RADIOLOGICAL DOSE ASSESSMENT FOR THE LEAKING COIL REPLACEMENT ON THE NECSA RADIOACTIVE EFFLUENT EVAPORATOR FACILITY

Charles Kros

A Research Report
submitted to the Faculty of Science,
University of the Witwatersrand, Johannesburg,
in partial fulfilment of the requirements for the degree of Master of Science.

School of Physics

December, 2013

DECLARATION

I declare that this research report is my own, unaided work. It is being submitted for the Degree of Master of Science in the University of the Witwatersrand, Johannesburg. It has not been submitted before for any degree or examination in any other university.

_____

Charles Gustav Kros

6 December 2013

ABSTRACT

In this study a dose assessment is used to demonstrate conformance to national and international dose limits for workers and meets the Necsa ALARA goals for a radiological repair task. The dose assessment methodology is based on international standards, principles and criteria and involves the process of determining radiological dose, through the use of exposure scenarios, bioassay results, monitoring data, source term information, and pathway analysis.

The radiological task is the replacement of the leaking steam coil on the radioactive effluent evaporator facility at Necsa. The effluent treatment facility, its operation, the origin of the radioactive effluent and hazards associated with the leaking coil are discussed.

The dose assessment is supported by measurement of actual radiological conditions in the area where the task will be performed using suitable and calibrated instrumentation. The assumptions were limited to the physical phenomena associated with the behaviour of materials and available from national and international studies. The importance of proper planning of all the tasks associated with the replacement task as well as sources of inaccuracy and uncertainty associated with the calculated doses are discussed.

The results of the assessment are evaluated in terms of ALARA, namely the safety fundamental principles of justification, optimisation and limitation of facilities and activities. Other dose reduction options, such as personal protective clothing and equipment, were considered to show that the doses conform to the ALARA objectives of Necsa and other operation optimisation measures.

DEDICATION

Dedicated to my loving wife, Liezl, and sons, Eric and Etienne.

ACKNOWLEDGEMENTS

I want to specially thank my wife, Liezl, and my two sons, Eric and Etienne, for the many years of patience with me and this study. I trust our family will have grown closer during this period and share in this achievement.

I want to thank the University of Witwatersrand, in particular Prof. John Carter and Mr James Larkin for their guidance, lecturing and support during this period.

I want to thank Necsa for providing me with the opportunity to perform this study through allowance of time and provision of funding. In particular, I want to thank my internal supervisor and mentor, Mr Abrie Visagie, for whom I have the utmost respect. Abrie has demonstrated, through the many years that I have known him, the ability to disseminate his immense knowledge of radiation protection and nuclear licensing, which provided an ideal platform for my own development in radiation protection.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS

ALARA	As Low As Reasonable Achievable
AMAD	Activity Median Aerodynamic Diameter
BSS	Basic Safety Standards [IAE96]
EPR	European Pressurised Water Reactor
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IEC	International Electrotechnical Commission
ISO	International Standards Organisation
keV	Kilo electron volt
LEMS	Liquid Effluent Management Services department at Necsa
man.Sv	man-sievert (unit for collective dose)
MW	megawatt
µSv	microsievert
µSv/h	microsievert per hour
mSv	millisievert
mSv/h	millisievert per hour
Necsa	The South African Nuclear Energy Corporation Ltd
NNR	National Nuclear Regulator (of South Africa)
PWR	Pressurised Water Reactor
RP	Radiation Protection
RSA	Republic of South Africa
SAFARI-1	South African Fundamental Atomic Research Installation,

	designation as number 1 being the first one in South Africa
Sv	sievert
Sv/h	sievert per hour
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
VRF	Volume Reduction Factor

GLOSSARY

Activity median aerodynamic diameter	The value of aerodynamic diameter of particles such that 50% of the airborne activity in a specified aerosol is associated with particles smaller than the AMAD, and 50% of the activity is associated with particles larger than the AMAD. The AMAD is used for particle sizes for which deposition depends principally on inertial impaction and sedimentation (i.e. typically those greater than about 0.5 μm) [IAE07].
Collective dose	This is the sum of all of the individual doses to members of the population.
Deterministic effects	Effect caused by high doses, often acute, which appear if the dose exceeds a threshold value, for example harmful tissue reactions.
Geotropism	Geotropism is an analogue meter's response due to orientation in a gravitational field whether radiation is present or not [ROL06].
Immobilisation	The conversion of waste into a waste form by solidification, embedding or encapsulation [IAE07]. Immobilisation reduces the potential for migration or dispersion of radionuclides during handling, transport, storage and/or disposal. The most extensively used methods are cementation and bituminisation [IAE83].
Loose (removable) surface contamination	Radioactive material that can be removed from surfaces by non-destructive means, including casual contact, wiping, or washing. This does not include radioactive material that is fixed and that requires physical means to remove the radioactive material.
Mock-up	A representation of the actual scenario where work has to be

	performed with no radioactive material present. Typically used for training of workers to perform a task.
Resuspension factor	The quantitative relationship between the concentration of loose surface contamination and consequent atmospheric concentration above the contaminated surface [CEM09]
Stochastic effects	Effects that may be caused by high or low doses which may be observed as a statistically detectable increase in the incidences of these effects occurring long after exposure, for example cancer or heritable effects.

CHAPTER 1

INTRODUCTION

"*Occupational exposure to ionising radiation can occur in a range of industries, medical institutions, educational and research establishments and nuclear fuel cycle facilities. Adequate radiation protection of workers is essential for the safe and acceptable use of radiation, radioactive materials and nuclear energy."* Taken from [IAE99a]

1.1 Occupational exposure of individuals

The statement above, taken from [IAE99a], required '*adequate protection'* for workers against the risk of ionising radiation. Before implementing such protection measures, an assessment of the radiological hazards need to be performed, namely a dose assessment.

Dose assessment can be defined as the process of determining radiological dose, through the use of exposure scenarios, bioassay results, monitoring data, source term information, and pathway analysis. These dose assessments are performed regularly as a means of evaluating the potential, planned or subsequent dose(s) received or to be received by personnel and/or the public from normal and accident conditions and is a condition imposed on operators by the regulatory authorities.

Occupational exposure is defined by the Basic Safety Standards [IAE96] as "*all exposures to workers received or committed during the course of their work, with the exception of those exposures which are excluded or practices which are exempted from regulations*".

1.2 Dose assessment and ALARA

"*The IAEA was created in 1957 in response to the deep fears and expectations resulting from the discovery of nuclear energy. Its fortunes are uniquely geared to*

this controversial technology that can be used either as a weapon or as a practical and useful tool." [www.iaea.org]

The International Atomic Energy Agency's (IAEA) statute authorises it "to establish safety standards to protect health and minimise danger to life and property" [IAE06]. Member states of the IAEA, of which South Africa has been a member state since its creation in 1957, can then apply these safety standards by means of regulatory provisions.

The following sections will elaborate on the fundamentals of radiological safety, the international system employed through to national legislations and how it is implemented at the South African Nuclear Energy Corporation Ltd (Necsa).

1.2.1 Fundamental safety principles

The IAEA has approved the publication of safety standards in the Safety Fundamentals categories on "the safety of nuclear installations", "safety of radioactive waste management" and "radiation protection and the safety of radiation sources", respectively, between 1993 and 1995.

In 1995 the IAEA Board initiated the revision of these safety standards with the aim of combining them in a unified set of principles to present a common safety philosophy. In 2006, the IAEA Fundamental Safety Principles No. SF-1 [IAE06] was approved, for promulgation by the IAEA's Board of Governors as the primary publication in the IAEA Safety Standard Series.

The Safety Standard Series comprises the Safety Fundamentals, Safety Requirements and Safety Guides in a tiered structure with a scientific underpinning to support decisions concerning safety of life and property.

The Fundamental Safety Principles [IAE06] states the fundamental safety objective and ten safety principles. The former applies to all circumstances that give rise to any form of radiation risks. The latter applies throughout the lifetime of all radiation facilities and activities. Facilities and activities include the safety of nuclear facilities and any places where nuclear material is produced, processed,

used, handled, stored or disposed of in radiation risk scenarios where radioactive material is produced, used, imported, exported, transported and other activities such as decommissioning, waste management and remediation.

Fundamental safety objective

The fundamental safety objective of the IAEA reflected in this publication [IAE06] is to protect people and the environment from the harmful effects of ionising radiation *"without unduly limiting the operation of facilities or the conduct of activities that give rise to radiation risks"*. Ten safety principles have been formulated to achieve this objective. The ten safety principles are:

Principle 1: Responsibility for safety

The holder of an authorisation, known as the licensee, has the prime responsibility for safety. "Safety" refers to the protection of people and the environment according to the fundamental safety objective and includes safety under normal operational conditions and accident conditions.

Principle 2: Role of the government

An effective legal and governmental framework for radiation safety must be established and sustained. This should include an independent regulatory body. This requires government to adopt within its legal framework provisions for legislation, regulations, and other standards and measures to fulfil its national and international obligations and to establish an independent regulatory body.

Principle 3: Leadership and management for safety

Effective leadership and management for safety matters must be established and sustained. This applies to all organisations concerned with, and facilities and activities that give rise to, radiation risks. This has to be demonstrated at the highest levels in the organisations and achieved by an effective management system. The management system also has to ensure the promotion of a safety culture that governs the attitudes and behaviour in relation to safety of all individuals concerned.

Principle 4: Justification of facilities and activities

Justification requires that the benefits that nuclear installations or other facilities and activities yield must outweigh the radiation risks to which they give rise. An example of this is the decision to embark on a nuclear power programme. Medical radiation exposure of patients is a special case in that the benefit is primarily to the patient.

Principle 5: Optimisation of protection

Optimisation is achieved when the highest level of safety can reasonably be achieved without unduly limiting utilisation. To determine whether radiation risks are As Low As Reasonably Achievable (ALARA), all such risks, whether arising from normal operations or from abnormal or accident conditions, must be assessed using a graded approach *a priori* and periodically reassessed, during the lifetime of facilities and activities.

Principle 6: Limitation of risks to individuals

Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk or harm. This is achieved by the application of dose and risk limits, supplemented by the application of the principles of Justification (Principle 4) and Optimisation (Principle 5).

Together these three principles form the basis for the system of radiation protection which will be discussed in Section 1.2.2.

Principle 7: Protection of present and future generations

People and the environment, present and future, must be protected against radiation risks. The potential consequences, for the present and for the future, of current activities have to be considered in judging how adequate the measures to control radiation risks to people and the environment are. This applies also to radioactive waste management to avoid an undue burden on future generations.

Principle 8: Prevention of accidents

The most harmful consequences arising from facilities and activities have come from the loss of control over nuclear reactor cores, nuclear chain reactions,

radioactive source or other source of radiation. Consequently, measures have to be taken to prevent the occurrence of accidents or abnormal conditions and to ensure that the likelihood of an accident having harmful consequences is extremely low.

The primary means of preventing and mitigating the consequences of accidents is through a concept called "defence in depth". Defence in depth is the principle of implementing a combination of a number of consecutive and independent levels of protection which would fail independently before any harmful effects could be caused to people or the environment. If any single level of protection were to fail, the subsequent level of protection or barrier would be available. When properly implemented, defence in depth ensures that no single failure, be it technical, human or organisational, could lead to consequences with harmful effects. It will also provide the assurance that any number of combinations of failures are of low probability.

Principle 9: Emergency preparedness and response

The licensee, employer, regulatory body and government have to establish arrangements for emergency preparedness and response for nuclear or radiation emergencies at the scene, and all levels including the international level. Consideration has to be given to all reasonably foreseeable events when developing emergency response arrangements.

Principle 10: Protective actions to reduce existing or unregulated radiation risks

Radiation risks may arise in situations other than in facilities and activities that are in compliance with regulatory control, such as situations or activities that were never subject to regulatory control in the past. In such situations, if the radiation risks are relatively high, consideration has to be given to whether protective actions can reasonably be taken to reduce radiation exposures and to remediate adverse conditions. The protective actions will have some foreseeable economic, social and, possibly, environmental costs and may entail some radiation risks (e.g. to workers carrying out these protective actions) and must be considered carefully and be optimised and justified.

1.2.2 System of radiological protection

Radiological protection deals with two types of harmful effects [ICR07]:

- Deterministic effects (e.g. harmful tissue reactions) are caused by high doses, often acute, which appear if the dose exceeds a threshold value and
- Stochastic effects (e.g. cancer or heritable effects) which may be caused by high or low doses and may be observed as a statistically detectable increase in the incidences of these effects occurring long after exposure.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNCSEAR) compiles assesses and disseminates information on the health effects of radiation and on levels of exposure to radiation from different scientific studies and reports. The International Commission on Radiological Protection (ICRP) uses UNSCEAR reports and other international reports to publish its recommendations on radiation protection.

In 2007, ICRP Publication 103 entitled "*The 2007 Recommendations of the International Commission on Radiation Protection*" [ICR07] was released, formally replacing the 1990 Recommendation version of the same report. Commonly known as the "System of Radiological Protection", this system aims primarily to protect human health. This is reflected in its health objectives: *"to manage and control exposures to ionising radiation so that deterministic effects are prevented, and the risks of stochastic effects are reduced to the extent reasonably achievable"* [ICR07]. In this context, this research report provides for several changes to tissue weighting factors and updates to the radiation detriment, but importantly maintain the three fundamental principles of radiation protection namely justification, optimisation and the application of dose limits.

These three fundamental principles of radiation protection and how they should be applied in planned occupational exposure situations (see Section 1.2.3) are discussed further.

Justification

The principle of justification: *Any decision that alters the radiation exposure situation should do more good than harm* [ICR07].

This is in line with fundamental safety principle 4 from Section 1.2.1 and in simple terms means that within the decision to increase or decrease levels of radiation exposure or a risk of potential exposure, the expected change in radiation detriment as well as other risks (e.g. costs, societal benefits) should be included. This is also referred to as the net benefit, and is implied to yield a positive net benefit.

Optimisation

The principle of optimisation: *The likelihood of incurring exposures, the number of people exposed, and the magnitude of their individual doses should all be kept as low as reasonably achievable, taking into account economic and societal factors* [ICR07].

Optimisation is not minimisation of dose, but the result of a forward-looking iterative evaluation aimed at preventing or reducing exposures under prevailing circumstances that involves:

- evaluation of the exposure situation,
- selection of an appropriate value for the constraint or reference level,
- identification of the possible protection options,
- selection of the best option under the prevailing circumstances and
- implementation of the selected option.

Dose Limitation

The principle of dose limitation: *The total dose to any individual from regulated sources in planned exposure situations other than medical exposure of patients should not exceed the appropriate limits recommended by the Commissi*on [ICR07].

These dose limits are determined by the regulatory body, taking account of international recommendations, and apply to workers and members of the public in planned exposure situations. In the Republic of South Africa (RSA) the National Nuclear Regulator (NNR) has set and published these dose limits in the Government Gazette [NAT99a].

1.2.3 Exposure situations

The ICRP in its Recommendations evolved from the previous process-based protection approach using *practices* which add doses, and *interventions* that reduce doses, by moving to an approach based on the exposure situation [ICR07]. They now use a situation-based approach to characterise the possible situation where radiation exposure may occur as *planned, emergency*, and *existing exposure situations*, and apply the fundamental principles of justification and optimisation of protection to all of these situations.

The three situations are:

- *planned exposure situations*, which are situations involving the planned introduction and operation of sources, which includes situations that were previously categorised as *practices*,
- *emergency exposure situations*, which are unexpected situations such as those that may occur during the operation of a planned situation, or from a malicious act, requiring urgent attention and
- *existing exposure situations*, which are exposure situations that already exist when a decision on control has to be taken, such as those caused by natural background radiation.

1.2.4 Dose limits

In its new Recommendation [ICR07] the ICRP furthermore reinforce the principle of optimisation of protection, which should be applicable in a similar way to all exposure situations, subject to the following restrictions on individual doses and risks:

- *dose* and *risk constraints* for planned exposure situations and
- *reference levels* for emergency and existing exposure situations.

The Recommendations also include an approach for developing a framework to demonstrate radiological protection of the environment.

Most noteworthy is the confirmation in the Recommendations that the existing dose limits remains unchanged as it is deemed to provide an appropriate level of protection. For occupational exposure in planned exposure situations, the limit is 20 mSv per year, averaged over defined 5 year periods, with the further provision that the effective dose should not exceed 50 mSv in any single year.

1.2.5 A national perspective

The National Nuclear Regulator (NNR) is the regulatory body constituted to regulate nuclear activities in the RSA, as established by the NNR Act [NAT99b] in 1999. The NNR published regulations on safety standards and regulatory practices [NAT99a] in 2003 which were based on the then existing IAEA Basic Safety Standards [IAE96] of 1996. The new recommendations of the ICRP [ICR07] in 2007 led to the publication of a new IAEA Safety Standard [IAE11] in 2011. Several changes from the previous standard have been covered in Sections 1.2.2, 1.2.3 and 1.2.4.

The national regulations might appear outdated in terms of international standards but remain in place until such time as a revised version is issued. Most relevant is the fact that the dose limits remain unchanged in the recent Recommendations and standards, as confirmed in Section 1.2.4.

The national regulations [NAT99a] in its Section 3 on *Principal radiation protection and nuclear safety requirements* does cover the ten fundamental safety principles from Section 1.2.1 and the radiation protection system from Section 1.2.2. For this work Section 3.3 of the regulations is most relevant since it prescribes *a priori* safety assessment: "*Measures to control the risk of nuclear damage to individuals must be determined on the basis of a prior safety assessment which is suitable and sufficient to identify all significant radiation*

hazards and to evaluate the nature and expected magnitude of the associated risks, with due regards for the dose and risk limits in Annexures 2 and 3." [NAT99a].

1.2.6 As Low As Reasonably Achievable (ALARA)

ALARA is defined in the IAEA Safety Glossary [IAE07] and the new Basic Safety Standards [IAE11] under *optimisation of protection and safety* as:

"The process of determining what level of protection and safety makes exposures, and the probability and magnitude of potential exposures, 'as low as reasonably achievable, economic and social factors being taken into account' (ALARA), as required by the International Commission on Radiological Protection System of Radiological Protection."

The concept of ALARA stems from the second principle of radiation protection (see Section 1.2.2) namely Optimisation. Optimisation is described by Cember [CEM09] as an operating philosophy which urges actual operational dose limits for any radiological activity to be more restrictive than the maximum recommended dose limit. This will require processes, equipment (such as shielding, ventilation, etc.), and other operational factors to be designed so that workers do not exceed the operational dose limit based on a cost-benefit analysis to derive the optimal operational solution. For this reason the ICRP [ICR07] recommended the use of dose constraint which is lower than the dose limits.

Demonstration of ALARA is achieved by application of a process that involves a series of steps to ensure that doses are carefully managed throughout the work. This does not necessarily mean that dose has been minimised, as minimising dose in isolation could have an unreasonable impact on other key factors such as project timescales or cost. The requirement is, therefore to minimise the overall project risk (as far as reasonably achievable) by optimising the key elements of dose, cost, project timescale, number of workers, environmental discharges and conventional safety.

The licensee in terms of the regulations [NAT99a] must therefore be able to demonstrate that it has:

- assessed the risk,
- estimated the detriment by means of assessment,
- performed an evaluation or cost-benefit analysis and
- taken action to avert the risk where appropriate.

The key point is to ensure that dose or risk assessment and subsequent justification is robust enough to stand up to scrutiny and cross examination in a Court of Law, defending recommendations and advice given.

Although the ALARA concept is sometimes misused, it usually consists of the planned and systematic application of common sense. Reasonable measures to reduce worker and public dose usually lead to reasonable measures to assure an optimised process. Two specific examples are:

- to prepare for maintenance work on high dose-rate systems, it is a common practice to *train on a mock-up*. This assures that the workers are near the top of the learning curve when they have to do it for real. It is also good practice for other reasons, since mistakes can lead to equipment damage or worse, increased doses to personnel and
- in addition to reducing collective dose, the ALARA concept requires that the number of individuals that are exposed be optimised. This can be achieved by *distribution of work* amongst several workers but also by ensuring that no individual is exposed unnecessarily. Experience has shown that management will typically come to depend on a few individuals to work on certain jobs. This is a poor practice, since additional workers may identify ways of improving the work scope, and an organisation often gets into troubled situations when the key employee is unavailable.

ALARA should thus not be considered as a waste of resources with no benefits. Rather, it is just one aspect of a well run and optimised facility. ALARA will be discussed in more detail in Chapter 2.

