

https://dspace.org/

Derivation of Operational Intervention Levels for the early phase of radioactive material at Koeberg Nuclear Power Station

Trollope, Ian Douglas

2015-01-29
http://hdl.handle.net/10539/16810

Downloaded from a DSpace repository. https://dspace.org/

Derivation of Operational Intervention Levels for the Early Phase Release of Radioactive Material at Koeberg Nuclear Power Station

Ian Douglas Trollope

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

Johannesburg, 2014

DECLARATION

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



Ian Douglas Trollope

8th day of October 2014 in Cape Town

ABSTRACT

An investigation was performed to look at a method to develop easy to use field survey measurements to assist decision makers in the process of deriving public protective actions. This method could be used at a nuclear power plant if certain accident conditions are known. International values for operational intervention levels (OIL's) do exist and are recommended to be employed if station specific data has not been derived. No values exist specific to Koeberg Nuclear Power Station and as a result, this became an ideal opportunity to derive station specific values.

It was firstly necessary to decide on a specific accident type and hence an applicable accident release fraction. A suitable accident software dispersion code was applied to calculate the organ doses for the selected accident type. It was also decided to use two different wind dispersion criteria to further refine the results. Due to the complexities of dose distribution within the body it was also necessary to look at the gamma dose in isolation as this would be the measurement radiation type utilised as a limit in the field either using installed radiation monitors or by physical measurement performed by station Radiation Protection staff.

Comparisons were done with thyroid and lung dose versus gamma dose to arrive at ratios for this specific accident type. This would then be indicative of the total dose to each organ as a result of a single field measurement.

Conclusions were drawn on the results obtained and recommendations were made for when this type of data may be suitable for use in the unlikely event of a nuclear accident.

DEDICATION

To my wife Ronelle, my mother Jean and the rest of my family and colleagues whose belief in me and invaluable support gave me the strength and motivation to continue until completion.

TABLE OF CONTENTS …….. Page

DECLARATION…..……...…….... ii

ABSTRACT….……...…….... iii

DEDICATION…….…….. iv

LIST OF FIGURES…...…….. vii

LIST OF TABLES…...…….. ix

CHAPTER ONE – INTRODUCTION……... 1

1.0 General Introduction…….. 1

1.1 The Fukushima Incident……. 2

1.2 Introduction to Emergency Action Levels……...….. 3

1.3 Introduction to Operational Intervention Levels……...…. 4

1.4 Project Outline……... 5

CHAPTER TWO – THEORETICAL CONSIDERATIONS……...…… 6

2.0 Terminology……... 6

2.1 Emergency Planning…….. 6

2.1.1 Emergency Preparedness……... 6

2.1.2 Emergency Preparedness Approach Adopted by Eskom……... 6

2.1.3 Emergency Planning Zones…… 8

2.2 Biological Effects……... 10

2.2.1 Deterministic Effects…….. 10

2.2.2 Stochastic Effects……... 11

2.3 Dose Terminology…….. 12

2.3.1 Absorbed Dose……... 12

2.3.2 Equivalent Dose……. 12

2.3.3 Effective Dose……...……. 13

2.3.4 Committed Equivalent Dose……...….... 14

2.3.5 Committed Effective Dose……...…….. 15

2.4 Basis for Types of Dose Limits……..……...……. 15

2.4.1 Projected Dose……... 15

2.4.2 Averted Dose…….. 16
2.4.3 Reference Levels…….. 17
2.4.4 Residual Dose……. 19
2.5 Dose Limits…… 19
2.5.1 Koeberg Procedural Dose Limits…….. 19
2.5.2 Legal Dose Limits…….. 21
2.5.3 Dose Limits Based on Generic Criteria……. 22
2.6 Operational Intervention Levels..…….. 26
2.7 Emergency Action Levels…….. 26
2.8 Urgent Protective Actions…….. 27
2.8.1 Evacuation…….. 27
2.8.2 Sheltering……... 27
2.8.3 Iodine Prophylaxis……. 28
2.9 Longer Term Protective Actions…… 28
2.10 Field Measurements……... 28
2.10.1 Photoelectric Absorption……... 29
2.10.2 Compton Scattering……..... 30
2.10.3 Pair Production……..... 31

CHAPTER THREE – PC COSYMA AND INPUT DATA…….. 33

3.0 Introduction……..... 33
3.1 Atmospheric Dispersion and Deposition…… 33
3.2 The Gaussian Plume Model…….……... 33
3.3 Meteorological Data and its Sampling…….... 35
3.4 Koeberg Source Term…….……..... 36
3.5 Reference Accident for the Project……...……... 39
3.6 Accident Release Fraction…….. 41
3.7 Structure of PC COSYMA …..…….. 42
3.7.1 General Assumptions……...…….. 43
3.8 The Input Interface …..…….. 43
3.9 Description of the Radionuclide Release……... 44
3.10 Exposure Pathways and Dose Assessment Calculations……...……. 44

3.11 Radioactive Decay Model…….. 46
3.12 External Dose from a Passing Cloud……. 46
3.13 External Dose from Deposited Radionuclides……... 46
3.14 Inhalation Dose from a Passing Cloud…….. 47
3.15 Inhalation Dose from Resuspended Radionuclides…… 47

CHAPTER FOUR - RESULTS…… 48

4.1 Overview…… 48
4.2 Analysis……... 85

CHAPTER FIVE - GENERAL DISCUSSION AND CONCLUSION 92

5.1 Discussion…….. 92
5.2 Improvements……. 95
5.3 Concluding Remarks…….. 96

REFERENCES…… 98

LIST OF FIGURES

Figure 2.1: Koeberg Emergency Planning Zone…….. 8
Figure 2.2: Deterministic Effects…….. 10
Figure 2.3: Stochastic Effects……... 11
Figure 2.4: Projected Dose Determined after Start of an Accident…….. 16
Figure 2.5: Avertable dose…… 17
Figure 2.6: System of Generic Criteria and Operational Criteria…….....…… 23
Figure 2.7: Photoelectric Absorption…….……...…… 29
Figure 2.8: Compton Scattering……..……...… 30
Figure 2.9: Pair Production……..…….…… 31
Figure 2.10: Annihilation……..……..……..…… 32
Figure 3.1: Gaussian plume dispersion model for a continuous point source……. 34
Figure 4.1: 1 Day Dose – D Dispersion Category – Gamma Only……...…... 50
Figure 4.2: 1 Day Dose – D Dispersion Category…… 52
Figure 4.3: 1 Day Dose – F Dispersion Category…… 54
Figure 4.4: 2 Day Dose – D Dispersion Category…… 56
Figure 4.5: 2 Day Dose – F Dispersion Category…… 58
Figure 4.6: 7 Day Dose – D Dispersion Category…… 60
Figure 4.7: 7 Day Dose – F Dispersion Category…… 62
Figure 4.8: 30 Day Dose – D Dispersion Category…….. 64
Figure 4.9: 30 Day Dose – F Dispersion Category…….. 66
Figure 4.10: 50 Year Dose – D Dispersion Category……... 68
Figure 4.11: 50 Year Dose – F Dispersion Category……... 70
Figure 4.12: D Dispersion Category – Thyroid Dose / Gamma Dose Ratios…….. 76
Figure 4.13: D Dispersion Category – Lung Dose / Gamma Dose Ratios……... 77
Figure 4.14: F Dispersion Category – Thyroid Dose / Gamma Dose Ratios…….... 83
Figure 4.15: F Dispersion Category – Lung / Gamma Dose Ratios……. 84

LIST OF TABLES

Table 1.1: Examples of the Default OILs for Field Survey Measurements..…… 5
Table 2.1: Recommended Radiation Weighting Factors…….. 13
Table 2.2: Recommended Tissue Weighting Factors……... 14
Table 2.3: Generic Intervention Levels…… 17
Table 2.4: Emergency Intervention Levels for Urgent Protective Actions……... 20
Table 2.5: Generic criteria for acute doses for which protective actions and other response actions are expected to be taken under and circumstances to avoid or minimise severe deterministic effects…….. 24
Table 2.6: Generic criteria for protective actions and other response actions in emergency exposure situations to reduce the risk of stochastic effects……. 25
Table 3.1: The Final Inventory for AFA-3GA, 4,4% Enriched Fuel……..... 37
Table 3.2: Koeberg Accident Release Categories…..……... 40
Table 3.3: Nuclide Grouping Scheme used in PC COSYMA Calculations……..... 40
Table 3.4: Summary of Release Fractions for RC-3……..……... 42
Table 4.1: 1 Day Dose - D Dispersion Category - Gamma only……..…… 49
Table 4.2: 1 Day Dose - D Dispersion Category…….. 51
Table 4.3: 1 Day Dose - F Dispersion Category…….. 53
Table 4.4: 2 Day Dose - D Dispersion Category…….. 55
Table 4.5: 2 Day Dose - F Dispersion Category…….. 57
Table 4.6: 7 Day Dose - D Dispersion Category…….. 59
Table 4.7: 7 Day Dose - F Dispersion Category…….. 61
Table 4.8: 30 Day Dose - D Dispersion Category……... 63
Table 4.9: 30 Day Dose - F Dispersion Category…… 65
Table 4.10: 50 Year Dose - D Dispersion Category……. 67
Table 4.11: 50 Year Dose - F Dispersion Category……. 69
Table 4.12: 1 Day - D Dispersion Category Gamma Ratios…… 71
Table 4.13: 2 Days - D Dispersion Category Gamma Ratios……... 72
Table 4.14: 7 Days - D Dispersion Category Gamma Ratios……... 73
Table 4.15: 30 Days - D Dispersion Category Gamma Ratios……. 74

Table 4.16: 50 Years - D Dispersion Category Gamma Ratios…… 75
Table 4.17: 1 Day - F Dispersion Category Gamma Ratios……. 78
Table 4.18: 2 Days - F Dispersion Category Gamma Ratios……... 79
Table 4.19: 7 Days - F Dispersion Category Gamma Ratios……... 80
Table 4.20: 7 Days - F Dispersion Category Gamma Ratios……... 81
Table 4.21: 50 Years - F Dispersion Category……. 82
Table 4.22: 1 Day Ratios - D Dispersion Category……. 85
Table 4.23: 7 Day Ratios - D Dispersion Category…….....…. 86
Table 4.24: Recommended OILs with D Dispersion Category using Deterministic Limits……..... 88
Table 4.25: Recommended OILs with D Dispersion Category using Stochastic Risk Limits……...…….. 89
Table 4.26: Recommended OILs with F Dispersion Category using Deterministic Limits……..... 90
Table 4.27: Recommended OILs with F Dispersion Category using Stochastic Risk Limits……...…….. 91

CHAPTER ONE – INTRODUCTION

1.0 General Introduction

The generation of electricity using nuclear fission has been a controversial topic in the public domain especially since the Three Mile Island accident in Dauphin County Pennsylvania, United States on the 28th of March 1979. After this accident there was also the Chernobyl accident which occurred on the 26th of April 1986. As a result, many years have passed with very little expansion of the nuclear network. The technology could not be ignored as a viable means of generating electricity. This resulted in improvements made in the design of new generation reactors and safety reassessments performed on operating units. Some of the older stations and plants that did not make the necessary changes to enhance the safety of the station were removed from production and are in the process of decommissioning.

By the beginning of 2011, national regulators and governments all over the world were generally of the opinion that nuclear power was adequately explored to be declared a safe and viable means for meeting the energy needs of the increasing world population. Large Power Stations have not been built in South Africa for many years which has and will continue to strain the existing demand for electricity. As a quick, however expensive mitigation, diesel turbine generators were constructed and have been used extensively to ensure that the country's electricity demand is met. This is however a very expensive option which negatively impacts on the environment and the cost of electricity to the consumer.

Eskom's fleet currently relies heavily on coal fired power stations. Due to the fact that fossil powered stations result in a large carbon footprint, world leaders are being pressurised to make use of renewable energy sources. Nuclear power does not fall into the renewable category due to the production of radioactive waste products as well as the risks associated with accident conditions. Nuclear power is a relatively cost effective and environmentally friendly alternative during normal operation; however initial building costs and the time required to complete the building and commissioning of such a project is challenging.

Nuclear power has been produced efficiently and safely in South Africa since Unit One was synchronised with the national grid in 1984 proving that this type of technology can be operated safely in the South African context. Eskom has invested extensively on accident mitigation modifications in the last 10 years to ensure that Koeberg is as safe as the safest plants in the world.

The South African Government, Eskom and other stakeholders are in the process of investigating different technologies to increase electricity capacity applicable to the needs of the South African market. Nuclear is seen to be a viable source of the production of electricity to ensure a reliable electricity supply to the consumers.

1.1 The Fukushima Incident

On 11 March 2011 there was an earthquake measuring 8.9 on the Richter scale with the epicentre 180 km north east of the Fukushima District “Daiichi” nuclear power station (INPO, 2011). Three of the four units were running at full power at the time of the earthquake. The units tripped and shut down as per the design basis after the earthquake. The earthquake caused substantial damage to the power lines resulting in a loss of off-site power. All the units emergency diesel generators started and safety and cooling systems were being powered and the plant was in a safe configuration. The operators were busy investigating any damage to the plant after the earth quake. About approximately half an hour later, a series of tsunami waves up to a height of about 14 m causing considerable damage and shut down of all station emergency generators resulting a total loss of on-site electrical power. This incident has once again resulted in an international review of all operational nuclear power stations in terms of being able to mitigate against natural disasters of this magnitude.

Determining the level of emergency preparedness in the event of an accident resulting in the release of radioactive material from a Nuclear power Station is extremely difficult. Previous accidents resulting in radiological releases to the environment such as Chernobyl and the more recent Japanese Fukushima event

have proven that it is necessary to be able to handle such an accident at very short notice. Plans need to be in place to mitigate the emergency situation at the facility to ensure that there is no release of radioactive material from the facility.

It is extremely difficult to plan for any eventuality especially when dealing with external events that initiate an accident such as earthquakes and Tsunamis. The international expectation, however, is that there need to be plans in place to deal with “Beyond Design Basis” events such as what happened at the Fukushima Daiichi plant.

The principle of defence in depth ensures that a number of independent safety systems would need to fail to result in a release of radioactive material to the environment. This however, will not be discussed in detail for the purpose of this project. There is an international school of thought amongst emergency planners that suggest that the criteria to decide on various protective actions taken during an accident need to be simple with little or no calculations required during the early phase of decision making. The thinking behind this is that the decision makers will be under extreme stress and their focus will be on mitigation of the incident to prevent a release of radioactive material to the environment. These extreme circumstances will significantly degrade their ability to carry out complex modelling when errors may result in catastrophic consequences.

1.2 Introduction to Emergency Action Levels

Emergency Action Levels (IAEA, 2007) are typically plant operational indicators, which, if exceeded, will result in an action that needs to be taken in terms of public protection. Emergency action levels have been recommended in various IAEA documents (IAEA, 2007; IAEA, 2011) and prescribe various plant conditions which, can be used to consider the classification of an accident. An example of the use of plant conditions was with the Fukushima accident, such as loss of both on and off site power supplies, resulting in no core cooling systems being available. As a result, an emergency condition was declared, which resulted in notification and evacuation of the public prior to the release of radioactive

material to the environment. These action levels are becoming increasingly important to decision makers, however in this research project, an alternative approach will be taken, which is detailed more in Chapter two.

1.3 Introduction to Operational Intervention Levels

An alternative method for determining the degree and magnitude of an emergency is by taking a number of radiological measurements during the early phase of a release and then comparing them against a number of predetermined values also known as operational intervention levels (OILs) which, if exceeded automatically result in a number of predetermined response actions. These actions can include evacuation, sheltering or iodine prophylaxis which will be explained in more detail in Chapter two.

This method of using measured radiation results after a release, however, can be counterproductive as the evacuation of the public during a release will result in relatively high doses to the effected population, including the fact that they would then need to be monitored for contamination and possibly decontaminated if required. This method is however widely recognised and utilised once again to recommend public protective actions.

To further complicate this method, there are different field measurement of OILs for different technologies and accident types. The International Atomic Energy Agency have recommended generic OIL's for stations which do not have accident specific OILs (IAEA, 2011; IAEA, 2005; IAEA, 1997). To show an example of the actual operational use of OILs, see Table 1.1 below with the measurable criteria and the expected response or protective action.

