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OCCUPATIONAL DOSE CONTROL in Nuclear Power Plants

An Overview

by

C. Viktorsson, J. Lochard
M. Benedittini, J. Baum, T.A. Khan

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971) and New Zealand (29th May 1973).

The Socialist Federal Republic of Yugoslavia takes part in some of the work of the OECD (agreement of 28th October 1961).

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all European Member countries of OECD as well as Australia, Canada, Japan and the United States. The commission of the European Communities takes part in the work of the Agency.

The primary objective of NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonisation of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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LE CONTROLE DES DOSES
D'IRRADIATION PROFESSIONNELLE
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Rapport de synthèse

FOREWORD

The data on individual and collective doses are key indicators of the radiological situation at a nuclear plant, and are often used to assess the overall performance of the plant in relation to radiation protection. This, however, is only part of the information required to implement an effective dose-control programme. It should be complemented by information on methods for the reduction of doses in the plants and on task-related doses.

The purpose of this report is to give an overview of the occupational exposure situation and of achievements in the field of occupational dose control at nuclear power plants in the OECD Member countries.

TABLE OF CONTENTS

1. INTRODUCTION

Operation and maintenance of nuclear power plants imply the exposure of workers, i.e. plant personnel and outside workers, to ionising radiation. The average individual doses and the collective doses give an indication of the radiological conditions at a plant and are, therefore, often used by plant operators and regulators to assess the overall performance of the plant operation in relation to radiation protection.

To implement an effective dose control programme, knowledge of the parameters that govern the exposure of workers is required. For example, it is necessary to know the dose rates at various places in the plant and details about the working conditions and requirements. Another type of information that is particularly useful in this context is task-related dose information, both from the plants own past history and from the experience of a large cross section of other plants. Co-operation and information exchange not only within a particular country, but also between utilities in different countries would be of importance to facilitate this process.

In its present programme on occupational exposure the NEA Committee on Radiation Protection and Public Health (CRPPH) recommended that the NEA coordinate international efforts towards occupational dose reduction in nuclear facilities, particularly nuclear power plants, in order to enable utilities and authorities in Member countries to gain from each other's experience. One way of doing this would be to create a mechanism for exchanging information and experience on occupational dose data, particularly in the field of "high-dose jobs".

The present overview report on occupational dose control should be seen in this context and has been prepared to support the implementation of such a mechanism on an international level.

2. PURPOSE AND SCOPE

The purpose of the report is to give an up-to-date overview of the dose situation for workers and the achievements in the field of occupational dose control at nuclear power plants. It will therefore:

a) Review dose trends and the current situation as regards the exposure of workers;

b) Identify types of work, operations and tasks giving a significant contribution to the exposure of workers (high-dose jobs);

c) Identify the principal factors influencing the exposure of workers;

d) Examine current available dose-reduction techniques, methods and strategies, including the application of the ALARA principle;

e) Discuss the role of past experience in connection with occupational dose control.

Table 1. Power Reactors Used in the OECD Countries
Reactors under Construction shown in brackets [NEA 1989]

(Net GWe) (en GWe nets)

COUNTRY	BWR		PWR		GCR		HWR		PAYS
	Units Tranche	Capacity Puissance	Units Tranche	Capacity Puissance	Units Tranche	Capacity Puissance	Units Tranche	Capacity Puissance	
Belgium	-	-	7	5.5	-	-	-	-	Belgique
Canada	-	-	-	-	-	-	18 (4)	11.9 (3.5)	Canada
Finland	2	1.4	2	0.9	-	-	-	-	Finlande
France	-	-	49 (8)	49.4 (10.8)	4	1.5	-	-	France
Germany F.R.	7 (1)	6.9 (1.2)	13	14.3	-	-	-	-	Allemagne, R.F.
Italy	1	0.8	1	0.3	-	-	-	-	Italie
Japan	18 (7)	14.1 (6.3)	16 (6)	11.8 (6.3)	1	0.2	-	-	Japon
Netherlands	1	0.1	1	0.4	-	-	-	-	Pays-Bas
Spain	2 (2)	1.4 (1.9)	7	5.7	1	0.5	-	-	Espagne
Sweden	9	7.1	3	2.6	-	-	-	-	Suède
Switzerland	2	1.3	3	1.7	-	-	-	-	Suisse
United Kingdom	-	-	- (1)	- (1.2)	38 (1)	10.0 (0.6)	1	0.1	Royaume-Uni
United States	37 (4)	31.9 (4.0)	70 (14)	62.9 (17.0)	-	-	-	-	Etats-Unis
OECD Total	79 (14)	65.0 (13.4)	172 (29)	155.5 (35.3)	44 (1)	12.2 (0.6)	19 (4)	12 (3.5)	Total OCDE

COUNTRY	FBR(1)		HTR(2)		Others/Autres		Total		PAYS
	Units Tranche	Capacity Puissance	Units Tranche	Capacity Puissance	Units Tranche	Capacity Puissance	Units Tranche	Capacity Puissance	
Belgium	-	-	-	-	-	-	7	5.5	Belgique
Canada	-	-	-	-	-	-	18 (4)	11.9 (3.5)	Canada
Finlan	-	-	-	-	-	-	4	2.3	Finlande
France	2	1.4	-	-	-	-	55 (8)	52.3 (10.8)	France
Germany F.R.	- (1)	- (0.3)	1	0.3	-	-	21 (2)	21.5 (1.5)	Allemagne, R.F.
Italy	-	-	-	-	-	-	2	1.1	Italie
Japan	- (1)	- (0.3)	-	-	1	0.2	36 (14)	26.3 (12.9)	Japon
Netherlands	-	-	-	-	-	-	2	0.5	Pays-Bas
Spain	-	-	-	-	-	-	10 (2)	7.6 (1.9)	Espagne
Sweden	-	-	-	-	-	-	12	9.7	Suède
Switzerland	-	-	-	-	-	-	5	3.0	Suisse
United Kingdom	1	0.3	-	-	-	-	40 (2)	10.4 (1.8)	Royaume-Uni
United States	-	-	1	0.2	-	-	108 (18)	95.0 (21.0)	Etats-Unis
OECD Total	3 (2)	1.7 (0.6)	2	0.5	1	0.2	320 (50)	247.3 (53.4)	Total OCDE

1) FBR = Fast Breeder Reactor
2) HTR = High Temperature Reactor

The information in this report pertains to all main reactor types which exist in the OECD countries. However, because of their preponderance, a greater emphasis is put on questions related to light water reactors. The intention of the report is to give a general overview of the main lines and trends of developments without any claim to giving a complete picture of all developments in the field.

3. NUCLEAR POWER PLANTS IN OPERATION AND UNDER CONSTRUCTION IN THE OECD COUNTRIES

By the end of 1988 there were approximately 430 nuclear power reactors in operation throughout the world of which 320 were in the OECD Member countries. The main reactor types used are:

- Light Water Reactors (LWR) of two types - Pressurised Water Reactors (PWR) and Boiling Water Reactors (BWR)
- Heavy Water Reactors of a pressurised tube type (HWR)
- Gas Cooled Reactors (GCR).

LWRs are dominating in number, and account for about 75 per cent of all reactors in the OECD area. LWRs are at present operated in 11 of the 13 OECD countries using nuclear power. The other main reactor types, HWRs and GCRs, are mainly operated in Canada and the United Kingdom, respectively. Table 1 gives additional data on the nuclear power situation in the OECD Member countries (NEA 1989).