1.2.7 Dose assessment

The IAEA in its Safety Glossary [IAE07] describes assessment as the process, and the result, of analysing systematically and evaluating the hazards associated with sources and practices, and associated protection and safety measures. Dose assessment is subsequently the assessment of the dose(s) to an individual or group of people.

Dose assessments are often referred to as the ALARA pre- and post-job studies. The pre-job ALARA study is an evaluation of the dose(s) and the means employed to reduce this dose(s). This task is complex since the dose is normally determined by many factors, e.g. source term, geometry, composition, conventional hazards, work conditions, etc. Such situation is made even more complex when a changing and/or unknown environment is encountered, as is the case during decommissioning or non-routine maintenance activities. In the latter cases, several assumptions will need to be made to simulate the expected environment using prior experience (personnel, literature, etc) or experimental results. Later in this research report significant effort will be devoted to the assumptions made and the rationale used in this context.

The post-task study will be the culmination of the actual measured conditions and activities into dose(s) received, as well as a comparison against the pre-job study.

By way of example, in the design and development of the European Pressurised Water Reactor (EPR) a large effort was made to improve the plant design with respect to radiation protection using the experience gained during the design of former generations of Pressurised Water Reactor (PWR) in France and Germany, and their current operation. Keeping the radiation exposure of personnel to an acceptable level is one of the main objectives of the EPR design. Detailed dose assessment and improvements in the design have led to decrease the target for the

collective exposure from 0.75 man.Sv per year to 0.5 man.Sv per year on average [BAU06].

1.2.8 Demonstrating ALARA at Necsa

Necsa, as a licensee of the NNR to operate various facilities on the Pelindaba site in the Northwest Province of South Africa, is required to demonstrate compliance to the regulations [NAT99a]. Internal procedures are compiled to govern the operations on the Necsa site.

The limitation of doses to workers and members of the public from Necsa operations [NEC01] are aligned with the regulations [NAT99a] and international standards [ICR07]. As required by the regulations [NAT99a], Necsa has set its own dose constraint as an ALARA objective to ensure that exposures are maintained as ALARA. The Necsa ALARA objective requires that the average annual effective dose to the occupationally exposed workforce does not exceed 4 mSv for all routine operations.

For any *ad-hoc* radiological task (i.e. task with radiological exposure potential for which no standard work procedure exist, normally non-routine tasks) to be performed, Necsa requires a radiological protection work permit to be completed [NEC02]. The completion of the work permit entails gathering of information to perform a dose assessment of the task and provide recommendations on radiation protection measures to ensure doses will conform to the ALARA programme [NEC03]. Work permits are authorised by senior RP personnel. Furthermore, if the individual dose is estimated to exceed 1 mSv, then a formal ALARA pre-planning and review is required. An operational constraint, known as an *ALARA goal,* is also defined for such a task. This ALARA goal is a predetermined dose and is authorised by management.

An ALARA programme [NEC03] is also in place which consists of the following elements:

- requirements for ALARA training and orientation,

- planning and control over radiological tasks to track and review exposures on a regular basis,
- design criteria for new facilities or modifications to existing facilities to meet a set dose constraint,
- continuous dose management and optimisation by means of individual ALARA goals and regular ALARA reviews and
- optimisation of radioactive waste mainly by waste prevention and limitation of waste quantities generated.

Individual workers at Necsa will thus be subjected to continuous monitoring against ALARA goals based on existing operations and experience; both of which will undergo regular review to demonstrate compliance to dose constraints, optimisation and continual improvement.

This study involves a dose assessment for a task which has been identified to have potentially high doses for workers, in excess of 1 mSv. Therefore, the Necsa approach would require formal ALARA pre-planning which is achieved by a dose assessment which must demonstrate:

- conformability to the ALARA goal and
- exposure of workers has been optimised.

1.3 Statement of the problem

The effluent treatment plant at Necsa was constructed in the 1960's [NEW79]. At the time the evaporator was regarded as '*one of the most modern in the world*' [NEW79]. For the past 50 years, the evaporation facility has been operated virtually unaltered, which speaks a lot towards the quality of workmanship and materials employed in its construction, as well as regular maintenance performed.

However, in early 2008, failure of the steam coil in the secondary isolated circuit resulted in the concentrate leaking from the primary to the secondary circuit. The secondary circuit, which is not expected to contain any radioactive material, is

released to the industrial effluent system which acts as a barrier system before release to the environment, through regulated practices. The radioactive leakage was subsequently detected in the industrial effluent before release to the environment. More descriptive information on the system will be supplied in Chapter 2.

Fortunately, the damage to the coil initially did not necessitate halting operations, but adversely affected the efficiency of the process (as a result of steam leaking into the evaporator primary circuit).

Replacement of the steam coil will require a substantial amount of work to be performed in the evaporator room, namely Cell 4, which in itself is regarded as a confined space. The entire area is radioactively contaminated with elevated ambient and localised high external radiation levels. In order to confirm that worker exposures have been optimised, justified and are within regulatory limits, an iterative process is proposed in this research report.

1.3.1 Objective

The objectives of this study is to perform a radiological dose assessment and evaluation of the occupational exposure to workers involved in the replacement of a steam coil inside an evaporator used for the treatment of radioactive effluent. This includes suggestions for improvement in the light of ALARA (pre-task) and recommendations for radiological protection during performance of the task.

1.3.2 Methodology

The methodology is summarised in the form of a flow diagram in Figure 1.1.

The items identified in the flow chart appear in a sequential order, and are elaborated upon below:

- Defining ALARA goals: the ALARA goal for a task needs to be set as per Section 1.2.8.

- Gathering radiological information: in order to perform a dose assessment, a radiological surveillance is required, primarily to determine the extent of the radiological hazards as well as to perform a pre-task dose assessment for the surveillance team.

- A detailed work plan schedule is required to perform the dose assessment. The required information is a description of task to be performed (in chronological order), number of persons required to perform task, names of identified individuals and duration of task.

- Assumptions (with appropriate justifications) need to be formulated to simulate the exposure scenarios. Examples of these assumptions are:
  - air exchange rate,
  - breathing rate,
  - dose conversion factors and
  - airborne release fractions.

- The dose assessment is performed using internationally accepted calculation methodologies and formulae which will include the contributions from direct external radiation, internal radiation (ingestion, inhalation) and the total effective dose.

- Comparison with the ALARA goal: the calculated individual doses are compared against the ALARA goal. In this work, it is expected that the doses will exceed the ALARA goal. Based on this outcome, additional ALARA recommendations will be made as to the measures that should be used to decrease doses and the relative effect these will have, e.g. personal protection clothing and equipment, area classification, access and egress, contamination monitoring, personnel monitoring and review during the progress.

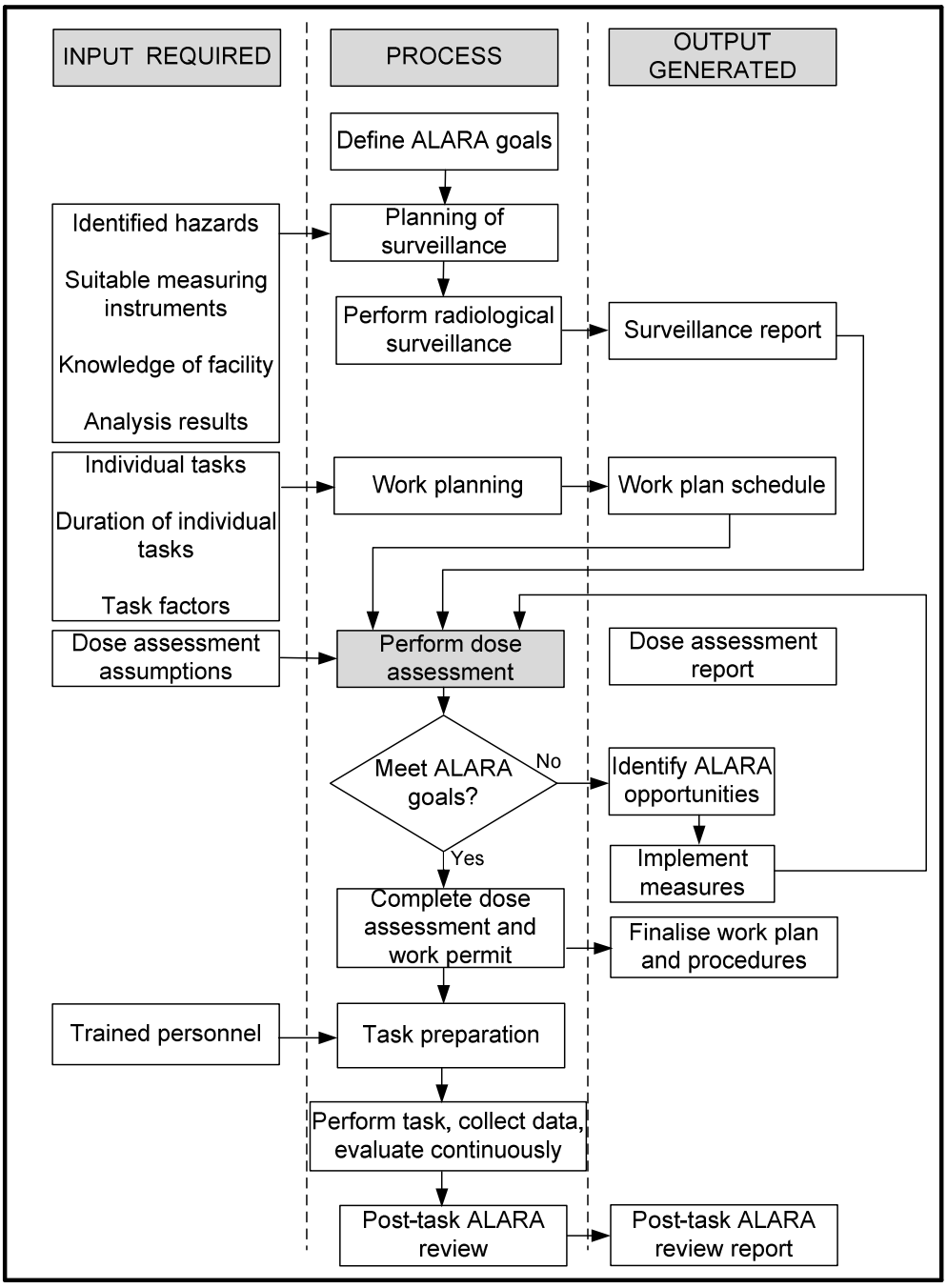

Figure 1.1: Methodology used in this research report.

- Complete work permit: upon completion of the dose assessment and compliance to the ALARA goal, the dose assessment is reviewed and authorised.

- Performance of the task can commence subject to completion of task preparations following the recommendations and provisions set, which will include strict monitoring and record keeping.

- Post-task review: this is required as an evaluation of the effectiveness of the measures implemented, and should include discussion on lessons learned and suggestions for improvement in future. This is outside the scope of this report as the author was not involved in this action.

1.4 Research Report structure and chapter outline

According to the methodology described above, the layout of the study is diagrammatically presented in Figure 1.2.

The chapter layout structure of the remaining part of the present study is as follows:

Chapter Two – Theory part consisting of:

- background on the origin and handling of liquid effluent at Necsa,
- more background on the requirements of the dose assessment to be performed,
- physics of interaction of radiation with matter in order to support the measured results, derive quantitative information for the dose assessment and identify radiological hazards,
- dose assessment methodology is expanded to cover exposure pathways and the assessment for these pathways and
- identify ALARA options based on information gather above.

Chapter Three – Consisting of:

- a dose assessment using the measured data, assumptions and formulae in Chapter 2,
- detail on the exposure scenarios,
- interpretation of the external, internal and total doses calculated and
- final radiological protection recommendations and demonstration of compliance to ALARA constraints.

Chapter Four – Conclusions and recommendations are presented.

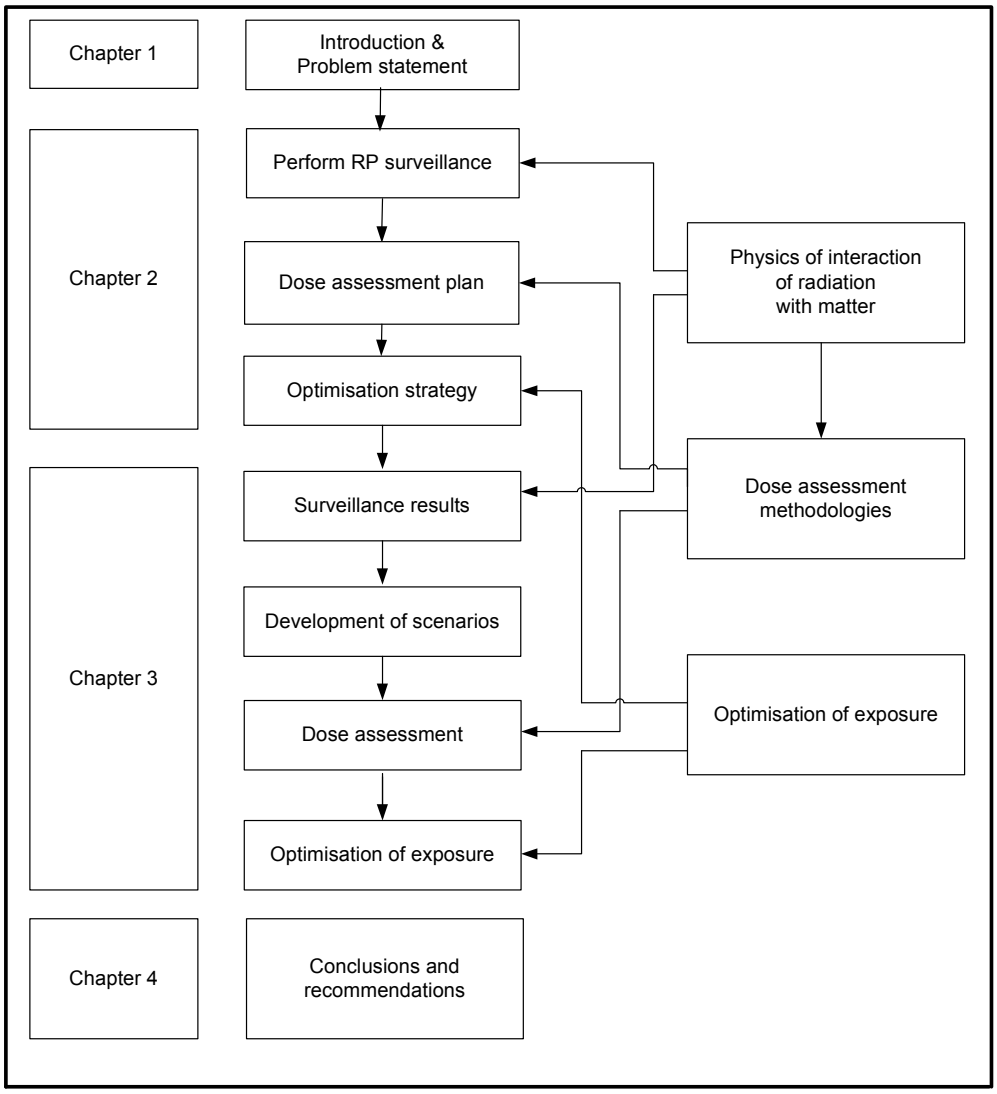

Figure 1.2 : Diagrammatic presentation of Chapter layouts used in this report.

CHAPTER 2

THEORETICAL CONSIDERATIONS

In this chapter details will be provided on the handling of liquid effluent at Necsa and the problem with the evaporator coil which has arisen. Theory will be provided on the physics of interaction of radiation with matter, dose assessment methodologies and optimisation of exposure which will support the implementation thereof in Chapter 3.

2.1 Treatment of liquid radioactive waste

The nuclear industry has been established for over six decades. Its activities give rise to both liquid and solid wastes from various processes. In volume, liquid wastes tend to be much greater than solid wastes [IAE84] which emphasises the need for treatment processes which will reduce the storage capacity.

Many treatment processes exist in industry and the selection of process depends on several factors [IAE84]:

- characteristics of the liquid wastes,
- requirements for discharge to the environment,
- available technologies and cost,
- conditioning of concentrates resulting from the treatment and
- storage and disposal of the conditioned concentrates.

The treatment processes can be categorised into 4 main categories, namely filtration to removed suspended matter, chemical precipitation, ion exchange and evaporation [IAE84]; however, a combination of these are normally encountered in industry. One of the most common of these is ion exchange, which is a well developed technique that has been employed for many years in both the nuclear and other industries [IAE02]. All of these processes result in some form of

concentrate of which the most common are sludges, spent ion exchange media and concentrates from evaporation.

The primary purpose of liquid waste treatment is to reduce the volume of the waste. The IAEA [IAE83] defines the volume reduction factor (VRF) as follows:

$$VRF = \frac{Volume\ of\ waste\ treated}{Volume\ of\ concentrate}. \quad (2.1)$$

Theses concentrates require immobilisation which is defined as the conversion of waste into a waste form by solidification, embedding or encapsulation [IAE07]. The primary purpose of immobilisation is to reduce the potential for migration, release or dispersal of radioactive material during handling, transport, storage and/or disposal. The most extensively used methods are cementation and bituminisation [IAE83]. The former consists of mixing the liquid waste with cement to form a solution within a container and allowing the mixture to set. The latter consists of mixing the liquid waste with bitumen at elevated temperatures which, after evaporation and cooling, results in a solidified mixture inside containers.

2.2 Liquid effluent at Necsa

Liquid effluent in the context of the Necsa operations refer to radioactively contaminated aqueous waste generated by the operations in nuclear facilities at Necsa, and accepted for processing by the Liquid Effluent Management Services (LEMS) department. Processing of liquid effluent involves different handling operations based on classification in terms of the activity concentration of the effluent and can included evaporation followed by solidification or even authorised discharge to the environment.

2.2.1 Classification of radioactive liquid effluent

Liquid effluent is transferred from the generating facility to LEMS mainly by pipeline and collected in dedicated receiving tanks. Effluent is classified [NEC04] in terms of its activity concentration into either:

- low activity effluent,
- medium activity effluent or
- industrial effluent.

Low activity effluent is effluent for which the gross alpha-decay activity concentration is between 10 and 100 $Bq\ell^{-1}$ or the gross beta-decay activity is between 40 and 4000 $Bq\ell^{-1}$. Medium activity effluent is effluent for which the gross alpha-decay activity concentration exceeds 100 $Bq\ell^{-1}$ or the gross beta-decay activity exceeds 4000 $Bq\ell^{-1}$.

The Industrial Effluent is the type of effluent that contains low levels of radioactivity (less than 10 $Bq\ell^{-1}$ of gross alpha-decay activity or less than 40 $Bq\ell^{-1}$ of gross beta-decay activity), which requires no additional treatment and is destined for authorised discharge into the nearby Crocodile River based on low environmental impact.

The classification of liquid effluent in terms of activity concentration is based on internationally accepted practice. It should be noted that the specific activity concentrations applied at Necsa are much lower than those applied in, for example, India [RAJ06] being due to Necsa operating a research reactor *versus* the power reactors in India which operate on much higher power levels and generate liquid effluent in larger quantities and of higher activity concentrations. However, the treatment processes applied are similar, as mentioned in Section 2.1.

2.2.2 Treatment of low activity effluent

The low activity effluent is treated through a chemical precipitation process to reduce the concentration of dissolved solids. Precipitate is flushed into radioactive sludge drying beds and is treated as solid radioactive waste after drying. The Low Activity Effluent is transferred to interim holding tanks, subjected to sampling, radioactive analysis and discharged into the Crocodile River upon conforming to authorised discharge criteria. The criteria are based on dose impact to the public as set by the NNR in the regulations [NAT99a].

2.2.3 Treatment of medium activity effluent

Medium activity effluent cannot be discharged to the environment since its activity is high enough to cause substantial health effects to the members of the public. This effluent subsequently needs to be subjected to more intensive treatment processes as discussed in Section 2.1.

The medium activity effluent treatment facility consists of receiving and storage tanks and an evaporator system. Here, the purpose of the evaporator is to reduce the volume of the contaminated effluent. The evaporation process results in the activity being concentrated at the bottom of the evaporator as a condensate and the cleaner effluent is captured at the top of the evaporator. Condensate from the evaporator is then removed from the evaporator to separate holding tanks where, after analysis against low activity effluent classification criteria in Section 2.2.1, treated as low activity effluent according to Section 2.2.2.

The concentrate from the evaporator is removed at regular intervals and immobilised as solid radioactive waste. More information will be provided in Section 2.2.5.

2.2.4 Origin of medium activity radioactive effluent at Necsa

There are several nuclear facilities on the Necsa site, most of which generate radioactive effluent in some form, be it from cleaning, laboratory or process operations.

