University of Southern Queensland Faculty of Engineering & Surveying

Nuclear Instruments in Mineral Processing

A dissertation submitted by

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towards the degree of

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Abstract

This research project was conceived out of the author's own self-interest in seeking to answer the following:

"Why use instruments that utilize ionising radiation as part of the measurement system?"

This argument was heavily weighted by the author's own safety concerns. Subsequently the research parameters within this document were set on ascertaining the utility and the appropriateness of instruments that use ionising radiation as part of the measurement system.

Instruments utilizing ionising radiation as part of the measurement system are typically labelled as "nuclear instruments". This labelling at times is technically incorrect as not all forms of ionising radiation are generated from the nucleus; notably this generalisation has been continued throughout this document.

Importantly research has found that the use of ionising radiation as part of a measurement system in a mineral processing application is a safe practice.

Nuclear instruments provide reliable measures for often difficult applications where few genuine alternatives exist. These devices have significantly enhanced production and reduced costs. Taking all things into consideration it may be argued that this technology is under-utilized. The costs associated with regulatory compliance were found not to be oppressive to large scale operations. However the extra burden of regulatory compliance on smaller scale operations may pose a significant deterrent to the implementation of these measurement systems.

Alternatives were found for the process measurements of in-pipe slurry density and conveyor mass flow. For both process measurements the alternatives are believed to provide better accuracies, thereby questioning the appropriateness of the use of nuclear instruments in these applications.

The major parts of this dissertation cover;

- the fundamentals of radiation and its measurement,
- radiation safety and regulation,
- measurement applications,
- alternatives,
- recommendations, and
- the author's conclusions.

The benefits bestowed by the use of nuclear instruments throughout industry are impressive; greatly enhanced production at reduced cost, with the addition of close to real-time assays. Arguments against the use of nuclear instruments appear superfluous to the author, as efficient production on a large scale can not be envisaged without inclusion. University of Southern Queensland Faculty of Engineering and Surveying

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I further certify that the work is original and has not been previously submitted for assessment in any other course or institution, except where specifically stated.

TIMOTHY WILLIGEN

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Signature

Date

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First and foremost I would like to sincerely thank my wife Michelle, and my family for their patience and understanding whilst I undertook this task. I wish to also thank Dr Andrew Maxwell, who as my project supervisor gave me the confidence and encouragement when I needed it most. In addition I would like to express my thanks to the staff of Newmont Mining for their assistance in completing my research project.

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University of Southern Queensland October 2012

Contents

Abstract	i		
Acknowledgments	v		
List of Figures	xiii		
List of Tables	xvii		
Nomenclature	xix		
Chapter 1 Introduction 1			
1.1 Introduction	1		
1.2 Research aims and objectives	3		
1.3 Dissertation Outline	6		
Chapter 2 Literature Review 9			
2.1 Chapter Overview	9		
2.2 The applicable nuclear physics	10		

CONTENTS

00111		• • • •
2.3	Radiation detection and measurement	12
2.4	Nuclear instruments	14
2.5	Radiation safety, legislation and regulation	16
2.6	Non – nuclear instrument alternatives	18
2.7	Questions raised from the literature review	19
Chapte	er 3 Project Implementation	20
3.1	Chapter Overview	20
3.2	Methodology	21
Chapte	er 4 The Basics of Nuclear Physics	23
4.1	Chapter Overview	23
4.2	Ionising Radiation	23
4.3	Isotopes and Radioactive Decay	24
4.4	Radioactive Decay Mechanisms	32
	4.4.1 Alpha emission	34
	4.4.2 Beta emissions and electron capture	36
	4.4.3 Spontaneous fission	38
	4.4.4 Gamma Emission, Internal Conversion and X-rays	39
4.5	Quantum Numbers	41
4.6	Decay Schemes	43

4.7	Chapter Summary	45			
Chapte	er 5 Radiation's Interaction with Matter	46			
5.1	Chapter Overview	46			
5.2	Penetrating Power	47			
5.3	Interaction with Matter	47			
	5.3.1 Heavy charged particles	48			
	5.3.2 Electrons	51			
	5.3.3 Positrons	54			
	5.3.4 Gamma rays and x-rays	54			
	5.3.5 Neutrons	59			
5.4	Chapter summary	64			
Chapte	Chapter 6 Radiation units, intensity and attenuation				
6.1	Chapter Overview	65			
6.2	Activity	65			
6.3	Intensity	66			
6.4	Exposure rate and Dose Equivalent rate	71			
6.5	Attenuation	73			
	6.5.1 Linear and Mass attenuation coefficient methods	73			
	6.5.2 Half thickness methods	74			

viii

CONT	ENTS	ix
	6.5.3 Transmission tables	74
6.6	Chapter summary	75
Chapt	er 7 Radiation safety, legislation and regulation	76
7.1	Chapter Overview	76
7.2	Radiation Safety	76
7.3	Legislation and regulation	84
7.4	Chapter Summary	86
Chapte	er 8 Radiation sources	87
8.1	Chapter Overview	87
8.2	Naturally occurring radioactive materials	87
8.3	Artificial isotopes	88
8.4	X-rays	91
8.5	Source sizing	94
8.6	Chapter Summary	96
Chapt	er 9 Detectors	97
9.1	Chapter Overview	97
9.2	Geiger Muller tubes	97
9.3	Scintillation Techniques	103
	9.3.1 Scintillators	103

	9.3.2 Light detection	107
9.4	Semiconductor techniques without scintillation	113
9.5	Chapter Summary	116
Chapte	er 10 Measurement	117
10.1	Chapter Overview	117
10.2	Measurement fundamentals	117
	10.2.1 Measurement Model	118
	10.2.2 Statistics of Measurement	119
	10.2.3 Operational effects on error	123
	10.2.4 Signal processing	123
10.3	Chapter Summary	126
Chapte	er 11 Measurement Applications	127
11.1	Chapter Overview	127
11.2	Nuclear Instruments in Mineral Processing	127
	11.2.1 Process Instruments	128
11.3	Process Measurement	129
	11.3.1 Level	129
	11.3.2 Density	133
	11.3.3 Belt-weighers	136

11.4	Analytical Instruments	137
	11.4.1 Concentration by direct radiation measurement	137
	11.4.2 X-ray fluorescence analysis	138
	11.4.3 Neutron Activation Analysis	142
11.5	Chapter Summary	144
Chapte	er 12 Evaluation, performance and alternatives	145
12.1	Chapter Overview	145
12.2	Site review	145
12.3	Evaluation of source type	147
12.4	Cost of ownership	150
12.5	On-pipe density	152
12.6	Nuclear belt-weighers	156
12.7	Chapter Summary	159
		100
Cnapte	er 13 Recommendations	160
13.1	Introduction	160
13.2	Source selection	161
13.3	On-pipe density	162
13.4	Nuclear belt-weighers	163
13.5	Installation Specific	163

CONTENTS	xii
13.6 Maintenance	. 165
13.7 Employee education	. 165
Chapter 14 Conclusions and Further Work	166
14.1 Achievement of Project Objectives	. 166
14.2 Conclusions	. 168
14.3 Further work	. 169
References	170
Appendix A Project Specification	177
Appendix B Dose Limits and Maximum Permissible Exposure Levels	179
Appendix C Australian Regulatory Authorities	183
Appendix D Registration of Radioactive Substances	186

List of Figures

4.1	Uranium–238 decay chain (Department of Mines and Petroleum (2010))	26
4.2	Binding energy curve, plotting binding energy per nucleon against atomic mass. (Massachusetts Institute of Technology $(2006a)$)	30
4.3	Chart of nuclide stability in respect to half-life. (adapted from Nationial Nuclear Data Centre, Brookhaven National Laboratory (2012)) \ldots	31
4.4	Chart of Nuclides in respect to decay type. (adapted from Nationial Nuclear Data Centre, Brookhaven National Laboratory (2012))	33
4.5	Energy level spectrum of ¹² C. (Lawrence Berkeley National Laboratory (2003))	42
4.6	Decay scheme of cobalt-60. (Left - Nationial Nuclear Data Centre, Brookhaven National Laboratory (2012)) (Right - Massachusetts Insti- tute of Technology $(2006c)$)	44
5.1	Penetrating abilities of ionising radiation. (The Australian Journal of Emergency Management, Colella, Logan, McIntosh & Thomson (2005))	47
5.2	Bragg Plot for 1.8 and 4.3 MeV alpha particles in SiC.(Kinoshita, Iwami, ichi Kobayashi, Nakano, Tanaka, Kamiya, Ohi, Ohshima & Fukushima 2005)	51

LIST OF FIGURES

5.3	Transmission curves for beta particles emitted from ${}^{35}S$, ${}^{60}Co$ and ${}^{204}Tl$	
	beek (2007) Delft University of Technology)	53
5.4	Gamma ray interaction processes with respect to photon energy and atomic number Z. (Massachusetts Institute of Technology $(2006b)$)	55
5.5	Compton scattering. (National Aeronautics and Space Administration, Science Mission Di- rectorate. (2010))	56
5.6	Attenuation of gamma and x-rays through air. (Massachusetts Institute of Technology $(2006b)$)	58
5.7	Neutron interactions with respect to energy. (Hamilton (2006)) \ldots	62
6.1	Radiation weighting factors for neutrons vs neutron kinetic energy. (In- ternational Commission on Radiological Protection (2007))	68
6.2	Transmission table for caesium-137 (Omega (n.d.))	75
8.1	Radiation source physical view. (Berthold Technologies 2004) $\ \ldots \ \ldots$	90
8.2	Radiation source sectional view. (Berthold Technologies 2004) $\ . \ . \ .$	90
8.3	Radiation source physical view.(Labspace n.d.)	92
8.4	Operating principle of an x-ray tube. (U.S. Geological Survey 2001)	93
9.1	Geiger Muller tube overview. (Gamma Geiger Counter 2012)	98
9.2	Gas filled detector plot of ion multiplication against applied voltage; (a): 1 MeV β particles; (b): 100 keV β particles. (Podgoršak 2005)	99
9.3	Example of count rate against voltage for a Geiger Muller tube. (University of the Fraser Valley 2012)	102

 \mathbf{xiv}

LIST OF FIGURES

9.4	Scintillator detection principle.	103
9.5	Photomultiplier tube assembly complete with scintillator. (Saint - Gob- ain Crystals 2011)	108
9.6	Operational view of a photomultiplier tube. (University of Illinois 2011)	109
9.7	Overview of a lithium drifted silicon detector. (Northern Arizona University 2006)	7 115
9.8	Full width at half maximum for a Gaussian distribution	115
10.1	Nuclear sensor model applicable to mineral processing environments	118
10.2	Pulse counting circuit with discriminator. (Johansen & Jackson 2004, Fig. 5.9)	125
11.1	Application of point level switches. (Vega Grieshaber KG 2011)	130
11.2	Continuous level measurement arrangements. (Berthold Technologies 2012)	132
11.3	On-pipe density arrangement. (Berthold Technologies 2010)	133
11.4	Nuclear belt weigher general arrangement. (Ohmart/VEGA Corporation 2012)	136
11.5	Uranium discriminator at the Ranger Uranium Mine. (OB rien 2010) $\ .$.	138
11.6	Principle of XRF analysis. (Tawada Scientific 2010)	139
11.7	WDXRF and EDXRF methods for slurry analysis. (Outotec Oyj 2008)	141
11.8	EDXRF spectral plot, count-rate (x 1000) against energy for zinc slurry. (Outotec Oyj 2008)	142

LIST OF FIGURES

11.9 Neutron Activation Analysis spectral plot, count-rate against energy.	
(University of Missouri Research Reactor 2008) $\ldots \ldots \ldots \ldots \ldots$	143
11.10 PGNAA installed on a conveyor. (Thermo Fisher Scientific Inc. 2008) $.$	144
12.1 Boddington belt-weigher comparisons	158

List of Tables

2.1	Literature review of Nuclear Physics	11
2.2	Literature review of Radiation Dection and Measurement	13
2.3	Literature review of Nuclear Instruments	15
2.4	Literature review of Radiation safety, legislation and regulation $\ . \ . \ .$	17
2.5	Literature review of Non-nuclear measurement methods	18
6.1	Inter-relationship between radiation measurement unit descriptors $\ . \ .$	67
6.2	Radiation weighting factors. (International Commission on Radiological Protection (2007))	68
6.3	Radiation measurement units	70
6.4	Exposure rate constants for gamma sources. (Knoll (2010, p. 57) and the Canadian Nuclear Safety Commission (2011))	72
6.5	Dose rate constants for gamma sources. (Unger & Trubey (1982))	72
7.1	Radiation doses in perspective. (Australian Radiation Protection and Nuclear Safety Agency $(2011a)$)	78

LIST OF TABLES

7.2	Health risks associated with low doses. (Australian Radiation Protection and Nuclear Safety Agency $(2011a)$)	78
7.3	Typical values of effective dose for various medical x-rays. (Australian Radiation Protection and Nuclear Safety Agency $(2011a)$)	79
7.4	Shielding materials dependent on radiation type	81
7.5	Radiation dose limits. ((Radiological Council $2012b$))	83
8.1	Artificial isotopes used in mineral processing. (Nationial Nuclear Data Centre, Brookhaven National Laboratory 2012)	89
9.1	Properties of commonly used scintillators. Knoll (2010) and Cooper (1986)1	.06
10.1	Confidence to standard deviation relationship for a Gaussian distribution.1	.20
12.1	Nuclear installations at the Boddington Gold Mine	.46
12.2	Annual individual nuclear installation costs	52
12.3	Density gauge comparisions	.54

Nomenclature

Abbreviations

ALARA	As low as reasonably achievable		
ARPANSA	Australian Radiation Protection and Nuclear Safety Agency		
CSIRO	Commonwealth Scientific and Industrial Research Organisation		
EDXRF	Energy Dispersive X-ray Fluorescence Analysis		
FWHM	Full Width at Half Maximum		
$\mathbf{G}\mathbf{M}$	Geiger Muller		
ICRP	International Commission on Radiological Protection		
NAA	Neutron Activition Analysis		
NASA	National Aeronautics and Space Administration		
NORM	Naturally Occurring Radioactive Material		
PGNAA	Prompt Gamma Neutron Activition Analysis		
RSO	Radiation Safety Officer		
\mathbf{SG}	Specific Gravity		
WDXRF	Wavelength Dispersive X-ray Fluorescence Analysis		
XRF	X-ray Fluorescence		

Units of Measure

$\mathbf{B}\mathbf{q}$	becquerel; SI measurement unit of activity, with one becquerel
	equal to a rate of one disintegration per second
m C/kg	coulomb per kilogram; SI measurement unit of exposure, based on charge per volume
Ci	curie; historical unit of activity: 1 Ci = 3.7 x 10^{10} Bq
${ m eV}$	electron volt; a standard energy measurement used in nuclear physics, one electron volt is equal to the kinetic energy gained by an electron from its acceleration through a potential difference of one volt. 1 eV = $1.602 \times 10^{-19} \text{ J}$
Gy	gray; SI measurement unit of absorbed dose, with one gray equal to the absorption of one joule of energy per kilogram of the absorbing material
R	roentgen; historical unit of exposure: equates to approximately $2.08 \ge 10^9$ ion pairs/cm ³ of air under standard conditions. 1 R = $2.58 \ge 10^{-4}$ C/kg
rad	historical unit of absorbed dose: 1 rad = 1 x 10^{-2} Gy
\mathbf{rem}	historical unit of equivalent dose: 1 rem = 1 x 10^{-2} Sv
Sv	sievert; SI measurement unit of equivalent dose, equal to the absorbed dose in gray multiplied by the weighting factor prescribed for the radiation form of interest
u	atomic mass unit; 1 u is the equivalent mass of one twelfth of a neutral Carbon atom's mass. 1 u = 1.6605×10^{-27} kg

Constants and Coefficients

С	$3 \times 10^8 \mathrm{~m/s}$
h	Planck's constant; $6.62\times 10^{-34}~{\rm J/s}$
γ	radioactive decay constant
Γ_{δ}	gamma constant
μ	linear attenuation coefficient; $\rm cm^{-1}$
μ_m	mass attenuation coefficient; $\rm cm^2/g$

Chapter 1

Introduction

1.1 Introduction

The basis of this research project was to determine the utility and appropriateness of ionising radiation measurement devices used within the mineral processing industries. Judgements have been made in respect to process applications, safety and viable alternatives. Mineral processing nuclear measurement systems exploit the properties of ionising radiation to provide pertinent measurements and are primarily used for;

- process instrumentation,
- workplace safety, and
- environmental monitoring

Much of this subject matter is also applicable to the industries of coal, cement and chemical production.

One of the main drivers of this project was to address the author's own understanding of what appeared to be a "dark science". The words "radiation" and "nuclear" even in these low exposure applications typically trigger unsubstantiated fears in the general public and in the workplace where installed. Included in this research is a review of

1.1 Introduction

radiation safety that addresses these fears; this has found that the biggest safety risks appear to revolve around radiation source security, transport and disposal.

Undertaking this project substantially committed the author to gaining an understanding of the applicable nuclear physics. This was an early requirement, so as the concepts of detection, measurement and safety could be put into context. In addition this knowledge also gave the author a foundation on which sound judgements could be made. Coverage of the applicable physics can be found in the early parts of this dissertation. Overall the acquirement of this knowledge took far longer than expected and consumed approximately 40% of all research effort.

Initially instrument review was based on the systems installed at Newmont Mining Corporation's Boddington Gold Mine. Further investigation additionally found that several more different applications exist in other mineral processing environments and these too have been included as part of this research.

The review of current nuclear technologies and their use, with respect to themselves and possible non-nuclear technologies will hopefully result in the implementation of more favourable outcomes.

In regard to the technical content of this dissertation, instruments utilizing ionising radiation as part of the measurement system are typically labelled as "nuclear instruments". This labelling at times is technically incorrect as not all forms of ionising radiation are generated from the nucleus. The author has chosen to continue this generalisation throughout this document.

1.2 Research aims and objectives

The scope of this research is based on the application of nuclear instruments within the mineral processing industries. The primary aim and objective of this research was to determine how useful and how appropriate nuclear instruments are in regard to the measurements they are employed to undertake. The aims and objectives for this project were compiled as follows:

Primary aim and objective

To successfully determine this primary aim and objective a method of judgement will be implemented; this judgement will be based on the following considerations:

- Benefit to process
- Installation, operating and disposal costs
- Workplace and public safety
- Comparisons with non-nuclear instrument installations

The judgement considerations proposed are believed necessary in determining the worthiness of an instrument with regard to a particular application. A breakdown of these considerations follows:

- 1. Benefit to process. This is a measure of what value an instrument will bring to the applicable process. Examples include the benefit a measurement will have on process control and the benefit a measurement will bestow on process management. Furthermore this value may also be attributed to workplace safety and environmental monitoring when taken in respect to operations involving naturally occurring radioactive materials (NORM).
- 2. Installation, operating and disposal costs. This consideration covers the cost of ownership of nuclear instruments; knowledge of this cost is essential when viewed in respect to the cost benefit analysis of the device. With regard to NORM applications such as workplace safety and environmental monitoring, it is envisaged these costs would be based on the initial purchase cost and costs of maintenance of the measurement devices. However most process and many analytical measurements will require the use of a radioactive source; this may

1.2 Research aims and objectives

include a substantial initial outlay, ongoing operational costs (including those in respect to workplace safety) and substantial disposal costs.

- 3. Workplace and public safety. Implications in the use of nuclear instruments must be taken in regard to workplace and public safety. The use of nuclear instruments in the workplace may pose a risk to employees and the public. Subsequently safety is paramount and is governed by legislation and codes of practice. Meeting these requirements places additional costs on the user and these must be taken into account when deciding the worthiness of a specific instrument.
- 4. Comparisons with non-nuclear instrument installations. The benefits of nuclear instrumentation cannot be made unless a comparison is drawn in respect to non-nuclear instruments. This comparison may be as basic as to state that no other alternative exists, through to a comparison that reflects the superiority of one device with respect to another.

Secondary aims and objectives

Delivery of the primary aim and objective will fulfil many of the secondary aims and objectives. The secondary aims and objectives include:

- An introduction to applicable nuclear physics
- An education in the use and applications of nuclear instruments
- A survey of site specific nuclear installations
- Recommendations with regard to current nuclear installations
- Determination of alternative measurement solutions

A break down of the secondary aims and objectives follows:

- 1. An introduction to applicable nuclear physics. To understand the benefits and downfalls of a nuclear instrument it is a necessity to gain and insight into the fundamentals of nuclear physics; this coverage is an essential requirement if quantitative judgements are to be made. Furthermore this provides an educational benefit to both the author and reader.
- 2. An education in the use and applications of nuclear instruments. The research component of this project by its very nature will provide the author and reader

with an insight into the usage and type of instruments readily found in a mineral processing environment.

- 3. A survey of site specific nuclear installations. Although only a snapshot of one particular processing plant, this should provide an indication of what nuclear instruments may be typically found in most mineral processing plants. It will also give some relevance to radioactive source type and the safety implications.
- 4. Recommendations with regard to current nuclear installations. These recommendations will be predominantly generated from the site specific survey. Recommendations may include a judgement on the instruments suitability for the task, installation arrangements, maintenance and possible workplace safety issues.
- 5. Determination of alternative measurement solutions. Time permitting, alternative measurement methods may be proposed for the installations covered under the site survey. These solutions may be innovative or readily available in the marketplace.

Anticipated outcomes

It is expected that successful completion of the said aims and objectives may result in;

- making the world a safer place,
- improved instrument selection,
- a reduction in regulatory compliance, and
- education of my peers

1.3 Dissertation Outline

This dissertation has been completed over 14 chapters. An overview of the content follows:

Chapter 2 – Literature review, details the primary sources of literature that were used in the compilation of this dissertation. Sections include;

- applicable nuclear physics,
- radiation detection and measurement,
- industrial nuclear instruments,
- radiation safety and regulation, and
- non-nuclear measurement methods
- Chapter 3 Methodology, discusses the approach as to how this research project was conducted.
- Chapter 4 The Basics of Nuclear Physics, introduces the fundamental concepts of ionising radiation. These concepts include; the origins of radioactivity, the different forms that occur and the associated energies.
- Chapter 5 Radiations interaction with matter, reviews the interaction that different radiation forms have with matter. This chapter discusses the theory behind radiation shielding and penetrability of matter with respect to radiation type.
- Chapter 6 Radiation units, intensity and attenuation, presents the measurement units associated with radiation. The intensity measurements of exposure, absorbed dose and equivalent dose are explained and their derivation outlined. Furthermore attenuation and the associated methods of calculation are defined.
- Chapter 7 Radiation safety, legislation and regulation, discusses radiation safety in respect to everyday exposure and the effects that may occur due to all levels of exposure. The issues of radiation source security are also raised. In addition Australian legislation and regulation are introduced.

- Chapter 8 Radiation sources, describes the sources of ionising radiation that may be found in a mineral processing environment. Sources types include; naturally occurring radioactive materials (such as uranium), radioactive isotopes and xrays.
- Chapter 9 Detectors, presents the types of radiation detectors typically used within the mineral processing industry. Detector types include; Geiger-Muller tubes, scintillation detectors and semiconductor techniques.
- Chapter 10 Measurement, introduces the fundamentals of radiation measurement. Inclusive are the measurement model, the statistics of measurement, operational effects and signal processing techniques.
- Chapter 11 Measurement Applications, presents the types of ionising radiation process measurement and analytical instruments that can be typically found in the mineral processing environments.
- Chapter 12 Evaluation, performance and alternatives, reviews instrument installations in respect to the measurement applications. Where possible, comparisons are made against viable alternatives.
- Chapter 13 Recommendations, gives an account of what the author believes are the best measurement solutions for current applications. In additional, issues that have been raised from theory or from the site survey are addressed with possible solutions.
- Chapter 14 Conclusions, notes research achievements against aims and objectives. Key addition findings of this research are also presented. A section "Further works" is included that considers the author's research and suggests the direction in which any further work may be worthwhile.

Supporting documents of this research may be found in the following appendices:

Appendix A – **Project Specification**, details the project specification for this research.

- Appendix B Dose Limits and Maximum Permissible Exposure Levels, details the radiation dose limits set by the State of Western Australia.
- **Appendix C Australian Radiation Regulatory Authorities**, provides the name and contact details of the radiation regulatory authorities within Australia.
- Appendix D Registration of Radioactive Substances, details the registration requirements for radioactive substances and lists the fee schedule set by the State of Western Australia.

Chapter 2

Literature Review

2.1 Chapter Overview

This chapter covers the literature that was most significant in aiding the completion of this research project. This review has been segregated into parts that reflect the differences in subject matter. These sections include:

- The applicable nuclear physics
- Radiation detection and measurement
- Nuclear instruments
- Radiation safety, legislation and regulation
- Non nuclear instrument alternatives
- Questions raised from the literature review

Of significance in this review is the fact that books written by authors in the areas of expertise remained the best and most dependable sources of information. Review has also shown that in general; university, national and international organisation websites contain trustworthy information.

2.2 The applicable nuclear physics

Initially it was difficult to differentiate between the quality and the relevance of reference materials with respect to the subject of "the applicable nuclear physics". This was primarily due to the subject in hand. In many cases the information was difficult to understand as it is derived from the point of view of a nuclear physicist, whilst other sources are too simplistic. Internet sources also often appeared dubious, with little supportive evidence of content.

Taking into account the above difficulties, it was decided the best approach would be to initially access information that would provide an introduction to the relevant nuclear physics in hardcopy from a university library. Books were accessed from both the Edith Cowan University and the University of Western Australia. Via this process, much of the author's initial understanding of nuclear physics has been derived from the book "Nuclear Physics: principles and applications" by Lilley (2001).

Table 2.1 shows the main sources of literature reviewed. From review, a good understanding of principles of radioactivity has been achieved and documented. Remarkably the radiation source types available for mineral processing instrument applications are limited; all are man-made, with many also being applicable for medical use.

Another unexpected revelation was the finding that gamma rays and x-rays are indistinguishable from each other and therefore there is no known method of detecting the difference between the two. Surprisingly the distinguishing difference between gamma rays and x-rays is solely due to their source of origin. Furthermore it has been found that gamma rays and x-rays can both share and cross over their designated sections of the electromagnetic spectrum.

Source	Title/link	Author	Review
Book	Nuclear Physics: Principles and	JS Lilley	Contains good informa-
	applications	(2001)	tion, substantially used
			when addressing the sub-
			ject of nuclear physics.
Book	Fundamentals of Physics	D Halliday,	An excellent reference di-
		R Resnick,	rected at those studying
		J Walker	university physics
		(2011)	
Internet	An Introduction to Radioactivity	R Lawson	Provides a good basic un-
	http://www.e-radiography		derstanding of the nu-
	.net/articles/Introduction		clear physics involved
	%20to $%20$ Radioactivity.pdf		
Internet	The ABCs of Nuclear Science	Lawrence	Provides a wealth of in-
	http://www.lbl.gov/abc/	Berkeley	formation in regard to ra-
		National	dioisotopes
		Laboratory	
Internet	DOE Fundamentals Handbook	U.S	A very good reference
	Nuclear physics and reactor the-	Department	that still explains the
	ory http://www.hss.doe.gov/	of Energy	concepts well even
	nuclearsafety/techstds/docs/	(1993)	though slightly dated
	handbook/h1019v1.pdf		
Internet	The nucleus and radioactive de-	CTA Lee,	University lecture notes
	cay http://www.ruf.rice.edu/	Rice Uni-	that cover the fundamen-
	ctlee/Chapter2A.pdf	versity	tals of nuclear physics
Internet	Radioactivity	VP Guinn,	An excellent read that
	http://www.sciencedirect.com/	University	details the fundamentals
	science/article/pii/	of Califor-	of radioactivity
	B0122274105006438	nia	

Table 2.1: Literature review of Nuclear Physics

2.3 Radiation detection and measurement

The detection of radiation and its measurement appears as a much specialised field; the literature found was often secular on detection method with very few sources giving adequate coverage. The review of many university sites revealed one consistent source of information. This information source was the book *"Radiation Detection and Measurement"* by Glenn F. Knoll (2010). Further review of this source led the author of this research to purchase a copy of the latest edition (4th) from John Wiley & Sons, Australia. This purchase has proven extremely worthwhile and has been extensively used throughout this research project. This book would be a noteworthy addition to any technical library.

Table 2.2 is a compilation of sources that provided good information for this project. Additionally close inspection of books by Cooper (1986) and Knoll (2010) found that both have devoted sections to the statistics of detection; both clearly state that radioactive decay is a statistical process and therefore the measuring process is also statistical. Hence many measurements of the same sample are required to deliver a measured value that can be stated with confidence.

