European ALARA Network Issue 8 - May 2000

European ALARA Newsletter

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Editorial

The Network is now entering its fifth year of life and is still in a growing phase with new countries participating to its Steering Committee : Finland at the end of 1999, Denmark in 2000.

From the beginning, the objective of the Network has been to ensure a better protection of the workers from ionising radiations both in the nuclear field and in a wide range of industrial and research applications where ionising radiation and natural or man-made radionuclides are used extensively. The primary mean of achieving that objective, has been through two issues of the European ALARA Newsletter and one ALARA Workshop each year.

The Newsletter is considered by many participants as a very useful way of sharing experience in different countries, both in the nuclear and non-nuclear fields. The dissemination of good ALARA practices as well as practical examples of lessons learned from radiological incidents is particularly appreciated, and the number of individuals or institutions asking to be put on the mailing list is increasing. This, the eight issue of the European ALARA Newsletter, provides a synthesis of the 3rd Workshop on “Managing Internal Exposure”, and we plan that the next issue will give an overview of the new regulations concerning radiological protection in Europe.

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Editorial Board

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The first two Workshops gave rise to nearly 20 recommendations to the Commission, the regulatory bodies and the other involved parties. Stemming from these recommendations, several new projects have been started. For example the second Workshop on “Good Radiation Practices in industry and research” identified that most of the radiological accidents occur in research and industrial uses of radiation and that a large fraction of the high individual doses are in this sector. This is particularly the case in industrial radiography and irradiators; therefore the Network will focus in 2001 on these sectors in order to define possible improvements in terms of - industrial tools, - workers training, - good practices dissemination, - and modification of regulations. The proceedings of the first Workshop on " ALARA and Decommissioning” are now available in the issue 108 of the EC publication ”Radiation Protection”. The proceedings of the other Workshops will be published in the same format.

The 3rd Workshop on “Managing internal exposure”, which took place last November in Munich at BfS facilities, is a direct output from the first two Workshops which pointed out that non negligible internal exposures can occur during decommissioning and for those industries using Naturally Occurring Radioactive Materials (NORMs). The 80 participants of the Workshop, from 12 countries, were specialists of internal dose measurements, representatives of regulatory bodies and managers in charge of risks management within different types of facilities (nuclear industry, phosphate enterprises, radio-pharmaceutical industry, nuclear medicine services in hospital…). These participants considered that when the exposures are “predictable”, the ALARA principle can be applied to controlling the doses but many efforts remain to be done to find or develop adequate dose monitoring techniques (particularly personal air samplers) and strategies appropriate to the optimisation of radiological protection. The workshop also identified parallels with the development of the application of ALARA for external exposure in the 80's and in particular the need for case studies on the application of ALARA for internal exposure.

Most facilities dealing with radiological risk management have to deal with many other type of occupational risks, eg. biological, chemical,…etc which are often managed quite differently both from the regulatory and the practical points of view. The Steering Group has considered therefore that it should be fruitful for radiological protection specialists to exchange information with experts of the other risks and to benefit from each other experience in order to improve workers’ safety. The next Workshop of the Network will be devoted to the management of occupational radiological and non radiological risks.

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Observations and recommendations from the 3rd EAN Workshop on "Managing internal exposure"

Christian Lefaure, John Croft, Jean-Pierre Degrange

❑ Introduction

The European ALARA Network held its third Workshop devoted to "Managing Internal Exposure" in November 1999 at Bundesamt fur Stralhenschutz (BfS), Institut fur Stralhenhygiene facilities in Munich. The 80 participants from 12 countries were specialists of internal dosimetry, representatives of regulatory bodies, managers in charge of risks management within different types of facilities (nuclear industry, phosphate industry, radio-pharmaceutical industry, nuclear medicine services in hospital…). More than thirty oral presentations and twenty posters gave rise to very fruitful discussions and led to a final panel session where the most important lessons learned from the Workshop were formulated into conclusions and recommendations.

❑ Predictable and accidental internal exposures

Within the Western European radiological protection culture an approach often advocated is to minimise internal exposure in all situations. However the Workshop identified that there are two main categories of situations and that these are amenable to different approaches.

The first type of situation involves "predictable internal exposures". Such exposure situations are often encountered in the front end of the fuel cycle (mining, uranium refinement, fuel fabrication…) as well as in the industries using Naturally Occurring Radioactive Materials (NORMs), and in some aspects of decommissioning. In all these situations, the exposure whilst not continuous does occur with a reasonably predictable pattern and the air contamination is often the result of the worker's own activities. The estimated doses are, with few exceptions, lower than the dose limits, but can reach a significant fraction of it *"Doses are typically in the range 0-5 mSv, and in some cases, 10 mSv or more. Where doses are high, the internal component is the dominant exposure pathway"* (Hipkins, Shaw). In many such situations an approach that minimised dose from internal exposure might result in excessive costs or an increase in external exposure that exceeded the savings in dose from internal exposure. In such situations the ALARA principle can be applied to controlling the doses and therefore must be applied.

The second type of situation covers "accidental internal exposures" or "probabilistic internal exposures". These exposure situations correspond to work activities which, if no preventative measures are taken, could result in significant internal exposure. The probability of such exposure is often low but, if intakes do occur, the dose limit could be exceeded. The tendency then is to apply a broad "cautionary principle" approach and seek to eliminate exposures via both engineering methods (containment…) and the use of personal protective equipment. *"The design should remove the worker from the hazard by appropriate engineered reliable barriers. The use of appropriate technology, remote operation and maintenance should provide an operating system where human intrusion is minimised."* (Simister). In such situations, the main objective is the minimisation of the probability of occurrence of the accident.

❑ Lack of statistics

From the papers presented and the discussion sessions it was clear that within Europe for internal exposure there is little data on the numbers of workers concerned (even to an order of magnitude) as well as on their internal doses distribution. As may be seen in Table 1, internal exposures are not always included into the national statistics regarding occupational exposure and, even in those countries where they are included into these statistics, the data are far from being exhaustive, particularly for the industries related to NORMs.