The high activity concentration in medium activity effluent is an indication that this type of effluent is generated from specialised processes or processes where potential for high activity levels are expected. The two facilities responsible for the bulk of this effluent are the isotope production facility and the SAFARI-1 research reactor.

Isotope production facility

The isotope production facility at Necsa, operated by NTP Radioisotopes SOC Limited, is a modern hot cell complex where radioisotopes are manufactured and packaged. Radioisotopes are used in various chemical forms in a large number of medical applications, such as dynamic and static diagnostic studies which include imaging of the heart, brain, thyroid, liver, lungs, kidneys and bone.

The isotope production facility is by far Africa's largest producer of a range of medical isotopes that are used for diagnostic purposes and therapeutic treatment of cancer and many millions of people have benefited from these medical isotopes. The most important of these isotopes for Necsa, is the radioisotope Molybdenum-99 (^{99}Mo) which is used extensively as a raw material for ^{99m}Tc (the most important diagnostic nuclear medicine isotope).

The manufacturing process for these radionuclides involves several processing steps which generate waste in solid, liquid or gaseous form. The liquid waste is collected in waste storage tanks. Only when the tanks reach capacity and the radioactivity levels has decayed sufficiently, will this effluent be transferred by pipeline to LEMS for treatment.

SAFARI-1 research reactor

The SAFARI-1 research reactor at Necsa, constructed in the 1960s and commissioned in 1965 [NEW79], makes use of plate type fuel elements containing enriched uranium for operations. The enrichment grade of the uranium was initially high enriched uranium (enriched to > 20% in the ^{235}U isotope) and in 2009 conversion to low enriched uranium (enriched to < 20 % in the ^{235}U isotope) was completed.

SAFARI-1 is a 20 MW tank-in-pool type nuclear research reactor, owned and operated by Necsa. SAFARI-1 is an acronym for South Africa Fundamental Atomic Research Installation and is South Africa's only nuclear research reactor.

Contamination of the primary coolant water is caused by the slow diffusion of fission products from the fuel elements through defects in the fuel plates and by the production of activation products. The contamination is compensated for by removal due to radioactive decay (short lived radionuclides), deposition on the inner surfaces of primary circuit and by decontamination of primary coolant water. Decontamination of the primary coolant is required in order to minimise contamination of tools and equipment, as well as to reduce exposure of personnel.

The fission process in the reactor results in the generation of a suite of fission and activation products, as well as actinides. There are more than 40 different elements and more than 600 different isotopes formed in the fission process. These radioactive isotopes are produced in different quantities and possess different physical and chemical properties, as well as biological effects. As a result, fission products can be classified accordingly as noble gasses, halogens, metals and actinides.

The inventory of nuclides in a reactor is mainly determined by the power level of the reactor and the irradiation time. Short lived isotopes reach their equilibrium concentrations soon after reactor start up while the other isotopes continue to accumulate during irradiation. The main source of radiation in the SAFARI-1 reactor is due to the gamma radiation from these fission and activation products.

The fission inventory of reactors is available in the literature being based on reactor power level and is normally calculated by using a computer modelling codes (eg. ORIGEN [http://scale.ornl.gov]), as is the case for SAFARI-1. The following represent the most important radionuclides based on activity expected:

- activation products: ^{54}Mn, ^{58}Co, ^{60}Co
- noble gas fission products: ^{133}Xe
- fission products: ^{90}Sr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{134}Cs, ^{137}Cs.

Returning to the subject of decontamination of coolant water, some of the general methods available for the treatment and decontamination of primary coolant are chemical precipitation and ion exchange [IAE84] [RAJ06] [OHW67]. The choice

of a suitable technique is determined by the chemical composition of the waste solution and by economic factors. SAFARI-1 employs mixed-bed ion exchangers for this purpose. This is a proven technology with literature dating back to the 1950's [NAC56] [OHW67] quoting decontamination factors of greater than 10^3 for mixed- bed ion exchangers.

The potential usefulness of inorganic ion-exchangers has been proven in various areas of chemical processing before it was utilised in nuclear fuel cycle technology, especially in the separation and fixation of fission products and actinides and in the treatment of effluents from nuclear power plants [IAE84]. The process involves exchange of ionic species between the liquid and solid matrix containing ionisable polar groups. Inorganic ion-exchangers have received attention for these purposes because of their strong chemical affinity, high retention capacity for cation radionuclides and high resistance to radiation.

An interesting study [SIN97] has even shown this technology to be "the most effective" for the removal of iodine from liquid effluents and for spent fuel reprocessing [NAV89]. Studies were conducted in the 1960's to remove radioactive contamination from milk [EDM64] by means of an ion exchange process. Zeolite ion exchangers were used extensively in the cleanup of large volumes of contaminated water at Three Mile Island after the Unit 2 reactor incident [CAM83].

The technology might be considered old, but it is still in use internationally. In 2002, the IAEA published a technical report [IAE02] where it is claimed: "*With respect to economy and efficacy, ion exchange stands between the other two major liquid waste treatment processes of evaporation and chemical precipitation. While evaporation may yield higher decontamination factors, it is also more costly than ion exchange. The development of new ion exchangers is narrowing the gap in decontamination factors between evaporation and ion exchange."*

When the ion exchanger columns become fully loaded (saturated), they are regenerated by using strong acids or bases, yielding high concentrations of

radioactive liquid waste with a high salt content. The activity concentration of this liquid waste is typically higher than 1 $MBq\ell^{-1}$ which classifies it as medium activity effluent in terms of the criteria in Section 2.2.1. This medium activity effluent is collected in holding tanks at SAFARI-1 before transfer to LEMS (see Figure 2.1). SAFARI-1 is responsible for the bulk of the medium activity effluent received by LEMS.

2.2.5 Treatment of medium activity effluent at Necsa

In the 1960's, NECSA commissioned an evaporation facility for the treatment of medium activity effluent [NEW79] as a volume reduction technique through evaporation, as described in Section 2.1. The process of evaporation, where water is removed in the vapour phase of a process leaving behind non-volatile components such as salts and most radionuclides, is a proven method yielding good decontamination (typically 10^1-10^4) and good concentration factors. A schematic presentation of the treatment process is shown in Figure 2.1.

Figure 2.2 and Figure 2.3 show the lower and upper sections of the evaporator, respectively.

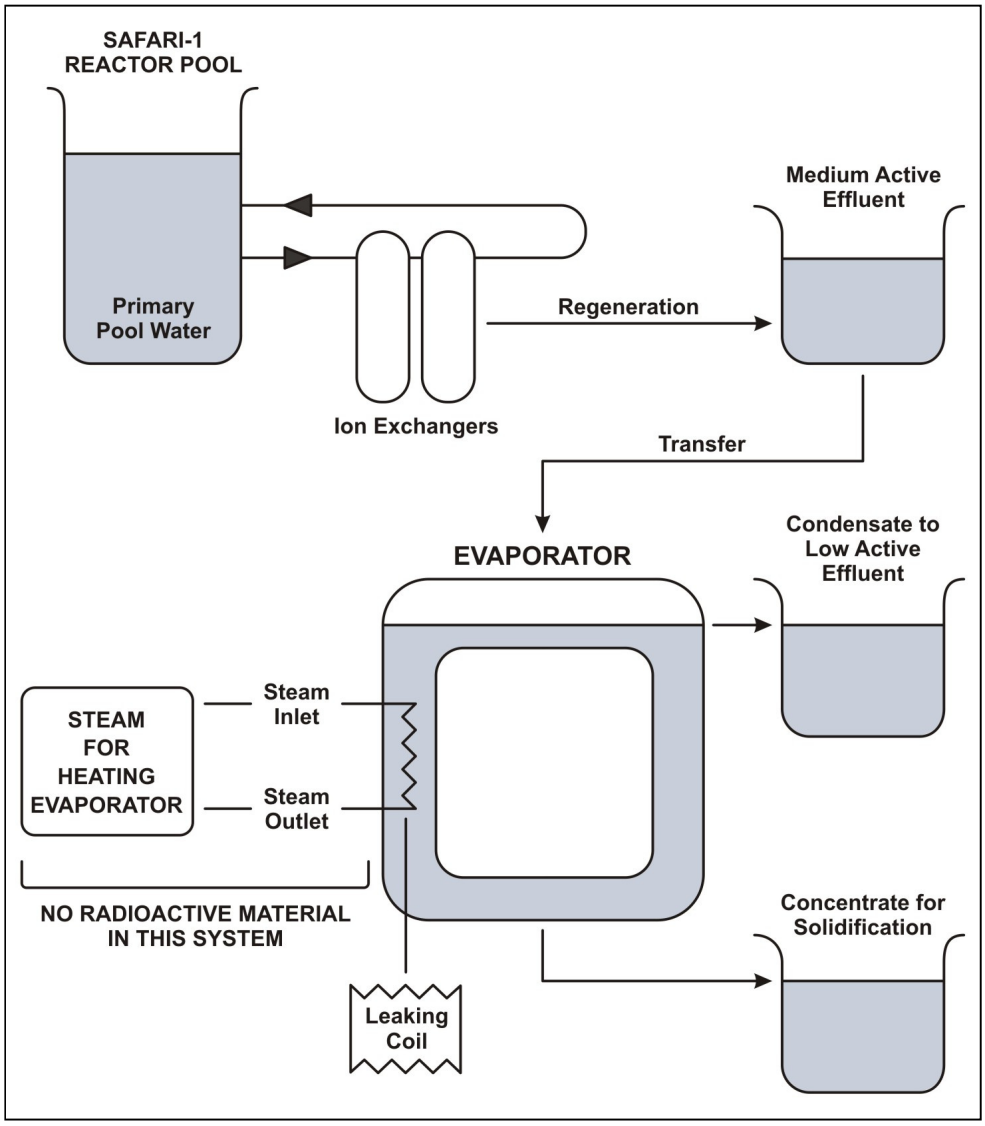

Figure 2.1: Schematic representation of medium activity effluent treatment process, showing the SAFARI-1 reactor pool, ion exchangers and evaporator.

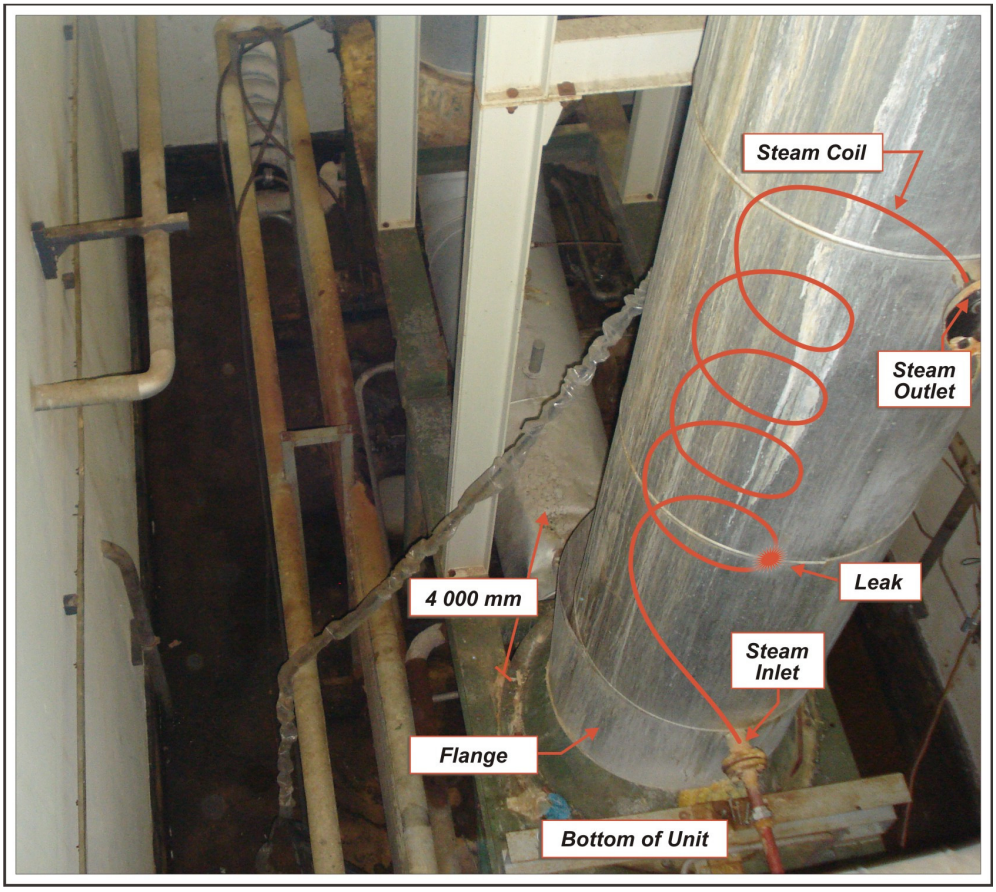

Figure 2.2 : Photograph of the lower section of evaporator and steam coil inside the evaporator taken by the author.

Figure 2.3 : Photograph of the upper section of the evaporator taken by the author.

2.3 Problem with the Necsa evaporator

The steam coil in the evaporator forms part of a secondary system where steam is used as the heating medium in the lower section, also known as the downcomer section, as illustrated in Figure 2.2. It was discovered in 2008 that the steam coil has a leak into the primary system (evaporator). The immediate consequence (steam leaking into primary) is loss of evaporator efficiency.

However, during the end of day cooling down phase, steam condenses and causes the medium activity effluent concentrate to leak into the secondary system. Subsequently, upon the beginning of day start-up, the blowing down of steam results in the leaked concentrate leaking into the industrial effluent system where no activity is expected, according to the classification criteria in Section 2.2.1. Fortunately, the industrial effluent is subjected to sampling and analysis before evaluation for authorised discharge to the environment. It was during this evaluation process that the leaked activity was found and upon investigation, the leaking steam coil was identified.

The repair or replacement of the steam coil will require maintenance work to be performed in an area with radioactive contamination present in potentially high concentrations, as well as elevated ambient and localised high external exposure levels.

The radiological hazards can be predicted from the knowledge of the presence of dominant fission and activation products in SAFARI-1's medium activity effluent. This assumption is based on the fact that more than 90% of medium activity effluent received by LEMS in terms of volume, originated from SAFARI-1 [LEM13]. Notwithstanding, the hazards need to be quantified by means of a radiological survey. A dose assessment will be performed based on the survey results as well as a detailed work plan. The level of sophistication and corresponding degree of effort given to these assessments should be representative of and reflect the magnitude of the radiation problem concerned.

It is expected that the individual doses from external and internal (ingestion and inhalation) exposure will exceed 1 mSv and that an ALARA review would be required, as per Section 1.2.8. Such review needs to be performed to define, quantify and implement additional engineering, administration and RP measures in order to reduce and optimise the individual doses to acceptable levels. Furthermore, an individual ALARA goal for this task has been set at 2.5 mSv.

2.4 Physics of basic interactions of radiation with matter

This section provides an overview of the basic interactions of radiation with matter to obtain a better understanding in the sections which follow in this work in order to perform a dose assessment.

2.4.1 Overview of atomic and nuclear structure

Rutherford postulated, in 1911, that a nuclear atom consists of a heavy nucleus and negatively charged electrons situated around it and proved this experimentally. The nucleus was conceptualised as being composed of positively charged protons and sufficient equally-charged negative electrons. In 1913, Bohr postulated an atomic model where the orbiting electrons move in different orbits with varying energy levels. Bohr's atomic model ultimately led to the construction of the periodic table of elements [CEM09].

Ninety-two naturally occurring elements exists in nature with uranium having the highest atomic number (Z = 92). Elements with higher atomic numbers can be produced artificially. If the same element has different number of neutrons, it is called an isotope of the original element. Most elements contain several isotopes which can be stable or unstable. Unstable isotopes will spontaneously undergo radioactive decay to become stable.

According to Bohr's model, electrons move in orbits around the nucleus and at different energy levels. The electrons in the inner orbits require more energy to be removed than electrons in the outer orbits. The process of removing electrons from its orbits is called ionisation and the energy required to achieve this is called the ionisation energy, or the binding energy of the electron. By removing one of

the outmost electrons completely, the atom is "ionised" which results in one free electron leaving the original atom slightly different in mass but with a net positive charge.

2.4.2 Radioactive decay

It was already stated above that unstable isotopes will undergo radioactive decay in order to reach a lower energy state. Radioactive decay will result in 'radiation' emitted by the atom and the result will be a new atom, called the "daughter product".

The decay processes of interest are alpha decay, isobaric transitions (where the atomic mass number of the parent and daughter is the same) and isomeric transitions (where the atomic number of the parent and daughter is the same). The processes are discussed individually below with focus on the latter process as it is of interest for this work.

Alpha decay

An alpha particle is a highly energetic positively charged helium nucleus that is emitted from the nucleus of an unstable atom when the proton-to-neutron ratio is too high.

Isobaric transitions

There are three common forms of beta decay:

- β^--decay or beta emission is spontaneously produced in the nucleus by the transformation of a neutron into a proton and a single negative electrically charged particle (identical to an electron) and an anti-neutrino; and is ejected from the nucleus of the atom at very high speed according to the equation:

$$ {}^{1}_{0}\mathrm{n} \rightarrow {}^{1}_{1}\mathrm{H} + {}^{0}_{-1}\mathrm{e} + \upsilon \quad , \tag{2.2} $$

  where υ is the anti-neutrino;

- β^+-decay or positron emission is the transformation of a proton into a neutron and a single positive electrically charged beta particle is emitted from the nucleus of the atom; and a neutrino; according to the equation:

$$ {}^{1}_{1}\mathrm{H} \rightarrow {}^{1}_{0}\mathrm{n} + {}^{0}_{+1}\mathrm{e} + \upsilon \quad , \tag{2.3} $$

where υ is the neutrino;

- Electron Capture (EC) where the nucleus of the atom captures one of its own orbital electrons to cause the transition of an atomic proton into a neutron and the emission of characteristic X-rays of the daughter.

Isomeric transitions

The two forms of isomeric transitions are:

- Gamma rays are characteristic electromagnetic radiation which are emitted from the nucleus when the excitation energy of the nucleus is released; and
- Internal conversion is a process whereby an excited nucleus of a gamma emitting atom may rid itself of excitation energy resulting in the emission of characteristic X-rays and Auger electrons.

2.4.3 Sources of gamma rays

The gamma decay process is the most prominent decay process of interest to the treatment of liquid effluent in the evaporator at LEMS and the dose assessment which follows in this work. An understanding of the sources of radiation is therefore required.

Gamma rays following beta decay

As discussed in Section 2.4.2, beta decay can lead to some form of de-excitation by the daughter nucleus through the emission of a gamma-ray photon whose energy is essentially equal to the difference in energy between the initial and final

nuclear states. Examples of these are gamma-ray calibration sources used for instrument calibrations such as ^{137}Cs and ^{60}Co as shown in Figure 2.4.

Annihilation radiation

When the parent nucleus undergoes β^+ decay and this positron combines with a normal negative electron in an absorbing material, then both disappear and are replaced by two oppositely directed 0.511 MeV electromagnetic photons known as annihilation radiation. By way of example, the decay of ^{22}Na to ^{22}Ne is also shown in Figure 2.4.

Gamma rays following nuclear reactions

The result of nuclear reactions, such as the absorption of thermal neutrons by typical nuclei, can result in the production of characteristic gamma rays. This practice is used in nuclear reactors or extensively in industry to fabricate radioactive sources.

Bremsstrahlung

When fast electrons interact in matter, part of their energy is converted into electromagnetic radiation in the form of Bremsstrahlung. This process is of importance for the production of X-rays from conventional X-ray tubes.

Characteristic X-rays

The process of internal conversion which causes the characteristic X-rays within atoms was discussed in Section 2.4.2.