Source	Title/link	Author	Review
Book	Radiation detection and measure-	GF Knoll	An excellent resource.
	ment (4th Edition)	(2010)	Covers all areas includ-
			ing the basics of nu-
			clear physics, requires
			the reader having gained
			knowledge from previous
			chapters.
Book	Introduction to nuclear radiation	PN Cooper	Introduces the different
	detectors	(1986)	methods of detection.
			This is not an in depth
			reference, but it is a
			good beginning. Also
			covers basic nuclear the-
			ory
Book	Radioisotope Gauges for Indus-	GA Johansen	An excellent resource.
	trial Process Measurement	& P Jackson	Covers all areas includ-
		(2004)	ing the basics of nu-
			clear physics, requires
			the reader having gained
			knowledge from previous
			chapters.
Internet	Geiger Muller Tubes	Centronic	An excellent review of
	http://www.centronic.co.uk/		the operation, design
	downloads/		and types of radiation
	Geiger_Tube_theory.pdf		that can be detected us-
			ing Geiger Muller tubes
Internet	Radiation detectors (Saint-	Saint-	Manufacturer of radia-
	Gobain Crystals)	Gobain	tion detectors. Provides
	http://www.detectors.saint-	Crystals	additional technical in-
	gobain.com/		formation

Table 2.2: Literature review of Radiation Dection and Measurement

2.4 Nuclear instruments

A literature review of nuclear instruments currently in use within the mineral processing industries was commenced firstly based on the instruments found at the Boddington Gold Mine. Through review of these devices by manufacturer, it was further possible to ascertain what other types of nuclear instruments may currently be in service principally due to device synergies in manufacture. Table 2.3 covers the main information sources that have been used. Very few references were found that even came close to covering the full range of instruments available, notably the greatest coverage found was given by the International Atomic Energy Agency's (2005) publication *"Technical data on nucleonic gauges"*. The book *"Radioisotope Gauges for Industrial Process Measurement"* by G. Johansen and P. Jackson (2004) has been included in the section as well as the prior. This book would also be a noteworthy addition to any technical library.
Source	Title/link	Author	Review
Book	Radioisotope Gauges for Indus-	GA Johansen	An excellent resource.
	trial Process Measurement	& P Jackson	Gives good insight into
		(2004)	the operational theory of
			many applications.
Book	Operating Manual Density Meter	Berthold	O & M manual for an
	LB444	(2009)	on-pipe density gauge.
			Covers the fundamentals
			of operation well.
Internet	Technical data on nucleonic	International	An excellent apprecia-
	gauges	Atomic En-	tion of what devices are
	http://www-pub.iaea.org/	ergy Agency	currently used in general
	MTCD/publications/PDF/	(2005)	industry
	$te_1459_web.pdf$		
Internet	Ronan Engineering Measure-	Ronan En-	An overview of process
	ments Division, Library of	gineering	instruments using nu-
	technical documents.	Measure-	clear techniques. It
	http://www.ronanmeasure.	ments	also clearly states sev-
	$com/tn_radiat.html$	Division	eral limitations with re-
			gard to nuclear belt
			weighers
Internet	Berthold Technologies	Berthold	Instrument manufactur-
	https://www.berthold.com/	Technolo-	ers' web site
		gies	
Internet	Thermo Scientific http://www.	Thermo Sci-	Instrument manufactur-
	thermoscientific. com	entific	ers' web site
Internet	Vega Americas	Vega Ameri-	Instrument manufactur-
	http://www.ohmartvega.com/	cas	ers' web site
Internet	Ronan Measurements Division	Ronan Mea-	Instrument manufactur-
	http://ronanmeasure.com/pages/	surements	ers' web site
		Division	

Table 2.3: Literature review of Nuclear Instruments

2.5 Radiation safety, legislation and regulation

The perusal of literature relevant to radiation safety, legislation and regulation has found that within Australia the regulation of radioactive devices and radiation safety is primarily a state and territory responsibility. On a Commonwealth level this role is undertaken by the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA). Additionally codes of practice for Australian jurisdictions are published by ARPANSA and cover the areas of safety, transport, storage and disposal. Table 2.4 shows the information sources that were instrumental in completing this section of this dissertation. Notably all are internet sources of high quality from highly respected or government organisations.

Title/link	Author	Review
Radiation Safety Regula-	Radiological	A PowerPoint presenta-
tion in Western Australia	Council	tion that covers radia-
http://www.dmp.wa.gov.au		tion safety and responsi-
/documents/duncan.ppt		bilities in Western Aus-
		tralia
Government of Western Aus-	Radiological	Statutory authority of
tralia Radiological Council	Council	Western Australia that
http://www.radiologicalcouncil.		provides requirements
wa.gov.au/		and fact sheets.
Radiation Protection and Nu-	ARPANSA	National radiation safety
clear Safety Agency (ARPANSA)		site that also publishes
http://www.arpansa.gov.au/		codes and practices.
Western Australian Radia-	Government of	Covers radiation safety
tion Safety Regulations 1983.	Western Aus-	legislation and regulation
http://www.slp.wa.gov.au/	tralia, State	in Western Australia
$legislation/statutes.nsf/main_{-}$	Law Publisher	
$mrtitle_1969_homepage.html$		
International Atomic Energy Or-	International	A well-known interna-
ganisation	Atomic Energy	tional organisation, that
http://www.iaea.org/	Organisation	publishes wide rang-
		ing recommendations,
		including safety and
		security guides.
International Commission on Ra-	International	A well-known interna-
diological Protection	Commission	tional organisation, that
http://www.icrp.org/	on Radiologi-	publishes wide ranging
	cal Protection	recommendations on ra-
		diation safety.

Table 2.4: Literature review of Radiation safety, legislation and regulation

2.6 Non – nuclear instrument alternatives

Investigation into non-nuclear instrument alternatives found that once again good books are a worthwhile asset when undertaking a task such as this. An excellent source of information was the *Instrument Engineers' Handbook*" by B.G. Lipták. In principle this large textbook is an extensive coverage of all process instruments, and covers the theory of operation and applications. This book would stand proud in any Instrument Engineers' technical collection.

Additional alternatives were also found through the use of the internet and more specifically by perusal of process instrument manufacturers' websites. Table 2.5 lists the most useful information sources reviewed.

Source	Title/link	Author	Review
Book	Instrument Engineers' Hand-	BG Lipták	An excellent in-depth
	book: Process Measurement and	(2005)	book that covers a mul-
	Analysis		titude of instrument
			solutions.
Book	Instrumentation for Process Mea-	NA Andersor	An older book that cov-
	surement and control	(1997)	ers many traditional con-
			cepts in process measure-
			ment.
Internet	Emerson Process Measurement	Emerson	Instrument manufactur-
	http://www2.emersonprocess.com	Process	ers' website.
		Measure-	
		ment	
Internet	Krohne Messtechnik GmbH	Krohne	Instrument manufactur-
	http://krohne.com/	Messtech-	ers' website.
		nik GmbH	
Internet	Vega Grieshaber KG	Vega	Instrument manufactur-
	http://www.vega.com	Grieshaber	ers' website.
		KG	

Table 2.5: Literature review of Non-nuclear measurement methods

2.7 Questions raised from the literature review

From the literature reviewed, the initial questions raised included:

- Is there a definitive requirement for the use of radioactive isotope sources and if so are they necessarily the best selection? If gamma rays and x-rays are effectively the same would it not be better to use x-ray sources that can be simply disabled by the removal of power?
- Should the use of nuclear belt weighers be condoned?
- Should employees within mineral processing plants be given a better eduction in regard to radiation safety?
- Are there significant statistical measurement issues with respect to the use of nuclear instruments?

Completion of this dissertation has answered these questions. Answers to these can be found in the relevant sections on measurement and/or in the recommendations and conclusions sections.

Chapter 3

Project Implementation

3.1 Chapter Overview

As a significant part of this research project was out of discipline, an additional education in the applicable nuclear physics was required at the early stages of the project. This in itself took an extensive period of time to complete and consumed a significant amount of research resources. This education was completed in conjunction with the initial review of the nuclear instruments installed at the Boddington Gold Mine. Overall this research is based extensively on theory, and on the review of existing nuclear and non-nuclear instruments available. The following section details the methodology taken in achieving project completion.

3.2 Methodology

Successful project completion required the development of a strategy that was well ordered and in the main required the acquisition of knowledge in certain areas before the next task could be undertaken. The following details the task description, how the task was achieved and the order of implementation:

- 1. Education in the relevant fields of nuclear physics. This was achieved through a literature review with the appropriate knowledge documented in the dissertation.
- 2. Assessment of what nuclear instruments are used in the mineral processing industries. This was achieved by a site survey of Boddington Gold Mine and a wide ranging review of industry literature.
- 3. Education in regard to the relevant nuclear measurement techniques. This was achieved through a literature review with the appropriate knowledge also documented in the dissertation.
- 4. Education in the field of radiation safety. This was achieved by literature review and where appropriate through discussion with the site Radiation Safety Officer.
- 5. Appraisal of nuclear instruments used within the mineral processing fields. Judgements were made with regard to the following:
 - Benefit to process.
 - Inherent characteristics such as reliability and accuracy. This was achieved via literature review of manufacture's information, site history and discussion with site personnel.
 - Installation, operating and disposal costs. This required liaison with suppliers and Boddington Gold Mine staff to determine costs.
 - Workplace and public safety.
- 6. From the above processes a list of recommendations for the current nuclear instrument applications were generated. Recommendations include a judgement on the instrument's suitability for the task, installation arrangements, maintenance and possible workplace safety issues.

- 7. With regard to items 5 & 6, performance models for both nuclear and traditional instrumentation where applicable were developed. These are based on the previous findings, literature review and discussion with specialists on site.
- 8. With time permitting, viable measurement/technology alternatives were proposed for Boddington Gold Mine nuclear instrument installations. These were based on the previous determined models, the literature review of alternative technologies and the author's own knowledge.

With project completion, it was found the above methodology was adequate. However in respect to the total additional knowledge required for successful project completion, the research time allotted was in some cases found to be severely under estimated.

Chapter 4

The Basics of Nuclear Physics

4.1 Chapter Overview

This chapter covers the basics of ionising radiation. Throughout the chapter the reader is exposed to the origins, types and energies of these radiations. The concepts of activity, decay and half-life are also introduced. In addition decay mechanisms are covered to an extent that will give the reader the preliminary knowledge for upcoming chapters that form the basis of radiation measurement. The writings of this chapter are to a significant extent based on interpretation of the book *Nuclear Physics, principles and applications* (Lilley 2001).

4.2 Ionising Radiation

To understand the measurement principles behind nuclear instruments one must first have an understanding of ionising radiation. Ionisation energy as quoted by D.Halliday, Resnick & Walker (2011, p. 1113) is "the energy required to remove the most loosely bound electron from a neutral atom". The amount of energy required to achieve this depends on the atom type and ranges from approximately 4 to 24 eV (electron volts). With reference to the definition; ionising radiation has enough energy to remove an electron from the atoms of a target material. This form of radiation can be divided into two types, particulate and electromagnetic. In respect to this document the relevant forms of ionising radiation are:

- 1. Particulate alpha particles, beta particles and neutrons
- 2. Electromagnetic gamma rays and x-rays

Sources of ionising radiation occur both naturally and through man's application of science. Background radiation is the natural sources of ionising radiation that we are regularly exposed to in our day to day lives; the value of this radiation at any location is variable and is primarily influenced by geology, latitude and altitude. The main contributors of background radiation are the ongoing radioactive decay of the isotopes of uranium and thorium, and cosmic rays. Uranium and thorium are a legacy of universal creation, whilst cosmic rays consist predominately of protons and alpha particles from outer space, which interact with the Earth's atmosphere producing gamma rays and electrons.

The majority of ionising radiation sources within the mineral processing industry are radioactive isotopes; in a very few specialised applications x-rays are artificially produced through the use of x-ray tubes.

4.3 Isotopes and Radioactive Decay

Elements can have several different configurations know as isotopes. Isotopes of the same element are distinguished from each other by their different mass number A. All isotopes of a particular element have the same number of protons Z, but differ in the number neutrons N they have within the nucleus. Notably as the isotopes of an element have the same number of electrons they act chemically identical.

Protons and neutrons are particles of a nucleus and are regularly referred to as nucleons (non-definitive particles of the nucleus). Therefore the mass number A is equal to the number of nucleons, which is in turn equal to the sum of protons and neutrons within the nucleus.

Isotopes of an element can either be stable or unstable. Stable elements do not change (mass number) over time. Unstable isotopes are known as radioactive isotopes and are unstable due to the composition (the number of protons and neutrons) within the nuclei.

According to quantum physics a nucleus can exist at several discrete energy states. Nuclei at an excited state will de-excite. The nuclei of radioactive materials are in an excited state and dependant on their energy state will emit electromagnetic radiation (gamma rays) and or particulate radiation (alpha particles, beta particles and neutrons) to obtain a lower energy state.

The energy unit associated with ionising radiation is the electron Volt, eV; as defined by Knoll (2010, p. 3) one electron volt is "the kinetic energy gained by an electron by its acceleration through a potential difference of one volt". The equivalent energy value in joules is: $1 \text{ eV} = 1.602 \times 10^{-19} \text{ J}$

The spontaneous release of ionising radiation and the subsequent loss of energy is known as radioactive decay. During radioactive decay the "Parent" nuclide (type of nucleus identified by the number of protons and neutrons) alters to become a more stable "Daughter" nuclide. This daughter nuclide may also be radioactive and if so, the decay process will continue until stability is reached. When decay occurs through many transformations, the process is a chain of events and this process is commonly known as a decay chain. Some of these decay chains can be quite extensive and may take extremely long periods of time. An example is the radioactive decay of uranium-238 (238 U) to lead-206 (206 Pb), which can be viewed in Fig. 4.1.



Figure 4.1: Uranium-238 decay chain (Department of Mines and Petroleum (2010))

The rate of decay of a radioactive isotope is referred to as activity. This activity is expressed in the SI unit becquerel (Bq), with one becquerel equal to a rate of one disintegration per second. An older unit of measurement of activity is the curie, with one curie equal to 3.7×10^{10} Bq; it is still quite common to come across the use of this unit and many older sources can be found marked with the source activity in mCi. The rate of decay is based on probability and by using an isotope's decay constant γ , the rate of decay can be calculated with:

$$\frac{dN}{dt} = \gamma N \tag{4.1}$$

From the above the amount of radioactive nuclei after a period of time can be determined using the equation:

$$N(t) = N_0 e^{-\gamma t} \tag{4.2}$$

where N_0 = the original number of radioactive nuclei at t_0

 $\gamma = \text{decay constant}$

t = time; s

The common radioactive term half-life, is the amount of time required to pass until a radioactive substance has half of its original activity. Half-life can also be calculated using the decay constant as shown in the following formula:

$$t_{1/2} = \frac{\ln 2}{\gamma} \tag{4.3}$$

An additional term often used in regard to radioactivity is specific activity; this refers to the activity per mass and is usually quoted in becquerels per gram (Bq/g).

Coulomb potential energy is a measure of the energy within a nucleus that is resultant of the constant interaction between, the attractive nuclear forces of the neutrons and the repulsive electromagnetic forces between protons. The Coulomb energy is greatest near the edge of the nucleus creating a boundary referred to as the Coulomb barrier. Isotopes that eject charged particles will have their half-lives altered by the effect of the Coulomb barrier.

The energy required to break the nucleus into to its component parts is defined as its binding energy. Science has shown that the mass of a nucleus is always less than the sum of its components. This difference in mass is known as the mass defect. Knowing the mass defect and using Einstein's equation of $E = mc^2$ the binding energy of a nucleus can be determined. The relevant equation in respect to binding energy is:

$$m(A,Z) = Zm_p + Nm_n - \frac{B}{c^2}$$
 (4.4)

where m(A, Z) =mass of the nucleus

Z = number of protons $m_p = \text{mass of a proton; } 1.67262158 \times 10^{-27} \text{ kg or } 1.007276 \text{ u}$ N = number of neutrons $m_n = \text{mass of a neutron; } 1.67492735 \times 10^{-27} \text{ kg or } 1.008665 \text{ u}$ B = binding energy (J or MeV) $c = \text{speed of light; } 3 \times 10^8 \text{ m/s}$

Note: $1 \text{ u} = 1.6605 \times 10^{-27}$ kg. Nuclear and atomic masses are regularly referred to in atomic mass units (u), with 1 u the equivalent mass of one twelfth of a neutral carbon atom's mass. This has the benefit of easier energy calculations, with the mass of $1 \text{ u} = 931.5 \text{ MeV}/c^2$.

From the above it can be seen that in reference to $E = mc^2$ the binding energy can be calculated using:

$$B = \Delta m c^2 \tag{4.5}$$

where B = binding energy (J or MeV) $\Delta m =$ the mass defect c = speed of light; 3×10^8 m/s

Calculating the binding energy using the alpha particle as an example:

1. Firstly using standard engineering units:

$$m(A,Z) = Zm_p + Nm_n - \frac{B}{c^2}$$

$$\frac{B}{c^2} = Zm_p + Nm_n - m(A,Z)$$

$$\frac{B}{c^2} = (2 \times 1.673 \times 10^{-27}) + (2 \times 1.675 \times 10^{-27}) - 6.644 \times 10^{-27}$$

$$B = (0.0503 \times 10^{-27}) \times c^2$$

$$B = (0.0503 \times 10^{-27}) \times (3 \times 10^8)^2$$
$$B = 4.527 \times 10^{-12} \text{ J}$$

Converting to MeV:

$$B = 4.527 \times 10^{-12} \text{ J} \times \frac{1 \text{ MeV}}{1.602 \times 10^{-13} \text{ J}}$$
$$B = 28.26 \text{ MeV}$$

2. Secondly using atomic mass units and the conversion factor of 1 $u = 931.5 \text{ MeV}/c^2$:

$$m(A, Z) = Zm_p + Nm_n - \frac{B}{c^2}$$

$$\frac{B}{c^2} = Zm_p + Nm_n - m(A, Z)$$

$$\frac{B}{c^2} = (2 \times 1.007276 \text{ u}) + (2 \times 1.008665 \text{ u}) - 4.00150675 \text{ u}$$

$$B = (0.0304 \text{ u})c^2$$

$$B = (0.0304 \text{ u})c^2 \times \frac{931.5 \text{ MeV}}{1 \text{ u} \times c^2}$$

$$B = 28.29 \text{ MeV}$$

An increase of nucleons in a nucleus will result in an increase in binding energy. Subsequently the binding energy is often expressed in binding energy per nucleon;

Binding energy per nucleon
$$=$$
 $\frac{\text{total binding energy}}{\text{number of protons and neutrons}}$ (4.6)

A binding energy curve for stable nuclei can be viewed in Fig. 4.2. In this figure binding energy per nucleon is plotted in respect to the mass number. Cooper (1986, p. 3) states "Atoms that are most stable have the highest value of average binding energy per nucleon".



Figure 4.2: Binding energy curve, plotting binding energy per nucleon against atomic mass. (Massachusetts Institute of Technology (2006 a))

4.3 Isotopes and Radioactive Decay

Fig. 4.3 shows a "Chart of Nuclides" that depicts the stability of nuclei with respect to half-life; nuclei are plotted in reference to their number of protons and neutrons. Stability can be found to follow the middle of the band; this band is regularly referred to as the "line or valley of stability". Nuclei can have an equal number of neutrons and protons, be neutron rich or neutron poor. Nuclei away from the line of stability are unstable and achieve stability through the process of radioactive decay. Isotopes with an atomic number greater than 83 are inherently unstable. The limiting atomic number of naturally occurring isotopes is 92; all isotopes above these numbers are man made and are designated as transuranic elements.



Figure 4.3: Chart of nuclide stability in respect to half-life. (adapted from Nationial Nuclear Data Centre, Brookhaven National Laboratory (2012))

4.4 Radioactive Decay Mechanisms

There are several different methods of radioactive decay. With respect to mineral processing the following are relevant:

- Alpha decay
- Beta decay and Electron capture
- Spontaneous fission
- Gamma rays and Internal Conversion
- X-rays

Additionally other forms of radioactive decay include proton and neutron emissions. The following brief description of proton and neutron emissions is based on the text *"Radiation Mechanics; Principles and Practice"* (Hussien 2001, p. 39). Proton emission may occur under the following two circumstances:

- 1. Beta-delayed proton emission emission of a proton from a daughter nucleus that is in a highly excited state after beta decay.
- 2. Extremely proton rich isotopes may emit protons to gain stability.

Proton emission is a rare form of decay and is only possible due to man made isotopes. Additionally neutron emission is also a rare form of decay and may also occur in two similar manners as with proton emission:

- 1. Beta-delayed neutron emission emission of a neutron from a daughter nucleus that is in a highly excited state after beta decay.
- 2. Extremely neutron rich isotopes may emit neutrons to gain stability.

Notably both proton and neutron emission are not possible without artificial isotopes. Furthermore both emission types may also eject other types of particles from the nucleus. Fig. 4.4 shows a "Chart of Nuclides" that depicts the decay type in reference to an isotope's number of protons and neutrons.



Figure 4.4: Chart of Nuclides in respect to decay type.

(adapted from Nationial Nuclear Data Centre, Brookhaven National Laboratory (2012))

4.4.1 Alpha emission

The radioactive decay of most heavy nuclei (isotopes with an atomic number > 82) results in the emission of alpha particles. Alpha particles are effectively the nuclei of Helium atoms. The released energy of decay takes the form of kinetic energy shared between alpha and daughter particles. Presuming the nucleus is at rest at the time of decay, using the laws of energy conservation and momentum the kinetic energy of product particles can be determined using the following equations:

$$E_{\alpha} = Q_{\alpha}(\frac{m_D}{m_D + m_{\alpha}}) \tag{4.7}$$

$$E_D = Q_\alpha - E_\alpha \tag{4.8}$$

where E_{α} = alpha energy

 E_D = daughter energy Q_{α} = decay energy m_D = mass of the daughter particle m_{α} = mass of the alpha particle

As the alpha particle is much lighter than the daughter, the alpha particles energy is usually approximately 98% of the total decay energy. Via the measurement of the energies of decay it is possible to determine the parent nuclide, alternatively it is also possible to determine the daughter product if the parent is known in conjunction with the alpha energy value.

Determination of the energy of decay is best explained by way of example. Based on the alpha decay of plutonium-239 (239 Pu) to uranium-235 (235 U):

239
Pu $\longrightarrow ^{235}$ U + 4 He

The energy released in decay can be calculated using two methods:

1. By calculating the difference in binding energy between parent and products. Using the equation of:

$$Q_{\alpha} = E_D + E_{\alpha} - E_P \tag{4.9}$$

where $Q_{\alpha} = \text{decay energy}$

 E_D = daughter energy E_α = alpha energy E_P = parent energy with ²³⁹Pu = 1806.914 MeV ²³⁵U = 1783.863 MeV ⁴He = 28.296 MeV

> $Q_{\alpha} = 1783.863 + 28.296 - 1806.914$ $Q_{\alpha} = 5.245 \text{MeV}$

2. By calculating the difference in mass energy between parent and final products. Using the equation of:

$$Q_{\alpha} = (m_P - m_D - m_{\alpha})c^2$$
 (4.10)

where $Q_{\alpha} = \text{decay energy}$

 $m_{P} = \text{mass of parent}$ $m_{D} = \text{mass of daughter}$ $m_{\alpha} = \text{mass of alpha particle}$ $c = 3 \times 10^{8} \text{ m/s}$ with ²³⁹Pu = 239.052157 u ²³⁵U = 235.043924 u ⁴He = 4.002602 u $Q_{\alpha} = (239.052 + 235.044 - 4.003)c^{2}$ $Q_{\alpha} = 0.005631c^{2}$ $Q_{\alpha} = 0.005631c^{2} \times \frac{931.5\text{MeV}}{c^{2}} \quad (\text{MeV conversion factor})$ $Q_{\alpha} = 5.245\text{MeV}$

4.4 Radioactive Decay Mechanisms

Notably alpha decay always results in transmutation; the creation of a more stable daughter element that may well also be radioactive. As previously mentioned the halflives of isotopes that emit charged particles are affected by the Coulomb barrier and this therefore also holds true for alpha particle decay. The height of the Coulomb barrier is associated with an increase in atomic number (protons). As the Coulomb barrier acts as a barrier to alpha decay, increasing values will result in increased half-lives.

4.4.2 Beta emissions and electron capture

Unstable nuclei with an atomic mass less than 82 will generally decay by beta emission or by electron capture.

Beta emissions can be positive or negative. Negative beta emissions release negatron (e^{-}) particles that are the equivalent of electrons, whilst positive beta emissions release positrons (e^{+}) the equivalent of positively charged electrons. Additionally positive beta emissions are always accompanied to some degree with electron capture. Electron capture will occur on its own, if discharge energies are less than 1.02 MeV.

Neutron rich isotopes will decay via negative beta emission. In this process a neutron within a nucleus will convert to; a proton, a negatron and an anti-neutrino (\bar{v}) . Alternatively proton rich isotopes will decay via positive beta emission, converting a proton to; a neutron, a positron and a neutrino (v). Neutrinos and anti-neutrinos are particles of minimal mass, with no charge and are additional particles created in the beta decay process that maintain the energy/mass balance during decay.

The equations for negative and positive beta emission are as follows:

Negative beta
$$n \longrightarrow p + e^- + \bar{v}$$
 (4.11)

Positive beta
$$p \longrightarrow n + e^+ + v$$
 (4.12)

The energy of beta emission differs from alpha decay, as the energy of negatrons and positrons is spread over a spectrum, with the maximum value equal to that anticipated from mass/energy calculations; subsequently average electron/positron energy values are approximately one third of this value. The spread of the energy spectrum is due to the inconsistent sharing of energy between discharged particles (i.e. negatrons and anti-neutrinos).

With beta decay the Coulomb barrier assists in the ejection of positrons and retards the discharge of electrons from the nucleus. However the Coulomb barrier cannot be used to give any indication of an isotope's half-life, as it is only one of many complex factors that require consideration when attempting calculation.

Beta emissions change the numbers of neutrons and protons within a nucleus, with the charge of nucleus composition summarized as follows:

- 1. Positive beta emission maintains the mass number A, but decreases the atomic number Z by one.
- 2. Negative beta emission maintains the mass number A, but increases the atomic number Z by one.

Electron capture is a process with may also occur as an alternative to positive beta decay in a proton rich nucleus. This process captures an electron from one of the atom's electron shells, usually from the innermost "K" shell; the result is the conversion of a proton in the nucleus to a neutron and neutrino. The process equation follows:

$$p + e^- \longrightarrow n + v$$
 (4.13)

As an electron is captured from one of the atom's shells, a vacancy now exists that will be filled by an electron from an outer shell, this effectively results in a cascade of electrons, moving from shell to shell filling vacant positions. This process of electron movement (change in energy level) gives rise to the emission of x-ray photons.

Electron capture is dissimilar to both positive and negative beta as the energy released is not spread across a spectrum; the energy released with electron capture is at a characteristic level and is bestowed on the neutrino.

Positive beta and electron capture often occur together and at different ratios that are isotope dependent. It is even possible to have positive beta, negative beta and electron capture occurring with the same isotope producing two different daughter products. The energy released in negative beta and electron capture decay is calculated using the difference between parent and daughter atomic masses and can be determined using the following equation:

$$Q_{\beta} = (m_P - m_D)c^2 \tag{4.14}$$

where $Q_{\beta} = \text{decay energy}$

 $m_P =$ mass of parent $m_D =$ mass of daughter $c = 3 \times 10^8$ m/s

Using atomic mass units (u) and applying the MeV conversion factor, the equation becomes:

$$Q_{\beta} = \Delta m c^{2} \times \frac{931.5 \text{ MeV}}{c^{2}}$$
$$Q_{\beta} = \Delta m \times 931.5 \text{ MeV}$$
(4.15)

Whereas the energy released in positive beta is slightly less due to the emission of a positron and neutrino; calculation is similar to above with a subtraction of 1.022 MeV applied to the MeV conversion factor. Equation 4.15 in this instance becomes:

$$Q_{\beta} = \Delta m \times 930.5 \text{ MeV} \tag{4.16}$$

An additional event that results from positive beta emissions is the production of annihilation radiation. This occurs when the emitted positron combines with an electron of an absorbing material as the positron comes to a stop and their combined mass is converted to two 0.511 MeV gamma rays that travel in opposite directions.

Beta decay always ends in transmutation due to a change in the number of protons within the nucleus.

4.4.3 Spontaneous fission

Very heavy nuclei may decay by spontaneous fission. When this occurs the heavier parent is split into two lighter daughter nuclides. Nuclides that decay by spontaneous fission are both alpha and neutron emitters. Spontaneous fission occurs because as the mass number A rises, the repulsive Coulomb forces due to neutron/proton interaction become overwhelming, splitting the atom in two.

With useful heavy nuclei, alpha emissions significantly out weigh emissions due to spontaneous fission. An example of this is the decay of californium-252, in this case only 3.1% of decay is due to spontaneous fission. As the mass number of nuclei rise, spontaneous fission becomes the more dominant form of decay and half-lives rapidly diminish.

4.4.4 Gamma Emission, Internal Conversion and X-rays

Very often after an initial decay event such as alpha or beta decay, the nucleus may be left in an excited state and the move to a more stable state may result in the discharge of a gamma ray photon. These photons (gamma rays) are electromagnetic waves that can have substantially larger energies than that of visible light. Gamma ray photons are generally emitted immediately ($< 10^{-9}$ s) after other decay mechanisms.

The transition to a lower state via gamma rays for some nuclei can take periods longer than nanoseconds; these nuclei are referred to as metastable isomers.