Table 1. Integration of individual internal dose assessment into national occupational exposure statistics

Country	Internal doses included into national statistics
Belgium	No
France	No
Finland	Yes
Germany	not yet
Italy	No
The Netherlands	Yes
Norway	Yes
UK	Yes
Spain	Yes
Sweden	Yes

Recommendation 1

The meeting identified that there was limited data, at the national regulatory body level, on the number of workers exposed to intakes and the profile of the dose received. It is recommended that the Commission and the regulatory bodies pursue efforts to improve the data. Of particular concern is the area related to doses from the use of NORMs.

❑ Impact of dose assessment complexity on inconsistencies and difficulties to manage and communicate

Unlike in external exposure, it is often difficult to predict the levels of intake and hence the doses associated with internal exposure, because many variables come into play. The problem is compounded by the difficulties encountered in accurately measuring the actual intakes of many isotopes. Over the years, research has improved our understanding of physical and biological characteristics of internal exposure and the accuracy of the pulmonary, digestive, biokinetic and irradiation models. However, progress in these fields is still needed and the Workshop gave the opportunity to different stakeholders to present their point of views on the necessary improvements.

A number of internal dosimetry specialists provided presentations covering inter-comparison exercises (e.g. 3rd EULEP/EURADOS European Inter-comparison, IAEA surveys of inter-comparisons) showing that the assessment methods used vary largely from one country to another and even from one utility to another. Large variations in the intake and dose assessment results were observed, essentially due to the variety of the different biokinetic models and software tools used. Misinterpretation of instructions (i.e. the exposure scenario) and inconsistencies between dose factors and models used (new -old or old -new) were also put in evidence. *"Internal dose inter-comparisons (mSv) reveal commonly larger differences in the results than measurements inter-comparisons (Bq)....Depending on the case, the differences vary from a factor ten to several thousands."* (Beyer, Dalheimer). *"As a result, the mixed use of different models and dose factors can lead to results which are not scientifically based and also lead to greater inconsistencies."* (Cruz Suarez, Gustafsson). These specialists concluded that there is a strong need for harmonisation of the evaluation procedures especially for the radionuclides with high radiotoxicity.

These uncertainties in the dosimetry results bring into question their utility in the practical management of the hazards. In the discussions, managers from different industries clearly expressed their needs. They are looking to have at their disposal measurement tools and standardised methods of interpretation corresponding to a good compromise between accuracy of the dose assessment and the ease of the use of monitoring and results. Therefore they require from the researchers that *"they recognise operational radiological protection services as customers, and aim for as simple and transparent models as possible"* (Britcher). They also consider as fundamental the participation of the persons to be protected in the management of their doses and the trust of these individuals into the dose monitoring and assessment systems. Hence there is a requirement that follow up procedures be as simple and as easy as possible to understand by the workers. They also advise to communicate with the workers in terms of mSv rather than Bq, in order to allow the workers to put into perspective the external and internal risks. *"Don´t speak about becquerels.... Tell a person, that an internal dose is so much in mSv and that it has the same effect as an equally large external dose."* (Sundell)

Recommendation 2

The assessment of internal exposure often involves a wide range of parameters, which can lead to complex mechanisms to assess doses. These complexities provide problems for the communicating of dose information to the workforce and others, and in the ongoing management of doses. Thus there is a judgement to be made between the scientific accuracy and the ease of assessment/operational usefulness of the data. Where doses are not a significant fraction of the dose limit the meeting was strongly of the opinion that the ease of assessment should be the dominant factor in determining approved dosimetry, with more complex measurement protocols only being invoked for higher doses. This recommendation is mainly directed to regulatory bodies and monitoring laboratories.

❑ **Needed qualities of measurement methods to implement ALARA**

When, three decades ago, the process of implementing ALARA for external exposure was just starting, the use of film badge dosimeters, or even TLDs, could not provide in most cases answers to the questions: when, where and how were the doses received? Without this information it was thus difficult to answer the question "What could be done reasonably to reduce individual and collective exposures? " Since then, much has been done in order to assess and follow up as realistically as possible the external doses per job, task, category of workers etc. Several generations of electronic dosimeters have been developed, feed back experience computerised data bases have been set up, ALARA programmes have been elaborated and implemented…

Concerning internal exposure, one fundamental question raised during the Workshop was then:

"Have we adapted tools and strategies to provide answers to the "when, where, how and what" questions for "predictable" internal exposures?" (Lefaure)

In other words, are the tools available for the assessment of individual and collective internal doses, enough realistic, sensitive and analytical to allow the identification of the main sources of exposure and the selection of the optimal protection options, for individual dose levels by far lower than annual limits? To achieve this, are the measurement intervals shorter than the task duration?

As far as bioassays are concerned, it was clear from the Workshop that they cannot (and are unlikely ever to be) able to provide operational monitoring for "predictable exposure situations". For many reasons (cost, burden of work, worker acceptability…), even when they are performed regularly, their frequencies are not less than monthly intervals. Very often, the incorporation time profile of the worker between two measurements is not known and this may result in significant uncertainties on the dose assessment. In many situations, measurements below the detection limit could be compatible with annual doses equal or higher to the annual dose limit; this has been illustrated in the case of natural Uranium processing or Ra 226 arising from insoluble sulfates in oil and gas facilities, where the results of yearly lung counting correspond respectively to 50 mSv (Degrange) or 400 mSv (Van Weers). For these reasons, the bioassays are not useful to set up any operational internal dosimetry system for "predictable doses". However, even for that type of exposure situations, the bioassays remain very useful to assess the dose after an incident or accident.

Several presentations from sectors as different as radio-pharmaceutics, nuclear or phosphorus industries (Ardissimo, Bricher, Degrange, Erkens, …) have then shown that air samplers are much more adequate for providing an operational dosimetry than bioassays. The Static Air Samplers (SAS) allow a daily follow-up of the "sources" but do not allow any analysis of the contribution of individual tasks to the overall exposure, nor an operational assessment of the workers exposure. Personal Air Samplers (PAS) are the only method, which in theory provide that capability, particularly as in many cases the intake is directly due to the workers activities; i.e. the workers activities produce particles in suspension at the workers breathing zone. Representatives of Sorin Biomedica, British Nuclear Fuel (BNFL), Comurhex Uranium refinement plant and Termphos phosphorus production plant have particularly pointed out this point.