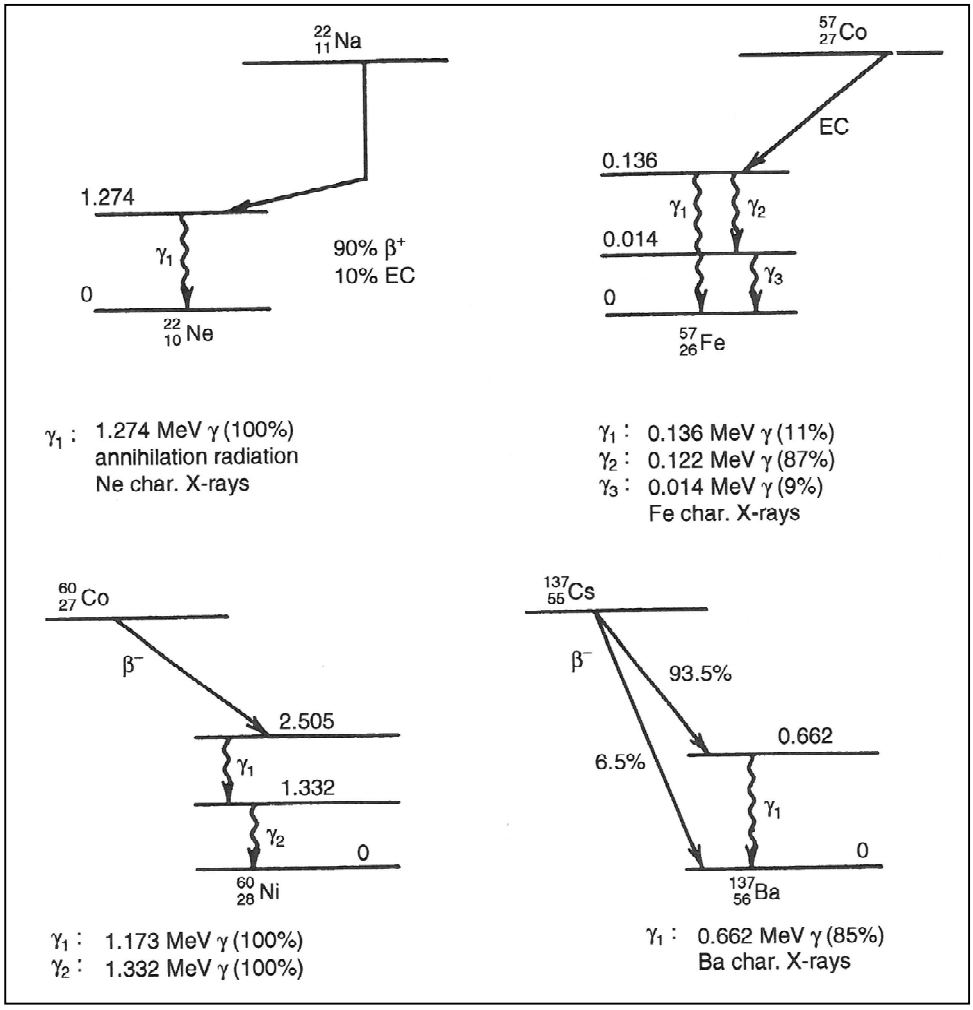

Figure 2.4 : Decay scheme for gamma-ray sources [KNO10].

2.4.4 Basic interaction mechanisms of ionising photons with matter

For radiation protection purposes an understanding of the interaction mechanisms whereby γ-rays can deposit energy into matter, especially the human body, is required. These mechanisms also play an important role in radiation measurements [KNO10].

The three major types of interaction mechanisms for γ-rays in matter are:

- photoelectric absorption,
- Compton scattering and
- pair production.

These are discussed in more detail below and are illustrated in Figure 2.5.

Photoelectric absorption

An ionizing γ-ray photon interacts with an absorber atom and transfers its energy to an atomic electron, ejecting that electron, known as an energetic photoelectron, from the atom. The kinetic energy of the resulting photoelectron is equal to the energy of the incident γ-ray photon minus the binding energy of the electron.

As can be seen in Figure 2.5, the transfer of energy takes place from the incident photon to the resulting electron. This process can thus be considered as a true absorption process. This is also the predominant mode of interaction for gamma rays (or X-rays) of relatively low energy [KNO10].

Compton scattering

An incoming ionising γ-ray photon is deflected through an angle with respect to its original direction while transferring a portion of its energy in an elastic scattering collision to an electron at rest in the absorbing material, which is then known as a recoil electron. Since all angles of scattering are possible, the energy transferred to the electron can vary from zero to a large fraction of the γ-ray energy.

This is the most predominant interaction mechanism for γ-ray energies typical of radioisotope sources.

Pair production

An incoming ionising γ-ray photon whose energy exceeds 1.022 MeV may, as it passes near a nucleus, spontaneously disappear, and its energy reappears as a positron and an electron, as illustrated in Figure 2.5. This positron and electron will be projected and again lose its kinetic energy by excitation, ionisation and bremsstrahlung. The positron will be annihilated in an interaction which will result in two 0.511 MeV photons (as described in Section 2.4.3) [TUR05].

Pair production is more probable at higher photon energies, generally in the region of tens of MeV, and can be considered an absorption process as the high energy photon will be converted into two 0.511 MeV photons and the dissipation of the remainder of the energy by the absorbing material.

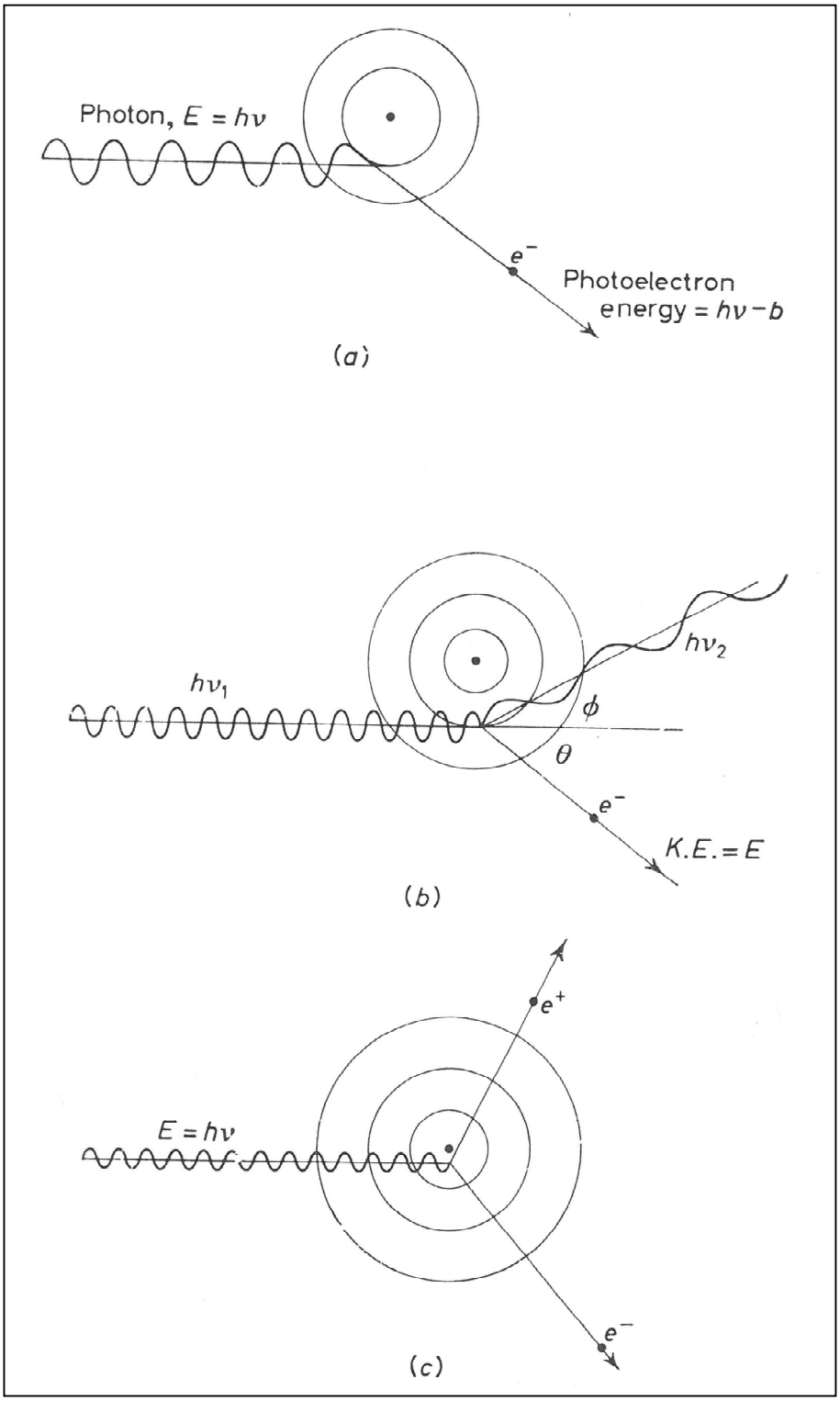

Figure 2.5 : Interaction of γ-ray photons with matter, showing in (a) Photoelectric absorption, (b) Compton scattering and (c) Pair production.

Relative importance of these interactions

Figure 2.6 illustrates the relative importance of the three main interactions as a function of the atomic number (Z) of the absorbing medium and the energy of the incident photon ($h\upsilon$) travelling at the speed of light. The energy of the incident photon, can also be expressed as $h\upsilon$ where h is Planck's constant and υ the frequency of the photon.

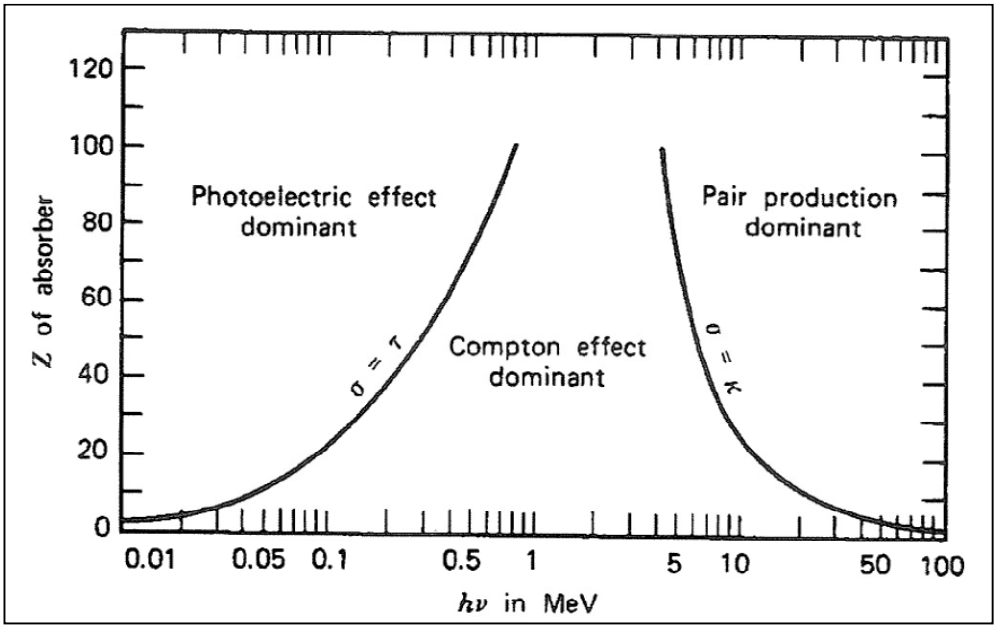

Figure 2.6 : Relative importance of the three major types of gamma ray interactions. Taken from [KNO10].

The three photon interactions allow for the interpretation of a number of properties of pulse-height spectra, as presented by J.E. Turner [TUR05]. This is illustrated in Figure 2.7 for the spectra from a ^{137}Cs gamma ray source with a gamma ray energy of 662 keV. The photons are actually released by the transition of the metastable (half-life 2.55 min) ^{137m}Ba daughter nucleus of ^{137}Cs to its ground state. The data in Figure 2.7 were collected with a 4 x 4 inch NaI(T1) crystal.

As discussed above, the transfer of energy normally occurs in one of two ways:

- a primary photon is absorbed (photoelectric absorption), and the photoelectron, Auger electrons, and fluorescence radiation are all absorbed in the crystal, or

- an incident photon that is Compton scattered one or more times is absorbed in the crystal.

The large narrow peak shown in Figure 2.7, known as the total energy peak in scintillation spectrometry, is due photoelectric absorption where the pulses are caused whenever the entire energy of an incident photon is absorbed in the crystal. The pulse is registered in the vicinity of 662 keV, which was expected and this also confirms that photoelectric absorption is an ideal process for measuring the energy of gamma rays.

The height of the pulse is a measure of the intensity of the energy and the spread of the width is a measure of statistical fluctuations in the conversion of the absorbed radiation energy into the number of electrons in the external circuit that registers the pulse. The combination of these is known as the energy resolution which is a measure of the effectiveness of the device and is discussed in Section 2.5.4.

The Compton edge, shown in Figure 2.7 at 478 keV, represents the maximum energy that an electron can acquire from Compton scattering by a primary photon.

Another characteristic feature of the spectrum in Figure 2.7 is the lower and wider part of the curve, known as the Compton scattering curve. This curve represents a continuum of pulses, mostly from single or multiple scatterings ranging from the most energetic electrons to electrons due to scattering through very small angles. The backscatter peak, shown at 187 keV in Figure 2.7, is caused from photons that are scattered into the scintillator from the surrounding neighbourhood and not from directly incident primary photons. These photons have been scattered mainly through large angles.

The size of the detector influences the prevalence of these effects. For example, the larger the detector, the more effective it captures the total energy of an incident photon, subsequently reducing the contribution of the continuum and increasing the relative size of the peak [TUR05]. The above effects are often masked by the finite energy resolution of the detector.

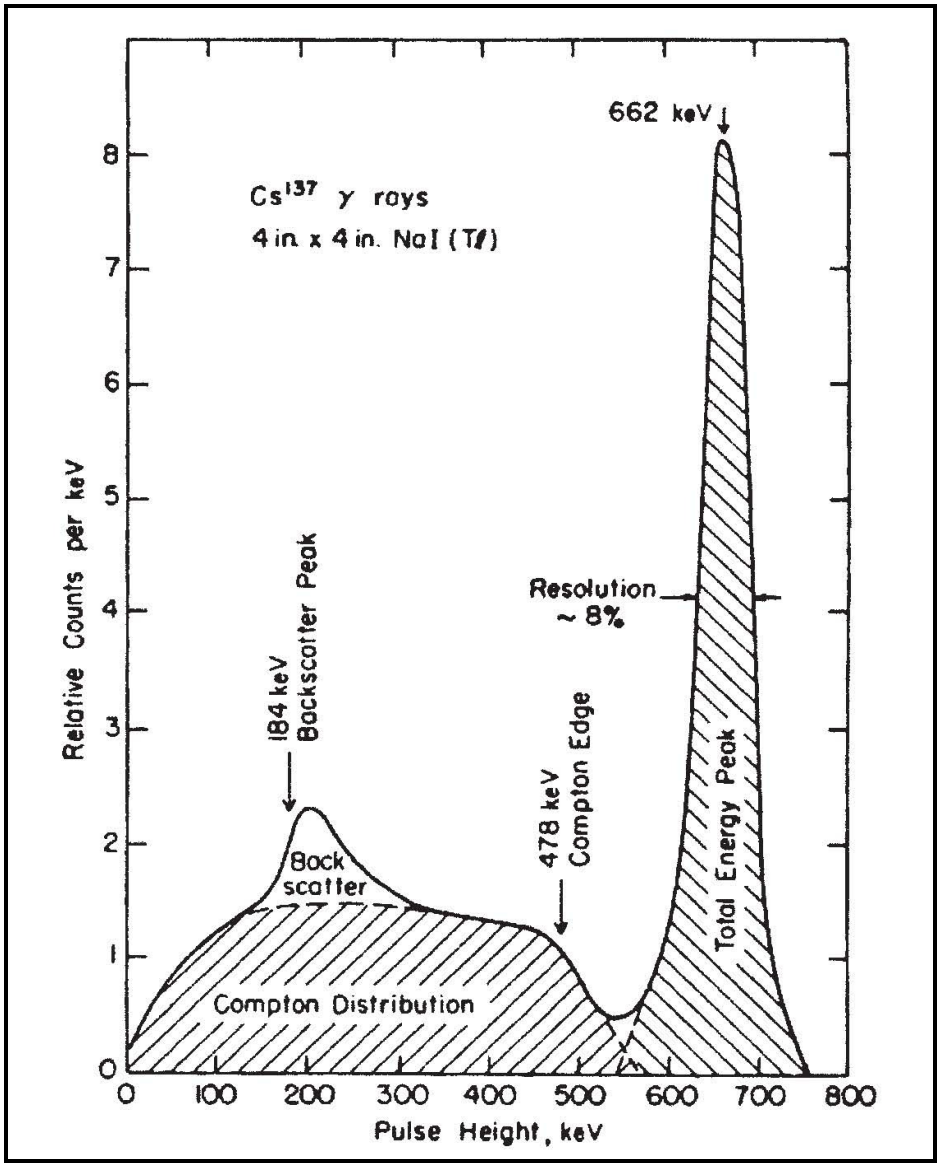

Figure 2.7: Pulse-height spectrum measured wih a 4 x 4 inch NaI(Tl) crystal scintillator exposed to gamma rays from ^{137}Cs. Taken from [TUR05].

2.5 Radiation detection and measurement

The physics presented in the preceding section serves as an aid in the understanding of the detection principles applicable.

2.5.1 Theory of detection and measurement

Radiation detection and measurement instrumentation in radiation protection includes a wide variety of applications, such as personnel monitoring, environmental monitoring or contamination measurements. This section will provide a broad overview of the theory of detection and measurement while the following section will focus on the instrumentation required for this work.

The principle of detection relies on one or more of the interactions discussed in Section 2.4.4 taking place inside a detector to create a charge within the electrical field of the detector. This charge is collected to form an electrical signal which is relayed to an electrical or mechanical device which is able to interpret it into some useful format.

Technological developments in the areas of material science, electronics, and computer technology have resulted in more sensitive, affordable, reliable, and user friendly instruments. This section will focus on two radiation measurement systems used in this work, namely:

- charged particle measuring instruments (subdivided into gas-filled counters and scintillation counters) and
- gamma-ray measuring instruments.

These systems provide the capability to measure, quantify and identify most forms of radiation from alpha-decay, beta-decay and gamma photons, respectively. Since neutrons are outside the scope of this work, the measurement of neutrons will not be discussed.

2.5.2 Measurement of charged particles

The most common charged particles for radiation protection measurements are β-particles (negative electrons, β^--particles) and alpha particles (which consist of two protons and two neutrons, the nucleus of a ^{4}He atom).

One important distinction between electrons emitted from the nucleus and alpha particles, which will contribute towards the capability to measure and identify particles, is the energy distribution of the charged particles emitted during the decay process. The energy from electrons is distributed over an energy continuum ranging from almost 0 eV to some maximum energy, usually referred to as E_{max}. The value for E_{max} from a beta energy spectrum could potentially be used to identify a radionuclide. In practice, with current instrumentation, it is difficult to get an accurate enough measurement of E_{max} to be used for conclusive identification. The contrary exist for alpha particles which are emitted at discrete energies. These energies are used in practice for identification purposes as they are characteristic of the parent radionuclide.

The detector in particle-counting instruments may be either a gas or a solid. In either case, the passage of an ionising particle through the detector results in energy dissipation by a burst of ionisation which is converted into an electrical pulse that actuates a readout device.

Gas-filled particle counters are all of similar design. There are three basic types of gas-filled detectors:

- ionisation chamber,
- proportional counter and
- Geiger-Müller counter.

They all operate on similar way but use different types of gases. Each has their own advantages and disadvantages in terms of size, operational costs and detection efficiencies. The selection of an operating voltage is critical for proper operation of a gas-filled counter.

In this work, a gas-flow proportional counter was used to measure the removable surface contamination on smear samples. Gas-flow proportional counters are operated at a bias voltage of normally ~ 600-800 V. The selection of voltage is important as the size of the output pulse is proportional to the high voltage across the detector. The correct voltage will ensure that only alpha particles are detected, commonly referred to as "alpha only" mode. If the bias voltage is increased to normally ~1500 V the counter will respond to both alpha and beta particles. Operation at this higher voltage is referred to as "simultaneous" or "α + β" mode.

Beta particles are more penetrating than alpha particles and are less affected by mass absorption. Beta particles have the ability to escape the active volume of a gas-flow proportional counter before all the energy from the β-particle is deposited within the detector and there is a higher probability of backscatter with beta particles which, depending upon the source backing material, can have a significant impact on the overall counting response with a proportional counter. The counting response is also dependent on the energy distribution of beta emissions and the interactions of the beta particles with the counting gas. Typical beta counting efficiencies of windowless gas-flow proportional counters are in the range 50-90%. The beta counting efficiency is a function of the average beta-decay energy which means there is no single beta counting efficiency to use for the calculation of the gross beta activity. In the calibration of these instruments, the beta efficiency for gross beta-decay measurements is determined with a standard reference material source prepared from ^{90}Sr (mean energy of 0.546 MeV) in secular equilibrium with its daughter ^{90}Y (mean energy of 2.281 MeV) [ICR08].

