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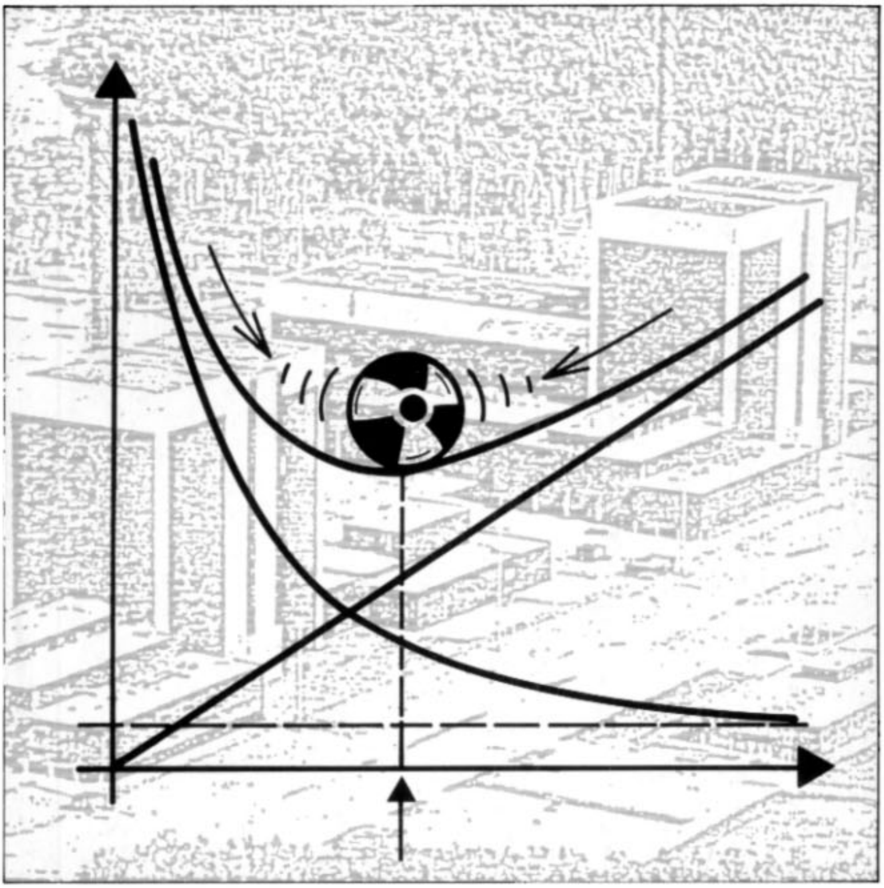
OPTIMIZATION OF RADIATION PROTECTION AT NUCLEAR POWER PLANTS



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OPTIMIZATION OF RADIATION PROTECTION AT NUCLEAR POWER PLANTS

Summary Report of the NKA Project RAS 410

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ABSTRACT

Doses to workers at the nuclear power plants in Sweden and Finland are generally low. The main causes of this situation include appropriate design and lay-out as well as successful operation of and radiation protection at the Nordic plants.

It was investigated whether the optimization principle recommended by the International Commission on Radiological Protection is being applied at the Nordic nuclear power plants or can its use be extended. The report summarizes the main results of the study.

Key words: optimization, ALARA, radiation protection, radiation shielding, in-service inspections, reactor operation, reactor design, nuclear power, energy production, Nordic co-operation, Denmark, Finland, Norway, Sweden

The project is a part of the safety programme sponsored by NKA, the Nordic Liaison Committee for Atomic Energy. The project work has been financed in part by the Nordic Council of Ministers, in part by the participating organizations.

SUMMARY

A considerable amount of experience in occupational radiation protection at nuclear facilities has been assembled in the Nordic countries. The radiation doses to workers at the sixteen operating Finnish and Swedish nuclear power plants are generally pretty low compared with the doses reported from other nuclear power plants in the world. The main causes of the prevailing situation include appropriate design and lay-out as well as successful operation of and radiation protection at the Nordic plants.

Doses to workers at the nuclear power plants shall be kept as low as reasonably achievable (ALARA) according to the optimization principle recommended by the International Commission on Radiological Protection (IRCP). This implies that doses shall be avoided if the cost of the corresponding countermeasure is reasonable in relation to the benefit from the enhanced radiation protection.

There is, however, a need for greater awareness in optimization, a clear and at the same time, deep understanding of what optimization means and of the possibilities for achieving practical improvements in radiation protection. It was, therefore, decided in the Nordic research project to investigate whether the ICRP optimization principle is being applied at routine interventions etc. at the Nordic nuclear power plants or can its use be extended. In the project certain practical applications were examined: radiation protection in in-service inspections, in the use of protective clothing and equipment, as well as in modifications of plant systems and constructions. The project was run by a group with members from radiation protection and safety authorities, nuclear industry and research bodies in the Nordic countries.

In the course of the work, a considerable amount of information about the features of radiation protection at the Nordic nuclear power plants was also collected.

In-service inspections are generally considered to be of great importance to plant safety, reliability and economy. This study indicates that optimization of radiation protection in the strict formalized sense is not usually done at the plants. Apart from dosimetry and other direct radiation protection measures, one of the most important actions for controlling doses during in-service inspections appears to be an active "work management" programme. Such a programme includes prior planning of the work to be performed, preparation of work places, as well as education and training of the personnel involved. Automatization of some parts of the inspections is considered an improvement also from the viewpoint of radiation protection. Another step forward would be the reconsideration of the traditional in-service inspection routines actually used. Such a reconsideration has also been initiated. It is important that the data on doses, on the cost of protection etc. which are available as a result of such developments are collected and evaluated systematically.

Radiation doses to workers are usually reduced by introducing additional boundaries for contaminated areas, by using protective clothing, respirators, temporary shielding etc. Optimization of radiation protection has been studied in cases where protective means are used against radioactive contamination or when temporary shielding is employed. In this study, a cost-benefit formalism was developed to aid decision about protection. As a by-product of this study, a calculational model for the spreading of radioactive contamination at a nuclear power plant was also worked out.

The results show that the protective measures applied today at the Nordic nuclear power plants are fairly similar to the outcome of the more formal optimization analysis performed. This indicates that the practical goals and means at the plants are in fact an ALARA-solution.

In the study of the optimization of radiation protection in modifications of plant systems and constructions at Nordic nuclear power plants, actions that have been taken to reduce doses were reviewed. The aim was to find common factors contributing to the relatively low radiation doses. More than 100 actions were studied, ranging from small modifications to bigger new constructions. The resulting view is that actions to reduce doses are, by and large, based on needs more direct than the optimization considerations. Such are e.g. the needs to avoid high local or general dose rates, or reasons related to the general safety goals of plant operation. Optimization is hardly ever done quantitatively. Instead, it is more like an intuitive process based on the experience and skill of the radiation protection staff.

The actions taken by such staff can generally be considered cost-effective. The study reveals that some of the actions involve rather high costs, reflecting the relatively high weight which the plant operators attach to factors which are not primarily related to radiation protection, but which also imply that doses are kept low.

A general outcome of the project is that a crucial factor from the licensee's point of view is the ease of optimization. What seem to be required are clear, simple and fast methods for routine optimization in the daily work at the nuclear power plants. In order to have a sound basis for the optimization of radiation protection and decision-making, two factors are essential: the availability of good technical competence among the personnel, and a thorough preparation of the protection as well as the decision. Special emphasis should be put on the search for options as well as on the assessment of the resulting doses and the costs involved.

Today, general guidance on the selection of formal optimization methods can be given. There are many details to be improved, however, in order to facilitate this process. There is a need for "rules of thumb" or some standardization that can be used in practical decision-making, i.e. in operational radiation protection work at nuclear power plants. There is also a need for data bases and computerized decision support systems that can be used in more complex cases.

The level of ambition in the radiation protection of workers at the Nordic nuclear power plants is very high. Consequently, it seems that the entire and actual cost of protection is not recognized or assessed in detail, since it is only a minor item in the overall plant economy.

In the project, suggestions are also made to further study optimization of radiation protection at the Nordic nuclear power plants.

SAMMANFATTNING

En avsevärd mängd erfarenheter från yrkesstrålskydd vid kärntekniska anläggningar finns idag i de nordiska länderna. Personalstråldoserna vid de sexton finska och svenska kärnkraftblock som är i drift är generellt sett låga jämfört med de doser som redovisas från andra kärnkraftblock världen över. Huvudskälen till detta förhållande bedöms vara en väl genomtänkt konstruktion och utformning, goda driftresultat och ett gott operationellt strålskydd vid de nordiska anläggningarna.

Doserna till den personal som arbetar vid kärnkraftanläggningar skall hållas så låga som det rimligen är möjligt (ALARA) i enlighet med optimeringsprincipen som rekommenderas av den internationella strålskyddskommissionen (ICRP). Detta innebär att en stråldos skall undvikas om kostnaden för en motåtgärd är rimlig i förhållande till den "nytta" som det förbättrade strålskyddet innebär.

I strålskyddsarbete krävs en klar och fördjupad insikt i vad optimering innebär och de möjligheter en optimering kan bidraga med för att förbättra praktiskt strålskydd. Med detta som utgångspunkt beslöt man inom ramen för det gemensamma Nordiska forskningsprogrammet undersöka om den av ICRP rekommenderade optimeringsprincipen rutinartat tillämpas vid de nordiska kärnkraftanläggningarna, samt om en utvidgad tillämpning är möjlig. Följande praktiska tillämpningar har undersökts: strålskyddet vid återkommande besiktning, vid användning av skyddsutrustning, samt i samband med ändringsarbeten i anläggningssystem och vid nybyggnation. Projektet har utförts av en arbetsgrupp bestående av representanter från strålskydds- och säkerhetsmyndigheter, kärnkraftindustri och forskningsinstitutioner i de nordiska länderna. Under arbetets gång har en avsevärd mängd information med anknytning till strålskydd i de nordiska kärnkraftanläggningarna insamlats.

Återkommande besiktningar bedöms ha stor betydelse för reaktorsäkerheten men även för drifttillgängligheten och ekonomin. Denna studie ger en antydan om att optimering av strålskyddet i strikt mening vanligen inte tillämpas vid anläggningarna. Bortsett från dosimetri och andra direkta strålskyddåtgärder, tycks en av de viktigaste åtgärderna för att kontrollera dosutvecklingen vid återkommande besiktningar vara tillämpning av ett aktivt arbetsstyrningsprogram. Ett sådant program innefattar förplanering av arbetet, förberedelser på arbetsplatsen samt utbildning och träning av berörd personal. Automatiserad provning inom vissa delar av besiktningssektorn bedöms även medföra fördelar ur strålskyddssynpunkt. Ytterligare framsteg skulle troligen uppnås om traditionella besiktningsrutiner omvärderades. En sådan omvärdering är också pågående. Av stor vikt för strålskyddet är att insamla och systematiskt utvärdera dosdata, kostnader för skydd osv. som denna pågående utveckling resulterar i.

Vanligen reduceras personalstråldoser genom kompletterande begränsningar för arbete i kontaminerade områden, till exempel genom användning av skyddskläder, friskluftutrustning, temporära skärmningar osv. Optimering av strålskyddet har studerats i fall där skyddsåtgärder använts vid radioaktiv kontamination eller där temporär skärmning tillämpats. I denna studie har en "konstnad - nytta" formalism utvecklats till stöd för beslutsfattandet i skyddsfrågor. En modell för beräkning av spridning av radioaktiv kontamination i kärnkraftanläggningar har även tagits fram.

De skyddsåtgärder som idag tillämpas vid de nordiska kärnkraftanläggningarna är i tämlingen god överensstämmelse med de resultat som erhölls när en mera formell optimeringsanalys genomfördes. Detta ger en indikation av, att de praktiska mål och åtgärder som tillämpas vid anläggningarna i själva verket är i enlighet med ALARA-principen.

I studien av optimering av strålskyddet vid ändringsarbeten och nykonstruktioner i de nordiska kärnkraftanläggningarna har man följt upp och granskat åtgärder som vidtagits för att reducera doserna. Syftet var att se om det fanns gemensamma faktorer som bidragit till de relativt sett låga stråldoser man har. Mer än 100 utförda åtgärder studerades, omfattande både små modifieringar och större nykonstruktioner. Studien visar att åtgärder som vidtas för att reducera doser oftast baseras på mer direkta behov än på optimeringsöverväganden. Sådana behov kan till exempel vara strävan att undvika höga lokala strålnivåer eller allmännivåer, alternativt orsaker med anknytning till de generella säkerhetsmålen för anläggningens drift. Optimering utförs praktiskt taget aldrig kvantitativt utan närmast som en intuitiv process, baserad på erfarenhet och skicklighet hos strålskyddsledningen och de åtgärder som vidtas kan vanligen anses vara kostnadseffektiva. Studien klargör också att en del åtgärder innebär relativt höga kostnader, vilket återspeglar den betydelse som anläggningsansvariga fäster vid faktorer vilka inte är direkt relaterade till strålskydd. Detta leder emellertid till att doserna blir låga.

Ett allmänt resultat från projektet är att lätthanterlighet är en avgörande faktor ur tillämpningssynpunkt för användning av optimering. Ett primärt krav är att klara, enkla och snabba metoder för rutinmässig optimering i det dagliga arbetet vid kärnkraftanläggningarna bör stå till buds. För att få en god bas för en optimering av strålskyddet och beslutsprocesserna krävs tillgång på hög teknisk kompetens och ett gott underlag för beslut i fråga om skydd. Speciell uppmärksamhet bör riktas på att söka alternativa lösningar, samt att fastställa resulterande doser och kostnader för de föreslagna lösningar.

Generella råd för val av formella optimeringsmetoder kan idag ges. Emellertid finns det många detaljer som kan förbättras för att underlätta optimeringsprocessen. Det finns ett behov av "tumregler" eller standardprocedurer som kan användas vid beslutsprocessen i det praktiska arbetet vid en kärnkraftanläggning. Det finns också behov av databaser och datoriserade stödfunktioner som kan användas i mer komplexa beslutssituationer.

Ambitionsnivån då det gäller strålskydd vid de nordiska kärnkraftanläggningarna är mycket hög. Detta har följdaktligen lett till att de verkliga kostnaderna för skydd inte alltid identifieras eller bestämmas i detalj eftersom de endast utgör en mindre andel av den totala ekonomin för anläggningen.

Projektet har framlagt förslag till fortsatta studier för strålskyddsoptimering vid de nordiska kärnkraftanläggningarna.

LIST OF CONTENTS

1 INTRODUCTION

The Nordic research project NKA RAS 410 "Application of the optimization principle to radiation protection at nuclear power plants", was carried out as a project of the safety research programme sponsored by NKA, the Nordic Liaison Committee for Atomic Energy, during the period from 1985 to 1989.

The main objective of the project was to work out how optimization could be applied to the radiation protection of workers at the Nordic nuclear power plants. This objective was striven at utilizing the existing knowledge and experience. The purpose was to clarify the optimization concept and, as far as possible, to map out a practicable methodology.

The project studied certain practical applications: radiation protection in in-service inspections, in the use of protective clothing and devices, and in modifications of plant systems and constructions. In addition, the optimization procedure and methodology were evaluated in general terms.

The project was run by a project group, whose members worked for the safety authorities, nuclear industry and research bodies in the Nordic countries (Finland, Sweden, Norway and Denmark).

This report is organized as follows: chapter 2 deals with the optimization procedure and methodology in radiation protection, as well as its practical application in relation to nuclear power plant operation worldwide. Some general conclusions are made. Reference is made to particulars which are more thorougly described in literature.

Chapters 3, 4 and 5 of the report describe the practical parts of the Nordic project, the applications mentioned above. Each of these chapters include their own conclusions and recommendations. A more comprehensive review of these subjects can be found in the separate project reports (which are listed among the references of each chapter).

Chapter 6 concludes the report in general terms.

2 OPTIMIZATION PROCEDURE

2.1 The optimization principle

The primary radiation protection guidelines worldwide are issued by the International Commission on Radiological Protection (ICRP). The ICRP has summarized its recommended basic system of dose limitation as follows:

The system has three components which are necessarily interrelated.

1) No practice shall be adopted unless its introduction produces a positive net benefit (The justification of the practice).
2) All exposures shall be kept as low as reasonably achievable, economic and social factors taken into account (The optimization of radiation protection).
3) The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances by the Commission (The limits of individual dose equivalent).