The adequacy of the different measurement methods to optimisation is summarised in Table 2 from Degrange.

Table 2. Adequacy of measurement methods to optimisation (Degrange)

	Air sampling				Bioassays	
	Collective	Individual			Urinary excreta	Lung retention
		5 L.h^{-1}	120 L.h^{-1}			
Measurement periodicity	1 day	5 days	5 days	1 day	30 days	180 days
Sources	Good	Average	Average	Average		
Tasks	Average	Good	Good	Very Good		
Operators	Insufficient	Good	Good	Very Good	Average	Very insufficient

The characteristics of the available SAS and PAS devices that are used up to now still raise many problems when optimisation of radiological protection is the objective.

Through a survey of many studies performed in different sectors, Witschger has shown that the ratio between PAS and SAS measurements may reach several orders of magnitude (see Figure 1)

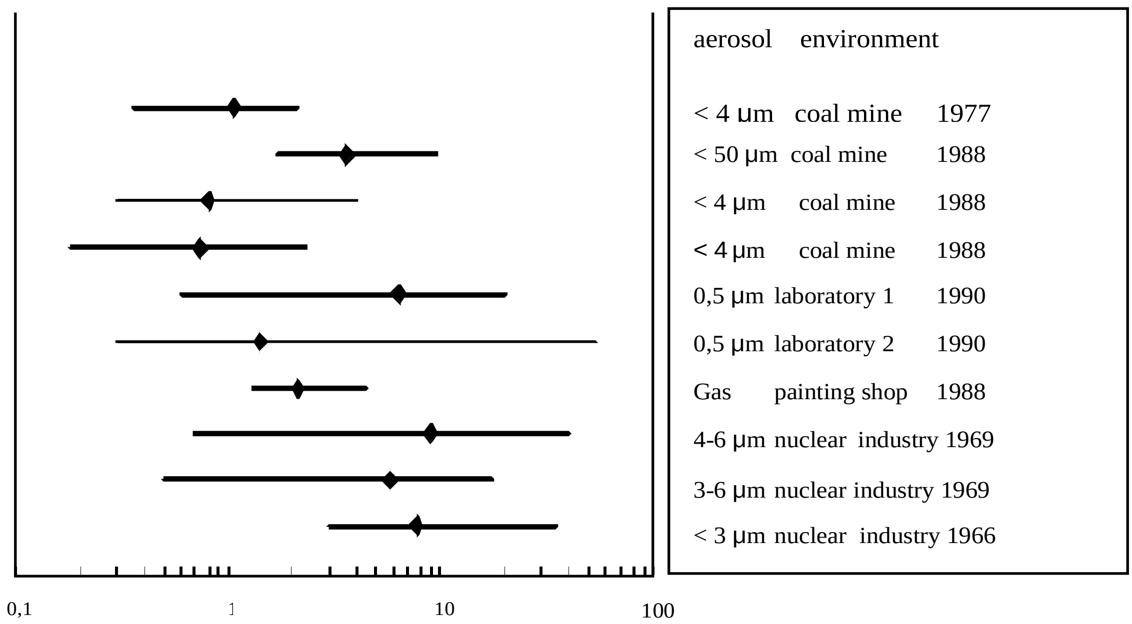

Figure 1 Ratios of personal measurement / static concentration measurement (Witschger)

In many circumstances it has been seen that SAS greatly underestimate doses. This is always the case when the dose is only due to the worker activity. Personal air sampling is then the only measurement method whose results might be close to the actual inhaled air concentration. However, when the air sampling rate is too low, when the particle size is high and when the radiotoxicity of compounds is important, the representativeness of the air sample in the PAS may be insufficient: a single particle may *"disrupt work and home life for nothing"*(Britcher). So even though the personal sampling measurement is usually considered much more representative of the aerosol in the breathing zone than the static sampling measurement, "*the personal sampling may(still) be inaccurate and imprecise"*(Witschger). Some new PAS sample the air at a high rate close to the breathing zone, but they are quite heavy (one kilo or more). Those problems explain that the unpopularity in most situations of the PAS among the workforce is not only due to the inconvenience of their use during the work, but also to the lack of trust and confidence in their results.

Therefore there is much to be done to select or develop adequate personal air sampling tools and to find monitoring strategies appropriate to the optimisation of radiological protection and acceptability from the workforce.

Recommendation 3

Not withstanding the above recommendation, there is a need to pursue efforts to improve the quality and accuracy of internal dose monitoring techniques (particularly personnel air sampler) to fit with the specifications needed for analytical task dosimetry. The meeting recommend to the Commission and regulatory bodies, that they support research in that area.

Where PAS are used as part of the implementation of the ALARA programme, the organisations that use them have developed strategies to ensure the acceptability of the measurement regimes by the workforce. Very often, PAS are only used during specific campaigns, when needed *"for analytical purpose"* (Sorin Biomedica) (Comurhex). However a few utilities have set up more formalised strategies, as illustrated by the figure 2 from BNFL, where PAS are sometimes routinely used.