The alpha counting efficiency is fairly constant over the range of 4 to 6 MeV. For this reason and the fact that alpha particles are emitted at discrete energies, the alpha counting efficiency is normally determined with ^{241}Am which has a mean alpha-decay energy of 5.479 MeV [ICR08].

Samples in the form of smears taken from surfaces are counted in the gas-filled detectors to determine the gross alpha-decay and gross beta-decay activities.

Another type of instrument to measure charge particles is the scintillation counter. As the name implies, the detector makes use of a transducer that changes the kinetic energy of an ionising particle into a flash of light. This light is viewed by photomultiplier tubes or photodiodes whose output is interpreted and counted. Scintillation counters are used extensively to count alpha and beta particles and gamma rays. Since the intensity of the light pulse is proportional to the energy of the particle, distinction can be made between alpha particles of higher energy and beta particles of lower energy by means of discrimination of the pulse height of the electronic signal.

For radiation protection usage, scintillation counters have been developed which can be used as portable and battery powered instrumentation. These are relatively inexpensive to maintain except for the thin foil used to cover the outer surface of the scintillator. The purpose of this foil is to keep external light from entering the detector enclosure yet thin enough not to attenuate too much of the lower energy particles. This foil subsequently can become damaged fairly easily and requires replacement.

2.5.3 Gamma-ray spectrometry and dose measuring instruments

The gamma-ray measuring instrumentation used in this work is confined to instruments used in the surveillance of the work area. Subsequently, this work does not include instruments used to measure personal dose such as dosimeters. For this work, gamma-ray instruments were used to perform radionuclide identification and to measure dose rate.

Scintillation detectors, similar to the technology discussed previously, can be used for the measurement of gamma rays as well. The proportionality of the light signal to the energy of the incident gamma rays, provided the capability to develop a practical detector that has a high efficiency for the detection of gamma rays and, at the same time, capable of sufficiently good energy resolution to be useful in separating the contributions of polyenergetic gamma-ray sources [KNO10]. All three of the interaction mechanisms in Section 2.4.4 play a role in the selection of suitable materials to construct scintillation detectors.

Crystalline sodium iodide, in which a trace of thallium iodide has been added in the melt, is commonly known as NaI(Tl) and remains the most common choice for detector material because of its exceptionally large scintillation light output, relatively low production cost and excellent efficiency. However, poor energy resolution makes it less ideal for complex spectra analysis applications. It is most commonly used for simple spectra or online monitoring systems.

The most popular type of scintillation detector currently used for gamma-ray spectrometry in laboratory conditions, and used in this work, is the coaxial high-purity germanium (HPGe) detector. HPGe detectors provide excellent energy resolution but must be kept at low temperatures, typically 77 K, by using liquid nitrogen to maintain such good resolution.

Semiconductor diode detectors provide a more modern technology which has great benefit, particularly in reducing the size of detectors and in relatively fast timing characteristics. Semiconductor detectors act as a solid-state ionisation chamber [CEM09]. The ionising gamma ray interacts with atoms in the sensitive volume of the detector to produce electrons by ionisation which are collected and converted to an output pulse.

2.5.4 Energy resolution

The energy resolution is a very important aspect in the spectrometry for the interpretation of complex spectra and radionuclide identification. Knoll provides a definition for energy resolution as [KNO10]:

$$R = \frac{FWHM}{H_0} \quad , \tag{2.4}$$

where

R = energy resolution expressed as a percentage,
$FWHM$ = full width at half maximum of the full-energy peak and
H_0 = mean pulse height corresponding to the same peak

It is assumed that background noise has been subtracted from the observed signal. A scintillation detector has an energy resolution of 3-10%. The lower this percentage, the better the detector will be at distinguishing between two energy peaks whose energies lie near each other.

2.5.5 Calibration and accuracy of measurement

Calibration can be defined as the process of exposing an instrument in a known exposure field and comparing the instrument reading to these know field values [CEM09], often by means of a calibration factor or curve on a calibration certificate. The principle of calibration requires the metrologist to determine the instrument deviation (or error) from the known value and to perform adjustment if these deviations are outside acceptable limits. Acceptable limits for instrument deviation are provided by the International Electrotechnical Commission (IEC) for many types of instruments in different applications.

Reference radiation fields for calibration and routine testing of instruments are recommended by the International Standards Organisation (ISO) in their standards. Portable radiation monitors, for example, must be calibrated to determine the instrument linearity using a ^{137}Cs gamma-ray source and the instrument response to different gamma-ray energies should be determined against a ^{241}Am and ^{60}Co gamma-ray sources.

The assessment of uncertainty in measurement is the basis for quantifying the measurement accuracy as well as an essential aspect of quality assurance. In practice, radiation protection operators rarely perform a series of measurements to perform any form of uncertainty evaluation. Much reliance is placed on the ICRU recommendation [ICR93] for single measurements that *"....in most cases, an overall uncertainty of one standard deviation of 30% should be acceptable*."

The situation is further complicated by several influencing factors, such as the different radiation energies or the instrument angular response. For this study, all measurements were well above background levels which also reduce uncertainties considerably.

2.6 Dose assessment methodology

The IAEA Safety Guide on Occupational Radiation Protection [IAE99a] recommends the prior radiological evaluation to include:

- an identification of the sources of routine and reasonably foreseen potential exposures,
- a realistic estimate of the relevant doses and probabilities and
- an identification of the radiological protection measures to meet the optimisation principle.

The initial step is commonly referred to as "characterisation" [LAB07] which includes *in situ* measurements of radiological conditions.

Literature on dose assessment also suggests emerging radiological protection challenges to arise in the near future. One such recent paper by Lazo [LAZ11] elaborates on the possibility of "individual-level risk assessment" based on the ICRP developing male, female, child and infant phantoms. This challenges the current "one-size-fits-all" approach to exposure management for gender- and age-averaged individuals.

2.6.1 Exposure pathways

Numerous exposure pathways have been identified and considered to calculate the radiological exposure of individuals. For occupational exposure the most significant exposure pathways involve external exposure due to penetrating radiation and internal exposure due to inhalation or ingestion of radioactive materials. Other exposure pathways, such as internal exposure due to contamination of wounds and dermal absorption, do exist but is normally not considered for normal exposure scenarios. These pathways are either representative of abnormal situations or of very low consequence and are generally ignored for the purpose of normal operation exposure scenarios.

External exposure pathways are normally the dominant pathway for a wide variety of beta-gamma emitting radionuclides. Internal exposure due to the inhalation of airborne radioactive material is typically an important exposure pathway for alpha emitting radionuclides. Internal exposure due to the ingestion of radioactive material is generally considered for scenarios where transfer of contamination from hands to mouth can occur or where contaminated foodstuffs are ingested. Ingestion pathways are typically important for relatively soluble beta emitting radionuclides.

In this work, the identified exposure scenarios are:

- external exposure pathway due to gamma emitting radiation and
- internal exposure pathway due to ingestion and inhalation of contaminated material.

2.6.2 Assumptions

The IAEA Safety Guide [IAE99a] requires the prior dose assessment to describe the situation "*as precisely as necessary*". For typical routine maintenance tasks, the situation is well defined and performed on a regular basis which allows for an iterative process of assessment and monitoring to ensure optimisation of exposure. This is not the case for *ad-hoc* type situations where information is vague and sometimes unpredictable.

To enable the analyst to perform a dose assessment in the absence of complete information on the situation, the analyst will need to make assumptions. These assumptions can be based on external consultancy, literature or even prior experience. Examples of such assumptions are the percentage of contaminated material which will be suspended into the air due to the maintenance activities. These assumptions will be provided and substantiated in Chapter 3.

2.6.3 External dose assessment

External radiation in the workplace is measured in the operational quantity ambient dose equivalent rate $H^*(10)$ for strongly penetrating radiation and

$H^*(0.07)$ for weakly penetrating radiation [IAE99a]. The operational quantity for individual monitoring, recommended in the BSS, is the personal dose equivalent $H_p(10)$ for deeply penetrating and $H_p(0.07)$ for weekly penetrating radiation respectively. By using the operational quantities $H^*(10)$ or $H_p(10)$, one obtains approximate values for effective dose. It should be noted that these are operational quantities for practical use in radiological protection introduced by the ICRU [ICR93].

Instrumentation to measure ambient dose equivalent rate or personal dose equivalent rate, is calibrated to perform this function. The results of such measurements are used to calculate the exposure of workers to external radiation.

The external exposure of workers is dependent on several variables which need to be defined in the exposure scenario. The variables are:

- prevailing ambient dose rate where the work is to be performed including the type and uniformity of the radiation field,
- geometry of the radiation source including distance between source and the worker and
- duration of work to be performed.

This will be applied in the assessment in Chapter 3.

2.6.4 Internal dose Assessment

Internal exposure of workers occurs when radioactive material is taken into the body by inhalation, ingestion or through the skin [IAE99c]. For occupational exposure of workers, the ICRP has developed models for describing the behaviour of radionuclides that enter the body either by inhalation or ingestion. For exposure pathways through the skin or open wounds, exposure is only likely to occur as a result of accidents. Except for tritium, which is readily absorbed through the skin, no internationally accepted models have therefore been developed for these exposure scenarios [IAE04].

To perform a prospective dose assessment for internal exposure information regarding the prevailing radiological conditions and radionuclides involved are required. The internal exposure of workers is dependent on several variables which need to be defined in the exposure scenario. The variables are:

- physical condition of the material to be handled (particle size distribution, loose contamination) and work to be performed (e.g. aggressive cutting work),
- quantity of material present and available for intake and
- duration of work to be performed.

Using the above information and assumptions on the behaviour of materials, an air concentration can be determined to calculate the inhaled quantity of radioactive material. Similarly, the quantity of ingested removable radioactive contamination when transferred from surfaces to hands, foodstuffs, cigarettes or other items that enter the mouth, can be calculated. The ingestion and inhalation doses are calculated by applying dose conversion factors available from the BSS to the ingested quantities.

The quantity adopted in the BSS for internal dose for radiological protection purposes are the effective dose [IAE96]. For occupational exposure, all exposed workers are adults and therefore the period of time over which the committed effective dose is assessed is 50 years, irrespective of the age at intake.

In the assessment in Chapter 3, the internal dose will be determined.

2.7 ALARA and the optimisation of exposure

The ALARA concept (introduced in Section 1.2.6.), the evaluation of ALARA and examples of scenarios where it was deemed appropriate, is discussed. Bevelacqua [BEV10] reasons that ALARA encompasses more than the traditional time, distance, shielding and planning approach. ALARA is a process of

quantitative and qualitative aspects which revolves around the dose to the individual. An effective ALARA programme requires management commitment and cooperation between facility workgroups. The development and sustainability of effective ALARA programs require the establishment and monitoring of goals, rewarding the successful achievement of those goals, and incorporating lessons learned from tasks that fail to meet their goals.

When is ALARA achieved?

As recent as 2005, the Nuclear Energy Agency (NEA) recommended in its report [OEC05] on optimisation in radiological protection: "*It would be useful if the ICRP would provide guidance as to the elements to be considered, from the scientific standpoint, when deciding that a particular approach is optimum or reasonable, and will result in doses that are ALARA*".

ALARA has been discussed extensively in many publications, but the general consensus appears to be the quantitative demonstration by means of safety assessment supplemented by a structured evaluation process. The NEA, in 2011, published a report on occupational radiological protection for new nuclear power plants [OEC11] in which such a structured approach in terms of ALARA is recommended, consisting of:

- data collection and networking on international level (by means of ALARA networks),
- ALARA design review committees and
- ALARA design checklists as an important tool for design analysis and assessment.

ALARA incentives

Bevelacqua [BEV10] suggests the implementation of a simple, cost effective incentive program where the attainment of ALARA goals is rewarded but warns against overemphasizing performance indicators. Such rewards are intended to be in the form of safety related items such as keyrings with slogans or sports items such as jackets.

Use of ALARA tools

Commercially several tools are available under the banner of ALARA tools such as portable shields, remote handling tools, etc. Recent developments include computer software to enable three dimensional modelling of scenarios to enable better planning of operations.

Several of this type of computer software is already available [VER07]. A popular choice appears to be the VISIPLAN 3D ALARA planning tool developed by the Belgian Nuclear Research Centre SCK•CEN in Belgium during the decommissioning project of its BR3 nuclear reactor [VER05]. This tool calculates the dose for different work scenarios taking into account worker position, work duration and subsequent geometry and source distribution changes in a three dimensional simulation of the workplace. This tool was applied successfully to dose optimisation problems in decommissioning and dismantling activities for typically hot cells containing highly radioactive sources at SCK•CEN and other nuclear installations [VER07].

Application of ALARA in the handling of radioactive effluent

The closure of the high level radioactive liquid waste farm at Idaho Laboratories [AIT05] necessitated improvements to the facility in order to demonstrate better radiological control and ALARA performance. The following are some of the identified improvements:

- use of remote operation and handling kept operators away from high radiation levels,
- washdown of equipment and structures prior to operations reduced the potential for contamination of personnel and internal exposure and
- use of containment tents around structures to reduce spread of contamination.

The outcome of this project after implementation of the identified ALARA principles, is reported to have reduced the personnel exposure from 0.076 Sv to 0.03 Sv.

CHAPTER 3

CALCULATION OF DOSES TO THE WORKERS, AND OPTIMISATION OF EXPOSURE

The main aim of this chapter is to perform the dose assessment for the replacement of the leaking coil in the evaporator.

First the dose assessment methodology which will be presented followed by the internal and external doses will be assessed. The assessment includes an interpretation of the measurement results, explanation of any assumptions and formulae used and the results of the calculations.

3.1 Introduction

As discussed in Section 1.2.3, the ICRP use a situation-based approach to characterise the possible situation where radiation exposure may occur as *planned, emergency*, and *existing exposure situations*, and apply the fundamental principles of justification and optimisation of protection to all of these situations.

The task to replace the leaking coil in the evaporator can be classified as a *planned exposure situation.*

In order to predict the planned exposure of workers for the leaking coil replacement, a dose assessment has to be performed. This dose assessment involves the calculation of the total effective dose to individual workers which

includes the contribution from direct external radiation and radiation as a result of radionuclides taken into the body.

3.2 Dose assessment methodology

The exposure of workers from the leaking coil replacement may occur in three main ways:

- external exposure to radiation from the radioactive materials,
- the inhalation of airborne particles of radioactive materials or
- the ingestion of radioactive materials.

Although other exposure pathways do exist, for example through skin absorption or open wounds, these pathways are excluded as stated earlier.

3.2.1 Principles and methodology

A study of available literature revealed several papers, documents and handbooks devoted to the subject of dose assessment [CEM09, IAE92, IAE04, IAE05]. The dose assessment methodology appears to be well established and consistent. Formulae used for internal and external dose calculations are also consistent however, there appears to be variability in the various factors used in the formulae, which will be addressed in the following sections of this chapter.

Although this leaking coil replacement task is regarded as a maintenance related action, it will entail the preparation, dismantling, removal and replacement of individual components, and will be performed by the Decommissioning Group at Necsa. This group has substantial experience in decommissioning projects and access to suitable equipment.

Given the nature of this project and the dose assessment required, the most appropriate methodology was selected to be that provided in IAEA Safety Series 111-P [IAE92].

3.2.2 Assumptions

Exposure scenarios are normally unique in nature and case specific. Variability can be expected in the physical behaviour of workers, materials, methods employed and time taken to complete a task. Formulae used for dose calculation contain factors to account for physical transfer, translocation, deposition, etc. of radiological materials, which can vary based on situation. Even these factors can vary based on the situation. Fortunately, many studies are available to assist with and justify the selection of these factors. For this reason, some assumptions need to be made. All assumptions are normally a balance between being conservative, showing best judgement and being pragmatic.

In the following sections, each factor used shall be substantiated by evidence on its selection, or justified in terms of the assumptions used.

3.2.3 Total effective dose

The total effective dose (E) is estimated from the following expression obtained from the ICRP [ICR07]:

$$E = H_p(10) + E(50) \quad , \tag{3.1}$$

where

$H_p(10)$ = the personal dose equivalent from external exposure at a depth of 10 mm in soft tissue and

$E(50)$ = the committed effective dose from internal exposure.

3.2.4 External dose assessment

To calculate the external doses to workers, information regarding the external radiological hazard is required. The results of a radiological surveillance will be used, along with information on the exposure scenario (position, distance from evaporator walls, stay time, etc) to calculate the external dose to the workers assuming a uniform whole body exposure [ICR07].

The personal dose equivalent $H_p(10)$, which forms a part in Eq. (3.1), is calculated as follows:

$$H_p(10) = \dot{H}_p(10) \times t \quad , \tag{3.2}$$

where

$\dot{H}_p(10)$ = the personal dose equivalent rate in Sv/h and

t = the exposure time in hours.

It is important that the measured personal dose equivalent rate be representative of the position that the worker will occupy during the task.

3.2.5 Internal dose assessment

To calculate the internal doses to the workers, an evaluation of the contamination hazard needs to be performed. The result of this surveillance will be used, along with some assumptions, and information on the exposure scenario, to perform a series of calculations.

The committed effective dose from internal exposure ($E(50)$), which forms a part in Eq. 3.1, is calculated as follows [ICR07]:

$$E(50) = \sum_j e_{j,inh}(50) \cdot I_{j,inh} + \sum_j e_{j,ing}(50) \cdot I_{j,ing} \quad , \tag{3.3}$$

where

$e_{j,inh}(50)$ = committed effective dose coefficient for activity intakes by inhalation of a radionuclide j,

$I_{j,inh}$ = activity intake of a radionuclide j by inhalation,

$e_{j,ing}(50)$ = committed effective dose coefficient for activity intakes by ingestion of a radionuclide j and

$I_{j,ing}$ = activity intake of a radionuclide *j* by ingestion.

Since the area is normally unoccupied, the prevailing radiological conditions will not be representative of the conditions when work is performed in the area. Therefore, it is important to make use of appropriate assumptions e.g. re-suspension factors, when performing the assessment to calculate for example the potential airborne activity concentrations.

Furthermore, when the containment of the evaporator is breached (cutting into it), the radiological condition is expected to change as the contaminated material is released into the room. This needs to be included in the internal dose assessment.

The internal dose assessment will calculated by taking the doses from inhalation and ingestion, as these are regarded as the relevant pathways using the formulae from Ref. [IAE92].

The committed dose from inhalation ($H_{inh,j}$) of radionuclide *j* is calculated as follows [IAE92]:

$$H_{inh,j} = V \; t \; e_{inh,j}(50) \; W(C_d \; C_{w,j} + C_{s,j} \; RF \; TF_{inh}) \quad , \tag{3.4}$$

where

V = the breathing rate of the worker in m^3h^{-1},

T = the duration of the task in hours,

$e_{inh,j}$ (50) = committed effective dose coefficient for activity intakes by inhalation of a radionuclide *j*,

W = the fraction of the material handled by the worker, taken to be 1 in this assessment,

C_d = the concentration of respirable dust in air in $g.m^{-3}$,

$C_{w,j}$ = the concentration of radionuclide j in the material in $Bq.g^{-1}$,

$C_{s,j}$ = the concentration of radionuclide j in the surface contamination

in Bqcm^{-2},

RF = the resuspension factor for surface activity in m^{-1} and

TF_{inh} = the transfer factor for the inhalation of surface activity (dimensionless).

The committed dose from ingestion ($H_{ing,i}$) of radionuclide i is calculated as follows [IAE92]:

$$H_{ing,j} = t \; e_{ing,j}(50) \; W(IC_{ing,j} + I_2 TF_{ing} C_{s,i}) \quad , \tag{3.5}$$

where

t = the duration of the task in hours,

$e_{ing,j}$ (50) = committed effective dose coefficient for activity intakes by ingestion of a radionuclide j,

W = the fraction of material handled by the worker (taken to be 1 in this assessment),

I = the rate of secondary ingestion of removable surface contamination in gh^{-1},

$C_{ing,j}$ = the concentration of radionuclide j in the material in Bqg^{-1},

I_2 = the rate of secondary ingestion of removable surface contamination in m^{2}h^{-1},

TF_{ing} = the transfer factor for ingestion of surface activity (dimensionless) and

$C_{s,j}$ = the concentration of radionuclide j present in the surface contamination in Bqm^{-2}.

3.3 Defining the exposure scenarios

The leaking coil replacement operation can be divided into the following separate groups of actions:

- preparatory work to be performed in the area to support the evaporator structure for the removal of a section from it,
- removal of lagging from evaporator,
- initial loosening of bolts,
- cutting by plasma torch of the defunct evaporator,
- collection and removal of all parts and waste generated from the area,
- placement of the sections of the new evaporator unit into place and
- completion of new installation including new lagging, torque of bolts, and removal of tools.