4. CURRENT SITUATION AND TRENDS IN OCCUPATIONAL EXPOSURE

4.1. Overview of Levels and Trends

Workers at nuclear power plants, plant personnel as well as contractors, are exposed to ionising radiation during plant operation and maintenance. Regulatory authorities require licencees or employers to carry out the monitoring of workers and ensure that individual doses remain within established limits set by the authorities. In addition to the "institutional" monitoring, most utilities run an "operational dosimetry" programme to control doses in particular jobs as well as to facilitate planning and follow up of doses.

The main contribution to occupational exposure in nuclear power plants comes from external irradiation. Doses from internal contamination appear to be a minor component to the exposure. For this reason, the doses referred to in this report are essentially those resulting from external irradiation. There is one exception, however; for the heavy water reactors a substantial part of the doses received by the personnel is due to internal contamination from the uptake of tritiated water vapour. In fact, about 20-25 per cent of the total annual dose in CANDU-reactors, which are heavy water reactors of a pressurised tube type, results from the tritium uptake [VIV 1986].

Apart from individual doses recorded by the different kinds of personnel dose meters, the total dose, or collective dose received by all workers in a plant is usually assessed, although this is not a legal requirement.

The concept of collective dose which is the sum of all recorded individual doses, is largely used to describe the overall radiological impact of practices involving radiation risks and is, therefore, relevant when describing the radiological impact of nuclear power operation. The unit of the collective dose is the manSievert (manSv). Throughout this report the term "collective dose" is used to mean collective effective dose equivalent and "individual dose" to mean individual effective dose equivalent.

The two indicators, the measured individual dose and the calculated collective dose, are used here to identify and summarise occupational dose trends.

Collective Doses

Using the concept of the collective dose, one may derive additional indicators, such as the annual collective dose per reactor unit, expressed in manSv/reactor-year. Other indicators, such as the annual collective dose per produced megawatt-year of electricity or per installed capacity, give also an indication of the radiological situation at the plant. However, throughout this report the annual collective dose per reactor unit is used.

The global dosimetric situation for the main reactor types used is given in Figure 1. It shows the average annual collective dose per reactor unit as a function of calendar year for each type of reactor. Data are included from all OECD countries with a nuclear power programme.

It appears from Figure 1 that the collective dose levels are quite different for the different reactor types, and that they are higher for LWRs (BWRs, PWRs) than for HWRs and GCRs. Furthermore, it seems that they have stabilised at a value which for PWRs is about 3 manSv/reactor unit, for HWRs about 1 manSv/reactor unit and for GCRs less than 1 manSv/reactor unit. For BWRs, the collective doses fluctuate between about 5 and 9 manSv/reactor unit, but the overall trend in particular since 1983 seems to be downwards.

The levels and trends shown in Figure 1 are dominated by the results experienced in countries with large nuclear power programmes such as the United States, France, Federal Republic of Germany and Japan for LWRs, the United Kingdom for GCRs and Canada for HWRs. Therefore, in order to understand the major reasons behind the levels and trends of occupational exposure shown one has to look at the corresponding developments in the countries operating these plants. Such a comparison may also show the influence of different design, operating strategy, maintenance and backfitting requirements on the exposure of workers.

Several reports have been published where statistics from various countries have been included [BNL 1988, CEP 1989, HIL 1987]. From these reports it can be seen that there is a slight downward trend of collective dose in PWRs since about 1982 in several countries such as the USA, Japan and the FRG, and a flattening out in others, like France. It also appears that the differences between "PWR countries" concerning average annual collective doses seem to be decreasing but is still about a factor of 6 in the year 1987 if one compares the country with the lowest exposure, Finland, and the country with the highest exposure, the USA.

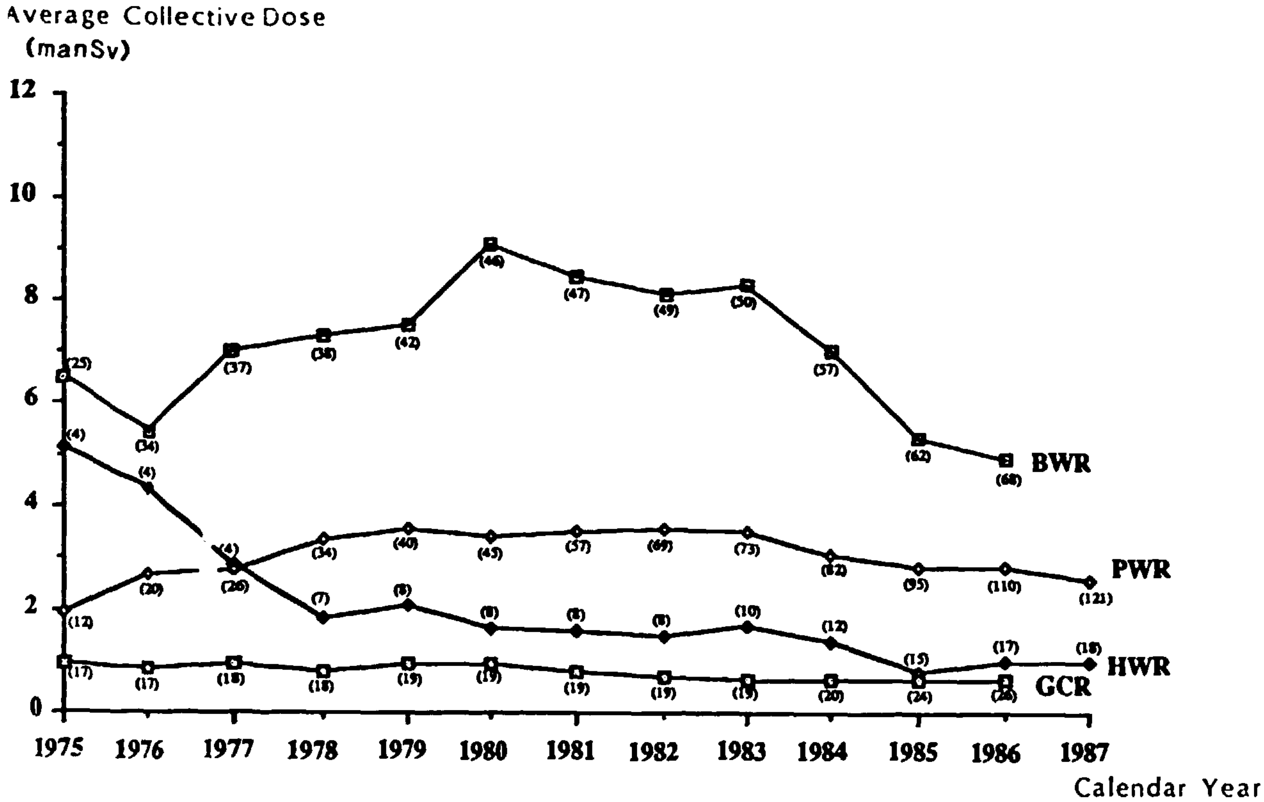

Figure 1. The Average Annual Collective Dose for the Main Reactor Types in the OECD Member Countries
[The number of units included in the calculation is shown in brackets]

The operation of BWRs involves the highest average annual collective doses of all four reactor types. The picture is dominated by two countries, the USA and Japan, which operate 70 per cent of all BWRs in the OECD countries. For the USA the peak value was reached in 1980 with an average collective dose of about 12 manSv/reactor unit. Since then there has been a downward trend in the USA and for 1986 the collective dose has decreased by a factor of about 2 [BNL 1988]. A further decrease has been experienced since then and the average collective dose was for 1988 about 5 manSv/reactor unit [INP 1989]. In Japan the average collective dose for BWRs was 3.6 manSv/reactor unit in 1986 as compared to 5.5 manSv/reactor unit in 1980, and 8 manSv/reactor unit in 1976 [NUS 1988]. In other countries, however, like Sweden and Finland, the operation of BWRs is characterised by low doses to the personnel. The average annual collective doses at Swedish and Finnish BWRs are below the average for BWRs as given in Figure 1 and are typically in the order of 1 to 2 manSv per reactor unit and year.