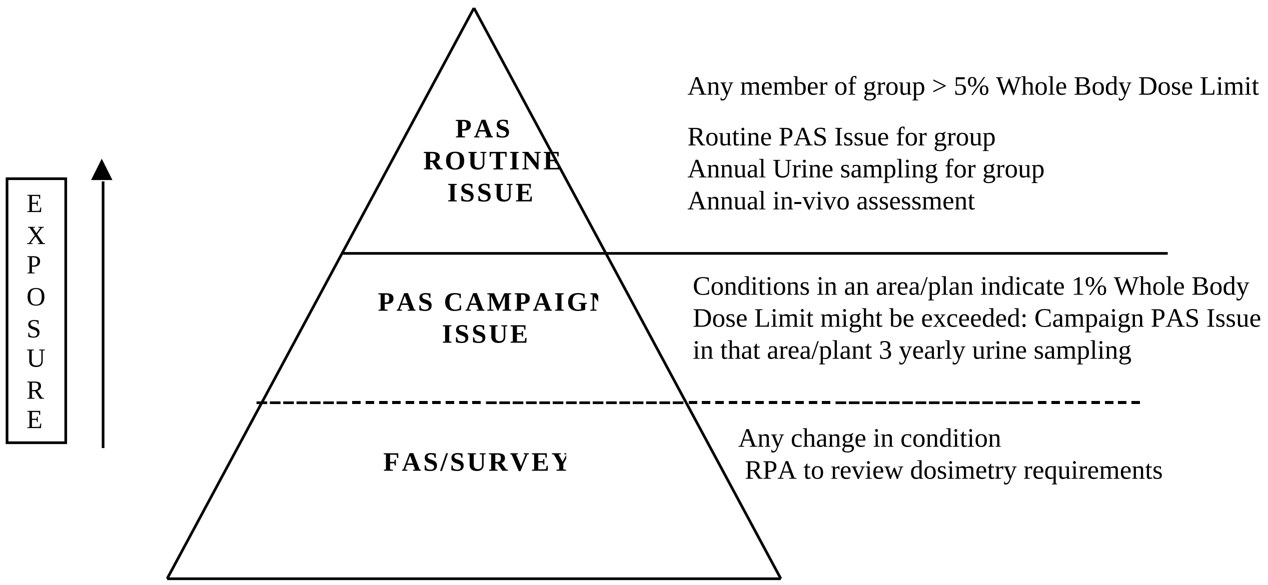

Figure 2. BNFL Sellafield internal dosimetry monitoring Programme (Britcher)

❑ ALARA implementation case studies

Less than ten percent of the presentations provided examples where the use of both PAS and SAS were used as input to a real analytical ALARA approach. One example was from Sorin Biomedica who has been able to select through such an analysis protection actions to efficiently reduce the predictable internal doses. In another case, Termphos used the monetary value of the man Sievert to check the cost efficiency of options. In these and other cases, the efficiency of the approach relied heavily on the involvement of both managers and workers. Some utilities have even set up what should be called an ALARA programme targeted at internal exposure. For example, Nycomed (alpha foil production for detectors) has had such a programme running since 1994. It has three major components: training and awareness of the workers, design and modifications of the workplaces and a global work management. This has resulted in a reduction of the collective dose from 57 to 19 mSv/year and the maximum individual internal dose from 9 to 2.6 mSv/year.

These examples demonstrated that implementing ALARA in the case of predictable internal exposures is possible and efficient. However, in most cases, ALARA is not applied even when it might be possible. It is thus necessary to demonstrate its potential through more case studies, in order to describe generic procedures and tools that will take into account the specificity of the ALARA approach applied to internal exposure.

Recommendation 4

The workshop identified parallels with the development of the application of ALARA for external exposure in the 1980's and in particular the need for case studies on the application of ALARA for internal exposure. These may involve both retrospective studies to identify important points in previous decisions as well as predictive case studies. They should cover the whole range of exposure scenarios e.g. NORM, nuclear fuel cycle, medicine, source production and transport... The meeting recommended that the Commission and regulatory bodies support such research.

Recommendation 5

The meeting noted that whilst the commitment, attitude and awareness necessary to implement ALARA, was now commonly in place for external exposure, the same could not be said for internal exposure. A number of case studies showed the positive impact of management explicitly committing themselves to applying ALARA to internal exposure, and the meeting urged all stakeholders, but particularly management, to adopt this approach.

❑ Conclusion

Many strategies have been proposed for the assessment and the follow-up of occupational internal doses, but these strategies have, in most cases, essentially dealt with respecting dose limits. The situation is hence quite similar with the development of the application of ALARA for external exposure in the 1980's. The participants to the Workshop expressed their hope that their recommendations will help to expedite the spread of an ALARA culture and to have adequate ALARA tools for internal exposure.

Recommendation 6

The meeting concluded that the Workshop had been successful in providing feedback between specialists in internal dosimetry, between operators and between the two groups. However the meeting also identified the need for ongoing exchanges. Therefore it is recommended that the Commission and regulatory bodies support the establishment of networking arrangements in this area.

ALARA Practice with the Refurbishment of the Belgian Reactor 2

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❑ Introduction

The BR2 reactor of the Belgian Nuclear Research Centre (SCK/CEN) at Mol, Belgium, was put into operation in January 1963. This Materials Testing Reactor is the SCK/CEN's most important nuclear facility and has operated for the past 32 years. A beryllium matrix, composed of irregular hollow prisms forming the reactor channels, acts as a moderator. Irradiation causes beryllium to become brittle and to swell. Cracking of the beryllium channels occurs by mechanical interaction, so that the whole matrix has to be renewed after an operation time depending on its utilisation. The facility has already been shutdown once in 1979-1980, to have its first beryllium matrix replaced. After another operating period of 15 years, the second beryllium matrix was reaching the end of its licensed life and had to be replaced to allow further operation. At the same time, the operation license required an overall inspection and re-qualification of the aluminium reactor vessel. So the BR2 reactor was shutdown at the end of June 1995 for an extensive refurbishment program. The beryllium matrix was replaced by a replica (previously used in the zero power reactor BR02) and the aluminium vessel was inspected and re-qualified for the envisaged 15 years life extension. Other aspects of the refurbishment program aimed at reliability and availability of the installation, safety of operation and compliance with modern safety standards. The reactor was restarted in April 1997.

❑ The implementation of the ALARA procedure during the refurbishment of the BR2

Philosophy
In recent years, the application of the ALARA procedure for special as well as for routine tasks became common practice at SCK/CEN. Since an important part of the refurbishment tasks had to be executed in moderate to very high radiation fields in places with difficult access, efforts were made by the ALARA Committee, the BR2 ALARA co-ordinator, together with the BR2 staff and workers to prepare each relevant subtask as an in-depth dose optimisation exercise.

Use of dose registration equipment
The BR2 system for automatic registration of doses, with electronic dosimeters, was extensively used during the refurbishment period. Each worker had to specify a task and subtask identification number as well as his personal ID before entering the area. When leaving the area, the data registered on the systemdata was dose, duration, and dose rates.

❑ Refurbishment tasks with high collective doses

Apart from maintenance work on cooling towers, in the water treatment building and in machine and ventilation control rooms, most of the refurbishment work was executed in the controlled areas of the BR2 reactor. The tasks resulting in the highest collective doses were :

- The replacement of the beryllium matrix ;
- The internal inspection of the primary heat exchangers ;
- The internal inspection of the reactor vessel ;
- The refurbishment of the neutron beam tubes ;
- Work on experimental installations in the reactor sub-pile room.