Table 1.1: Example of the Default OILs for Field Survey Measurements (Taken from Ref. [7])

OIL No	OIL Value	Response action if OIL is exceeded
OIL 1	Gamma Dose Rate >1 mSv/h at 1 m from surface	Immediately evacuate or provide substantial shelter; Register and provide for medical examination of evacuees.
OIL 2	Gamma Dose Rate 0.1 mSv/h at 1 m from surface	Temporarily relocate those living in the area within days; Reduce inadvertent ingestion; Register and estimate the dose to those who are in the area to determine if medical screening is warranted.

The outcome of this project is to calculate Koeberg specific OILs for a single accident type, which will be discussed later, for quick decision making during the early phase of a nuclear accident. The same process can then be followed for all postulated accident types and then included in Koeberg emergency operating procedures. This will result in OIL's being available for immediate use, thereby removing the need for decision makers to perform complicated calculations during stressful conditions.

1.4 Project Outline

Chapter two looks at and defines different terminology to assist the reader in understanding the concepts that are discussed. It is important for the reader to be able to understand the concepts that need to be taken into consideration to ensure that the outcome has been achieved. Chapter three provides the background and assumptions that were used to arrive at an outcome. The software programme known as PC COSYMA (Jones J.A., 1996) is also discussed in this chapter. In Chapter four the results are shown both graphically and in table formats. This was then analysed by the author and opinions and conclusions made in the discussion. This work may assist any other Emergency Planners in the nuclear field to attempt to simplify the work required by decision makers in the event of a nuclear incident or accident.

CHAPTER TWO – THE0RETICAL CONSIDERATIONS

2.0 Terminology

Due to the specialised terminology specific to Nuclear Power plants and the release of radioactive material, it is important to list and define those which will be utilised to achieve the expected outcome of this research project.

2.1 Emergency Planning

Emergency planning consists of two separate aspects, namely the preparedness and response. To measure the effectiveness of emergency preparedness defined in see Section 2.1.1, real time exercises are utilised to physically measure the effectiveness of the plan against specific predetermined objectives. The exercises are outside the scope of this report.

2.1.1 Emergency Preparedness

Emergency preparedness is the function of an operator of a facility to effectively plan the response requirements in the event of a nuclear accident. The main outcome is firstly the mitigation of the accident preventing the release of radioactive material to the environment, and secondly, the response capabilities in the event of a release. Koeberg has a dedicated team known as Emergency Management looking at on and off site consequences taking into consideration the various incidents and accidents types that could take place at the Koeberg Nuclear Power Station. This is looked at in a lot more detail in Chapter three.

Emergency preparedness in terms of accident mitigation from a plant perspective is also controlled by a separate dedicated accident management group. This group consists of highly skilled and trained plant operators, accident mitigation advisors, procedure specialists and system engineers. Their role ultimately is to prevent the release of radioactive material to the environment by protecting the reactor fuel from melting using applicable procedures and guides.

2.1.2 Emergency Preparedness Approach Adopted by Eskom

Eskom has adopted the IAEA approach with respect to roles and responsibilities in the event of a radiological emergency occurring at Koeberg (IAEA, 2002). This

is further supported by the Disaster Management Act, No 57 of 2002 (Minister of Provincial and Local Government, 2004), which requires that government has the ability to handle all disaster types within their jurisdiction. The National Nuclear Regulator Act, No 47 of 1999 (Minister of Minerals and Energy, 1999) requires that Eskom, as the holder of the nuclear license, has an agreement with relevant municipalities and provincial authorities to establish an emergency plan. A memorandum of agreement is in place signed by Local and Provincial Government and Eskom specifying the requirements of the act. The specific responsibilities of the Integrated Koeberg Nuclear Emergency Plan are documented (Eskom, 2011).

In terms of this research project only the establishment of radiological limits for the early phase of a nuclear accident will be calculated. The early phase is typically from hours before the possibility of a release until the time that the release has been terminated. The phase after the early phase is typically known as the late phase or recovery phase (IAEA, 2007). Various IAEA publications after the Chernobyl accident that cover emergency response (IAEA, 2011; IAEA, 2005) mention that the decision makers did not have clear guidelines to public protective action implementation limits during this early stage or phase of the accident. The IAEA has recommended that operators should derive station specific limits at which action should be taken for public protection by using operational intervention levels (OIL) specific to the utility (IAEA, 2011; IAEA, 1997; IAEA, 2005).

In line with the concept of developing OILs, it is recommended during the early phase of an accident that simple unambiguous limits are derived during the station emergency preparedness programme. In the event of an incident or accident, it is difficult to calculate or derive criteria while coping with a multitude of challenges when dealing with the emergency. Uncertainty and reversing or changing public protective action that have been ordered will result in public mistrust. It is of utmost importance that protective actions ordered to the public are accurate and correct. If actions are changed or not justified, the public may not obey

instructions, which could result in chaos or at the very least, make it very challenging for officials to carry out their tasks.

2.1.3 Emergency Planning Zones

Koeberg's formal emergency planning zones are specified in the plan (Eskom, 2011) at distances of 5 km (PAZ) and 5 to 16 km (UPZ). The centre point is in the centre of the plant between the two reactor buildings, and the zones are shown in Figure 2.1.

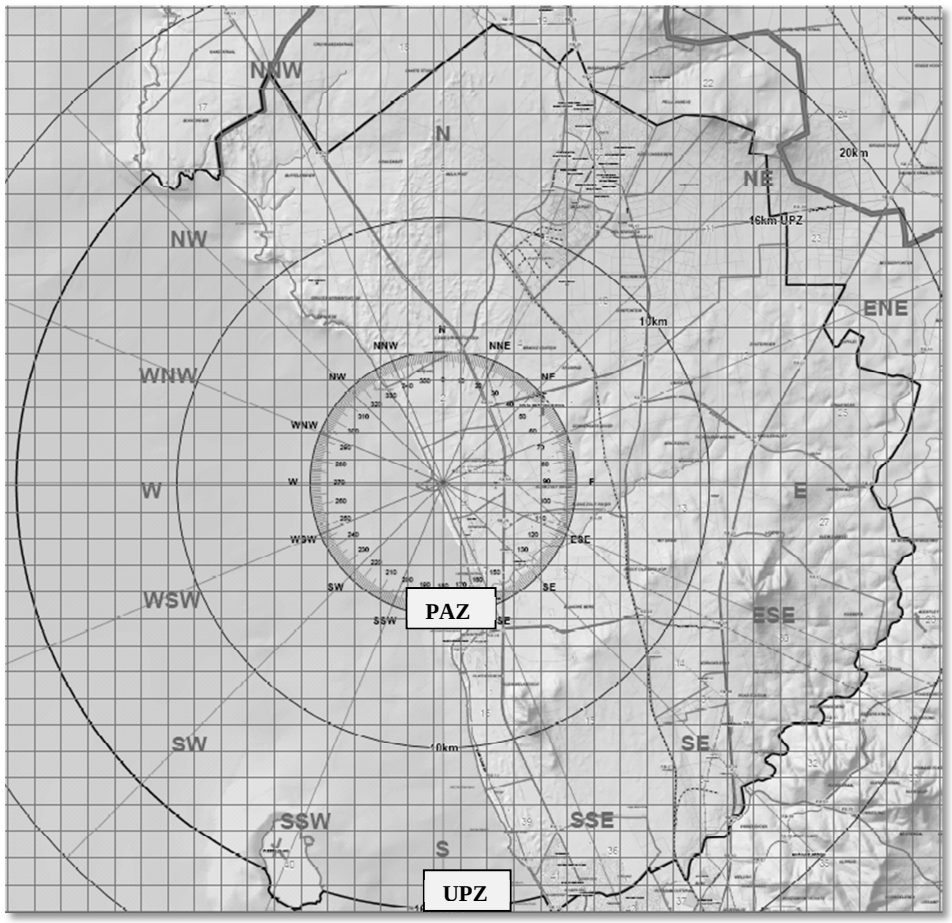

Figure 2.1: Koeberg Emergency Planning Zone

2.1.3.1 Precautionary Action Zone (PAZ)

The 5 km radius from the release point (IAEA, 2007; IAEA, 2003) is known as the Precautionary Action Zone (PAZ), see Figure 2.1, where arrangements must

be made to take protective actions such as evacuation or sheltering of the public. These actions, if possible, need to be taken prior to the start of any release to prevent the occurrence of severe deterministic effects (IAEA, 2007) should there be a significant release. In the case of Koeberg, on declaration of an accident that could affect the public, Koeberg will pre-emptively recommend evacuation in this zone. The limits that are utilised as triggers for the recommendation of protective actions are specified in the Koeberg Plan (Eskom, 2011).

2.1.3.2 Urgent Protective Action Zone (UPZ)

The area between 5 km to 16 km radius from the release point (IAEA, 2007; IAEA, 2003) is known as the urgent protective action zone (UPZ), see Figure 2.1. The requirement is also that arrangements must be in place to take protective actions in the event of a nuclear emergency and avert the dose to the public in accordance with international standards. In this case the international standard is the IAEA requirements document, GS-R-2 (IAEA, 2002). The driver behind the implementation of urgent protective actions is to ensure that more benefit than harm is achieved by the implementation thereof.

2.1.3.3 Longer Term Protective Action Zone (LPZ)

Koeberg has a longer term protective action zone (LPZ) (Eskom, 2011; Eskom, 2012) with a radius of 80 km from the centre of the plant. The reason for this zone is for longer term protective actions such as food banning and possibly having to move members of the public out of their homes or businesses temporarily until conditions improve either by decontamination or radioactive decay of the radioactive material. Eskom reviews the modelled impact of Koeberg on the public periodically with the last review being performed in 2011 (Eskom, 2011). The results conclude that the existing zone sizes remain applicable for Koeberg.

2.2 Biological Effects

The two objectives of an emergency plan in terms of biological effects after an exposure to radiation, are:

- To prevent the occurrence of deterministic effects, and,
- To keep the risk of stochastic effects as low as is reasonably achievable (IAEA, 2002).

The difference between deterministic effects and stochastic effects is explained in more detail in Section 2.2.1 and Section 2.2.2 respectively.

2.2.1 Deterministic Effects

Deterministic effects (IAEA, 2007), previously known as non-stochastic effects, are effects where there is a threshold dose, see Figure 2.2, from which effects will occur and there is a clear relationship between dose and effect. Examples of this are erythema (skin reddening), cataracts, radiation illness and death. These effects become more severe as the dose increases. Severe deterministic effects are effects that range from fatal exposures to those that are life threatening or the resultant effect results in a permanent injury that negatively influences the quality of life. Deterministic threshold dose limits are limits which should not be exceeded and every action must be carried out to ensure that individuals do not exceed these limits.

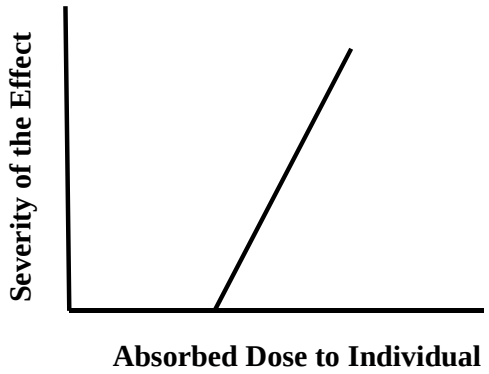

Figure 2.2: Deterministic Effects

2.2.2 Stochastic Effects

Stochastic effects (IAEA, 2007) are effects that occur based on the probability of occurring after an exposure, see Figure 2.3. The higher the dose the higher the probability of the effect occurring. Examples include solid cancers and leukaemia. It is assumed that there is no threshold dose, but the risk of occurrence increases with the amount of radiation exposure.

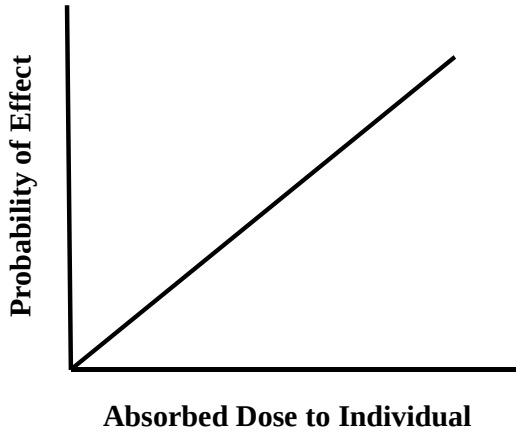

Figure 2.3: Stochastic Effects

The increase in probability for the occurrence of stochastic effects is dependent on the individuals exposed and it is extremely difficult to determine due to many factors including age, gender and radiosensitivity of the individual (ICRP, 2007). By using large groups of the population exposed to radiation incidents, reasonable statistical data is available, mostly after the Second World War with the detonation of nuclear bombs in Japan, by monitoring the amount of survivors who developed cancers in their lifetimes. This is then compared to a control group of the same amount of people that were not exposed to radiation. Statistical data is being researched in great depth after the Chernobyl accident with the conclusive evidence of the increase in thyroid cancers, especially when the people exposed were children and teenagers at the time of the exposure. It took varying times, however, for the cancers to form as the latency period varies again for different individuals.

2.3 Dose Terminology

It is of great importance to understand the units used together with the terminology used for the different exposure types for the research that is to be conducted. The types of exposures that are widely used in terms of biological damage are the equivalent dose / committed equivalent and effective dose which are discussed in 2.3.2 till 2.3.4. To appreciate the differences in dose types it is necessary to firstly explain the meaning of absorbed dose.

2.3.1 Absorbed Dose

Absorbed dose (IAEA, 2007; Herman Cember, 2009) is approximately equal to the mean concentration of absorbed energy in an irradiated organ or tissue. Absorbed dose can be defined as the mean energy (ΔE) imparted to matter of mass (Δm) by ionising radiation of any type.

$$\text{radiation absorbed dose} = \frac{\Delta \text{E}}{\Delta \text{m}} \qquad (2.1)$$

The basic unit of radiation is measured and expressed in terms of absorbed energy per unit mass of tissue. The unit of absorbed dose is the Gray (Gy) and is equal to an absorbed dose of one joule per kilogram (1 J/kg).

2.3.2 Equivalent Dose (H_T)

Equivalent dose (IAEA, 2007; Herman Cember, 2009) can be defined as the dose in an organ, T which is calculated in Eqn 2.2, by:

$$H_{\text{T}} = \sum_{\text{R}} w_{\text{R}} D_{\text{T, R}} \qquad (2.2)$$

where $D_{\text{T,R}}$ is the mean absorbed dose from a radiation type R averaged over a tissue or organ T and w_{R} is the radiation weighting factor for radiation type R, see Table 2.1 (ICRP, 2007). The unit of equivalent dose is the Sievert (Sv), which is equal to one joule per kilogram (1 J/kg).

Equivalent dose is basically a measure of the dose imparted to a tissue or organ, taking into consideration the biological effects due to the different types of radiation.

Table 2.1: Recommended Radiation Weighting Factors
(Taken from Ref. [14])

Radiation Type	**Radiation Weighting Factor (w_R)**
Photons	1
Electrons and muons	1
Protons and charged pions	2
Alpha particles, fission fragments, heavy ions	20
Neutrons	2-20 (Energy dependant)

2.3.3 Effective dose (E)

Effective dose (IAEA, 2007; ICRP, 2007) takes into consideration the overall harm (radiation detriment) to the body as a result of the dose imparted to individual organs and tissues. This provides a single dose measurement tool for radiation protection which also takes into consideration the radiosensitivity of the respective organs to radiation damage.

Effective dose is an extension of Section 2.3.2, where H_T is the equivalent in tissue T and w_T is the tissue weighting factor for tissue T, see Table 2.2. From the definition of equivalent dose, see Eqn 2.3, it follows that:

$$E = \sum_T w_T \sum_R w_R D_{T,R} \quad or \tag{2.3}$$

$$E = \sum_T w_T H_T \tag{2.4}$$

Summation is performed over the tissue equivalent doses, each multiplied by the appropriate tissue weighting factor w_T and the other symbols have the same meaning as in Eqn 2.2. The unit of effective dose is the Sievert (Sv).

Table 2.2: Recommended Tissue Weighting Factors
(Taken from Ref. [14])

Tissue	w_T	$\sum w_T$
Bone-marrow (red), colon, stomach, breast, remainder tissues*	0.12	0.72
Gonads	0.08	0.08
Bladder, oesophagus, liver, thyroid	0.04	0.16
Bone surface, brain, salivary glands, skin	0.01	0.04
	Total	1.00

* Remainder tissues: Adrenals, extra thoracic (ET) region, gall bladder, heart, kidneys, lymphatic nodes, muscle, oral mucosa, pancreas, prostrate small intestine, spleen, thymus and uterus/cervix.