For GCRs and HWRs, comparison by country is irrelevant in that essentially only two countries operate these types of reactors, the United Kingdom for GCRs and Canada for HWRs. Therefore, the curves in Figure 1 reflect essentially the experience in these two countries. It is seen that an important reduction in the collective dose has taken place for the HWRs. This is due to a major programme on dose reduction which was carried out in Canada, involving reactor operators as well as designers. Since the beginning of 1970 a considerable reduction in collective doses has been experienced, as it is shown in Figure 1 [VTV 1986]. This has led to a collective dose per reactor unit which is comparable to that for GCR reactors, the operation of which is characterised by very low doses to workers, collective as well as individual.

The differences in collective doses between countries are depending on many factors, e.g., the reactor type and age, the design of the reactors, the mode of operation, the level of maintenance and back-fitting and the efforts spent and strategy used as regards occupational dose control. Also the dose recording criteria and practices might influence the results to some extent. No attempt has been made in this report to quantify these different factors.

The downward trend in the collective exposure in the USA, which to a great extent dominates the overall picture as far as LWRs are concerned, is partly due to completion of many of the authority mandated safety actions. Also, new plants with various dose reduction measures incorporated in their designs have gone into operation. The increase in the capacity factors is also playing a role in keeping doses low [BNL 1988]. Last, but not least, one should keep in mind that lessons learned from past operation, ALARA efforts, as well as results from dose reduction projects are beginning to have a positive impact on the results worldwide in terms of reduced collective doses. The increased attention which is put on the question of occupational exposure by authorities, plant management and the workers themselves are additional factors influencing the situation.

Finally, there is a trend worldwide towards introducing larger reactors. Several studies have been conducted on the influence of reactor size and age on the resulting occupational exposure. A study from 1987 [HIL 1987] shows for PWRs in seven European countries, Japan and the USA, that modern plants (1-5 years old) experience a lower annual collective exposure than plants which are of an intermediate age (6-10 years old) and in particular old plants (11-16 years old). As regards the plant size the study shows that (for PWRs larger than 400 MWe) the collective dose tends to increase when the plant size increases. A study by the CEPN in France [CEP 1989] also concerning PWRs seems to confirm the above results. This type of comparison has to be interpreted with great care, however, because the results depend on many factors, for example how the classification of age and size is made. Nevertheless, it seems to be clear from these studies that lessons have been learnt from operating experience as well as from research projects, and improvements have been made and incorporated into new reactor designs.

Individual Doses

The individual doses to personnel working in nuclear power plants are monitored by dosemeters, usually of thermoluminescence (TLD) or film badge types. The average annual individual doses for personnel working in the four main types of nuclear reactors are shown in Figure 2.

It appears from Figure 2 that the levels of average individual doses to the personnel working in the reactors are about 3 to 5 mSv per year except for GCRs where a considerably lower average dose is experienced, which at least partly is linked to the design of this reactor type. The levels seem to be stabilising around the values mentioned. For the HWRs comparatively large fluctuations are experienced, partly depending on the relatively limited number of reactors of this type in operation and the policy of employing a smaller number of persons. For example, the number of exposed workers, i.e. those with a measurable dose during a year, in Canada is much lower compared to other countries, about 250 per reactor as compared to more than 1000 in some other countries [PIC 1989]. The sharp decrease in individual exposure in HWRs until 1978 is a result of the above mentioned dose reduction efforts which were introduced by the Ontario Hydro utility.

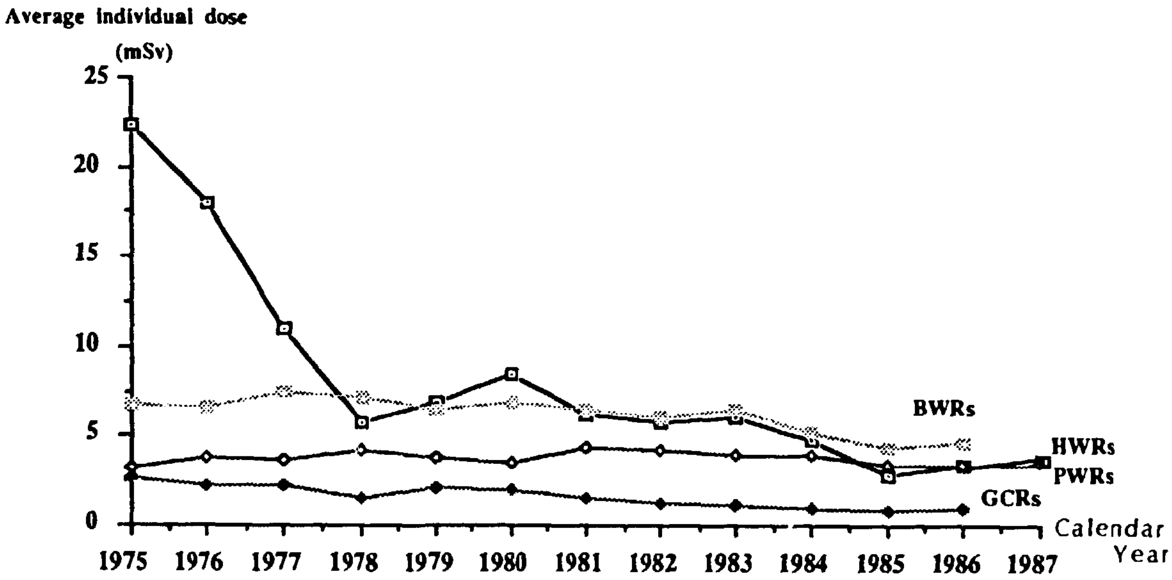

Figure 2. Average Annual Individual Doses for Personnel Working in the Main Reactor Types in the OECD Member countries (Information from France is not included)

The information given in Figure 2 has to be interpreted with great care since significant differences exist between countries in the way they record and report individual doses. In some countries all workers entering the plant area are given a dosemeter and are included in the calculation of the average doses, whereas in other countries only those who have worked in a radiological environment and received a measurable dose are included. Moreover, a possibility exists to deliberately keep individual doses low, as an average, by employing more and more people to do the work at the plants. It is, therefore, prudent not to draw any firm conclusions concerning the general radiological situation at each type of nuclear power plant from the statistics given.

Although the average doses to workers in nuclear power plants are fairly low (the individual dose limit recommended by ICRP is 50 mSv per year), there exists groups of workers that receive considerably higher doses than the average. Therefore, in most countries there are requirements to report doses in dose intervals in order to give the distribution of the doses among the workforce. An example of an individual dose distribution is given in Table 2, also indicating the number of persons monitored but with no measurable dose [NRC 1989].

Table 2

DISTRIBUTION OF INDIVIDUAL DOSES FOR WORKERS
IN US LWRs, 1986 [NRC 1989]

Dose Interval (mSv)	Number of persons	
	BWR	PWR
Not measurable	49.889	51.675
<1	17.456	30.523
1-2.5	6.168	10.428
2.5-5.0	5.093	8.280
5.0-7.5	3.036	4.822
7.5-10	2.135	3.069
10-20	5.099	5.599
20-30	1.429	1.244
30-40	354	239
40-50	45	30
>50	-	-

4.2. Dose Breakdown

The main part of the annual collective dose in operating nuclear power plants comes from maintenance, backfitting, inspections, refuelling and associated service activities. The critical periods are the annual shutdowns. For example, concerning LWRs, between 70 and 80 per cent of the annual collective dose can be attributed to the annual shutdowns [CEC 1987]. Also in GCRs and HWRs a significant part of the collective dose comes from maintenance and service operations during shutdown.