❑ BR2 refurbishment dose statistics

The main results are:

- Total registered collective dose : 343.5 man.mSv
- Total number of registered records : 26271
- Total manpower : (approximately) 250

Collective doses of specified subgroups
The following table lists the collective doses of pre-defined subgroups resulting from the BR2 refurbishment. It shows that most of the work (both in respect of time and collective dose) was executed by SCK/CEN personnel in area's with relatively low radiation levels.

subgroup	man.mSv	# records
BR2 operation	114.07	7990
BR2 maintenance	44.24	3371
BR2 experiment manipulations	34.34	1212
BR2 hot cells	47.40	2192
BR2 irradiations group	12.10	976
Technology Dept.	20.60	1737
Safety Dept & Radiation Control	14.13	1830
Other SCK/CEN personnel	10.44	1468
Contract workers	45.32	3796

Collective doses related to specific tasks
The predictions made in the preparatory study of the ALARA procedure were mostly based on simple calculations taking into account background radiation level and predicted execution time. Apart from the inspection of the internals of the primary heat exchangers, it would have been very difficult and not relevant to make more precise predictions based for instance on critical pathways in the controlled area or on more detailed working instructions.

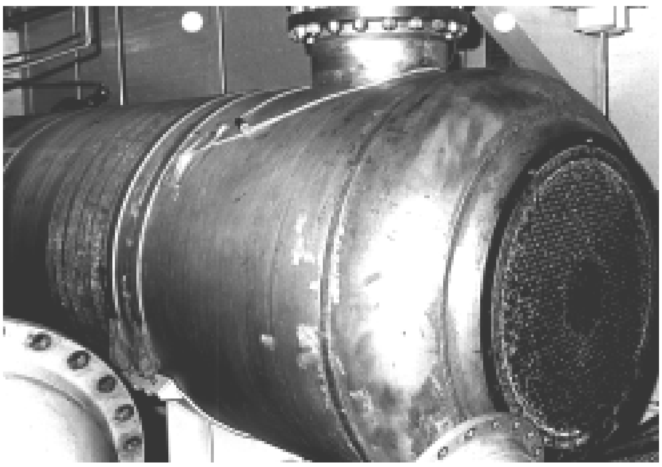
The Beryllium matrix which had to be replaced was highly active. A typical gamma scan gave average levels of about 0.8 Sv/h, with maximum dose rates up to 4 Sv/h just below mid plane level of the Be-channels. The unloading and storing of the Be-channels and the cleaning of the reactor vessel were executed under water. These tasks took one month and were executed by three teams consisting of 10 workers each.

Unloading of the Be matrix	Registered (man.mSv)	Predicted (man.mSv)
Preparatory work	15.53	
Unloading & storing + cleaning of vessel	19.60	
Total (30 workers)	35.13	196.40

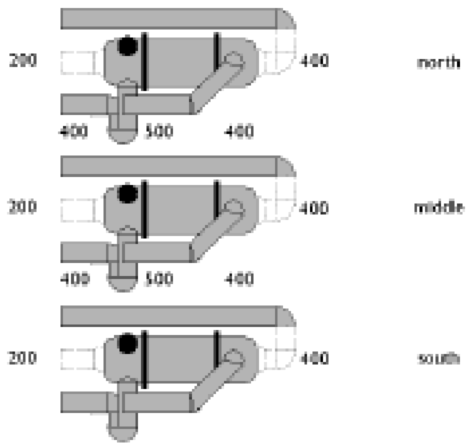
BR2 has a set of radial and tangential shielded tubes which can focus neutron beams directly escaping from the reactor core. It was decided to dismantle and unload about half of the available reactor neutron beam tubes and to close the left-over holes in the concrete shield of the reactor pool. The material was highly activated (aluminium and stainless steel, typically up to 500 mSv/h). The dismantling and unloading had to be done on the spot, in places with poor accessibility and directly in contact with the activated material. This explains the relative high individual doses received in short time periods.

Refurbishment neutron beam tubes	Registered (man.mSv)	Predicted (man.mSv)
Unloading of selected beam tubes	12.83	-
Maintenance other beam tubes	1.05	-
Total (5 workers)	13.89	

The inspection of the primary heat exchangers was scheduled towards the end of the refurbishment period. By that time, dose rate levels were significantly lower than at the moment the reactor stopped.
The following picture shows the entrance to the tubes (secondary coolant flow) after removal of the secondary circuit water collector of one of the three primary heat exchangers.



The radiation map shows the dose rate levels (in µSv/h) in the working area at the time of the inspection (top view)



Inspection primary heat exchangers	Registered (man.mSv)	Predicted (man.mSv)
Opening heat exchangers	1.65	
Video endoscope inspection of tubes	4.91	
US and penetrant weld inspection	3.59	
Closure of heat exchangers	0.76	
Total (18 workers)	10.91	38

The BR2 reactor vessel was inspected internally by way of a detailed visual and ultrasonic inspection program after removal of the internal parts (matrix and support structure). As the middle part of the aluminium reactor vessel (core level) is continuously highly activated, the vessel stayed covered with water during the whole refurbishment period and the inspection of the inside had to be done remotely. Dose rates on the working platform positioned on the vessel (open and filled with water) were of the order of 50 µSv/h. A gamma scan on the middle axis of the vessel (under water) gave values up to 350 mSv/h at mid plane level.
The removable vessel head was ultrasonically, die penetrant and visually inspected. As this part of the vessel has relatively low activation, the inspection could be executed in the working area next to the reactor pool. However, instructions were given to limit exposure time in direct contact with the material.

Inspection reactor vessel	Registered (man.mSv)	Predicted (man.mSv)
Preparatory work	0.67	
Inspection vessel	5.68	
Total (21 workers)	6.35	46.2

The loading and positioning of the new beryllium matrix was, as the unloading of the previous one, done with remote handling equipment. The loading was finished in a relatively short time (20 days; 2 operators/team) and, because of the very low activity of the Be-channels, didn't result in high exposures of the workers.