The energy level (state) to which a nucleus stabilises is one of probability; it depends on the attributes determined by quantum physics and the initial energy level. Energy levels of gamma rays typically used in the mineral processing industry are below 2 MeV.

Gamma emission is not the only method in which transition from a higher to lower energy state can occur. An alternative is Internal Conversion; in this process the energy of the excited nucleus ejects an electron from one of the atom's shells. The energy from this event is at discernible level, and furthermore this characteristic energy value is dependent on the shell from which the electron was ejected.

Electrons also have a binding energy to the nucleus. The kinetic energy of the ejected electron can be calculated using the following equation:

$$E_{el} = Q - B_{el} \tag{4.17}$$

where E_{el} = electron kinetic energy

Q = decay energy

 B_{el} = electron binding energy

Internal Conversion results in a vacancy in one of the atom's shells, as with electron capture the cascading electrons filling the vacancy result in energy emissions, but in this case taking the form of x-ray photons.

For all intents and purposes gamma rays and x-rays are the same; they even overlap in the electromagnetic spectrum. Both have no mass or charge and are pure energy. The difference between the two is their origin; gamma rays originate from the nucleus whereas x-rays originate from the electron shell.

There is no method that can distinguish between gamma and x-rays; however gamma rays do tend to have shorter wave lengths and therefore higher energies.

The correlation between gamma and x-ray wave length, and energy is as follows:

$$\lambda = \frac{hc}{E} \tag{4.18}$$

where λ = wave length; m

h = Planck's constant; 6.62×10^{-34} J/s $c = 3 \times 10^8$ m/s E = decay energy; J

Alternatively to determine using MeV, a constant can be substituted for hc allowing for the Joule to MeV conversion:

$$\lambda = \frac{1.24 \times 10^{-12}}{E} \tag{4.19}$$

where λ = wave length; m

E = decay energy; MeV

4.5 Quantum Numbers

The probabilities of both nucleon and electron energy levels (states) present at discrete levels have been determined through the study of quantum theory and experimentation. These energy states are represented by quantum numbers.

The following is predominantly based on an interpretation of an extract from "Nuclear Science-A guide to the Nuclear Science Wall Chart" produced by the Lawrence Berkeley National Laboratory (2003) of California.

The energy state of the individual nuclei is dependent on the composition of the nucleus. As the study of quantum physics is beyond the scope of this document, only a reference to composition of the quantum numbers is given; for the nucleus these are:

- Angular momentum quantum number (J); regularly referred to as nuclear "spin" represents the value of total angular momentum with reference to \hbar (Diracs constant); derived from Planck's constant divided by 2π . With angular momentum equal to the quantum number J multiplied by \hbar .
- Parity (P); a value that is "+" or "-", it is referenced to a comparison of energy states that result when the spatial coordinates of all nucleons are reversed. A "+" indicates no change, whereas a "-" indicates a difference.
- Isospin (T); gives an indication of the property change that would occur if neutrons and protons of a nucleus exchanged positions.

The full format for expressing quantum numbers for nucleus energy states is J^P , T (angular momentum, parity and isospin). Generally in the textbook sense isospin is not readily used and quantum numbers usually take a format of simply J^P (Angular momentum and parity).

Quantum numbers are often used as a method to show excitation levels and the associated energy. Fig. 4.5 is an example which depicts the energy level spectrum of carbon-12. Notably the ground state is the lowest rung of the diagram, with excitation levels increasing towards the top.



Figure 4.5: Energy level spectrum of ¹²C.(Lawrence Berkeley National Laboratory (2003))

At lower energy levels nuclei tend to decay by gamma ray emission, however as energy levels rise, it is possible for a nucleus to decay using other methods such as alpha or beta decay. In Fig. 4.5 a second excited state exists at 7.66 MeV, this is at a high enough level for alpha emission to occur with a daughter of ⁸Be.

Energy levels for electrons are also represented by the use of quantum numbers. The four quantum numbers in regard to electron energy levels are:

- The principle quantum number (n); this value represents the shell number the electron occupies. The innermost electron shell "K" is represented by n = 1 with n increasing by a value of one for each additional shell further from the nucleus. Additionally used to specify the energy value of an electron.
- Angular quantum number (l); used to represent the shape of an elliptical orbit. This integer value can range from zero to n - 1. Additionally breaks shells into sub-shell groups. Also used to specify the energy value of an electron.
- Magnetic quantum number (m) used to define the position of an orbital of given energy and shape. Additionally breaks sub-shells into orbitals, with each orbital containing a maximum of 2 electrons.
- Spin quantum number (s), used to represent the direction of spin around the axis of an electron. Values are $\pm \frac{1}{2}$, if an orbital is occupied by 2 electrons they shall have opposite signs.

Under the "Pauli exclusion principle" (Wolfgang Pauli 1945) no electrons of the same atom can have an identical set of four quantum numbers. Electron energy states can be depicted with similar drawings to that previously shown to the nucleus; however energy levels tend to be more in the keV range than the MeV range. Notably de-excitation of electrons generally results in the emission of x-rays.

4.6 Decay Schemes

A graphical method used to illustrate how a radioactive isotope decays is the "Decay Scheme". These drawings generally show the following:

- The parent nucleus
- The daughter nucleus

- Half-life of the parent nucleus
- Types of decay and their probability
- Energy levels
- Energy states as described by quantum numbers

Decay schemes for the isotope cobalt-60 can be seen in Fig. 4.6. These schemes show that decay is 100% negative beta and the way decay occurs has different probabilities. The decay scheme on the left is more complex and shows half-life in days and quantum energy levels. Whilst the decay scheme on the right is more simplified it depicts decay mechanisms well, showing the initial decay as negative beta and that de-excitation of the daughter nucleus is achieved via gamma ray emissions primarily at 1.173 and 1.332 MeV.



Figure 4.6: Decay scheme of cobalt-60.

(Left - Nationial Nuclear Data Centre, Brookhaven National Laboratory (2012)) (Right - Massachusetts Institute of Technology (2006 c))

4.7 Chapter Summary

Ionisation radiation has enough energy to remove the most loosely bound electron from a neutral atom. The amount of energy required to achieve this depends on the atom type and ranges from approximately 4 to 24 eV. One electron volt equates to the kinetic energy gained by an electron by its acceleration through a potential difference of one volt and is equal to 1.602×10^{-19} J. Ionising radiation can be divided into two types, particulate and electromagnetic. The common forms of particulate radiation are charged and uncharged, and include;

- alpha particles,
- beta particles, and
- neutrons

Ionising forms of electromagnetic radiation include gamma rays and x-rays, which have no charge, are pure energy and are emitted as photons. Both electromagnetic radiation forms are typically emitted after some other decay mechanism has occurred. It is impossible to distinguish between x-rays and gamma rays, but x-rays are generally at a lower energy.

Sources of ionising radiation occur both naturally and through man's application of science. Naturally occurring radiation is primarily due to cosmic rays and from the decay of radioisotopes that are a legacy of the Earth's formation.

Activity is expressed in the SI unit becquerel (Bq); with one becquerel equal to a rate of one disintegration per second. Half-life is the amount of time required to pass until a radioactive substance has half of its original activity.

Decay schemes are a graphical method used to illustrate how a radioactive isotope decays and typically show the half-life of the parent nucleus, daughter products, decay mechanisms, probability, and energy levels.

Chapter 5

Radiation's Interaction with Matter

5.1 Chapter Overview

Due to the many forms of ionising radiation, interaction with matter occurs in many different ways. In respect to earth sciences, through the acquired knowledge of this interaction, it has been possible to;

- develop specific radiation type detectors,
- develop protection measures, such as shielding,
- construct artificial sources of ionising radiation, and
- develop methods of elemental analysis

In addition, study in this field has made extremely significant contributions to medical science.

This chapter covers ionising radiation's interaction with matter. Comprehension of this subject will give the reader a good appreciation of the penetrability of the different forms of ionising radiation, an understanding of shielding methods and establish some fundamental measurement concepts.

5.2 Penetrating Power

The following is a brief overview of ionising radation's penetrating ability. The penetrating power of a specific form of ionising radiation is very much charge dependent; with uncharged forms having the greatest penetrating ability. A simplistic indication of the penetrating power of the different forms of radiation can be viewed in Fig. 5.1. As can be seen, the stopping distance is not only a function of energy, but also of radiation type.



Figure 5.1: Penetrating abilities of ionising radiation.(The Australian Journal of Emergency Management, Colella et al. (2005))

With the exception of neutrons, the higher the atomic mass of an absorber the shorter the stopping distance; this is due to the subsequent higher electron densities and the interaction that occurs with the other forms of radiation.

Many mathematical relationships have been developed over time to model the penetrating distance of radiations; with this distance dependent on the particle's energy and radiation type.

5.3 Interaction with Matter

As previously covered ionising radiations can be charged or uncharged; with the uncharged forms achieving the greatest penetration. The charged forms of interest include alpha and beta particulate radiations that result in the emission of heavy charged par-

5.3 Interaction with Matter

ticles, electrons (negatrons) and positrons. Uncharged forms include neutrons, gamma rays and x-rays. The following subsections cover the aforementioned radiation's interactions with matter.

5.3.1 Heavy charged particles

Alpha particles and protons are the emissions that are most commonly associated with heavy charged particles. Energy loss of a heavy charged particle is primarily due to coulombic interaction of the positive ions with the absorbing material's electrons. Direct collisions between charges and nuclei rarely occur, and in principle can be ignored. As a charged particle passes through the absorber, additional energy attributed to the absorber's electrons may move these electrons to higher energy states or even result in ionisation. The stripping of an electron from the absorber's atom results in an ion pair; consisting of the negative electron released and the subsequently created positive absorber ion. Because the laws of energy conservation must hold, the charged particle subsequently loses kinetic energy and will eventually come to a halt. As the particle comes to a halt the positive particle neutralises by the collection of electrons from the absorber.

Those electrons that have risen to a higher state may de-energise by emission of x-rays, whereas the ionised electrons released may cause further ionisation events or recombine with positive ions of the absorber.

The maximum amount of energy that can be transferred from a heavy charged particle to an electron will happen with the occurrence of a head on collision. In this case the maximum amount of energy transferred can be calculated with the following equation:

$$\Delta E = 4E \frac{m_e}{m} \tag{5.1}$$

where ΔE = energy transferred to the electron

E = energy of charged particle

- $m_e = \text{mass of the electron}$
- m = mass of the heavy particle

In all other cases the energy transferred will be less. The average distance to achieve full energy loss is surprisingly repeatable for particles of the same nature; this is due to the averaging effect that the many interactions have along the ionising tract.

Differences from average distance values are referenced as straggling, with values for alpha particles in the lower percentiles (< 3%). The average penetrating distance is often referred to as the "range" value. The properties that define a particles' range include;

- the kinetic energy of the particle,
- the mass of the particle,
- the charge of the particle, and
- the nature of the absorbing material

A heavy charged particle's range in any absorber material can be determined using the Bragg-Kleenman equation. Essentially by knowing the range of a heavy charged particle in a known material, commonly air, the range in another material can be determined using;

$$\frac{R_1}{R_2} = \frac{\rho_2}{\rho_1} \sqrt{\frac{A_2}{A_1}}$$
(5.2)

where R_1 = range of reference

 $R_2 = range of unknown$

 $\rho_1 = \text{density of reference; g/cm}^3$

- $\rho_2 = \text{density of unknown; g/cm}^3$
- A_1 = atomic weight of reference; g/mol
- $A_2 =$ atomic weight of unknown; g/mol

which becomes;

$$R_2 = \frac{\rho_1}{\sqrt{A_1}} \frac{\sqrt{A_2}}{\rho_2} R_1 \tag{5.3}$$

An example of the range determination of a 4 MeV alpha particle in aluminium follows. Firstly the range of an alpha particle in air at 15°C and at 1 atmosphere can be approximated from the following formula taken from Cember (1996, p. 132);

$$R_{air}[cm] = \begin{cases} 0.56E_{\alpha} & \text{for} \quad E_{\alpha} < 4\text{MeV} \\ 1.24E_{\alpha} - 2.62 & \text{for} \quad 4\text{MeV} \le E_{\alpha} < 8\text{MeV} \end{cases}$$
(5.4)
$$R_{air} = (1.24 \times 4) - 1.24 = 2.34 \text{ cm}$$

Taking the properties of air at 15°C and at 1 atmosphere:

where
$$\rho_{air} = 1.23 \text{ x } 10^{-3} \text{ g/cm}^3$$

 $\rho_{al} = 2.7 \text{ g/cm}^3$
 $A_{air} = 14.6 \text{ g/mol}$
 $A_{al} = 26.98 \text{ g/mol}$

$$R_{al} = \frac{\rho_{air}}{\sqrt{A_{air}}} \frac{\sqrt{A_{al}}}{\rho_{al}} R_{air}$$

$$R_{al} = \frac{1.23 \times 10^{-3}}{\sqrt{14.6}} \frac{\sqrt{26.98}}{2.7} 2.34$$

$$R_{al} = 1.44 \times 10^{-3} \text{cm}$$

The path of a heavily charged particle is generally straight due to the extremely large mass difference between the heavy particle and the electrons of the absorber. This path may distort towards the end of the particle's trajectory, as it has lost significant energy and the influence of the electrons has much more of an effect. Modelling of the rate of energy loss $\left(\frac{-dE}{dx}\right)$ for alpha particles has determined that the rate of energy loss (stopping power/ionisation) increases as the particle loses energy along the ionising tract. A plot depicting this interaction is referred to as a Bragg curve; an example for 1.8 and 4.3 MeV alpha particles in silicon carbide (SiC) is shown in Fig. 5.2. As can be viewed the rate of energy loss increases as the distance travelled increases; notably as the particle comes to a halt with neutralisation due to ion exchange, the rate of energy loss decreases rapidly.


Figure 5.2: Bragg Plot for 1.8 and 4.3 MeV alpha particles in SiC. (Kinoshita et al. 2005)

Although it can be shown that absorbers of higher atomic number provide greater attenuation, these materials are generally not used for shielding due to the extremely short distances heavy charged particles can travel in any material. The penetrating power of alpha particles is very low and subsequently the human skin acts as a sufficient barrier.

5.3.2 Electrons

Unlike alpha particles, electrons do not follow typically straight paths through matter. Electrons tend to scatter during interaction with the absorber's electrons due to their very light and equivalent mass. This scatter can be to the extreme, where the electrons can actually leave the absorber in the direction from which they incident, this occurrence is commonly referred to as backscatter. The depth of penetration for electrons substantially exceeds that of heavy charged particles of the same energy. Electrons have a much greater speed than heavy charged particles due to their much smaller

5.3 Interaction with Matter

mass, which generally results in significantly smaller energy losses during interaction. Notably however, direct collisions can consume much of an electron's energy. As a consequence of the variability of interaction, range determination for a single electron is not possible.

An electron's energy can be lost through not only ionisation and excitation interactions, but through the effects of acceleration. The changes in acceleration that occur as electrons pass through an absorber result in the emittance of bremsstrahlung radiation (x-rays); this is due to the phenomenon that all charged particles will emit electromagnetic energy when rapidly decelerated. Subsequently the energy loss of electrons is a sum of ionisation, excitation and radiative (electromagnetic) energy losses. Radiative energy losses are strongly influenced by the atomic number of the absorbing material and the initial energy. The ratio of the radiative to ionisation and excitation losses according to Knoll (2010, p. 43) approximates the initial energy multiplied by the atomic number, divided by 700:

$$\frac{(\frac{dE}{dx})_r}{(\frac{dE}{dx})_c} = \frac{EZ}{700} \tag{5.5}$$

where $(\frac{dE}{dx})_r$ = radiative losses $(\frac{dE}{dx})_c$ = ionisation and excitation losses

E =Initial energy in MeV

Z = atomic number

From this equation it can be interpreted that radiative losses are not the dominant losses unless materials of higher atomic number are used with significant initial energies.

Because of the inability to determine a single electron's range in a material, statistical methods using a large quantity of electrons have been required to determine the average penetration depth. As previously covered, beta emissions cover a large energy spectrum and therefore consideration in any shielding design must account for energies at the higher end of the spectrum. Transmission curves are regularly used to display the attenuation that occurs with beta particles. An example of a transmission curve for beta particles from several different radiation sources using aluminium as an absorber can be viewed in Fig. 5.3.



Figure 5.3: Transmission curves for beta particles emitted from ${}^{35}S$, ${}^{60}Co$ and ${}^{204}Tl$ sources using aluminium as an absorber. (Haemers et al. (2007) Delft University of Technology)

A material's absorption coefficient for beta emissions according to Knoll (2010, p. 45) may be defined using the following equation:

$$\frac{I}{I_0} = e^{-nt} \tag{5.6}$$

where $I_0 = \text{count rate without absorber}$

- I = count rate with absorber
- n = absorption coefficient
- $t = absorber thickness; g/cm^2$

The above equation takes into allowance the maximum spectral energy displayed by beta emissions.

5.3.3 Positrons

Positive beta emissions emit positrons; their interaction with matter is very much the same as electrons with exception that they will produce annihilation radiation. This occurs when the emitted positron combines with an electron of an absorbing material as it comes to a halt, and their mass is converted to two 0.511 MeV gamma rays that travel in opposite directions. When compared with negative beta emissions, positive beta emissions have significantly greater absorber pentration due to the secondary gamma rays produced.

5.3.4 Gamma rays and x-rays

As gamma ray and x-ray photons carry no charge they can not directly cause ionisation. Ionisation is possible due to the fast electrons that are created through the photon's interaction with matter. The interaction of gamma rays and x-rays with absorbing materials chiefly occurs through three separate processes, namely the photoelectric effect, Compton scattering and pair production. The dominance of one process over another is dependent on the photon's initial energy and the atomic number of the absorbing material. A chart depicting the dominance of an associated process in respect to photon energy and the atomic number of the absorbing material can be viewed in Fig. 5.4.

The photoelectric effect encompasses the total absorption of a gamma or x-ray photon by an absorbing material's atom, and the subsequent release of a fast electron from predominantly one of the absorber atom's inner electron shells. In addition to this initial emission, the absorbing atom may attempt to stabilise to a lower energy state by the release of more loosely bound electrons, or by the emission of x-rays as a free electron fills the vacancy and/or as electrons move to lower level shells to fill vacancies. The aforementioned ejected loosley bound electrons are commonly known as "auger electrons".



Figure 5.4: Gamma ray interaction processes with respect to photon energy and atomic number Z. (Massachusetts Institute of Technology (2006b))

The release of the initial fast electron (photoelectron) has a kinetic energy of:

$$E_{e^-} = E_\gamma - B_e \tag{5.7}$$

where $E_{e^-} = \text{fast electron's energy}$

 E_{γ} = photon energy

 B_e = binding energy of the electron

As with decay processes, the methods of de-excitation for a given energy may all occur and at different rates dependent on the absorber material. Notably the photoelectric effect occurs at lower photon energies.

Compton scattering occurs between the incident photons and an absorber's electrons. Unlike the photoelectric effect, the incident photons are not absorbed by absorber atoms, but deflected by free electrons at varying angles. The energy imparted on the free electron, now known as a recoil electron (due to subsequently imparted kinetic energy) is dependent on the scattering angle. This interaction also results in the scattering of the incident photons with a reduction in energy, which can be defined from Fig. 5.5 and by using the equation of:

$$hv' = \frac{hv}{1 + \frac{hv}{m_0 c^2} (1 - \cos\theta)}$$
(5.8)

where hv' =scattering photon's energy

hv = photon's initial energy

 $m_0 c^2$ = rest mass energy of the electron; 0.511 MeV

 θ = scatter angle in radians



Figure 5.5: Compton scattering.

(National Aeronautics and Space Administration, Science Mission Directorate. (2010))

In respect to the above, the kinetic energy of the free electron can be simply determined using:

$$E_k = hv - hv' \tag{5.9}$$

where E_k = kinetic energy of the free electron

hv = photon's initial energy

hv' =scattering photon's energy

Pair production is the third main process and is only possible when the energy of the incident photons exceeds twice the at rest mass energy of the electron (1.022 MeV). This process creates both an electron and a positron; their existence is due to the photon energy being absorbed into the nucleus, with the resultant conversion of energy to mass. For photon energies greater than 1.022 MeV, the additional energy exceeding the requirement for pair production is imparted on both the electron and positron as kinetic energy.

The likelihood of pair production remains low until the photon energy exceeds approximately twice the minimum level required for pair creation and is not normally significant until initial energies exceed several MeV. The aftermath of pair production will result in the further radiative emissions that are associated with negative beta (electron) interactions and positron interactions (annihilation radiation).

The attenuation of gamma rays and x-rays is greatly dependent on the aforementioned processes and the characteristics of the absorbing materials. Fig. 5.6 shows the attenuation of photons at different energies with respect to the interacting mechanisms and with air as the absorbing material. Calculations of attenuation can be completed using the parameter μ that represents the linear attenuation coefficient and is defined with the unit of inverse length (cm⁻¹). In similarity with beta transmission calculations μ replaces the absorption coefficient n and t now represents the absorber thickness. The linear attenuation coefficient can thereby be deduced using the equation of:

$$\frac{I}{I_0} = e^{-\mu t}$$
 (5.10)

where $I_0 = \text{count rate without absorber}$

- I = count rate with absorber
- μ = linear attenuation coefficient; cm⁻¹
- t = absorber thickness; cm

As attenuation is due to the sum of the effects of the interacting mechanisms:

$$\mu = \tau(\text{photoelectric effect}) + \sigma(\text{Compton}) + \kappa(\text{pair})$$
(5.11)



Figure 5.6: Attenuation of gamma and x-rays through air. (Massachusetts Institute of Technology (2006b))

With attenuation calculations it is more common to use mass absorption coefficients; a parameter that allows for the density of an absorbing material. This parameter can be viewed on the vertical axis of Fig. 5.6. The mass absorption coefficient can be determined using the following expression;

$$\mu_m = \frac{\mu}{\rho} \tag{5.12}$$

where $\mu_m = \text{mass}$ attenuation coefficient

 μ = linear attenuation coefficient; cm⁻¹ ρ = density; g/cm³ Attenuation calculations can now be completed using the equation:

$$\frac{I}{I_0} = e^{-\mu_m \rho t} \tag{5.13}$$

where $I_0 = \text{count rate without absorber}$

I = count rate with absorber

 $\mu_m = \text{mass attenuation coefficient; } \text{cm}^2/\text{g}$

 $\rho = \text{density}; \text{g/cm}^3$

t = absorber thickness; cm

The half thickness value of an absorber is commonly used to describe the required thickness of an absorbing material to reduce the radiation intensity by half. As gamma and x-ray radiations are of an exponential nature the half thickness can be determined using:

$$x_{\frac{1}{2}} = \frac{\ln 2}{\mu} \tag{5.14}$$

where $x_{\frac{1}{2}}$ = absorbing material's half thickness; cm

 μ = linear attenuation coefficient; cm⁻¹

Due to the exponential nature of gamma and x-ray radiations it can be seen that many half thicknesses may be required to attenuate the radiation to a safe level. Furthermore derived is the "inverse square law" which states that the intensity of gamma rays and x-rays is inversely proportional to the square of the distance from the source. In simply terms this may be interpreted as; by doubling the distance from a source, the radiation is reduced fourfold.

5.3.5 Neutrons

Neutrons are generally considered the most dangerous type of radiation due to penetrating ability and the high energies involved. Furthermore, neutrons can induce radioactivity in target materials. All neutron sources are artificial. As with gamma rays and x-rays, neutrons have no charge and therefore do not have electromagnetic (coulomb) interactions. Neutrons interact with the nucleus of an absorbing material's atoms, and may at higher energies produce secondary radiations that generally take the form of heavy charged particles. Due to the low probabilities associated with the nucleus' extremely small size in respect to the atom, neutrons can travel substantial distances before interacting.

Neutrons have a half-life of less than 15 minutes and may be divided into categories dependent on energies, according to L'Annunziata (2007, p. 253) these include:

- 1. Cold neutrons: < 1 meV
- 2. Thermal neutrons: < 0.5 eV
- 3. Epithermal neutrons: 0.5 eV 50 keV
- 4. Fast neutrons: > 50 keV
- 5. Medium energy neutrons: > 1 MeV
- 6. High energy neutrons: > 10 MeV

Note: Thermal neutrons are neutrons that are at thermal equilibrium with the associated environment.

Velocities of neutrons can be determined using basic kinetic formulas, in principle:

$$E = \frac{1}{2}mv^2 \tag{5.15}$$

The interacting mechanisms of neutrons include:

- Elastic scattering
- Inelastic scattering
- Absorption reactions

At lower energies, typically below 1 MeV neutron interactions generally involve elastic scattering. The majority of elastic scattering events involve the collision of the neutron

with a nucleus of the absorbing material, and the subsequent deflection of a neutron and a recoiling nucleus. In this interaction, momentum and kinetic energy are conserved between the neutron and nucleus. Any kinetic energy loss by the neutron is gained by the nucleus. Importantly elastic scattering results in no secondary radiation emissions. Rarer elastic scattering events may occur that involve the absorption of the neutron into the nucleus, and then the release of a neutron whilst still conserving kinetic energies.

Inelastic scattering involves the absorption a neutron into the nucleus, and then the release of a neutron of lower energy. This leaves the nucleus in an excited state, which will return to the ground state through the emission of gamma rays. Taking into account that the conservation of energy applies, the gamma ray energy will equal the initial neutron energy minus the energy of the neutron ejected from the nucleus.

Absorption reactions occur when the nucleus absorbs the neutron and may result in gamma ray, particle emissions and fission.

In a process known as "radiative capture" a neutron is absorbed into the nucleus with no particle release; this places the nucleus in a higher energy state that de-excites to the ground state through the emission of gamma rays. Radiative capture produces an isotope of the absorbing material with an increase in mass number of one.

An alternative to radiative capture is that of "particle ejection". In this case the neutron is absorbed and the nucleus de-energises through the ejection of heavy charged particles. This leads to the transmutation to another element that will in most cases transform back to the absorbing material's original composition through negative beta decay. The secondary radiations produced from this interaction mechanism are often used as a method to detect neutron radiation.

Fission is also an absorption mechanism and occurs due to the interaction of a neutron with a heavy nucleus. Neutrons with sufficient energy can facilitate fission in most elements that have an atomic number greater than 90. The process of fission involves the absorption of a neutron into the nucleus and the following de-excitation of the nucleus by separation into two or possibly more fission fragments of different atomic number. In addition to the creation of fission fragments, nucleus de-excitation also releases additional neutrons and gamma rays. Fission reactions release high energy values per incident; as an example, the fission of uranium-235 typically releases approximately 200 MeV of energy per interaction event.

Fig. 5.7 shows the dominant forms of neutron interaction, dependent on neutron type.



Figure 5.7: Neutron interactions with respect to energy. (Hamilton (2006))

The likelihood of a neutron interacting with a nucleus of an absorbing material is described through the use of the term cross section (σ); it is typically quoted in barns, with one barn equivalent to an area of 10^{-24} cm². Cross section values are different for each type of interacting mechanism, and are primarily dependent on the incident neutron's energy and the composition of the absorbing materials. The larger the cross section value the more likelihood there is of interaction.

The mean free path length (λ) of neutrons can be determined using macroscopic cross sections (Σ) that are derived from the aforementioned microscopic cross section values (σ) . The macroscopic cross section can be determined using:

$$\Sigma = N\sigma \tag{5.16}$$

where $\Sigma = \text{macroscopic cross section}$, units in inverse length; cm⁻¹

N = number of atoms/cm³ of the absorbing material

 σ = microscopic cross section of absorbing material; barns (10⁻²⁴ cm²)

Once the macroscopic cross section of the absorber has been determined the mean free path of a neutron in cm, can be simply determined by finding the reciprocal of Σ :

$$\lambda = \frac{1}{\Sigma} \tag{5.17}$$

where λ = neutron's mean free path length; cm

 $\Sigma = \text{macroscopic cross section; } \text{cm}^{-1}$

It must also be remembered that an allowance must be made where more than one type of interaction is possible; with respect to calculating mean free path length, the total macroscopic cross section is a sum of the individual interaction mechanism macroscopic cross sections. Additional allowances will be further required if the absorbing material is of varying chemical composition.

Study in the fields of cross sections has revealed that the most effective shielding materials for neutrons contain hydrogen. As the hydrogen nucleus contains only one proton, its mass is nearly identical to that of a neutron and due to this physical similarity, it is in theory possible to transfer the entire neutron's excess energy to the proton in a perfect collision in the process of elastic scattering. Elastic scattering results in the generation of heat, that may be insignificant or in larger quantities controllable. In the case of hydrogen inelastic scattering is not possible. Use of shielding materials of higher atomic number, such as lead reduces the likelihood of interaction and therefore results in a longer mean free path length. In addition any interaction will more likely result in some elastic scattering and the subsequent emission of gamma radiation as the absorber nuclei de-excite. As a consequence of these gamma emissions shielding arrangements are further complicated.

As previously stated the most effective neutron shielding materials contain hydrogen, examples include; water, paraffin and concrete.

5.4 Chapter summary

As can be clearly seen, radiation types vary greatly in their ability to penetrate an absorber, vary in energy levels and therefore vary in inherent risk. Prime attributes are dependent on charge, with the uncharged radiations of gamma rays, x-rays and neutrons having far greater depths of penetration. Notably effective shielding materials are type dependent; neutrons may be effectively stopped by materials containing substantial amounts of hydrogen where as gamma rays will pass straight through. In addition lead may stop gamma rays, but has little effect on neutrons.