2.3.4 Committed Equivalent Dose $H_T(\tau)$

Committed equivalent dose $H_T(\tau)$(IAEA, 2007) can be defined as:

$$H_T(\tau) = \int_{t_o}^{t_o+\tau} H_T(t)\mathrm{d}t \qquad (2.5)$$

Where t_0 is the time of intake, $H_T(t)$ is the equivalent dose rate at time t in organ or tissue T and τ is the time elapsed after an intake of radioactive substances. When τ is not specified, it will be taken to be 50 years for adults and 70 years for children. The unit of Committed Equivalent Dose is the Sievert (Sv).

2.3.5 Committed Effective Dose E(τ)

Committed Effective Dose $E(\tau)$ (IAEA, 2007) is defined as:

$$E(\tau) = \sum_{\mathrm{T}} w_{\mathrm{T}} H_T(\tau) \qquad (2.6)$$

Where $H_T(\tau)$ is the committed equivalent dose to a tissue T over the integration time τ and w_{T} is the tissue weighting factor for tissue T as listed in Table 2.2. When τ is not specified, it will be taken to be 50 years for adults and 70 years for children. The unit of Committed Effective Dose is the Sievert (Sv).

2.4 Basis for Types of Dose Limits

Dose limits are values of either effective dose or the equivalent dose to individuals from controlled practices that shall not be exceeded (ICRP, 2007). Limits have been derived and recommended based on different methodology and assumptions.

It is essential to understand the various terminologies that are utilised with respect to the derivation of limits that can be applied to assist in public protection by the introduction of protective actions (IAEA, 2007).

2.4.1 Projected Dose

Projected dose (IAEA, 2007) is the magnitude of calculated dose that an unsheltered person could be exposed to during the time that an individual is submersed within a radioactive plume, see Figure 2.4.

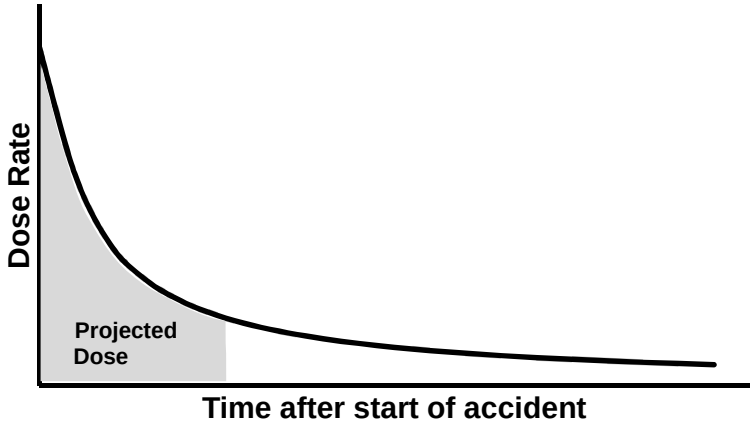

Figure 2.4: Projected Dose Determined after Start of an Accident

Various methods are utilised to project public dose, for example, by using information from 'in plant' monitors or readings in the field together with the dose projection software or by modelling using complex computer codes such as PC COSYMA (Jones J.A., 1996). Projected dose is currently utilised at Koeberg for early phase protective action decision.

2.4.2 Averted Dose

The use of the term Averted dose basically means the magnitude of the dose that is saved by implementing a protective action. An example of this is demonstrated in Figure 2.5, where if for example, people remained indoors with doors and windows closed between t_1 and t_2, during a release, a certain amount of dose is saved or averted as shown on the graph as Avertable dose. In this example it is also noted that sheltering in normal dwellings cannot avert the entire dose.

The averted dose principle was derived as part of justifying the implementation of a particular protective action. The philosophy of avoiding various levels of exposure does not always justify the risk of implementing the protective action. The intent is to justify the risks of implementing protective actions, such as being involved in an accidents and the psychological impact of being evacuated, for

example, which may result in more harm than the risks associated with a radiation exposure.

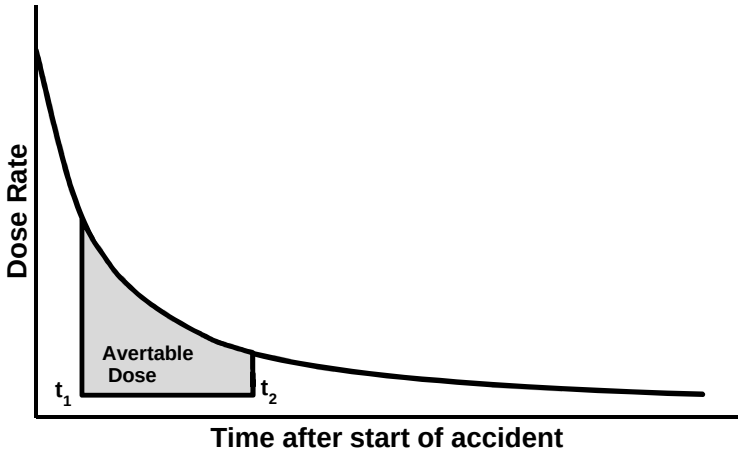

Figure 2.5: Avertable dose

The recommended Generic Intervention Levels are specified in the IAEA requirements document GS-R-2 (IAEA, 2002) and listed below in Table 2.3. These criteria are currently in use, however, averted dose is difficult to predict accurately due to factors such as evacuation times, release times, and building sheltering factors.

Table 2.3: Generic Intervention Levels
(Taken from Ref. [11])

Protective action	Generic intervention level
Sheltering	10 mSv
Evacuation	50 mSv
Iodine Prophylaxis	100 mGy

The guidelines for intervention levels and action levels in emergency exposure situations are based on the averted dose and on the principle that any intervention ordered must result in more good than harm.

2.4.3 Reference Levels

Reference levels (IAEA, 2007; ICRP, 2007) are used for optimisation of protection and safety in emergency exposure situations and in existing exposure

situations, typically as a result of a release of radioactive material. They are established or approved by the regulatory body or possibly even government, depending on the country's ability to handle restrictions or relocation protective actions.

For occupational exposure (not part of this report) and public exposure in emergency exposure situations and in existing exposure situations, a reference level serves as a boundary condition in defining the range of options for the purposes of optimisation in implementing protective actions. This allows the decision to be made on protective actions that will protect the public.

The reference level is a level of dose, which is judged to be inappropriate to occur at planned exposure situations, and below which the optimisation of protection and safety is implemented. The value of the reference level will depend upon the prevailing circumstances for the exposures under consideration. The optimised protection strategies are intended to keep doses below the reference level. When an emergency exposure situation has arisen or an existing exposure situation has been identified, actual exposures could be above or below the reference level. The reference level would be used as a benchmark for judging whether further protective measures are necessary and, if so, in prioritising their application. Optimisation is to be applied in emergency exposure situations.

In respect to Koeberg, at the time of completing this report, a position paper released by the National Nuclear Regulator (National Nuclear Regulator, 2012), for preparing an Emergency Planning Technical Basis for New Nuclear Installations was finalised. The recommended Generic Reference level (total effective dose a result of an accident) for the public, for new installations is 50 mSv. It is unsure at this stage as to the impact that it may have on Koeberg's emergency plan.

2.4.4 Residual Dose

Residual dose (IAEA, 2007; ICRP, 2007) is the dose that is expected to be incurred in the future after protective actions have been terminated or a decision has been taken not to implement protective actions. This project will focus on the early phase in providing operational intervention levels for the early phase implementation of protective actions. The resultant methods of calculating such criteria may be used in the calculation of the residual dose, post-accident, if deemed necessary.

2.5 Dose Limits

2.5.1 Koeberg Operational Procedural Dose Limits

The current Koeberg specific dose limits (Eskom, 2012) were initially derived from ICRP-63 (ICRP, 1991) where a generic projected dose limit to the whole body of 500 mSv within 24 hours or the averted average individual effective dose for the duration of the evacuation is 500 mSv or 5 Sv skin dose. This publication further recommended that for the most foreseeable accident situations, an optimised level of averted dose for evacuation will be lower, but not more than a factor of 10. ICRP-63 also states that on a generic basis that sheltering will almost always be justified, provided that an averted effective dose of 50 mSv can be achieved during the time considered feasible for sheltering. Once again, optimised levels will be lower but not by more than a factor of 10. In summary the limits as specified in Table 2.4 are based on these provisions. Due to reasons not to be discussed in this report Koeberg will always recommend protective actions at the lower intervention levels, for public protection.

Table 2.4: Emergency Intervention Levels for Urgent Protective Actions (Taken from Ref. [5])

Protective Actions	Whole Body (Effective) Dose [1, 4] (mSv)	Lung, Thyroid or any Single Organ Dose [1, 2] (mSv)
Sheltering & Stable Iodine Administration [3]		
Upper Dose Level	50	500
Lower Dose Level	5	50
Evacuation		
Upper Dose Level	500	5000
Lower Dose Level	50	50

NOTE [1] All dose values are based on projected unsheltered dose.

NOTE [2] For practical reasons, one thyroid intervention level is recommended for all age groups.

NOTE [3] Currently no plan exists to distribute potassium iodate tablets together with sheltering. If thyroid dose is less than 50 mSv, no potassium iodate issuing is required.

NOTE [4] When using gamma dose rates to calculate projected dose, it is assumed that effective dose rate is a factor of ten times that of the external gamma dose rate. This is in accordance with IAEA recommendations to take into account dose from inhalation.

Koeberg Iodine concentrations are currently being measured in the field to estimate thyroid equivalent dose. IAEA-TECDOC-955 (IAEA, 1997) assumes that in the absence of being able to calculate lung dose, that the dose to the lung will be a factor of 10 times higher than the whole body dose. This is the reason why Note [4] above was included in the limits for intervention.

These assumptions are generic and were the main driver for the selection of this project to ascertain the relevance of the assumptions in the Koeberg context. It is

also important to note that all of the above limits are calculated using unsheltered projected dose. The outcome of this project will hopefully be utilised to adjust the operational limits in Table 2.4 with Koeberg specific data.

2.5.2 Legal Dose Limits

The public legal dose limits specified in the IAEA document, GS-R-2 (IAEA, 2002) are currently being reviewed and finalised, however the values that will be recommended in the new requirements document already appear in documentation such as the IAEA GSG-2. This change will influence Eskom and the regulator has issued a National Nuclear Regulator Position Paper (National Nuclear Regulator, 2012) reflecting these updated limits. The current General Intervention Levels are prescribed in GS-R-2 and listed in Table 2.3 in terms of averted dose.

Sheltering of the public indoors is not recommended to exceed 2 days, however authorities may choose to advise sheltering at lower intervention levels or for shorter durations so as to facilitate further countermeasures such as evacuation.

The generic optimised intervention value for evacuation is 50 mSv of avertable dose in a period of no more than 1 week. Authorities may wish to initiate evacuation at lower intervention levels for shorter periods, and also where evacuation can be carried out quickly and easily, e.g. for small groups of people. Higher intervention levels may be appropriate in situations where evacuation would be difficult, e.g. for large population groups or if there is inadequate transport.

In some countries a value of 100 mSv of avertable dose is considered to be a more realistic level for temporary evacuation. The ICRP (ICRP, 1991) has recommended that evacuation would almost always be justified for an avertable dose of 500 mSv (or equivalent dose to the skin of 5000 mSv), and that the range of optimised values would be lower than this by no more than a factor of ten.

Koeberg currently has an agreement with the National Nuclear Regulator that after an event has occurred, the averted dose will be calculated depending on the protective actions implemented, based on projected dose and then checked against the averted dose limits.

2.5.3 Dose Limits Based on Generic Criteria

Generic criteria are required to prevent severe deterministic effects (IAEA, 2007) and have been established by the IAEA at levels of dose that are approaching the thresholds, see Table 2.5 (IAEA, 2011), for severe deterministic effects as well as levels to ensure acceptable risk of stochastic effects, see Table 2.6 (IAEA, 2011).

Precautionary protective actions should be implemented before the event if possible, especially in the event that there is a possibility of severe deterministic effects. In the event of a situation whereby deterministic effects are not the limiting factor, stochastic risk must be adequately managed and the justified protective actions must be optimised to ensure that the risk remains as low as reasonably achievable. The IAEA has also included recommendations in the event of an individual having been exposed to radiation in excess of these values. This is however outside the scope of this project as the intent of emergency preparedness is to have intervention levels to ensure that these limits are not exceeded.

Figure 2.6 shows how Emergency Action Levels, Operational Intervention Levels and observable conditions are all tools that can be used to initiate protective actions, however the Generic Reference level remains the overriding factor to ensure that the prescribed dose level to the public is not exceeded.

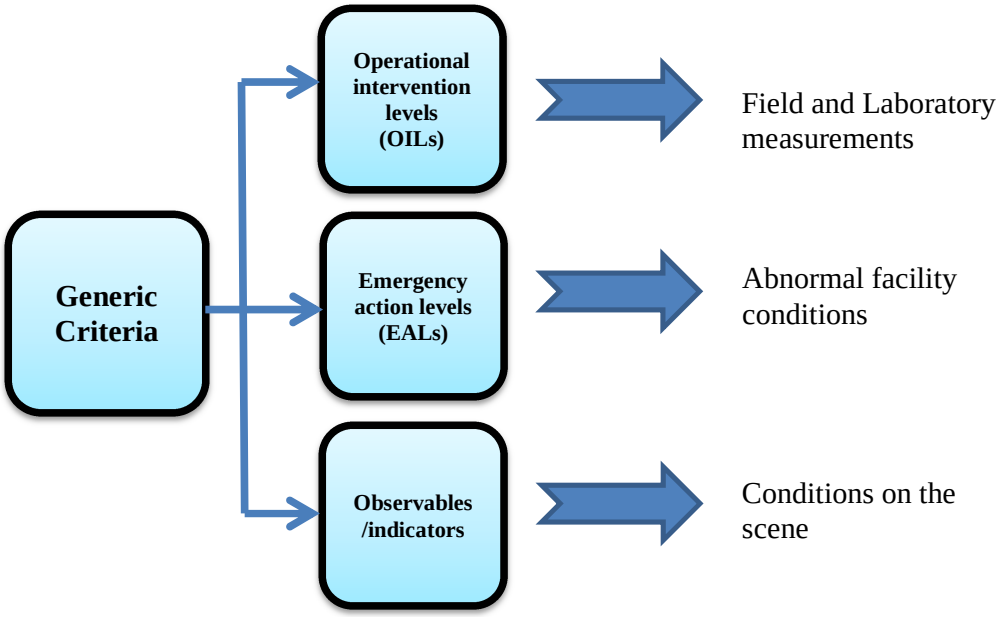

Figure 2.6: System of Generic Criteria and Operational Criteria (Taken from Ref. [7])

Table 2.5: Generic criteria for acute doses for which protective actions and other response actions are expected to be taken under and circumstances to avoid or minimise severe deterministic effects (Taken from Ref. [7])

External acute exposure (<10 hours)		**Examples of protective actions and other response actions**
Organ	**Generic Criteria (Dose)**	
$AD_{\text{Red marrow}}$[a]	1 Gy	**If the dose is projected:** **Take precautionary urgent protective actions immediately** **Provide Public information and warnings** **Carry out urgent decontamination** **If the dose has been received:** **Perform immediate medical examination, consultation and indicated medical treatment** **Carry out contamination control** **Carry out immediate decorporation[f] if applicable** **Carry out registration for follow up health monitoring** **Provide comprehensive psychological counselling**
AD_{Foetus}	0,1 Gy	
AD_{Tissue}[b]	25 Gy at 0.5cm	
AD_{Skin}[c]	10 Gy to 100cm^2	
Internal exposure from acute intake (Δ = 30 days)[d]		
Organ	**Generic Criteria (Dose)**	
$AD(\Delta)_{\text{Red marrow}}$	0,2 Gy for Z $\geq$ 90[e] 2 Gy for Z $\leq$ 89[e]	
$AD(\Delta)_{\text{Thyroid}}$	2 Gy	
$AD(\Delta)_{\text{Lung}}$[g]	30 Gy	
$AD(\Delta)_{\text{Colon}}$	20 Gy	
$AD(\Delta`)_{\text{Foetus}}$[h]	0,1 Gy	

[a] $AD(\Delta)_{\text{Red marrow}}$ represents the average weighted absorbed dose to internal tissues or organs(e.g. red bone marrow, lung, small intestine, gonads, thyroid) and to the lens of the eye from exposure in a uniform field of strongly penetrating radiation.

