which had least 1 intake regime with this absorption type. Subjects may have exposure to different absorption types in different regimes. See Table 1 for numbers of subjects with monitoring for a nuclide.

* a range of sets of parameter values were used dependent upon the building, examples are shown.

[+] The HRTM bound state was also used for this material the parameters values used were f_b 0.57 and s_b 0.21

Table 6 Assessed dose to the lung from alpha radiations*

		UKAEA	AWE	BNFL	CEA-COGEMA	SCK•CEN/BN	All Cohorts
	Number of assessments	138	213	318	31	11	711
Plutonium	Median lung dose (mGy)	1.0	6.2	0.8	3.6	0.1	1.3
	Max lung dose (mGy)	98.7	110.4	43.2	97.6	4.7	110.4
	Number of assessments	122	197	1040	44	6	1409
Uranium	Median lung dose (mGy)	0.2	3.5	2.3	0.2	3.5	2.2
	Max lung dose (mGy)	301.5	104.9	87.4	97.6	6.3	301.5
	Number of assessments	8	47	-	-	1	56
Other alpha	Median lung dose (mGy)	0.4	1.5	-	-	0	0.5
emitters	Max lung dose (mGy)	1.1	308.1	-	-	0	308.1
	Number of assessments	175	232	1246	50	18	1721
Total alpha	Median lung dose (mGy)	1.21	11.3	2.1	2.9	0.2	2.4
	Max lung dose (mGy)	304.9	315.6	87.4	97.8	6.3	315.6

* This is the committed dose to the date of death for cases and reference date for controls.