Chapter 6

Radiation units, intensity and attenuation

6.1 Chapter Overview

The terminology used in radiation measurements can be confusing. This chapter aims to clarify the use of terminology and introduce the concepts of intensity and attenuation. Furthermore the relationships between SI and the still commonly used legacy units are addressed.

6.2 Activity

As previously discussed in chapter 4, the activity of radiation sources in SI units is defined using the unit becquerel (Bq); with one becquerel equal to a rate of one disintegration per second. An older unit of measurement of activity that is still in common circulation is the curie (Ci), with one curie equal to 3.7×10^{10} Bq. The curie was derived as an equivalent value to the activity of 1 gram of radium.

6.3 Intensity

Radiation intensity is a value at a point that reflects the activity of a radioactive source, the distance from the source and any absorption prior to arriving at the point of measurement. Confusingly, intensity can be measured in many different ways characteristically dependent on an indicative reference, such as:

- 1. "Exposure", the number of ion pairs created as a result of gamma or x-ray interaction per cubic centimetre of air.
- 2. "Absorbed dose", this equates to the radiation energy absorbed per kilogram of the absorbing material.
- 3. "Equivalent dose", this equates to the absorbed dose multiplied by a weighting factor that will consider the damaging effects a particular type of radiation will incur on living tissue. This weighting factor is close to one for x-rays, gamma rays and beta emissions, but is significantly higher (x 20) for alpha particles. This simply states that the different forms of radiation can incur significant differences in biological damage, even though the same amount of energy is delivered.

As with many units of measure, the passage of time and the establishment of international standards have altered the preferred unit descriptor. Preferred descriptors are now all SI units, however the alternative units are still often found on existing equipment and in documentation. Furthermore the legacy units still appear in predominant use for engineering calculations. Table 6.1 covers the inter-relationship between both the SI unit and the legacy unit descriptors.

The measurement of "Exposure" is a measure of the charge per volume, measured as mass $\left(\frac{dQ}{dm}\right)$ that results from ionisation due to gamma or x-ray interaction. The SI unit for exposure is the *coulomb per kilogram* (C/kg). The legacy unit is the *roentgen* (R) which equates to the generation of approximately 2.08 x 10⁹ ion pairs per cubic centimetre of air under standard temperature and pressure conditions. Conversion between the SI and legacy unit can be achieved using;

$$1 \text{ R} = 2.58 \text{ x} 10^{-4} \text{ C/kg}$$

	Radioactivity	Absorbed	Equivalent	Exposure
		\mathbf{dose}	\mathbf{dose}	
Legacy	curie (Ci)	rad	rem	roentgen (R)
units				
SI units	becquerel (Bq)	gray (Gy)	sievert (Sv)	coulomb/kilogram
				(C/kg)

Table 6.1: Inter-relationship between radiation measurement unit descriptors

Radiation Measurement Units

The SI unit for "Absorbed dose" is the gray (Gy) and equates to the absorption of one joule of energy per kilogram of absorbing material. The legacy unit for absorbed dose is the rad; conversion between the SI and legacy unit can be achieved using;

$$1 \text{ rad} = 1 \ge 10^{-2} \text{ Gy}$$

Notably materials of different composition subjected to equal energy levels will incur differences in absorbed dose. Due to the difficulties in measuring absorbed dose, values are usually inferred from ionisation measurements.

"Equivalent dose" allows for the biological effects that occur with different types of radiation. As previously stated this is achieved by applying a weighting factor dependent on radiation type to the absorbed dose measurements. The SI unit for equivalent dose is the *sievert* (Sv) and the legacy unit is the *roentgen equivalent man* (rem). Conversion between these two units can be achieved using;

$$1 \text{ rem} = 1 \ge 10^{-2} \text{ Sv}$$

The weighting factors for the different types of radiation can be viewed in Table 6.2 and specifically for neutrons with reference to kinetic energy in Fig 6.1. Essentially for gamma rays and x-rays, the unity radiation weighting factor results in the corresponding units representing absorbed dose and equivalent dose being equal, subsequently; 1 Gy = 1 Sv and 1 rad = 1 rem.

Table 6.2: Radiation weighting factors.

(International Commission on Radiological Protection (2007))

Radiation type	Weighting factor
Photons	1
Electrons	1
Protons	2
Alpha particles, fission fragments,	20
heavy ions	
Neutrons	A continuous curve as a function of
	neutron energy, refer to Fig. 6.1



Figure 6.1: Radiation weighting factors for neutrons vs neutron kinetic energy. (International Commission on Radiological Protection (2007))

With respect to gamma rays and x-rays, the conversion of exposure to absorbed dose and vice versa is somewhat more difficult, as a factor must be attributed to allow for the absorbing materials composition (air, muscle, bone, etc) and the energy of the incident radiation. This conversion factor is often referenced to as the f factor. With regard to nuclear process instruments in industrial applications the f factor for air is constant over the energies of interest.

Manufacturers typically quote the intensity values required in air at the detector to satisfy the measurement application. To add further confusion these are not quoted in a standard manner. Intensity values may be quoted in exposure, absorbed dose or equivalent dose values and in SI or legacy units. As a consequence, the need to convert between intensity units is often required so as other necessary calculations (i.e. source sizing, shielding) can be achieved. Correlating f factor values for air with the associated values of R and C/kg are 0.876 rads/R and 33.97 J/C respectively. As an example the following converts exposure in air to absorbed dose in air using both SI and legacy units using the formula of;

$$D = fX \tag{6.1}$$

where
$$D =$$
 the absorbed dose; Gy or rad
 $f =$ f factor (value depends on the radiation's energy level and
the units of exposure used)
 $X =$ exposure; R or C/kg

Using an exposure value of 3876 R and the equivalent value of 1 C/kg:

D = fX	D = fX
$D=33.97~{\rm J/C}\times 1~{\rm C/kg}$	$D=0.876~{\rm rad/R}\times 3876~{\rm R}$
$D = 33.97 { m Gy}$	D = 3395 rad

As 1 Gy is equal to 100 rad, confirmation that these two values are equivalent is simply achieved by multiplying the SI unit by 100.

1 becquerel	=	1 disintegration per second
1 curie	=	$3.7 \ge 10^{10}$ disintegrations per second

Table 6.3: Radiation measurement units.

1 millicurie (mCi)	=	37 megabecquerels (MBq)
1 rad	=	0.01 gray (Gy)
1 rem	=	0.01 sievert (Sv)
1 roentgen (R)	=	0.000258 coulomb/kilogram (C/kg)

1 megabecquerel (MBq)	=	0.027 millicuries (mCi)
1 gray (Gy)	=	100 rad
1 sievert (Sv)	=	100 rem
1 coulomb/kilogram (C/kg)	=	3,880 roentgens (R)

With the continued and somewhat supported use of legacy measurement units for both activity and intensity, measurement conversions between legacy and SI units are frequently required. Table 6.3 provides an overview of measurement units and lists the commonly used conversions.

Radiation intensity values obey Newton's inverse square law. A simple statement acknowledging this fact is "If the distance between the radiation source and the radiation intensity measurement is doubled, the measured intensity of the radiation is quartered". The mathematical relationship is;

$$I = I_0 \frac{{d_1}^2}{{d_2}^2} \tag{6.2}$$

where I_0 = original intensity

I = the intensity at the second point

 d_1 = the distance from the source that I_0 was determined

 d_2 = the distance from the source that I was determined

A straight forward example follows, with;

$$I_0 = 10 \ \mu Sv$$

 $d_1 = 0.25 \ m$
 $d_2 = 0.50 \ m$

$$I = I_0 \frac{{d_1}^2}{{d_2}^2}$$
$$I = (10 \times \frac{0.25^2}{0.50^2}) \text{ } \mu\text{Sv}$$
$$I = 2.5 \text{ } \mu\text{Sv}$$

6.4 Exposure rate and Dose Equivalent rate

Exposure rates and dose equivalent rates from different types of gamma sources of differing activity can be determined using gamma constants/factors that have been scientifically measured. These values have been determined with reference to a source activity unit at a specified distance from the radiation source.

The relationship between source activity and exposure rate can be determined according to Knoll (2010, p. 57) through the use of the following equation:

$$X = \Gamma_{\delta} \frac{\alpha}{d^2} \tag{6.3}$$

where X = exposure rate in R/hr

 Γ_{δ} = exposure rate constant (gamma constant)

 α = source activity in mCi

d = distance, usually in cm

Typically older calculations have been undertaken in terms of exposure rate, with the exposure rate constant (Γ_{δ}) given in (R.cm²)/(hr.mCi) at 1 cm or (mR.m²)/(hr.mCi) at 1 m from the radiation source. Calculations using SI units use the same general format as the equation for exposure rate, but usually determine dose equivalent rate in mSv/h, with the gamma constant given in (mSv.m²)/(h.MBq) at 1 m from the source. Table 6.4 and Table 6.5 give the gamma constants for exposure rate and dose equivalent rate calculations of radiation sources regularly used in mineral processing industries.

Table 6.4: Exposure rate constants for gamma sources.

(Knoll (2010, p. 57) and the Canadian Nuclear Safety Commission (2011))

Radiation source type	Exposure rate constant Γ_{δ} at 1 cm	
	$(\mathrm{R.cm}^2)/(\mathrm{hr.mCi})$	
cobalt - 60	13.2	
caesium – 137	3.3	
americium – 241	1.3	

Table 6.5: Dose rate constants for gamma sources.

Radiation source type	Dose equivalent rate constant Γ_{δ}	
	at 1 m (mSv.m ²)/(h.MBq)	
m cobalt-60	$3.67 \ge 10^{-4}$	
caesium – 137	$1.07 \ge 10^{-4}$	
americium – 241	$8.48 \ge 10^{-5}$	

(Unger & Trubey (1982))

An example calculating the equivalent dose rate at distance of 0.6 m from a 9250 Mbq caesium-137 source and using gamma dose equivalent rate constant of 1.07 x $10^{-4} (\text{mSv.m}^2)/(\text{h.MBq})$ follows. The equation from Knoll remains unchanged, with SI units substituted;

$$X = \Gamma_{\delta} \frac{\alpha}{d^2}$$

where
$$X = \text{dose rate; mSv/h}$$

 $\Gamma_{\delta} = 1.07 \times 10^{-4} (\text{mSv.m}^2) / (\text{h.MBq})$
 $\alpha = 9250 \text{ MBq}$
 $d = 0.6 \text{ m}$
 $X = \Gamma_{\delta} \frac{\alpha}{d^2}$
 $X = 1.07 \times 10^{-4} \frac{\text{mSv.m}^2}{\text{h.MBq}} \times \frac{9250}{0.6^2} \frac{\text{MBq}}{\text{m}^2}$

$$X = 1.07 \times 10^{-4} \frac{\text{ms/m}}{\text{h.MBq}} \times \frac{3200}{0.6^2}$$
$$X = 2.75 \text{ mSv/h}$$

6.5 Attenuation

The attenuation due to an absorber can be determined using a number of methods, these include;

- 1. calculations using mass or linear attenuation coefficients
- 2. calculations using half thickness values, and
- 3. transmission tables

As covered in Chapter 5, the half thickness value of an absorber is used to describe the required thickness of an absorbing material to reduce the radiation intensity by half. Using the following equation the half thickness value can be calculated if the linear attenuation coefficient is known, or vice versa if the the half thickness value is known:

:

$$x_{\frac{1}{2}} = \frac{\ln 2}{\mu} \tag{6.4}$$

where $x_{\frac{1}{2}}$ = absorbing material's half thickness; cm

 μ = linear attenuation coefficient; cm⁻¹

6.5.1 Linear and Mass attenuation coefficient methods

Linear absorption coefficient methods are rarely used due to additional complexities. Using mass absorption coefficients is relatively simple and may be completed with the equation;

$$\frac{I}{I_0} = e^{-\mu_m \rho t} \tag{6.5}$$

where $I_0 = \text{count rate without absorber}$

I =count rate with absorber

- $\mu_m = \text{mass}$ attenuation coefficient; cm²/g
- $\rho = \text{density}; \text{g/cm}^3$
- t = absorber thickness; cm

By setting $I_0 = 1$, the ratio of attenuated value to the original value can be determined.

6.5.2 Half thickness methods

Calculating attenuation using half thickness values which are readily available for different isotopes is relatively simple. The attenuation factor can be calculated using the following equation;

$$S = 0.5^n \tag{6.6}$$

where S = attenuation factor

n = number of half thickness layers

An example using the above equation:

Determining the attenuation achieved from using 50 mm of lead shielding with a Cs-137 source that has a half thickness value of 8 mm. This half thickness value has been taken from the Canadian Nuclear Safety Commission (2012) Radiation Safety Data Sheet for Cs-137. Firstly calculating the number of half thickness layers n:

$$n = \frac{50}{8}$$
$$n = 6.25$$

- -

substituting n into the equation;

$$S = 0.5^{6.25}$$

 $S = 0.013$

By using 50 mm of lead the original intensity has been reduced by 98.7%

6.5.3 Transmission tables

Transmission tables are another method of determining attenuation. A transmission table plots transmission in percent in respect to the absorbing material and its thickness. An example of a transmission table for Cs-137 can be viewed in Fig. 6.2. Using the this transmission table in respect to the previous example a slightly higher value of approximately 1.3% of original intensity is found. An additional benefit when using transmission tables is that a correlation can be easily draw between an absorbing material's density and its subsequent attenuation.



Figure 6.2: Transmission table for caesium-137 (Omega (n.d.))

6.6 Chapter summary

A prominent issue that remains is the use of legacy units in describing units of measure; this is primarily due to the reluctance of the United States to adopt the SI measurement system in this physical field.

Measurements of exposure are related to charge, whereas dose measurements are related to the dose deposited in a material. Notably the value of equivalent dose allows for the biological effects that differing radiation types of the same intensity have, by using a method of weighting on absorbed dose values. Attenuation materials are dependent on radiation type, with materials of high atomic number such as lead most suitable for the shielding of gamma rays and x-rays. Methods for attenuation calculations include:

- calculations using mass or linear absorption coefficients
- calculations using half thickness values, and
- transmission tables

Chapter 7

Radiation safety, legislation and regulation

7.1 Chapter Overview

Exposure to ionising radiation can be extremely hazardous to the human body at elevated levels. This chapter gives the individual a concept of the risk involved in respect to radiation exposure based on historical data. Methods that will reduce radiation exposure are also reviewed. Furthermore Australian legislative and regulatory requirements aimed at limiting workplace and public exposure are also introduced.

7.2 Radiation Safety

The effects of excessive exposure to ionising radiation are very much dependent on the radiation type and quantity. As previously reviewed in Chapter 6, with respect to *equivalent dose* a weighting factor is attributed to the type of radiation, with alpha particles having a greater biological effect by a factor of 20 over gamma rays of the same energy. In addition a weighting factor is also attributed for different parts of the human body that may be exposed, with this value referenced as *effective dose;* weighting factors for different parts of the body can be readily found in publications of the International Commission on Radiological Protection (ICRP).

The human body is constantly subjected to ionising radiation through interaction with the environment. This naturally occurring radiation is commonly referred to as natural background radiation, with the contributing sources of:

- Cosmic rays radiation from space
- Terrestrial and airborne radiation radiation evolving from the Earth's formation

Cosmic rays evolve from outer space and according to National Aeronautics and Space Administration, Goddard Space Flight Center (2012) consist primarily of protons (90%) and alpha particles (9%). Cosmic rays are highly penetrating and exposure is very much a function of altitude; the higher the altitude the greater the exposure.

Terrestrial and airborne radiation exists due to the reactions that formed the Earth. Much of this radiation can attributed to the decay of uranium, thorium, radium and the associated daughter products. Of note, one of the products from these decay chains is the radioactive gas radon, which decays by alpha emission and can be readily ingested. This is of great concern and under continuous monitoring in mining and processing environments where this is of known to occur. Assessments by the United States Environmental Protection Agency (2012) has found that exposure to radon as a background source is the second leading cause of lung cancer within the USA. In regard to the previous statement, the Australian Radiation Protection and Nuclear Safety Agency (2011b) indicate radon's contribution to background radiation in Australia is generally a smaller component than that of overseas locations.

As an indicative guide the radiation doses that humans may receive are shown in Table 7.1. Additionally Table 7.2 shows the current Australian Radiation Protection and Nuclear Safety Agency's (ARPANSA) consensus on the risks associated with low dose exposures.

Table 7.1: Radiation doses in perspective. (Australian Radiation Protection and Nuclear Safety Agency (2011a))

Source Of Exposure	Exposure
Natural Radiation (terrestrial and airborne)	1.2 mSv per year
Natural Radiation (cosmic radiation at sea level)	0.3 mSv per year
Total Natural Radiation	1.5 mSv per year
Seven hour aeroplane flight	$0.05 \mathrm{~mSv}$
Chest X-ray	$0.04 \mathrm{mSv}$
Nuclear Fallout (from atmospheric tests in 50's & 60's)	$0.02 \mathrm{mSv} \mathrm{per} \mathrm{Year}$
Chernobyl (people living in Control Zones near Chernobyl)	10 mSv per year
Cosmic Radiation Exposure of Domestic Airline Pilot	2 mSv per year

Man's Exposure to Ionising Radiation

Table 7.2: Health risks associated with low doses. (Australian Radiation Protection and Nuclear Safety Agency (2011a))

Health Risks Arising from Low Doses of Ionising Radiation

Effect	\mathbf{Risk}	Normal Incidence	
Risk of cancer from 1 mSv of radiation	1 in 17,000*	57 in 77000**	
Risk of severe hereditary effect from $1~{\rm mSv}$	1 in 77,000	1,770 in 77,000	
of radiation			
*Age standardised lifetime probability for whole population			
**Age standardised lifetime incidence rate for whole population			
(not necessarily fatal)			
The risk of obtaining cancer from 1 mSv of radiation exposure is equivalent			
to the risk of getting cancer from smoking approximately 100 cigarettes.			

7.2 Radiation Safety

Table 7.3: Typical values of effective dose for various medical x-rays. (Australian Radiation Protection and Nuclear Safety Agency (2011a))

Medical x-ray type	Dose	Medical x-ray type	Dose
	mSv		mSv
Body CT	10.6	Thoracic Spine	1.0
Cardiac Angiography	7.0	Pelvis/Hip	0.6
Lower Gastrointestinal	6.0	Screening Mammography	0.4
Upper Gastrointestinal	3.0	Cervical Spine	0.2
Head CT	2.6	Limbs and joints	0.07
Lumbar Spine	2.0	Jaw	0.07
Cerebral Angiography	2.0	Chest radiography	0.025
Abdomen	1.0	Dental: Intraoral	0.008

Medical Exposures

Without doubt the largest dose of radiation the average person receives is due to medical diagnostic examinations, and in most instances few consider the risk. Table 7.3 depicts the approximate effective dose for some of these diagnostic examination procedures.

Taking into consideration the larger sources at the Boddington Gold Mine, the aforementioned tabulated values are considerably higher than what could be expected in the vicinity of site instrumentation. As an example a larger 9250 MBq caesium-137 sealed source installed on a 600 mm pipe for density measurement has the name plated value for equivalent dose at 1 m of 1.3 μ Sv/h. In addition legislation in most jurisdictions refers to ARPANSA's code of practice on the safe use of fixed radiation gauges, stating that the equivalent dose rate at 5 cm from the source housing shall be less than 500 μ Sv/h and at 1 m shall be less than 10 μ Sv/h. Ionising radiation interacts with cells in the human body in a number of ways, these include:

- Direct interaction with important cell molecular structures such as DNA, RNA and proteins.
- Interaction with the cytoplasm (fluid surrounding the nucleus of a cell) that results in the production free radicals. These free radicals may then further react with other important molecular cell structures such as DNA, RNA and proteins.

The consequences of the above interactions may result in cell death or cell mutation, which may degrade cell function and/or via cell multiplication eventually develop into cancers.

The effects of exposure to ionising radiation on the individual can be separated into two groups, namely deterministic and stochastic effects.

Deterministic effects are typically due to a high dose exposure, with the subsequent outcome related to the amount of exposure. Deterministic effects are the effects that may be directly linked to a radiation exposure event. These effects may include burns, cataracts and radiation sickness and will not be present until a certain dose is exceeded; typically this dose would exceed 1 Sv in a very short time.

Stochastic effects are the effects that may be attributed to chance. These are generally due to lower doses accumulated over time. Apportioning the cause of a particular illness to radiation is generally difficult, due to the other influences present in our environment and cancers may not appear until many years after exposure. The International Commission on Radiation Protection and others estimate the fatal cancer risk as 5% per sievert exposure for a population of all ages (World Nuclear Association 2012). In addition no evidence has been found in humans that radiation exposure may result in genetic abnormalities.

The effects and risks of low level radiation exposure are based on linear hypotheses, with data based on groups that have had significant exposures. Subsequently current practice maintains what may be perceived as a conservative approach, considering any dose to have risk associated with it, even though "Some evidence suggests that there may be a threshold below which no harmful effects of radiation occur" (World Nuclear Association 2012).

Radiation	Shielding material
alpha	None skin is a sufficient protective layer, may be of concern
	when ingested.
beta	Aluminium or Plastic/Perspex higher density materials may
	result in the secondary production of bremsstrahlung radia-
	tion.
neutrons	Materials that have a high concentration of hydrogen atoms;
	water, paraffin and concrete
gamma rays &	Materials of high atomic number; typically lead and steel
x-rays	

Table 7.4: Shielding materials dependent on radiation type.

Shielding Materials for Radiations

Radiation exposure may be limited by:

- Limiting exposure time.
- Increasing distance between individuals and the radioactive source.
- Shielding; applicable shielding materials will significantly attenuate exposure levels via the radiation's interaction with the shielding material.

Studies of radiation's interaction with matter have determined that shielding requirements are dependent on radiation type. Applicable shielding materials can be viewed in Table 7.4.

An extensive search of radiation incident reports covering decades, published by both ARPANSA and the Radiological Council of Western Australia failed to find a single radiation induced injury from exposure due to a fixed static source (sealed radioisotope) used in a mineral processing environment. Although no radiation induced injuries due to x-rays have been associated with mineral processing in Australia, instruments using x-ray tubes have resulted in significant radiation injuries overseas and in other industries. Whilst x-ray tube instruments are easily switched off; the x-ray output is generally of lower energy, but has high intensity and all installations require phys-

7.2 Radiation Safety

ical isolation by enclosures and interlocks. These devices should never be run with enclosures removed and/or with interlocks bypassed.

Great care must be taken in respect to the security of, transportation and disposal of radioactive devices. Worldwide, radioactive devices have been lost or stolen with dire consequences, and in several instances have resulted in death. Although most sources used in mineral processing applications are at the lower end in activity, an unshielded radioisotope in close proximity could still have fatal consequences. Throughout the world many of these lost sources have been found in scrap metal and have even been further processed. On the subject of security, any terrorist act using only a small radioactive device in the form of a "dirty bomb", would result in hysteria and anxiety within the general public, cause major disruption, require a significant clean up effort and may possibly have a significant economic impact (Colella et al. 2005).

Dose limits have been set by regulators inline with International Atomic Energy Agency recommendations. A basic overview of the dose limits set by the Radiological Council of Western Australia can be viewed in Table 7.5; the full fact sheet may be viewed in Appendix B.

Survey meters are used to take measurements of exposure rates and are available for all radiation types. When conducting gamma surveys, many newer meter models may be capable of determining dose rate between different types of isotope; however older models due to detector type are typically calibrated to match the isotope of interest (often Cs-137) and allowances will be required for the ionising energy differences between isotopes. Table 7.5: Radiation dose limits. ((Radiological Council 2012b))

Radiation worker dose limits

- in any period of 5 years, an average effective dose of 20 millisieverts per year.
- in a period of 12 months, an effective dose of 50 millisieverts.

• in any period of less than 12 months but not less than 1 month, an effective dose of the amount which is the product of 50 millisieverts and the ratio of that period in weeks to 52 weeks.

• in any period of less than 1 month, an effective dose of 1/12 of 50 millisieverts.

Non-radiation worker dose limits

- \bullet in any period of less than 1 month, an effective dose of 1/12 of 50 millisieverts.
- in any period of 5 years, an average effective dose of 1 millisievert per year.
- in any period of 12 months, an effective dose of 5 millisieverts.
- in respect of an area which such persons might continuously occupy
- an effective dose of 20 microsieverts in any 1 hour; and
- an effective dose of 250 microsieverts in any period of 7 days.

7.3 Legislation and regulation

Within Australia the regulation of radioactive devices and radiation safety is primarily a state and territory responsibility. On a Commonwealth level this role is undertaken by the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), a Federal Government agency created under government legislation in 1998 and covers identities such as the Defence Force, CSIRO and Federal Police. The details for each jurisdiction can be viewed in Appendix C.

Each jurisdiction has passed some form of "Radiation Safety Act" that covers the use of radioactive materials and generally includes the requirements of;

- Registration
- Radiation Safety Officer
- Radiation Management Plan
- Radioactive Waste Management Plan
- Licensing

In addition the use of ionising equipment within mining and mineral processing may also be subject in each location to some form of Mining Act, Nuclear Waste Storage Act and Transportation Act.

Following enactment, relevant regulations, recommendations and codes of practice have been established, these generally include;

- Radiation safety
- Source security
- Reporting requirements
- Transportation

One of the aims of the Federal Government when creating ARPANSA was to establish more uniformity across all jurisdictions. Existing standards, recommendations, national codes of practice and safety guides are being revisited and are now placed under ARPANSA's Radiation Health Committee. Relevant publishings are generally listed under the "Radiation Protection Series" and include:

- Recommendations for Limiting Exposure to Ionising radiation and National Standard for Limiting Occupational Exposure to Ionising Radiation (Recommendations & National Standard) - Radiation Protection Series No. 1
- Safe Transport of Radioactive Material (Code of Practice) Radiation Protection Series No. 2
- Safe Transport of Radioactive Material (Safety Guide)) Radiation Protection Series No. 2.1
- Radiation Protection and Radioactive Waste Management in Mining and Mineral Processing - Radiation Protection Series No. 9
- Monitoring, Assessing and Recording Occupational Radiation Doses in Mining and Mineral Processing (Safety Guide) - Radiation Protection Series No. 9.1
- Security of Radioactive Sources (Code of Practice) Radiation Protection Series No. 11
- Safe Use of Fixed Radiation Gauges (Code of Practice) Radiation Protection Series No. 13
- Management of Naturally Occurring Radioactive Material (Safety Guide) Radiation Protection Series No. 15

Even though national standards are in place, regulations and practices may vary somewhat from jurisdiction to jurisdiction; this is particularly applicable to the transport of radioactive materials when crossing boundaries.

7.4 Chapter Summary

From an appreciation of the levels of radiation involved with respect to regulation and codes of practice, it is believed the use of radioactive devices within mineral processing environments within Australia is a safe practice.

Radiation exposure may be limited by reducing exposure times, increasing distances between individuals and sources of radiation, and through the use of appropriate shielding. Shielding must be chosen dependent on radiation type.

Radiation regulation in Australia is applied by jurisdiction at Commonwealth, state and territory levels; with each having individual acts and regulations in place. Although national standards are in place, regulations and practices may vary from jurisdiction to jurisdiction.
Chapter 8

Radiation sources

8.1 Chapter Overview

The sources of radiation that may be found in the mineral processing environment are not soley confined to measurement applications, but may also arise due to the process. This chapter reviews the sources that may be found and their origin. With regard to mineral processing there are three main radiation sources of interest, these include;

- 1. naturally occurring radioactive materials that are part of the process,
- 2. artificial isotopes, and
- 3. generated x-rays

In addition on very rare occassions radioactive tracers may be used to investigate process flows.

8.2 Naturally occurring radioactive materials

Naturally occurring radioactive materials (NORM) are inherent to the process and typically arise in the mining and processing of uranium and mineral sands. In addition other processes such as alumina production and the floatation of copper may result in increased concentrations of NORM in the waste products.

The isotopes of interest include; uranium, thorium and the associated decay products. Many radiation types may be involved when dealing with NORM and will generally include alpha, beta and gamma radiation emissions.

Measurements of intensity are monitored to ensure workplace and public safety. Additionally intensity measurements are also often used when mining uranium ores to determine ore grades; this data is further used to manage the treatment process.

8.3 Artificial isotopes

All isotopes used in mineral processing applications are artificial, with many being the by-products of nuclear reactors. The isotopes of interest are shown in Table 8.1 with reference to physical characteristics and applications. Many different decay mechanisms and energies levels are possible during an isotopes decay, with ratios and values varying significantly; subsequently in this table decay mechanisms and energy values are limited to those most prominent.

The shielding materials for different isotopes are dependent on radiation type. Notably caesium-137 and cobalt-60 isotopes are the most common sources found in both mineral processing and industrial environments.