In 1983 the ICRP published a committee report on the optimization of radiation protection /2-1/. The report (ICRP Publication 37) mainly deals with the method of cost-benefit analysis although it clearly states that this is only one of many possible methods. In 1984 the ICRP Committee 4 established a Task Group to produce a more comprehensive report on the optimization of radiation protection and especially on the methodology of optimization. This report was submitted to the main Commission in 1988 and was published in 1989/2-2/.

2.2 Optimization methodology

According to the ICRP Publication 37, the concept of optimization is connected to the concepts of "effective dose equivalent", "collective radiation dose", "health detriment" and "cost". The consideration of other, more subjective components of the detriment have also been discussed. For optimization assessment, the ICRP has adopted a working hypothesis, according to which the relationship between small radiation dose contributions and the resultant increase in risk is linear.

In addition to the ICRP, the criteria involved in the optimization of radiation protection have been dealt with in several scientific evaluations. These evaluations give additional information on how to apply optimization to radiation protection.

In the ICRP Publication 37, the Commission presents the use of cost-benefit analysis for obtaining the optimum level of radiation protection by maximizing the net benefit B in the equation:

$$B = V - (P + X + Y) \qquad (2.1)$$

where V is the gross benefit, P the production costs of the practice, X the cost of radiation protection and Y the cost of the detriment.

The Publication 37 presents a simplified way of performing such an analysis, which is carried out under the assumption that the gross benefit and the production costs are independent of the collective dose S. This has led to the use of a "differential cost-benefit analysis":

$$\left.\frac{dX}{dS}\right|_{S_o} = -\left.\frac{dY}{dS}\right|_{S_o} \qquad (2.2)$$

where the optimization condition is fulfilled at a value S_o.

The cost of the detriment Y is expressed by the ICRP in the form:

$$Y = \alpha S + \beta \Sigma_j n_j f_j (H_j) \qquad (2.3)$$

indicating other components of radiation detriment besides the objective health detriment.

Optimization of radiation protection is an aid for decision-making. It is intended to clarify and to quantify radiation protection factors and to systematize trade-offs between the factors. Systematic ALARA (As Low As Reasonably Achievable) procedures will be used in optimization of radiation protection when applied to operational situations or design /2-2/, /2-3/, /2-4/.

This is illustrated by Figure 2.1.

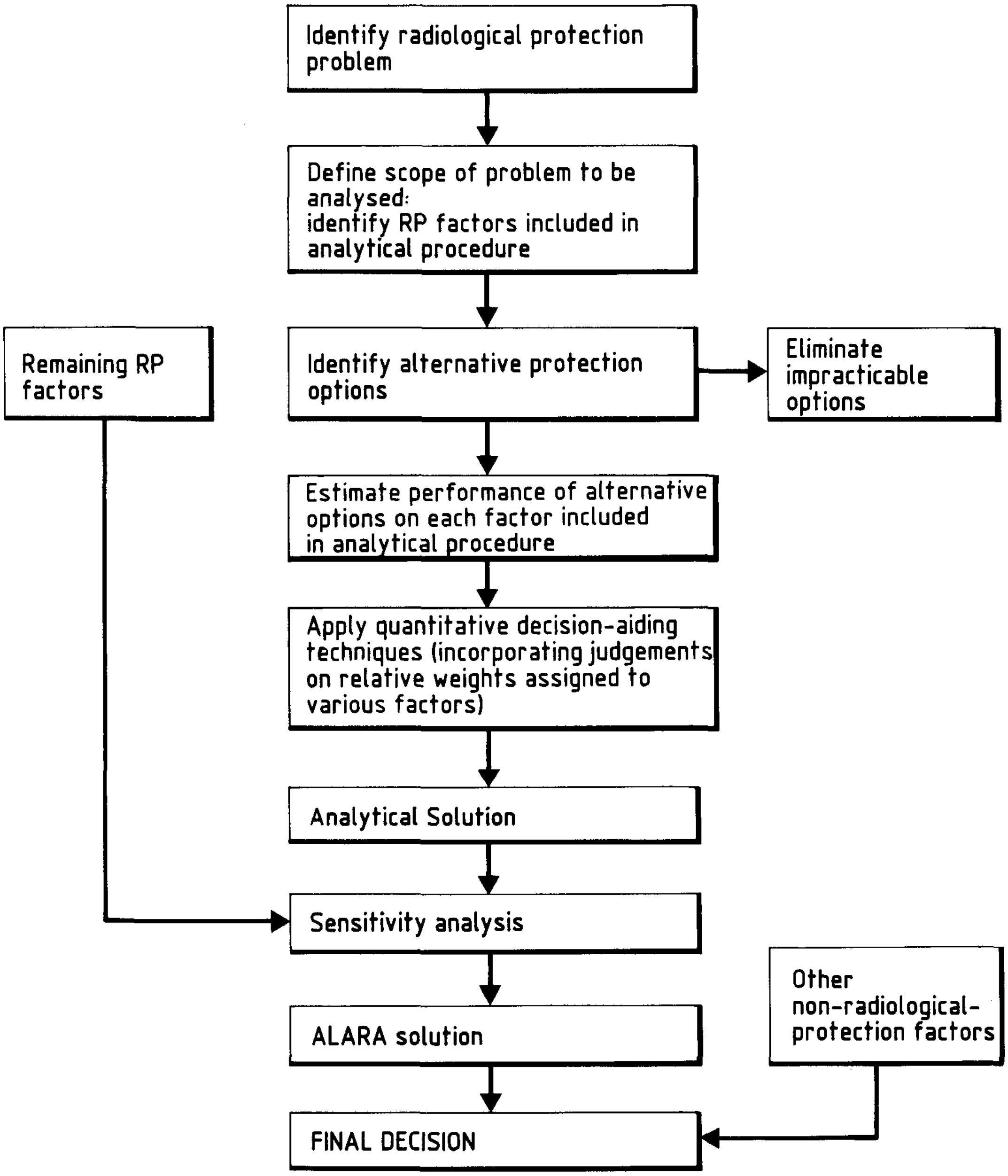

Figure 2.1. ALARA-procedure

In the following, each step of the ALARA procedure will be briefly considered with a few remarks:

Identification of the radiological protection problem

One should identify situations where exposures could be more closely examined. These can be e.g. cases which over the past years have not received any special attention or other cases which seem to hold potential for dose reduction.

Defining the scope of the problem

This can generally be quite complicated. In cases where only the radiation protection of workers at nuclear power plants is concerned, however, defining the scope is usually a fairly straigthforward step.

Sometimes such radiological protection factors that are difficult to treat quantitatively must be excluded from the quantitative analysis, but not from the entire ALARA procedure.

Identification of alternative options

Identification of alternative options is an elementary step in the ALARA procedure.

Estimation of performance

The set of factors introduced in the ALARA procedure can vary. The essential factors involve e.g. collective and individual doses as well as costs.

All relevant factors are not always directly measurable and they can be found out by using special models.

Choice of the quantitative decision - aiding technique

This can be a confusing step. There are many techniques not very commonly known. Among other things, the choice of the technique depends on the experience of the analyzer and the scope of the problem. Here we refer to the discussion later in this chapter.

Sensitivity analysis

Uncertainties of all relevant factors should be studied through a sensitivity analysis. The result of a sensitivity analysis indicates which factors have the greatest influence on the results.

The ALARA solution and the final decision

The outcome of an ALARA procedure shall be evaluated with respect to all factors excluded from the analytical procedure.

The ALARA procedure will clarify the main factors before the ultimate decision is made.

Another important extension to the presentation of the ICRP Publication 37 is the development work carried out by the National Radiological Protection Board (NRPB) in Great Britain. One purpose of the NRPB has been to give advice on the quantitative values associated with the detriment (the cost assigned to collective doses) /2-5/. The NRPB recommends that its minimum base-line cost of unit collective dose be multiplied by an additional factor, related to the individual dose distributions encountered in specific instances.

To illustrate the ideas of the NRPB, we refer to the following simplified table /2-3/.

Table 2.1 Proposed costs of unit collective dose for radiation workers

Individual annual dose band (mSv)	Cost of unit collective dose made up of individual doses within the band (1980 UK£/manSv)
- 5	4 000
5 - 15	20 000
15 - 50	100 000

This can also be illustrated with the following figure which gives the recommended value for the multiplication factor as a continuously increasing function of the individual dose level /2-5/. The numerical basis for the curve and for the base-line cost is a matter of judgement.

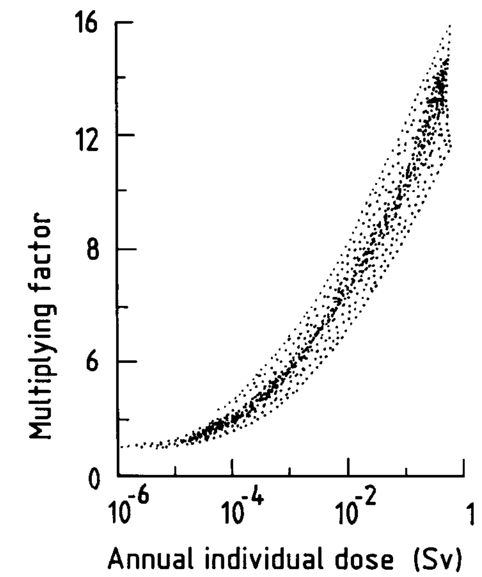

Figure 2.2. Multiplier to be applied to base-line detriment costs as a function of annual individual dose from the source

In addition to utilizing cost-benefit procedures or other aggregative methods for the optimization of radiation protection, organizations in some countries (e.g. CEPN in France) have in more detail examined the use of the decision theory and thereby also the use of multiattribute and multicriteria analyses as a tool for decision-making purposes. These are also described in ICRP Publication 55 /2-2/.

2.3 Practical applications at nuclear power plants worldwide

At a nuclear facility, the radiation exposure of workers can be reduced:

a) by reducing the dose rate in workplaces, the contributing factors and the corresponding means include

- radiation source (selection of materials, filtering, cleaning, corrosion monitoring, water chemistry, decontamination etc.)
- shielding/protection (radiation shields, protective equipment etc.)

b) by reducing working time, the contributing factors and the corresponding means include

- technical solutions (selection of components, tools, robotics, maintenance and operational measures etc.)
- detailed planning of work
- training/mock-up procedures.

The main radiological issues are often connected with major modifications at a nuclear power plant. Radiation protection is then inherently associated with e.g. engineering, production and nuclear safety.

Decisions to be made have in many cases been based on rather simple optimization methods, performed primarily by means of a cost-benefit or a cost-effectiveness analysis.

In Europe, ALARA-procedures linked with the optimization of radiation protection at nuclear facilities have been applied e.g. in the CEC countries, Sweden and Finland. This work has been thoroughly reported in scientific seminars, the latest of which was in 1988 /2-6/. The practical applications describe many interesting details of radiation protection, dealing with e.g. maintenance, work management and robotics at nuclear power plants.

In the USA, collection of data from special radiation work at nuclear power plants is partly systemized, facilitating a search for data in a special radiation protection database run by the ALARA Center at the Brookhaven National Laboratory. The emphasis in BNL work is on research and development concerning occupational dose control. Furthermore, a project for the development of a data collection, retrieval and analysis system is under way at the OECD/NEA. This work aims at facilitating the exchange of information on dosimetric data and dose-reducing methods between participants in the system and thus increasing the opportunities to learn from each other's experience.

2.4 Radiation protection optimization for employees at nuclear power plants, conclusions

A review of information on optimization methodology and its practical application carried out in the project produced the following results.

Essential findings include:

I If the optimization problem of radiation protection can be confined to cover the radiation protection of workers only, the procedure is normally quite well defined. The optimization criteria are acceptable; inaccuracies in detriment/dose equivalent, costs etc. make up the boundaries for decision-making. However, different groups of workers often have different working conditions, which have to be considered within an ALARA-procedure.

II The optimization of radiation protection and radiation protection actions shall be based on

- good technical competence
- thorough preparation (emphasis shall be put on the search for options and assessment of doses and costs involved)
- where applicable, some quantitative optimization method (see III)

III The choice of the quantitative optimization method shall be based on answers to the following considerations:

- what is the estimated radiological importance of the problem to be analyzed?
- do the radiation protection measures to be analyzed have a significant effect on the safety or production of the plant?
- are there "other factors" involved?
- is it easy to define the actual scope of the problem?
- what is the level of knowledge of the factors involved; are they qualitative or quantitative?

The following matters may also have an effect on decision-making and the choice of the optimization method:

- the time available for decision-making
- the radiation protection contribution available and its limitations, if any.

In selecting the optimization method, the quantitative decision-aiding methods could be considered in the following order:

- multi-attribute utility methods or multicriteria methods
- cost-benefit analysis (or other aggregative methods)
- cost-effectiveness analysis

A well-known phrase "engineering judgement" is not included in this list intended for an issue important from the standpoint of radiation protection. However, it may be used in decision-making at the basic level of operational dose control. It can also be considered relevant for decision-making, if it contains a structured and quantitative understanding of the problem.

It is generally suggested that the ALARA-procedure should be implemented depending on the level of decision-making. A more sophisticated quantitative decision-aiding method should be used when making decisions at higher levels.

A good command of the method to be applied is necessary. This involves both the techniques and the criteria.

2.5 References

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3 OPTIMIZATION OF RADIATION PROTECTION IN RELATION TO IN-SERVICE INSPECTION

3.1 Introduction

Inspections are carried out periodically during the operation of nuclear power plants in order to maintain a high level of safety and reliability in plant operation. Some of these inspections are called in-service inspections and they are performed on the basis of regulations issued by the safety authorities. The main purpose of in-service inspections is to reduce the probability of accidents by examining plant systems and components for any deterioration in order to judge whether the plant is acceptable for continued operation or whether some actions must be taken. Emphasis is placed on examining the critical parts of the "primary systems" because of their importance to safety and the potential severity of the consequences in case of a failure.

In-service inspections are also considered important to plant availability. Early detection of material deterioration makes it possible for the plant owner to prepare for preventive maintenance in order to minimize the risk of forced shutdowns and to introduce well-timed planning in case a replacement of major components seems to be necessary. For these reasons plant owners may introduce additional inspections complementary to the safety-related in-service inspections.

A third type of inspection of a similar kind is made up by inspections related to repairs. When replacing a component in a safety classified system it is normally mandatory for the plant owner to carry out inspections to prove the proper installation of the new component.

The inspections mentioned above are carried out mainly during plant outage periods by specially trained technicians - material testers.

The techniques of inspection vary from manual, as is normally the case concerning pipings and components, to highly sophisticated, remote controlled techniques which are used, for example, for the inspections of the reactor pressure vessel and the internals.

The implementation of the inspections as well as their preparatory and support operations involve significant exposure of the personnel involved, due to the need to enter areas with a high level of radiation and the necessity for the personnel to be very near to activated or contaminated systems and components for significant periods of time.

It is therefore a delicate question how high occupational doses should be allowed for performing inspections in nuclear power plants in order to reduce the probability of accidents and to maintain or improve plant availability. As regards regulatory in-service inspections this could be expressed as follows: how high doses should be allowed for workers in nuclear power plants year after year to examine components in order to reduce the probability of exposing the plant personnel and the public in case of an incident releasing radioactive substances.

3.2 Project activities

The objectives of the study were to review the applicability and limitations of the optimization principle of radiation protection presented by the ICRP (the ALARA principle) in relation to in-service inspections. It was also considered useful if some practical radiation protection guidelines could be developed for decision-making in in-service inspections.

In order to meet the above objectives, the following areas were studied concerning in-service inspections:

- regulations and practices;
- radiation protection experience;
- radiation protection measures;
- ALARA aspects;

The study was limited to in-service inspections of systems outside the reactor pressure vessel for the reason that the inspections of the reactor pressure vessel are remote-controlled and therefore do not cause significant occupational exposure.

The concept of in-service inspection was used in the study to cover all three types of inspections described in section 3.1.

The material in the study covered mainly the years from 1981 to 1985, although experience concerning important developments in inspection activities were included from years later than 1985.