Loading and positioning of the new beryllium matrix ; closing vessel	Registered (man.mSv)	Predicted (man.mSv)
loading matrix	4.17	
closing of the reactor vessel	0.79	
Total (12 workers)	4.96	14.7

Comparison of the predicted task related doses with the registrations
The total collective dose of the above mentioned tasks (without beam tubes results) is 58 man.mSv. The total of predicted collective doses is 296 man.mSv (about five times higher).

❑ Comparison with the previous refurbishment of BR2 (1979/80)

Comparison of the dose results

	1979/80	1995/96
General data		
duration	18 months	21 months
total registered dose (man.mSv)	~710	343
# of workers with year dose > 10 mSv	2	0
# of records > 1 mSv	10	0
Task specific registered total doses	(man.mSv)	(man.mSv)
unloading of experiments	69	< 5
unloading of Be-matrix	155	35
refurbishment beam tubes	not executed	14
inspection of primary heat exchangers	~300	11
inspection of reactor vessel	248	6.3
loading of Be-matrix	~25	5

Analysis of the differences between 2 refurbishments
Due to the lack of a computerised real time dose registration system in 79/80, only the major dose results were followed up at that time. Therefore the total collective dose was in fact higher than 710 man.mSv.

The first reason for the lower collective dose connected to the unloading of the experimental devices in 95 was the fact that, in 95, after unloading, the experiments were removed out of the reactor building instead of storing them to one side in the reactor pool. This resulted in a significantly lower background radiation level in the working area around the pool during the whole period of the 95/96 refurbishment.

The unloading of the reactor matrix took two months in 79/80. In 95/96 this job was executed in one month. The use of better distance manipulation tools resulted in an easier unloading, while a better filtering of the pool water and the placement of covers in the bottom of the Be-channels (at the level of the bottom support grid) permitted an easier collection of Be pieces after unloading of the channels. Also in 95/96, this collection was done once after the complete unloading of the matrix while, in 79/80, it was repeated after each unloading of a channel. Finally, in 79/80, a detailed gamma-scan of the matrix channels was made before unloading. This was in fact a dose consuming task which was not really necessary, as it was known that the interpretation of the matrix radiation map would not significantly alter the unloading procedure. Scans were previously made during routine operation, had given a good assessment of the radiation levels in the matrix.

In spite of the fact that the inspection of the primary heat exchangers in 79/80 resulted in a 300 man.mSv collective dose, the outcome of the inspection was unsatisfactory. This was due to the lack of specialised equipment, which was needed to inspect the tube bundles internally. For the inspection of 96, a 6.3 mm color video endoscope was bought. This, in combination with a better preparation of the work on the spot and the help of experienced contract workers, resulted in a significantly lower collective dose. Also importantly the inspection results finally led to a correct interpretation of the aluminium corrosion phenomena.

The internal inspections of the reactor vessel in 79/80 and 95/96 were both successful. However, thanks to automatic remote control inspection equipment, the collective dose could now be limited to a level 40 times lower than in 79/80.

In 79/80, three teams took care of the three consecutive tasks during the unloading of the old matrix (unloading of a channel - loading of the channel in a transport tube - transporting the tube to the storage pond), as well as during the loading of the new one (the other way round). This resulted in a lot of people "waiting" near the reactor pool while the unloading of a channel (most time consuming of the three consecutive tasks) was in progress. In 95/96 the three tasks, both during loading and unloading, were executed by only one team. In comparison with 79/80, the team carried out the work in less time and with less collective dose.

❑ Conclusions: lessons learned

Conclusions regarding individual doses
The individual and collective doses received during the refurbishment were relatively low. This is illustrated by the database results (summarised in previous paragraphs) as well as by the following statements :

- None of the refurbishment workers (SCK/CEN or contract workers) exceeded the “maximal” individual cumulative dose of 10 mSv/y advised by the SCK/CEN;
- None of the contract workers received an individual dose that would have prevented them being used for other contract work for the rest of the following 12 months (taken into account ICRP recommended maxima) ;
- None of the SCK/CEN workers received an individual monthly dose that would have made them unavailable for future refurbishment tasks (taken into account the SCK/CEN recommended maximum of 1 mSv / month).

Looking back, it is also important to note that the number of enlisted workers was certainly not excessive, given the strict work scheme and the extensive number of tasks scheduled during the 21 months of the BR2 refurbishment.

Conclusions regarding the predicted task specific collective doses
Dose predictions seem to be systematically too high. A study of specific tasks and related ALARA procedures afterwards identified that the dose over estimates were mostly due to an over estimation of the time needed to execute a specific task. Also, in a lot of cases, not enough

radiation level data was available to calculate detailed dose predictions. This was merely due to limited access to - and/or high radiation levels at - the specific location were the work had to be done.

Conclusions regarding the use of the ALARA principle during the BR2 refurbishment of 95/96

Although for some specific refurbishment tasks the "ALARA exercise" was rather difficult, one can certainly say that the introduction of the ALARA procedure was well accepted by the BR2 management and workers.

It is obvious that a real dose optimisation procedure is only useful if the scheduled task is not too much traced and simplified regarding to task steps, number of workers and/or planning. During the BR2 refurbishment, apart from the above mentioned extensive tasks, a lot of rather simple jobs were foreseen in controlled areas. In this case, it is of course advisable to try to do the ALARA exercise for a larger set of related tasks, permitting more freedom in doing a multi-criteria analysis. One can understand that this is not always possible and that often the preparatory study work is too complicated and expensive for the expected benefit. Instead of trying to force a lot of simple tasks into a more global framework, it is often more productive to do a "limited" ALARA exercise : checking the experience of the workers, specifying protective measurements, studying radiation maps together with the workers and - finally - asking for feedback after the execution of the task. Direct communication with the workers themselves got them used to the ALARA philosophy and improved their understanding about radiological protection in the broad sense.

As a conclusion, we can say that, both for the extensive and complicated tasks, and for the more simple jobs, the application of the ALARA procedure, in combination with the feedback of statistical analysis of registered doses into scheduled tasks, has led to a clear reduction of received individual and collective doses during the refurbishment of BR2.

The BR2 was successfully restarted in April 1997 for another period of about 15 years, contributing to international research & development programmes involving industry, regulatory bodies and multiparty collaboration in the nuclear field.