The physical size of a radioactive source used in mineral processing applications is quite small, typically less than 2 cm long and 1 cm in diameter. With the exemption of specialised applications such as those found in mineral analysis, gamma producing isotope sources are contained inside of a source holder that encapsulates the source with sufficient shielding so as to attenuate the radiation in all but the direction of incidence. Collimation is also used to provide shielding and to focus the radiation beam on the measurement point. In addition, to allow for safe access, storage and transportation a shutter mechanism within the source holder can be actuated so as to shield the radiation in the incident direction. Fig. 8.1 and Fig. 8.2 show typical physical and sectional housing arrangements.

Table 8.1: Artificial isotopes used in mineral processing.

(Nationial I	Nuclear Data	Centre, Bro	okhaven Nat	ional Laborator	y 2012)
--------------	--------------	-------------	-------------	-----------------	---------

Isotope	Decay	Emission	Half life	Applications
		Energy	(years)	
		(MeV)		
Fe-55	EC 100%	$\gamma 0.126$	2.744	XRF
Co-60	β^- 100%	$\beta^{-} 2.824$	5.271	Level, density, belt
		$\gamma \ 1.173$		weighing
		$\gamma \ 1.332$		
Cd-109	EC 100%	$\gamma 0.088$	1.26	XRF
Cs-137	β^- 100%	$\beta^{-} 1.175$	30.17	Level, density, belt
		$\gamma 0.662$		weighing
Pu-238	$\alpha \ 100\%$	lpha 5.593	87.7	XRF
Am-241	$lpha \ 100\%$	$\alpha 5.638$	432.6	XRF
Cm-244	$lpha \ 100\%$	$\alpha 5.901$	18.1	XRF
Cf-252	α 96.91%	α 6.216	2.645	NAA
	SF 3.09%	neutrons:		
		2.3 (average)		

Notes*:

1. Decay mechanisms: α - alpha; β^- - negative beta; γ - gamma;

EC - electron capture; SF - spontaneous fission

2. Applications: XRF - X-ray fluorescence; NAA - Neutron Activation Analysis



Figure 8.1: Radiation source physical view.(Berthold Technologies 2004)



Figure 8.2: Radiation source sectional view.(Berthold Technologies 2004)

Due to the nature of gamma and x-rays, lead is used as a commercially viable shielding material. The higher the emission energy the greater the thickness of shielding required, subsequently for the same source activity cobalt-60 sources require a greater amount of shielding than caesium-137 sources. Even though the radioactive source may be physically very small, the required source holder can be very heavy and this in itself provides significant risk when considering the installation, maintenance and transportation of the device. As an example of the weights involved, Berthold's CR series of source holders can weigh from approximately 31 kg through to approximately 170 kg.

Radioactive isotopes are typically used in the all nuclear measurement applications associated with mineral processing fields. Advantages include reliability and no requirement for power supplies.

8.4 X-rays

Alternatively ionising electromagnetic radiation in the form of x-rays can be produced through the use of x-ray tubes.

The fundamental principle of an x-ray tube is based on the interaction of fast electrons with a target material that will as a result of the interaction produce:

- 1. Bremsstrahlung radiation. X-rays emitted as a consequence of the rapid deceleration that occurs when the fast electrons interact with a target material of high atomic number.
- 2. Characteristic X-rays. A small proportion of fast electrons colliding with a target material will typically eject electrons from the inner K shell of the target material's atoms; as a result electrons of higher energy from the outer shells of these atoms race to fill the vacancy, the energies released in achieving a stable state are emitted in the form of x-ray photons. X-rays resulting from the transition to a stable state are said to be characteristic as the energies released are elemental dependent.

Due to the interactions involved, the energy imparted will vary and thereby results in the production of a continuous x-ray spectrum. The maximum spectrum x-ray energy value is directly related to the applied high voltage; with the maximum value in keV equal to the peak voltage value, kVp applied. Fig. 8.3 depicts the x-ray output spectrum that can be expected from an x-ray tube operated at 150 keV with a tungsten target anode.

Filters may be used to attenuate parts of the spectrum, but this also has the effect of reducing the output intensity of the tube. The construction of an x-ray tube primarily



Figure 8.3: Radiation source physical view.(Labspace n.d.)

consists of a glass or metal ceramic tube under vacuum that contains; a cathode that produces a source of electrons and an anode made of the target material that collects and interacts with the electrons emitted by the cathode. X-rays leave the tube via a beryllium window that serves the purpose of sealing the tube and providing a path that has a minimal effect of x-ray energies. The cathode generally consists of a tungsten heating element that liberates electrons when current is applied. These electrons are substantially accelerated by the application of a high voltage across the cathode and anode. By placing a positive high voltage with reference to the cathode on the anode, electrons are accelerated due to the forces of attraction. Tube voltage forms are dependent on design; all newer devices are typically DC with all AC x-ray tubes only producing x-rays when the positive half cycle is imposed on the anode. The basic concept of the x-ray tube can be viewed in Fig 8.4.



Figure 8.4: Operating principle of an x-ray tube. (U.S. Geological Survey 2001)

Energy levels of the emitted radiation are heavily influenced by the velocity of the incident electrons impacting the anode. With acceleration dependent on the high voltage applied, the higher the voltage applied the greater the acceleration and as a consequence the greater the outgoing x-ray photon energy.

The intensity of the x-rays produced is directly proportional to the tube current, in addition the number of x-rays produced is greatly influenced by the value of the high voltage applied. According to Stanford University (n.d.) an x-ray tube's maximum intensity can be determined using Ulrey's (1918) formula of:

$$I = AiZV^2 \tag{8.1}$$

where I = intensity (relative units)

- A =empirical constant
- i =tube current
- Z = atomic number
- V = high voltage applied

To obtain good control of x-ray intensity requires the use of precise voltage supplies for both high voltage and filament circuits, and precise components to control filament current.

The heat energy that arises from electron interaction with the anode is substantial; typical values are approximately 99% of all input energy and hence for all but the smallest x-ray tubes some form of anode cooling circuit is required. Cooling is often achieved through the use of a copper backed target anode and a water cooling circuit. In addition the target material of the anode is more often than not tungsten; this is specifically due to its inherent high melting temperature and its high atomic number. The major advantages of x-ray tubes include:

- The ability to isolate the source of radiation by simply switching off the electrical power supply which results in minimal transport, storage and disposal issues.
- As x-rays have a focal point, source shielding requirements are minimized.

The high radiation intensity values that may be produced at an x-ray tube's output window are of great concern; this significant safety risk is reduced through the use of shield enclosures surrounding the tube and associated interlocks. Currently x-rays are only used in the field of elemental analysis. The applicability of the use of x-ray tubes in mineral processing applications is limited due to;

- the maximum energy output with respect to economics,
- cooling requirements,
- power requirements, and
- reliability

8.5 Source sizing

Manufacturers typically quote the intensity values required in air at the detector to satisfy an application. Radioisotope source selection is typically made application dependent on attenuation. To evaluate the activity of the source required, the amount of attenuation between source and detector will require assessment. This attenuation value on most occasions will be calculated based the inverse square law for air using gamma constants and the previous discussed methods for determining attenuation of absorbing materials other than air. Minimum source sizing with no allowance for decay can be found using many different methods based on the determination of attenuation covered in Chapter 6. The method preferred by the author is an adaptation of the following dose rate equation from Berthold Technologies (2003) LB 444 density meter operating manual, revision 4;

$$X = \Gamma_{\delta} \frac{\alpha}{d^2} S \tag{8.2}$$

where X = dose rate; mSv/h

 Γ_{δ} = specific gamma ray dose constant; (mSv.m²)/(h.MBq) α = source activity; MBq d = free air distance; m S = attenuation factor

This equation can be manipulated to determine the minimum source size required for detector operation, with X now equal to the dose rate required at the detector in mSv/hr.;

$$\alpha = \frac{Xd^2}{\Gamma_{\delta}S} \tag{8.3}$$

An example determining the minimum source size required using caesium-137, with a free air distance of 0.6 m, vessel walls of 12 mm steel and 1 μ Sv required at the detector follows:

Firstly determining the attenuation factor: Half thickness value for steel according to Berthold Technologies (2003) is 14 mm, total steel thickness equals $2 \ge 12$ mm, and therefore the number of half thickness layers is equal to 1.7143. Using the equation;

$$S = 0.5^{n}$$
 (8.4)
 $S = 0.5^{1.7143}$
 $S = 0.3048$

now with $X = 1 \,\mu\text{Sv/h}$; dose rate required at the detector $\Gamma_{\delta} = 1.07 \times 10^{-4} \,(\text{mSv.m}^2)/(\text{h.MBq})$; specific gamma dose constant d = 0.6 m; free air distance S = 0.3048; attenuation factor $\alpha = 1 \frac{\mu\text{Sv}}{\text{h}} \times 0.6^2 \text{m}^2 \times \frac{1}{1.07 \times 10^{-4}} \frac{\text{h.MBq}}{\text{mSv.m}^2} \times \frac{\text{mSv}}{1000\mu\text{Sv}} \times \frac{1}{0.3048}$

$$\alpha = 11.04 \text{ Mbq}$$

The calculated activity in this case is the bare minimum required to achieve satisfactory operation. Sources will be typically sized to meet a specific life, if the working life is set to match the half-life this calculated minimum activity value would require doubling. In addition sources come in a range of set values and should be selected with "ALARA" principles in mind. As a rule manufactures/suppliers (not the client) calculate source sizes to match specific applications.

8.6 Chapter Summary

Naturally occurring radioactive materials (NORM) are inherent to the process and may include several different radiation emissions.

Sealed radioactive isotopes used in measurement applications are by far the most common radioactive sources. Of these caesium-137 and cobalt-60 are the most prevalent and are used as gamma sources. The major advantage of these sealed sources is reliability and the non-requirement of a power supply.

X-ray tubes have limited applications and are used typically in the field of elemental analysis. The major advantage of x-ray tubes is the ability to remove the source of radiation by switching off the power, which results in minimal transport, storage and disposal issues.

Calculating the activity of the source required for an installation must take into account the exposure/dose required at the detector, the amount of attenuation between source and detector and the design life of the application.

Chapter 9

Detectors

9.1 Chapter Overview

This chapter covers the types of detectors typically used to detect radiation in mineral processing environments. Predominately the radiation detection methods used for process measurement and analytical instruments throughout mineral processing activities are based on the detection of gamma rays and x-rays. In the main this is achieved through the use of Geiger-Muller tubes and scintillation detectors. Past methods of detecting radiation have included the use of ion chambers, which are in some way similar in operation to Geiger-Muller tubes. Ion chambers have not been covered in this document as it appears no modern devices use this measurement detection technique due to the lower sensitivities involved. In addition within this chapter modern detection technique susing semiconductors are also reviewed.

9.2 Geiger Muller tubes

The use of Geiger-Muller (GM) tubes within mineral processing is very much restricted to point level and survey meter applications. This restricted use is due to the output of the tube, as it can only act as a counter of ionising events and can not differentiate between ionising energy levels. The following description of the operation and

9.2 Geiger Muller tubes

characteristics of Geiger Muller tubes is based on information obtained from "Radiation Detection and Measurement" (Knoll 2010, chap. 4-7). GM tubes are not a new technology; they are rugged, simple and economical with a proven track record and hence are still popular for tasks solely dependent on registering a count rate below a few hundred pulses per second. Importantly it must be further recognized that some counting data may be lost due to the inherent nature of radioactive decay and the device. GM tubes are part of a family of radiation detectors commonly known as gas filled detectors, this family also includes ionisation chambers and proportional counters. Ionisation chambers and proportional counters have the ability to differentiate between energy types, but both also require more complex and more sensitive electronics whilst being nominally fragile. The GM tube consists of a very thin axial wire anode within a gas filled cylindrical cathode. The general arrangement of a GM tube can be viewed in Fig. 9.1, note supply voltage varies with tube design.



Figure 9.1: Geiger Muller tube overview.(Gamma Geiger Counter 2012)

Gases inside the GM tube are normally noble gases of either helium or argon. The first ionising energies of these gases from Knoll (2010) are 24.5 keV and 15.7 keV respectively. These fill gases are generally under a slight vacuum of approximately 0.7 atmospheres.

The GM tubes principle of operation utilises the phenomenon that when ionising radiation enters or interacts with a suitably gas filled tube that has a sufficiently high electric field applied, the initial ionising event will be multiplied due to the subsequent excitation of gas molecules. The electric field is achieved by the application of high DC voltage in the vicinity of 500 to 2000 volts. *Gas multiplication* is only possible when a significantly substantial electric field is present. Fig. 9.2 depicts a plot of two different radiation energies with respect to gas filled detector type and multiplication factor; this plot shows that the Geiger region has a multiplication factor in the vicinity of 10^9 to 10^{10} and that in this region the different energies can not be distinguished from each other.



Figure 9.2: Gas filled detector plot of ion multiplication against applied voltage; (a): 1 MeV β particles; (b): 100 keV β particles. (Podgoršak 2005)

The initial ionising event will create free electrons and positive ions along the ionising tract that are referred to as ion pairs, the number of ion pairs produced is proportional and dependent on the energy of the incident radiation. Free electrons will rapidly move to the positively charged anode, whilst much slower moving positive ions will travel towards the cathode. As the electrons rapidly move towards the anode they are accelerated and gain additional kinetic energy due to the high electric field, subsequent collisions impart energy on gas atoms which quickly (nanoseconds) de-energise via photon discharge with an energy that is within the visible to ultra-violet range. These

9.2 Geiger Muller tubes

lower energy photons interact with the fill gas and possibly tube walls to create many more free electrons; this action creates an "Avalanche" of electrons around a portion of the anode wire, with further avalanches occurring along and within the vicinity of the anode wire due to photon interaction. This whole process results in "Geiger discharge", a pulse output of several volts that occurs within microseconds.

With regard to the electrons, the positive ions move extremely slowly and will stop the Geiger discharge; this is due to the positive ion build up that consequently reduces the electric field strength. After the discharge pulse it is possible that the neutralisation of the positive ions on the cathode wall will result in the production of a free electron thus resulting in false detection and continuous avalanches. To avoid this situation a quenching gas of approximately 10% of tube volume is often introduced. This chosen gas has a high charge exchange rate with the positive ions of the primary fill gas, thereby inhibiting atoms of positive charge interacting with the cathode wall. Common quenching gas such as ethanol and halogen. The disadvantage of using an organic quenching as such as ethanol is that it is eventually consumed, which results in the GM tube having a finite life. The use of halogens has the advantage in that they are capable of replenishing themselves at some time after disassociation and resolve the issue of finite tube life due to quench gas.

Alternative to the use of quenching gas is the use of an external circuit to switch off the electric field after the occurrence of an output pulse, positive ions neutralise with the cathode wall, but any resulting free electrons will not cause Geiger discharge. This can be achieved through the use of an RC circuit, a large resistance in the order of 10 to 100 megaohms is placed in series with the high voltage supply and in combination with the capacitance of the tube and associated electronics (refer to Fig. 9.1, R1). After the pulse discharge, the RC circuit creates a time delay in the order of milliseconds that will prevent the supply reaching its full potential and hence prevents gas multiplication.

Dead time in a GM tube is chiefly due to the time it takes for the positive ions that stopped the Geiger discharge to dissipate, this dead time is required so as to allow the recognition of a second ionisation event; dead time is usually in the range of 50 - 100µs. In respect to other forms of radiation detection this is a significant amount time.

GM tube design is dependent on the use of the equipment. For the detection of lower

9.2 Geiger Muller tubes

energy radiations a fragile thin window type is usually employed. Tubes designed for higher radiation energy levels such as gamma rays, commonly have no window and rely on the interaction of gamma rays with the cathode wall. Due to use at higher energy levels GM tubes installed in mineral processing applications have efficiencies in the low percentages. Bismuth with an atomic number of 83 has been a popular choice for the cathodic material, as the likelihood of gamma ray interaction with a tube's walls is increased with an increase in the wall material's atomic number. Notably as the energy level decreases the efficiency of the GM tube increases. Taking the aforementioned into account, GM survey meters are often used to indicate exposure and therefore calibration must match the source of radioactivity; this is due to the fact that even though unlike sources may have similar activities, differing emission energies occur from unlike isotopes.

To achieve Geiger discharge, a minimum ratio of electric field strength to gas pressure must be achieved and this is dependent on the gas mixture within the tube. A reduction in gas pressure has the desired effect of increasing this ratio.

The GM tube's primary design equation takes the form of Diethorn's equation; this models the relationship between the tube's physical characteristics, the fill gas characteristics, the ion multiplication factor, the applied voltage and lastly the ratio of electric field strength to gas pressure. The equation is as follows:

$$\ln M = \frac{V}{\ln(b/a)} \cdot \frac{\ln 2}{\Delta V} \left(\ln \frac{V}{pa \ln(b/a)} - \ln K \right)$$
(9.1)

where M = the gas multiplication factor

V	=	applied voltage

- a = anode radius
- b = cathode radius
- p = gas pressure in atm
- ΔV = the potential difference through which an electron moves between successive ionising events

$$K$$
 = gas constant that represents electric field to gas pressure ratio

 ΔV and K are both gas constants of the fill gas and are referred to as Diethorn parameters. A list of common fill gas constants can be found in Table 6.1 of "*Radiation Detection and Measurement*" (Knoll 2010, p. 172).

Determination of the high voltage setting is achieved by initially plotting the count rate against an adjustment in voltage. Once these values have been taken, the Geiger plateau should now be recognizable and high voltage settings should be set in the middle of the Geiger plateau. In the example shown in Fig. 9.3, the setting should be approximately 950 Vdc; it must be remembered that the plateau's values will change with tube design.



Figure 9.3: Example of count rate against voltage for a Geiger Muller tube. (University of the Fraser Valley 2012)

Due to the GM tube's relatively large amplitude output pulse (volts), the counting electronics are relatively simple, with no signal amplification generally required.

Ultimately even with methods in place to prevent the consumption of quenching gases, GM tubes still only have a finite life due to the reactions that occur, these include and are not limited to contamination of the fill gas and changes to the anode surface.

9.3 Scintillation Techniques

In essence all radiation measurement devices used in mineral processing other than and in some cases including point level, use scintillation methods for radiation detection. A scintillator consists of a material that has the required characteristic of converting incident radiation to visible light pulses. Through the use of light detection methods after this conversion, it is possible to not only detect the ionising radiation, but to additionally gain energy information of the said radiation. The concept of scintillation detection can be viewed in Fig. 9.4; the system basically consists of a scintillator and a light sensor. The visible light produced is generally inline with lower end of the visible spectrum with a wavelength of 400 nm through to the upper end of the green light spectrum wavelength of approximately 560 nm.



Figure 9.4: Scintillator detection principle.

9.3.1 Scintillators

The emission of visible light photons is a result of the de-energisation from higher energy states. Dependent on scintillator type all radiations of general interest, including neutrons can be detected.

The quality of the raw phenomenon (visible light) data is a function of the material make up of the scintillator, ideal properties include:

- 1. High efficiency in energy conversion; kinetic energy to light.
- 2. Energy input to light output should be proportional and therefore linear.

- 3. Transparency to the material's output wavelength.
- 4. Short decay times on the return to initial molecular states; this is an important requirement, as the induced fluorescence must switch off quickly if fast count rates are to be measured.
- 5. Have good optical qualities, with a refractive index close to that of glass.
- 6. The ability to be produced at a size sufficient to suit detector production.

Currently no scintillator can meet all of the above requirements for a particular application, so choice is often a matter of compromise.

The efficiency of the scintillator is a measure of the light output energy against the incident energy. Typically scintillator efficiencies are quite low, often under 10%; an example of efficiency determination follows:

Using a NaI(TI) inorganic crystal with a light yield of 38000 photons per MeV of incident radiation and an output wavelength of 415 nm: Firstly determining output energy per photon;

$$E_{out} = \frac{1.24 \times 10^{-6}}{\lambda} \tag{9.2}$$

where E_{out} = output energy; eV/photon λ = wavelength; m 1.24×10^{-6} = constant for determination of energy in eV

$$E_{out} = \frac{1.24 \times 10^{-6}}{415 \times 10^{-9}} = 2.988 \text{ eV/photon}$$

Scintillator efficiency is equal to:

$$\eta = \frac{n \times E_{out}}{E_{in}} \tag{9.3}$$

where η

 η = efficiency

n = number of photons per MeV incident E_{out} = output energy; eV/photon E_{in} = Incident radiation energy in this case; 1 × 10⁶ eV

$$\eta = \frac{38000 \times 2.988}{1 \times 10^6} = 0.114$$

9.3 Scintillation Techniques

Scintillators can be separated into two groupings; organic and inorganic. Each has their advantages and disadvantages; notably organics have faster response times, whereas inorganics generally have the better output properties of efficiency and linearity. It should also be noted that no scintillator is fully linear over its entire energy range. Table 9.1 shows a property comparison of commonly used scintillators; this data is a compilation taken from both Knoll (2010, p. 230, p. 238) and Cooper (1986, p. 58). Light output is relative to that of the thallium iodine doped inorganic crystal sodium iodine, NaI(TI).

Organic scintillators can be further subdivided into three categories, crystalline, liquid and plastic. According to Knoll (2010, p. 228) organic scintillators are generally useful for detecting alpha and beta radiation, additionally they can be readily adapted to detect fast neutrons. Organics are primarily excited by the incoming ionising radiation that changes state at a molecular level and de-excite by the release of visible light.

Pure organic crystals that are commonly used as scintillators are Anthracene and Stilbene. Both of the aforementioned have limitations which include; fragility, efficiency in response to radiation due to crystal grain alignment, and size limitations.

Liquid organic scintillators are created through the dissolution of an organic scintillator with an appropriate solvent; the benefit of this is that a large scintillator may be produced that can withstand radiation doses that would otherwise damage many other types of scintillator. These liquids are usually encapsulated in glass.

Plastic scintillators are the most popular type of organic scintillator and are created by the dissolution of an organic scintillator with a suitable solvent, which is subsequently polymerized to form a solid plastic. Benefits of plastic scintillators include large production sizes and the inherent ability to be easily shaped. A major disadvantage is that damage to the scintillator may occur over time due to radiation exposure, this will result in lower and poorer light output quality. Organic scintillators will not provide the required output when the incident radiation takes the form of gamma or x-rays without modification. Successful modification of liquid and plastic scintillators can be achieved through the dosing of elements of high atomic number, such as lead or tin. Modification of the organic scintillator has a downside as it results in a reduction in light output; however the result is a method that can detect gamma or x-rays at a relatively low expense with a fast response.

Name	Type	\mathbf{Use}	$\mathbf{Max} \ \lambda$	Density	Decay	Relative
			(nm)		\mathbf{time}	light
					(ns)	output
NaI(TI)	inorganic	gamma &	415	3.67	230	1.00
	crystal	x-rays				
CsI(TI)	inorganic	gamma rays	540	4.51	680	0.49
64%	crystal	& heavy par-				
		ticles				
BGO	inorganic	gamma &	480	7.13	300	0.13
$\mathrm{Bi}_4\mathrm{Ge}_3\mathrm{O}_{12}$	crystal	x-rays				
LiI(Eu)	inorganic	neutrons	470	4.08	1400	0.23
	crystal					
ZnS	Crystalline	alpha parti-	450	4.09	200	1.30
	powder	cles				
$LaCl_3(Ce)$	inorganic	gamma &	350	3.79	28	>1.00
	crystal	x-rays				
$LaBr_3(Ce)$	inorganic	gamma &	380	5.29	26	>1.00
	crystal	x-rays				
Anthracene	organic	alpha, beta,	447	1.25	30	0.435
$\mathrm{C}_{14}\mathrm{H}_{10}$	crystal	gamma &				
		fast neutrons				
Stilbene	organic	gamma &	410	1.16	4.5	0.217
$\mathrm{C}_{14}\mathrm{H}_{14}$	crystal	fast neutrons				
NE213	organic	fast neutrons	425	0.874	3.2	0.339
BC-501	liquid					
NE102A	plastic	alpha, beta,	423	1.032	2.4	0.283
BC-400		gamma &				
		fast neutrons				
NE908	glass	slow neutrons	395	2.674	75	0.09

Table 9.1: Properties of commonly used scintillators. Knoll (2010) and Cooper (1986)

9.3 Scintillation Techniques

Inorganic scintillators consist of crystals commonly grown in a laboratory or manufacturing facility. The greatest use of these scintillators is in the detection of gamma rays and x-rays.

Inorganic scintillators' methods of scintillation are somewhat different to that of organic scintillators, with energy levels dependent directly on the crystal lattice. Electrons of un-doped inorganic crystals have distinct energy bands at which they can exist; additionally these bands are separated by forbidden zones in which electrons can not exist. Ionising radiation will have the effect of lifting electrons from the lower state bound in the valence band to the conductance band where electrons are free to roam. However on de-excitation the electrons will return to the valence band with insufficient light emissions or with emissions outside the visible spectrum. To achieve light emissions an activator is added that creates another possible energy level within the forbidden zone of the crystal. This has the effect of the returning a de-exciting electron via a smaller energy step to the forbidden zone before returning to the valence band, with the subsequent emission of visible light. A very common inorganic scintillator is thallium iodine doped sodium iodine, NaI(TI); although first bought to attention in 1948 (Knoll 2010, p. 239) it is still in wide use today. It is still popular due to cost economics and manufacturing volumes.

Many newer inorganics can provide substantial improvements in output, resolution and with faster count rates, but are often deemed unnecessary due to the additional cost. The majority of scintillators found in industrial environments are inorganic crystals; this is primarily due to the measurement applications of determining gamma ray and x-ray radiations.

9.3.2 Light detection

The light emitted from a scintillator is converted to a suitable electrical output using either a photomultiplier tube or photodiodes. Photomultiplier tubes are by far the more common of the two methods, but are less rugged, less compact and require a high voltage supply. Increasing advances in semi-conductor technology have seen an increase in the popularity of solid state photodiode assemblies.

Photomultiplier tubes

In principle the function of the photomultiplier tube is to convert incoming photons to electrons and then significantly multiply these electrons so as a usable electrical output signal can be produced. A typical photomultiplier tube complete with a NaI(TI) scintillator is illustrated in Fig. 9.5.



Figure 9.5: Photomultiplier tube assembly complete with scintillator. (Saint - Gobain Crystals 2011)

Operation of the photomultiplier tube is best expressed with reference to an image and this can be viewed in Fig. 9.6. Incident photons interact with the photocathode to release electrons (often referred to as photoelectrons). Through the application of a high voltage, a voltage gradient is achieved across resistors that supply internal dynodes (electrodes). The applied voltage results in the attraction and the acceleration of the released electrons to the first dynode, on reaching the first dynode addition electrons are released, effectively resulting in multiplication. Due to the applied voltage gradient this process repeats again and again until the electrons arrive at the anode giving a



current pulse output that can be further amplified and interrogated by electronics.

Figure 9.6: Operational view of a photomultiplier tube. (University of Illinois 2011)

Lilley (2001, p. 157) states that multiplication of a 10 stage tube can be in the vicinity of 10⁷. Knoll (2010, p. 282) further states that much higher multiplication factors are possible with high yield tubes. The multiplication factor is a function of high voltage. Adjustment of the high voltage will see little to no counts at lower values, a plateau of reasonably consistent counts at the correct operating voltage and an avalanche of counts if set too high. The action of multiplication is not fully repeatable as the electron release is a statistical function about a mean. It can be further stated that when operated at the right point, across the range of input to output values photomultiplier tubes give linear outputs.

The high voltage DC supply and resistor network as shown in Fig. 9.6 are located external to the photomultiplier tube and connect to the pins on the phenolic base (reference Fig 9.5).

The photocathode of a photomultiplier tube is extremely thin, often semi-transparent and commonly made of bialkali; additionally the thickness must be consistent across its surface so as to maintain a uniform sensitivity over the entire area. Efficiency of the photocathode is usually quoted using its quantum efficiency (QE) with:

$$QE = \frac{\# \text{ photoelectrons emitted}}{\# \text{ incident photons}}$$
(9.4)

Generally a photomultiplier tube's quantum efficiency is less than 30%. Quantum efficiency has a direct relationship to the incident wavelength and therefore the incident wavelength must be taken into consideration when selecting a photomultiplier tube. Adapted from Knoll (2010, p. 290) manufacturers typically quote tube performance specifications at a set operating high voltage, these generally include:

- Overall luminous sensitivity, in amps per lumen (A/lm) a ratio of anode current to lumens; determined by subjecting the photocathode to the luminous flux of a tungsten filament at a standardized temperature.
- Cathode luminous sensitivity, in amps per lumen (A/lm) a ratio of photocathode current to lumens; determined by subjecting the photocathode to the luminous flux of a tungsten filament at a standardized temperature.
- 3. Overall radiant sensitivity, in amps per watt (A/W) a ratio of anode current to power intensity from a given wavelength applied to the photocathode.
- 4. Cathode radiant sensitivity, in amps per watt (A/W) a ratio of photocathode current to power intensity from a given wavelength applied to the photocathode.
- 5. Dark current, in amps (A) measurement of anode current without illumination of the photocathode.
- 6. Anode pulse rise time, in seconds a measure of the time required for the anode current to climb from 10 % to 90% of its maximum value once illumination is applied to the photocathode.
- 7. Anode pulse width, in seconds measurement of pulse width at half maximum peak due to the application of a pulse of light.

Rise times and pulse widths limit the count rate of the device. Hamamatsu (2005, p. 19) states that most photomultiplier tubes can measure count rates of up to $3 \ge 10^6$ counts per second.