[b] Dose delivered to 100 cm^2 at a depth of 0,5 cm under the body surface in tissue due to close contact with a radioactive source (e.g. source carried in the hand or pocket.

[c] The dose is to the 100 cm^2 dermis (skin structure at a depth of 40 mg/cm^2 below the body surface.

[d] $AD(\Delta)$ is the RBE weighted absorbed dose delivered over the period of time Δ by the intake that will result in a severe deterministic effect in 5% of exposed individuals.

[e] Different criteria are used to take account of the significant difference in the radionuclide specific intake threshold values for the nuclides in these groups.

[f] The generic criterion for decorporation is based on the projected dose without decorporation. Decorporation is the biological processes, facilitated by a chemical or biological agent, by which incorporated radionuclides are removed from the human body.

[g] For the purposes of these generic criteria, 'lung' means the alveolar-interstitial region of the respiratory tract.

[h] For this particular case **Δ`** means the period of in utero development.

Table 2.6: Generic criteria for protective actions and other response actions in emergency exposure situations to reduce the risk of stochastic effects (Taken from Ref. [7])

Projected Dose that exceeds the following generic criteria: Take urgent protective actions and other response actions		
Organ	**Generic Criteria (Dose)**	**Examples of protective actions and other response actions**
H_{Thyroid}	50 mSv in the first 7 days	Iodine thyroid blocking
E	100 mSv in the first 7 days	Sheltering; evacuation; decontamination; restriction of consumption of food, milk and water; contamination control; public reassurance
H_{Foetus}	100 mSv in the first 7 days	

Projected Dose that exceeds the following generic criteria: Take protective actions and other response actions early in the response		
Organ	**Generic Criteria (Dose)**	**Examples of protective actions and other response actions**
E	100 mSv per annum	Temporary; decontamination; replacement of food, milk and water; public reassurance
H_{Foetus}	100 mSv for the full period of in utero development	

Dose that has been received and exceeds the following generic criteria: Take longer term medical actions to detect and to effectively treat radiation induced health effects		
Organ	**Generic Criteria (Dose)**	**Examples of protective actions and other response actions**
E	100 mSv in a month	Screening based on equivalent doses to specific radiosensitive organs (as a basis for medical follow-up); counselling
H_{Foetus}	100 mSv for the full period of in utero development	Counselling to allow informed decisions to be made in individual circumstances

Note: H_{τ} – equivalent dose in an organ or tissue T; E – effective dose.

2.6 Operational Intervention Levels

Operational intervention levels (IAEA, 2007) are calculated levels, which will either be measured by instruments in the field or determined by laboratory analysis, which corresponds to a generic intervention level, see Table 2.4. They are typically expressed in terms dose rates or activity and are derived to ensure both the limiting of public dose, as well as preventing the unnecessary implementation of protective actions, which could result in more harm than good.

An operational intervention level is used immediately and directly (without further assessment) to determine the appropriate protective actions on the basis of an environmental measurement.

Operational Intervention Levels are calculated based on the assumption that the most sensitive groups, namely, infants and pregnant women in terms of foetus protection and they assume normal life in the contaminated area for seven days. Exposure due to deposition of ground contamination and inhalation of re-suspended radioactive material also need to be considered. Ingestion of radioactive material is usually not considered critical in terms of dose, as there will be usually an automatic banning of all home grown foods or crops grown in the affected area.

2.7 Emergency Action Levels

Emergency Action Levels (IAEA, 2007; ICRP, 1991) are indications of abnormal plant conditions which would indicate an emergency situation that would result in an emergency being classified as either an Alert, Site or General Emergency (IAEA, 2002; Eskom, 2011; Eskom, 2012). An Emergency Action Level could represent an instrument reading or the status of a piece of equipment, such as a safety system that is not operational. The Emergency Action Levels are intended to be used as a tool for decision makers to ensure the timely implementation of urgent protective actions, prior to any release of radioactive material from the

utility. They depict an abnormal state of the plant which could result in the release of radioactive material into the environment.

Observable indicators such as fires, flooding of key areas of the plant and explosions are not strictly action levels, but typically have the same function.

2.8 Urgent Protective Actions

Urgent Protective Actions (IAEA, 2007) are interventions that are intended to avoid or reduce public dose due to the release of radioactive material into the environment. Urgent protective actions are actions that need to be taken promptly, normally within hours in order to be effective, and the effectiveness of which will be significantly reduced if delayed. Examples of urgent protective actions are discussed in Section 2.8.1 to Section 2.8.3.

2.8.1 Evacuation

Evacuation (IAEA, 2007) can be defined as the rapid removal of people from an area to avoid or reduce short term radiation exposure in an emergency. If permanent re-settlement of the public is required, the term relocation is typically used. Evacuation is a precautionary protective action where automatic evacuation is recommended at the declaration of a general emergency (Eskom, 2011) where the potential exists for the release of radioactive material to the environment.

2.8.2 Sheltering

Sheltering (IAEA, 2007) is when a structure is used for protection against exposure to a plume or the deposition of radioactive material. It is an urgent protective action used to provide shielding against external exposure and to reduce the inhalation of airborne contamination.

2.8.3 Iodine Prophylaxis

Iodine prophylaxis (IAEA, 2007) is also an urgent protection, which requires the administration of a compound of stable iodine (potassium iodate or potassium iodide) to prevent or reduce the uptake of radioactive iodine by the thyroid gland in an event of an accident involving radioactive iodine.

2.9 Longer Term Protective Actions

Examples of longer term protective actions (IAEA, 2007) are relocation, decontamination of infrastructure, food banning, follow up health monitoring and psychological counselling. Operational Intervention Levels can also be established for longer term protective actions by measuring actual deposition samples after the radioactive plume has passed.

2.10 Field Measurements

Radiation Protection personnel are expected to perform measurements of plume samples. One of the main objectives of this project is also to reduce the amount of time that is spent by emergency workers in the plume. Currently the measurement of iodine is done by taking air samples for analysis. If iodine exposure can be assessed from gamma measurement, then the time spent in the plume would be drastically reduced, which would then result in significant dose reduction.

Gas filled ionisation detectors are utilised for measurements which rely on ionisation of the gas inside the detector to produce an output current, which is measured by an ammeter. As the intent of this project is focussed on the measurement of gamma radiation, it is appropriate to give an overview of how ionisation occurs when gamma radiation enters a detector.

2.10.1 Photoelectric Effects

When a photon of relatively low energy encounters an orbital electron near the nucleus of an atom, the photon is completely absorbed and transfers all of its energy to the electron and ceases to exist. The electron is ejected from the atom. The ejected electron, called a photo-electron then loses its energy by normal charged particle interactions, see Figure 2.7. When an electron in close orbit is ejected, another electron from higher orbit will fill the vacated space and a low-energy characteristic X-ray will be emitted.

Photo-electric absorption (Knoll, 2010) is the most common process by which low energy photons lose their energy. The energy of the ejected photo-electron can be calculated as per Eqn 2.7

$$E_e = hv - E_b \qquad (2.7)$$

Where:

E_e = electron energy

hv = photon energy

E_b = binding energy of the electron to its orbital shell

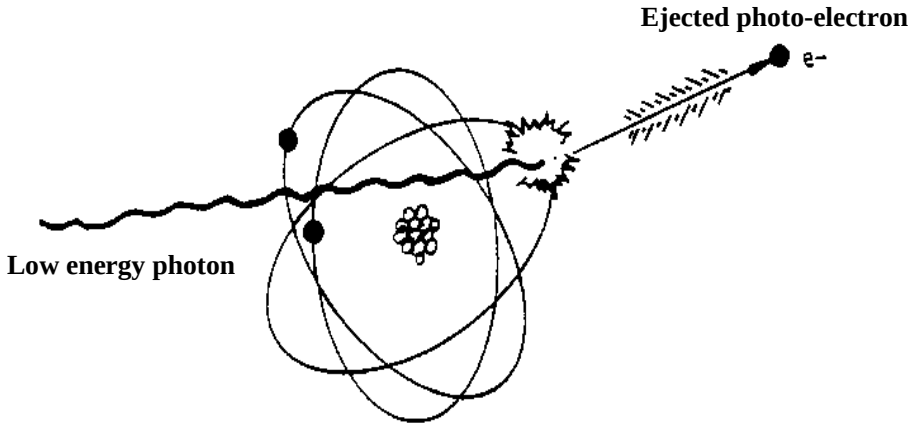

Figure 2.7: Photoelectric Effects

2.10.2 Compton Scattering

Compton scattering (Knoll, 2010) occurs when an incident photon interacts with an orbital electron. It is the predominant interaction mechanism due to the energies of typical radioactive sources. During Compton scattering only a portion of the photon energy is transferred to the electron. A photon of reduced energy is scattered off and the electron ejected, see Figure 2.8. The probability of Compton scattering is roughly proportional to the density of the absorbing material, and usually occurs for electrons in outer orbits around large nuclei.

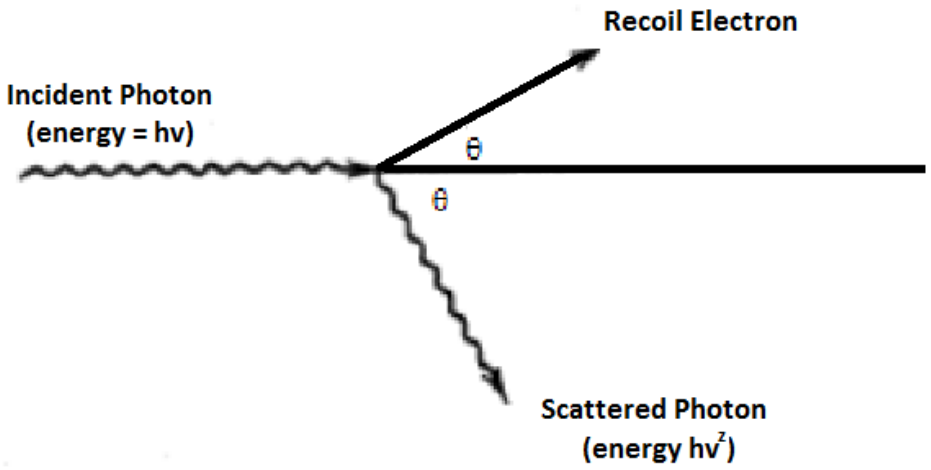

Figure 2.8: Compton Scattering

For a given scatter angle θ the following formula applies:

$$hv^z = \frac{hv}{1 + \frac{hv}{mc^2}\,(1 - cos\,\theta)} \tag{2.8}$$

Where mc^2 is the rest mass energy of the electron (0.511 MeV). For small scattering angle θ, very little energy is transferred.

2.10.3 Pair Production

If a high energy photon interacts with the strong coulomb forces near the nucleus of an atom, the photon will disappear and a positron/electron pair will be created, see Figure 2.9. The energy of an electron or positron can be shown to equal 0.511 MeV (using $E = mc^2$). Since two masses are created, pair production (Herman Cember, 2009) cannot occur if the energy of the photon is less than 1.02 MeV. For this reason, pair production is probable only for high-energy photons.

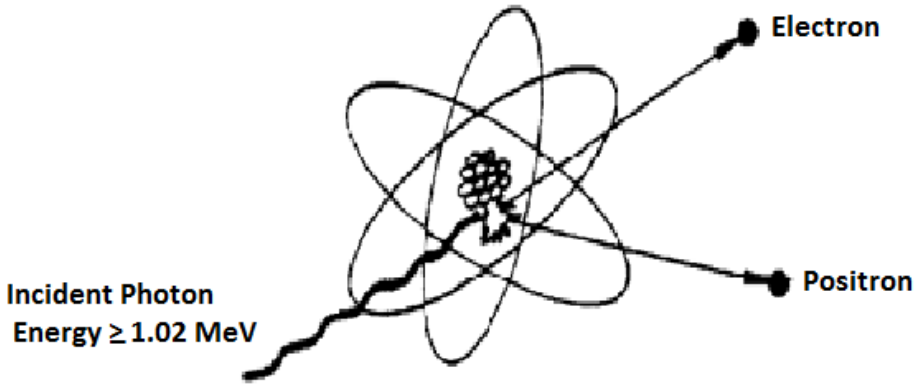

Figure 2.9: Pair Production

The electron loses energy as it travels and will slow down and be captured by an atom. The positron will slow down and absorb a free electron. In this process, called annihilation, see Figure 2.10, the positron and free electron cease to exist and create two photons of 0.511 MeV each, which could once again lose their energy by compton scattering or photoelectric effect.

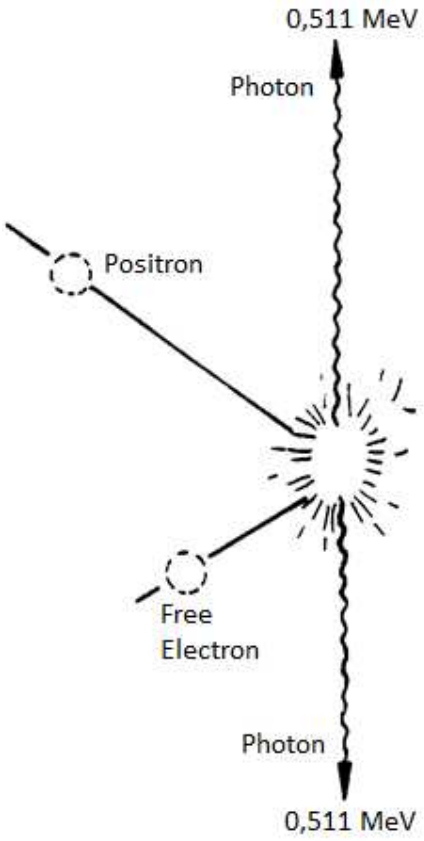

Figure 2.10: Annihilation

CHAPTER THREE – PC COSYMA AND INPUT DATA

3.0 Introduction

This chapter describes the relevant sections and the input parameters of the software modelling code, PC COSYMA (Jones J.A., 1996).

3.1 Atmospheric Dispersion and Deposition

Radionuclides are released to the atmosphere as either a fine aerosol or as a vapour that will create a plume. As the plume is carried downwind, turbulence in the atmosphere will mix 'clean' air into it, thereby diluting and spreading it in both the crosswind and vertical directions. Both processes, i.e., transport and diffusion are often summarised in the word 'dispersion'. The principal question is, what will the concentration profiles (crosswind and vertical) of the dispersing plume look like. As the plume passes, radioactive material settles on the ground and is known as deposition which would also be a source of exposure to an individual.

Many simple theoretical formulations of dispersion predict that these concentration profiles will have a Gaussian shape. Although the assumptions of the simple theories do not hold for the real atmosphere, the Gaussian shape has been found empirically to be approximately valid in many situations and it forms the basis of the Gaussian plume model, which is widely used in consequence assessment. For each individual plume it is usual practice to assume that it continues to travel in the direction of the wind prevailing at the time of the release, while at the same time taking into account the temporal changes in other wind characteristics, such as wind speed, stability category and the occurrence of rain.

3.2 The Gaussian Plume Model

One of the most commonly used mathematical models for determining ground level concentration, is the Gaussian plume, straight line trajectory model, see Fig 3.1.