If one examines the distribution of the annual collective dose between utility personnel and outside personnel, including contractors, there is a difference between LWRs on the one hand and GCRs and HWRs on the other. For the latter reactor types, which experience much lower collective doses, most doses are received by utility personnel, the relation between doses to utility and outside personnel being in the order of 70 to 30. For LWRs, on the other hand, the relation is the opposite, i.e, about 30 to 70.

The main reason for this difference is the way in which the reactors are designed and operated. The HWRs and GCRs are of a multi-unit design and with the ability to do refuelling and some other maintenance work during operation. (Some of the GCRs, however, are at present refuelled off-load). These features of HWRs and GCRs greatly influence the personnel management strategy, for example by reducing the pressure on the utilities during shutdown periods to employ a large number of contractors. To a large extent, the work can instead be done by permanent utility staff and be spread more evenly throughout the year [BRO 1989, VIV 1989].

To illustrate some of the above facts Table 3 and Figure 3 are included.

Table 3

DISTRIBUTION OF COLLECTIVE DOSES BETWEEN DIFFERENT TYPES OF WORK FUNCTIONS AT A TYPICAL CANDU-STATION WITH 8 OPERATING UNITS IN 1988 [VIV 1989]

WORK FUNCTION	COLLECTIVE DOSE (manSv)	
Operators	Internal	0.76
	External	0.57
	Total	1.33
Control Maintenance	Internal	0.27
	External	0.48
	Total	0.75
Mechanical Maintenance	Internal	0.79
	External	0.48
	Total	2.27
Service Maintenance	Internal	0.14
	External	1.48
	Total	0.30
Others*	Internal	0.10
	External	0.05
	Total	0.15
Attached*	Internal	0.52
	External	1.40
	Total	1.92
Station Total	Internal	2.62
	External	4.16
	Total	6.78
Fuel Handling**	Internal	0.26
	External	0.61
	Total	0.87

* Others includes: Variuos services activities
Attached includes: Utility Construction Workers, Contractors, Health Physicists

** The Fuel Handling dose information is included in the station total under the Work Groups "Control Maintenance" and "Mechanical Maintenance".

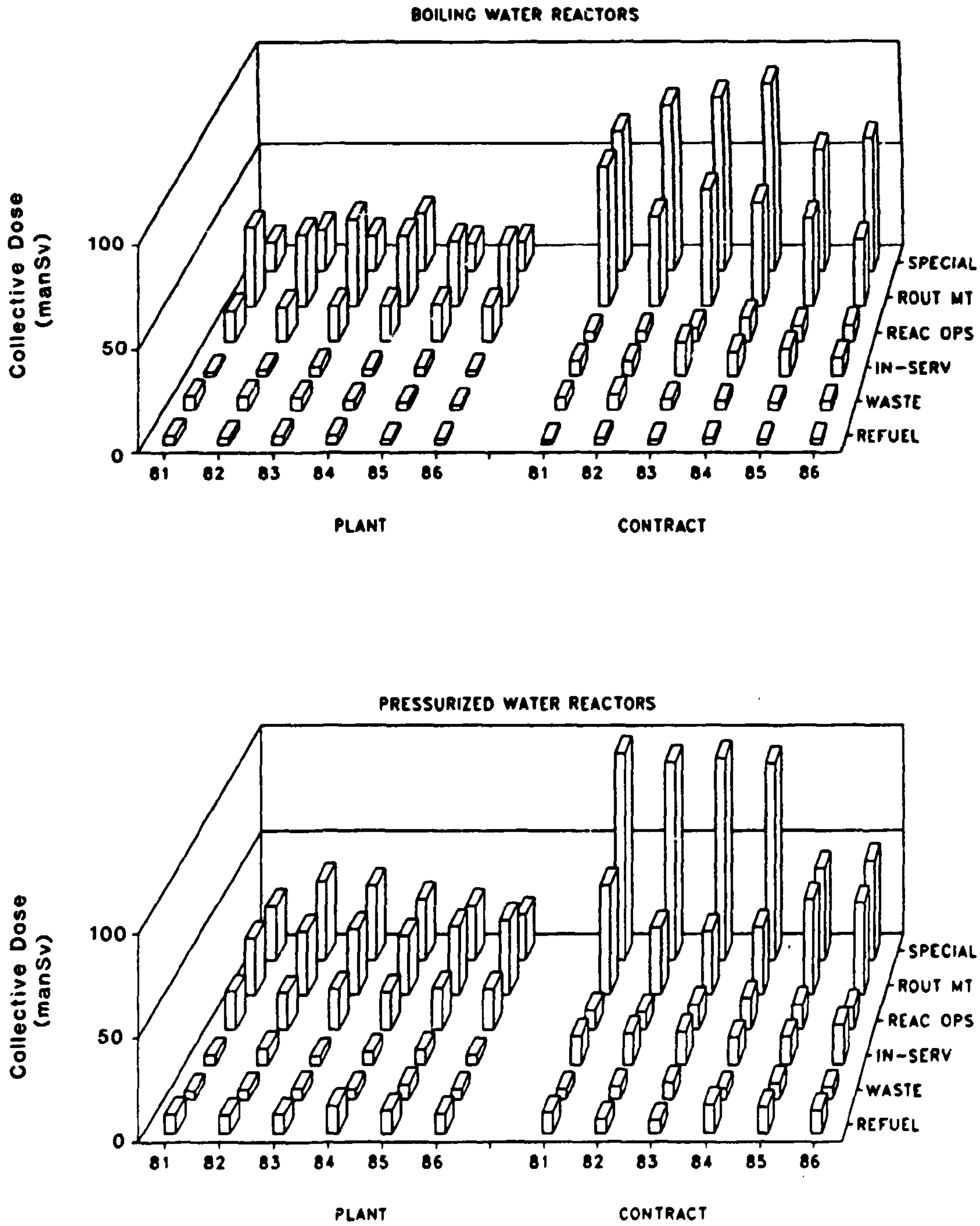

Figure 3. Distribution of the Collective Dose by Work and Personnel Type for US LWRs, 1981 - 1986 (NRC 1989)

Figure 3 which concerns light water reactors shows the distribution of collective dose between plant personnel and contractors for the main works contributing to the collective dose in US reactors and illustrate that:

i) Special maintenance is contributing significantly to the overall exposure of workers at US LWRs, and

ii) Contractor personnel in US LWRs experience a considerably higher collective dose than plant personnel.

A similar situation is experienced in other OECD countries operating light water reactors.

The doses received at HWRs are illustrated in Table 3 by an example which shows the distribution of collective doses among work functions in a typical CANDU-reactor station with 8 reactor units.

From Table 3 it can be seen that:

1) Maintenance activities contribute significantly to the annual collective doses;

2) Attached personnel, including contractors, only account for about 30 per cent of the collective dose; and

3) Internal doses mainly from tritium contribute significantly to the collective dose.

Finally, operations associated with on-load refuelling are among the most dose intensive routine tasks in CANDU-reactors.

As pointed out above, a main contributor to the annual collective dose is maintenance. Different types of maintenance activities contribute also to the distribution of individual doses, placing maintenance workers as one of the highest exposed groups in nuclear power operation for all types of reactors. In Figure 4 are shown average individual doses for some categories of EdF personnel working at French PWRs.

A further dose breakdown would be to analyse the different works in detail. For example, one of the most important maintenance works in PWRs, steam generator work, includes sub-works of different kinds, repetitive as well as those of a more special character. Eddy current testing, for example, is an operation which at present is done on a routine basis to test the condition of steam generator tubes. Examples of more specialised operations related to steam generator maintenance are sleeving and tube plugging.