3.3 Results of the study

The results presented in the following are mainly based on a report entitled "Optimization of Radiation Protection in Material Testing at Nuclear Power Plants" /3-1/ and partly on a study prepared for ABB Atom and the National Institute of Radiation Protection in Sweden on "Advanced Pipe Inspection Systems, State-of-the-Art" /3-2/. The material for the first study was collected mainly with questionnaires, from available dose reports and through interviews with radiation protection managers and material specialists at all the Nordic nuclear power plants, whereas the second is a report mainly based on the information from a conference on advanced technology in the field of in-service inspections.

3.3.1 In-service inspections in nuclear power plants

Components subject to in-service inspection are examined with various methods classified as visual, volumetric and surface methods. The most time-consuming methods and thus of special interest from a radiation protection point of view are the volumetric methods, particularly ultrasonic testing and to some extent radiography. Ultrasonic tests seem to be used more and more frequently instead of radiography as volumetric tests. To a large extent this is because of the increased demands to validate the accuracy of the test method to be used for a certain component. More reliable methods for radiography are, however, being developed in some countries /3-3/.

In-service inspections are carried out according to established general programmes. These programmes specify the extent of inspections, the method to be used, the frequency of inspections, etc.

The programmes regulated by the authorities in Finland and Sweden differ somewhat with respect to the basis for their development although in practice they appear quite similar. The Finnish programme is more directly based on the American ASME-Code whereas the Swedish programme, which originally was developed in the early seventies by a working group for in-service inspections, is partly based on the ASME and partly on Swedish conventional norms for pressurized components. These programmes are in both countries supplemented with utility programmes based on operating experience. It should be pointed out that presently a new programme is being implemented in Sweden concerning regulatory inspections; its impact is still to be seen. This question is discussed in more detail in section 3.3.4.

In-service inspections are carried out mainly during outage periods and mainly by contractor personnel employed by a few companies specialized in material testing. These people, rather few in number, are working in one or, more often, in several plants during a year, which means that they receive doses at different plants. In addition, there is a transfrontier exchange of personnel in this field.

How extensive are the in-service inspection programmes? One way of answering the question is to give the number of inspection points, i.e. welds, fixing points for pipings, etc., which have to be tested each year at a plant. This number varies from one year to another depending on, for example, the age of the plant, but is typically of the order of 300-600. These points are inspected using one or several of the methods referred to above. The time needed for inspection varies depending on the test method, accessibility, radiological conditions, experience and training of the material tester, etc, but is in the order of hours rather than minutes per inspection point. The requirements for preparatory work must be added to this, if the total time requirement for in-service inspections is to be estimated.

3.3.2 Radiation doses due to in-service inspections

Important factors contributing to radiation doses, collective as well as individual, obtained from in-service inspections include:

- the extent of the inspections and the spatial distribution of inspection points within the plant;
- test methods;
- radiological conditions;
- accessibility;
- radiation protection measures;
- work management considerations.

No attempt has been made to quantify these factors as regards the effect on the resultant doses, but some of them are discussed in some detail later in the report. However, some results concerning collective and individual doses in relation to inspections are given first.

Collective doses

Radiation doses caused by in-service inspections were found to be considerably dependent on the plant in question and the doses vary from one year to another. As an average for the years 1981 to 1985, doses of in-service inspection in Nordic nuclear power plants constituted testing, and preparatory work included, between 5 and 25 per cent of the annual collective dose.

The annual collective doses per reactor unit for in-service inspections varied between 0.01 and 0.4 manSv in the years 1981 to 1985 with an average for the years studied of 0.13 manSv.

Table 3.1 shows the average annual total collective doses and the average annual collective doses for in-service inspection for 13 nuclear power plants in Finland and Sweden in the years 1981 to 1985. The in-service inspection dose as per cent of the annual total collective dose is also included. The reactor units concerned are given in brackets.

Table 3.1

Nuclear Power Plant	Aver. Annual Collective Dose per Reactor Unit [manSv] Total	In-service Inspection	Percent of Collective Dose for In-service Inspection
Barsebäck, BWR (B1, B2)	0.75	0.09	12
Forsmark, BWR (F1, F2)	0.59	0.07	11
Oskarshamn, BWR, (O1, O2)	1.13	0.11	10
Ringhals*			
BWR, (R1)	3.2	0.20	6
PWR, (R2, R3)	1.8	0.17	9
Olkiluoto, BWR, (TVO1, TVO2)	0.48	0.11	22
Lovisa, PWR, (L1, L2)	0.79	0.20	24

* Ringhals 4 was taken into operation in 1982-1983 and is not included in the table, nor are the BWRs F3 and O3, which were taken into operation in 1985.

As can be seen from Table 3.1, the average annual collective doses for in-service inspection were quite similar for the boiling water reactors (BWR), except for Ringhals 1, but higher for the pressurized water reactors (PWR). The relatively high figures for Ringhals and Lovisa were influenced by:

- extensive inspections of the steam generators at Ringhals PWRs (R2, R3)
- extensive inspection of the welds in the primary system of the Lovisa PWRs
- relatively high exposure rates in Ringhals 1, particularly within the reactor containment, where the inspection of the primary circuit had also caused some access problems.

Finally, it is seen from Table 3.1 that the doses from in-service inspections are about equal in Finland and Sweden, considering BWR's and PWR's separately. Presented as per cent of the annual collective doses they were higher in Finnish reactors than in Swedish ones, which might indicate the inspection programmes in Finland to be more extensive than in Sweden. It was not possible, however, to investigate this in detail within the project.

Table 3.2 shows the trend between the years 1981 and 1985 in occupational exposure in the Nordic nuclear power plants for the total annual collective dose and for the total annual collective dose due to inservice inspection. The in-service inspection doses are also given as per cent of the total doses.

Table 3.2

Year	Total Annual Collective Dose at the Nordic NPPs [manSv]		Per cent for In-service
	Total	In-service Inspection	
1981	14.9	1.4	9
1982	13.2	1.7	13
1983	17.2	2.0	12
1984	14.7	1.8	12
1985	13.0	1.8	14

No increasing or decreasing trend either in exposure due to in-service inspection or in the total exposure can be concluded from Table 3.2.

In the project it was not possible to systematically distinguish between doses from regulatory in-service inspections, utility inspections and inspections related to repair work because of the different practices established at that time for reporting doses in the plants in the two countries. However, it was estimated that regulatory in-service inspections contribute as an average to the total 50 per cent in BWR's and 30 per cent in PWR's.

The collective dose for in-service inspection is composed of contributions from different occupational groups: material testers, insulation personnel, service personnel, etc. It was found that in most plants almost 90 per cent of the collective dose for in-service inspection was received by material testers and insulation personnel.

Individual doses

The average annual individual doses for workers in nuclear power plants in Finland and Sweden are low and far below the regulatory dose limits. However, for the personnel participating in in-service inspections the situation might be different considering the relatively limited number of people doing this type of specialized work and the fact the many of them work at several plants during a year. Therefore, it was investigated if there were differences of significance between the individual doses received by material testers and insulation personnel, and the doses received by other occupational groups employed at Nordic nuclear power plants.

The annex in section 3.6 of this chapter gives average and maximum annual individual doses for a number of occupational categories. From that compilation it could be concluded that the mean average annual individual doses for material testers at one site seemed to be between 1 and 3 mSv, and for insulation personnel between 2 and 7 mSv in the years 1981 to 1985. It must be observed that the figures given for insulation personnel include all the work done by this group and not only the part belonging to in-service inspection. A distinction was not possible due to dosimetric routines at the time of the study. It should be noted that these figures apply to different reactor sites and it may well be that the same persons are included in many of them.

This means that the figures cannot be used to calculate country averages; instead, information obtained from national registers should be used. When this is done and these two groups of workers are compared to the average worker, statistically speaking, it can be concluded that the "average" material testers have a radiological situation comparable to that of an average nuclear power plant worker. However, a limited number of individuals in this group receive relatively high individual doses, particularly when evaluating indications of cracks and a longer than normal time has to be spent in close proximity to the test object. These high individual doses need continuous attention in order to minimize the healthy risk of them and also the risk of not being able to use some of these highly skilled technicians all around the year.

Moreover, the insulation personnel receives significantly higher individual doses than the average nuclear power plant worker. The insulation personnel together with health physicists make up the highest exposed groups working at Nordic nuclear power plants from the point of view of individual dose distribution, and their doses have also to be carefully followed.

3.3.3 Work management and radiation protection measures in relation to in-service inspection

A number of measures are taken by plant health physicists to control occupational doses during in-service inspection as well as during nuclear power plant operation in general. The protective measures related to in-service inspections are of a general nature, apart from the fact that inservice inspections are performed simultaneously in many places at a plant and they might therefore warrant some extra effort to co-ordinate radiation protection measures.

The most important measures were found to be as follows:

- Work planning and co-ordination. The health physicist should be involved in the planning of the inspections at an early stage. The health physics staff and the personnel responsible for material testing should therefore discuss the topic between themselves and also with the plant outage planning group, which is the group responsible for planning all the activities during outages. This means that the various aspects related to the performance of the inspections and the necessary radiation protection measures would be reviewed and taken into consideration already when the outage is planned. In this way there would be more flexibility to perform the inspection in the best possible radiological conditions. Another aspect of work planning and co-ordination is the preparation of working places and the training of the personnel involved in the inspections. The working places should be properly prepared and equipped with easily removable insulation and they should be identifiable according to a system known by the personnel. Training of the personnel, including the work supervisors, in station lay-out, radiation protection and ALARA thinking would be another good way of saving doses.

- Direct radiation protection measures. Direct measures that are used to reduce exposure levels include system flushing, decontamination and shielding. Shielding is usually done by using lead shields, water shields or by keeping a contaminated pipe or component, which is subject to inspection or otherwise affecting the exposure rates, filled up with water. For surface contamination, the most common protective measure is the use of extra boundaries for contaminated areas, i. e. to separate the contaminated areas from areas with normal contamination. If elevated levels of air contamination are detected, extra protective equipment such as respirators are used. The use of protective equipment in Nordic nuclear power plants has been studied in the project and is described in chapter 4 of this report.

- Dosimetry. Health physicists and the personnel involved consider personal self-reading dose meters in addition to the TL-dosemeters or film badges to be very useful tools in controlling individual doses. This is particularly true when performing, for example, an ultrasonic examination in an area with relatively high radiation fields. This touches on the question of the quality of the inspection when performed in high radiation fields. There may be doubts about the reliability of such tests even among material testers, and this is one good reason for developing automatized tools or equipment to minimize the time which needs to be spent in high radiation areas /3-4/. More about automatization later.

In conclusion it can be said that there are a number of measures which are taken or could be taken to save doses in in-service inspection. An alert health physics group that has taken a serious view of the aspects related to work management and co-ordination, training and informing workers, etc., can do a lot to control and save doses during nuclear power plant operation /3-5/. Finally, it is important to draw attention to the possibility of reducing some doses by shielding. In chapter 4 of this report it is recommended that the use of lead shields could be somewhat increased. This recommendation is based on an optimization analysis.

3.3.4 Status of the application of optimization of radiation protection in in-service inspections

Factors affecting decision-making in relation to radiation protection measures for in-service inspections include:

- radiological conditions, such as ambient and surface exposure rates and levels of contamination;
- time required for the inspections and preparatory work;
- dose records of the individuals involved;
- costs of radiation protection measures.

Moreover, it was found that ALARA thinking governs decision-making for protective actions. Optimization seemed to be used qualitatively with the purpose of keeping doses as low as reasonably possible using experience and engineering judgement.

It was also found that ALARA seemed to be applied in the sense of:

- taking radiation protection measures which also minimize the dose for the total work, for example by also considering the dosimetric cost to people who carry out protective actions when deciding on a particular action;

- trying to direct the inspections to points with the lowest exposure rates if alternatives exist.

The above examples of measures could be classified as measures which are taken from a practical point of view within the scope of normal resources for radiation protection.

A second step would be to require additional measures for dose reduction such as measures to reduce the exposure rates in general at the power plant or actions to develop less time-consuming inspection techniques. The first of these measures has a strong bearing on the overall collective doses in the power plant, and as long as the radiological situation in the Nordic power plants is relatively favorable from an operational point of view, and in an international context, such radical measures as to decontaminate reactor systems may not be considered "optimized" by plant management. Measures to control and reduce system dose rates have been carefully considered during the design stage, for example by using components with very low cobalt contents.

As concerns actions to develop less time-consuming inspection techniques, steps are being taken in many countries, also in the Nordic countries. A number of companies are at present offering equipment for remotely controlled in-service inspections. Such equipment has also been used in power plants in the Nordic countries /3-2, 3-4/. However, the experience so far indicates that there are introductory problems with this equipment when used in real plant conditions, particularly in the application of transducers to components. However, the on-going technical development together with increased training of personnel will certainly improve the situation and thus make it worthwhile to put some effort into introducing this equipment, particularly for the inspection of components requiring special consideration with respect to radiation protection. It must be noted though, that the primary driving force in the development of this equipment seems to be a need to improve the quality of inspections, and not so much to develop equipment optimized from a radiation protection point of view.

A third type of ALARA measure would be to question the present programme of in-service inspections and to perform an optimization analysis in order to find out how many inspections should be performed to correspond to optimum safety and health for the public and the workers. Such studies have been conducted in some countries, for example in Canada /3-6/. It is not clear to us, however, to what extent the results of these studies have been used to revise existing regulatory programmes. In Sweden a revision of the present programme for in-service inspections is under way in order to take into consideration the experience gained of inspections and the opinion expressed by some reactor safety experts that the emphasis of the inspections must be optimized /3-7/.

The revised programme implies in practice that inspections would concentrate on pipings and components which theoretically or according to operating experience have a comparatively high risk of failure or where a failure would have major consequences. What the revised regulations mean in practice is still to be seen, but it seems to be the opinion of the operators, including health physicists at the plants, that this might be a fruitful way of optimizing the inspections. The experience of the application of these new programmes from a radiation protection point of view will be closely observed by the utilities according to the requirements expressed by the radiation protection authority, which reviewed the programme before it was issued. The authority has pointed out the necessity of introducing remotecontrolled techniques in order to avoid, if possible, an increase in the exposure of workers as a consequence of the application of the new programme /3-8/.

Discussion

Optimization of radiation protection in nuclear power plant operations, such as in-service inspection is rather complex because it affects not only the radiation protection of the workers but also the protection of the public and the economy of power plant operation.

Conceptually the optimization problem can be expressed as follows:

Case A. The programme is fixed, i.e., a safety level established, and in the optimization procedure one has to weigh the possible inspection strategies within the framework of the programme and the corresponding outcome in terms of workers' exposure;

Case B. The programme is to be fixed, which means that a proper balance between the safety and protection of the public and the safety and protection of occupational groups working with in-service inspections has to be reached.

Radiation protection actions within the context of normal resources are considered ALARA by many health physicists and it seems therefore to be the opinion of these people that spending more money on protective actions in relation to in-service inspections would not be ALARA. What ALARA means in this respect is, however, not clear. Does it appear as a result of a comparison with other jobs at the plant, as a result of an international comparison, or as a result of a "cost-of-protection" reflection? A model was developed within the project to calculate the cost of protective actions in relation to in-service inspection. Only those costs that are specific to in-service inspection are included, not the costs of a general type, such as for dose meters, protective overalls, etc. The costs of special resources and equipment that are needed and considered "extra" for the in-service inspection activity are, however, included.

Thus the model for the total annual cost K of extra radiation protection measures will be:

$$K = k_1 t_1 + k_2 t_2 + s_1 a_1 + s_2 a_2 + s_3 a_3 + m + u \qquad (3.1)$$

where

k_1 , k_2 = cost per hour for personnel
t_1 , t_2 = the number of working hours
s_1 , s_2 = cost of lead shield
a_1 , a_2 = the number of lead shields
s_3 = cost of other shields
a_3 = the number of other shields
m = cost of measuring instruments
u = cost of special equipment.

This model was used to calculate the extra costs of radiation protection in relation to in-service inspection and they were found to be roughly SEK 100-400 (FIM 70-300 or US$ 20-70) per inspection point, or in total about SEK 100 000 (FIM 70 000 or US$ 20 000) per year and per reactor.

The results of these cost calculations stir up many interesting questions, for example, is the money spent in an optimum way concerning occupational exposure in general at the plant, or are there other ways of spending it which would give a higher dose saving? Would the level of collective exposure in in-service inspections decrease significantly if extra resources were spent in line with the present α-value of the order of SEK 100 000 (FIM 70 000) per manSv officially recommended in the Nordic countries, taking into consideration that the average collective exposure in in-service inspections is only about 0.15 manSv per year and reactor (and the marginal dose saving caused by "extra" radiation protection measures only a share of this figure)?