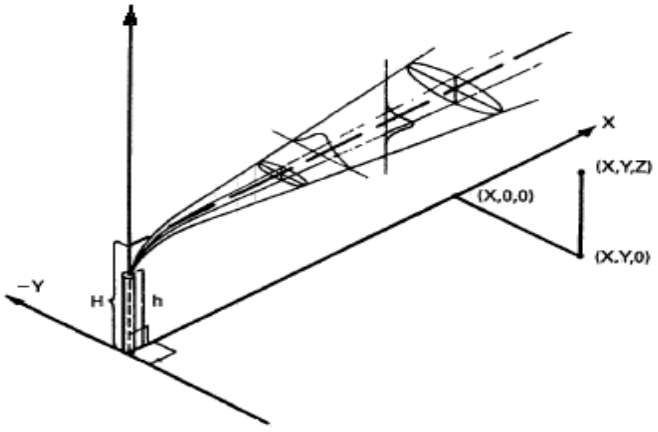

Figure 3.1: Gaussian plume dispersion model for a continuous point source

This model assumes that the contaminant is normally distributed around the central axis of the plume in both vertical and horizontal directions. It is also assumed that atmospheric stability and wind speed determine the atmospheric dispersion characteristics of the contaminants in the downwind direction. This model is described by the Pasquill-Gillford equation; see Eqn 3.1 which is used by the software PC COSYMA (Jones J.A., 1996).

$$\mathrm{X}(x,y) = \frac{Q}{\pi\sigma_y\sigma_z\mu} e^{\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2}+\frac{H^2}{\sigma_z^2}\right)\right]} \qquad (3.1)$$

Where

X(x,y) = ground level concentration in Bq per cubic meter at point (x,y),
x = downwind distance on plume centre line in metres
y = crosswind distance in metres
Q = emission rate in Bq/s
$\sigma_y\sigma_z$ = horizontal and vertical standard deviations of contaminant concentration in the plume in metres
μ = mean wind speed at level of plume centre line in m/s, and
H = Effective plume height in meters

Due to the fact that the temperature of the released gas has a much higher temperature than the surrounding air, the plume will rise up higher than the release point. The height where the plume levels out, is known as the effective

plume height, and is a function of the volume of air released and the temperature reduction rate of the plume with increased height.

The atmospheric dispersion plays a very important role in the transport of the radioactive material through the environment once it is released from the core. The atmospheric dispersion calculations provide estimates of air concentrations of radionuclides following discharges into the atmosphere. These air concentrations of various nuclides are then used to calculate the doses to man. PC COSYMA uses the Gaussian plume model to describe the movement of radionuclides through the atmosphere. The model that is used in this analysis is a segmented Gaussian plume model, which allows for hourly changes in the wind speed and direction, stability category and rainfall rate affecting the dispersing material. The model assumes that the conditions affecting the plume are determined from those at a single meteorological station representing conditions in the region of interest. The model allows for material released over a long period by describing the release as a series of hourly phases that are treated separately. PC COSYMA allows the release to be divided into six phases that need not be in sequential hours. The Gaussian plume model applies to the dispersion of material over short distances from a nuclear site. Typically, the Gaussian model is applied to distances less than 100 km. This model is accepted by the NNR and provides sufficient modelling capabilities for emergency preparedness at Koeberg.

3.3 Meteorological Data and its Sampling

Most consequence-modelling codes simulate the atmospheric dispersion of the released radioactive material by using the Gaussian dispersion model to calculate ground-level instantaneous and time-integrated airborne concentrations and deposited levels of radioactivity. This is done as a function of time and distance from the point of release. The Gaussian model is generally used in such a way as to allow changes in atmospheric stability, wind speed, wind direction and precipitation for each successive hour of travel time. The hourly weather data required as inputs are usually generated by processing data collected at the reactor

site itself or at a nearby weather station in this case Koeberg Nuclear Power Station.

3.4 Koeberg Source Term

For any type of modelling of radioactive material in the environment, it is important to know the concentration and makeup of the contaminants. The initial inventories of the various radionuclides were calculated for a full core of the AFA-3GA fuel assemblies using the ORIGEN-S computer code (Eskom, 2009). The ORIGEN-S code is part of the SCALE 4.2 package (RSIC, 1993). This code computes time dependent concentrations and source terms of a large number of isotopes, which are generated through different processes that take place inside a nuclear reactor. The initial source term is calculated at an average burn-up after three cycles in the core, assuming that the reactor runs at continuous full power with maintenance off line durations conservatively calculated as 20 days.

The ORIGEN-S radionuclide output contains in excess of 1000 radionuclides that could be released to the environment. It is however not practical to consider all of the released nuclides for two reasons. Firstly many isotopes have a very short half-life and therefore decay before any significant dose is received and secondly, many of the nuclides are not volatile so the fraction of those nuclides released is much lower than that of other nuclides.

The data libraries used in PC COSYMA (Jones J.A., 1996) contain information on 196 nuclides, which are likely to be the most important contributors to the doses, and hence the risks of most accidents at nuclear facilities. The source term that was used for the input data is listed in Table 3.1.

Table 3.1: The Final Inventory for AFA-3GA, 4,4% Enriched Fuel (Taken from Ref. [2])

Nuclide	Total Activity (Bq)	Nuclide	Total Activity (Bq)	Nuclide	Total Activity (Bq)
Ag-108m	8.31×10^{7}	Eu-152	0	Na-22	0
Ag-110	4.51×10^{17}	Eu-152m	0	Na-24	1.97×10^{12}
Ag-110m	1.25×10^{16}	Eu-154	0	Nb-91m	0
Ag-111	1.94×10^{17}	Eu-155	0	Nb-92	2.42×10^{14}
Am-241	3.24×10^{14}	Eu-156	0	Nb-92m	0
Am-242	2.14×10^{17}	Fe-52	0	Nb-93m	4.41×10^{11}
Am-242m	1.91×10^{13}	Fe-55	9.59×10^{14}	Nb-94	1.57×10^{13}
Am-243	6.24×10^{13}	Fe-59	6.47×10^{13}	Nb-94m	3.05×10^{12}
Ar-41	3.01×10^{5}	Fe-60	0	Nb-95	4.80×10^{18}
Ba-139	5.01×10^{18}	Hf-175	0	Nb-95m	3.47E+16
Ba-140	4.83×10^{18}	Hf-181	0	Nb-97	4.87×10^{18}
Be-10	8.75×10^{6}	Hf-182	0	Nd-147	1.83×10^{18}
Be-7	0	Hg-197	0	Ni-59	1.23×10^{12}
Ce-141	4.57×10^{18}	Hg-197m	0	Ni-63	1.94×10^{14}
Ce-143	4.23×10^{18}	Hg-203	0	Ni-65	3.42×10^{14}
Ce-144	3.39×10^{18}	I-125	0	Np-237	9.69×10^{11}
Cm-242	1.19×10^{17}	I-129	9.06×10^{10}	Np-238	1.09×10^{18}
Cm-243	7.15×10^{13}	I-130	1.10×10^{17}	Np-239	5.45×10^{19}
Cm-244	1.09×10^{16}	I-131	2.75×10^{18}	P-32	4.94×10^{12}
Cm-245	1.15×10^{12}	I-132	3.97×10^{18}	P-33	2.02×10^{8}
Cm-246	3.60×10^{11}	I-133	5.60×10^{18}	Pm-147	3.54×10^{17}
Cm-247	1.57×10^{6}	I-134	6.15×10^{18}	Pm-148	7.56×10^{17}
Cm-248	5.70×10^{6}	I-135	5.24×10^{18}	Pm-148m	8.23×10^{16}
Co-56	0	Kr-83m	3.40×10^{17}	Pm-149	0
Co-57	0	Kr-85	2.85×10^{16}	Pm-151	0
Co-58	2.21×10^{15}	Kr-85m	7.18×10^{17}	Po-208	0
Co-58m	0	Kr-87	1.38×10^{18}	Po-210	3.60×10^{3}
Co-60	6.10×10^{14}	Kr-88	1.94×10^{18}	Pr-143	4.13×10^{18}
Co-60m	1.28×10^{15}	La-140	5.02×10^{18}	Pr-145	2.88×10^{18}
Co-61	1.50×10^{13}	La-141	4.56×10^{18}	Pu-236	2.41×10^{12}
Cr-51	6.27×10^{15}	La-142	4.41×10^{18}	Pu-238	8.30×10^{15}
Cs-134	5.16×10^{17}	Mn-52	0	Pu-239	8.66×10^{14}
Cs-134m	1.53×10^{17}	Mn-52m	0	Pu-240	1.23×10^{15}
Cs-135	1.26×10^{12}	Mn-53	0	Pu-241	3.37×10^{17}
Cs-136	1.55×10^{17}	Mn-54	1.68×10^{14}	Pu-242	5.04×10^{12}
Cs-137	3.10×10^{17}	Mn-56	3.09×10^{15}	Ra-226	1.65×10^{2}
Cs-138	5.12×10^{18}	Mo-101	4.68×10^{18}	Rb-86	6.05×10^{15}
Cu-64	4.53×10^{14}	Mo-93	1.26×10^{10}	Rb-88	1.97×10^{18}
Cu-66	9.97×10^{13}	Mo-99	5.02×10^{18}	Rb-89	2.52×10^{18}

Nuclide	Total Activity (Bq)	Nuclide	Total Activity (Bq)
Re-184	0	Tc-99m	4.40×10^{18}
Re-184m	0	Te-125m	9.00×10^{15}
Re-186	1.09×10^{12}	Te-127	2.92×10^{17}
Re-186m	0	Te-127m	3.80×10^{16}
Re-188	8.17×10^{8}	Te-129	8.71×10^{17}
Re-188m	7.78×10^{8}	Te-129m	1.29×10^{17}
Rh-103m	3.79×10^{18}	Te-131	2.44×10^{18}
Rh-105	2.68×10^{18}	Te-131m	3.98×10^{17}
Ru-103	4.20×10^{18}	Te-132	3.91×10^{18}
Ru-105	2.91×10^{18}	Te-133	3.29×10^{18}
Ru-106	1.44×10^{18}	Te-133m	2.04×10^{18}
Sb-124	4.22×10^{15}	Te-134	4.64×10^{18}
Sb-125	4.27×10^{16}	U-234	3.71×10^{10}
Sb-126	3.26×10^{15}	U-235	8.42×10^{10}
Sb-127	2.96×10^{17}	U-238	8.28×10^{11}
Sb-128m	4.88×10^{17}	V-48	0
Sb-129	8.84×10^{17}	V-49	0
Sb-130m	1.29×10^{18}	W-183	0
Sb-131	2.30×10^{18}	W-183m	5.86×10^{11}
Sc-44	0	W-185	2.73×10^{12}
Sc-44m	0	W-187	8.85×10^{9}
Sc-46	2.47×10^{12}	Xe-131m	2.96×10^{16}
Sc-47	4.03×10^{12}	Xe-133	5.56×10^{18}
Sc-48	3.74×10^{11}	Xe-133m	1.75×10^{17}
Se-75	0	Xe-135	1.33×10^{18}
Si-32	1.06×10^{3}	Xe-135m	1.10×10^{18}
Sr-89	2.62×10^{18}	Xe-138	4.62×10^{18}
Sr-90	2.24×10^{17}	Y-90	2.44×10^{17}
Sr-91	3.27×10^{18}	Y-90m	4.63×10^{13}
Sr-92	3.54×10^{18}	Y-91	3.37×10^{18}
Sr-93	4.02×10^{18}	Y-91m	1.90×10^{18}
Ta-179	0	Y-92	3.56×10^{18}
Ta-182	5.83×10^{14}	Y-93	4.12×10^{18}
Ta-182m	1.05×10^{12}	Zn-65	2.43×10^{11}
Ta-183	1.65×10^{15}	Zn-69m	3.79×10^{5}
Tc-101	4.68×10^{18}	Zr-89	1.62×10^{10}
Tc-95	0	Zr-93	5.91×10^{12}
Tc-95m	0	Zr-95	4.76×10^{18}
Tc-96	0	Zr-97	4.83×10^{18}
Tc-99	3.77×10^{13}		

It is not possible to consider all 196 nuclides in PC COSYMA calculations due to the CPU time and disk storage, which would be required. The calculation programs are limited to considering not more than 60 nuclides in the source term. This is required to be done by the user who must first specify the inventories and release fractions of those nuclides, which are to be considered for inclusion in the source term.

3.5 Reference Accident for the Project

There are generally three types of accidents (Eskom, 2009) associated with the operation of nuclear power plants that may result in people being exposed to radiation. These accident types are:

- Fuel Melt Accidents: These accidents involve the melting of the core and also fuel melt in the spent fuel pool;
- Fuel Handling Accidents: These involve fuel damage as a result of dropping of a fuel assembly during the movement of used fuel, and
- Waste Treatment Accidents: These involve accidents involving radioactive waste treatment.

It is important to realise that due to the design of the nuclear power station, accidents can occur in a variety of manners, from large breaks in piping or components to loss of external and internal power supplies. The latest incident, which occurred at the Fukushima plant in Japan, has added a whole new element to emergency planning, which require in depth evaluation of possible risks due to external events such as earthquakes and tsunamis.

External events have not been included in this project since the release of radioactivity for different accident types are similar, irrespective of the cause of the initial accident. A dedicated Koeberg group models the amount of activity released from the containment under various conditions to produce a report known as the Koeberg Level 3 Probabilistic Safety Analysis (PRA) Study, No. PSA-R-T16-01 (Eskom, 2009). Future modelling will also evaluate the risk associated

with events that are created by external events such as tsunamis, flooding, tornadoes, etc.

As part of the modelling process, the response of the containment to severe accidents has been grouped into eight release categories (RC). These different release categories are designated RC-1 to RC-8 and are defined in Table 3.2.

Table 3.2: Koeberg Accident Release Categories
(Taken from Ref. [2])

Release Category	Description of Release Category
RC-1	Containment integrity is maintained. Containment spray system operable. This results in a slow release of radioactivity into the external environment at the containment design leak rate.
RC-2	Containment not isolated. Containment spray system is operable. This results in a slow filtered release of activity into the external environment.
RC-3	Containment is by-passed or not isolated. The containment remains unisolated, resulting in a continuous release into the external environment. Containment spray is impaired resulting in an unfiltered release.
RC-4	Early Containment failure. This corresponds to the total and early loss of containment and early loss of containment integrity and a direct release of radioactivity into the external environment within a few hours of the start of the accident.
RC-5	Late containment failure. Containment spray system operable. This results in direct release of radioactivity into the external environment and is considered to occur approximately one day into the accident.
RC-6	Late containment failure. Containment spray function impaired. This results in direct release of radioactivity into the external environment and is considered to occur approximately one day into the accident.
RC-7	Basemat melt-through. Containment spray operable. This results in a ground level release of radioactivity into the external environment and is considered to occur days into the accident.
RC-8	Basemat melt through. Containment spray system impaired. This results in a ground level release of radioactivity into the external environment and is considered to occur several days into the accident.

RC-3 type incidents have the highest postulated dose impact on the public, even though it is not the most probable in terms of risk calculated in The Koeberg level

3 PSA studies (Eskom, 2009). For the purpose of this project, using the release data for this accident type is a conservative assumption that will envelope all other accident types. This research project is limited to the assessment of the consequences of accidental releases to the atmosphere.

3.6 Accident Release Fraction

PC COSYMA further groups isotopes into 11 different bins to assign release fractions (Eskom, 2009). The grouping scheme is only used to determine the amount of each relevant isotope released to the environment. PC COSYMA groups together nuclides with similar radiological characteristics in order to reduce calculation time. The source term calculation deals with this by grouping together elements with the same release fractions. The grouping of isotopes used in PC COSYMA is shown in Table 3.3

Table 3.3: Nuclide Grouping Scheme used in PC COSYMA Calculations (Taken from Ref. [2])

Group Number	Isotopes
1	Kr, Xe, Ar
2	I, Br
3	Te, Se, Po
4	Sr, Ra, Be
5	Mo, Fe, Co, Tc, V, Cr, Mn, Nb, Ta, W, Ru, Rh, Re, Ni
6	Cs, Na, Rb, Cu
7	Ba
8	La, Sc, Y, Pr, Nd, Pm, Sm, Eu, Am, Cm
9	Ce, Hf, Zr
10	Sb, Hg, Zn
11	U, Np, Pu

Accidents differ greatly in terms of plant conditions at the time of an incident. There would be a significant difference in release to the environment if no safety systems were operational versus if there was, for example, some core cooling taking place. The release height as well as the temperature of the escaping material will also result in very different public doses. The fractions of the initial

inventory released for a RC-3 accident are shown in Table 3.4. Phases 1-6 are in units of hours from the start of the release.