In addition to the aforementioned performance specifications, manufacturers also often stipulate the following parameters of their devices;

- 1. recommended operating voltage,
- 2. maximum operating voltage,
- 3. wavelength of maximum response,
- 4. gain,
- 5. number of dynodes,
- 6. photocathode efficiency, and
- 7. dimensions

Issues with photomultiplier tubes include;

- 1. susceptibility to magnetic fields,
- 2. fragility, and
- 3. poor uniformity of the photocathode

Due to the low energy of electrons in transit within the tube, external magnetic fields can significantly affect the flight of electrons. This issue can be counteracted to a large extent by the use of a magnetic shield.

Photomultiplier tubes are quite fragile and therefore are extremely susceptible to shock. It is highly recommended that photomultiplier tubes are not installed under these conditions.

Poor uniformity of the photocathode often occurs in larger diameter photomultiplier tubes and can lead to significant differences in count resolution. One of the methods used to avoid this is to use a smaller photomultiplier tube with some device that will focus the light.

9.3 Scintillation Techniques

Photomultipliers have a long life under intended operating conditions; however this life span may be reduced by operation at too high a voltage, high input illuminance and operation at high temperature. A photomultiplier tube should never be exposed to strong light sources (daylight) with voltage applied, as this may cause irreversible damage.

Overall when used correctly photomultiplier tubes provide a tried and proven method for the conversion of light pulses to an electrical signal, with an output pulse amplitude proportional to the light intensity.

Photodiode semiconductor techniques

Although not as popular as photomultiplier tubes, semiconductor techniques are becoming more common. Semiconductor techniques based on the photodiode still require the use of a scintillator. Advantages of photodiode technologies over photomultiplier tubes include;

- 1. greater ruggedness,
- 2. reduced size,
- 3. significant immunity to magnetic fields
- 4. response to a wider range of wavelengths, and
- 5. operation at much lower voltages

Photodiodes can offer higher quantum efficiencies (60-80%) than the photocathode of a photomultiplier tube, however unlike the photomultiplier tube no further multiplication occurs, so the signal is quite small and requires further amplification.

Photodiodes are generally produced from silicon or germanium. The basis of operation of a photodiode is the creation of electron-hole pairs when the semiconductor is exposed to photons. The photodiode is connected in reverse bias with a dc voltage applied; in general the higher the applied voltage the faster the speed of response.

Conventional photodiodes are not well suited for this application as they provide no gain and have poor signal to noise ratios. In addition the semiconductor material can

9.4 Semiconductor techniques without scintillation

react to ionising radiation that may pass through or bypass the scintillator, resulting in output pulses that are very much larger than those produced from the light emitted by the scintillator.

Improvement of the above is achieved by the use of avalanche photodiodes. The operating principle remains the same as the conventional diode, but by increasing the applied voltage, a resultant multiplication of electron-hole pairs (avalanche) occurs. This has the benefit of increasing both the output current and the signal to noise ratio. Repeatable avalanche photodiode output requires both an extremely stable applied voltage and temperature compensation.

An advancement in the use of avalanche photodiodes is to use them in an array with a higher applied voltage that will put the output into runaway (Geiger mode) when a photon incidents the diode. This arrangement is commonly referred to as a silicon photomultiplier or solid state photomultiplier. Construction of the device consists of an array of extremely small avalanche photodiodes as an integral part of a cmos semiconductor device. The number of photodiodes can number in the thousands with diameters of less than 100 microns. The concept of design is based on probability, by reducing the size of the photodiodes and through the use of an array, a photodiode pulse per incident photon is envisaged.

Precise operating voltages are not so much of an issue as these devices run in Geiger mode; however temperature still has an effect on device gain.

Notably these devices are photon counting devices, with similar gains to those of photomultiplier tubes.

9.4 Semiconductor techniques without scintillation

Semiconductor techniques for radiation detection are relatively new and are rapidly evolving. There are currently semiconductor techniques other than avalanche photodiode arrays that can directly measure gamma and x-ray radiations without scintillators; these are generally not well suited for mineral processing applications due to the higher energies involved and/or due to intrinsic restrictions, such as a requirement for cooling. However some spectroscopic analytical methods are implemented that do use semiconductor techniques.

A common semiconductor detector used in radiation spectroscopy is that using the lithium drifted silicon semiconductor. In principle this is a p-n junction diode with a large depletion zone made possible by the introduction of lithium ions into the p-type semiconductor material during the production process. This results in large n-type depletion layer adjacent to the p-type material which is the detector volume of this device.

Electrons in the outer shell of atoms in the semiconductor are known as valency electrons and are said to be distributed in the valency band. A higher energy level band known as the conduction band also exists; this band contains free electrons that can move under the influence of an external potential field. There is a definitive energy gap between the valency and conductive bands, and it is possible to move electrons from the valency band to the conductive band if enough energy is applied. The interaction of x-ray photons with the semiconductor crystal can provide energies high enough to promote valency band electrons to the conduction band. These freed electrons may also impart enough energy to release other electrons from within the semiconductor crystal. In addition the holes left after electron promotion may also move under the influence of a potential difference across the semiconductor. The movement of electrons and holes due to x-ray interaction results in the generation of electrical current pulses. Bias voltages across the detector according to Knoll (2010, p. 469) are in the vicinity of 500 to 4000 V. A simple schematic outlining the construction and operation of a lithium drifted silicon detector can be viewed in Fig. 9.7. Elevated temperatures will produce significant leakage currents in the device that can result in significant noise and will additionally result in the undesirable relocation of lithium ions within the semiconductor. It is essential that these detectors are kept cold; this is usually achieved through the use of liquid nitrogen.

Furthermore radiation damage will eventually result in detector failure.

Devices used for radiation spectroscopy that measure energy distribution return a curve that is representative of the incident energy level. The curve is a typically statistical with counts plotted against energy. The better the energy resolution the greater the



Figure 9.7: Overview of a lithium drifted silicon detector. (Northern Arizona University 2006)

peak, with less deviation either side of the mean energy level. Energy resolution is defined by dividing the full width at half maximum value (FWHM) by the incident energy value. FWHM is the differential energy value taken at half the peak amplitude value; the concept for a Gaussian distribution and the associated value of FWHM can be viewed in Fig. 9.8. Analytical detectors that use energy differentiation methods



Figure 9.8: Full width at half maximum for a Gaussian distribution.

are typically chosen to have the best energy resolution, so as to provide the greatest capability of distinguishing between elements using characteristic energies. As a consequence, at this current time lithium drift silicon detectors remain the detector of choice in this application due to superior energy resolution.

9.5 Chapter Summary

A multitude of methods can be used to detect radiation; detection methods used throughout mineral processing activities are primarily based on the detection of gamma rays and x-rays. In the majority of installations Geiger-Muller tubes and scintillation detectors in conjunction with conventional photomultiplier tubes are used for radiation detection.

Geiger-Muller tube use is very much restricted to point level and survey meter applications. Scintillators in conjunction with photomultiplier tubes can cover most measurement applications. Analytical applications that differentiate between characteristic energies require detectors that possess good energy resolution; this is required so as to provide the greatest capability of distinguishing between elements. The best energy resolution detectors use semiconductor techniques, and require constant cooling to minimise noise and to keep the characteristics of the semiconductor stable.

Chapter 10

Measurement

10.1 Chapter Overview

Nuclear measurement systems have many factors influencing accuracy, precision and repeatability. The contents of this chapter covers the fundamentals of measurement for ionising radiation measurement systems.

Raw signals of radiation intensity are taken as a number of counts and hence all measurements are of a statistical nature. The effects on measurement of background radiation and other inherent attributes are also considered, with the latter part of this chapter dedicated to the basics of signal processing.

10.2 Measurement fundamentals

The factors influencing the accuracy, precision and repeatability of measurement are considerable. Of significant influence is the source of radiation; in nearly all cases the source is a radioactive isotope, where decay is of a statistical nature. In addition, a major external influence on measurement is that of background radiation and in most cases this may be deemed as consistent. In regard to the statistics of measurement, a Gaussian distribution model (Knoll 2010, p. 71) can generally be used due to the typically large mean count values measured.

10.2.1 Measurement Model

When considering the model of a radiation measurement system, background counts can be considered as noise and may have a significant effect when the background counts are at high levels in comparison to the desired signal. Instances where high background count levels may occur include; NORM as part of the process and installations within the vicinity of other nuclear instrument systems.

Contamination inputs to nuclear sensor systems with reference to the COMETMAN acronym may include:

- Nuclear background radiation
- Thermal changes in temperature
- Mechanical changes in pressure
- Electrical changes in electromagnetic field

A model of a typical nuclear measurement system with inputs and outputs referenced to the COMETMAN acronym is shown in Fig. 10.1.



Figure 10.1: Nuclear sensor model applicable to mineral processing environments.

10.2 Measurement fundamentals

Within the sensor system itself, signal manipulation may be affected by changes in parameters such as a drift in high voltage supplies, aging of components and changes in pressure and temperature.

In addition due to the nature of measurement some counts will be inherently lost, this is a result of the detector/electronics not being capable of differentiating between counts (photons) that may arrive at approximately the same time; this may be significant at higher count rates and is non-linear with attenuation. This issue arises due to the dead time of a detector. To allow for the recognition of a subsequent pulse, a certain period of time (dead time) is required after the leading pulse. If the dead time does not elapse the subsequent pulse will not be recognized and any further pulses may also not be recognized until the dead time does elapse. If the dead time does not elapse and counts continue to arrive, the detector will not count and is said to be paralysed. A method to prevent paralysation (non-paralysed) is to reset the detector to count after the dead time would have elapsed, thereby capturing many of the lost counts and possibly registering the maximum count rate of the detector.

To achieve the best possible sensitivity, Johansen & Jackson (2004, p. 249) state radioisotopes of the lowest possible energy to meet an application should be used; this not only reduces the effects of scattering, but also provides a greater signal change specific to the attenuation and the isotope type. This is one of the prime reasons why Cs-137 sources with gamma energies of 0.662 MeV are often chosen in preference to Co-60 sources with the energies of 1.117 MeV and 1.332 MeV.

10.2.2 Statistics of Measurement

Statistical methods must be appreciated when considering nuclear measurement applications. Without considering background radiation, variations in measured count rates are mostly due to the random decay of isotopes. These variations result in some uncertainty of measurement that can generally be addressed using a Gaussian distribution model.

Plots from Johansen & Jackson (2004, p. 146) of relative error (standard deviation) as a function of μx (linear attenuation factor and absorber thickness) and the number of incident photons show that the relative error is at a minimum when the signal is

K ^a	Confidence Limit (%)	
0.5	38.3	
0.68	50.0	
0.8	57.6	
1.00	68.3	
1.20	77.0	
1.40	83.8	
1.65	90.0	
1.81	93.0	
1.96	95.0	
2.00	96.0	
2.33	98.0	
2.58	99.0	
3.00	99.7	
^a Number of standard deviations		

Table 10.1: Confidence to standard deviation relationship for a Gaussian distribution.

attenuated by 86%. This is often unrealistic in design. In addition these plots also reveal that increasing the number of incident counts will also achieve a reduction in error.

In support of the last statement, according to Knoll (2010, p. 84) 'the total number of recorded counts completely determines the fractional error to be associated with that measurement' and furthermore 'the time required to achieve a given fractional error will increase as the inverse square of the desired statistical precision'. Simply put, to reduce error and increase the probably that a measurement is correct, the number of counts taken to determine a measurement must be significantly increased. Deviation from the mean value can be determined as a fraction value by dividing the deviation of a count rate by the count rate. Confidence is the likelihood the measurement is correct, and is related to standard deviation of a Gaussian distribution as shown in Table 10.1. By designating an acceptable fractional error and setting a confidence limit, the number of counts to meet these criteria can be determined using the following equation:

$$n = \left(\frac{K}{F}\right)^2 \tag{10.1}$$

where n = number of counts

K = confidence value, expressed in standard deviation units

F =fractional error

As an example determining the number of counts required to achieve a fractional error within 10% with a confidence limit of 68.3% (1 standard deviation).

$$n = \left(\frac{1}{0.1}\right)^2$$
$$n = 100$$

Reducing the fraction error to 1% with the same confidence limit of 68.3%

$$n = \left(\frac{1}{0.01}\right)^2$$
$$n = 10000$$

Retaining the fraction error at 1% and increasing the confidence level to 96% (2 standard deviations).

$$n = \left(\frac{2}{0.1}\right)^2$$
$$n = 40000$$

As can be seen from the above, to reduce the error and increase the confidence in measurement values requires a much greater count rate, and therefore requires a much longer count time. Additionally the uncertainty of measurement for individual installations will require assessment of the measurement with the inclusion of background radiation.

In comparison for a given number of counts, accuracy is the same independent of how the counts are taken, i.e. accuracy over many shorter time spans or over one longer time span will be the same if the two total time durations are identical.

Also of note is the fact that regular computations on raw data (counts) will give results

that will also follow a Gaussian distribution. Mean values may be easily be traced through computational steps, however standard deviation (σ) values will follow the error propagation formula of:

$$\sigma_u^2 = \left(\frac{\partial u}{\partial x}\right)^2 \sigma_x^2 + \left(\frac{\partial u}{\partial y}\right)^2 \sigma_y^2 + \left(\frac{\partial u}{\partial z}\right)^2 \sigma_z^2 + \dots$$
(10.2)

where u = total counts

x, y, z = independent variables

Examples that follow of simple computations include subtraction and multiplication by a constant.

1. In the case of determining standard deviation of a net count which is calculated by subtracting the background counts from the measure count that is inclusive of background counts:

$$u = x - y$$
$$\frac{\partial u}{\partial x} = 1$$
$$\frac{\partial u}{\partial y} = -1$$

substituting into the error propagation formula;

$$\sigma_u^2 = (1)^2 \sigma_x^2 + (-1)^2 \sigma_y^2$$
$$\sigma_u = \sqrt{\sigma_x^2 + \sigma_y^2}$$

This will result in higher standard deviation than would be calculated by just taking the square root of the net count.

2. Multiplication by a constant:

$$u = Ax$$
$$\frac{\partial u}{\partial x} = A$$

substituting into the error propagation formula;

$$\sigma_u^2 = (A)^2 \, \sigma_x^2$$
$$\sigma_u = A \sigma_x$$

This will result in the calculated standard deviation being A times the original standard deviation.
10.2.3 Operational effects on error

Nuclear instruments in general have become heavily software based, more reliable and more accurate since inception. However one of the major sources of error when using these devices is due to the very nature of measurement. In most cases nuclear measurements are inferred. The input data for calibration relies heavily on the interpretation of sampling, which in itself can deviate significantly due to factors such as;

- Operator experience
- Process conditions
- Sampling arrangements

The use of manufactured standards for calibration purposes may be possible, but in most cases these are not easily instituted. Analytical devices may use standards that are typically produced by the equipment manufacturer. Due to the many factors that can affect calibration, manufacturers now rarely quote in respect to accuracy, instead preferring to quote in regard to repeatability.

In the majority of applications, homogeneity of the sample stream is a desirable attribute that will deliver better measurement results; this is chiefly due to the ability to use longer counting periods. Rapidly changing attributes of the sample medium will require faster response times that can only be achieved by reducing the number of counts taken; considering the measuring statistics, this will result in a reduction in accuracy. Notably with reference to Nyquist's sampling theorem Johansen & Jackson (2004, p. 164) the count time must be at least half the smallest time constant in the process.

10.2.4 Signal processing

Signal to noise ratios may be significantly increased by increasing source activities so as to generate higher counts; however this action is not undertaken without serious consideration as it goes against the major safety principles of ionizing radiation use. With the exception of NORM within the measurement stream, the noise that can be

10.2 Measurement fundamentals

attributed to background radiation may be substantially reduced by the placement of shielding around the detector. Signal to noise ratios are also very much dependent on detector type. Typically Geiger-Muller tubes and conventional photomultiplier tubes have high gain of the input signal and consequently have high signal to noise ratios, whereas semiconductor techniques generate much lower signals which require further amplification and are therefore much more susceptible to electronic noise.

A common method used to minimise signal degradation is the use of preamplifiers located close to detectors. As part of the design the preamplifier provides a low impedance connection between the detector and a downstream amplifier. Subsequently the connection distance between a preamplifier and a downstream amplifier can be significant due to the low impedance output provided by a preamplifier, and this distance may be further extended through the use of cables with larger cross sectional areas.

The manipulation and interpretation of input signals is dependent on both the desired measurand and the type of detector used.

With respect to the process, Geiger –Muller tubes are predominately used in point level (switch) applications with very little to no signal manipulation. Switch activation is defined by a certain count value, with the hysteresis between on and off values also typically set using some value of count difference. Erroneous activation is generally accounted for by this hysteresis value and by the inclusion of an on or off time delay. As previously mentioned GM survey meters are often used in a safety capacity to indicate exposure, to achieve this survey meters need to be calibrated to match the source of radioactivity. Best results are achieved using long count times with the values returned dependent on the functionality of the meter, these displayed values may include count rate and any of the exposure units; in most instances (due to source prevalence) the meter will be calibrated to the energy levels associated with Cs-137.

Process measurements using radioisotopes include density, continuous level and belt weighing applications, and as a rule all use scintillation methods for detection. The output takes the form of electrical pulses, with the number pulses per length of time proportional to the intensity of the radiation received. In the majority of applications a NaI photomultiplier tube is used with a charge-sensitive preamplifier. Charge-sensitive preamplifiers integrate the input charge and produce an output with amplitude propor-

10.2 Measurement fundamentals

tional to the accumulated charge. Further signal processing of the preamplifier output is generally completed at a downstream evaluation unit. The preamplifier signal is modified by a shaping amplifier that amplifies and filters the signal to suit further signal processing operations and to increase the signal to noise ratio. Filtering operations within the shaping amplifier are typically completed through the use of active filters, with output pulses having a close resemblance to that of a Gaussian curve. Further signal processing involves the use of a discriminator that suppresses noise pulses by limiting pulse counting to those pulses that exceed a certain amplitude. Fig. 10.2 shows the typical layout of a discriminator circuit used for pulse counting; the discriminator is represented using a comparator and the monostable one shot is used as noise suppression, the one shot prevents the registration of multiple pulses that may occur due to noise triggering the comparator. Timing of the one shot is characteristically set to match the width of the shaper output pulse. Once into memory, the count rate can further manipulated to determine the required process measurement.



Figure 10.2: Pulse counting circuit with discriminator. (Johansen & Jackson 2004, Fig. 5.9)

Analytical devices are spectroscopic instruments used to determine elemental concentrations in slurries and ores. The fundamental principle behind analytical methods is the measurement of characteristic energies which may be used to determine both the element type and concentration within a sample.

X-ray fluorescent (XRF) methods are used in both slurry and ore analysis. In this method the sample is excited with a suitable form of ionising radiation, which typically results in the ejection of an electron from the inner shell of some of the sample atoms. Electrons in the outer shells drop to fill the inner shell vacancy, and as a consequence characteristic energies are released in the form of x-rays as the atom de-excites to a stable state.

Neutron activation analysis is a method that is restricted to ore analysis and is typ-

10.3 Chapter Summary

ically installed on feed conveyors. The ore is bombarded with neutrons that has the effect of producing radioisotopes of the existing elements, the subsequent decay of these isotopes emit characteristic energies that can be detected to determine element type and concentration.

In analytical applications the process of pulse counting measurement is similar to that of the previously discussed methods using detectors, preamplifiers and shaping amplifiers; however now a Single Channel Analyser (one energy of interest) or a Multiple Channel Analyser (multiple energies of interest) replaces the discriminator and is used to determine the number of counts at a particular energy. The channel analyser sets upper and lower pulse amplitude limits to separate the counts of different energy; with the elemental determination of concentration within a sample a function of the radiation intensity at defined energies.

10.3 Chapter Summary

All devices used in the mineral processing environment use measurements of count to derive the measurand of interest, and in nearly all cases the source is a radioactive isotope where decay is of a statistical nature. Statistical approaches must be used with radiation measurements, with the uncertainty of measurement generally addressed using Gaussian distribution models. Reducing the error and increasing the confidence in measurement values can be achieved by the collection of a much greater number of counts, which will subsequently require much longer count times. Background radiation counts can be considered as noise and may have a significant effect when at high levels in comparison to the desired signal. In addition, the uncertainty of measurement for individual installations must be considered with the inclusion of background radiation. Although all detectors deliver pulse outputs, the manipulation and interpretation of these signals is dependent on both the desired measurand and the type of detector used. Furthermore much of the signal manipulation and interpretation is now completed in software resulting in far greater functionality.

Chapter 11

Measurement Applications

11.1 Chapter Overview

The use of nuclear instruments throughout large processing industries is common place. The applications covered in this chapter although aimed specifically at what may be found in mineral processing environments, may also be relevant to other industries such as; coal, cement and chemical production. The applications reviewed are by no means a complete coverage, but those reported on include by far the majority of installations. In the main these devices have arisen to meet a need, with often no viable measurement alternative; over time these devices have been proved reliable, with sufficient accuracy and have significantly enhanced production.

11.2 Nuclear Instruments in Mineral Processing

Nuclear measurement systems that are applicable to mineral processing applications are primarily used for process instrumentation, workplace safety and environmental monitoring. Workplace safety and environmental monitoring are of major significance when it comes to the processing of naturally occurring radioactive materials (NORM), such as uranium ores and mineral sands. Mineral processing plants that process NORM in Australia include the Ranger Uranium Mine of ERA (Jabiru, N.T.) and BHP Billiton's (Olympic Dam, S.A.). As previously mentioned the calibration for the majority of these instrument installations is an inferred process, with calibration achieved from values returned from sampled media; as a result most manufacturers now rarely quote accuracy, but quote in regard to repeatability.

11.2.1 Process Instruments

Process instruments can be divided into two categories, process measurement and analytical; both of these types can be further subdivided and defined.

Process measurement instruments cover physical measurement devices such as level, density, flow and pressure. Analytical instruments by definition are directed at some measurement in reference to process composition and cover the simpler fields such as pH and conductivity through to the more complex fields such as spectrometry.

Mineral processing process measurement instruments using nuclear principles include;

- Point and continuous level devices
- Density meters
- Belt weighers

Mineral processing analytical instruments using nuclear principles include;

- Concentration by direct radiation measurement, NORM applications
- X-ray fluorescence (XRF) spectrometry
- Prompt gamma neutron activation analysis (PGNAA)

Due to the extremely specialised nature of the aforementioned analytical instruments, these instruments will only be given a brief coverage in this document.

11.3 Process Measurement

Nuclear process measuring instruments are typically used for indication, control and metallurgical accounting. These devices have proven to be reliable and are generally installed in difficult measurement applications. A significant advantage of these devices over most non-nuclear instruments is the non-contact measurement methods employed.

11.3.1 Level

Point level measurements are generally employed for the tasks of process and equipment protection. Applications include blocked chute (high level) detection under crushers and low level detection on feeders. Detection of a high level in a crusher discharge chute will inhibit feed to the crusher; this will allow the blockage to clear naturally or will prevent a further worsening of the situation. Detection of a low level on feeder will result in feeder shutdown; this will protect the feeder from falling impacts by maintaining a sufficient protective layer of ore.

Point level measurement is regularly achieved through the use of a cobalt-60 or caesium-137 gamma radiation source and a Geiger-Muller tube or scintillation detector arrangement. Most often, Geiger-Muller tubes are the norm for detection in these applications due to the benefits of lower cost and greater ruggedness. However scintillation detection methods are becoming more popular, as these devices are much more sensitive than Geiger-Muller tubes and will therefore detect changes with the use of sources of substantially lower activity.

Nuclear point level devices are typically used in the aforementioned applications, as non-contact methods are employed and dusty measurement environments pose no issue to this detection method.

Radioactive sources for point level applications can be quite powerful as the undertaken measurements often involve equipment with hefty walls that are often lined with wear plates. Typical installations can be viewed in Fig. 11.1, with both high and low level applications are shown.



Figure 11.1: Application of point level switches. (Vega Grieshaber KG 2011)

Interpretation of Fig. 11.1 shows that when under normal operating conditions, the incident radiation reaching the detector is at a maximum for a high level switch and at minimum for a low level switch. Subsequently high level switches will indicate process abnormalities when count rates fall below predetermined values. The opposite is true for low level switches, with process abnormalities highlighted when count rates rise above predetermined values.

The differences in detector technology can be appreciated by a review of the dose required at the detector. Interpretation of the installation and operation guide for a Thermo-Scientific PNF point level system (Thermo Fisher Scientific Inc. 2011), using a Geiger-Muller tube for a typical mineral processing application with a response time of

11.3 Process Measurement

3 seconds requires a radiation field of approximately 0.4 mR/hr (3.5 μ Sv/hr); whereas according to Berthold Technologies (2012) a scintillation detector used as part of a point level system can require as little as 1 μ Sv/hr (0.11 mR/hr). As a consequence it is possible to significantly reduce the source size and still achieve the desired outcome using scintillation techniques.

The use of scintillation detectors for point level mineral processing applications with substantial vibrations have largely been overlooked in the past, this has been primarily due to fragility of the photomultiplier tube. It is now possible to acquire fibre optic scintillators that isolate the photomultiplier from these intense vibrations, thereby increasing detector sensitivity with a reduction in source activity.

The use of nuclear techniques for continuous level in mineral processing environments is more of a rarity. In the main, alternative technologies can handle most applications; with the more difficult typically accomplished using strain gauges/load cells. The use of nuclear techniques for continuous level is generally limited to custom solutions and hazardous environments. These installations take advantage of one or more of the following:

- non-contact measurement
- isolation from hazardous environments
- ability to work in dusty environments
- ability to maintain operation with some build-up on vessel walls

Installation examples include level measurement in refractory lined reactor vessels where elevated temperatures are encountered and autoclaves used for high pressure leaching, where both elevated temperatures and pressures occur. Vessels containing liquids operated at elevated temperature and pressure will have gas densities above the liquid that are dependent on operating pressure. This will result in an increase in gamma ray attenuation and where possible the level instrument should be calibrated under normal operating conditions to allow for this effect.

As with many point level applications the size of the source can be substantial due to the vessel walls that must be penetrated. The most commonly used modern detector technology uses scintillation techniques with either a conventional or solid state photomultiplier. The installation arrangements can be:

- point detector point source
- point detector rod source
- rod detector point source, or
- rod detector rod source

Each system has some advantage over the other and an overview of these systems can be seen in Fig 11.2. As with point level installations the amount of radiation received at the detector decreases with rising levels.



Figure 11.2: Continuous level measurement arrangements. (Berthold Technologies 2012)

A variation of point level and continuous level applications is the detection of an interface between products. Detection of the interface is achievable as the products each have different absorption coefficient values. As a consequence the attenuation of the radiation source will differ for each product and the differences in count rate can be used to determine the interface level.

11.3.2 Density

In the majority of mineral processing plants, on-pipe slurry density measurement instruments represent the greatest number of application specific radioactive devices installed. The system typically consists of a Cs-137 or Co-60 radiation source mounted on one side of a pipe and a scintillation detector mounted on the opposing side, diagrammatically this can be viewed in Fig. 11.3. The principle of measurement is that with all parameters fixed, the amount of radiation received at the detector is dependent on the specific gravity of the product within the pipe. The detector count rate will decrease with an increasing specific gravity.



Figure 11.3: On-pipe density arrangement. (Berthold Technologies 2010)

On-pipe density instruments are used primarily for the following reasons;

- density indication and control,
- mass flow measurement, and
- metallurgical accounting

Mass flow measurement is usually completed in conjunction with an electromagnetic flowmeter. Most density transmitters have the option to input a signal from the flowmeter and output mass flow; however on larger mineral processing installations the signals from both the density transmitter and flowmeter are typically returned to a control system where data manipulation is performed.

The measurement of density is based on the following equation;

$$I = I_0 e^{-u_m \rho x} \tag{11.1}$$

where I = count rate from the detector

 I_0 = un-attenuated count rate μ_m = mass attenuation coefficient; cm²/g ρ = liquid density; g/cm³ x = absorber thickness; cm

With relevance to the above equation one may expect changes in mass absorption coefficients with slurry composition would be of great concern, however Gillum (2008, p. 442) states very small changes in mass absorption coefficient are insignificant and that calibration will provide the solution for water carrier slurries.

Notably density gauges by the manufacturer Berthold Technologies allow multipoint calibration, that take into consideration the non-linearity of returned count rate due to detector characteristics through the use of polynomial equations.

Temperature compensation may be used with density instruments to allow for density fluctuations due to temperature change. However in all installed examples considered none had this functionality fitted. This is due to a number of considerations which include;

- the fact that slurry temperatures remain reasonably consistent,
- the degree of difficulty in calculating temperature coefficients,
- the requirement to place a temperature detector in the process path, and
- the capital expenditure

11.3 Process Measurement

Good measurement is achieved when density changes are slower than density meter response; therefore the more homogenous the process the better. Many density transmitters have algorithms installed so as to switch response rates (reduce count times) when rapid changes in process occur in an effort to reduce error.