To manage radiation protection activities at a nuclear power plant it is necessary, as has been pointed out earlier, to know the details of the works to be carried out as well as the details of the work places including the radiation levels at these places. Therefore, in many plants extensive planning, surveillance, supervision and follow-up systems have been established, giving the possibility for those responsible for radiation protection to follow day by day the outcome of a particular work in relation to plans set up in advance and get information to enable them to establish links between a particular dose for a person or persons and a job. Knowledge of such links is of paramount importance in managing and controlling doses at nuclear power plants. Such a system will make available different types of data of relevance to radiation protection to be used, for example, when evaluating performance of the works or tasks. An example of such data is shown in Table 4. This type of dose data, complemented by information on the number of persons exposed, the work method used and the protection applied are necessary when evaluating the performance of a particular job.

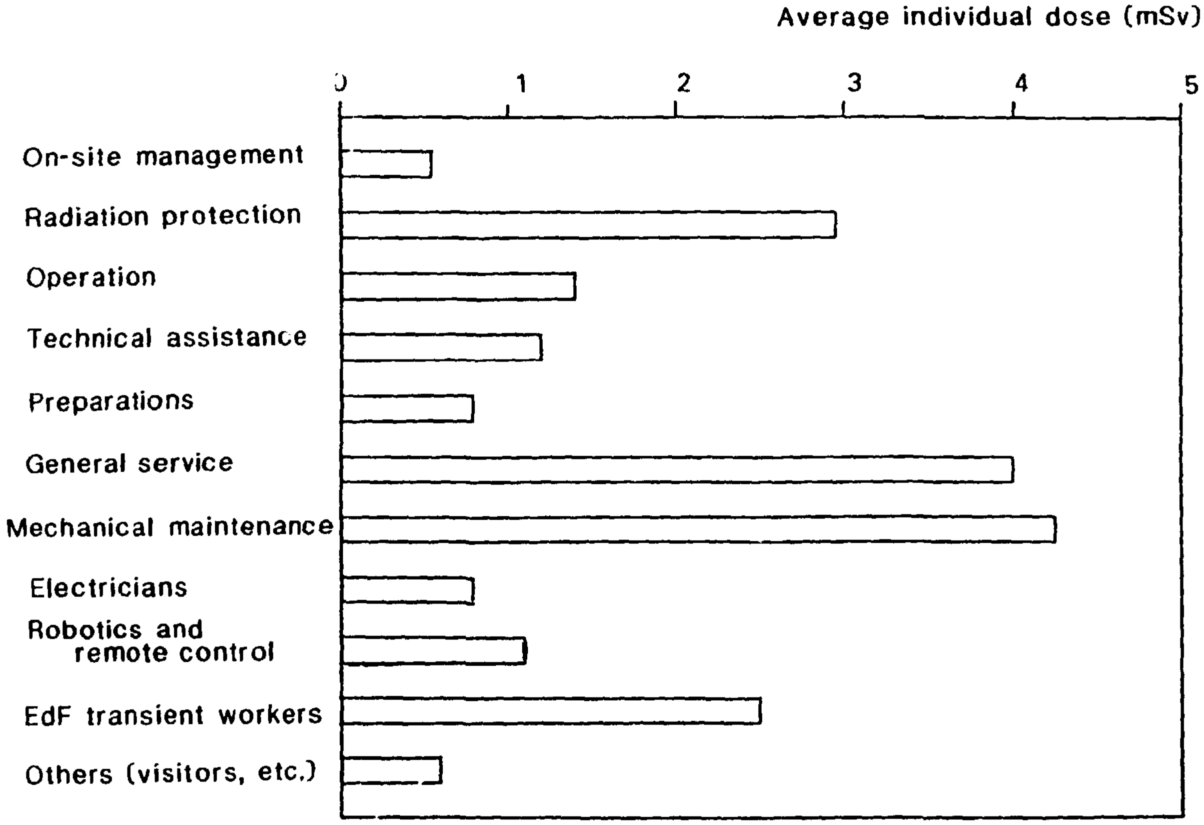

Figure 4. The Average Individual Dose for Different EdF Personnel Categories in 1987 [EdF 1987]

Table 4

AN EXAMPLE OF TASK RELATED DOSE DATA [ERI 1988]

SHOT PEENING IN RINGHALS 4 1987	mmanSv
Preparation, mounting equipment, test runs	28.7
Mounting of plate, rubber carpet in channel heads	2.2
Mounting, demounting and turning of spider	12.3
Reparation of spider	2.4
Work on cameras	3.4
Work on trunk	6.5
Work on pusher, retur cyclone	0.9
Work on hoses	6.3
Work on bead generator	5.5
Work on strainer	1.1
Almen tests, etc	8.7
Filter changes	0.7
Cleaning and inspection of channel head	14.4
Unspecified doses	23.5
Clear out and transportation of equipment	15.4
Erection and dismantling of tents	7.9
Decontamination of equipment	7.9
Health physics	11.6
TOTAL:	159.4

The relevance of this type of information is further discussed below.

5. OCCUPATIONAL DOSE CONTROL

5.1. General

A reduction in occupational exposures at nuclear power plants is desirable, not only because it affects the health and safety of plant personnel, but also because it enhances the safety and reliability of plants by making their operation more efficient and economical. Considerable effort is therefore being spent in this area.

As the collective dose is the sum of all individual doses, its value will be influenced by the dose rates, the duration of jobs and the number of involved workers. These factors can be influenced by technical as well as administrative measures. For example, the dose rates are mainly dependent on how well the sources of the exposure are managed. The number of man-hours is in turn dependent on work requirements, work organisation and training, use of robotics, etc.

The following discussion which summarises some of the findings of recent research and development work, can be conveniently split into two parts, the control of sources and the control of exposure duration. Furthermore, safety requirements are treated separately. Finally, state-of-the-art ALARA programmes are reviewed and the importance of establishing such programmes for the purpose of controlling occupational exposure is stressed.

5.2. Control of Sources

Control of sources of exposure means control of radioactive products, their production, transport, dispersion and removal from the reactor systems. In LWRs the main sources of radiation fields are the corrosion products, and to some extent, the fission products. It has been estimated that about 85 per cent of the collective dose at LWRs is due to radioactive corrosion products. Of these the main contributor is cobalt-60. A key factor in influencing exposure is the reduction in radiation field build-up. Different techniques are being used which fall within four main categories:

i) cobalt replacement and reduction
ii) preconditioning
iii) water chemistry control
iv) decontamination

Cobalt is present as a low-level impurity in structural materials and at high levels in certain hardfacing alloys used for components which must resist wear. Sources of cobalt originate inside the reactor core as well as outside the core. It has been estimated that "in-core cobalt" makes a significant contribution to contamination of reactor systems outside the core, and, therefore, it should be of importance to try to eliminate the in-core sources of cobalt such as the cobalt content in fuel assemblies and core support structures. Research and development is consequently in this direction (BNL 1988, BNL 1989, DUB 1985). There is also a tendency to specify lower cobalt levels than were previously common or to avoid cobalt as far as practicable in out of core structures, both in new designs and in replacement components. The following table lists some examples of components where cobalt reduction is of importance and where technology is available (EPR 1989).

Table 5

EXAMPLES OF COMPONENTS
WHERE COBALT REDUCTION TECHNOLOGY IS APPLICABLE AND AVAILABLE

- Co-free alloys for pins/rollers of BWR control blades
- 400 series stainless steel for feedwater flow control valves
- Zircaloy fuel grid spacers for PWRs
- Low cobalt Inconel 600/690 for replacement steam generator tubing (3 times reduction)
- Low cobalt 304 stainless steel for BWR control blade tubing/sheathing (2-5 times reduction)

The importance of preconditioning and use of high-temperature filters to reduce the deposition of corrosion products on system surfaces is being studied in man countries. It is still too early to draw conclusions concerning the benefits of installing filters to reduce the overall radiation fields, but the research continues [DUB 1985] and some promising results are available [BNL 1989]. Preconditioning, however, is commonly used, for example when replacing large components. Preconditioning methods have been applied to large replacement components, and there is always an element of preconditioning in the start-up procedures for new reactors.