Table 3.4: Summary of Release Fractions for RC-3
(Taken from Ref. [2])

Group	Phase 1 0-1 h	Phase 2 1-2 h	Phase 3 2-3h	Phase 4 3-4h	Phase 5 4-12h	Phase 6 12-24h
1	$5.60x10^{-01}$	$2.21x10^{-01}$	$1.04x10^{-01}$	$1.65x10^{-02}$	$7.41x10^{-04}$	$2.13x10^{-05}$
2	$1.97x10^{-01}$	$2.97x10^{-02}$	$8.42x10^{-03}$	$6.45x10^{-03}$	$1.75x10^{-02}$	$1.36x10^{-02}$
3	$2.28x10^{-01}$	$1.60x10^{-02}$	$4.30x10^{-03}$	$1.55x10^{-03}$	$4.33x10^{-03}$	$1.03x10^{-03}$
4	$1.13x10^{-02}$	$1.06x10^{-03}$	$4.10x10^{-03}$	$2.32x10^{-04}$	$3.06x10^{-04}$	$5.57x10^{-06}$
5	$2.27x10^{-02}$	$2.74x10^{-03}$	$1.37x10^{-03}$	$1.59x10^{-03}$	$3.40x10^{-04}$	0
6	$1.81x10^{-01}$	$4.72x10^{-03}$	$1.52x10^{-03}$	$1.07x10^{-03}$	$7.83x10^{-03}$	$4.33x10^{-03}$
7	$2.12x10^{-02}$	$1.31x10^{-03}$	$2.08x10^{-03}$	$3.68x10^{-04}$	$5.05x10^{-04}$	$5.03x10^{-06}$
8	$1.41x10^{-04}$	$2.59x10^{-05}$	$5.83x10^{-04}$	$3.26x10^{-05}$	$4.94x10^{-05}$	$8.56x10^{-07}$
9	$9.87x10^{-04}$	$2.07x10^{-04}$	$7.92x10^{-03}$	$4.54x10^{-04}$	$6.66x10^{-04}$	$1.27x10^{-05}$
10	$1.09x10^{-01}$	$1.26x10^{-02}$	$3.02x10^{-02}$	$1.02x10^{-02}$	$1.87x10^{-02}$	$1.59x10^{-03}$
11	0	$7.27x10^{-07}$	$4.55x10^{-05}$	$8.68x10^{-06}$	$9.81x10^{-06}$	$1.75x10^{-07}$

3.7 Structure of PC COSYMA

PC COSYMA is a highly modularised (Jones J.A., 1996) program with each module dealing with a particular aspect of the analysis such as health effects calculation, meteorological conditions, etc. The code consists of three major interfaces, namely, the Input Interface, the Calculation Programs, and the Results Interface, each with one or more modules, which allows the user to interact with the system. In addition to these interfaces the system also includes a series of libraries containing data which are needed in some parts of the calculations. Examples of data contained in the libraries are the various dose rates per unit deposit required by dose models used in PC COSYMA to calculate the doses at various distances due to each exposure route.

3.7.1 General Assumptions

- Irradiation of the exposed population through ingestion of foodstuffs is not considered as it is assumed that food banning of home grown crops is implemented;
- No emergency plan protective actions, other than that of food banning to mitigate the accident have been assumed. This is a conservative approach, and,
- The transient population and population projections are also not considered.

Default values are provided for all the input parameters. The input interface performs a number of checks on the validity of the input data.

3.8 The Input Interface

The input interface (Jones J.A., 1996) is the starting point of PC COSYMA. The input interface uses a series of menus to guide the user through the process of identifying the required endpoints, and setting values for the input parameters needed to calculate the quantities required. The input interface also determines which of the Calculations Programs are required and sets up a system to run those programs without further intervention from the user. Deterministic calculations were selected as well as site specific information such as:

- The source terms i.e., the amount of material released, and the timings of the release and the height of the release;
- Spatial distribution data;
- Dispersion characteristics, namely D for neutral atmospheric conditions and F referring to moderately stable atmospheric conditions (Herman Cember, 2009).
- Deposition characteristics of the various nuclides; and
- Meteorological conditions for a 2 year period at Koeberg Power Station weather station.

3.9 Description of the Radionuclide Release

The calculation begins with a description of the characteristics of the radionuclide release, including the following:

- Total activity of each radionuclide released to the environment;
- The amount of energy associated with the release;
- The duration of the release;
- The time of the release after accident initiation;
- The warning time;
- The height of the release; and
- The frequency of occurrence predicted for the accident.

3.10 Exposure Pathways and Dose Assessment Calculations

Following the accidental releases of radioactivity into the environment, the public may be exposed to the radiation through various pathways. This section briefly describes the methods used in dose assessment calculations in the absence of countermeasures for various types of radiation to various organs within the human body. The methods for calculating doses in the presence of countermeasures will not be described here. Following an accidental release of radioactive material to the atmosphere, people can be irradiated by a number of routes from material both in the atmosphere and after its deposition to the ground. In PC COSYMA calculations, the following routes of exposure are considered:

- External irradiation from radioactive material in the passing plume or cloud, referred to as cloudshine;
- External irradiation from radioactive material deposited on the ground, referred to as groundshine;
- External irradiation from radioactive material deposited on skin and clothing;

- Internal irradiation from radioactive material inhaled directly from the passing plume; referred to as Inhalation;
- Internal irradiation from radioactive material inhaled following resuspension of the ground deposit, referred to as Resuspension; and,
- Internal irradiation from radioactive material ingested following the contamination of foodstuffs by radioactive material deposited from the plume. This pathway is not considered in this study as it is assumed that banning of home grown food would be implemented in the downwind sector.
    - There is, however some ingestion from radioactive material that has settled on individuals lips, fingers, etc., but this is of minor importance.

For estimating deterministic effects, the most important of these pathways are:

- Inhalation from the cloud;
- Cloudshine; and
- Short-term exposure from contaminated ground (hours to days).

For estimating stochastic health effects, the important pathways include:

- External exposure from contaminated ground (short and long term);
- Inhalation exposure from the passing cloud; and
- Subsequent resuspension of radionuclides.

When determining doses, PC COSYMA takes into account all these exposure routes. The doses calculated by PC COSYMA are those for a reference adult. For each pathway, the dosimetric models are used to evaluate not only effective dose but also the equivalent dose in each of a number of organs and tissues that are of significance for evaluating the health impact on the exposed population.

The outputs used for this project were to calculate the doses received by members of the public over various time intervals from 1 day up to 50 years following an accident.

3.11 Radioactive Decay Model

Radioactive decay is the process by which a radionuclide disintegrates to form different species (daughters). Thus, for short-lived radionuclides, the radioactive decay may cause significant reduction in the concentration. The concentrations calculated by PC COSYMA allow for radioactive decay and ingrowth of daughter products during the travel period of the plume. This is accounted for by multiplying the air concentration in the Gaussian Equation by a correction factor.

3.12 External Dose from a Passing Cloud

The dose model (Jones J.A., 1996) for external gamma irradiation assumes a semi-infinite cloud of uniform radionuclide concentration. The gamma dose from material in the plume is calculated by:

$$Dn(x, y, z) = Cn.\, cn(x, y, z) \qquad (3.2)$$

Where Dn(x,y,z) is a radiation dose from the exposure to cloudshine at point (x,y,z); Cn is known as cloud-dose conversion factor; and cn(x,y,z) is the time-integrated airborne concentration of nuclide n at point (x,y,z).

The cloud dose conversion factors for particular radionuclides have been pre-calculated and are located in the data library of PC COSYMA. The dose calculated by this equation is corrected to allow for the plume not being semi-infinite or of uniform concentration. The correction factor takes into account the shape and size of the plume.

3.13 External Dose from Deposited Radionuclides

The external gamma dose from deposited material (Jones J.A., 1996) is calculated by multiplying the amount of material deposited by a quantity, the dose per unit deposition, which is obtained from a data library. A suitable library, giving the dose per unit deposit at a series of times for each of a large number of nuclides,

including the contribution from daughter products formed after deposition, is provided in the code.

3.14 Inhalation Dose from Passing Cloud

The dose from material inhaled from the plume (Jones J.A., 1996; Eskom, 2009) is calculated as the product of the time averaged air concentration, breathing rate (3.3×10^{-4} m^3/s) and dose per unit activity inhaled, which is obtained from PC COSYMA library. The inhalation rate, which can be set by the user, should be an average value for adults, as that is the age group for which doses are calculated in PC COSYMA and which is included in the data library of dose per unit intake.

3.15 Inhalation Dose from Resuspended Radionuclides

The dose from inhalation of resuspended material (Jones J.A., 1996) is also calculated as the product of the time averaged air concentration, breathing rate and dose per unit activity inhaled. The relationship between amount of material deposited and the resuspended air concentration depends on a number of factors. A simple model is used in PC COSYMA, so that the air concentration at a specified time after deposition is obtained using a resuspension factor, R, (the ratio of air concentration to deposition with units of m^{-1}) given by:

$$R = ae^{-bt} + c \qquad (3.3)$$

Where a, b and c are parameters, which can be set by the user, and t is the time since deposition. The air concentration inside and outside buildings may be different, and PC COSYMA incorporates this by using a suitable reduction factor, which may be set by the user, when calculating inhalation doses for people who are indoors. However, for the current study, the air concentrations calculated here are those for outdoor situations.

CHAPTER FOUR – RESULTS

4.1 Overview

This research project focus was on the dose accrued to the following organs, namely, lung, thyroid, foetus, skin and bone marrow. The output data also included effective dose as a result of the exposure. Ingestion was excluded as it can be assumed that the consumption of affected food in the area will be prohibited, and therefore the dose due to ingestion need not be considered. Dose comparisons were performed for one, two, seven and thirty days, as well as a 50 year lifetime dose.

It was also necessary to decide on the outcome of such an assessment as the methods used throughout the last few decades have changed tremendously as newer research data was utilised to derive operational limits. In terms of the way forward it is sensible to use the latest IAEA references (IAEA, 2011), which would mean that this information could be used by the Koeberg Emergency Plan considering the latest references for the determination of Operational Intervention levels.

To determine gamma dose as a result of cloudshine and groundshine, it was necessary to change the input data by excluding resuspension and skin dose. This was performed and all data collated into Table 4.1 and graphically represented in Fig 4.1. Tables 4.2 to 4.21 are the results obtained for the various PC COSYMA runs for the different wind categories and at different distances as specified. The results have also been shown graphically and are represented in Fig 4.2 up to Fig 4.15.

Table 4.1: 1 Day Dose (Sv) - D Dispersion Category - Gamma Only

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow
0.35 km	33.1	36.8	29.3	31.8	31.2
0.80 km	12.0	13.3	10.5	11.5	11.3
1.20 km	7.02	7.82	6.15	6.75	6.61
2 km	3.46	3.86	3.02	3.34	3.26
3 km	1.94	2.16	1.69	1.87	1.82
4 km	1.27	1.42	1.10	1.22	1.19
5 km	0.925	1.03	0.803	0.892	0.868
6 km	0.703	0.786	0.61	0.679	0.661
7 km	0.556	0.621	0.482	0.537	0.522
8 km	0.453	0.506	0.392	0.437	0.425
9 km	0.378	0.422	0.327	0.365	0.355
10 km	0.323	0.361	0.279	0.312	0.303
11 km	0.278	0.31	0.241	0.268	0.261
12 km	0.242	0.271	0.210	0.234	0.227
13 km	0.215	0.240	0.186	0.207	0.201
14 km	0.194	0.217	0.168	0.188	0.183
15 km	0.184	0.206	0.160	0.178	0.173
16 km	0.173	0.193	0.150	0.167	0.162
17 km	0.166	0.185	0.143	0.160	0.155
18 km	0.158	0.177	0.137	0.153	0.149
19 km	0.148	0.165	0.128	0.143	0.139
20 km	0.141	0.158	0.122	0.136	0.133
21 km	0.137	0.153	0.118	0.132	0.128
22 km	0.131	0.146	0.113	0.126	0.123
23 km	0.125	0.140	0.108	0.121	0.117

Using the above mentioned results, as expected due to the penetrating power of gamma radiation, the results are similar for the different organs within the body. To simplify the process the effective gamma external dose versus lung and thyroid organ equivalent doses, a relationship between external effective dose and the equivalent doses can be used. This is a reasonable assumption as the end point relationship is a measurable dose rate in the field, which can be related to lung and thyroid equivalent dose.

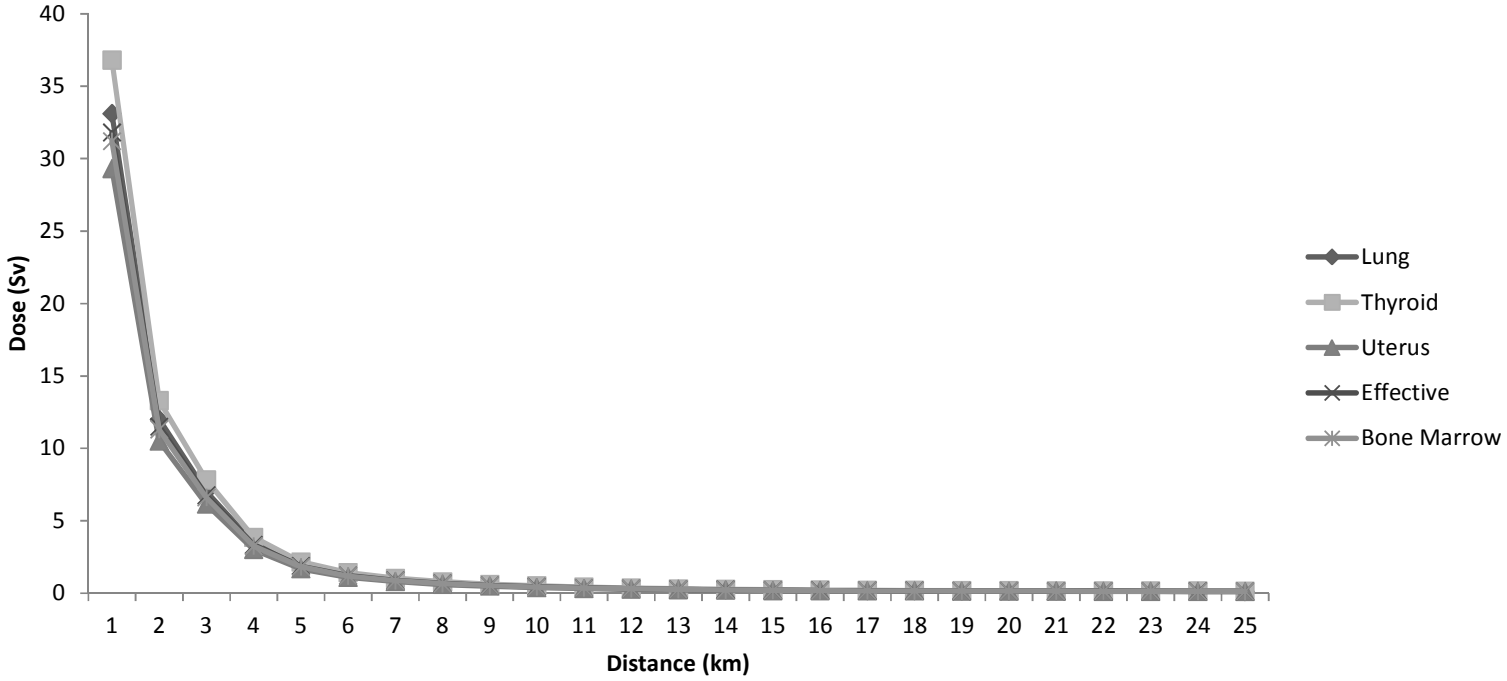

Figure 4.1: 1 Day Dose (Sv) – Gamma only (Cloudshine and Groundshine)

Table 4.2: 1 Day Dose (Sv) - D Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	62.7	257	30.7	51.7	32.6	31.8
0.80 km	22.1	87.8	11.0	18.3	11.7	11.5
1.20 km	12.8	49.6	6.42	10.6	6.87	6.75
2 km	6.20	23.5	3.15	5.14	3.38	3.34
3 km	3.44	12.8	1.76	2.85	1.89	1.87
4 km	2.24	8.23	1.15	1.86	1.24	1.22
5 km	1.62	5.86	0.835	1.34	0.90	0.892
6 km	1.23	4.42	0.635	1.02	0.685	0.679
7 km	0.975	3.47	0.502	0.805	0.541	0.537
8 km	0.795	2.82	0.408	0.655	0.441	0.437
9 km	0.664	2.34	0.341	0.547	0.368	0.365
10 km	0.567	1.99	0.291	0.466	0.314	0.312
11 km	0.489	1.71	0.25	0.401	0.27	0.268
12 km	0.427	1.49	0.218	0.35	0.236	0.234
13 km	0.379	1.32	0.193	0.31	0.209	0.207
14 km	0.344	1.2	0.175	0.281	0.189	0.188
15 km	0.326	1.13	0.166	0.267	0.179	0.178
16 km	0.307	1.06	0.156	0.25	0.168	0.167
17 km	0.294	1.01	0.149	0.24	0.161	0.16
18 km	0.281	0.967	0.143	0.229	0.154	0.153
19 km	0.265	0.911	0.133	0.215	0.144	0.143
20 km	0.253	0.869	0.127	0.205	0.138	0.136
21 km	0.244	0.833	0.123	0.198	0.133	0.132
22 km	0.234	0.797	0.118	0.189	0.127	0.126
23 km	0.225	0.765	0.113	0.182	0.122	0.121