One of the major sources of error when using these devices is due to the inferred nature of measurement. The input data for calibration relies heavily on the interpretation of sampling, which in itself can deviate significantly due to factors such as;

- Operator experience
- Process conditions
- Sampling arrangements

Additional sources of error include;

- scale build up in lines that may be reduced by material selection,
- background radiation from other sources in the area, and
- changes in count rate due to decay.

Allowances for changes in count rate due to decay appears readily available through software for all modern devices and simply requires the installer to input the isotope source type in software.

In all viewed instances on the Boddington Mine site and through the review of manufacture's literature, installations use NaI scintillators with conventional photomultiplier tubes to detect gamma radiation. One major limitation of these detector assemblies is that they are very susceptible to shock, and there have been instances onsite where detector failure has occurred due to shocks induced from pump cavitation. In fact a least one detector remains out of service due to line shocks and requires installation review to determine a suitable solution.

Overall these instruments appear well suited for tasks where the consistency of the process varies considerably from well ground slurry through to larger size aggregates. Time constants often exceed 20 seconds in these applications in an attempt to minimise statistical error, and therefore for an installation to be accurate requires the presentation of a typically homogenous process. Most installations are placed out of the vicinity of personnel and at times can be difficult to access when maintenance is required; however this is generally intentional so as to limit possible radiation exposures.

11.3.3 Belt-weighers

Nuclear belt weighers normally consist of a reasonably compact frame, generally with one or two sources mounted on the section above the conveyor belt and with a rod detector mounted on the section below the belt. A drawing of general assembly can be viewed in Fig. 11.4.



Figure 11.4: Nuclear belt weigher general arrangement. (Ohmart/VEGA Corporation 2012)

Operation of nuclear belt weighers is similar to that of density gauges, using the same fundamental equations to determine mass. The source is generally Cs-137 or Co-60; with an increase in mass returning a decreased count rate. Source activities tend to be greater than those used in density gauges due to the larger specific gravities encountered. The major advantages of these installations are the small footprint they occupy and the ease of installation.

As this measurement is one of mass flow, the conveyor speed must be incorporated into the measurement system and this is usually achieved through the use of a speed sensor mounted on the tail end of the conveyor. Integration of mass flow values provides totals in the unit of mass.

Ronan Engineering Measurements Division (2000) claim that 'Basic repeatability is \pm 0.1% of full scale where conveyer speed and conveyor loading geometry are repeatable and loading is maintained between 60% and 80% of full scale for a minimum of 30 minutes.' Notably this statement takes into no consideration the effects of belt speed and belt type. Johansen & Jackson (2004, p. 215), make what appear to more realistic statements, stating accuracy is typically \pm 3% and that rapid counting times of a few tenths of milliseconds are required to obtain the best accuracy. From the previous section on measurement, short count times not only increase error, but also reduce confidence in the measurement.

From interpretation of the above, most installations in mineral processing could never obtain high accuracy simply due to the load profiles that are often presented, let alone the consideration that many installations would include high belt speeds.

11.4 Analytical Instruments

Analytical measurements are used to determine sample concentrations. Compared with traditional techniques, these methods provide fast feedback than can be used to make process management decisions and in many applications automatically through a control system. As this is a much specialised area of instrumentation only a brief overview will be given.

11.4.1 Concentration by direct radiation measurement

In NORM mining applications it is often possible to determine the concentration of radioactive products by direct measurement of radioactive emissions. These methods are typically restricted to uranium mining applications, using gamma radiations to de-

11.4 Analytical Instruments

termine uranium concentration.

Examples include instruments determining out of mine ore grade before processing occurs. In reference to this example, an instrument package referred to as a Uranium ore discriminator consists of gamma detectors mounted over a truck stop; this installation can be viewed in operation at the Ranger Uranium Mine in Fig. 11.5. When a loaded haul truck is parked underneath the discriminator, gamma measurements are used to determine concentration. The haul truck can then be directed to dump at the appropriate stockpiles so as the ores can be blended to recipe that will optimise production.



Figure 11.5: Uranium discriminator at the Ranger Uranium Mine. (OBrien 2010)

11.4.2 X-ray fluorescence analysis

X-ray fluorescence is a form of spectrometry that can be used to determine elemental concentration. Applications include slurry and solids analysis. Analysis is fast in comparison to tradition technologies and the information obtained, often within minutes, can be used to implement automatic controls. Dependent on the method of analysis it is possible to achieve measurement with detection limits down to a few parts per million.

11.4 Analytical Instruments

Outotec Oyj (2008) state the typical measurable concentration range for slurries is 0.004% - 100% by weight. Although it is possible to measure elements with an atomic number as low as 4 in solids, it is according to Outotec Oyj (2008) impractical to measure elements in slurry with an atomic number less than 20.

The principle of x-ray fluorescence has its foundation based on the characteristic energies that are emitted as a consequence when electrons are dislodged from the inner shells of an atom when subject to some form of ionising radiation. Characteristic energies are released as electrons in the outer shells of the ionised atom fall to fill the vacancies of the dislodged electrons; these characteristic energies in the form of x-rays are released so as the atom can gain stability. The principle of x-ray fluorescence can be viewed in Fig. 11.6; it should be noted that the incident radiation in the form of low energy x-rays can also be achieved through the use of alpha emitting radioisotopes. Radioisotope selection for the analyser is dependent on the element of interest.



Figure 11.6: Principle of XRF analysis. (Tawada Scientific 2010)

11.4 Analytical Instruments

In principle an electron can be dislodged from any of an atom's electron shells, however in most instances electrons are ejected from the inner K and L shells. The energy values of the x-ray photons emitted is dependent on the energy difference between the shell from where the electron to fill the vacancy is sourced and the shell from where the vacancy occurred. Subsequently the emitted x-ray photons have a range of energy values that are elementally independent (characteristic). As with the nucleus, electrons have binding energy and the incident energy needs to exceed a value that is dependent on shell type if electrons are to be ejected from a specific shell. Acknowledging these energy requirements, it is therefore possible to have some selectivity in what electrons are ejected when subject to ionising radiations.

XRF analysis can be undertaken using two different methods that are often completed in unison; these methods are wavelength dispersive (WDXRF) and energy dispersive (EDXRF). WDXRF methods use one or more 'Bragg' crystals to filter the input energies to very narrow elemental bands of interest, thus eliminating other elemental energies that are close to the energies of interest. Detection in most cases is completed using scintillation techniques. This method gives better resolution and accuracy and is often used in analytical laboratories. EDXRF has lower detection limits, but a complete energy spectrum is obtainable, which through the use of multi-channel analysers can be used to determine the concentrations of many elements within the sample. Detection in modern applications is completed using a lithium drifted silicon semiconductor that requires cooling to minimise noise in the detector. A diagram showing both methods in the application of slurry analysis can be viewed in Fig. 11.7. With respect to detection limits, Outotec's Courier SL series documentation (2008) states 'The WDXRF-measurement channel's detection limit is 3-30 ppm for most elements in slurry samples. Using the EDXRF-measurement channel, the minimum detection limit is typically 100 – 500 ppm.'



Figure 11.7: WDXRF and EDXRF methods for slurry analysis. (Outotec Oyj 2008)

In text, characteristic x-rays are labelled with the shell from which they originate (K, L, M, and N, etc) and with a Greek subscript that signifies from which shell the vacancy filling electron comes from. As electron shells may also contain electron orbits of different energies, a further subscript is attached as required so as to differentiate energy levels within shells. In particular order Greek subscripts indicate the number of rings that the vacancy filling electron migrates; as an example an electron migrating one shell, from the L shell to fill a vacancy in the K shell would be labelled K_{α} and migrating from the M shell to the K shell labelled K_{β} . Fig. 11.8 shows an example of the spectral plot from an EDXRF analyser.

XRF analysis is commonly found on large mineral processing sites, as the management of process can be actioned quickly; whereas in comparison, laboratory analysis can take extended periods of time and provides little opportunity to implement automatic controls.



Energy-dispersive channel EDXRF, 0.12% Zn.

Figure 11.8: EDXRF spectral plot, count-rate (x 1000) against energy for zinc slurry. (Outotec Oyj 2008)

11.4.3 Neutron Activation Analysis

Neutron activation analysis is typically used to analyse mineral concentrations where bulk in-feed analysis will prove highly beneficial. This allows for process management, and in cases such as iron ore may be used to determine the requirement for further treatment and/or blending schedules based on ore quality. Case studies presented at the Canadian Mining Magazine Congress by Kurth (2009) of Scantech show an accuracy within 0.5% of laboratory analysis is possible for on-belt ore processing applications with a metal content between 48 - 50%. Additionally unlike XRF techniques, neutron activation analysis is possible on slurries with low atomic number to an accuracy of approximately 0.5% (Thermo Fisher Scientific Inc. 2007). The principle of operation is that when a sample is irradiated with a neutron source, the induced radioactivity in the sample will have predictable decay. Decay can be prompt or delayed, with count rates plotted against decay energies via the use a gamma detector in conjunction with a multi-channel analyser. From these values the chemical composition of the sample can be established. Fig. 11.9 depicts the plot of a sample subject to neutron activation analysis.



Figure 11.9: Neutron Activation Analysis spectral plot, count-rate against energy. (University of Missouri Research Reactor 2008)

With respect to mineral processing operations, neutron activation analysis is always prompt as long time delays can not be tolerated. Hence these instruments fall under the category of prompt gamma neutron activation analysis (PGNNA). In all viewed manufacture's literature the neutron source is Cf-252 with a half life of 2.65 years that decays through spontaneous fission, emitting both alpha particles and neutrons. A picture showing a PGNNA installed for on-belt analysis can be viewed in Fig. 11.10.



Figure 11.10: PGNAA installed on a conveyor. (Thermo Fisher Scientific Inc. 2008)

11.5 Chapter Summary

As can be ascertained from the myriad of applications, nuclear instruments provide reliable measures for often difficult applications. Most applications have arisen out of necessity and generally few genuine viable alternatives exist. The ability of analytical instruments to accurately determine concentration has allowed the implementation of process control, which has resulted in better quality control and with substantial economic benefit.

Chapter 12

Evaluation, performance and alternatives

12.1 Chapter Overview

This chapter is dedicated to the evaluation and performance of nuclear instruments currently installed in mineral processing applications. No attempt has been made to review analytical instruments, as these are specialised devices that are often application specific and henceforth performance will often vary with location and installation. In addition the merits of the two most popular gamma producing sources, CS-137 and Co-60 are reviewed in respect to each other.

12.2 Site review

The Boddington Gold Mine is a copper and gold mineral processing site that uses nuclear measurement systems in the applications of point level, on-pipe density, beltweighing and XRF. The numbers of each device type can be viewed in Table 12.1. As in most plants, on-pipe density covers the largest proportion of nuclear installations.

Instrument	Number of	Source type	Manufacturer
Application	devices		
Level switch	7	Cs-137	Berthold
On-pipe density	33	Cs-137	Berthold
Belt-weigher	4	Cs-137	Ronan
XRF	2	x-ray tube	Outotec Oyj
	1	AM-241	Thermo Scientific

Table 12.1: Nuclear installations at the Boddington Gold Mine.

From investigations the use of nuclear point level has been restricted to bin, chute and feeder levels and have proven to be reliable. Alternative devices using time of flight and other beam-break technologies at this point in time appear not viable. Installations where these alternatives have been installed or trialled are often marginal with greater unreliability and often require human intervention (maintenance) to place these devices back into service.

Review of site instrumentation indicates that the only form of density measurement onsite is on-pipe density and is completed using nucleonic devices. It is believed that viable alternatives may exist for many of these installations; however in some instances such as mill cyclone feed with a large aggregate content, the non-contact benefit of nucleonic instruments may still provide the optimum measurement solution.

As stated in Chapter 11; in all density installations NaI scintillators are used with conventional photomultiplier tubes to detect gamma radiation. One major limitation of these detector assemblies is that they are very susceptible to shock, and there have been instances onsite where detector failure has occurred due to shocks induced from pump cavitation. In fact a least one detector remains out of service due to line shocks and requires installation review to determine a suitable solution.

In some density installations radiation sources have been placed too close together and background counts significantly affect measurement accuracy. This was observed by measurement change when adjacent gauge sources were isolated. This fault can be attributed to poor plant design.

12.3 Evaluation of source type

Nuclear belt weighers have been installed on a conveyor under ore sizing screens, with the oversize placed on the belt and returned to crushing circuits. The feed onto the conveyor does not have a consistent profile or geometry and these facts on their own, even without consideration for high belt speeds appears to strongly suggest that these installations are a poor fit for the measurement application.

Evaluation of on-pipe density and belt-weighers shall be considered further in this chapter with respect to alternatives.

From an access to equipment perspective it has been found that on occasion maintenance activities in the vicinity have been restricted due to the presence of a radioactive source, and hence can influence equipment availability, maintenance planning and workshop utilisation. This is chiefly due to the requirement that radiation sources must be isolated by the RSO or an appointed deputy; as with any 24/7 operation at times it is difficult to have the required personnel available.

A generalised survey of employee knowledge in respect to radiation safety within maintenance and operational departments at the workshop level was found to be minimal; with many personnel being highly concerned for their safety when in the vicinity of these devices. These unwarranted concerns could be simply addressed with a little additional workplace training. Notably operations in which naturally occurring radioactive materials are found the employer has an obligation to provide adequate training (Australian Radiation Protection and Nuclear Safety Agency 2005).

12.3 Evaluation of source type

Caesium-137 and cobalt-60 isotopes are the most common ionising radiation sources found in both mineral processing and industrial environments. The following attempts to evaluate the pros and cons of each, so as it is possible to make informed decisions on what may be the better isotope choice for a gamma measurement application. Firstly considering gamma energy levels, Cs-137 has gamma energies of 0.662 MeV, whereas Co-60 has gamma energies of 1.173 and 1.332 MeV. With reference to Johansen & Jackson (2004, p. 249) to obtain the maximum sensitivity requires the selection of the source with lowest possible energy, in all cases this would be Cs-137. An additional advantage of lower source energies is that the quantity of shielding material required can be substantially less, somewhat reducing the manufacturing costs associated with shielding and reducing source holder weight.

However these are not the only considerations that must be acknowledged in source selection. The benefits associated with Co-60 having higher source energies include a greater depth of penetration and therefore reduced activity requirements. To demonstrate the effect of emission energy on source activity; if 7.5 μ Sv/h is required at a detector placed 1 m from the source with air as the only attenuating medium and using the equation:

$$\alpha = \frac{Xd^2}{\Gamma_{\delta}} \tag{12.1}$$

where $X = 7.5 \,\mu\text{Sv/h}$; dose rate required at the detector $\Gamma_{\delta} = (\text{mSv.m}^2)/(\text{h.MBq})$; specific gamma dose constant d = 1.0 m; free air distance

Source type Cs-137 with $\Gamma_{\delta} = 1.07 \times 10^{-4} \text{ (mSv.m}^2)/(\text{h.MBq})$: $\alpha = 7.5 \frac{\mu \text{Sv}}{\text{h}} \times 1.0^2 \text{m}^2 \times \frac{1}{1.07 \times 10^{-4}} \frac{\text{h.MBq}}{\text{mSv.m}^2} \times \frac{\text{mSv}}{1000 \ \mu \text{Sv}}$ $\alpha = 70.1 \text{ Mbq}$

Source type Co-60 with $\Gamma_{\delta} = 3.70 \times 10^{-4} \text{ (mSv.m}^2)/(\text{h.MBq})$:

$$\begin{array}{lll} \alpha & = & 7.5 \frac{\mu \text{Sv}}{\text{h}} \times 1.0^2 \text{m}^2 \times \frac{1}{3.70 \times 10^{-4}} \frac{\text{h.MBq}}{\text{mSv.m}^2} \times \frac{\text{mSv}}{1000 \ \mu \text{Sv}} \\ \alpha & = & 20.3 \text{ Mbq} \end{array}$$

As can be appreciated in this case the activity of the Co-60 source is less than a third of the activity of the Cs-137 source. In addition the half-life of Co-60 is 5.27 years in comparison to Cs-137's half-life of 30.17 years; from the authors view Co-60 appears a much safer option although operation life may be substantially reduced. Instrument design life does not necessarily match half-life and source sizes may be increased to extend design life in the case of Co-60, or reduced in the case of Cs-137 thereby limiting both activity and design life. Ultimately source size will be limited by legislation. Furthermore with respect to on-pipe density gauges the standard deviation in density due to statistical errors according to Johansen & Jackson (2004, p. 146) can be determined using:

$$\sigma_{\rho} = \frac{1}{\mu_m x \sqrt{I\tau_1}} \tag{12.2}$$

where σ_{ρ} = standard deviation in density I = count rate from the detector μ_m = mass attenuation constant; cm²/g τ_1 = time constant; s x = internal diameter of the pipe; cm

This equation simply determines the error in density \pm (g/cm³) within \pm one standard deviation. From the author's interpretation, if a Cs-137 and a Co-60 were initially sized to deliver the same number of counts at the detector, by simply manipulating the number of counts or the time constant, it is possible for Co-60 to achieve similar accuracies to Cs-137 if the product if fully homogenous. As an example, implementing Equation 4.2 to achieve similar accuracies with water as the attenuating medium;

$$\frac{1}{\mu_m \sum_{S_{137}} x \sqrt{I_1 \tau_1}} = \frac{1}{\mu_m \sum_{Co60} x \sqrt{I_2 \tau_2}}$$

The mass attenuation coefficients for both isotopes with repect to water are: Cs-137 = 0.089 cm²/g Co-60 = 0.063 cm²/g

$$\frac{1}{0.089x\sqrt{I_1\tau_1}} = \frac{1}{0.063x\sqrt{I_2\tau_2}}$$
$$\sqrt{I_2\tau_2} = \frac{0.089x\sqrt{I_1\tau_1}}{0.063x}$$
$$I_2\tau_2 = 1.997(I_1\tau_1)$$

From the previous calculation, to achieve the same accuracy as with Cs-137, the activity of a Co-60 source could be doubled, the time constant doubled or a combination of both could be increased to achieve the factor of 2. Doubling the activity would be the least desirable outcome as it moves away from "ALARA" principles.

The major benefits of use with Co-60 in comparison to Cs-137 are the reduced activity required to complete the task and the substantially shorter half-life of 5.27 years. The choice of Co-60 sources over Cs-137 would significantly reduce the risk to the public in the event of source loss and reduce the corporate risk associated with ownership.

Another factor that favours Co-60 source selection is the recognition that ever increasing detector sensitivities have been achieved over time and therefore instead of source replacement at the end of effective product life, it may be possible to simply exchange the detector. Furthermore Co-60 appears to have better synergies with source reconditioning times and electronic product life.

12.4 Cost of ownership

Ownership costs related with the use of nuclear devices varies and is site dependent. This is primarily due to the number and type of devices installed. The greater the number of sources the more the apportioning of cost can be spread across each. Of great surprize to the author was the minimal cost in annual registration/licencing of these devices. Annually in Western Australia a fee is payable, the registration encompasses all devices on site, with the cost based on total installed activity. The maximum cost is currently set at \$830 for 1 year (Radiological Council 2012a). A copy of this fee schedule can be found in Appendix D.

Additionally there is a requirement for each site to have at least one Radiation Safety Officer (RSO) that must be qualified and recognised by the relevant authority. A small fee is generally also payable annually for the licencing of the RSO; this cost is currently set in Western Australia at \$55 for 1 year (Radiological Council 2012c). Training costs for qualification as an RSO are currently estimated at \$3000 without the additional incidentals of accommodation, meals, etc. Employment costs associated with the appointed RSO again vary on site and with the number of devices installed. This position may be full time and require the professional qualifications of an engineer

as may be the case with naturally occurring radioactive materials, through to minimal intervention with a qualification achieved through training; that typically takes less than one week to complete and that is recognised by the relevant authority. From this the author estimates the employment costs of an RSO could range from approximately \$10,000 to \$250,000 per year.

Furthermore a secure radiation store is required on-site for the safe storage of outof-service sources; initial costs may be low as typically a sea container is sufficient. However some cost allowances should be included for security monitoring.

The individual costs per year based purely on individual installation are estimated in consideration to the following:

- Compliance costs
- Source refurbishment and disposal costs

Compliance costs include the cost of annual source wipe tests and assay, equipment inspection and signage. Wipe test costs are estimated at \$40 per source from the Government of Western Australia, Department of Health (2012).

Source effective service life times have been estimated at 15 years for Cs-137 and 7 years for Co-60, with refurbishment required for Cs-137 sources at least once in their lifetime.

Correspondence with Beckley (2012) of Krohne; the agent for Berthold Technologies within Australia has established disposal costs for both Cs-137 and Co-60 sources for a 300 mm on-pipe density gauge at approximately \$6000 each. The author expected significant differences in disposal costs between the two source types, but this may be explained by the fact that Berthold Technologies manufactures and recycles radioactive sources. On this information, source disposal cost for all installations has been set at \$6000.

Refurbishment of source containers requires, overhaul, testing and hazardous transport and has been given a subjective cost of \$5000.

The annual costs of ownership specifically for each source type have been determined using the aforementioned costs averaged over the effective service life. The estimated costs association with Cs-137 and Co-60 source types can be viewed in Table 12.2. It should be noted that the acquisition costs and unforeseen maintenance costs have not been included in these calculations. It could be envisaged that Cs-137 installation with an estimated system life of 15 years would require replacement of the detector and possibly electronic assemblies at some stage. Towards the end of design life these parts may become hard to source and at greater expensive as they may be deemed as legacy parts.

Cost description	Cs-137	Co-60
Wipe test	\$40	\$40
Inspection, 4 hours @ \$120/h	\$480	\$480
Signage replacement every 4 years; cost of	\$220	\$220
signage and 4 hours @ $120/h$		
Cost of refurbishment \$5000/15 years	\$333	n/a
Cost of disposal \$6000/effective life	\$400	\$857
Total	\$1473	\$1597

Table 12.2: Annual individual nuclear installation costs.

Acquisition costs based on quoted prices of on-pipe density systems from the Krohne Group (2012) would suggest the prices of measurement systems using either Cs-137 or Co-60 isotopes are extremely similar, and from the authors view should be treated as equal. Of major benefit when considering nuclear instruments is that the total measurement system cost does not change significantly with a change in the distance of the radiation path.

12.5 On-pipe density

With respect to on-pipe nuclear density gauges the sources of choice for slurry density measurement are Cs-137 or Co-60. Notably at the Boddington Gold Mine all density installations use Cs-137 isotope sources. As previously stated, of the two source types Cs-137 produces the better measurement result.

Alternatives to the use of these nuclear devices do exist, but all require process contact and will suffer from the effects of abrasion. The following instruments were found as alternatives:

- Coriolis flow meter principle of operation involves the measurement of the deflection of an oscillating tube driven by an exciter. Liquid flows through the tube with the frequency of oscillation representative to the liquid density; with the frequency of oscillation inversely proportional to density. Furthermore this device can also return flow rate thereby eliminating the need for an additional flowmeter that is typically installed.
- Vibrating fork methods measurement is based on the premise that the resonant frequency of the vibrating fork times change with changes in liquid density.
- SCIAM DM3 density meter in principle this system weighs a set volume within a section of pipe using a load cell.

One positive aspect of the three above alternatives is that calibration is traceable. These three alternatives have been reviewed with the aid of correspondence from the associated manufacturers. Comparisons have been draw between these alternatives and nuclear density devices using both Cs-137 and Co-60 isotopes. These comparisons can be viewed in Table 12.3. The data in the table for the Coriolis flowmeter is based on that of the largest possible internal diameter of 80 mm; all other data is based on the nuclear installation of 324 mm steel pipe with 9.5 mm walls and with a maximum slurry SG of 2.0.

	Berthold LB444	Berthold LB444	Emerson	Krohne	SCIAM
	Cs-137	Co-60	7828	Optimass 7000	DM3
Technology	Nuclear Cs-137	Nuclear Co-60	Vibrating fork	Coriolis	Load Cell
	1850 Mbq	740 Mbq			
Accuracy	not quoted	not quoted	$\pm 1.0 \ \mathrm{kg/m^3}$	$\pm 0.5 \ \mathrm{kg/m^3}$	$\pm 0.5\%$
					of full scale
Repeatability	$\pm 1.66 \text{ g/l}$	± 1.66 g/l	$\pm 0.1 \ \mathrm{kg/m^3}$	$<\pm 0.1\%$	$\pm 0.1\%$
				of actual flow	of full scale
Possible Range	0 to > 3000	0 to > 3000	0 to 2000	400 to 2500	1000 to 3000
	$\rm kg/m^3$	$\rm kg/m^3$	$\rm kg/m^3$	$\rm kg/m^3$	$ m kg/m^3$
Cost	pprox \$20,000	pprox \$20,000	pprox \$10,000	pprox \$22,000	$\approx \$30,000$
Life expectancy	15 years	7 years	>2 years	>2 years	>5 years
Time constant	60 s	87 s	not quoted	not quoted	45 ms
Orientation	any	any	any	any	horizontal
Ease of install	2	2	1	3	51
Note – Ease of inst	all rated 1 to 5 on au	thor's judgement, wi	th 1 rated the easiest		

Table 12.3: Density gauge comparisions.

12.5 On-pipe density

154

Data sourced from: Krohne Group (2012), Emerson Process Management (2012) and SCIAM Worldwide (2012)

12.5 On-pipe density

From all appearances the alternatives would return better measurement values and would give a better reflection of what is actually occurring in the process.

As previously stated the major benefit of nuclear installations is non-contact measurement; review of the costs of ownership and acquisition also suggests that over the period of ownership nuclear devices have the lowest overall cost.

From the author's observations a significant increase in accuracy could be assured if a Coriolis flowmeter was used and with comparable overall costs if the meter life was greater than five years. Placement of a Coriolis flowmeter on smaller lines (< 80 mm) would also eliminate the additional cost of the flowmeter that is usually installed with a dedicated density gauge to measure mass flow. Notably the maximum measurable SG for this Coriolis flowmeter is 2.5.

Deciphering product information the author believes the Micro motion 7828 and SCIAM DM3 appear comparable in cost over the longer period, but with a greater overall cost outlay than both nuclear and coriolis methods. The Micro motion 7828 is by far easier to install, whereas the DM3 appears much better suited to the larger particle sizes (coarse aggregate) in the product. Both should be considered application dependent.

Before any judgement can be passed the author believes a trial in a large tails line of the Micro motion 7828 would be of benefit in determining actual gauge life expectancy. Although on a cost basis the author would not choose to install the DM3; in processes where it is difficult to achieve good sampling for calibration and for where the addition process information may be of benefit, the DM3 appears as a sound solution. Considering this, the DM3 may be an excellent choice for process density measurement on mill discharge/cyclone feed applications where large aggregate sizes are possible.

The life expectancy of the alternatives is directly related to the effects of abrasion and hence process velocity. Life expectancy of devices installed in the process is typically process dependent, therefore product life can be subjective and can often only be determined by trial. Information from the Krohne Group (2012) on the Coriolis flowmeter recommends that the flow velocity should be below 4 m/s to assure long meter life. Additionally the Coriolis flowmeter could be used for density measurement in larger installations if installed as a side stream or in a recycle line.

12.6 Nuclear belt-weighers

The use of nuclear belt-weighers in the eyes of the author is contentious. Systems are already in use that are non-nuclear and have better accuracies. The only benefits that can be highlighted for these instruments are the small footprint that these devices occupy and apparent ease of installation. The statement "ease of installation" should be considered in-line with addition regulatory requirements that must be completed to place the equipment into operation and forever after until instrument disposal. Furthermore a typical non-nuclear weigh frame installation can usually be completed within one day.

Issues also exist with calibration. Calibration is inferred from mass rate against count rate. At high tonnages it is typically only possible to calibrate against another beltweigher; which is generally of the non-nuclear type.

From Chapter 11, Ronan Engineering Measurements Division (2000) claims that for a nuclear belt-weigher: 'Basic repeatability is $\pm 0.1\%$ of full scale where conveyer speed and conveyor loading geometry are repeatable and loading is maintained between 60% and 80% of full scale for a minimum of 30 minutes.' Notably this statement takes into no consideration the effects of belt speed and belt type. Johansen & Jackson (2004, p. 215)) make what appear to more realistic statements, stating accuracy is typically \pm 3% and that rapid counting times of a few tenths of milliseconds are required to obtain the best accuracy. From the previous section on measurement (Chapter 10), short count times not only increase error, but also reduce confidence in the measurement.

A review on the installation of a nuclear belt-weigher located on Conveyor No.9 (CV9) at the Boddington Gold Mine follows:

CV9 takes feed from the oversize of ore sizing screens that feed ball grinding mills. The mass flow on CV9 is measured by a nuclear belt-weigher (WI631018DRAW), with conveyor discharge placed onto either Conveyor No.10 (CV10) or Conveyor No.11 (CV11) to return the oversize ore to the crushing circuit for size reduction. Non-nuclear belt-weighers using load cells are installed on both CV10 and CV11, both are known to be accurate and are typically within 1% of actual when calibrated with roller chains. Fig. 12.1 shows a comparison between the three belt-weighers with increasing load. As can

be seen the weight on CV9 is always high and the weights on CV10 and CV11 match each other on discharge switch-over. A small time delay can be viewed between reading from CV9 and the other conveyors, in addition it can clearly be seen that the error attributed to the belt-weigher on CV9 is non-linear with load. The error on the readings of CV9 ranges in this figure from approximately 24% at low tonnes to approximately 2% at high tonnes; with inconsistencies in error at similar tonnages most likely due changes to the load profile or ore size.