Proper planning of the replacement task by the operators has produced a task list detailing the following information for each of the above actions:

- detailed description of each task to be performed,
- location of the workers and number of workers relative to the equipment,
- estimated duration of each task and
- identification of the workers (in case a worker is expected to perform multiple tasks).

This information is critical to the dose assessment as it defines the actions of the individual worker involved.

The identified actions were then grouped into scenarios with similar exposure characteristics:

Scenario 1

Workers are exposed in a scenario where radiation levels are elevated and where surface contamination is present. Actions associated with all preparatory work before the containment of the evaporator is compromised (cut). Exposure pathways encompass external irradiation from the evaporator contents, the inhalation of contaminated dust and the inadvertent ingestion of contaminated material (e.g. by the hand to mouth transfer pathway).

Scenario 2

Workers perform cutting and removal operations which cause aggressive resuspension of volume contaminated material in area with elevated radiation levels. Exposure pathways encompass external radiation from the evaporator contents, the inhalation of contaminated dust and the inadvertent ingestion of contaminated material (e.g. by the hand to mouth transfer pathway).

Scenario 3

Workers place new components in place and complete the installation. The exposure pathways include external radiation from the surrounding evaporator contents, the inhalation of contaminated dust and the inadvertent ingestion of contaminated material (e.g. by the hand to mouth transfer pathway) with significantly lower levels of contamination present.

The scenario are summarised in Table 3.1.

Table 3.1: Exposure scenarios considered and relevant exposure pathways

Scenario	Description	Number of workers	Duration of task(s)	Relevant exposure pathways
1	Preparatory work to support the evaporator structure; Removal of lagging; Initial loosening of bolts.	4	16 hours	External exposure Inhalation of dust Inadvertent ingestion
2	Cutting by plasma of the defunct evaporator; Drumming and removal of all parts and waste generated from the area;	3	3 hours	External exposure Inhalation of dust Inadvertent ingestion
3	Replacement of the new evaporator unit in sections into place; Completion of new installation including new lagging, torque of bolts, removal of tools.	4	14 hours	External exposure Inhalation of dust Inadvertent ingestion

3.4 Radiological surveillance instrumentation

The radiological surveillance instruments used for the surveillance were determined based on the radiation properties of the nuclides known to be present in the facility, using information obtained from the generators of the effluent (see Section 2.2.4). The nuclides are mainly fission and activation products, which are known beta and gamma irradiators.

The instruments used in the surveillance are limited to instrument available at the time of the surveillance and are listed in Table 3.2. Fortunately, all of the handheld instruments were equipped with digital display meters. Many instruments still in use today make use of analogue meters which introduces geotropism [ROL06] and the problem of unstable needle response. All instruments were calibrated to international standards [IEC02][IEC09][ISO99].

Table 3.2 : List of radiological measuring instrumentation

Measurement type	**Physical properties**	**Instrument(s)**
Contamination levels	Total Surface contamination	Thermo Scientific Electra connected to DP2R/4A probe
	Removable surface contamination	Gas proportional counter
Radiation levels	External radiation	Thermo Electron Corporation Interceptor
Sediment analysis	Nuclide identification	High purity germanium (HPGe) detector
	Activity concentration	

Measurement of contamination levels

The instrument selected for the measurement of the total surface contamination present on the external surfaces of the work area, was the Thermo Scientific Electra, which is a digital, microprocessor based ratemeter. The ratemeter was connected to the Thermo Scientific DP2R/4A “dual phosphor probe” which

responds to alpha, beta and gamma radiation. The instrument is illustrated in Figure 3.1. The probe consists of an inorganic zinc sulphite (ZnS) scintillator which has very high scintillation efficiency used primarily for alpha particles [KNO10], and a BC-400 plastic scintillator for the detection of medium to high energy beta and gamma radiation. A photomultiplier tube is used for amplification of the light pulse, as described in Section 2.5.2. Based on calibration certificates, the typical detection efficiencies observed for this instrument is 20 % to 30 % for surface activity.

To determine the levels of removable surface contamination (as a fraction of the total surface contamination measured in the preceding paragraph), the area where the work is to be performed as well as the surrounding area where workers will move, were subjected to smear testing. Smear testing is the process of wiping the suspected contaminated surface with a filter paper to collect the surface contaminated matter and then measuring the activity on the paper [CEM09].

At Necsa, this would require a 100 cm^2 area to be 'wiped' with a dry 47 mm diameter smear paper and sent to the RadioAnalysis laboratory, which will measure the activity on the paper with a calibrated gas proportional counter, as discussed in Section 2.5.2. The measured activity is corrected for a 10% collection efficiency to determine the removable surface contamination [ISO88] [IAE08]. This is a generally acceptable practice, and experimentally confirmed by Klein [KLE92].

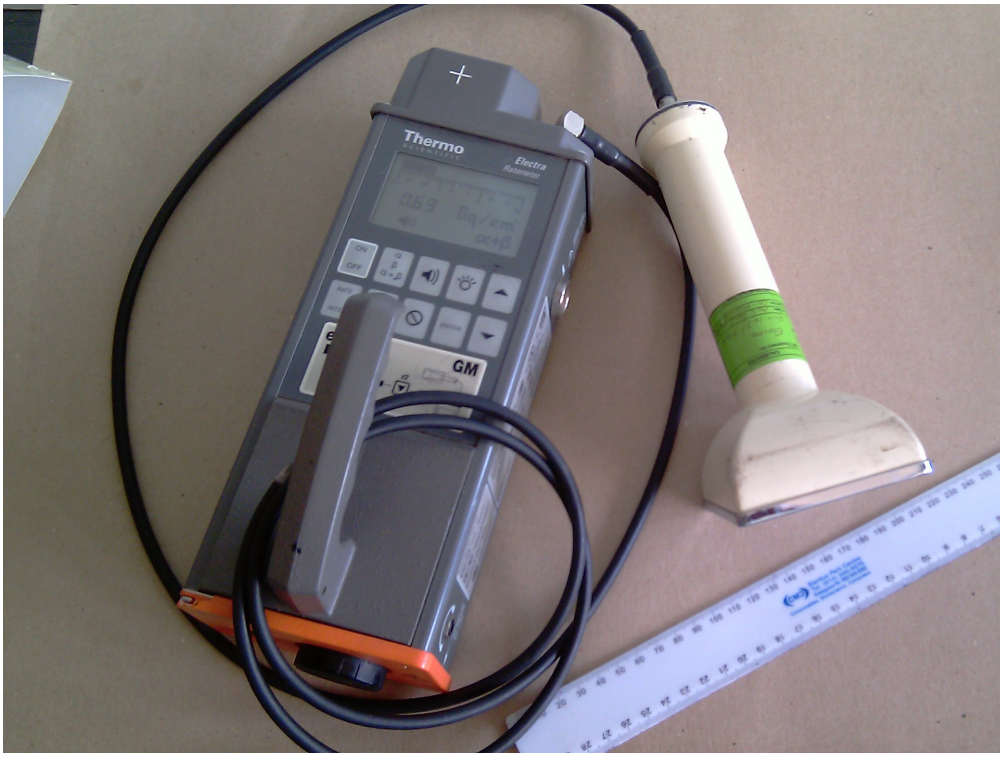

Figure 3.1 : Portable contamination monitor

Measurement of radiation levels and nuclide identification

The instrument used for the measurement of radiation levels was the Thermo Electron Corporation Interceptor, depicted in Figure 3.2. This is a portable radiation detector suitable for the measurement of gamma dose rate with isotope identification capability through three Cadmium-Zinc-Telluride (CZT) detectors.

Only one of the CZT detectors is used for isotope identification with a quoted gamma-ray energy range of 30 keV – 1.5 MeV with an energy resolution of 2.4% - 3.5% FWHM for the 662 keV gamma ray for ^{137}Cs, when measuring a gamma-ray energy spectrum on 1024 channels [INT08]. CZT detectors consist of room temperature semiconductors that directly convert X-ray or gamma-ray photons into electrons for detection. It is a new technology which sacrifices resolution for size [FRA05].

The other CZT detectors are used for the measurement of radiation levels. The instrument was calibrated by an accredited calibration laboratory to measure ambient dose equivalent rate as required by the International Electrotechnical Commission in IEC 60846-1 [IEC09].

Figure 3.2 : Thermo Electron Corporation's Interceptor portable radiation detector.

Analysis of sediment

The RadioAnalysis laboratory at Necsa is equipped to perform accurate identification of radionuclides and quantification of the activity concentration in a sample using sophisticated instrumentation which is operated under laboratory conditions. This technique is known as gamma spectrometric analysis and was introduced in Section 2.5.3.

In a paper on the history of radiation detection instrumentation [FRA05], Frame describes the first germanium detector in 1963 to have "*truly revolutionised the way we do things*". Today high purity germanium (HPGe) detectors is the norm

for laboratory analysis because of many advantages, including the ability to warm to room temperature compared to older germanium systems which had to be continuously maintained at low temperatures [KNO10].

The RadioAnalysis laboratory performs analysis using a calibrated coaxial high purity germanium (HPGe) detection system.

3.5 External dose calculation

The external doses will be calculated using the results from a radiological surveillance and a defined set of exposure scenarios for which a set of formulae and assumptions are needed. The exposure scenario for external exposure is described in Table 3.1 based on the tasks and workers defined in Section 3.3.

3.5.1 Interpretation of measurement data

A radiological surveillance was performed in the area where the task will be performed. It is important to note that the area was subjected to cleaning with a high pressure water spray apparatus to remove as far as practically possible loose contamination. Since the potential for effluent leakage from the facility exist, the room was designed to accumulate any such effluent in a bunded area. The effluent from this wash-down facility was thus collected in the bunded area and rerouted as medium activity effluent back into the process. This is considered an ALARA measure since the potential for exposure was reduced.

Measurements were performed by a radiation protection officer using the Thermo Electron Corporation Interceptor to determine the external dose rates as well as some spectrometric measurements (discussed in Section 2.5.3). Ambient dose equivalent rate measurements were performed at four radial positions at three intervals along the vertical axis of the evaporator at a distance of 500 mm from the outer surface of the evaporator shell, as detailed in Figure 3.3. The distance of 500 mm from the outer surface was considered representative of the location that a worker's body would be.

3.5.2 Identification of relevant nuclides

Spectrometric measurements were performed using the Thermo Electron Corporation Interceptor discussed in Section 3.4. It is important to note that the instrument is not capable of quantifying the result of such measurement as such measurements are highly dependent on the configuration of use (source detector distance, physical size of source, etc.).

Table 3.3 summarises the results of measurements taken with the Thermo Electron Corporation Interceptor placed within four angular points on evaporator at three different positions.

Spectrometric measurements were performed using the Thermo Electron Corporation Interceptor discussed in Section 3.4. It is important to note that the instrument is not capable of quantifying the result of such measurement as such measurements are highly dependent on the configuration of use (source detector distance, physical size of source, etc.).

Table 3.3: Results of measurements with Interceptor on evaporator at 500 mm

Position	Radial measurements of personal dose equivalent rate (µSv/h)				Average dose equivalent rate (µSv/h)
	0°	90°	180°	270°	
Position A	206	198	244	236	221
Position B	170	195	166	161	173
Position C	266	245	230	215	239

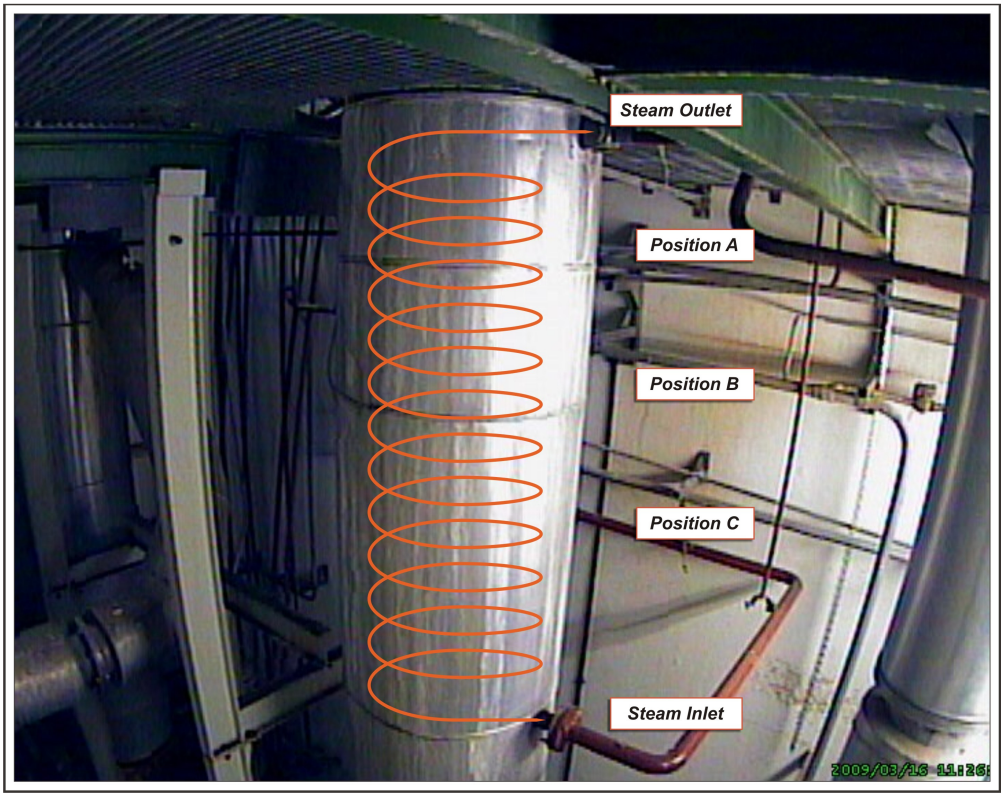

Figure 3.3: Positions where dose rate measurements were performed on evaporator.

The spectrum is illustrated in Figure 3.4. showing the large Compton backscattered peak as well as smaller peak, which the software was able to identify as ^{103}Ru and ^{95}Nb with high confidence and ^{95}Zr with lower confidence, as shown in Figure 3.5. Other gamma emitting radionuclides may also be present but in concentrations which are too low to distinguish by this instrument.

The radionuclides were identified by its dominant gamma ray energies [ICR07]:

- ^{103}Ru at 497.1 keV with intensity of 91.0 %,
- ^{95}Nb at 765.8 keV with intensity of 99.8 % and
- ^{95}Zr at 756.7 keV with intensity of 54.4 %.

The knowledge of the radionuclides present and the gamma ray energies at which it is detected provides confidence in the instrument being able to measure accurately at these energies, which is close to the 662 keV gamma ray energy of ^{137}Cs.

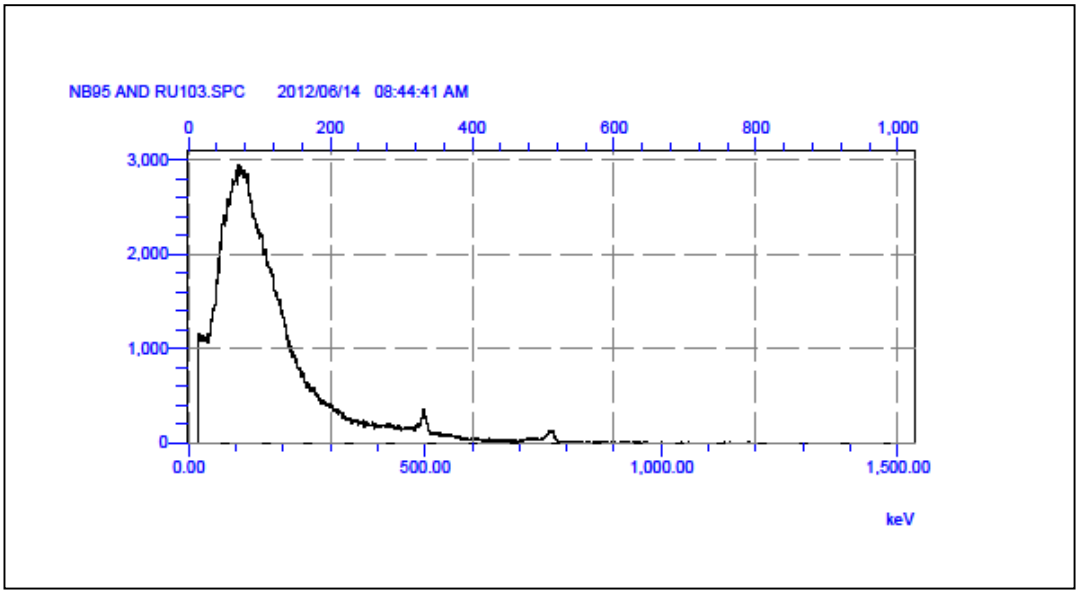

Figure 3.4: Graphic of spectrum drawn with Interceptor portable radiation detector as displayed by instrument.

2012/06/14 02:51:14 PM

Identification results:

Spectrum Name: Nb95 and Ru103.SPC

File Name:

Nuclide Name:	Nuclide Type:	Confidence Indication:	Confidence Description:
Ru-103	Industrial	99	High
Nb-95	Industrial	90	High
Zr-95	Industrial	54	Low

Figure 3.5: Results of spectrometric analysis performed by Interceptor portable radiation detector as displayed by instrument.

3.5.3 Assumption used for external dose calculation

For the calculation of the external dose to the workers, some assumptions regarding the exposure scenarios and the measurement results apply. These are discussed further.

Correction for decay of radionuclides

The radionuclides identified in Section 3.5.2 have half-lifes of 39.26 days for ^{103}Ru, 34.991 days for ^{95}Nb and 64.032 days for ^{95}Zr [ICR07].

No correction for radioactive decay will be applied as it was assumed that the work will be performed within a short space of time (typically one week) and that the radionuclides contributing to the measured external dose rate, reported in the preceding section, will remain present in the evaporator as a result of the ongoing operation of the evaporator.

Exposure times

The exposure times provided in Table 3.1 are based on the time periods planned by the project team based on a detailed work analysis.

It is furthermore assumed that all the workers will be exposed for the duration of the periods given**.** This is a conservative assumption as not all of the workers will be exposed all of the time; a certain task might require only one worker at a time to be exposed.

Ambient dose equivalent rates

The measured dose rates are considered representative of the exposure scenarios presented for Scenarios 1 and 2. For Scenario 3 the dose rates after removal of the defunct evaporator unit is not known prospectively. It is conservatively assumed that the dose rate will not change. After completion of the removal tasks, it is recommended that the dose rate be measured and the results either confirmed or re-evaluated.

The external dose to the personnel is calculated using Eq. (3.2) and the information provided in the preceding sections.

The result of the calculation of the external dose to the workers are summarised in Table 3.4. The results indicate that the doses for scenarios 1, 2 and 3 are directly related to the duration of the respective tasks.

Table 3.4: Results of external dose calculations

Scenario	**Personal dose equivalent rate (µSv/h)**	**Duration of task(s) (hours)**	**Personal dose equivalent (µSv)**
1	211	16	3376
2	211	3	633
3	211	14	2954

3.6 Internal dose calculation

The internal doses were calculated using the results from a radiological surveillance, a defined set of exposure scenarios, a set of formulae and assumptions (where needed) as discussed in the preceding sections.

Following the washdown of the facility, loose contamination has been confirmed to remain present on surfaces such as walls, railings, grids and the evaporator outer surface. The risk from internal exposure, however, will be significantly increased once the integrity of the confinement of the evaporator is breached, that is when the evaporator is cut to remove the damaged sections.

3.6.1 Interpretation of measured data

The contamination levels were determined by smear samples and sediment analysis, and are used in the calculations of the internal exposure to workers.

Total surface contamination levels

The intention of the surveillance was to use the Thermo Scientific Electra with DP2R/4A scintillation detector, described in Section 3.4, to measure the total surface contamination levels.

As stated in Section 3.4, the instrument responds to "medium to high energy beta and gamma radiation". The high prevalent radiation levels reported in Table 3.3 will result in the instrument showing high readings in the beta contamination mode which the instrument is not able to distinguish from gamma rays. Therefore, the instrument is not suitable to quantify the total beta surface contamination levels. In the alpha mode, the instrument readings are not affected by the high radiation levels. Measurements in the alpha mode indicated that alpha surface contamination is very low in the room or on the equipment.

Removable surface contamination results

The analysis result of smear samples, described in Section 3.4, taken on the external surfaces of the evaporator were alpha contamination of 200 Bqm^{-2} and beta contamination of 461 000 Bqm^{-2}.

As confirmed with the handheld measurement, the alpha contamination levels are very low.