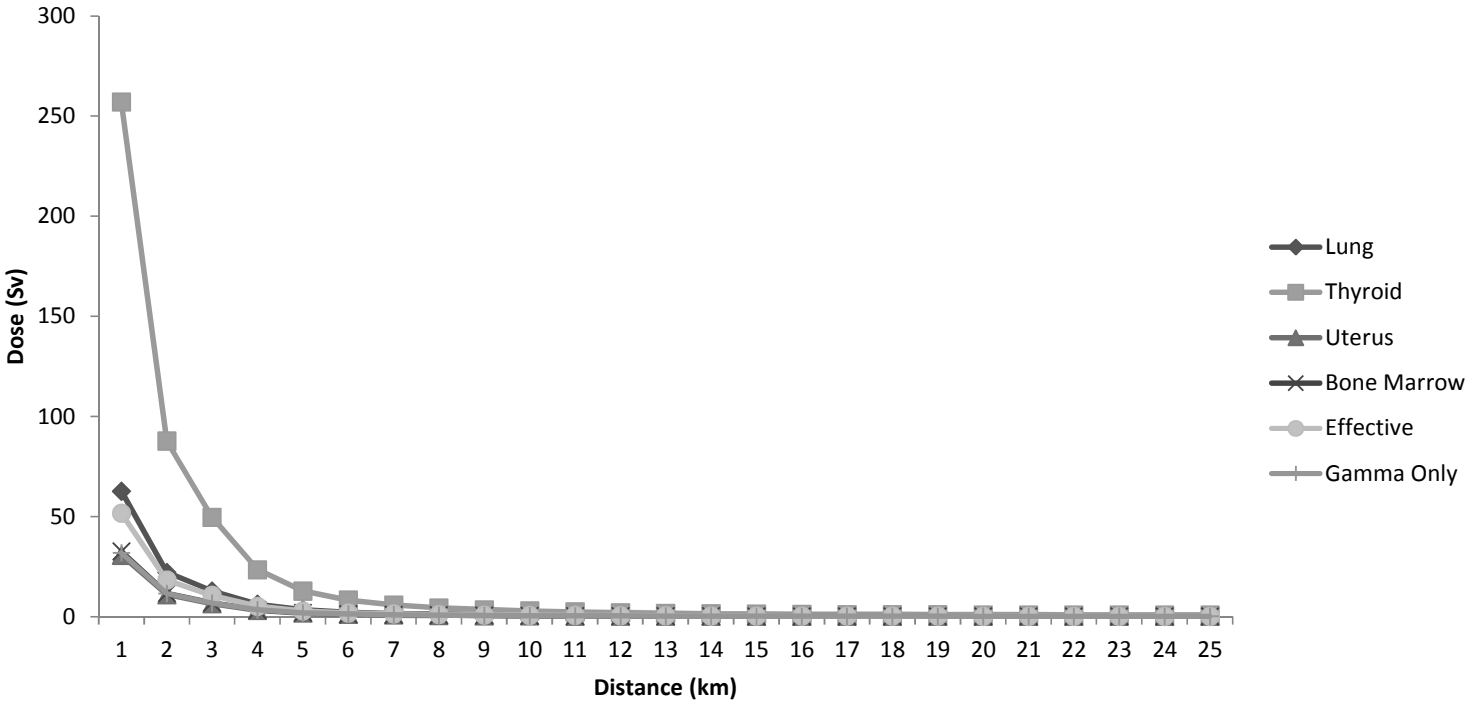

Figure 4.2: 1 Day Dose (Sv) – D Dispersion Category

Table 4.3: 1 Day Dose (Sv) - F Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	141	577	65.5	113	69.5	67.5
0.80 km	65.1	259	30.9	52.5	32.9	32.1
1.20 km	41.5	161	20	33.5	21.3	20.9
2 km	21.9	82.1	10.6	17.6	11.3	11.1
3 km	12.6	45.9	6.11	10.1	6.55	6.45
4 km	8.31	29.8	4.04	6.65	4.33	4.27
5 km	6.04	21.2	2.94	4.83	3.17	3.13
6 km	4.60	15.9	2.24	3.67	2.41	2.38
7 km	3.90	13.3	1.89	3.10	2.04	2.02
8 km	3.48	11.7	1.68	2.75	1.82	1.80
9 km	3.13	10.5	1.51	2.47	1.63	1.61
10 km	2.87	9.49	1.38	2.25	1.49	1.48
11 km	2.63	8.63	1.26	2.06	1.36	1.34
12 km	2.42	7.89	1.15	1.88	1.24	1.23
13 km	2.25	7.28	1.06	1.74	1.15	1.13
14 km	2.09	6.73	0.982	1.62	1.06	1.05
15 km	1.96	6.25	0.913	1.50	0.985	0.973
16 km	1.84	5.84	0.857	1.41	0.924	0.913
17 km	1.73	5.45	0.798	1.32	0.861	0.85
18 km	1.64	5.11	0.75	1.24	0.809	0.799
19 km	1.55	4.81	0.707	1.17	0.762	0.752
20 km	1.48	4.54	0.668	1.11	0.721	0.711
21 km	1.46	4.34	0.672	1.10	0.728	0.721
22 km	1.39	4.11	0.637	1.04	0.69	0.683
23 km	1.33	3.91	0.608	0.998	0.658	0.652

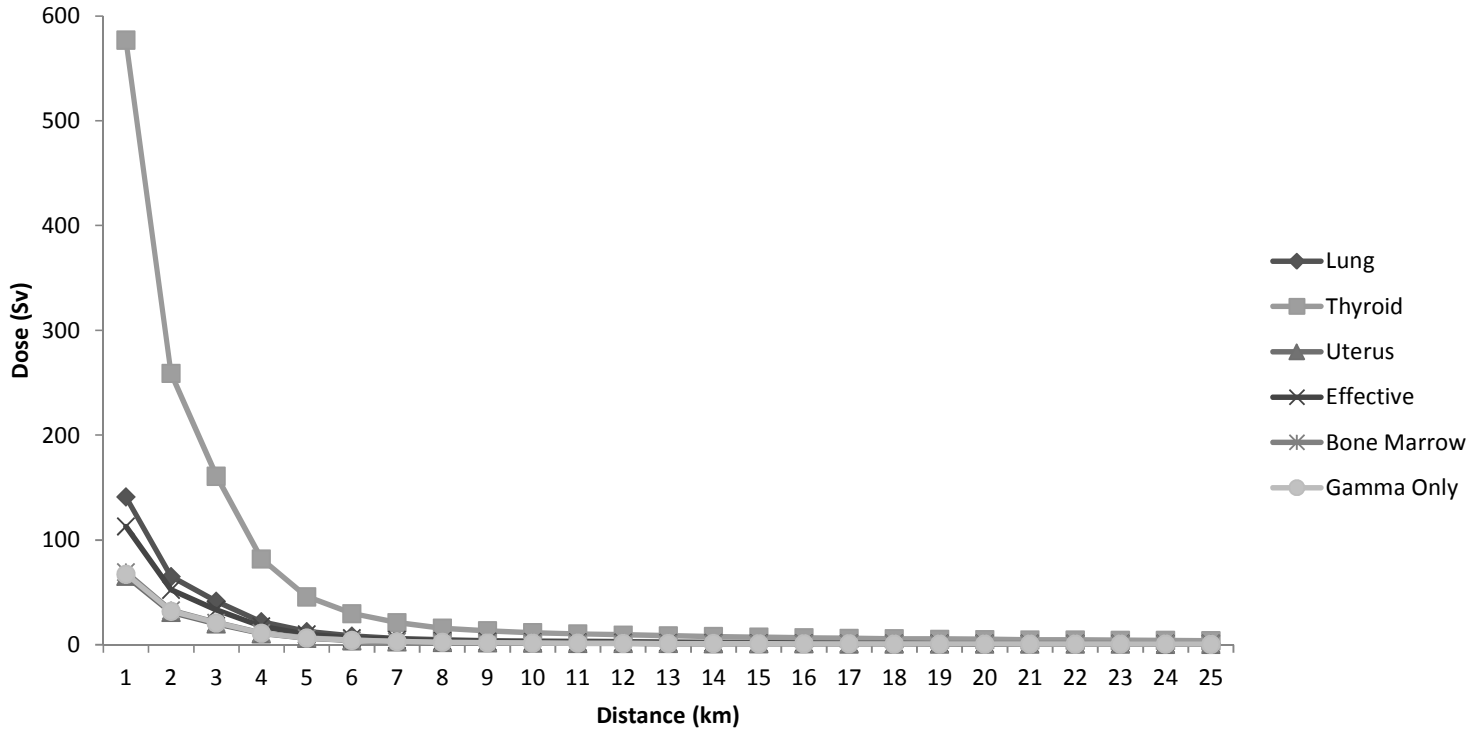

Figure 4.3: 1 Day Dose (Sv) – F Dispersion Category

Table 4.4: 2 Day Dose (Sv) - D Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	88.1	352	42.3	70.0	45.0	43.9
0.80 km	30.8	120	15.0	24.5	16.0	15.6
1.20 km	17.7	67.7	8.64	14.1	9.25	9.08
2 km	8.52	32.0	4.2	6.79	4.51	4.43
3 km	4.70	17.4	2.32	3.74	2.50	2.46
4 km	3.06	11.2	1.52	2.44	1.63	1.61
5 km	2.21	7.97	1.10	1.75	1.18	1.17
6 km	1.68	6.01	0.832	1.33	0.896	0.885
7 km	1.33	4.73	0.657	1.05	0.708	0.699
8 km	1.08	3.84	0.535	0.855	0.576	0.569
9 km	0.903	3.19	0.446	0.713	0.481	0.475
10 km	0.77	2.71	0.38	0.607	0.41	0.405
11 km	0.664	2.33	0.327	0.523	0.353	0.349
12 km	0.581	2.03	0.286	0.457	0.308	0.304
13 km	0.516	1.80	0.253	0.405	0.273	0.27
14 km	0.469	1.63	0.23	0.367	0.248	0.245
15 km	0.445	1.54	0.218	0.348	0.235	0.232
16 km	0.417	1.45	0.204	0.326	0.22	0.217
17 km	0.40	1.38	0.195	0.313	0.211	0.208
18 km	0.383	1.32	0.187	0.299	0.202	0.199
19 km	0.361	1.25	0.175	0.281	0.189	0.187
20 km	0.346	1.19	0.167	0.268	0.181	0.178
21 km	0.333	1.14	0.161	0.259	0.174	0.172
22 km	0.319	1.09	0.155	0.248	0.167	0.165
23 km	0.307	1.05	0.148	0.238	0.16	0.158

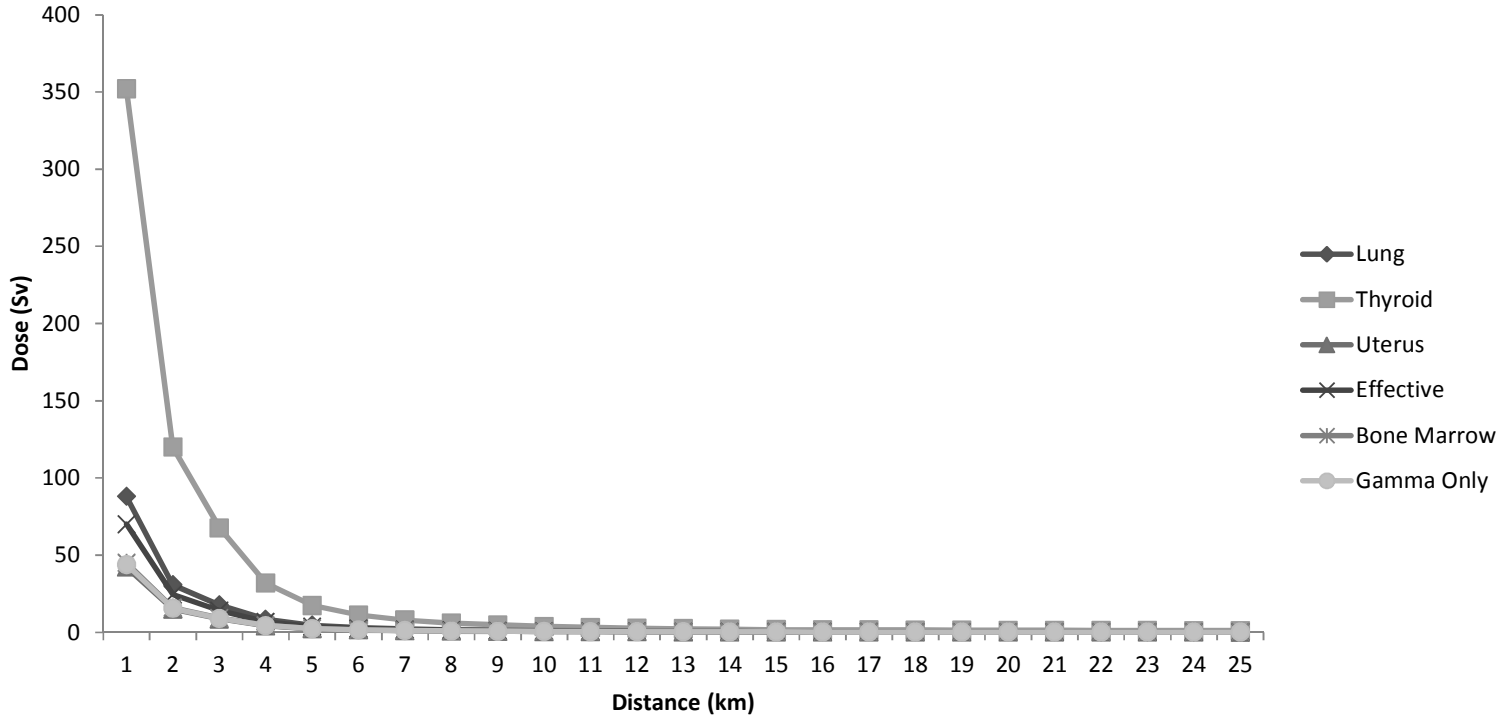

Figure 4.4: 2 Day Dose (Sv) – D Dispersion Category

Table 4.5: 2 Day Dose (Sv) - F Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	200	798	91.3	154	97.1	94.6
0.80 km	91.8	357	42.6	71.1	45.4	44.3
1.20 km	58.1	221	27.2	45.1	29	28.4
2 km	30.4	113	14.3	23.5	15.3	15.0
3 km	17.4	62.8	8.17	13.4	8.77	8.61
4 km	11.5	40.8	5.38	8.81	5.78	5.68
5 km	8.32	29.0	3.90	6.36	4.20	4.13
6 km	6.33	21.8	2.97	4.83	3.19	3.14
7 km	5.36	18.2	2.50	4.07	2.69	2.65
8 km	4.78	16.1	2.22	3.61	2.39	2.36
9 km	4.31	14.3	2.00	3.24	2.15	2.12
10 km	3.94	13	1.82	2.96	1.96	1.93
11 km	3.61	11.8	1.66	2.7	1.79	1.76
12 km	3.33	10.8	1.52	2.48	1.64	1.61
13 km	3.1	9.97	1.41	2.29	1.51	1.49
14 km	2.89	9.23	1.3.0	2.13	1.4.0	1.38
15 km	2.7	8.57	1.21	1.98	1.31	1.28
16 km	2.55	8.01	1.14	1.86	1.22	1.20
17 km	2.40	7.48	1.06	1.74	1.14	1.12
18 km	2.27	7.02	0.998	1.64	1.08	1.06
19 km	2.16	6.61	0.942	1.55	1.02	0.997
20 km	2.05	6.24	0.891	1.47	0.961	0.943
21 km	2.01	5.95	0.884	1.44	0.956	0.942
22 km	1.91	5.64	0.839	1.37	0.907	0.893
23 km	1.84	5.36	0.801	1.31	0.866	0.853