From these investigations nuclear belt-weighers do not appear as a good measurement solution. Nuclear belt weighers may prove viable at lower tonnages and with consistent profiles, however this is rarely the norm in mineral processing environments.

On a positive note, the investigation of nuclear belt-weighers in material handling has revealed that similar techniques could be used to determine mass flow with screw feeders. The author's inclination is that such a system should work well with the attributes of lower speeds, consistent feed and fixed profile.

From this review, the author does not condone the use of nuclear belt-weighers, primarily due to the fact that non-nuclear systems of better accuracy exist.




12.7 Chapter Summary

From the research into currently available nuclear instruments, it has been found that the statistical error of Co-60 can be adjusted to match that of Cs-137 sources in truly homogenous applications by increasing source strength or by increasing measurement time. Therefore it may be possible to replace Cs-137 sources with Co-60 and have little effect on measurement.

With respect to the costs associated with nuclear instruments, these were found to be much less than expected. In the cost comparisons between on-pipe nuclear density gauges and the possible alternatives, the nuclear solution appears more cost effect over the long term.

It is believed that all alternative instruments investigated with respect on-pipe nuclear density gauges will return better measurement results. Furthermore it appears an alternative is available for each on-pipe application, but in all cases at additional long term cost to the owner.

Finally the author believes there is no requirement for the use of nuclear belt-weighers and design should allow for installation of non-nuclear belt-weighers where measurement is required.

Chapter 13

Recommendations

13.1 Introduction

Most importantly this research has been found that the use of ionising radiation as part of a measurement system in the workplace is a safe practice. The use of specialised instruments based on nuclear principles has significantly enhanced production, whilst reducing operational costs. In addition many of these instruments are often the only viable measurement solution. Subsequently the author does not subscribe to the ideal of a nuclear free workplace.

However the author does believe improvements can be made. The best outcomes will be delivered by prudent judgement that will often need to take into perspective the desired result. Choices will typically pivot on; the desired measurement accuracy, reliability, overall cost and corporate ideology.

The choice to use nuclear instruments may be fundamentally driven by site specific parameters, such as plant size and workforce demographics.

On smaller sites, where no analytical devices use ionising radiation as part of a measurement system, the cost of using instruments based on nuclear principles may be cost prohibitive due to the additional costs of regulatory compliance. In addition the site workforce would typically be of the smaller nature with defined skill sets that may inhibit the choice of Radiation Safety Officer (RSO). Further issues of the transitional nature of the workforce may also make it difficult to fulfil regulatory compliance. In contrast the extra regulatory costs incurred per additional instrument installation on a site already faced with regulation are minimal. Large sites typically have large workforces with multiple personnel capable of filling the role of RSO. Furthermore the transitional nature of the workforce has little effect, as generally several personnel have been adequately trained to fulfil the role of RSO.

13.2 Source selection

Often nuclear instruments are installed for control and indication where an allowable margin of error is acceptable. Considering this, the author recommends the consideration of Co-60 sources instead of Cs-137 for measurement applications. As recognised in Chapter 12, it is possible for Co-60 to achieve similar accuracies to Cs-137 if the product is fully homogenous by manipulating either source size and/or time constant values.

The recognised benefits of Co-60 over Cs-137 taken from Chapter 12 include:

- The reduced activity required to complete the task and the substantially shorter half-life of 5.27 years. This would significantly reduce the risk to the public in the event of source loss and reduce the corporate risk associated with ownership.
- Recognition that ever increasing detector sensitivities have been achieved over time and therefore instead of source replacement at the end of effective product life, it may be possible to simply exchange the detector.
- Technological change over product lifetime may provide better measurement alternatives.
- Co-60 appears to have better synergies with source reconditioning times and electronic product life.

Where Cs-137 is deemed the isotope of choice due to measurement concerns, the author recommends design life is set at 10 years. This would not reduce the associated risks

as significantly as by choosing Co-60, but it would considerably lower these risks and realise the other benefits that may be associated with the choice of Co-60.

13.3 On-pipe density

With the largest proportion of nuclear instruments on a mineral processing site typically employed to determine pipeline density it has been found that alternatives do provide better measurement.

The reliability of modern instruments is typically dependent on electronics and it can be extremely difficult to predict which instrument is more reliable than another. The author does not believe product life or the benefit of non-contact measurement should be included as a measurement of reliability. The attractive aspect of non-contact measurement with a nuclear instrument allows for repairs to the measurement system whilst the process continues to run. However most systems employed in mineral processing have some form of redundancy, as in they can continue to run without density measurement or a standby system allows for process shutdown so as repairs can be completed. On top of the reported statistical error associated with the nuclear instrument, sampling with respect to calibration can significantly contribute to measurement error. The use of nuclear on-pipe density gauges requires that good sampling arrangements and techniques are implemented. Notably sampling should represent the count period of the device.

Where accuracy is the major concern with respect to process optimisation and metallurgical accounting, the author recommends that an alternative be considered with respect to overall cost. If the cost benefit is recognisable, the author recommends the following installations:

- Slurry with course aggregate, typically mill/cyclone feed SCIAM Worldwide DM3
- Concentrate lines < 80 mm Krohne Optimass 7000 coriolis flowmeter
- All other applications Emerson Process Management, Micro motion 7828 or Krohne Optimass 7000 coriolis flowmeter with a side stream or recycle line.

It should be noted that other manufacturers may supply equivalent instruments to the Optimass 7000 and the Micro motion 7828. The DM3 is a relatively new device and appears still to achieve wide recognition.

Ideally all the alternatives listed will be installed at low line velocities to promote long service life.

13.4 Nuclear belt-weighers

The author does not recommend the use of nuclear belt-weighers in any mineral processing application. There are systems already in use that are non-nuclear and have better accuracies. This statement is not made lightly and has considered both measurement principles and calibration difficulties. Furthermore these devices are an impediment to conveyor maintenance and raise unnecessary safety concerns.

13.5 Installation Specific

Research as well as the review of the Boddington Gold Mine has found these installation specific issues:

- Detector selection
- Measurement affected by background counts
- Detector and source mechanism failure due to shock
- Requirement for protection from the environment

The author recommends that all new installations of process measurement instruments use scintillation methods for detection. Although the use of Geiger Muller tubes may still be deemed as a safe practice, scintillation methods allow the use of significantly smaller radiation sources that are a better fit with "ALARA" principles.

13.5 Installation Specific

Review has found that even though the beams of radioactive sources are collimated and these sources are well shielded, sources located in the vicinity of each other can generate significant amounts of background radiation that may severely affect measurement. This is of particular concern with higher activity sources. The author notes in design that the distances between sources must be vigorously considered so as background radiation generated from other sources has a minimal affect.

Where detectors and sources are subject to severe shock, as in pump cavitation and the like, it is foreseeable the detector typically a scintillator/photomultiplier combination will fail after a very short period of time. In all likelihood damage may also occur to the protection mechanisms (shutters) of the source as well, thereby preventing source isolation. Solutions to this problem include mechanical isolation from the shock, installation of alternative measurement solutions and the standing-off (air gap allowing non-interacting mechanical movement) of the detector and source from the origin of the shock.

The author has found in many instances that source holders and detectors are poorly protected from the surrounding environment. Most source holders tend to have some form of protection placed over them to minimise process attack. Regularly this protection is lacking and within years, the integrity of the source holder is questionable; source holder refurbishment is then required at considerable cost. Detectors may also suffer from the process environment and often have no added protection; significantly it is not uncommon for moisture ingress to arise as an issue. Detectors are typically rated at 50 °C and on hot day, in full sun this rating can be exceeded and therefore some form a sun protection should be installed. The effects of the process environment for both source holders and detectors may be minimised by the installation of additional protection; typically mechanical or by a better choice of installation location. The author strongly suggests a new system be reviewed within a short period of installation so as to determine if environmental protection is an issue, and if so this should be addressed as soon as possible.

13.6 Maintenance

A reflection of currently installed nuclear devices has resulted in these maintenance recommendations:

- Nuclear instruments should be installed so as to readily allow for maintenance of equipment in the area. Simply put, nuclear sources should not be placed near pumps, valves and flanges so as their presence has the least amount of effect on maintenance activities.
- Research has found that Geiger Muller tubes only have a finite life; therefore the implementation of preventive maintenance where tubes are exchanged before failure may eliminate unforeseen equipment downtime.
- Source life may be extended by upgrade to a more sensitive detector.

13.7 Employee education

The author's final recommendation is in regard to employee education. It was found that employee knowledge in respect to radiation safety within maintenance and operational departments at the workshop level was minimal, with many personnel being highly concerned for their safety when in the vicinity of these devices. Furthermore at other sites where these employees had held previous positions and nuclear devices were also used, the consensus was that the prior training was also lacking. Subsequently the author strongly recommends that addition training should be considered at all mineral processing sites where ionising radiation sources are utilised.

Chapter 14

Conclusions and Further Work

14.1 Achievement of Project Objectives

Project commencement greeted the author with the realisation that the agreed project specification was to some extent open-ended. This was solely due to the range of research that required undertaking. Considering this, the author believes that practically the project specification has been satisfied.

The objective of determining the effects on the business factor of profitability have on not been heavily addressed, as this would require investigation on a case by case basis, which the author deemed could not be completed in the allocated time. The aims and objectives of the project specification have been addressed as follows:

Investigation the applicable nuclear science – Chapters 4 & 5 introduce the concepts of ionising radiation, the forms that generally exist and radiations interaction with matter. Chapter 6 summarizes radiation units in use and considers attenuation.

Identification of the applicable measurement science – Chapter 8 investigates the forms of radiation used as part of the measurement system and their generation. Chapter 9 covers the range of detectors available for the detection of ionising radiation and their measurement principles. Chapter 10 introduces the measurement model, the role of statistics in radiation measurement and the basics of signal processing. **Determination of where nuclear instruments are used and their applicability** – In the latter parts of Chapter 10 nuclear instruments a briefly introduced in respect to measurement fundamentals. Chapter 11 follows and gives a substantial account of the use of and application of nuclear instruments within the mineral processing industries.

Nuclear Instrument Safety – Chapter 7 addresses the safety aspects associated with the use of nuclear instruments. The subject of risk is raised and placed in comparison with known outcomes and regular exposures.

Identification of relevant legislation and regulation – Chapter 7 identifies legislative and regulatory requirements. The burden of these requirements is also discussed in chapters 12 & 13.

Site survey of nuclear instrumentation at the Boddington Gold Mine – This has been touched on in Chapter 10 and specifically covered in Chapter 11.

Performance evaluation of both nuclear and non-nuclear instruments – Chapter 12 focuses on the utility, appropriateness, installation and ongoing costs associated with process measurement devices. Chapter 13 follows with a list of proposed recommendations.

Recommendations in respect to nuclear instrument installations – Chapter 13 is an in-depth coverage of what the author believes are suitable recommendations in respect to current nuclear installations and other possible alternatives.

Alternatives to nuclear instruments – Chapter 12 investigates what are believed as viable alternatives, these are further considered under recommendations in Chapter 13.

14.2 Conclusions

From this research it has been ascertained that the use of ionising radiation as part of a measurement system in a mineral processing application is a safe practice. Safety concerns associated with installed nuclear instruments are typically unfounded, with the biggest safety risks appearing to revolve around radiation source security, transport and disposal.

Nuclear instruments provide reliable measurements for often difficult applications, where generally few genuine alternatives exist. These devices have significantly enhanced production and reduced costs. On the subject of spectrometry, analytical instruments using nuclear principles quickly determine concentration with high accuracy, allowing the implementation of process control that has resulted in better quality control and with substantial economic benefit.

The measurement of slurry density within pipes appears as the one area in which viable alternatives exist. It is believed these alternatives reviewed in Chapter 12 would provide better measurement results, with acceptable reliability, but at a higher overall cost. Investigation into the use of x-rays for radioisotope source replacement has found that the use of x-rays is not practical due to low output energies that restrict penetrability.

With reflection the author believes there is further scope for the use of instruments based on nuclear principles within mineral processing fields. Additionally in regard to processes that have variable interfaces, such as in floatation and the use of thickeners, one would think the use of ionising radiation could possibly return excellent process information. However a brief investigation found very little readily available information on this subject of "Process Tomography".

This research has covered a significant amount of subject matter. Throughout this research decisions have had to be made primarily based on meeting the project specification in the time period available, and as such some content could have been more in-depth. The coverage given is believed to be more than adequate, and therefore within all practical means the project specifications have been fulfilled.

14.3 Further work

In regard to this research further works that may be considered are:

- Further investigation into suitability of the alternatives for in-pipe density measurement; with possible trials to determine effective working life.
- The design and implementation of installation specific alternatives that off-theshelf instruments have failed to provide. This may range from the simplicity of using cameras to detect chute blockages, through to technical challenging analytical solutions.
- Investigation, design and implementation of process tomography solutions in industrial applications.

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Appendix A

Project Specification

University of Southern Queensland

FACULTY OF ENGINEERING AND SURVEYING

ENG4111 Research Project PROJECT SPECIFICATION

FOR: TIMOTHY WILLIGEN

- TOPIC: THE REQUIREMENT FOR NUCLEAR INSTRUMENTS IN MINERAL PROCESSING
- SUPERVISOR: Dr. Andrew Maxwell

INDUSTRY REPRESENTATIVE: Antonio Mancuso

PROJECT AIM: To investigate the utility and appropriateness of ionising radiation measurement devices within Mineral Processing. Specifically to examine the utility and functional properties of both nuclear and traditional instrumentation, examining both localised performance comparisons, but also effects on safety, corporate regulatory compliance, and effectiveness and profitability impacts of processing sub stages where nonnuclear alternatives are viable.

PROGRAMME: <u>Issue D, 16th April 2012</u>

- 1. Determine where nuclear instruments are used and their applicability in the Mineral Processing Industry. Investigate the applicable nuclear science associated with the radioactive sources used in measurement systems. Identify and research applicable measurement science used within processing specific measurements.
- 2. Identify and research relevant legislation regarding radiation within mineral processing, impact on instrument performance, safety, and installation costs.
- 3. Conduct a site survey of nuclear instrumentation at available installations (e.g. Boddington Gold Mine).
- 4. Develop performance models for both nuclear and traditional instrumentation, examining: utility; technological appropriateness (fit-for-purpose); installation and ongoing cost; and hazard/safety compliance.
- 5. Contrast and compare these models identifying crossover from nuclear to traditional instrumentation, and the effects on business factors including profitability, measurement reliability, and safety.
- 6. Generate recommendations in respect to nuclear instrument installations.

As time permits:

7. Propose viable measurement/technology alternatives for survey sites.

AGREED:

_____(Student) ______(Supervisor)

Examiner/Co-examiner _____

Appendix B

Dose Limits and Maximum Permissible Exposure Levels



RADIATION SAFETY ACT

Dose Limits and Maximum Permissible Exposure Levels Fact Sheet

Schedule I of the Radiation Safety (General) Regulations prescribes -

dose limits for -

- occupational and public exposure to ionising radiation
- exposure of pregnant workers to ionising radiation

maximum occupational and public exposure levels for -

- radiofrequency radiations
- ultraviolet radiation
- 50 Hz electric and magnetic fields

Ionising Radiation

The dose limits for radiation workers are -

- in any period of 5 years, an average effective dose of 20 millisieverts per year;
- in a period of 12 months, an effective dose of 50 millisieverts;
- in any period of less than 12 months but not less than 1 month, an effective dose
 of the amount which is the product of 50 millisieverts and the ratio of that period in
 weeks to 52 weeks;
- in any period of less than 1 month, an effective dose of 1/12 of 50 millisieverts.

The dose limits for a **radiation worker** who has notified her employer that she is **pregnant** are –

- for external radiation exposure, an equivalent dose to the surface of her abdomen for the remainder of her pregnancy of 2 millisieverts; and
- for internal radiation exposure, 1/20th of the Annual Limit on Intake (ALI) determined by reference to the values set out in the publication entitled "Dose Co-efficients for Intakes of Radionuclides by Workers" being ICRP Publication 68 published for the International Commission on Radiological Protection.

Revision	2.1	Date	22 May 2012		
	Page	1 of 3			

The dose limit in a **single planned special exposure** referred to in regulation 24(2) is an effective dose of 100 millisieverts.

The dose limits for persons other than radiation workers are -

- in any period of 5 years, an average effective dose of 1 millisievert per year;
- in any period of 12 months, an effective dose of 5 millisieverts; and
- in respect of an area which such persons might continuously occupy -
 - an effective dose of 20 microsieverts in any 1 hour; and
 - an effective dose of 250 microsieverts in any period of 7 days.

Non-ionising Radiation

Persons occupationally or non-occupationally exposed to the following radiations shall not be exposed to –

- 50/60 Hz electric and magnetic fields which exceed the limits specified in the publication entitled "Interim Guidelines of the Limits of Exposure to 50/60 Hz Electric and Magnetic Fields (1989)" published by the NHMRC in December 1989;
- low frequency electromagnetic radiation with frequencies from 3 kHz to 100 kHz which exceed the limits specified in the publication entitled "IEEE Standard for Safety Levels with respect to Human Exposure to Radio Frequency Electromagnetic Fields, 3 kHz to 300 GHz" published as IEEE C95.1-1991 by the Institute of Electrical and Electronics Engineers;
- radiofrequency fields with frequencies from 3 kHz to 300 GHz which exceed the limits specified in the publication entitled "Maximum Exposure Levels to Radiofrequency Fields – 3 kHz to 300 GHz" published by the Australian Radiation Protection and Nuclear Safety Agency in May 2002; or

(This item does not apply to exposure resulting from the normal operation of microwave ovens which comply with the radiation emission limit set out in the microwave oven standard).

• ultraviolet radiations with wavelengths from 180 nm to 400 nm which exceed the limits specified in the publication entitled "Occupational Standard for Exposure to Ultraviolet Radiation (1989)" published by the NHMRC in December 1989.

Legislation

Complete sets of the Act and Regulations are available from the State Law Publisher, www.slp.wa.gov.au.

Contact Us

Mail	The Secretary Radiological Council Locked Bag 2006 P O NEDLANDS W A 6009
Office	Grace Vaughan House 227 Stubbs Terrace, Shenton Park W A 6008
Phone	+61 8 9388 4999
Facsimile	+61 8 9382 0701
Email	radiation.health@health.wa.gov.au
Web	www.radiologicalcouncil.wa.gov.au

Appendix C

Australian Regulatory Authorities

Appendix C

Australian Radiation Regulatory Authorities Source: Australian Radiation Protection and Nuclear Protection Agency, 2012

Australian Capital

Commonwealth

Chief Executive Officer	Territory
Australian Radiation Protection	Manager Radiation Safety
and Nuclear Safety Agency	Health Protection Service
PO Box 655	ACT Health Directorate
MIRANDA NSW 1490	Locked Bag 5005
Email: info@arpansa.gov.au	WESTON CREEK ACT 2611
Telephone: (02) 9541 8333	Email: <u>hps@act.gov.au</u>
Web: www.arpansa.gov.au	Telephone: (02) 6205 1700
	Web: www.health.act.gov.au
New South Wales	Northern Territory
Manager Hazardous Materials, Chemicals and	Manager Radiation Protection

Manager Hazardous Materials, Chemicals and	Manager Radiation Protection
Radiation Environment Protection Authority	Radiation Protection Section
PO Box A290	Department of Health
SYDNEY SOUTH NSW 1232	GPO Box 40596
Email: radiation@epa.nsw.gov.au	CASUARINA NT 0811
Telephone: (02) 9995 5959	Email: envirohealth@nt.gov.au
Web: www.environment.nsw.gov.au/radiation/index.htm	Telephone: (08) 8922 7152
	Web: www.health.nt.gov.au

Queensland

Director Radiation Health Unit Queensland Health PO Box 2368 FORTITUDE VALLEY BC QLD 4006 Email: radiation health@health.qld.gov.au Telephone: (07) 3328 9987 Web: www.health.qld.gov.au/radiationhealth/

Tasmania

Senior Health Physicist Radiation Protection Unit Dept of Health and Human Services GPO Box 125 HOBART TAS 7001 Email: radiation.protection@dhhs.tas.gov.au Telephone: (03) 6222 7256 Web: www.dhhs.tas.gov.au/peh/radiation_protection

Northern Territory

Manager Radiation Protection Radiation Protection Section Department of Health GPO Box 40596 CASUARINA NT 0811 Email: envirohealth@nt.gov.au Telephone: (08) 8922 7152 Web: www.health.nt.gov.au

Victoria

Team Leader, Radiation Safety Department of Health GPO Box 4541 Melbourne VIC 3001 Email:radiation.safety@health.vic.gov.au Telephone: 1300 767 469 Web: www.health.vic.gov.au/environment/radiation/

Western Australia

Secretary

Radiological Council

Grace Vaughan House

277 Stubbs Terrace

SHENTON PARK WA 6008

Locked Bag 2006 PO NEDLANDS WA 6009

Email: radiation.health@health.wa.gov.auTelephone:(08) 9388 4999Web: www.radiologicalcouncil.wa.gov.au

Appendix D

Registration of Radioactive Substances

RADIATION SAFETY ACT

FIRST APPLICATION FOR REGISTRATION RADIOACTIVE SUBSTANCES

Application forms are enclosed for the registration of premises and radioactive substances. Please complete all applicable sections and return the signed form with the fee* to the Radiological Council, Locked Bag 2006 P O Nedlands W A 6009. **Faxed applications or photocopies of applications cannot be accepted.**

Registration applies to the premises where radiation is to be used as well as the types, form and quantities of the radioactive substances that are to be used or kept on those premises or at field sites. The applicant must provide an inventory on the supplementary form of all radioactive substances, as well as the names, qualifications and relevant training of persons who will be using the substances.

Radiation Safety Officer (RSO):

The applicant must provide full details of the qualifications, training and experience of the person nominated to hold this position. The nominee may be required to attend an appropriate radiation safety training course, pass an examination in radiation safety, and may be required to hold a licence as the person responsible for the use of radiation on the premises. *If the nomination is approved by the Radiological Council, the Registrant is required to inform the RSO of this appointment in writing.* Separate RSOs may be nominated for x-rays, lasers and transilluminators.

* **Fees:** The fee schedule should be provided with this letter. Fees are GST free. A receipt will only be issued on request.

Payment should be made to the Radiological Council by cheque, money order or cash. **Purchase orders and credit cards cannot be accepted**. Electronic funds transfer is not available.

If further assistance is required, please telephone (08) 9388 4999.

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Ms H Upton Secretary, Radiological Council

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RADIATION SAFETY ACT 1975 RADIOACTIVE SUBSTANCES REGISTRATION FEES

The Tax Invoice table below should be completed and returned with your registration application. A copy should also be kept for your records.

Fee Exemption: The fee is **NOT** payable if you are a State Government organisation recognised *by Treasury* as '*non-chargeable*' or you are applying for, or renewing, an Exemption from Registration. Hospitals are chargeable organisations.

Payment should be made to the Radiological Council by cheque, money order or cash and should accompany the application. *A receipt will be issued on request.*

Radioactive Substances (Maximum Activity on premises)	1 year fee	3 year fee
40 Gigabecquerels	\$ 138	\$ 277
400 Gigabecquerels	\$ 277	\$ 553
4 Terabecquerels	\$ 553	\$1106
> 4 Terabecquerels	\$ 830	\$1659

FEE SCHEDULE

NOTE If your application concerns one or more of the following only, the applicable fee is — \$138 for 1 year or \$277 for 3 years.

Sales and/or Service	You are only registering premises where devices containing radioactive substances are sold or serviced and their presence on the premises is temporary
Mining and Milling of Radioactive Ores	You are registering premises used solely for the mining and milling of radioactive substances. If other radioactive substances are also used, stored, or dealt with on the premises, determine the fee from the Fee Schedule above based on the total activity of those substances. That fee is then the sole fee payable.
Transport of Radioactive Substances (Storage)	You are registering premises where radioactive substances are stored temporarily during transport. Registration is NOT required for storage of less than 24 hours provided a licensed person is responsible.

TAX INVOICE FEES ARE GST FREE

Total Maximum Activity	1 year registration	3 year registration	GST	TOTAL	
	\$	\$	N/A	\$	

Please return the application form WITH the payment. For continuity, a renewal application MUST be received BEFORE the expiry date (section 37(2) of the Radiation Safety Act)

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FORM RS10 WESTERN AUSTRALIA

RADIATION SAFETY ACT 1975

INITIAL APPLICATION FOR REGISTRATION¹ OF PREMISES

in which

RADIOACTIVE SUBSTANCES

are to be used, stored or manufactured

Please complete **BOTH** this application and the supplementary form. Return both signed forms with the fee (see Note (c)). Where space is insufficient for any item, attach additional signed sheets.

1.	Name and Address of the Applicant. (See Note (a)). Renewal not	ices will be s	ent to t	his add	ress		
		Tel					
		Fax					
	e-m	il					
2.	Location of the Premises Subject to Registration						
3.	Occupation, nature of business, etc.						
4.	Name, qualifications and experience of the person nominated to responsibilities are given in regulations 18 and 19 of the Radiatio title, first name, next initial, last name and date of birth.	be the Radi 1 Safety (Ger This infor	ation S teral) I mation	afety C Regulat helps i)fficer ions). I in corre	(This p Please ctly ide	person's duties and give the nominee's entifying records.
5.	Names, qualifications and experience of persons licensed or othe Please complete the details on the attact	rwise autho hed SUPPLE	rised to	o deal v TARY fo	with the	e radio	active substances
6.	Purpose(s) for which the radioactive substances are to be used o <i>Please complete the details on the atta</i>	r manufactu ched SUPPL	red. EMEN	TARY f	orm.		
7.	Particulars of the radioactive substances to be used, stored or m <i>Please complete the details on the atta</i>	anufactured	on the	premi s TARY fo	ses. orm.		
 8. 9. 10. 11. 12. 	Please provide answers to Items 8 - 12 For radioactive substances obtained regularly, state the quantity in o What radiation monitoring instruments are available on the prem monitoring of personnel? State the nature and proposed method of disposal of radioactive wa What protective equipment is available and what handling techniqu What is the location of the storage facility and of what is it construct	on a separa ach shipmen ises and what ate or of seale es are propos ted? Is it use	ate sign t and the at arran ed sour- ed for a	ned sh ne frequ ngemen ces whi the radi ny othe	eet nency of ts have ach are r toactive er purpo	f supply been no long substa	y. made for radiation ger required. nces?
NOT	ES:						
a)	The 'owner' of the premises (the applicant) is defined in the Act to include attorney, agent, manager, foreman, supervisor, or other person in charge or representing himself to be acting for the owner.	the hirer, lesse r having contr	e, borro ol or ma	wer, bai anageme	lee, mort ent there	tgagee i of, and	n possession and any any person acting or
b)	A SCALE PLAN of the premises must be provided with the application sh manufactured or stored. The plan must show the purpose of all adjacent substances, additional information on waste lines, laboratory facilities, surfa-	owing the loca areas and the ce finishes and	tion wh nature d ventila	ere the 1 of the c ation is a	radioacti construct also requ	ve subs ion mat ired.	tances are to be used terials. For unsealed
c)	The fee schedule is attached. (Exemptions from Registration are exempt from	m fees).			F	NOUI	RIFS
d)	Return to Radiological Council, Locked Mail Bag 2006, P O Nedlands V	V A 6009			Tel:	(08) 9	388 4999
Th sig re	nis application and any supplementary forms cannot be processed gned by the person in whose name the premises are to be re- sponsibilities of that person (and the RSO) are set out in the Act and the	without beir jistered. Th e regulations	ng ne	radi Pl	Fax: iation.he	(08) 9 ealth@l ied, fax	382 0701 nealth.wa.gov.au ked or e-mailed
NA	ME of Applicant			ap Ple	ase reti signe	urn the ed origi	ot be accepted. completed and nal forms.
rU	SITION of Applicant						
SIC	SNATURE of Applicant]	Date		
Off	fice use: Fee Paid Receipt No		F	Period		Date	

¹ and/or Exemption from Registration

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FORM RS10 June 2012

SUPPLEMENTARY FORM RS10

Copy this form if space is insufficient, or provide a separate list

NAMES of PERSONS LICENSED to handle or otherwise deal with the radioactive substances

Please complete this section

LAST NAME	FIRST NAME(S)	TITLE	POSITION HELD	LICENCE NUMBER		

RADIOACTIVE SUBSTANCES

Complete this section ONLY if an inventory has NOT been provided with the renewal application OR that inventory is INCOMPLETE

NUCLIDE (eg ⁶⁰ Co)	SOURCE SERIAL No.ª	FORM	MAXIMUM ACTIVITY on PREMISES or activity in	PURPOSE or USE	LOCATION on PREMISES	DEVICES OR EQUIPMENT WHICH CON THE RADIOACTIVE SUBSTANCES (where applicable)		VT WHICH CONTAIN E SUBSTANCES licable)	
			each device listed			MANUFACTURER	MODEL	SERIAL No.	URANIUM (kg) ^b

^asealed sources

^bwhere depleted uranium is used as radiation shielding

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