Volume contamination results

A visual inspection (through a sight glass) confirmed that the inside of the evaporator is covered with a layer of dry sediment even after the evaporator has been subjected to repeated flushing. The potential exists for this sediment to become airborne during the cutting and removal task and pose an internal contamination hazard for workers. Subsequently, a sample of this sediment was extracted from the evaporator and subjected to gamma spectrometric analysis. The result of the laboratory analysis is attached as Appendix A and summarised in Table 3.5.

Table 3.5: Nuclide identified with associated activity concentrations

Radionuclide	**Activity concentration [MBq/kg]**
^{60}Co	11
^{95}Nb	440
^{95}Zr	250
^{103}Ru	230
^{106}Ru	39
^{125}Sb	30
^{137}Cs	1.7
^{140}Ba	2.6
^{140}La	3.8
^{141}Ce	2.8
^{144}Ce	74

The RadioAnalysis laboratory performed analysis using a calibrated coaxial high purity germanium (HPGe) detection system, described in Section 3.4. Results in Table 3.5 indicate high activity concentration of ^{95}Nb, ^{95}Zr and ^{103}Ru, with other radionuclides in lower concentrations. This confirms the results of the handheld instrument in Section3.5.2.

The presence of these radionuclides was expected. Although fission product yields are well known, their relative contribution to a mixture that contains corrosion activation products is not easy to predict, according to Thind [THI01]. In his paper, Thind suggest that ^{95}Zr and ^{95}Nb are created from two sources, namely fuel and fuel cladding, and therefore might constitute a significant fraction of the total activity in the mixture, as observed in the analysis result in Appendix A.

The radiological hazards of ^{95}Zr and ^{95}Nb are also well known and studied, as confirmed in a paper by Fletcher [FLE68] on studies performed on rats and dating back to 1968, in which references to papers dating back to 1947 are made.

3.6.2 Assumptions used for internal dose calculations

For the calculation of the internal doses to the workers, some assumptions regarding the exposure scenarios and the measurement results apply. These are discussed further.

Correction for decay of radionuclides

As stated in Section, 3.5.3, no correction for radioactive decay will be applied.

Exposure times

The exposure times provided in Table 3.1 are based on the time periods planned by the project team based on a detailed work analysis.

It is furthermore assumed that all the workers will be exposed for the duration of the periods given**.** This is a conservative assumption as not all of the workers will be exposed all of the time; certain task might require only one worker at a time to be exposed.

Activity concentration of volume contaminated material

The activity concentration as determined from analysis of the sediment inside the evaporator will be used in all calculations, assuming a uniform distribution inside the evaporator as a result of the operations inside the evaporator.

Activity concentration of surface contaminated material

The results of the measured removable surface contaminated levels will be used for the calculations. The alpha contamination levels are very low and will be excluded as its contribution is considered to be less than 0.1% of the total activity.

The radionuclide present in the beta surface contamination are assumed to be similar relative radionuclide contribution to that of the scale analysed since no other radionuclides is expected in the room and contamination is spread as a result of breaches in the containment of the evaporator during typical maintenance tasks. Therefore, the surface contamination will be scaled according to the relative radionuclide contribution, as shown in Table 3.6.

Studies on surface phenomena associated with the particle size distribution of the materials has shown that many factors can influence the actual inhaled or ingested activity, such as the type of material, its physical and chemical properties, and the origin and possible later redistribution of the contamination. For this assessment, the generic approach is deemed appropriate and no additional factoring for these phenomena will be applied.

Table 3.6: Scaled surface contamination activity

Radionuclide	**Scaled beta surface activity (kBq/m^2)**
^{60}Co	4.67
^{95}Nb	187
^{95}Zr	106
^{103}Ru	97.7
^{106}Ru	16.6
^{125}Sb	12.8
^{137}Cs	0.72
^{140}Ba	1.10
^{140}La	1.61
^{141}Ce	1.19
^{144}Ce	31.4
Total	**461**

Breathing rate of workers

The gender averaged breathing rate for workers of 1.1 m^3h^{-1} was used [ICR09].

Committed effective dose conversion factors

The dose conversion factors published in the Basic Safety Standards [IAE96] were used for the radionuclides identified in the analysis. For inhalation calculations, the factors for 5 μm AMAD (activity median aerodynamic diameter) were used for the default lung retention class recommended by the ICRP [ICR96] and given in [IAE96].

The selected dose conversion factors are summarised in Table 3.7.

Table 3.7: Dose conversion coefficients for identified radionuclides

Radionuclide	**Lung retention class**	**Inhalation dose conversion factor**	**Ingestion dose conversion factor**
		[Sv/Bq]	**[Sv/Bq]**
^{60}Co	M	7.1×10^{-9}	3.4×10^{-9}
^{95}Nb	M	1.3×10^{-9}	5.8×10^{-10}
^{95}Zr	M	3.6×10^{-9}	8.8×10^{-10}
^{103}Ru	M	1.9×10^{-9}	7.3×10^{-10}
^{106}Ru	M	1.7×10^{-8}	7.0×10^{-9}
^{125}Sb	M	3.3×10^{-9}	1.1×10^{-9}
^{137}Cs	F	6.7×10^{-9}	1.3×10^{-8}
^{140}Ba	M	1.6×10^{-9}	2.5×10^{-9}
^{140}La	M	1.5×10^{-9}	2.0×10^{-9}
^{141}Ce	M	2.7×10^{-9}	7.1×10^{-10}
^{144}Ce	M	2.3×10^{-8}	5.2×10^{-9}

Concentration of respirable dust in air

This concentration of respirable dust in air (also known as dust load) is dependent upon the type of work to be performed. A value of 1 x 10^{-3} gm^{-3} is recommended by the IAEA [IAE92] for building renovation actions.

Resuspension factor for surface activity

The phenomena of resuspension from surface activity have been widely studied and several reports are available from literature.

The resuspension factor is defined as the quantitative relationship between the concentration of loose surface contamination and consequent atmospheric concentration above the contaminated surface [CEM09]. This handbook also notes the variation in the resuspension factor to be between 10^{-4} and 10^{-8} m^{-1} and recommends 1 x 10^{-6} m^{-1} as a 'reasonable' value.

As early as 1953 a report by the United States Atomic Energy Commission [BAI53] illustrated the variability in the factors, with ranges from 2.5 x 10^{-5} to 1.9 x 10^{-4} m^{-1} for long continuous operations and a maximum activity of 2.0 x 10^{-3} m^{-1} subject to fairly extreme operational conditions for short periods. As more studies were performed in later years and more knowledge gained, later publications showed significantly reduced factors.

The IAEA has several documents where resuspension is encountered. In one document dating back to 1992, a value of 1 x 10^{-6} m^{-1} is used for typical decommissioning scenarios [IAE92].

In a more recent IAEA publication from 2008, the Advisory Material for the Transport Regulations [IAE08], reference is still made to research reports spanning back to 1970 and makes a recommendation of 5 x 10^{-5} m^{-1} as the value "recommended for general use by the IAEA" [IAE08]. This value is subsequently also used in this work and in the IAEA publication [IAE08] to justify the derivation of surface contamination limits in the IAEA Transport Regulations [IAE09a].

In the IAEA Tecdoc 1616 [IAE09b], an article by F. Jourdin is published where a study was performed on resuspension for a range of radionuclides and conditions which included in-situ experiments. This article concludes with statements on the very high variability of measured resuspension factors and low accuracies of model predictions.

Another source is the DOE Handbook [DOE94] where resuspension scenarios for a variety of materials in different physical forms are reported. This report recommends a resuspension factor of 4×10^{-5} m^{-1} for solids in powder form.

In 2001 Necsa reported to the NNR on data collected during local decommissioning activities over a period of time [AA5524]. The average calculated resuspension factor was 1.7×10^{-6} m^{-1} which was rounded to 2.0×10^{-6} m^{-1}. This value was accepted by the NNR and is currently still in use for safety assessment purposes.

Given the variability of the resuspension factors and the Necsa specific scenarios encountered, for this study a resuspension factor of 2.0×10^{-6} m^{-1} is used.

Transfer factor for the inhalation of surface activity

This is the fraction of surface contamination available for resuspension and is a dimensionless value. Although the IAEA recommends 1×10^{-6} in [IAE92], Necsa has agreed with the NNR to use a value of 1 [AA5524]. At the time it was reasoned that the calculations considered removable surface contamination present on the surface of which 100% is available for resuspension.

For this study, a transfer factor of 1 will be used.

Rate of secondary ingestion of removable surface contamination

The IAEA makes the assumption that adult workers will ingest 10 mg of contamination per hour [IAE92].

It furthermore recommends the use of a secondary ingestion rate of 1×10^{-4} m^2h^{-1} for the ingestion of removable surface contamination [IAE92] based on a study performed by Healy in 1971 [HEA71].

Transfer factor for ingestion of surface activity

This dimensionless factor is given as 1 x 10^{-2} by the IAEA in [IAE92].

The internal doses to the workers were calculated using Eqs. (3.4) and (3.5) and the information provided in the preceding sections.

3.6.3 Results

Results of the calculation of the internal dose to workers are summarised in Table 3.8. The results are presented graphically in Figure 3.6 and Figure 3.7. From the graphic in Figure 3.6 the following observations are made:

- the total internal dose contributions from the radionuclides with the highest activity concentration, namely ^{95}Nb (16.7%), ^{95}Zr (16.8%) and ^{103}Ru (18.2%), is responsible for almost half of the internal exposure,
- the relative contribution from ^{144}Ce, at 30.1%, is the largest single radionuclides contribution which is as a result of both the inhalation and ingestion dose conversion factors for ^{144}Ce being the highest of the nuclides analysed and
- ^{106}Ru has a significant contribution of 18%.

From Figure 3.7, the following observations can be made:

- the inhalation dose is higher than the ingestion dose by a factor of 6.99 for the workers in scenarios 1 and 3 and
- the ingestion dose is higher than inhalation dose by a factor of 2.86 for workers in scenario 2.

The latter is due to the assumption that workers in scenario 1 will be exposed to airborne sediment during aggressive cutting actions.

Table 3.8: Results of internal dose calculations

Scenario	Radionuclides	Inhalation dose (Sv)	Ingestion dose (Sv)	Total committed internal dose (Sv)
1	^{60}Co	1.17×10^{-9}	2.54×10^{-10}	1.42×10^{-9}
	^{95}Nb	8.56×10^{-9}	1.74×10^{-9}	1.03×10^{-8}
	^{95}Zr	1.35×10^{-8}	1.50×10^{-9}	1.50×10^{-8}
	^{103}Ru	6.54×10^{-9}	1.14×10^{-9}	7.68×10^{-9}
	^{106}Ru	9.92×10^{-9}	1.86×10^{-9}	1.18×10^{-8}
	^{125}Sb	1.48×10^{-9}	2.24×10^{-10}	1.71×10^{-9}
	^{137}Cs	1.70×10^{-10}	1.50×10^{-10}	3.21×10^{-10}
	^{140}Ba	6.22×10^{-11}	4.42×10^{-11}	1.06×10^{-10}
	^{140}La	8.53×10^{-11}	5.17×10^{-11}	1.37×10^{-10}
	^{141}Ce	1.13×10^{-10}	1.35×10^{-11}	1.27×10^{-10}
	^{144}Ce	2.55×10^{-8}	2.62×10^{-9}	2.81×10^{-8}
	Total	**6.70×10^{-8}**	**9.58×10^{-9}**	**7.66×10^{-8}**
2	^{60}Co	2.58×10^{-7}	1.12×10^{-6}	1.38×10^{-6}
	^{95}Nb	1.89×10^{-6}	7.66×10^{-6}	9.55×10^{-6}
	^{95}Zr	2.97×10^{-6}	6.60×10^{-6}	9.57×10^{-6}
	^{103}Ru	1.44×10^{-6}	5.04×10^{-6}	6.48×10^{-6}
	^{106}Ru	2.19×10^{-6}	8.19×10^{-6}	1.04×10^{-5}
	^{125}Sb	3.27×10^{-7}	9.90×10^{-7}	1.32×10^{-6}
	^{137}Cs	3.76×10^{-8}	6.63×10^{-7}	7.01×10^{-7}
	^{140}Ba	1.37×10^{-8}	1.95×10^{-7}	2.09×10^{-7}
	^{140}La	1.88×10^{-8}	2.28×10^{-7}	2.47×10^{-7}
	^{141}Ce	2.50×10^{-8}	5.96×10^{-8}	8.46×10^{-8}
	^{144}Ce	5.62×10^{-6}	1.15×10^{-5}	1.72×10^{-5}
	Total	**1.48×10^{-5}**	**4.23×10^{-5}**	**5.71×10^{-5}**
3	^{60}Co	1.02×10^{-9}	2.22×10^{-10}	1.24×10^{-9}
	^{95}Nb	7.49×10^{-9}	1.52×10^{-9}	9.00×10^{-9}
	^{95}Zr	1.18×10^{-8}	1.31×10^{-9}	1.31×10^{-8}
	^{103}Ru	5.72×10^{-9}	9.99×10^{-10}	6.72×10^{-9}
	^{106}Ru	8.68×10^{-9}	1.62×10^{-9}	1.03×10^{-8}
	^{125}Sb	1.30×10^{-9}	1.96×10^{-10}	1.49×10^{-9}
	^{137}Cs	1.49×10^{-10}	1.31×10^{-10}	2.81×10^{-10}
	^{140}Ba	5.44×10^{-11}	3.87×10^{-11}	9.31×10^{-11}
	^{140}La	7.46×10^{-11}	4.52×10^{-11}	1.20×10^{-10}
	^{141}Ce	9.89×10^{-11}	1.18×10^{-11}	1.11×10^{-10}
	^{144}Ce	2.23×10^{-8}	2.29×10^{-9}	2.46×10^{-8}
	Total	**5.86×10^{-8}**	**8.38×10^{-9}**	**6.70×10^{-8}**

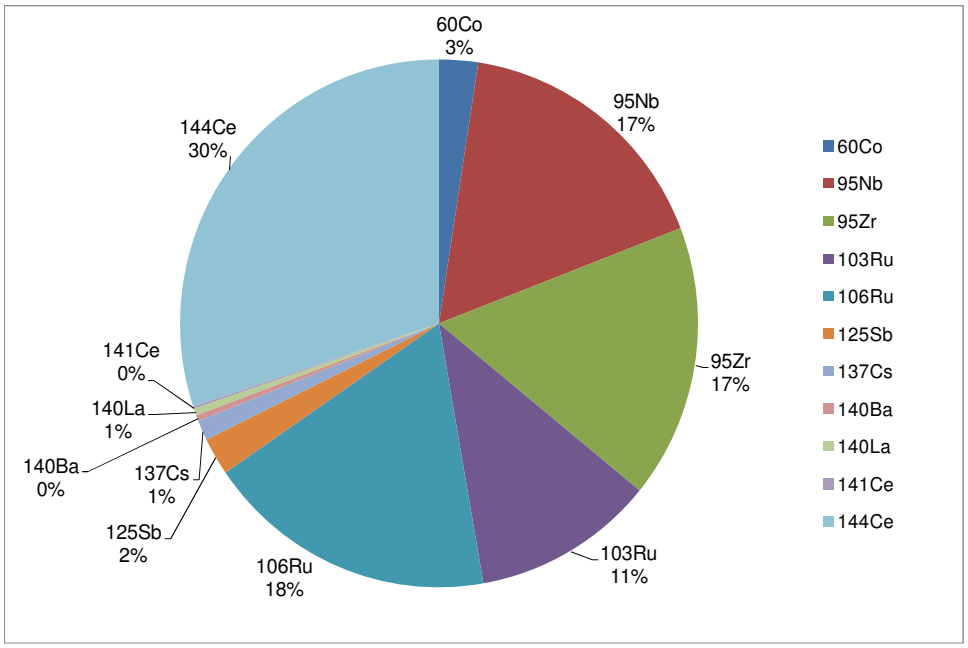

Figure 3.6: Graphic showing inhalation and ingestion dose per radionuclide as a percentage of the total committed effective dose.

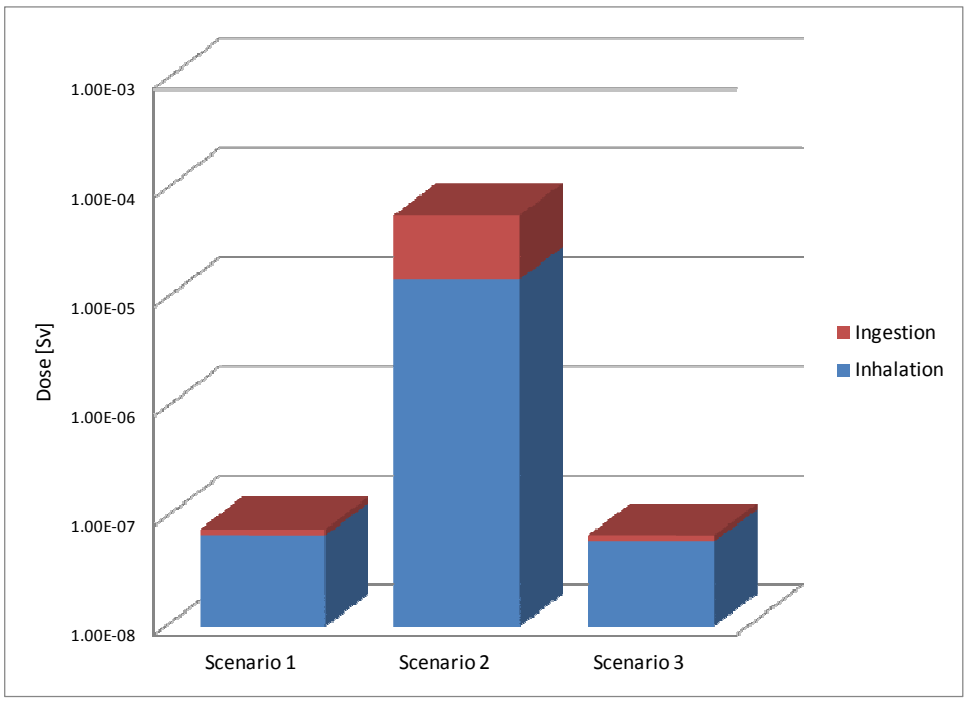

Figure 3.7 : Graphic showing internal dose per worker category.

3.7 Total effective dose

The total effective dose is calculated using Eq. (3.1). The results are provided in Table 3.4 and Table 3.8 for external and internal exposure respectively and is summarised in Table 3.9.

Table 3.9: Results of total effective dose calculations

Scenario	Personal dose equivalent (Sv)	Committed effective dose (Sv)	Total effective dose (Sv)
1	3.38×10^{-3}	7.66×10^{-8}	3.38×10^{-3}
2	6.33×10^{-4}	5.71×10^{-5}	6.90×10^{-4}
3	2.95×10^{-3}	6.70×10^{-8}	2.95×10^{-3}

For scenarios 2 and 3, the committed effective dose, from internal exposure to the worker, is insignificant compared to the external exposure. For scenario 1, the internal dose represents 8.3% of the total exposure to the workers as a result of the potential exposure to airborne sediment.

3.8 Optimisation of Exposure

The Necsa approach to optimisation, discussed in Section 1.2.8, relies on two aspects, namely conformance to the ALARA goal and demonstration of sufficient optimisation.

The ALARA goal, as defined in Section 2.3 for an individual has not been met since the projected doses in Section 0 exceed 2.5 mSv for any of the identified worker categories.

Optimisation options already considered for this task included:

- a need for a formal ALARA planning and review since the estimated individual dose exceeds 1 mSv,
- the use of a dose constraint in the form of the ALARA objective which is lower than the dose limit,
- the washdown of the facility prior to work being performed which resulted in decontamination of work surfaces to reduce the potential for internal exposure,
- the execution of an *a priori* dose assessment (this research report) to enable decision making in terms of exposure levels and protective measures and
- the use of limited personal protective equipment and clothing.

The latter was not considered in the dose assessment and will result in less hindrance to the worker and less time to dress/undress. This option is further discussed in Section 3.8.1.

3.8.1 Further optimisation options to reduce internal exposure

The use of personal protection clothing and equipment will result in reduced doses to workers in contaminated environments due to increased control over the inadvertent ingestion and inhalation of contaminated materials. This is not simple to quantify directly since many variables can influence the level of protection, such as the work procedures followed and the differences between methodologies followed by different workers. One measure which can result in significant reduction in the internal exposure, is the use of respiratory protection. A respirator can provide a protection factor of 10 for inhalation while an air hood can provide a protection factor of 1000 for inhalation, as published in the American National Standards as assigned protection factors [ANS92]. An air hood is typically a half body suit which supplies external fresh air to the worker thereby eliminating the inhalation of contaminated air from the work place.