The ALARA-experience gained with the refurbishment - and the lessons learned - will permit an even more effective implementation of the ALARA procedure in the daily life safety culture at SCK/CEN.

Radiological Protection in the Radioactive Incident of ACERINOX in Spain

José T. Ruiz, Juan M. Campayo (LAINSA)

❑ Introduction

At the end of May of 1998 a Cs-137 source was melted accidentally in one of the stainless steel production plant furnaces that the ACERINOX company has in Cadiz (Spain).

Once the presence of radioactive contamination was detected, a number of organisations provided assistance. These included LAINSA, an expert company in decontamination and dismantling of radioactive and nuclear facilities with experience in radioactive emergencies, the regulatory body, CSN, and the waste management utility, ENRESA. They have evaluated the situation and implemented first radiological protection measures:

- Evaluation of the contamination in the plant
- Control of the access of people, vehicles and materials to the contaminated zones
- Delineation and signing of all areas where radioactivity was detected
- Control of radiation in the gases extracted by the smoke clearing system.

The recovery operation for the affected facilities began immediately: even before the formal approval from CSN of a Performance Plan, to decontaminate the affected facilities.

Decontamination took 5 months, and 50.000 man-hours were necessary to perform the whole work (20% corresponding to radiological protection activities). The total collective dose was about 60 man.mSv.

❑ Objectives

The objectives established in the Performance Plan, previously mentioned, were:

- To avoid contamination outside the Plant.
- To guarantee the Radiological protection of the professionally exposed workers, the personnel of Acerinox and the public in general
- To control the decontamination activities according to the Radiological Protection standards.
- To ensure that the generated radioactive waste remained in safe conditions as far as their manipulation, storage and transport are concerned.

❑ Affected facilities

Since the very beginning the contamination had affected the smoke dust that circulates through the conduits of the gas extraction system of the electrical arc furnace nº 1 and to the shared clearing system for furnaces no 1 and no 2 (Figure 1).

Table 1. Levels of initial radiation in the main areas

SYSTEM	Average Dose Rate mSv/h	Maximum Dose Rate mSv/h
Electrical arc furnace no 1 and gas ducts extractions	0.5	1.8
Natural Cooler and stark arrester	0.02	0.05
Bag filter nº1	0.05	0.1
Bag filter nº2	0.02	0.03
Silos A and B	0.03	0.1

Table 1 summarises the detected values of radiation in the affected systems. The measured activities in samples taken in the smoke dust, before the beginning of the decontamination were in the range 800 to 2000 Bq/g.

❑ Radiological Criteria

According to the Performance Plan approved by the CSN the final state of the facilities would be such that:

- The maximum permissible dose in any zone of the factory did not exceed the value 1 mSv in an annual period.
- The derived values from surface contamination were such that they did not exceed 4 $Bq.cm^{-2}$, in those areas where their measurement was possible.

Due to the dimensions of the facility and the great number of affected zones it was not easy to establish a strict and unique access control. Thus, in the first phase those zones with higher dose rates and requiring greater movements of people were identified. The measures adopted were based on two general approaches:

- Immediate Intervention: action to remove radioactive material, decontaminating the zone, remove systems, equipment, etc, or.
- Isolation of these areas, by establishing alternative access and routes.

❑ Works development

The objective established for the final state of the facilities had to fulfil two requirements; the production of the Steel Works had to continue and it was necessary to cope with the radiological protection principles.

Therefore, in the first phase decontamination was limited to clearing line no.1, allowing normal production to continue on the other clearing line. In that phase most of the very low activity contaminated wastes were generated

Next decontamination of the electrical Furnace no.1 was undertaken, followed by the Bag filter no.2 and silos. In these phases, less smoke dust wastes were extracted but metallic wastes, refractory bricks of the furnace, etc., were generated. Dry decontamination techniques (vacuum cleaning, grinding, etc.) were used to avoid the generation of liquid wastes that would have been difficult to treat in that facility.

❑ Radiological control and ALARA studies

The main activities of LAINSA were as follow:

Control of effluents
Isokinetic samples were taken from the gas evacuation systems. The results showed that the values, prior to dispersion and diffusion in the atmosphere, were less than the lower limit of detection: 0.6 mBq/l. This monitoring was continuous until the decontamination of the smoke clearing systems was completed.

Radiological control of Decontamination work
The criteria for radiological protection control of the programme are summarised in the Table 2.

Table 2. Radiological protection criteria

Individual dose Constraints: 0.3 mSv per day; 1 mSv per week; 3 mSv per month **ALARA studies** If anticipated collective dose is higher than 10 mmanSv, **Use of electronic dosimeters** Works with dose rate greater than 30 µSv/h
Control of exposed time In an ambient dose rate higher than 150 µSv/h.
Control of environmental contamination Before and during the execution of the works with risk of producing dust. With values between 3.75 % and 37.5 % of the LDCA, face mask will be used. With values greater than 37.5 % of the LDCA air-fed equipment will be worn the ventilation conditions will be improved.
Control of surface contamination Surface contamination limit in zones in which the measurement is feasible < 4 Bq/cm^2

The radiological state of areas, equipment or systems were described in the corresponding Radiation Work Permit, where a dose estimation was also made.

❑ Controls of access

RP technicians from the UTPR - a specialised radiation protection company authorised to perform radiation protection tasks and provide specific activities such as decontamination - monitored the entrance and exit of personnel, materials and wastes, and controlled the accesses to the work zones. The controlled zones in the work places and the waste storage areas were periodically monitored, to assure that the established radiological conditions were fulfilled.

❑ Occupational exposure

All the personnel involved in decontamination operations in Acerinox were classified as Professionally Exposed Personnel to ionizing radiation and used TLDs. The total collective dose was 60 man-mSv. For the 5 months period, the average individual dose was 0.6 mSv and the maximum individual dose was 3.5 mSv.