These questions cannot be answered easily. It is obvious, however, that the present situation concerning exposure in in-service inspections is a result of years of experience where many factors have been taken into account, and it is not likely that, as has been indicated above, much exposure could be saved by spending extra money on protective actions in line with the above α-value. On the other hand, it is always worthwhile to question priorities in relation to protective actions, within the in-service inspection activity itself or between different activities carried out during nuclear power plant operation. It should also be pointed out that individual doses place a very important constraint on optimization of protection.

Concerning case A above, a typical example of optimization is the introduction of automatized inspection equipment. Some experience of the practical application of this equipment in plant conditions has already been obtained. It is too early to say, however, what effects the application will have on the collective dose to plant workers. It is also to be kept in mind that the use of the present generation of this equipment cannot replace all manual inspections, but hopefully the ones which are the most difficult to perform.

The experience which will be available in some years concerning the use of automatized inspection tools will give valuable data for future optimization studies. Moreover, very few studies seem to have been conducted so far to optimize this equipment itself with respect to radiation protection. This should be looked into and be one factor when deciding which equipment to introduce.

Case B is much more complicated because the probability of accidents enters the discussion. The problem is now the following: How much in detriment or dose to workers should be spent to prevent possible future accidents?

The optimization problem can be illustrated with the following figure:

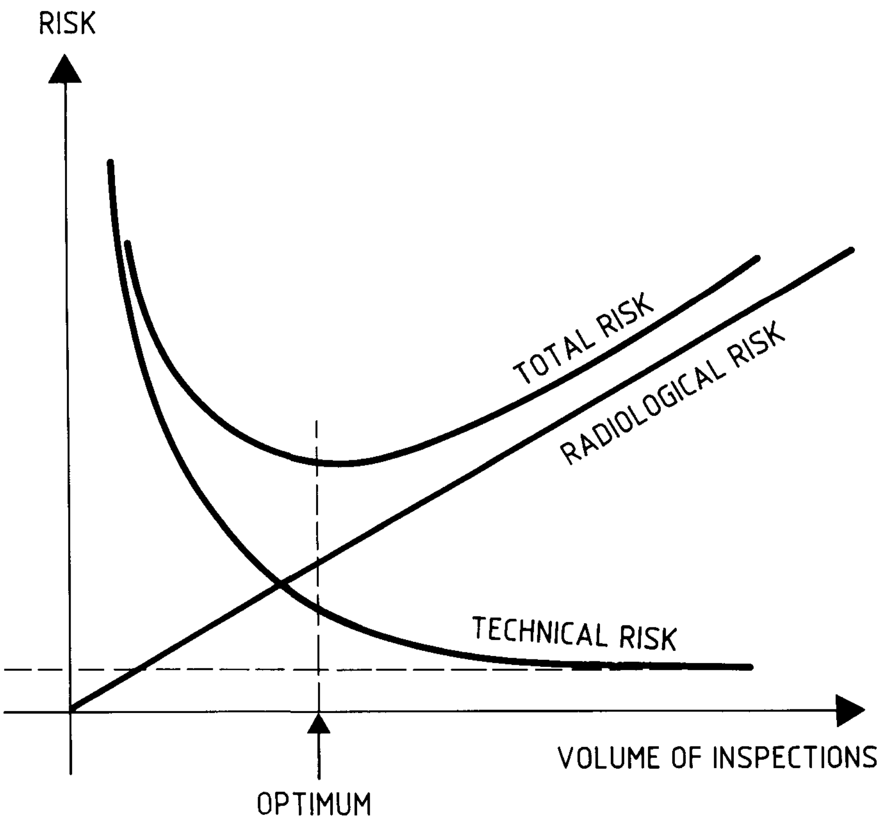

Figure 3.1 A general optimization problem.

The figure is very schematic and only intended to show the potential conflict which exists when establishing requirements for nuclear safety measures or maintenance programmes involving the exposure of workers. When work requirements in a nuclear power station are increased, the exposure of workers also increases in many cases. At some point the exposure of workers becomes a dominant risk factor, overriding the extra benefit, which is gained by the work requirements. This is what the figure intends to show. No attempts were made in the project to quantify the risks in order to make the two curves comparable.

One problem in this optimization analysis is the difficulty of quantifying the inspections in terms of a reduced probability of accidents. The question is further complicated by the fact that the inspection frequency is also connected to plant availability, which might be considered affected if the inspection frequency is reduced /3-9/. These questions have been discussed within the project and have also been dealt with when introducing the new inspection programme in Sweden. The new programme was introduced in order to increase the safety of the plants. However, it was also realized that it might increase the exposure of personnel involved in in-service inspection activities. Therefore, in the decision-making process, the extent of the inspections was a critical parameter and already here a qualitative optimization was made when choosing the extent and frequency of inspections. It was also said that experience, technically and from a radiation point of view, must be carefully observed in order to make it possible in a few years to reconsider the programme if the "optimum" was drastically missed /3-8/.

In optimization of radiation protection one has to consider several options to reduce doses. The value of the optimization strategy is not only to get absolute estimates of possible solutions but to get a tool for ranking different options. It may well be that spending money on the development of a more efficient work management system gives as much in reduced doses as does the introduction of automatized equipment but for much less money.

3.4 Conclusions and recommendations

In-service inspections are considered to be of great importance as concerns plant safety, reliability and economy. A rather extensive programme has therefore been established to fulfil the demands by the authorities and plant owners. Inspections are performed mainly by outside contractor personnel. Moreover, about half of the collective dose for the inspections is due to different kinds of assistance work for the inspections, insulation work being the main part of this. As an average, in-service inspections cause about 0.13 manSv per year and per reactor. This corresponds to about 15 per cent of the average annual collective dose for the reactors in the Nordic countries. The individual doses for personnel involved in the inspections are sometimes high and need continuous consideration regarding protective actions applied. This is particularly true for the insulation personnel and a part of the inspection personnel, particularly those who go from one plant to another during the year.

One of the most important measures for controlling doses from in-service inspections seems to be to establish an active "work management" programme, which mostly consists of work planning, co-ordination, training and introducing ALARA thinking among all involved. Optimization in the strict formalized sense seems not to be applied at the plants during plant operation. Instead, practical optimization based on experience and engineering judgement is applied. This has led to the present situation in in-service inspection. On the other hand, formal optimization might have given similar results. In any case, the weighing between individual doses and the collective dose is always difficult, and some guidance would need to be developed in this respect.

Recent developments in the Nordic countries on reconsidering parts of the established routines for in-service inspection are noted with satisfaction by the health physicists at the plants. Radiation protection aspects including the problems of working in high radiation areas, are partly the driving force in the development of new inspection techniques. The main driving force, however, seems to be the need to increase the reliability of component testing where the application of a manual technique does not give satisfactory results. Great attention is also paid to the reconsideration of the traditional programmes in this field. A new programme which might cause reduced doses to the personnel is being implemented in Sweden using partly this new inspection technique. Experience of the use of automatized tools for in-service inspection and of the implementation of the new programme has to be evaluated regularly. It is important that the data which will consequently become available is collected in a way useful for future ALARA-studies. The data referred to relates to collective and individual doses received by inspection and service personnel (insulation, scaffolding and health physics personnel) in adapting and using the new technique, cost of protection, cost of new equipment, quality of inspections, etc. The cost estimate model developed within the project may be useful in this context.

3.5 References

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3.6 ANNEX

TABLE 3.6.A: Average and Maximum Individual Doses in In-Service Inspection for some Occupational Categories Working in the Nordic Nuclear Power Plants

Year 1981	Average/Maximum Individual Dose [mSv]					
	Nuclear Power Plant					
	BI - B2	FI - F2	OI - O2	RI - R4	TVO I- TVO II	Lo I- Lo II
Occupational Category						
Material Testers	2.1/13.7	1.0/4.1	1.6/4.4	2.7/15.8	1.0/5.0	1.6/5.2
Health Physicists	2.3/7.6	1.8/8.4	4.2/11.9	17.1/39.5	1.1/3.8	2.6/6.3
Mech.repair personnel	2.2/31.6	2.1/	2.6/20.3	6.0/39.1	1.2/10.2	1.9/11.3
Service personnel	1.3/16.3	0.9/4.9	1.8/8.2	4.4/34.2	-.0/4.6	1.8/8.0
Operators	1.2/8.6	0.8/4.2	1.0/4.6	2.5/15.7	0.7/5.8	0.5/1.4
Elec.Instrumentation personnel	0.9/8.0	0.7/4.1	2.2/11.5	1.8/9.8	0.4/1.4	0.9/4.5
Chemists	0.6/2.5	0.4/1.2	1.4/3.5	1.1/5.1	0.2/0.3	-
Insulation personnel	4.6/17.7	2.9/14.0	3.6/10.3	12.5/29.9	2.2/7.5	4.0/10.8

TABLE 3.6.B: Average and Maximum Individual Doses in In-Service Inspection for some Occupational Categories working in the Nordic Nuclear Power Plants

	Average/Maximum Individual Dose [mSv]					
Year 1982	Nuclear Power Plant					
	BI - B2	FI - F2	OI - O2	RI - R4	TVO I- TVO II	Lo I- Lo II
Occupational Category						
Material Testers	1.0/3.5	1.3/24.2	1.0/4.0	2.4/16.9	1.8/9.0	2.9/18.3
Health Physicists	2.3/6.6	2.3/7.2	2.8/9.7	10.2/35.5	1.8/7.3	4.8/11.5
Mech.repair personnel	1.3/15.2	1.7/17.0	2.4/22.5	6.2/41.5	1.0/7.4	4.1/23.4
Service personnel	1.0/7.3	0.9/7.8	2.1/12.4	2.7/24.6	1.3/5.6	2.8/14.9
Operators	1.2/11.2	0.6/4.5	0.9/6.5	2.2/11.8	0.7/10.6	0.8/2.2
Elec.Instrumentation personnel	0.9/8.4	0.7/5.7	1.7/12.6	1.8/14.4	0.4/2.9	1.3/8.5
Chemists	0.4/0.8	0.3/0.5	1.5/4.3	1.1/8.3	0.5/1.0	-
Insulation personnel	3.4/11.2	2.1/7.9	4.6/13.7	5.3/25.2	4.2/14.7	7.0/19.7

TABLE 3.6.C: Average and Maximum Individual Doses in In-Service Inspection for some Occupational Categories working in the Nordic Nuclear Power Plants

Year 1983	Average/Maximum Individual Dose [mSv] Nuclear Power Plant					
	BI - B2	FI - F2	OI - O2	RI - R4	TVO I- TVO II	Lo I- Lo II
Occupational Category						
Material Testers	2.4/10.2	0.8/7.3	2.3/12.9	5.0/21.3	1.4/6.7	2.0/7.4
Health Physicists	3.1/14.0	1.3/4.2	5.1/14.1	10.6/34.1	1.5/4.3	3.4/8.9
Mech.repair personnel	2.2/14.7	1.1/6.8	3.4/25.8	5.9/35.3	1.1/11.3	2.7/10.1
Service personnel	1.4/11.9	0.8/3.9	2.4/10.7	2.9/21.5	1.0/4.8	2.4/11.6
Operators	1.2/7.4	0.5/2.5	1.1/14.9	2.2/11.7	1.0/16.8	0.7/2.5
Elec.Instrumentation personnel	2.9/16.7	0.5/3.7	2.1/10.7	2.4/16.5	0.4/3.0	1.3/9.9
Chemists	0.8/3.0	0.1/0.2	1.1/2.3	1.7/7.7	0.4/1.2	-
Insulation personnel	6.2/12.2	1.1/4.5	3.5/18.2	7.1/22.3	2.5/9.3	3.7/15.4

TABLE 3.6.D: Average and Maximum Individual Doses in In-Service Inspection for some Occupational Categories working in the Nordic Nuclear Power Plants

Year 1984	Average/Maximum Individual Dose [mSv]					
	Nuclear Power Plant					
	BI - B2	FI - F2	OI - O2	RI - R4	TVO I- TVO II	Lo I- Lo II
Occupational Category						
Material Testers	2.1/7.3	1.0/4.6	2.1/7.9	2.3/9.7	1.2/6.0	2.4/8.5
Health Physicists	3.3/12.3	2.4/7.8	4.1/18.6	7.1/31.2	1.3/3.8	5.9/11.8
Mech.repair personnel	2.5/18.6	1.1/12.5	2.9/22.7	5.4/41.5	1.3/8.6	3.3/13/1
Service personnel	1.6/9.6	1.0/6.7	2.6/13.2	2.5/24.6	1.2/7.2	3.8/15.2
Operators	1.5/6.4	0.7/3.1	1.1/6.0	1.5/11.5	0.8/12.4	0.6/2.2
Elec.Instrumentation personnel	1.3/7.5	0.7/2.9	1.9/10.1	2.7/15.1	0.5/1.7	1.6/6.4
Chemists	0.9/2.2	0.4/1.3	2.1/6.2	0.9/4.9	0.4/1.1	-
Insulation personnel	5.7/16.1	1.3/5.1	2.8/14.9	2.7/16.1	2.9/9.0	10.6/17.4

TABLE 3.6.E: Average and Maximum Individual Doses in In-Service Inspection for some Occupational Categories working in the Nordic Nuclear Power Plants

Year 1985	Average/Maximum Individual Dose [mSv]					
	Nuclear Power Plant					
	BI - B2	FI - F2	OI - O2	RI - R4	TVO I- TVO II	Lo I- Lo II
Occupational Category						
Material Testers	1.7/4.6	1.0/8.0	2.2/9.8	3.4/13.8	1.2/7.3	1.2/4.2
Health Physicists	2.6/8.6	2.2/7.1	3.7/16.1	5.2/20.4	1.2/3.4	4.3/10.3
Mech.repair personnel	1.4/11.3	0.9/10.2	2.6/16.8	5.2/41.9	1.1/7.3	2.0/8.4
Service personnel	1.3/6.9	0.9/6.4	2.5/15.1	2.3/21.7	1.2/5.2	2.8/12.9
Operators	1.0/6.0	0.6/3.9	1.3/6.2	1.2/5.1	0.7/9.7	0.5/2.6
Elec.Instrumentation personnel	0.9/3.7	0.5/2.2	1.9/16.6	1.9/17.3	0.5/2.3	2.3/12.1
Chemists	1.1/4.5	0.3/1.0	3.9/14.4	0.9/5.5	0.1/0.2	-
Insulation personnel	3.4/10.8	1.9/7.4	4.4/18.3	5.0/21.5	1.7/6.5	4.0/11.1

4 OPTIMIZATION OF RADIATION PROTECTION IN RELATION TO USE OF PROTECTIVE METHODS AND EQUIPMENT

4.1 Introduction

Special radiation protection measures are needed when workers dismantle components containing radioactive substances or work in locations classified as orange (in Sweden yellow) or red. This colour classification indicates an increasing risk of radioactive contamination and direct radiation. Green (in Sweden blue) areas usually have not any special restrictions.

If there is surface contamination in the working place, workers must be protected against radioactive dirt and at the same time the spreading of the dirt must be prevented. This can be achieved by cleaning the working place and/or by covering working surfaces and radioactive components with plastics.

The most common protective method against the spreading of surface contamination is the adoption of a shoe boundary area. The shoe boundary separates the more contaminated (orange and red) areas from the parts of the controlled area which have a "normal" contamination level (green). The shoe boundary area can be located around a single component under repair or it can cover a larger area. There can also be shoe boundaries inside each other.

Additional protective clothing is worn when the shoe boundary is crossed. The extra personal protective equipment may include extra shoe covers (cloth or plastics), boots, gloves (cloth or rubber), extra overalls (cloth or paper), rainwear and hoods. The protective equipment can be either disposable or reusable. When one returns from the area inside the shoe boundary, the extra protective equipment is taken off.

The extra protective equipment may also include respirators, which can be either filtering or fresh-air type. Filtering respirators are half or full masks which employ dust or gas filters or their combinations. In fresh-air equipment, the user gets fresh and clean air through a hose or the air comes from a portable tank.

The measures taken to protect workers against external direct radiation include keeping the distance from radiation sources as long as possible, minimization of working time, and various radiation shields attenuating radiation. Radiation dose rate in the working place is usually reduced by means of movable lead shields, particularly lead blankets.