However, by not using protective clothing also minimised the time required to perform the task in that less time is spent dressing and undressing; less

physiological stress is placed on the body, less restriction on movement is experienced and worker efficiency is improved [BEV10].

Although the dose assessment only considered the physical aspects of the task to be performed, several operational radiation protection practices will ensure the doses to the workers will be controlled and even reduced during performance of the work. Examples of these are:

- covering work area with plastic to further reduce spread of contamination and contamination of tools,
- waste minimisation practices and
- good housekeeping to reduce movement and expedite cleanup after work is completed.

The use of personal protective clothing should also be put into perspective. For the three scenarios in this study, the use of protective clothing would not benefit the scenarios where the internal exposure is already very low.

3.8.2 Further optimisation options to reduce external exposure

The calculated doses are largely as a result of external exposure, thus significant measures to reduce exposure time (working faster, task training) and reducing exposure (positioning relative to high exposure rates) can be considered.

The most effective way to reduce the external exposure to workers is to consider traditional time, distance, shielding and planning approach. The workspace is restricted in terms of space, which makes changes in terms of distance from the source of radiation and the use of shielding not viable. The most plausible solution is to improve on time through planning of work execution. A proposal was accepted to reduce the allocated period for workers in Scenario 1 and 3 by 30% to 11.2 hours and 9.8 hours respectively. When the doses are calculated, result in meeting the ALARA goal, as reflected in Table 3.9. Worker 2 was not affected as the projected time to perform these tasks is of short duration.

The measures to achieve this reduction in time are based upon improved planning:

- extensive training of all workers in the procedures,
- training on mock-ups to practice reduction of the time taken to complete the tasks and to confirm task duration,
- identification of problem areas and
- the use of electronic dosimetry to perform live tracking of doses which will enable radiation protection personnel to continuously evaluate the progress of the tasks against the doses incurred and to proactively initiate control measures or invigilation.

The latter measure introduces a recent technological advancement, which enables the remote tracking of exposures via a radiofrequency link between the electronic personal dosimeter worn by the worker and a computer located in another room. This is a very useful tool for tracking; however, cognisance should be taken of the differences experienced between doses recorded with thermoluminescence and electronic dosimeters, noting that the thermoluminescence dosimeter is still regarded as the official means of dose recording in South Africa. In a paper by Vanhavere [VAN01] based on studies performed at the SCK•CEN in Belgium, the electronic dosimeter recorded lower doses. In a more recent publication by Poston [POS05], a switch to electronic dosimeters in the near future is however projected.

Table 3.10: Results of total effective dose calculations after implementation of optimisation considerations

Scenario	**Personal dose equivalent (Sv)**	**Committed effective dose (Sv)**	**Total effective dose (Sv)**
1	2.37×10^{-3}	7.66×10^{-8}	2.37×10^{-3}
2	6.33×10^{-4}	5.71×10^{-5}	6.90×10^{-4}
3	2.07×10^{-3}	6.70×10^{-8}	2.07×10^{-3}

3.8.3 Recommendations for radiation protection programme

The last steps in the iterative evaluation in Section 1.3.2 is the completion of the dose assessment and work permit as well as to finalise the work plan and procedures where all the outcomes of the assessment culminate in the recommendations for execution of the task and the radiation protection programme.

CHAPTER 4

CONCLUSIONS AND RECOMMENDATIONS

"Planning is an unnatural process. It is much more fun to do something. The nicest thing about not planning is that the failure comes as a complete surprise rather than preceded by a period of worry and depression" – Sir John Harvey-Jones

The conclusions and recommendations resulting from this study are presented in this chapter.

4.1 Conclusions from calculations

The results of the dose assessment presented in Chapter 3 and summarised in shows the highest total effective dose to 2.37 mSv which is within the national and international dose limits for workers and meets the Necsa ALARA goal of 4 mSv.

The dose assessment was supported by measurement of actual radiological conditions in the area where the task will be performed using suitable and calibrated measuring instruments. The benefit of this was that the assumption needed to perform the dose assessment was limited to the physical phenomena associated with the behaviour of materials and available from national and international studies.

Another aspect which provides confidence in the results is the exposure scenarios which were defined during a proper planning of all the tasks associated with the replacement of the evaporator coil.

Sources of inaccuracy and uncertainty associated with the calculated doses were discussed in Chapter 3 and are limited to uncertainty in the measurement data and assumptions. This should be objectively viewed in light of the large variations in accuracy expected in personnel monitoring. Poston [POS05] reports accuracy in the order of -50% to +100% for routine monitoring in general.

It is worth noting that the calculations perform in this research report are based on international methods and formulae. However, the assumptions used are based on scenarios which are not always applicable or an accurate reflection of the actual scenario, although considerable effort was made to obtain realistic values. Subsequently, the results obtained include conservatism. The post task review will afford the RP professional the opportunity to compare the outcome of the personnel monitoring with the projected outcome, taking into consideration the alignment between the planned work plan and the actual work plan.

4.2 Conclusions on ALARA

In Chapters 1 through Chapter 3, much emphasis has been placed on the requirements, controls and implementation of ALARA. The Fundamental Safety Principles on ALARA, which were discussed in Section 1.2.1 provided the basis for evaluating the outcomes of the task to replace the evaporator coil:

Principle 4: Justification of facilities and activities

The evaporator facility treats medium activity effluent as a service to SAFARI-1 and NTP. The radiological exposures involved in the replacement of the evaporator coil are justified to ensure the continued operation of these facilities, which produce inter alia medical isotopes.

Principle 5: Optimisation of protection

Optimisation has been demonstrated in Section 3.8. The exposures to workers were calculated in an *a priori* dose assessment. The doses conform to the ALARA objectives of Necsa and other operation optimisation measures. Other dose reduction options, such as personal protective clothing and equipment, were considered.

Principle 6: Limitation of risks to individuals

An *a priori* dose assessment illustrated that doses will be within dose limits for workers and within ALARA constraints set internally by Necsa.

The implementation of several engineering and administrative controls resulted in the calculation of doses which are considered ALARA.

Continued implementation of ALARA

The methodology used to perform the dose assessment in this study provides an illustration of the effective use of many aspects of radiation protection to enable the licensee to demonstrate ALARA in its operations.

The success of continued ALARA efforts is widely reported. In the United States, continuous improvement in the ALARA programmes over the past 25 years has reduced the average annual measurable dose per occupational worker at commercial nuclear power reactors and other facilities from 6.6 mSv to 1.4 mSv [BLE11], more than a four-fold decrease. This is illustrated in Figure 4.1.

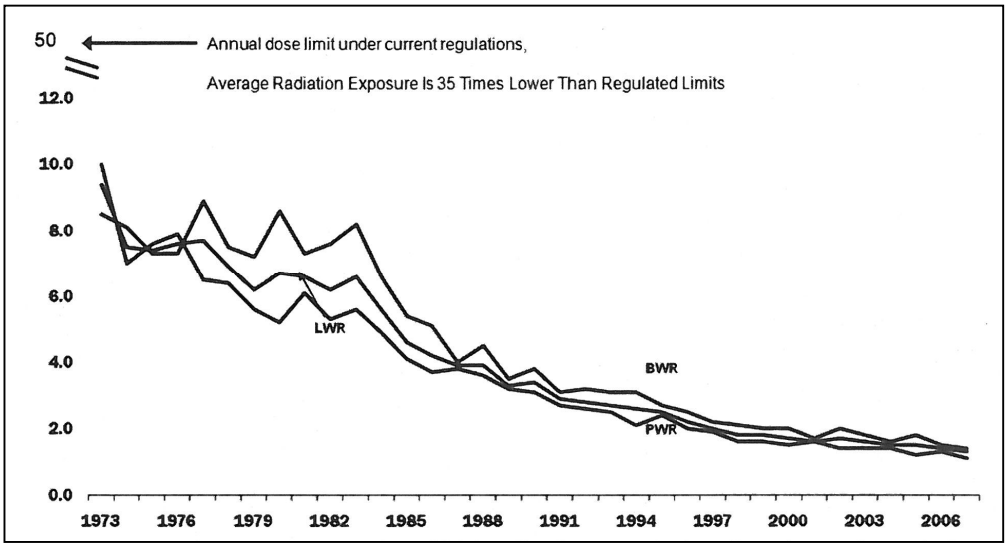

Figure 4.1: Average measurable dose per occupational worker at commercial nuclear power reactors and other facilities in the USA, 1973-2006 in mSv. Taken from [BLE11].

REFERENCES

[AIT05] S. B. Aitken et al, "High Level Waste Tank Closure Project: ALARA Applications at the Idaho Engineering and Environmental Laboratory", Health Physics Journal, (2005).

[ANS92] American National Standard Institute, "American National Standard for Respiratory Protection", ANSI Z88.2-1992 (1992).

[BAI53] J.C. Bailey and R.C. Rohr, "Air-borne contamination resulting from transferable contamination on surfaces", United States Atomic Energy Commission, (1953).

[BLE11] M. R. Blevins and R. Andersen, "Radiation Protection at U.S. Nuclear Power Plants – Today and Tomorrow", (2011).

[BAU06] E. Baumann and I. Terry, "The EPR: A clear step forward in dose reduction and radiation protection", (2006).

[BEV10] J.J. Bevelacqua, "Practical and effective ALARA", (2010).

[CAM83] D.A. Campbell et al, "Evaluation of the use of Zeolite mixtures in the submerged demineraliser system", Proceedings of the International Zeolite Conference, Reno, Nevada, (1983).

[CEM09] H. Cember, "Introduction to Health Physics", 4th Edition, (2009).

[DJS96] D.J. Strom, "Ten Principles and Ten Commandments of Radiation Protection", (1996).

[DOE94] DOE Handbook, "Airborne Release Fractions/Rates and Respirable Fractions for Non-reactor Nuclear Facilities", DOE-HDBK-3010-94, (1994).

[EDM64] L.F. Edmondson, "Ion exchange processes for removing radioactive contamination from milk", (1964).

[FRA05] P.W. Frame, “A history of radiation detection instrumentation”, (2005).

[FLE68] C.R. Fletcher, “The radiological hazards of Zirconium-95 and Niobium-95”, (1968).

[HEA71] J.W. Healy, “Surface Contamination: Decision Levels”, (1971).

[IAE02] International Atomic Energy Association, “Application of Ion Exchange Processes for the treatment of Radioactive Waste and Management of Spent Ion Exchangers”, IAEA Technical Report Series No. 408, (2002).

[IAE04] International Atomic Energy Association, “Methods for Assessment of Radiation Doses due to Intakes of Radionuclides”, IAEA Safety Report Series No. 37, (2004).

[IAE05] International Atomic Energy Association, “Derivation of Activity Concentration Levels for Exclusion, Exemption and Clearance”, IAEA Safety Report Series No. 44, (2005).

[IAE06] International Atomic Energy Association, “Fundamental Safety Principles”, IAEA Safety Standards No. SF-1, (2006).

[IAE07] International Atomic Energy Association, “Terminology Used in Nuclear Safety and Radiation Protection”, IAEA Safety Glossary, (2007).

[IAE08] International Atomic Energy Association, “Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material”, IAEA Safety Standard No. TS-G-1.1, (2008).

[IAE09a] International Atomic Energy Association, “Regulations for the Safe Transport of Radioactive Material”, IAEA Safety Standards No. TS-R-1, (2009).

[IAE09b] International Atomic Energy Association, “Quantification of Radionuclide Transfer in Terrestrial and Freshwater Environments for Radiological Assessments”, IAEA TECDOC 1616, (2009).

[IAE11] International Atomic Energy Association, “Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards”, IAEA Safety Standards No. GSR Part 3 (Interim), (2011).

[IAE83] International Atomic Energy Association, “Conditioning of Low- and Intermediate-Level Radioactive Wastes”, IAEA Technical Report Series No 222, (1983).

[IAE84] International Atomic Energy Association, “Treatment of Low- and Intermediate-Level Liquid Radioactive Wastes”, IAEA Technical Report Series No 236, (1984).

[IAE85] International Atomic Energy Association, “Inorganic ion exchangers and absorbents for chemical processing in the nuclear fuel cycle”, IAEA TECDOC 337, (1985).

[IAE92] International Atomic Energy Association, “Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities”, IAEA Safety Series No. 111-P-1.1, (1992).

[IAE96] International Atomic Energy Association, “International Basic Safety Standards for Protection against Ionising Radiation and for the Safety of Radiation Sources”, IAEA Safety Series No. 115, (1996).

[IAE99a] International Atomic Energy Association, “Occupational Radiation Protection”, IAEA Safety Guide No. RS-G-1.1, (1999).

[IAE99b] International Atomic Energy Association, “Assessment of Doses to the Public from the Ingested Radionuclides”, IAEA Safety Report Series No. 14, (1999).

[IAE99c] International Atomic Energy Association, “Assessment of Occupational Exposure Due to Intake of Radionuclides”, IAEA Safety Standard Guide No. RS-G-1.2, (1999).

[ICR07] International Commission on Radiological Protection, “The 2007 Recommendations of the International Commission on Radiological Protection”, ICRP Publication 103, (2007).

[ICR93] International Commission on Radiation Units and Measurements, “Quantities and Units in Radiation Dosimetry”, ICRU Report No. 51, (1993).

[ICR96] International Commission on Radiological Protection, ”Age-dependent Doses to Members of the Public from Intakes of Radionuclides, Part 5. Compilation of Inhalation and Ingestion Dose Coefficients”, ICRP Publication 72, (1996).

[ICR08] International Commission on Radiological Protection, “Nuclear Decay Data for Dosimetric Calculations”, ICRP Publication 107, 2008).

[IEC02] International Electrotechnical Commission, “Radiation protection instrumentation - Alpha, beta and alpha/beta (beta energy >60 keV) contamination meters and monitors”, IEC 60325 Third edition, (2002), International Standard.

[IEC09] International Electrotechnical Commission, “Radiation Protection Instrumentation – Ambient and/or directional dose equivalent (rate) meters and/or monitors for beta, X and gamma radiation – Part 1: Portable workplace and environmental meters and monitors”, IEC 60846-1 Ed 1.0, (2009), International Standard.

[INT08] Thermo Fisher Scientific Inc., “InterceptorTM User Manual”, Rev 4.8, (2008).

[ISO88] International Standards Organisation, “Evaluation of surface contamination – Part 1: Beta-emitters (maximum beta energy greater than 0,15 MeV) and alpha-emitters”, (1988) (ISO 7503-1).

[ISO99] International Standards Organisation, “X- and gamma reference radiation for calibrating dosemeters and doserate meters and for determining their response as a function of photon energy - Part 3 : Calibration of area and personal dosemeters and the measurement of their response as a function of energy and angle of incidence”, (1999) (ISO 4037-3).

[KLE92] R. C. Klein, I. Linins, E. L. Gershey, “Detecting Removable Surface Contamination”, (1992).

[KNO10] G. F. Knoll, “Radiation Detection and Measurement”, (2010).

[LAB07] T. Labarta, “Aspects of operational radiation protection during the dismantling of nuclear facilities relevant for the estimation of internal doses”, (2007).

[LAZ11] E. Lazo, “New nuclear build and evolving radiation protection challenges”, (2011).

[NAC56] F.C. Nachod et al, “Ion Exchange Techology”, Academic Press Inc, (1956).

[NAT99a] National Nuclear Regulator Act, "Regulations in terms of Section 36, read in conjunction with Section 47 of the National Nuclear Regulator Act, on Safety Standards and Regulatory Practices", (1999) (Act No. 47 of 1999).

[NAT99b] National Nuclear Regulator Act, Republic of South Africa, (1999), (Act No. 47 of 1999).

[NAV89] J.D. Navratil, "Ion exchange technology in spent fuel reprocessing", (1989).

[NEW79] A.R. Newby-Fraser, "Kettingreaksie", (1979).

[OEC05] Nuclear Energy Agency, "Optimisation in Operational Radiological Protection", NEA Report No. 5411 (2005) (ISBN 92-64-01050-5).

[OEC11] Nuclear Energy Agency, "Occupational Radiological Principles and Criteria for Designing New Nuclear Power Plants", NEA Report No. 6975 (2011) (ISBN 92-64-01050-5).

[OHW67] K. Ohwada, "Removal of fission products and their complexing agents from degraded solvent by ion exchange method", (1967).

[POS05] J. Poston, "External Dosimetry and Personnel Monitoring", (2005).

[RAJ06] K. Raj, "Radioactive waste management practices in India", Nuclear Engineering and Design, (2006).

[ROL06] J. Rolph, "Effect of Geotropism on Instrument Readings", (2006).

[SIN97] P.K. Sinha et al, "Removal of radioiodine from liquid effluents", (1997).

[THI01] K.S. Thind, "Monitoring methods and dose assessment for internal exposures involving mixed fission and activation products containing actinides", (2001).

[VAN01] F. Vanhavere and M. Coeck, "Comparison between Thermoluminescence and Electronic Dosimetry Results at the Belgian Nuclear Research Centre", (2001).

[VER05] F. Vermeersch, "ALARA Pre-Job Studies Using the Visiplan 3D ALARA Planning Tool", (2005).

[VER07] F. Vermeersch, "3D planning tools and their applicability to occupational exposure at a fusion power plant", Restricted contract report under the European Fusion Development Agreement (SCK•CEN-R-4541) (2007).

[WAGGO] L. Waggoner and J. Kurtz, "Dose Reduction Techniques", (Date unknown).

NECSA INTERNAL DOCUMENTS

[AA5524] Necsa communication to the NNR by G.R. Liebenberg, 2001, "Calculation of Resuspension Factors for Decommissioning Projects", AA5524

[NEC01] SHEQ-INS-8080 Rev 6, 2010, "Radiation Dose Limitation"

[NEC02] SHEQ-INS-8120 Rev 7, 2010, "Safe Work Procedures and Radiation Protection Work Permit System"

[NEC03] SHEQ-INS-8180 Rev 5, 2010, "ALARA Programme"

[NEC04] SHEQ-INS-8260 Rev 4, 2010, "Management of radioactive effluent and discharges at the Pelindaba site"

[LEM13] Internal e-mail correspondence with LEMS Department manager on 2013-01-29

APPENDIX A – Analysis Report

RadioAnalysis
Building 1600
P O Box 582
Pretoria 0001

Telephone: + 27 12 305 5243
Facsimile: + 27 12 305 5944

necsa
We're in your world
NUCLEAR SERVICES

Contact: ***Mr W Blofield***
Company: ***LEMS Operations***
Address: ***Building P2400***
Pelindaba-West
Necsa

Date: ***24 February 2009***
Report number: ***Final***
Pages: ***2***
Our reference: ***RA-09423***
Your reference:

Analysis Report

Radioactivity analysis of a powder

Compiler: ***D Kotze***
Technical signatories: ***J Smit, M Raven***

The views and opinions of authors expressed in this report do not necessarily state or reflect those of Necsa. The liability of Necsa is limited to the "General Conditions of Sale", which is available on request.

1. SERVICE

Gamma analysis of a high active powder sample.

2. SAMPLE PREPARATION AND ANALYSIS

Product	Description	Procedure	Completed	Assayer
050111	Gamma analysis: Identify and quantify all lines	WIN-101	23/08/2008	D Kotze

- A powder sample inside a sealed injection vial was submitted for analysis on 20 August 2008.
- The mass of the sample was determined by difference in weight of the vial with and without the powder.
- The radioactivity of the powder was measured with a calibrated HPGe detector (RA-FS-102) at a distance of approximately 50 cm from the face of the detector.

3. RESULTS

The table below lists the nuclides identified and corresponding activities. Reference date for radioactivity is 22 August 2008.

Nuclide	Activity (MBq/kg)
^{60}Co	11
^{95}Nb	**440**
^{95}Zr	**250**
^{103}Ru	**230**
^{106}Ru	39
^{125}Sb	30
^{137}Cs	1.7
^{140}Ba	2.6
^{140}La	3.8
^{141}Ce	2.8
^{144}Ce	74

.

4. QUALITY ASSURANCE

4.1 RadioAnalysis is a SANAS accredited laboratory (Testing Laboratory T0111) based on ISO/IEC Standard 17025. All analytical methods are documented in the RadioAnalysis Quality System.

4.2 This report is a compilation of results obtained from different test reports individually signed by the SANAS Technical Signatories for the respective methods. The compiler is the Technical Expert for all the methods. The individual test reports are available upon request.

4.3 The RadioAnalysis Laboratory keeps the original signed hard copy of this report on record for three years.