Table 3 shows the results of the operational dose (electronic dosimeters) for the critical tasks. 40 percent of the total collective dose was associated to the operations of decontamination of the electrical arc furnace nº 1 and of the gas ducts, where doses rates were the highest. The next critical group consists of the individuals dedicated to the wastes segregation and preparation (23% of the total collective dose). In this case, the number of people and the time used were more significant than the dose rates. As far as the internal dosimetry was concerned, two programs of monitoring were set up (whole body monitoring) , the first a few days after the start of works, to verify the suitability of the adopted protection measures. The second at the end of the work to confirm the absence of contamination. In the all cases the results were less than the recording level, 0.5 mSv.

Table 3. Operational doses for critical tasks

TASK	DOSES (man-mSv)
Electrical arc furnace nº1 and gas ducts extractions	16.1
Natural Coolers	3.1
Bag filter nº1	5.3
Bag filter nº2	2.3
Scaffolding installation and stripping	3.4
Silos	0.5
Wastes Handling	9.7
Total	40.4

❑ Waste management

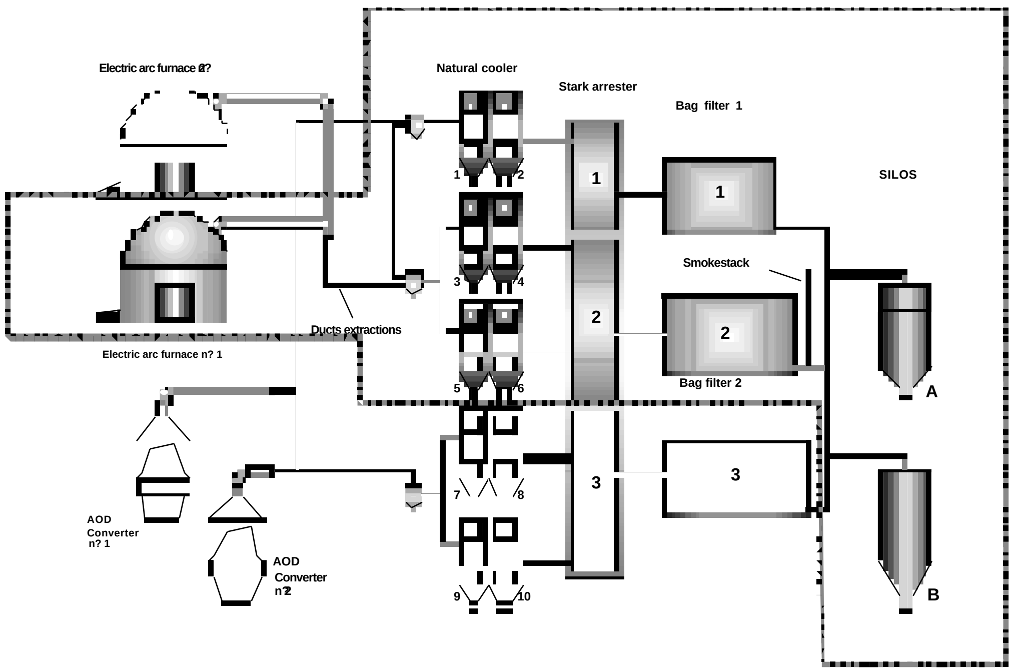
The wastes produced were put into two types of containers. The smoke dust was put into 1 m^3 big-bags, whereas metallic, plastic wastes, paper, etc., were put into 220 liters drums. Each waste was identified, labeled, and measured. The parameters registered for each container were the content, weight, size, origin, specific activity, etc. These wastes were stored within the facility in a place with the suitable radiological and physical security conditions. A significant percentage of the waste was checked with spectrometrical measures to determine the specific activity and to evaluate the decontamination process.

❑ Conclusions

The incident in Acerinox in May 1998 did not involve illegal risks of exposure to ionizing radiation for the workers, or for the public, nor for the environment. The adopted radiological protection measures in the decontamination work were effective (no internal contamination). Also, the external doses remained at very low levels, thanks to the strict application of the established criteria of radiological protection from the beginning of the works.

Finally we would like to note that the immediate intervention made in ACERINOX, has demonstrated the capacity of response and co-ordination between companies and institutions in an incident without precedent in Spain.
Approximately 2000 Ton of low level activity wastes were produced in the decontamination operation at ACERINOX. (smoke dust 91%, fiber cement panel 4%, refractory bricks 2%, compressible waste 2%, metallic waste 1%).

Figure 1. Contaminated facilities



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ALARA NEWS

4th European ALARA Workshop "Management of Occupational and non radiological risks: lessons to be learned"

Antwerpen
Belgium, 20-22 November 2000

The aim of the 4th Workshop is to provide an opportunity to put radiological risk management into context with the management of other occupational risks, by engaging interested parties (managers, workforce, contractors, regulatory bodies, communicators etc in the exchange of information and experience.

The objectives are:

By means of case studies from a range of different work activities (nuclear, chemical, petro-chemical, biological, engineering etc…):

- to review the approaches to risk management, both for single or multiple types of risk(s) ;
- to identify the significant factors (technical, legal, economical, social, health impact, ethical, ...) in the decision making processes;
- to examine how the different interested parties impact on the risk management process (at the regulatory body, corporate and workstation levels...), including risk perception.

In order to pursue these objectives it is envisaged to have keynote speakers covering the different work activities as well as the points of views of interested parties, and to devote a large part of the Workshop to work in small discussion groups. Therefore the number of participants will be restricted to a maximum of 50.

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Feedback fom the course on optimisation of radiological protection in the design and operation of nuclear facilities

St-Petersburg, Russia, 18-21 October 1999

The CEPN and NRPB have organised on behalf of the International Atomic Energy Agency and European Commission the ninth course on Optimisation of Radiological Protection in the Design and Operation of Nuclear Facilities. The first seven of these courses were used to be "shuttle" courses in France, Sweden, Germany, and Spain. At the end of the seventh it was decided to stop the organisation of such courses in EC countries, as they had provided the catalyst for national courses. However it was suggested that other courses should be organised aimed at central and eastern countries audiences. As a result the first of these courses took place at Prague in September 1997. The St Petersburg course was the second one focused on that audience. It has allowed representatives of regulatory bodies and utilities from Armenia, Bulgaria, Hungary, Lithuania, Romania, Russian Federation, Slovakia, and Ukraine to make fruitful exchanges of feedback experiences and to demonstrate the improvement of the occupational exposure situations in their Nuclear Power Plants since the previous course.

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