4.2 Project activities

The purpose of this study was to examine the ways in which the optimization of radiation protection has been applied to the use of protective methods and equipment for the protection of workers.

The background information for this study was acquired in the form of answers from all the Nordic nuclear power plants to a questionnaire. The results were completed and adjusted with a supplementary questionnaire. Partly on the basis of the answers to these two questionnaires and on the basis of discussions with health physics personnel in Nordic nuclear power plants, a master's thesis was completed at the University of Turku. This very thorough and extensive thesis was abridged to the NKA's project report which forms the basis of paras 4.3 - 4.5 in this chapter /4-1/.

In addition a special exercise was made to develop a model for the calculation of the spreading of radioactive contamination in a nuclear power plant /4-2/. The results are described in para 4.6.

4.3 Use of protective methods and equipment at the Nordic nuclear power plants

4.3.1 Background information

The following data have been obtained by means of questionnaires to the Nordic nuclear power plants and they are mainly based on the situation in the years 1983 to 1986. They give some figures of the use of protective methods and equipment. The tables combine the data on all units at each power plant. Because reactor type, power and the number of units at a plant site vary, the numerical data presented in the tables are not wholly comparable with each other. It must also be remembered that the total duration of an annual maintenance outage and the maintenance operations performed vary from year to year. The protective equipment at the plants are also of somewhat different types.

For background information the numbers of workers, collective radiation doses of workers and durations of outages at the plants are shown in Tables 4.1, 4.2 and 4.3.

Table 4.1 Number of persons working in the controlled area during operation and outages.

Power plant	Number of workers (operation/outage)		
	1983	1984	1985
Barsebäck (2)*	300 / 1000	300 / 1000	300 / 900
Forsmark (2)	150 / 700	150 / 1000	250 / 1300
Loviisa (2)	400 / 950	350 / 950	350 / 900
Olkiluoto (2)	450 / 1350	450 / 1450	450 / 1400
Oskarshamn (2)	350 / 1000	300 / 850	400 / 1200
Ringhals (4)	800 / 2000	800 / 2000	800 / 2000

* The number of plant units taken into account in this study.

Table 4.2 Collective doses of workers in the years 1983 - 1985 during operation and outages.

Power plant	Collective doses (operation/outage) (manSv) 1983	1984	1985
Barsebäck	0,4 / 1,8	0,9 / 1,0	0,3 / 0,7
Forsmark	0,2 / 0,6	0,5 / 0,8	0,3 / 0,7
Loviisa	0,1 / 1,3	0,1 / 1,8	0,1 / 1,0
Olkiluoto	0,2 / 0,8	0,2 / 1,0	0,2 / 0,7
Oskarshamn	0,3 / 2,4	0,4 / 1,6	0,7 / 2,0
Ringhals	2,4 / 6,7	1,3 / 4,9	1,0 / 5,0

Table 4.3 Total duration of outages.

Power plant	Total duration of outages (d) 1983	1984	1985
Barsebäck	81	61	33 1)
Forsmark	44	65	46
Loviisa	54	67	41
Olkiluoto	64	47	46
Oskarshamn	84	77	161
Ringhals	120	140	140

1) Outage only at one unit.

The duration of the outage was determined by the longest maintenance operation, or the so-called critical path. In case no extensive special operations were performed during the annual maintenance, the critical path consisted of refuelling and reloading.

4.3.2 Extra boundaries for contaminated areas

The establishment of an extra boundary for a contaminated area (shoe boundary) is mainly based on the amount of surface contamination. Working time, number of workers, size of working object and level of gamma radiation are other factors which might affect this decision. For example, cases in which the duration of the work is very short and the number of persons participating in the work is small, can be handled without an extra shoe boundary.

The number of extra shoe boundaries established during operation and outages at plants in 1985 and 1986 are shown in Tables 4.4 and 4.5.

Table 4.4 Total number of permanent and temporary extra shoe boundaries during operation.

Power plant		Number of shoe boundaries permanent / temporary 1985	1986
Barsebäck	(2)	20 / 15	20 / 15
Forsmark	(2)	10 / 5 - 15	30 / 20 - 30*
Loviisa	(2)	5 / 10	5 / 10
Olkiluoto	(2)	10 / 20	10 / 20
Oskarshamn	(2)	30 / 1 - 5	30 / 1 - 10*
Ringhals	(4)	20 / 40	20 / 45

* Three plant units were taken into account instead of two in 1985.

Table 4.5 Total number of extra shoe boundaries during outages.

Power plant		Number of shoe boundaries 1985	1986
Barsebäck	(2)	90	100
Forsmark	(2)	100	200*
Loviisa	(2)	20	20
Olkiluoto	(2)	80	70
Oskarshamn	(2)	60	90*
Ringhals	(4)	80	80

* see Table 4.4.

4.3.3 Extra protective clothing

The criteria for the use and selection of extra protective clothing have been very similar at all Nordic nuclear power plants. Protective clothing is required when the amount of contamination exceeds the limit for the green (blue) classification. The value and spreading of contamination and the quality of the work to be performed have a bearing on the selection of extra protective clothing. If contamination is restricted to the component that is being worked on, it may be sufficient just to wear gloves. If contamination has spread to the whole room, the gloves are supplemented with extra shoe covers and overalls. If contamination is wet or the component is highly contaminated, protective clothing which suits the situation, such as boots, rubber gloves and rainwear, are used. At the plants, there are many jobs every year the nature of which is known in advance. So the need for protective clothing is determined by experience.

Heavy protective clothing can prolong work to some extent. When the project involves a high level of gamma radiation, protective clothing is selected so that the resulting total dose is as low as possible. Ambient conditions (temperature, humidity) have also an effect on the selection of the type of protection.

Tables 4.6 and 4.7 illustrate the average numbers of disposable and reusable protective clothes used annually.

Table 4.6 Average number of disposable gloves (cloth and rubber), overalls, hoods and rainwear used annually.

Power plant	Cloth gloves (pair)	Rubber gloves (pair)	Overalls	Hoods	Rain-wear
Barsebäck (2)*	15000	75000	1000	2000	400
Forsmark (2)	37000	49000	1800	3200	200
Loviisa (2)	26000	14000	400	-	200
Olkiluoto (2)	45000	40000	2100	100	100
Oskarshamn (2)	7000	61000	600	2800	600
Ringhals (4)	24000	73000	1000	1000	50

* The number of plant units.

Table 4.7 Average number of reusable shoe covers, boots, overalls, hoods and rainwear used annually.

Power plant	Shoe covers (pair)	Boots (pair)	Overalls	Hoods	Rain-wear
Barsebäck	12000	200	6000	-	-
Forsmark	10000	80	4200	3500	-
Loviisa	3500	40	2200	-	200
Olkiluoto	10000	50	5000	100	20
Oskarshamn	3000	160	2500	3000	80
Ringhals	2000	20	2900	1000	120

4.3.4 Respirators

There are both filtering respirators and fresh-air equipment in use at the Nordic nuclear power plants. Filtering respirators (half and full masks) can be provided with a dust or gas filter. The same filter can be used several times if its service life and the activity level does not prevent this. At least at some plants, each filter is used only once. The annual numbers of filtering respirators and filters in use are illustrated in Table 4.8. The number of fresh-air equipment with overpressure is shown in Table 4.9.

Table 4.8 Average annual number of filtering respirators (masks) and filters in use.

Power plant		Masks	Filters
Barsebäck	(2)*	550	1900
Forsmark	(2)	400	5100
Loviisa	(2)	200	400
Olkiluoto	(2)	350	2500
Oskarshamn	(2)	450	1900
Ringhals	(4)	600	2900

* The number of plant units.

Table 4.9 Average annual number of fresh-air equipment with overpressure in use.

Power plant	Fresh-air equipment
Barsebäck	20
Forsmark	50
Loviisa	5
Olkiluoto	10
Oskarshamn	35
Ringhals	20

The selection of the type of respirator is primarily determined by the concentration and quality of air contamination. Other contributing factors include ambient working conditions (temperature, humidity), the amount of surface contamination, working time and the quality of the job to be performed. If, for example, both air contamination and a high temperature prevail, a fresh-air suit may be used for working. Even if the level of gamma radiation were relatively high, sufficiently effective equipment is chosen to protect workers against airborne contamination, despite a possible prolongation in the working time.

4.3.5 Lead shields

The most common lead shields at Nordic nuclear power plants are lead blankets, sheets and bricks. Table 4.10 shows the average number of lead blankets used at the various power plants.

Table 4.10 Average annual number of lead blankets in use.

Power plant		Lead blankets
Barsebäck	(2)*	350
Forsmark	(2)	1000
Loviisa	(2)	300
Olkiluoto	(2)	300
Oskarshamn	(2)	300
Ringhals	(4)	1250

* The number of plant units.

The purpose of lead shielding is to keep the total dose resulting from the work as low as possible. It is mostly the duration of the job that determines the need for shielding. No actual dose rate limit has been indicated, which would directly require shielding.

4.3.6 Whole-body counting

The internal exposure of persons working in the controlled area is monitored by means of whole-body counting. In particular, representatives of so-called risk groups shall be monitored (e.g. radiation monitoring personnel, cleaners and maintenance personnel for valves). According to the usual practice, the persons to be measured wash themselves and change into clean overalls or coats before the whole-body counting. Some 100 to 500 workers go through a precision whole-body counting at a plant annually.

The internal exposures of workers at Nordic nuclear power plants have been relatively low. For instance, in the years 1983 - 1988 there had been only a few workers in whom the detected amount of radioactivity in the body was more than 0,5% of the ALI-value combined with a specific radionuclide. The radionuclides that have been detected in the body of a worker at a nuclear power plant are typically ^{60}Co, ^{58}Co and ^{54}Mn.

4.4 Optimization of the use of temporary shielding

4.4.1 The α-value for temporary shielding

There are three basic means offered by radiation protection for reducing the dose caused by external radiation.

The radiation dose can be reduced by minimizing the working time in the radiation field. The dose can be reduced by staying as far from radiating objects as possible.

The third means of protection is to reduce radiation with the help of various radiation shields. The most commonly used shields are movable lead blankets or sheets.

The general objective is to make individual and collective doses as low as reasonably achievable. In a room with gamma radiation, the need for shielding is determined by the dose rate, working time and the number of workers. In addition, one must consider the dose received by the installers of the radiation shields and possibly the more difficult working conditions and the longer working time because of the shields. It may also be necessary to move the shields several times in the working location during the work.

To calculate the α-value of a lead blanket, we must know the total costs (ΔX) caused by the use of the blanket and the dose reduction (ΔS) obtained with the blanket. The ratio $\Delta X/\Delta S$ is the α-value of the use of the lead blanket.

The total costs of the lead blanket are estimated for one use, even though the uses may be very different. It is assumed that the blanket will be used for five years, three times a year and that it will cost FIM 300 - 800 (SEK 400 - 1200).

Lead shielding (moving, installation and dismantling of lead blankets) costs about FIM 100 (SEK 150)/h per person doing the work. Usually a lead blanket is quick to install and dismantle. Installation costs for one blanket are small, only a couple of marks (crowns). Costs arising from the maintenance and cleaning of the blanket and its storage are also quite small. Thus the total costs arising from the use of a lead blanket are almost wholly made up by the price of the blanket for one time of use. If we add some other expenses to this, the average total cost ΔX of a lead blanket comes to about FIM 40 (SEK 60) for one use.

The lead's half value layer (HVL) for ^{60}Co (1.17 MeV, 1.32 MeV) is 1.2 cm and for ^{137}Cs (0.662 MeV) 0.65 cm. The mean energy of γ-radiation usually lies between 0.6 - 0.8 MeV. This means that a normal lead blanket reduces the original dose rate (without leading) to about one half of the initial value.

In the following text, the duration of the work is expressed as hours instead of manhours (manh) just for the sake of simplicity. It means that when the duration of the work is for example 4 h, there can be 2 persons working for 2 h or 1 person for 4 h. Dose always means collective dose. Below we shall examine cases in which the use of a lead blanket results in a dose reduction between 33% and 50% compared with a situation in which there is no shielding. The doses to workers performing the lead shielding and the possible prolongation of the working time because of the blanket have already been taken into consideration in calculating the dose reduction. The calculated α-value curves in the two different cases have been plotted in Figure 4.1.

The curves show that in all cases in which the dose rate is at least 1 mSv/h and the work takes at least 2 h, a lead blanket has a positive protective value if we use FIM 80 000/manSv as the α-value in the comparison. If the work takes only an hour, the initial dose rate must be at least 2 mSv/h to make it worthwhile to use a lead blanket, taking the various cases into account. The longer the working time, the lower the dose rate required for useful protection. A crucial factor in the calculations is the total cost of the blanket for each time it is used.

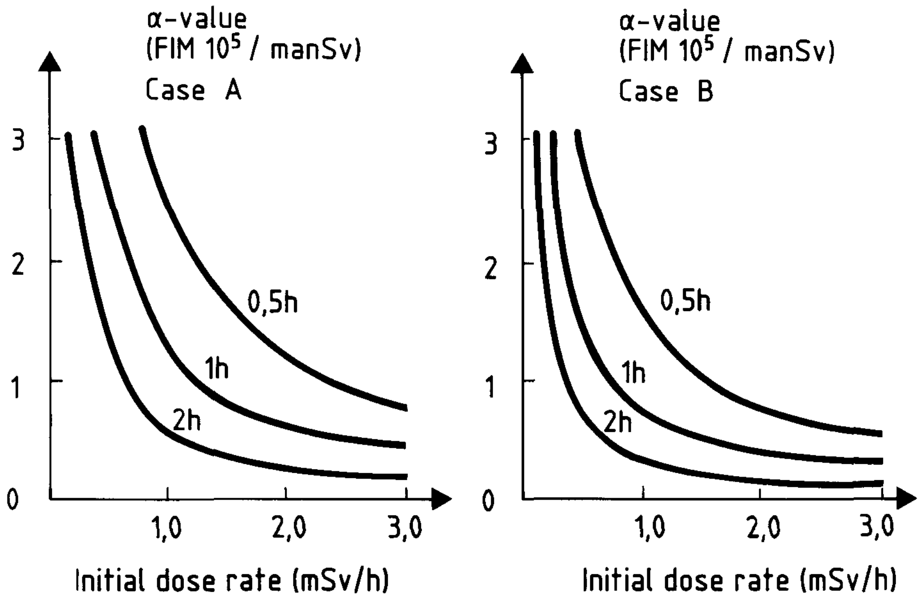
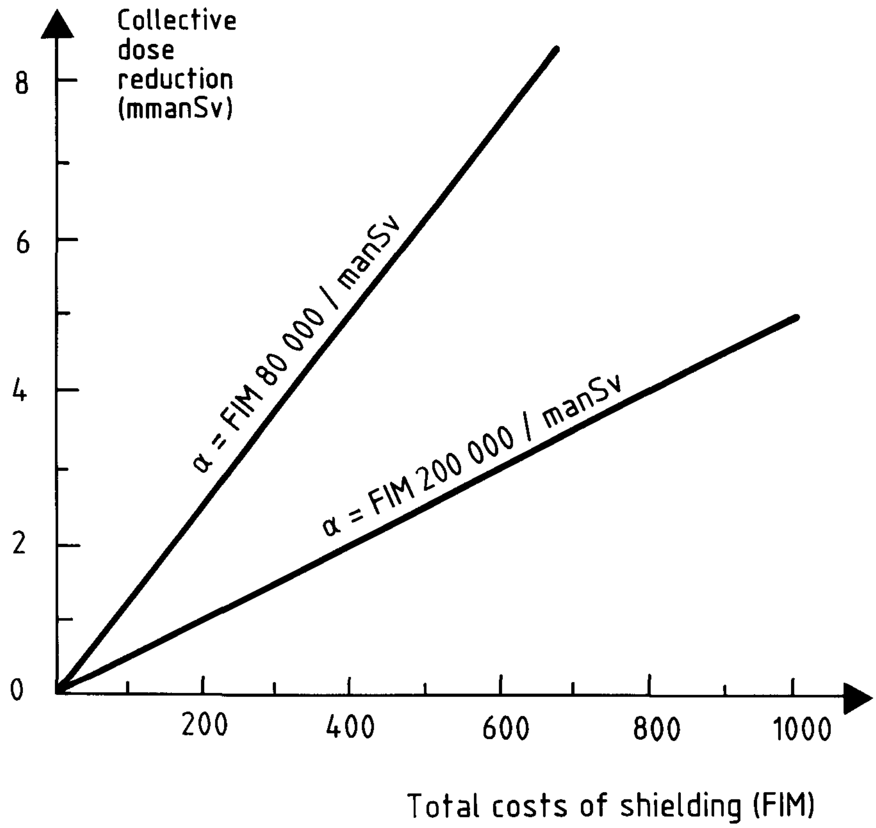


Figure 4.1 The dependence of the α-value of one lead blanket on the dose rate and the working time. The total costs caused by the use of the lead blanket for each time are FIM 40. The dose reduction achieved through leading is 33% in case A and 50% in case B.