It was recognised rather early that the chemistry of the primary water is an important factor as regards the build-up of corrosion products in the reactor system. Large resources have been devoted to try to understand the processes which govern these phenomena. Strict guidelines specific to the reactor type and the particular design are therefore applied in order to control and reduce radiation build up, by minimising transport and activation of cobalt containing corrosion products [BNL 1989].

Chemical decontamination is now a well-established technology for removing radioactive contamination and thereby reducing the radiation fields in the work places. So-called dilute reagent solutions have been developed which remove activity effectively and which preserve materials' integrity. A number of successful decontaminations for BWR piping systems and PWR steam generator channel heads have been carried out. For HWRs in Canada, full primary system decontamination of the reactor with the fuel in place is routinely being carried out.

5.3. Control of Exposure Duration

The problem of reduction of time in radiation fields or the exposure duration is being approached by different methods. One is to improve the reliability of materials so that they require less inspection and maintenance, another is to develop remote tools and robotics, and a third approach is to introduce efficient work management programmes including work planning, training, optimisation considerations, etc.

In recent years considerable progress has been made in developing and qualifying better materials for nuclear power plants. Materials have been developed that improve the reliability of components, make them less susceptible to corrosion, reduce their cobalt content significantly and make them more amenable to decontamination. Improvements have also been made in the reliability of major components, allowing systems to be designed for significant dose savings. The design of internal reactor coolant pumps for BWRs is one such example. Inspection and maintenance of external primary circuits of BWRs of old design is one significant contributor to the collective dose. However, by replacing them by internal recirculation pumps new designs have significantly reduced this source of radiation.

For PWRs the paramount problem experienced so far is the cracking in the tubes of the steam generators, which on many occasions have forced the utilities to make expensive repair operations and sometimes to replace the whole set of these large components. The repair and replacement operations require large doses. The replacements of the lower assembly in four American plants, Surry 1 and 2 and Turkey Point 3 and 4, for example, have caused collective doses of between 13 and 21 manSv for each reactor [MOR 1989, Figure 5].

To combat these problems research and development work has been directed in several directions, some more appropriate for new plants or those replacing their steam generators, others more suited to improve the resistance of existing generators and thereby extend their service life. The development of more corrosion resistant materials as well as processes to treat existing tubes, such as shot peening, are two examples.

Robotics and remote systems are beginning to play an increasingly important role in the nuclear reactor field. Their use is especially important in accident situations when high radiation fields are present. They are now also used in more routine tasks to reduce the occupational radiation exposure of personnel, for example in the maintenance of steam generators and for in-service inspections of highly contaminated piping and components [BNL 1989].

Apart from measures to reduce the dose rates and critically consider all working requirements including proposed safety measures, there are aspects related to work management which need to be included in the programme to control doses at nuclear power plants. Non-technical factors which have an influence on the radiation doses are, for example, work planning and co-ordination, training, optimisation of work forces and efficient health physics practices. Moreover, an excellent way of controlling doses in relation to critical jobs is the use of real-time dosimetry and job-related dosimetry. Job-related dosimetry if used properly has proved very useful also in giving indications on where efforts should be directed next time a similar task has to be carried out.

Work management in a broad sense, i.e., including all measures to control and reduce the exposure time and the number of manhours in radiological environment, is recognised as part of occupational dose control. However, it has received in the past less attention then the area of measures to reduce sources of exposure. Several examples of good results of work management exist however (WAH 1988), and further work in this area should be encouraged in order to obtain the best possible "working culture" in the area of nuclear power plant operation with positive influences on radiation protection and nuclear safety.

5.4. Nuclear Safety Requirements

In recent years, accelerated efforts to improve nuclear safety have resulted in a large variety of requirements affecting the design and operation of nuclear installations. These requirements, formulated by designers and utility operators or imposed by regulatory authorities, may involve exposing personnel to tasks in the presence of radiation beyond those originally expected to be associated with the normal running of the plants. This situation results in an increase of actual doses being incurred by workers. The benefit of this is represented by a decrease of the potential exposure of the public and the workers, due to the possible decrease of the probabilities of accidents and/or the projected doses associated with these accidents.

There has been a growing concern among experts that, in some cases, this situation seems to show an imbalance, not only in the actual level of risk attributed to two different groups of people (workers and population), but also in the response to two different needs (safety of plants and protection of workers) both of which are regulatory requirements.

In order to study this particular question an NEA group of experts was established a few years ago. In the following are given some of the main findings of this expert group [NEA 1988].

The group thought that part of the differences that exist between countries in the levels of exposure of workers is to be attributed to differences in nuclear safety and regulatory requirements in the various countries. In particular the group concluded that the decision-making process leading to the nuclear safety measures is specifically focussed on the reduction of risks to the public, and, in many countries, it does not normally include an explicit consideration of the possible implications for the protection of the workers. On the other hand, the group concluded that safety measures which reduce the probability and/or consequences of accidents may well be beneficial to the workers who might be involved with in-plant accident recovery and clean-up operations.

The difficulties of quantifying the net benefit of introducing new safety requirements were also pointed out. Several factors have a bearing on this, for example the scarcity of the specific data which are needed to assess the relation and the balance between the occupational exposure and the nuclear safety requirements. In fact, although there has been a major increase in the availability of occupational dose data for high dose jobs and specific tasks, it is still difficult to assign the portion of such doses that should be allocated to nuclear safety versus productivity improvement, for example.

Finally, the group recommended among other things the following:

i) Before introducing any in-plant nuclear safety-related measure which would entail a significant worker dose expenditure for its implementation and future maintenance, regulatory authorities should consider the inclusion in their decision-making process of a procedure of analysis which determines whether the relevant benefits and detriments are optimised.

ii) The continuing development of methods and techniques (i.e., materials selection, plant layout, improved inspection techniques, robotics, etc.) to reduce occupational exposure should be encouraged.

iii) Progress of safety technology and operational techniques should be regularly reviewed with a view to appreciating whether simplifications in regulatory requirements (i.e., in-service inspection frequency) can be warranted.

iv) The assessment and recording of task-related dose data and the establishment of more uniform dosimetric procedures aimed at inter-plant comparability of dose data should be encouraged.

v) The development and standardisation of electronic dosimetry and automatic data management systems for the determination of task-related doses should be accelerated.

vi) An international exchange of information on occupational exposure data and optimisation procedures should be prompted. Differences among these data should be studied to evaluate the potential role played by different design, operation and safety philosophy approaches, dosimetric and equipment reliability data uncertainties, and other relevant parameters in producing these discrepancies. In this respect it would be very useful to plant designers and safety authorities if efforts were made to create an international task-related data base.

As pointed out in the introduction, the NEA has taken up this last point in its programme of work in order to contribute to the process of facilitating internationally information exchange on occupational dose data, in particular task related doses and dose reduction techniques.

5.5. ALARA Programmes

For a long time, the optimisation principle, introduced in 1977 by the International Commission on Radiological Protection (ICRP), also referred to as the ALARA principle, remained a subject of theoretical debates within the radiation protection community. Then, as the emphasis was more and more laid on maintaining individual and collective occupational exposures as low as reasonably achievable rather than just assuring compliance with dose limits, the optimisation principle has progressively gained ground during the last years within operational radiation protection programmes [CRO 1988].