In studying the profitability of using lead shields one must estimate the average dose rate in the working place before and after shielding, the number of workers in the place and those engaged in shielding work, and the duration of the work and that of preparing lead shielding. The dose reduction achieved with shielding must be assessed on the basis of this information.

By estimating the total costs induced it is possible to calculate an α-value for each case. It is essential that the total dose is reduced significantly (Figure 4.2).



<u>Figure 4.2</u> The figure shows the minimum reduction that should be achieved in the collective dose when we know the total costs of lead shielding. This has been shown with two different α-values.

Practice has shown that temporaray shielding is used more often in jobs of long duration than in short ones. This practice also conforms to the ALARA-principle.

4.4.2 A general way to calculate the α-value

In a general case the α-value of using lead shields can be estimated as follows. It is assumed that lead shielding reduces the total dose received without shielding by a half. This reduction takes into account the possible longer working time due to lead shielding, the doses received by the workers performing the shielding, etc. The α-value curve is plotted in a hypothetical case where the total costs of lead shielding are FIM 1 and the total working time, i.e. the number of man-hours, is 1 h (Figure 4.3). For the desired initial dose rate it is possible to get a preliminary α'-value from the curve. This in turn can be converted to an α-value corresponding to different situations. By multiplying the α'-value obtained from the figure with the total cost of leading ΔX in the case under examination and by dividing this result with man-hours T used in the work, we get the true α-value or the case in question, i.e.

$$\alpha = \alpha' \frac{\Delta X}{T} \qquad (4.1)$$

If for example the average dose rate in the working place is 2 mSv/h, them α'-value obtained from Figure 4.3 is about FIM 1000/manSv. If ΔX = FIM 40 and T = 0.5 h, the final α-value will be 1000 x 40/0.5 FIM/manSv = FIM 80 000/manSv (cf. Figure 4.1, case B).

In cases where the dose reduction is not one half, the prosedure is as follows: The α-value calculated as above is multiplied by factor 1/2k, where k indicates the real dose reduction factor received with shielding. In this case the final α-value is

$$\alpha = \frac{1}{2k} \alpha' \frac{\Delta X}{T} \qquad (4.2)$$

If k = 1/2, we come to formula (4.1). If k = 1/3 (i.e. the dose reduction is 33%), the α-value is then 3/2 times the α-value obtained with formula (4.1). In the case examined above, the α-value will be FIM 120 000/manSv when k = 1/3 (cf. Figure 4.1, case A).

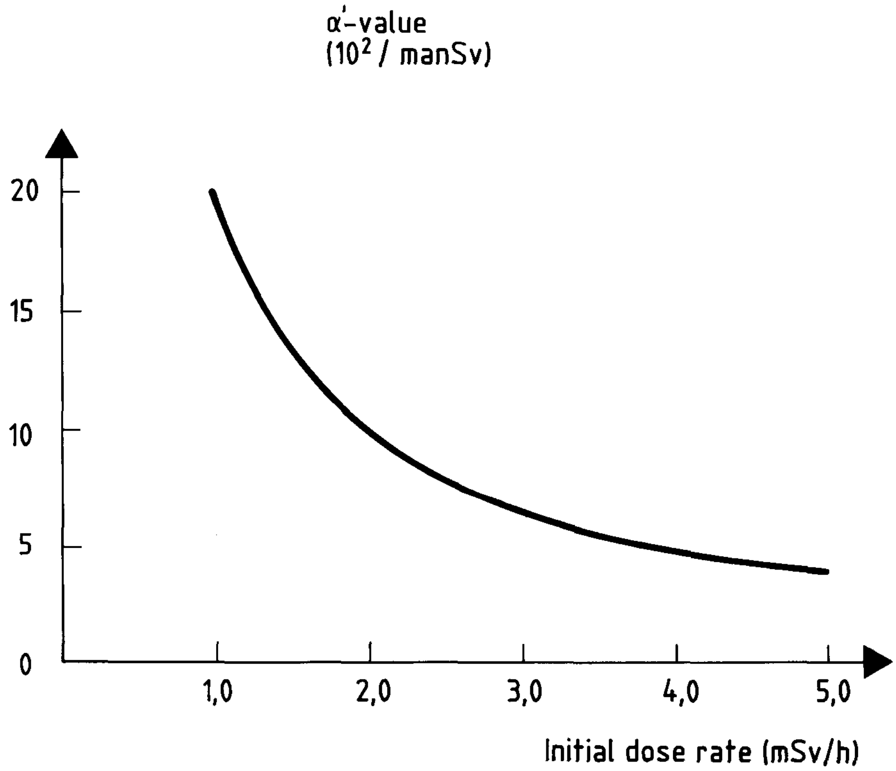


Figure 4.3 The curve shows the price of the dose saved with lead shielding and its dependence on the initial dose rate with following assumptions, the cost of shielding is FIM 1, the number of manhours is one and shielding reduces the total dose with 50%. The final α-value is obtained by multiplying the α-value obtained from the figure with the total cost of lead shielding ΔX and dividing it with man-hours T.

4.5 Optimization of the respiratory protective equipment

4.5.1 The α-value for respirators

When working surfaces are contaminated there can always be radioactive aerosols in the air and they can get into the body. To avoid this, one must use respirators.

Half and full masks with suitable filters are the most commonly used respirators. Fresh-air equipment are also used to some extent, provided with a suit, a mask or a hood.

The choice of the type of respirators and filters depends primarily on the concentration and quality of air contamination, but the object of work and the conditions of the working place also affect the choice. If the face is also to be protected, it is best to choose a full mask instead of a half mask. In humid and hot conditions for example, it is well-founded to use a fresh-air suit.

The effectiviness of respirators is given as a decontamination factor, DF, which shows the fraction to which the air impurity content falls when air passes through the respirator.

The first step is to calculate the dose prices ("α-values") that can be saved by using the respirator. The α-values are calculated for one use. The average service life of half and full masks is about five years and it is assumed that the masks will be used ten times a year. The average service life of fresh-air equipment is about ten years, but a hood, a mask or a suit has only a life of one or two years. It is assumed that the equipment will be used three times a year. In calculating the costs of respirators for one use, one must note that the filters are replaced after each use.

Thus the filters account for most of the cost of using half and full masks. The costs of washing and servicing the respirators must also be included. Table 4.11 shows the assumed decontamination factors and the estimated total costs of the respirators under examination for one use. However, it should be remembered that the decontamination factors may be higher or lower than the values below.

Table 4.11 Decontamination factors and estimated costs of respirators.

Respirator	Total costs for one use (FIM)	Decontamination factor DF
Half mask and dust filter	12	30
Full mask and dust filter	35	100
Fresh-air equipment and a suit, hood or mask	250	1000

In calculating the doses saved with the use of respirators, the following formulas are used. The internal dose S' received when the respirator is used is

$$S' = \frac{K \cdot T}{DF} \times \frac{0.05}{2000} \qquad (4.3)$$

where K is the average air contamination (DAC), T is the time that the respirator is used (h), DF is the decontamination factor of the respirator and 0.05 / 2000 is the conversion factor (Sv/(DAC x h)). When the respirator is not used, the internal dose S'' received is

$$S'' = K \cdot T \times \frac{0.05}{2000} = DF \cdot S' \qquad (4.4)$$

Thus the dose saved by using the respirator is ΔS

$$\Delta S = S'' - S' = S'(DF - 1) \qquad (4.5)$$

The formula (4.5) for the dose saved by using a respirator can also be written as

$$\Delta S = S'' - S' = K \cdot T \, \frac{0.05}{2000} \, (1 - \frac{1}{DF}) \qquad (4.6)$$

It is seen from this expression that when the decontamination factor DF is high, the dose reduction is almost exactly the same as S''. When the decontamination factor is 30, the dose saved with the respirator is 3,3% smaller than the dose reduction in a case in which the protection achieved is 100%.

Now we can calculate the α-values by using various values for airborne contamination and for the time that the respirator is worn (Figures 4.4, 4.5 and 4.6). <u>Any prolongation in the working time that may be caused by the use of the respirators has not been taken into account.</u>

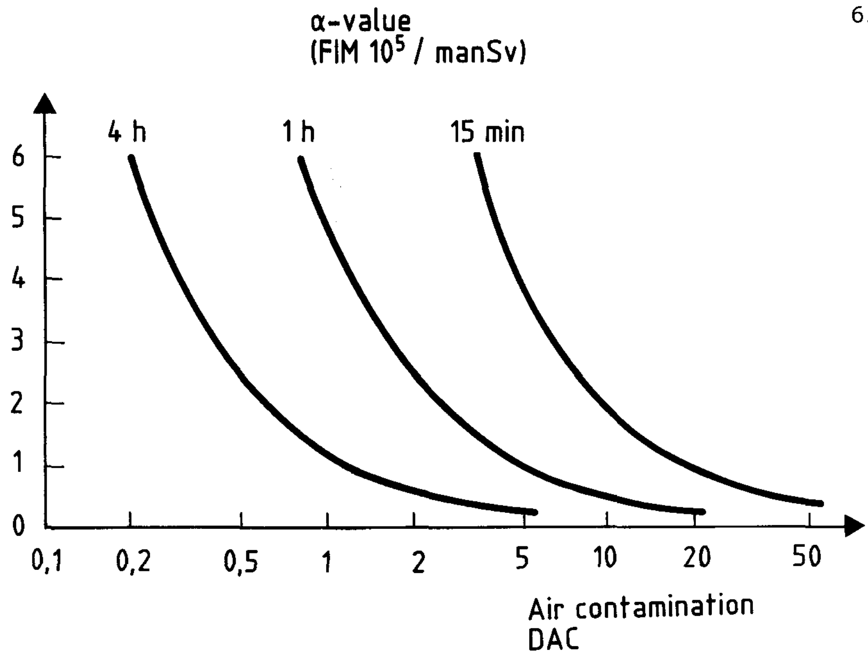


Fig. 4.4 The α-value of a half mask provided with a dust filter and its dependence on air contamination and time of use. The decontamination factor of the respirator is 30 and total costs for one use are FIM 12.

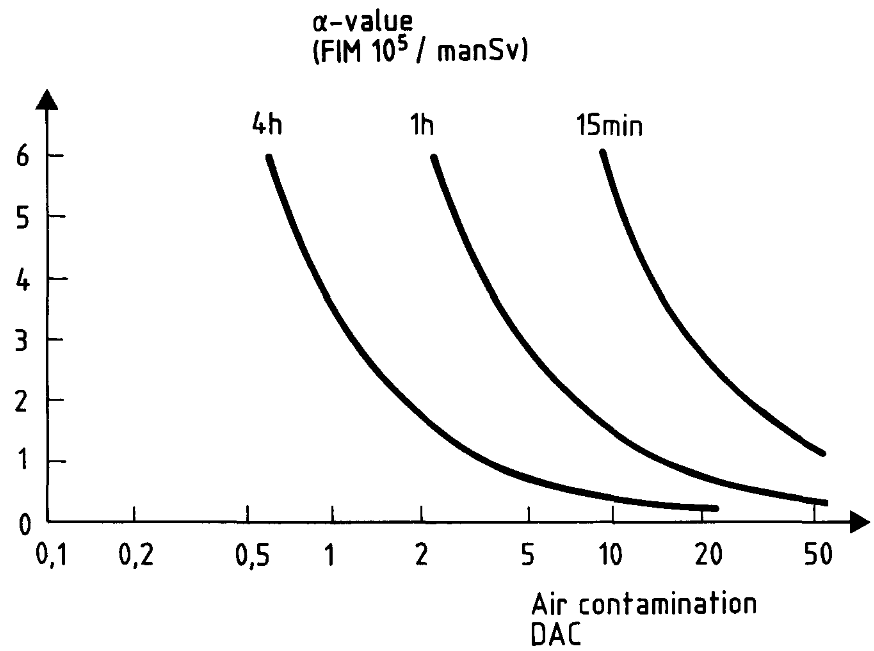


Fig 4.5 The α-value of a full mask provided with a dust filter and its dependence on air contamination and time of use. The decontamination factor of the respirator is 100 and the total costs for one use are FIM 35.

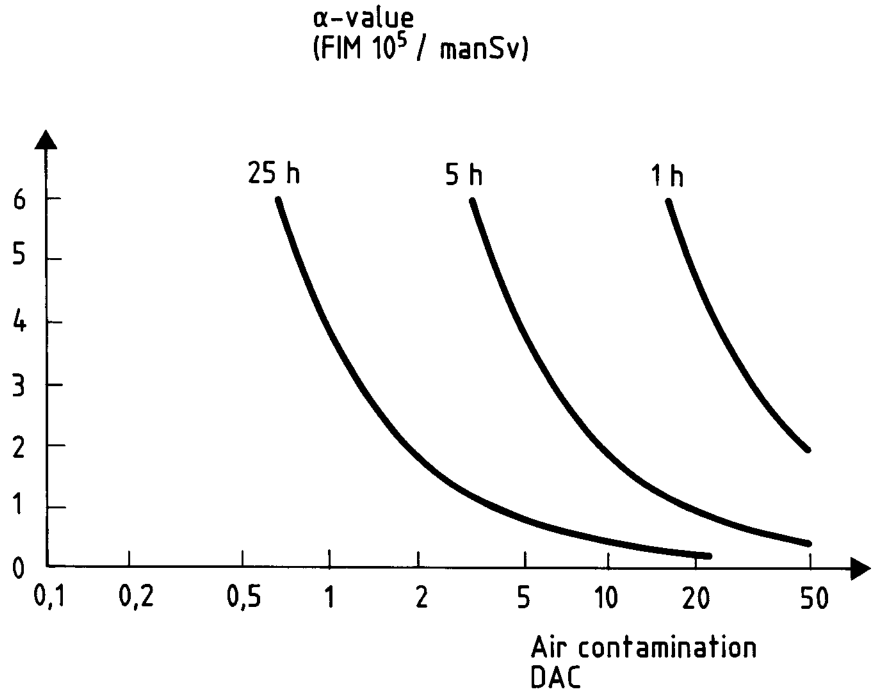


Figure 4.6 The α-value of fresh-air equipment provided with a hood, a mask or a suit and its dependence on air contamination and time of use. The decontamination factor of the respirator is 1000 and the total costs for one use are FIM 250.

4.5.2 A general way to calculate the α-value

In fact the figures examined above can be replaced with one figure and a curve. By suitably working up the α'-values obtained from this curve we can come to the α-value for the situation that is studied (Figure 4.7).