The rationale in applying ALARA in nuclear power plants is to achieve a level of exposure low enough to guarantee a sufficient protection of the workers but preserving at the same time the economical viability of the installations. This means to first identify and then implement the dose control techniques and methods that will assure the best trade-off between the reduction of doses and the increase of protection costs. This selection process, which is the heart of the optimisation of protection principle, explains why the use of decision-aiding techniques, such as cost-benefit analysis, has so heavily focussed the attention of those involved in the promotion of ALARA. However, from the practical implementation point of view, the means to achieve ALARA goals basically depend on motivational and organisational arrangements. In this context, the decision-aiding process to select protection actions is only a part of what constitutes ALARA programmes.

Looking at the experience so far, there are many different approaches to provide an effective framework for implementing ALARA [MUN 1988]. These range from general recommendations as part of classical radiation protection programmes to very formal procedures applied in a systematic way. Formal ALARA programmes are much more developed in the USA than in other countries, particularly in European countries, but a large variety of arrangements can be found. Whatever the degree of formalism which is finally adopted, the basic components of an ALARA programme can be grouped under three major headlines: commitment, organisation and tools, procedures and data.

Commitment to ALARA

The key to a successful ALARA programme is the commitment at all levels of those who are in charge of regulating, controlling, managing and operating radiation protection in nuclear power plants. This commitment does not mean necessarily that the ALARA procedures to be implemented effectively need to be codified in detail and become part of regulations. This aspect has been a matter of misunderstanding in the past and probably explains the "prudence" which still characterises many organisations. Instead, ALARA is primarily a "state of thinking" that must pervade the various levels of management and workforce.

The ALARA principle has been progressively incorporated into national regulations where in most cases it appears as a general requirement. Different views exist in the various countries about how this requirement can be transformed into practical rules at the regulatory level. In general, a large flexibility is left to utilities to develop ALARA programmes which best fit with their specific needs.

Apart from the regulatory constraint, the motivation of the various actors can only be gained when a clear management support exists. This support means that the management is not only a strong advocate of the ALARA philosophy but also provides the technical and financial inputs. The management commitment must also be translated into responsabilities which are distributed among the key personnel. Results from a survey in US nuclear power plants show that the commitment of the management is considered as the cornerstone of a successful ALARA programme in 95 per cent of the stations [DIO 1985].

Awareness of personnel is the complementary element of management commitment to ALARA. The basic ways to gain this awareness among the personnel are the following:

- Education and training;
- Communication and information about the objectives and the performances;
- Incentives and suggestions to provide a vehicle for the workers to express their concern and thereby influence and improve the situation.

Practical arrangements to implement these various functions are numerous and many programmes already exist.

ALARA organisation

A strong commitment from both management and workforce can be sufficient to achieve ALARA goals. However, specific organisational structures may be useful to strengthen operating efficiency. There is no universally applicable organisation. Each organisation will have to develop its own structure. It is important that ALARA structures, to be effective, rely on the health physics organisation and aim at improving the co-ordination between the different individuals or organisational components involved in radiation protection. A significant aspect of this need for co-ordination is, for example, between designers and operators and, among the latter, between utility personnel and contractor workers in cases where large numbers of contractors are used. From the experience so far the most commonly used structures are the following:

- ALARA committees. The purpose of these committees is to provide a multi-disciplinary forum for the discussion of the main radiation protection issues and choices. The members of the committees are generally the responsable persons of the various sectors involved in the operations concerning the plants (construction, maintenance, engineering) including the radiation protection sector. The role of the committees is mainly to make recommendations and work out ALARA objectives for those responsible for the plant operation and the utility management. The committees should also be in charge of proposing the strategy for improving radiation exposure conditions, when such actions are considered necessary.

- ALARA co-ordinators. These are basically in charge of implementing the ALARA procedures (described in the next paragraphs). They not only form the link between the ALARA committees and the various people involved in the operations, but also ensure the interface between these. ALARA co-ordinators are working in close collaboration with the health physics sections and decide about corrective actions to be undertaken during the course of the actual operations. Well trained co-ordinators are considered as fundamental elements for a successsful ALARA programme.

ALARA tools and procedures

No ALARA programme can work routinely and efficiently without the support of specific tools and procedures which aim at providing a structured and clear approach to decision making related to ALARA. A large variety of ALARA tools have been developed to deal with the various facets of radiation protection problems of nuclear power stations at the design and operational stages. These techniques are certainly not universally used but are gaining wider acceptance. Despite their variety, it is however possible to group them into three large categories.

- ALARA reviews and audits. These tools allow the identification of problems by screening situations according to structured approaches. They are generally based either on "check lists" or "analytical trees" and can be applied at any step of the operations (pre-job or post-job reviews, for example). The use of such tools may result from management initiatives (ALARA committees or co-ordinators) or systematically where the anticipated exposure for an operation is greater than a pre-established value.

- Predictive ALARA plans. Their role is to provide the operators with the dose profile to be expected from the actual work together with acceptable levels of variation and trigger levels for action. ALARA plans generally allow the fixing of targets that will serve as reference to motivate and to gain the commitment of operators.

- The "ALARA procedure". ALARA audits and reviews or predictive ALARA plans do not optimise radiation protection by themselves. They only allow the identification of problem areas which need consideration. But at all levels where these techniques are applied, usually a number of alternative protective actions exist, e.g., choice of type and layout of equipment, special tools, whether to decontaminate the working area, trade-offs between individual or collective doses. The so-called "ALARA procedure" is a structured approach that can help in assessing and comparing protective actions in a systematic and coherent way with a view to identifying the one which leads to the best allocation of resources within the existing constraints [CRO 1988].

The tools and procedures briefly described here and above cannot be effectively applied without the use of ALARA criteria which are expressed either in terms of reference values (the monetary cost of the detriment for example) or performance indicators. These criteria are generally specific to the utility and developed within its internal code of practice to give guidance to the operators.

ALARA data bases

The establishment of dose and cost data bases is of prime importance in the pursuance of the various ALARA tools and procedures. There are two potential inputs for establishing such data bases: modelling of the situation and use of past experience data. The most common way seems to be by the use of past experience. However, modelling techniques can play a complementary role especially for estimating ambient dose rates in the various areas of the plants. Different systems relying on computer codes have been developed or are still under development to estimate ambient dose rates in areas where mapping based on direct measurements is impossible, as well as to determine the relative contributions of the different components to the radiation field in a given area.

As far as past experience is concerned, most of the relevant information results from the various dose monitoring systems in use in power plants. The dosimetry systems introduced in the first operating nuclear power plants were intended primarily for monitoring and the control of individual exposures of workers for regulatory purposes, i.e., the demonstration that dose limits had not been exceeded. Such a posteriori systems are unfortunately not able to provide the kind of data required in ALARA programmes, because no link can be reasonably established between the doses, generally evaluated every month, and the different jobs performed by workers day by day. The weaknesses of these systems pushed utilities to develop operational monitoring systems (like "automatic job dosimetry" for example) based on TLDs or integrating dose meters to be used in addition to the regulatory individual dosimetry. This type of dosimetry allows a link to be made between the daily doses received by workers and a set of generic operations taking place during the operation and maintenance of the plants. Although these systems represent a real improvement, their usefulness for ALARA evaluation critically depends on the tasks being undertaken and their coding. Based on the need for more relevant monitoring systems, operators are now pushing towards the development of a third generation system, the real-time dosimetry, able to have a direct access to the fraction of individual dose and time associated with the various elementary tasks carried out during a given operation.

6. LEARNING FROM EXPERIENCE

The role of past experience

Examination of the evolution of exposures associated with routine or special repetitive maintenance operations in nuclear power plants shows a decreasing trend for collective exposures throughout successive operations. This evolution is generally accompanied by a simultaneous reduction of the mean individual doses for the various categories of operators involved. Figure 5 illustrates the typical "dose profile" for a large maintenance operation, steam generator replacement. As can be seen, significant progress has been made with regard to personnel exposure, which is closely related to the reduction of the work duration.