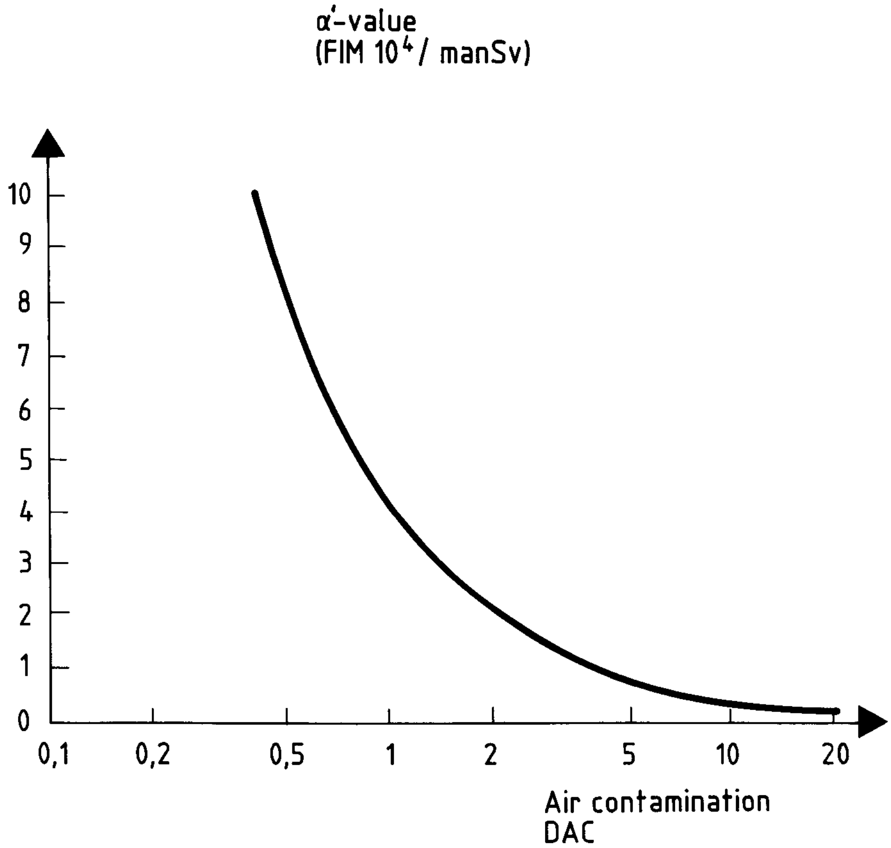


Figure 4.7 The curve shows the price of the dose saved with respirators and its dependence on airborne contamination with certain assumptions.

The figure shows the dependence of the α'-value on the average airborne contamination with the following assumptions. The total cost of the use of respirators is FIM 1, the respirators are used for 1 h and the decontamination factor is 50. Figure 4.7 can be applied to various situations as follows. The α'-value obtained from the figure with a certain value of airborne contamination is multiplied with the real operating costs of the respirator(s) ΔX. By dividing this result with the combined time the respirators are used, i.e. man-hours T, we come to the final α-value for the case in question. (c.f. formula (4.1). The effect of the decontamination factor of the respirator (DF > 30) is of so little importance that it need not be taken into account. Let us assume for example that ΔX = FIM 12, T = 4 h and the air contamination K = 1 DAC. The preliminary α'-value obtained from Figure 4.7 is in that case about FIM 40 000/manSv. By multiplying FIM 40 000/manSv by factor 12 and by dividing it by 4, we come to the final α-value FIM 120 000/manSv. The result is approximately the same as the α-value obtained from Figure 4.4 in a corresponding case.

4.6 A calculational model for the spreading of radioactive contamination

4.6.1 Spreading of radioactive substances at a nuclear power plant

At the moment the risk of internal doses from radioactive contamination is small in Nordic nuclear power plants. The purpose of this exercise was to develop a model and a computer program for the spreading of contamination and exposure of workers to contamination in a nuclear power plant. It was hoped that the model could be used as a tool in planning the use of protective equipment.

Small amounts of radioactive substances are released inside a nuclear power plant in certain situations. Maintenance work and repairs often involve disassembly of contaminated components, which can release radioactive water or steam.

Also in machining components, radioactive particles (e.g. grinding dust) can be released into the environment. Radioactive water spilled on the floor can evaporate, and the radioactive substances remaining on the floor can easily be released into the room air. In order to prevent the spread of air and surface contamination, a working area can be isolated using plastics, applying underpressure or separating the area with a shoe boundary.

Radioactive particles can, however, be carried over the shoe boundary by contaminated workers or by air in the case of poor ventilation in the area. Outside the confined area the contamination spreads wider and wider while the concentration of it becomes lower.

4.6.2 Modelling

In the model developed for this work radionuclide transport and determination of the concentration of radioactive substances is treated by means of a compartment model. The kinetics of the transfer of radionuclides is expressed by a set of first-order differential equations. Each equation expresses the amount, inventory, of radioactive substances in the compartment at a given moment. The transfer of a radioactive substance from one compartment to another is expressed by linear flows, where the amount of the substance transferred is directly proportional to the transfer coefficient and to the inventory in the source compartment.

The principle of the model is illustrated in Figure 4.8.

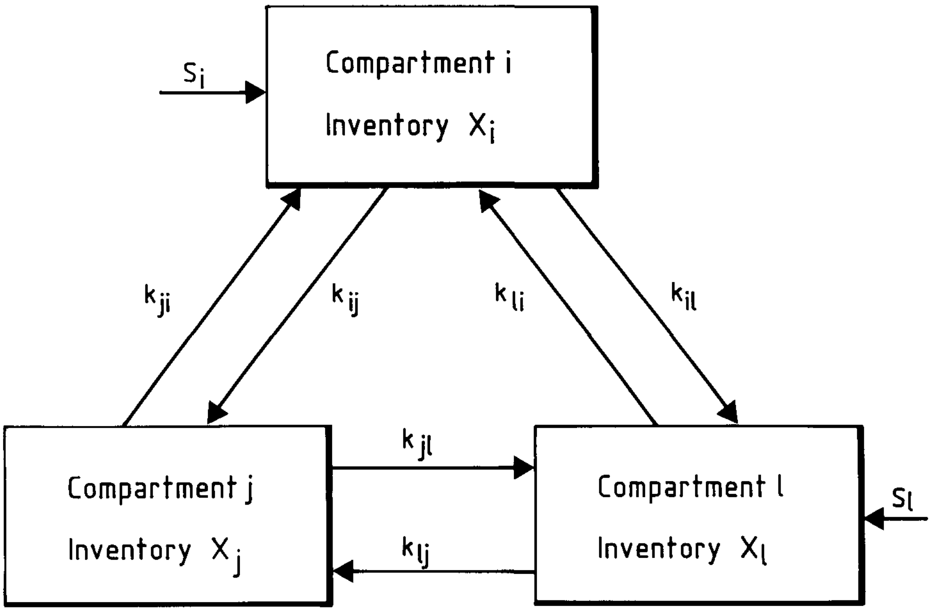


Figure 4.8 The principle of the compartment model

Figure 4.9 illustrates the way the model can be used. It is assumed that grinding work is done inside a shoe boundary.The working area is divided into an air compartment (1) and a surface compartment (2). The grinding results in air contamination (the release rate of activity must be known). The contamination can spread by air to the floor (k_{12}, k_{34}) to the air space outside the shoe boundary in the room (k_{13}), it can be resuspended from the floor (k_{21}) or spread on the floor (k_{24}, k_{45}).

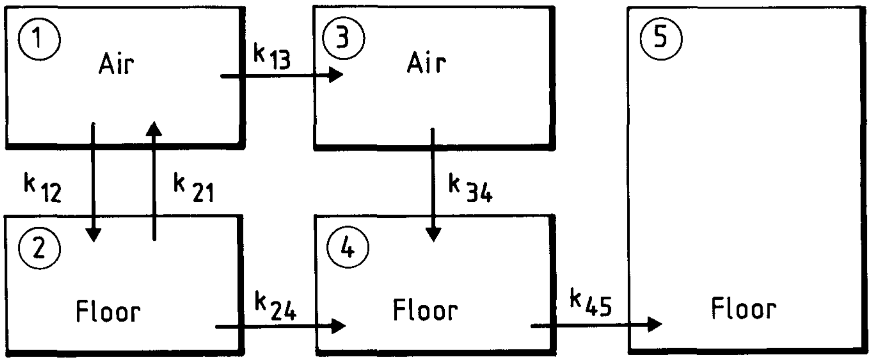


Figure 4.9 An example of the use of the model.

In calculating radiation doses, two dose pathways are to be considered: the external dose from surface contamination and the dose from inhaling contaminated air. The collective dose is obtained by multiplying the individual dose by the number of workers.

4.6.3 The computer code

The model was programmed into a computer code RAPRO (Radiation Protection Optimization) to calculate the spread of a radioactive substance and the consequent radiation doses inside a nuclear power plant. The goal was to compile a simple program to be used as a tool in radiation protection.

The program calculates the behaviour of only one radionuclide at a time; in this way any singularity of the calculated inverse matrice that may arise in certain situations is avoided. As input data, the nuclide half-life can be given, and the decrease in the concentration due to radioactive decay is taken into accout, when required. In the transport part of the program, the spreading of the radioactive substance in the compartment system is simulated time-dependently. The dose calculated is the product of the mean concentration, time spent and number of persons in the compartment, and dose coefficient.

As input data the program needs compartment sizes, source per time interval, number of persons working per time interval and the applicaple transfer coefficients. The maximum number of compartments is ten.

The transfer coefficients shall somehow be estimated, for instance they shall be derived from measurements in various working conditions in a nuclear power plant.

In this exercise, the model and the code were applied to an idealized grinding work situation. In that case, the purpose was to try to use realistic model parameters. The model requires further development and testing, especially parameters based on the results of measurements at plants to be used to calculate transfer factors, equilibrium activities and particle sizes.

4.7 Conclusions and recommendations

By conducting a complete optimization process in accordance with the ALARA procedure, one could find out all the factors that affect a case requiring radiation protection measures. At the same time one could identify the most essential factors and get a recommendation for suitable protection. The final decision on protective measures would be made then by using the results obtained from optimization.

In simple cases, such as optimization of radiation protection equipment, it seems unnecessary to go through the complete ALARA procedure. It is usually considered sufficient just to study the dose reduction or α-value achieved through protection. The final conclusions are drawn on the basis of the calculated α-value. In complicated cases, where decisions are made on a high level, the application of the whole ALARA procedure is recommendable.

Not all factors needed in the optimization in the use of radiation protection methods and protective equipment are always known very accurately. Thus it is not possible to give a reliable estimate of the dose reduction achieved through the use of most protective devices. There are cases where the comparison between alternative protective measures must be carried out more simply by comparing the advantages and disadvantages of the various measures.

The protective measures applied at the Nordic nuclear power plants are quite similar. The results now achieved in the optimization in the use of protective methods and protective equipment are not much different from the practice applied at the plants. Practice and experience have taught certain ways and methods which are fairly optimal. However, the use of extra shoe boundary areas and some protective equipment could perhaps be slightly reduced. The use of temporary shielding could be recommended.

The α-values calculated for the use of lead shields showed that the reduction in the total dose achieved with lead blankets is relatively advantageous. In practice, using lead shields usually results in a value that is clearly below FIM 80 000/manSv (SEK 120 000/manSv or US$ 20 000/ manSv). Consequently, the use of lead blankets could be somewhat increased.

The α-values calculated for the use of respirators are fairly high. They are generally FIM 100 000 - 500 000/manSv (SEK 150 000 - 750 000 or US$ 25 000 - 125 000/ manSv). Therefore respirators should be used with care and only when there is an actual need.

Because the risk of exposure to radiation can stir up fear, practical radiation protection should also take psychological factors into account. A feeling of safety can be increased also by using extra protective equipment fairly liberally, even if their use were not directly justified on the grounds of radiation protection.

4.8 References

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/4-2/ J Rossi, R Paltemaa, Optimization of Radiation Protection in Nuclear Power Plants, Contamination in working Environment, NKA Report RAS 410, 18.6.1987

5 OPTIMIZATION OF RADIATION PROTECTION IN RELATION TO PLANT SYSTEMS AND CONSTRUCTIONS

5.1 Introduction

Radiation doses of workers at Swedish and Finnish nuclear power plants are among the lowest in the world. At an overall level this can be attributed to the prevailing "technical culture", but in order to extract the deeper structure of the reasons for this situation, studies have been made focusing on the different aspects of radiation protection activities at the plants and in design.

It is very convenient to study practical optimization on a scale significantly smaller than the one that has to be considered when optimising a whole nuclear power plant. The problems in smaller optimization tasks are usually marginal, well defined and more suitable for study than the very intermixed and complex ones in large projects. This helps both to extract information about jobs done (looking back) and to achieve results when trying out optimization in practice on new problems (looking forward).

5.2 Project activities

The actual work concentrated mainly on common principles, rules and philosophies, enhancing exchange of experience, collecting information, hopefully leading to a common data base system, and developing rules and methods in order to make certain sectors of radiation protection work easier in the future.

The subproject NKA RAS 410:4 "Applicability of the optimization principle related to plant systems and constructions" was originally initiated in order to study principles and methods related to special dose-reducing actions at existing nuclear power plants. The dose-reducing actions in connection with new constructions, reconstructions and extensions were to be considered. It was later on decided to include dose-reducing actions in general in the study, in order not to lose valuable but possibly scarce information about optimization. The emphasis was to be put on both smaller and bigger reconstructions related to both buildings and parts of them, and to systems.

The study was established in the middle of 1985. The peak of the activity occurred during the first half of 1986, when the study of dose-reducing actions at the plants was performed /5-1/. Later activities were less intensive, mainly because of scarce manpower resources.

5.3 Applicability of optimization

The applicability of the ALARA principle within the area of this study has been an issue of discussion. Operational radiation protection people generally have a tendency to regard optimization as something that is really not needed for radiation protection decision- making at NPPs in operation, but which may be useful in design of new ones. Design people may resist optimization on various grounds, e.g. there may be a fear of having to cope with strict authority regulations, or a recognition of the very amount of work needed to achieve something by optimising. The present position of the ICRP is clear, in a pre-report /5-2/ discussing recommendations to be published it is stated that "Optimization of protection is an idea of very broad application. It can be used at all levels from simple day-to-day decisions to major analyses".

As a consequence of the development work in optimization methodology in recent years there is a clear trend in international recommendations and national legislation towards giving optimization a more significant role.

One major difficulty in discussing the applicability of the optimization principle is the fact that the different persons involved usually have very diverging views of the meaning of ALARA. In the following ALARA is primarily considered to have both quantitative content and a procedural meaning, and an effort will be made to bring new elements into the "optimization" as it has traditionally been done at the plants. This means nothing more complicated than a significant improvement of radiation protection decision-making in the direction of greater clarity, ease and efficiency.

5.4 Study of optimization at Swedish and Finnish plants

The main study was an attempt to map the optimization activities that had taken place at Swedish and Finnish nuclear power plants concerning extensions and reconstructions. At an early stage it was realized that it was not possible to separate the activities into different areas, such as those related to buildings and other civil constructions and those related to process systems. Instead, the scope of the study was chosen to be "the applicability of the optimization principle related to plant systems and constructions".

The overall radiation protection work at the Nordic plants is affected by the fact that both the dose levels and the radiation doses to workers are low in an international perspective. One reason why the situation could be considered optimal is the trend at Swedish plants to approach the ambition level 2 manSv/a GWe as a long time average. This does not mean that the level is optimal with respect to objective health effects (α-value), but puts a rather strong emphasis on the β-term of the ICRP-detriment model. There is not a directly corresponding ambition level for Finnish plants, but, instead, there is an intervention level of 5 manSv/a GWe, which in practice reflects a same type of ambition. Because of the various views of the reasons for the overall situation, there was a strong incentive to try to find common principles in plant-specific practices and design philosophies.

It was decided to collect an inventory of different radiation protection actions in order to extract the basic principles and guidelines used for judging dose-reducing actions. In order to get an undistorted picture of the practices at the various plants, it was also necessary to have a complete coverage of all kinds of radiation protection actions within the scope of the study.

A limited analysis of dose-reducing actions was performed using existing and easily retrievable information to plan the study and to get information about possible difficulties. As a result it was decided to study, as broadly as possible, both actions which were and were not taken. It was also decided not to study probabilistic events.

Information for the study was directly and indirectly gathered from all Nordic nuclear power plants. The time span of interest to the study was mainly about 8 years. A total of about 320 dose-reducing actions were collected, all of which could not be studied in detail. A screening procedure was applied, resulting in 110 actions for further study.

The questionnaire presented in Figure 5.1 shows the format used for the gathering of information. Table 5.1 gives explanations to the various requests.

Group			
	Object Action Reason Reference		No
1			
	Dose Red Act DRA reason Design limit Comment 1		DRA cost
2			
	Method doseevaluation/shielddesign Source model Nuclides Exposure time Comment 2		Shield model Activity
3			
	Dose/doserate before DRA Dose during DRA Real dose saving Comment 3		After Calc dose saving
4			
	Opt model Cost Comment 4	Used cost/dose	Real cost/dose
5			

Figure 5.1 The final questionnaire

The final questionnaire, as seen on the screen of the computer, is shown in the figure. It contains a total of 28 entries divided into 5 groups.

Group 1 describes the action that is taken, where it is taken and why.

Group 2 describes dose-reducing actions, reasons, costs and design limits.

Group 3 describes the method used for the evaluation dose rates and doses and for the design of the action.

Group 4 describes dose rates before, after and during the action.

Group 5 describes the type of optimization, if any, used for the action.

The different entries in the questionnaire, including some short explanations and comments are given in table 5.1.

Table 5.1 Information collected in the detailed analysis.

	Data entry	Comment
Group 1:	1. Object	Where does the action take place, plant, system?
	2. Action	What type of action (A-G, see 2.1)
	3. Reason	What is the reason: satety, RP, authority etc?
	4. Reference	Report, contact person etc
Group 2:	5. Dose reducing action	Actions A-G in more detail
	6. Dose reducing action reason	
	7. Dose reducing action cost	
	8. Design limits	Design limits for dose or dose rate
	9. Comment 1	
Group 3:	10. Method for dose eval /shield dim	Method for evaluation of doses/dose rate and for design of DRA action.
	11. Source model	Assumptions for design of
	12. Shield model	RP actions, conservatism versus realism
	13. Nuclides	"-
	14. Activity	"-
	15. Exposure time	"-
	16. Comment 2	"-
Group 4:	17. dose/dose rate before DRA	Estimation of the results attained by the actions
	18. After	"-
	19. Dose during DRA	"-
	20. Calculated dose saving	"-
	21. Real dose saving	"-
	22. Comment 3	
Group 5:	23. Optimization model	Type of optimization if any
	24. Cost of optimization work	Cost of the resources used for optimization
	25. Used cost/dose	Cost/dose used in optimization
	26. Real cost/dose	
	27. Comment 4	

Dose-reducing actions were divided into the following categories:

- shielding
- replacements / reconstructions
- installation or use of supporting equipment or systems
- dose measuring and alarm equipment
- documentation of active areas and sources
- other actions
- new constructions.

More detailed description of the statistics on answers and practical examples of filled-in questionnaires are given in the project report /5-1/.

5.5 Results of the study

The first part of this section presents in a concise form the results based on the project report and the last part points out some additional features which could be seen in the study, but which were not clearly stated in the report.

The following will present some observations regarding the various entries. These have been condensed from the corresponding section in the report of the study.

A. Object:

A large variety of objects were listed, but the majority of dose-reducing actions referred to objects causing high dose rates or doses, such as

- steam generators
- radwaste treatment systems
- primary circuit
- pools
- reactor tank during outages.

B. Action type / Dose-reducing action:

A distinction was made between two types of action. Action meant the action taken in general, perhaps not directly related to radiation protection. Dose-reducing action meant a specific action leading to a change in the radiation conditions.

The actions described were of a great variety. The most common dose-reducing actions were of the type 'shielding' and 'supporting equipment'.

C. Reason / Dose-reducing action reason:

The reasons for dose-reducing actions were generally very complex. Most of them were taken for pure radiation protection reasons, i.e. where individual or collective doses could be reduced by rather simple actions. These were generally connected with high local or general dose rates. Other major reasons for dose-reducing actions related to operation or safety. These were often concurrent with radiation protection reasons. The remaining reasons were connected to the need for replacement or new construction based on e.g. operational experience at the plant.

D. Design limits:

The design limits for most actions were the limits set by authorities, or various internationally accepted rules. Internal rules and "good practice"-rules were also contributing.

E. Methods used for the evaluation of doses and design of actions:

Doses and dose rates were in many cases measured and simple analytical models were used to design the dose-reducing actions. Engineering judgement was used for most actions having a low cost. Actions with higher costs were evaluated in more detail and sometimes computer codes were used.

F. Dose or dose rate before, after and during the action / Calculated saved dose / Real saved dose:

Somewhat surprisingly, this information had been difficult to obtain. One significant reason seemed to be the lack of documentation. Formal reports existed to a larger extent in connection with major new constructions, or when the action was expensive. However, rigorous work was done when planning the actions, which meant that information existed at the appropriate stage, but was afterwards difficult to find.

Evaluations of saved doses were in most cases not done.

Some retrospective analyses had been made when data on doses before and after the action had been available.

Most actions had resulted in lower doses. In a few cases the actions had caused higher doses in other areas.

G. Optimization model:

Formal optimization was almost never done. Most actions were performed because they were needed, they had a low cost, or they resulted in a better working environment with a lower accident risk. Most actions were judged as cost- effective, which in some cases implicitly meant a high cost/dose-ratio, reflecting the relative significance of other factors than objective health detriment. The α- and β-factors were not normally distinguished from each other, and other positive effects of the action, such as increased availability, were not explicitly taken into account. These might however, initiate the need for a dose-reducing action.

The report of the study did not as such represent any complete result of the project. It only formed a basis for one of the main objectives of the project, which was to develop optimization thinking. It could be seen from the results of the study that most of the radiation protection decisions had been made on other grounds than those related to radiation protection. Most of the 330 measures taken to improve radiation protection were related to components and systems giving rise to high dose rates, not directly to high doses. The reasons for the measures were generally complex. They were based on design philosophy, purely radiation protectional needs such as individual and collective doses, and on operational and safety points of view. There were limits guiding construction, but they were quite elementary, such as generic authority requirements or predefined dose rate zoning limits. Methods for calculation and estimation were sometimes used, ranging from "yes/no" and "engineering judgement" to computer-based analytical methods. Dose rates before and after the measure were sometimes considered, but an evaluation of the dose saved was only rarely available. Explicit optimization evaluations, for instance cost-benefit analyses, were even more rare. Formal optimization was almost never done.

Some of the measures were associated with high costs, but despite that they had generally been considered cost-effective.

One finding was that it was sometimes difficult to obtain the information required. It must certainly have been available at the time of decision-making, but, due to difficulties in documenting such information which is mainly for use in real time, it was often lost. No attempts were made to reconstruct the situation, because it was judged as unreasonable from the point of view of efficiency. This matter is discussed in view of future needs in section 5.8.

Although the study showed that almost no explicit, deliberate, concious and analytical optimization had been obtained within the scope of the study, it was certainly evident that optimization was an inherent part of radiation protection work both at the plants and in design in the Nordic countries. This apparent contradiction was a consequence of how optimization was defined. Optimization as expressed by the actual situation corresponds to the traditional, intuitive, non-structured and holistic comprehension of what optimization is and how it works. This picture is still completely valid, and has proved to be a good one, judging from the results. This is, after all, what optimization really is about. New elements have, however, lately been included in optimization (see chapter 2).

5.6 Implementing a working optimization culture

One of the implicit goals of the study was to try to find common principles and rules regarding optimization as it is carried out at different Swedish and Finnish plants. The study clearly showed that optimization is primarily a state of mind in the sense that the radiation protection officer should continuously ask himself "Have I done all that I reasonably can to keep doses low in this particular case?". This is well in line with the opinions of optimization experts and also what ALARA historically was meant to be. Several issues of importance can be pointed out. The main new element is the strive to more quantitative considerations in designs and decisions. This means that two different decision- makers should arrive at about the same optimal solution to the same problem. This is, of course, at present quite possible and even probable in some cases, but the possibility of arriving at vastly different solutions should be excluded. In this way there is a shift towards objectivity and rationality. There is also a need for greater awareness in optimization, a real, clear and, at the same time, simple and deep practical understanding of what optimization means and of the possibilites offered by optimization for achieving good results.

In connection with reconstructions and additional constructions considered here, the role of the quantitative meaning of optimization is emphasised, as decisions may contain matters of greater significance than in everyday routine matters.

This is relevant to both design-related and operation-related reconstructions. ICRP has made a distinction between the two, considering two different ALARA-procedure schemes, although it seems that reconstruction usually contains a large element of design.

To achieve a situation where an optimal amount of optimization is done in most radiation protection decisions of the type considered in this work, some prerequisites are needed. The elements required are the following (the list should not be considered exclusive):

A. Responsibility of experts to sort out and clarify the elements of optimization into an easily understandable form.

This process has already come a long way, and it seems reasonable to assume that the meaning of optimization of radiation protection will be well understood in the near future, despite some unresolved issues. The international trend seems to be a certain convergence towards a common understanding.

B. Information to practitioners about what optimization is.

The problem of transferring the understanding of what optimization is and how it is applied at the levels of the expert and the practitioner, is a difficult one. Care should be taken to keep the presentation completely neutral, in order not to develop negative attitudes or too optimistic expectations based on preconceptions of the usefulness of optimization. This is probably best done by giving real examples of how optimization should be applied, gradually giving control to the practitioner.

C. Motivation: everyone must be aware of a common optimization philosophy valid within different groups of interest.

There are several motivational factors to be addressed. One is the need to show the obvious superiority of analytical, quantitative optimization in certain cases. Provided that the optimization work is fast and easy, it cannot be bad, by sheer definition, to optimise. Another one is to give good reasons for what will be gained by optimising. Shielding requirements, for instance, can be relaxed in certain cases based on optimization arguments.

D. Providing clear and simple methods.

In order to accomplish optimization in practice, clear and simple (which means fast) methods should be provided and taught. Such methods are being developed constantly, but they are still to some extent not applicabe as such by practitioners without an unreasonable amount of work.

E. Providing tools.

This issue is treated in section 5.7.

F. Great care has to be taken not to lose the good results achieved so far in radiation protection.

Extensive application of analytical, quantitative optimization must be seen as a supplement to the present practice, and the new results must be judged against what is presently considered to be good results.

G. Applying mild pressure to open up the field.

To achieve a breakthrough in optimization, some instruments must be available. At first it seems feasible to arrange continuous or periodical gathering of information in order to establish a representative data bank, based on an optimization thinking. This will guarantee that those willing to cooperate will not lose information, by providing an easy way of documenting the optimization aspects of the decisions. On the other hand, it will direct the decision-maker or the designer to think in optimization terms.

5.7 Need for advanced tools, international parallels

It is clear that there is much room for optimization at the existing plants in cases where the reasons for dose-reducing measures are not trivial, obvious or inexpensive. Optimization should not be an end in itself, but should reflect a strive to save as much dose as possible as cheaply as possible without any great effort. To achieve this, a systematic methodology and appropriate tools of some kind must be available. The tools should ideally help in keeping track of relevant factors in the decision process, assist in taking all factors properly into consideration, make the designer consider issues that otherwise might not get proper attention, give efficient assistance in quantitative analytical work and be able to select one or more preferable options. A computer-based design of this kind has been presented in /5-3/. A tool of the expert system type, mainly assisting in assuring the completeness of the set of factors considered in ALARA studies, is presented in /5-4/. Decision-aiding tools have been presented in /5-5/.

In order to give a practical illustration of the international state of optimization within the area of this study, and to put the project work into perspective, some examples will be given. These concern both specific technical optimization studies and the development of tools.

A. Brookhaven National Laboratory, ALARA Center

Work has been done at the Center to establish data bases regarding dose causing jobs, efforts to lower the doses and particular ALARA aspects. Many of the jobs and measures are of the type considered in this subproject. Some of the work has been reported in /5-6/.../5-9/.

B. Sargent & Lundy: ALARA review expert system

An ALARA design review assistant expert system, "ADRA", has been developed by Sargent & Lundy and Commonwealth Edison Company /5-4/. It is a rule-based system based on a commercial expert system shell, which provides facilities such as diffuse reasoning and explanation. It is designed for use by an engineer or a health physicist with a good understanding of the plant systems and areas and a working understanding of radiation sources and their effects.

C. Specific optimization case studies

Some detailed case studies exist, exposing the ALARA procedure. One example, quite relevant to this project, is /5-10/. It also shows the amount of work involved in a thorough optimization study.

5.8 Future needs and plans

Although one recommendation of the project report was to continue the collection of information with more developed questionnaires, this has not been done so far. The need remains. A new questionnaire version has been sketched, as presented in Figure 5.2. The main intention is to push the radiation protection decision-maker towards a greater awareness of the optimization. The purpose of this is twofold; to promote optimization and to catch such real-time information about optimization that is otherwise difficult to state or document.

Please fill in the essentials at the time of planning or performing the action. Write additional information or comments overleaf !

Return to : Rolf Holmberg, Imatran Voima Oy, POB. 112, SF-01601 VANTAA, Finland.

DOSE REDUCING ACTION :	REASON :	SAFETY OR OPERATIONAL ASPECTS :	LIMITS (design, regulatory....) :

Radiation conditions before:

Main & other radiation sources :

COST OF ACTION :

Max. dose rate :	Average dose rate :	×	stay time × frequency × no. of persons	=	DOSE BEFORE :

Radiation conditions after :

Reduced dose rate calculation :

Max. dose rate :	Average dose rate :	×	stay time × frequency × no. of persons	=	DOSE AFTER :

Cost to save dose :

Dose cost model :

$$\text{Cost of dose unit :} \times \left(\text{Dose saved :} - \text{Dose from action :} = \text{NET SAVED DOSE:} \right) = \text{Value of dose saved :}$$

Date :	Unit :	Situation (power, outage) :	Building/ room / system / comp:	Additional documentation :	Contact person:

There is also a need to clarify the treatment of accident and other probabilistic radiation protection optimization. This was also recommended in the study, although the treatment of such issues was categorically excluded. There has been some development in this field, as expressed e.g. in the report /5-11/.

5.9 Conclusions and recommendations

The project study serves as a foundation for further work, as it gives the impression that the radiation protection optimization work, which certainly has been done at Nordic plants, is to a great extent, of a qualitative and intuitive nature. This is partly because the concept of optimization is still in a state of development.

Optimization at its best is a very useful tool for radiation protection decisions, having significant potential in radiation protection problems of the type considered in this project. Two issues favoring optimization are emphasized, the increasing requirements for efficiency and the greater need to preserve the image of clean, safe plants. An absolute prerequisite for a breakthrough in quantitative optimization is that studies and decision-making must not take much time or be difficult, however advanced.

Tools for everyday routine use by practitioner, based primarily on modern computer-related techniques, remain to be completed.

It seems, on the basis of the study, that there is potential for achieving improvements in some areas of radiation protection by introducing optimization as a tool. Such areas are:

- design work would ultimately be less demanding, because of the need to establish information bases, methods and generic results
- rationality in decision making
- more balanced designs
- cost savings
- computer-based optimization tools would include modules, which would improve estimation and calculation in other radiation protection work.

It is further recommended that a program should be established for studying how analytical, formal optimization could be applied to restricted radiation protection problems related to extensions and reconstructions in design and operation. The application should reflect the work on optimization that has been done internationally.

5.10 References

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6 CONCLUSIONS

The objective of the Nordic research project was to work out how optimization could be applied to the radiation protection of workers at the Nordic nuclear power plants.

The state-of-the-art in optimization of radiation protection was reviewed. A lot of information was available during the time interval the project was run. In fact, main information gathered reflects the progress in optimization of radiation protection which in 1989 has been reported by the International Commission on Radiological Protection.

The project gathered a lot of interesting information on the features of the practical radiation protection at the Nordic nuclear power plants.

The resulting view was that actions to reduce doses were mostly based on other needs than direct optimization considerations. Such needs were e.g. high local or general dose rates, operational, or safety-related reasons. Optimization had been like an intuitive process, based on the experience and skill of the radiation protection staff.

An important factor for controlling doses of workers seemed to be an active work management programme including work planning, preparation of working places and training of the personnel involved.

Actions taken were generally considered to be cost-effective. The study revealed that some of the actions involve rather high costs, reflecting the relatively high weight of factors other than objective health detriment.

General guidance on the selection of formal optimization methods can be given. However, there are many things to be considered which have a bearing on this. There is a need for rules of thumb or some standardization to be used in decision-making on the basic level of operational radiation protection work at a nuclear power plant as well as a need for data bases and computerized decision support systems in more complex cases.

The level of ambition in radiation protection of workers at Nordic nuclear power plants is high. Consequently regarding many practical situations and routines, it seems that the entire and actual costs of protection and protective options are not recognized or assessed in detail.

For instance, the following could be done in order to further study the basis for optimization of radiation protection at the Nordic nuclear power plants:

- thorough evaluation of occupational radiation doses and radiation protection work carried out,
- survey of the relations between radiation protection and operation,
- more comprehensive monetary valuation of radiation protection and occupational radiation doses.

The validation of a model for estimating the cost of radiation workers' doses, in line with that developed by the NRPB in England, could possibly be studied as a special issue.

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