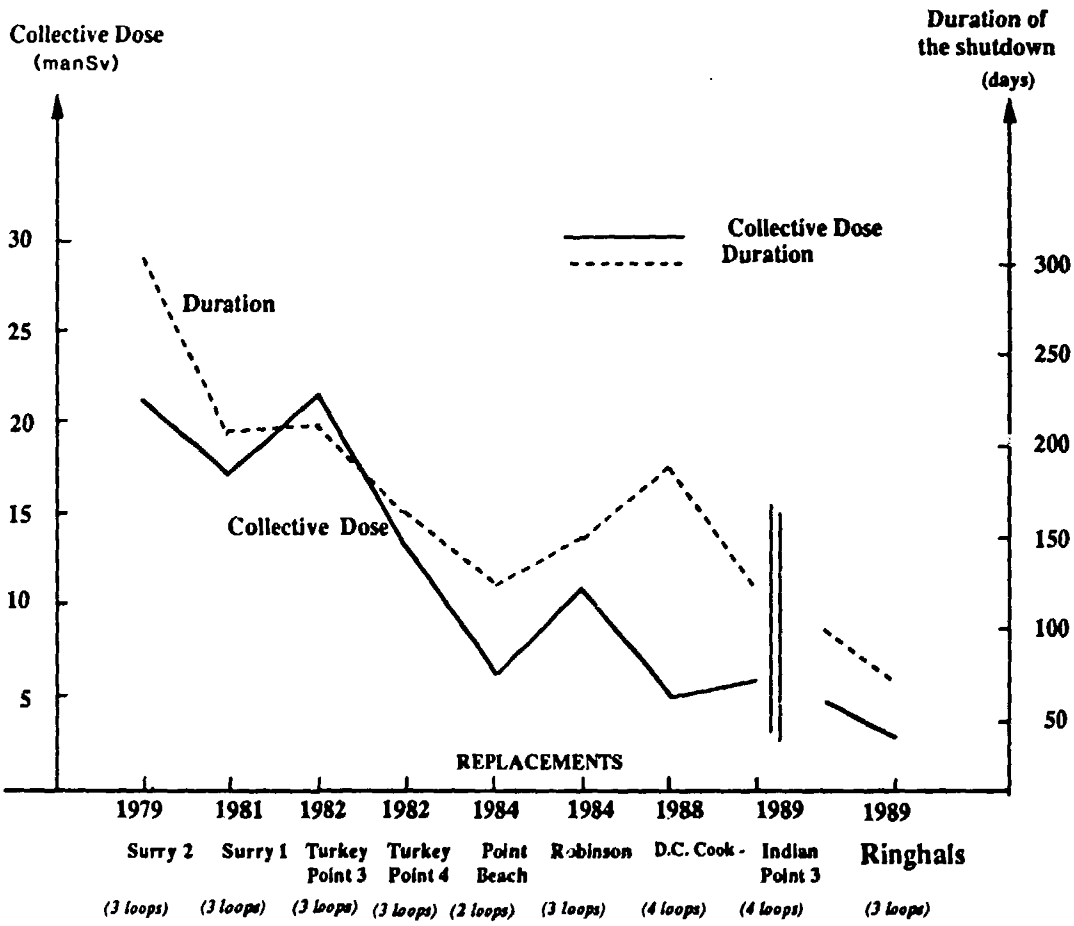

Figure 5. Evolution of the Occupational Collective Dose Associated with Steam Generator Replacement Operations

A more detailed analysis of this evolution, as one of many other operations, would show that the reduction of exposures results from changes in the technical approach, improvements in the reliability and the performance of tools but also from the "learning" effect which allows operators to integrate the lessons from previous operations. This last aspect is of particular importance when operations are performed at the same or similar stations with more or less the same personnel. Studies have also demonstrated that the dosimetric burden of the first operations can be substantially reduced if radiation protection considerations are present at the design and preparatory stages, as well as during work performance.

In an ALARA perspective, the objective of a good radiation protection practice for maintenance work should be to lighten the dosimetric burden of the learning phase of repetitive operations by identifying and implementing cost-effective protection actions to reduce sources and exposure times. This can only be achieved if designers and operators have all available information to estimate the consequences of their choices.

The shift from monitoring and control to management

It is worthwhile to emphasise here the impact of the introduction of the ALARA principle in the organisation and control of radiological protection. Putting forward the *a priori* estimation of the cost and effectiveness of dose control actions, ALARA introduces a predictive approach which relies on the quantification of the relations between conditions of exposure (sources, ambient dose rates, durations of exposure) and the doses to be received by the operators. This requirement presents a fundamental difficulty as in most cases readily available data are not in an appropriate form, i.e, task related. Even if the work of models, as mentioned above, can play an important role at this level, predictions have to be based on information from past experience. Consequently, predictive analysis, the follow-up of operations and past experience, are closely interrelated elements that should provide the foundations of the management of ALARA in practice.

As a matter of fact, the functions to be undertaken to achieve ALARA goals are very similar to those characterising management systems. These functions which are cyclical in nature and are repeated over and over again, each time improving the total performance of the protection, can be described as the following:

- Establishment of objectives (for instance target figures for a given operation);
- Measurement of performances (individual and collective doses related to elementary tasks);
- Comparison of performance measurement feedback information against objectives;
- Identification of the various causes for deviations (dose rates, job durations, mishaps...);
- Determination and implementation of corrective actions (training, organisation, special tools, change in design ...);
- Follow-up to ensure completion of corrective actions;
- Establishment of new objectives.

The procedures developed to implement the objectives (estimates, reviews, audits...) have been briefly described in the previous section. It is noticeable that this iterative management process can be directly integrated within the basic phases of any operation, i.e., planning and preparation (setting up of objectives), accomplishment and follow-up (measurement of performances), feedback (past experience analysis and corrective actions). By extension it is also possible to adopt the same sequential scheme for the evolution of the basic design of plants including backfitting actions as corrective actions. In that case the only fundamental difference is in the time scale for which the various steps are applied.

7. CONCLUSIONS

Operation and maintenance for nuclear power plants imply the exposure of workers. Special procedures, tools and training are needed for work in radiological environment requiring significant efforts and costs for the operators. On the other hand, experience also shows that good radiation protection, in addition to contribute to the health and safety of the personnel, also facilitates the operation of nuclear power plants and makes it more safe and economical. The optimization principle has proven to provide valuable guidance in radiation protection decision making and has been progressively incorporated into regulation and operational procedures in most countries.

Considering the evolution of the occupational radiation protection philosophy during the past decade, largely dominated by the ALARA way of thinking, it is paradoxical to observe that, whilst the difficulties in implementing the principle have been identified at the methodological level, one of the basic obstacles was more prosaically the lack of adequate data to estimate the consequences on individual and collective doses due to any modifications in the conditions of exposures. To reach such level of adequacy it is necessary to develop dosimetry systems allowing the identification of critical tasks or operations as well as the parameters on which to take action to reduce further exposures.

This ideally can only be achieved by means of specific measurements of contact and ambient dose rates at components within areas where operations are to be performed and by the systematic record of the working times and doses received by the workers according to each elementary task related to the operation. Furthermore, it is also essential to benefit from any experience available at other plants having performed similar tasks or operations.

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Reduction in occupational exposure at nuclear power plants is desirable not only in the interest of the health and safety of plant personnel, but also because it enhances the safety and reliability of the plants.

This report summarises the current trends of doses to workers at nuclear power plants and the achievements and developments regarding methods for