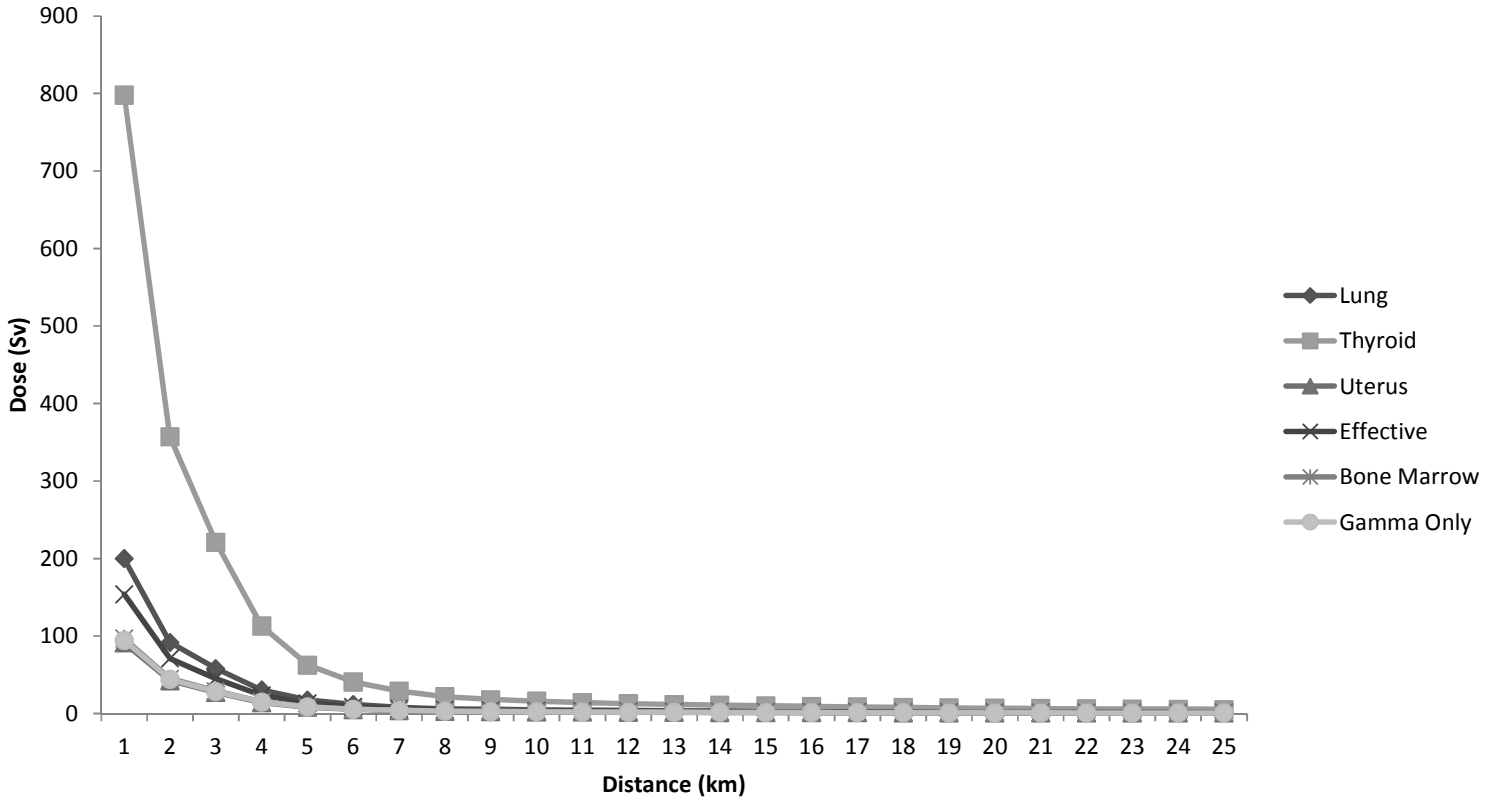

Figure 4.5: 2 Day Dose (Sv) – F Dispersion Category

Table 4.6: 7 Day Dose (Sv) - D Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	184	795	72.8	132	78.1	75.2
0.80 km	63.8	270	25.4	45.6	27.3	26.3
1.20 km	36.3	152	14.5	25.9	15.6	15.1
2 km	17.4	71.6	6.96	12.4	7.50	7.27
3 km	9.53	38.8	3.83	6.79	4.13	4.00
4 km	6.20	25.0	2.49	4.41	2.69	2.6
5 km	4.45	17.8	1.79	3.16	1.93	1.87
6 km	3.38	13.4	1.36	2.39	1.47	1.42
7 km	2.67	10.5	1.07	1.89	1.16	1.12
8 km	2.18	8.56	0.872	1.54	0.942	0.914
9 km	1.82	7.12	0.727	1.28	0.786	0.763
10 km	1.55	6.04	0.619	1.09	0.669	0.65
11 km	1.34	5.2	0.534	0.939	0.577	0.56
12 km	1.17	4.54	0.466	0.821	0.504	0.489
13 km	1.04	4.02	0.414	0.729	0.447	0.434
14 km	0.95	3.65	0.375	0.661	0.406	0.394
15 km	0.901	3.46	0.356	0.627	0.385	0.373
16 km	0.846	3.24	0.334	0.588	0.361	0.35
17 km	0.811	3.09	0.32	0.563	0.346	0.335
18 km	0.776	2.95	0.305	0.538	0.33	0.32
19 km	0.737	2.79	0.288	0.509	0.312	0.302
20 km	0.706	2.67	0.276	0.486	0.298	0.289
21 km	0.679	2.56	0.265	0.468	0.287	0.278
22 km	0.652	2.45	0.254	0.448	0.275	0.267
23 km	0.627	2.35	0.244	0.431	0.264	0.256

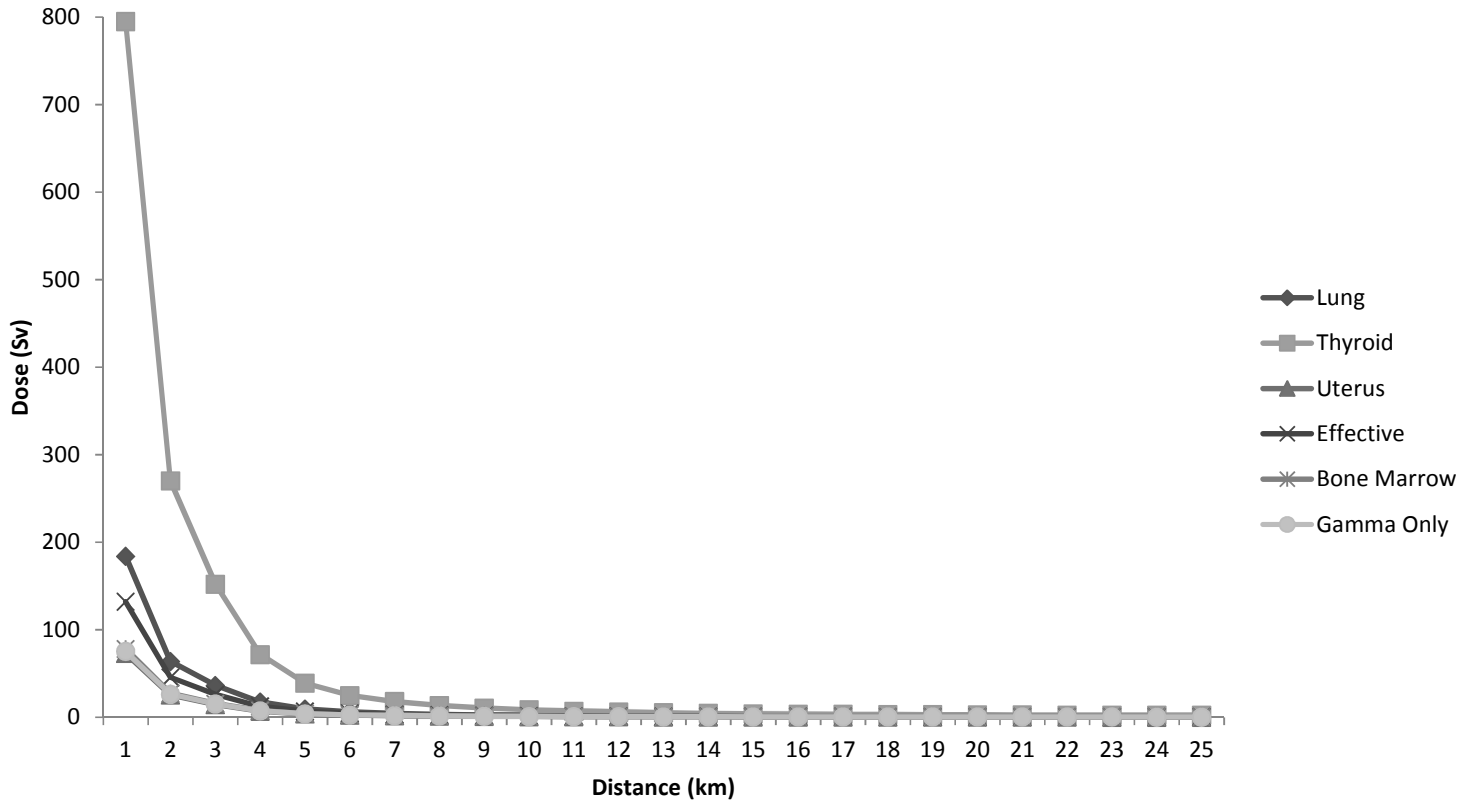

Figure 4.6: 7 Day Dose (Sv) – D Dispersion Category

Table 4.7: 7 Day Dose (Sv) - F Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	429	1830	159	297	171	165
0.80 km	195	813	73.3	135	78.6	75.8
1.20 km	122	502	46.3	84.5	49.7	48.0
2 km	63.4	254	24.1	43.7	25.9	25.0
3 km	36.1	142	13.7	24.7	14.7	14.3
4 km	23.8	91.8	8.99	16.2	9.69	9.37
5 km	17.2	65.1	6.48	11.6	6.99	6.77
6 km	13.1	49.0	4.91	8.82	5.31	5.13
7 km	11.1	40.9	4.14	7.41	4.47	4.33
8 km	9.89	36.1	3.68	6.59	3.97	3.85
9 km	8.94	32.3	3.3	5.92	3.57	3.46
10 km	8.18	29.2	3.01	5.39	3.26	3.15
11 km	7.52	26.6	2.75	4.93	2.98	2.88
12 km	6.96	24.3	2.53	4.53	2.73	2.64
13 km	6.49	22.5	2.34	4.2	2.53	2.44
14 km	6.07	20.8	2.17	3.91	2.35	2.27
15 km	5.69	19.3	2.03	3.64	2.19	2.12
16 km	5.38	18.1	1.91	3.42	2.06	1.99
17 km	5.08	16.9	1.79	3.21	1.93	1.86
18 km	4.82	15.9	1.68	3.03	1.82	1.75
19 km	4.59	15.0	1.59	2.87	1.72	1.66
20 km	4.38	14.1	1.51	2.72	1.63	1.57
21 km	4.23	13.4	1.47	2.63	1.6	1.54
22 km	4.05	12.7	1.40	2.51	1.52	1.46
23 km	3.89	12.1	1.34	2.40	1.45	1.40

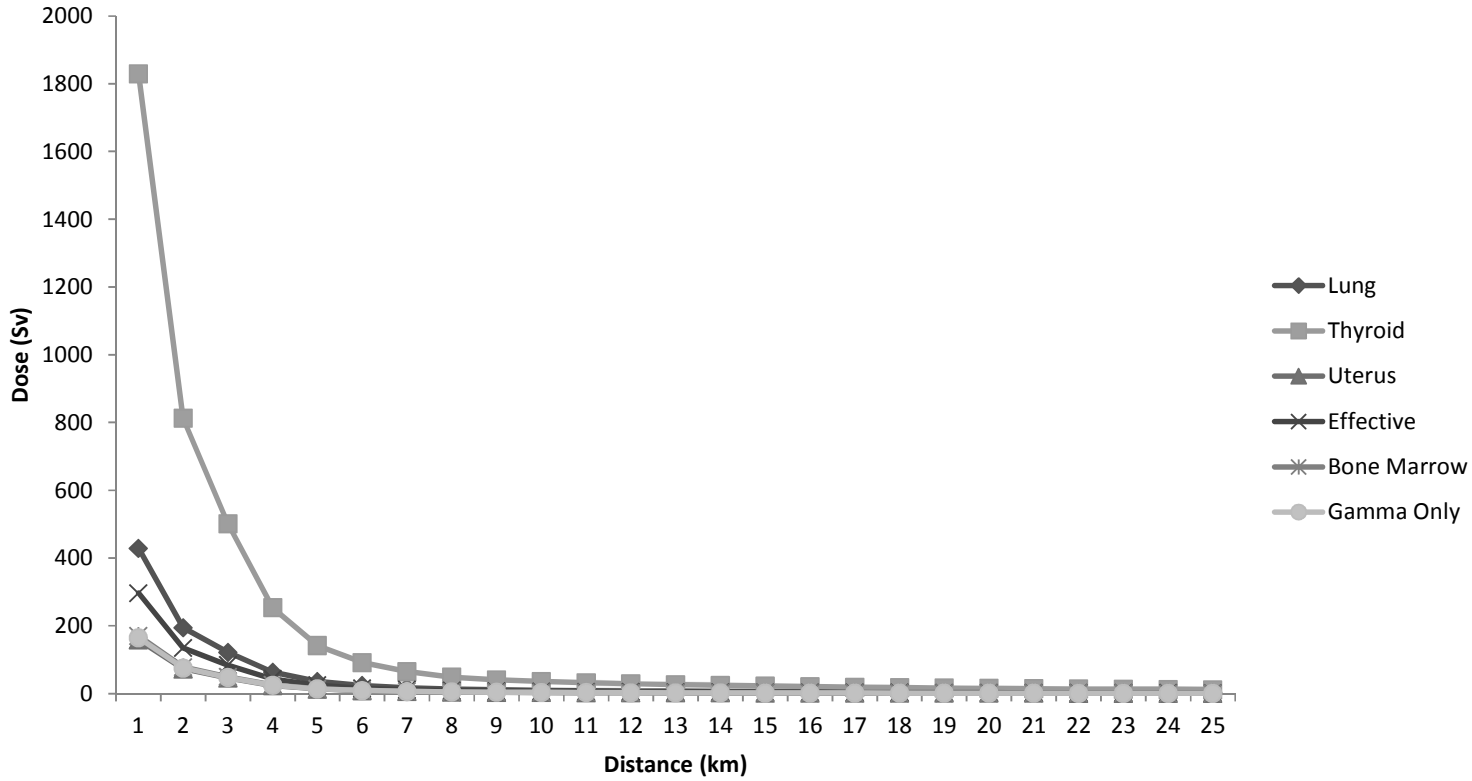

Figure 4.7: 7 Day Dose (Sv) – F Dispersion Category

Table 4.8: 30 Day Dose (Sv) - D Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	307	1200	115	205	125	119
0.80 km	106	409	39.9	70.4	43.2	41.3
1.20 km	60.0	230	22.7	39.9	24.6	23.5
2 km	28.6	108	10.8	19.0	11.7	11.3
3 km	15.7	58.6	5.94	10.4	6.44	6.18
4 km	10.2	37.8	3.86	6.73	4.18	4.01
5 km	7.32	26.8	2.76	4.81	3.00	2.88
6 km	5.55	20.2	2.09	3.64	2.27	2.18
7 km	4.40	15.9	1.65	2.88	1.79	1.72
8 km	3.59	12.9	1.35	2.34	1.46	1.40
9 km	3.00	10.8	1.12	1.95	1.22	1.17
10 km	2.55	9.12	0.956	1.66	1.04	0.996
11 km	2.21	7.86	0.825	1.43	0.895	0.859
12 km	1.93	6.86	0.721	1.25	0.783	0.751
13 km	1.72	6.08	0.64	1.11	0.695	0.667
14 km	1.57	5.52	0.581	1.01	0.632	0.606
15 km	1.49	5.23	0.551	0.958	0.599	0.574
16 km	1.40	4.90	0.517	0.899	0.562	0.539
17 km	1.34	4.68	0.495	0.86	0.538	0.515
18 km	1.28	4.46	0.473	0.822	0.514	0.493
19 km	1.22	4.24	0.448	0.78	0.487	0.467
20 km	1.17	4.05	0.429	0.746	0.466	0.446
21 km	1.12	3.88	0.412	0.716	0.448	0.429
22 km	1.08	3.72	0.395	0.687	0.43	0.412
23 km	1.04	3.57	0.38	0.661	0.413	0.396

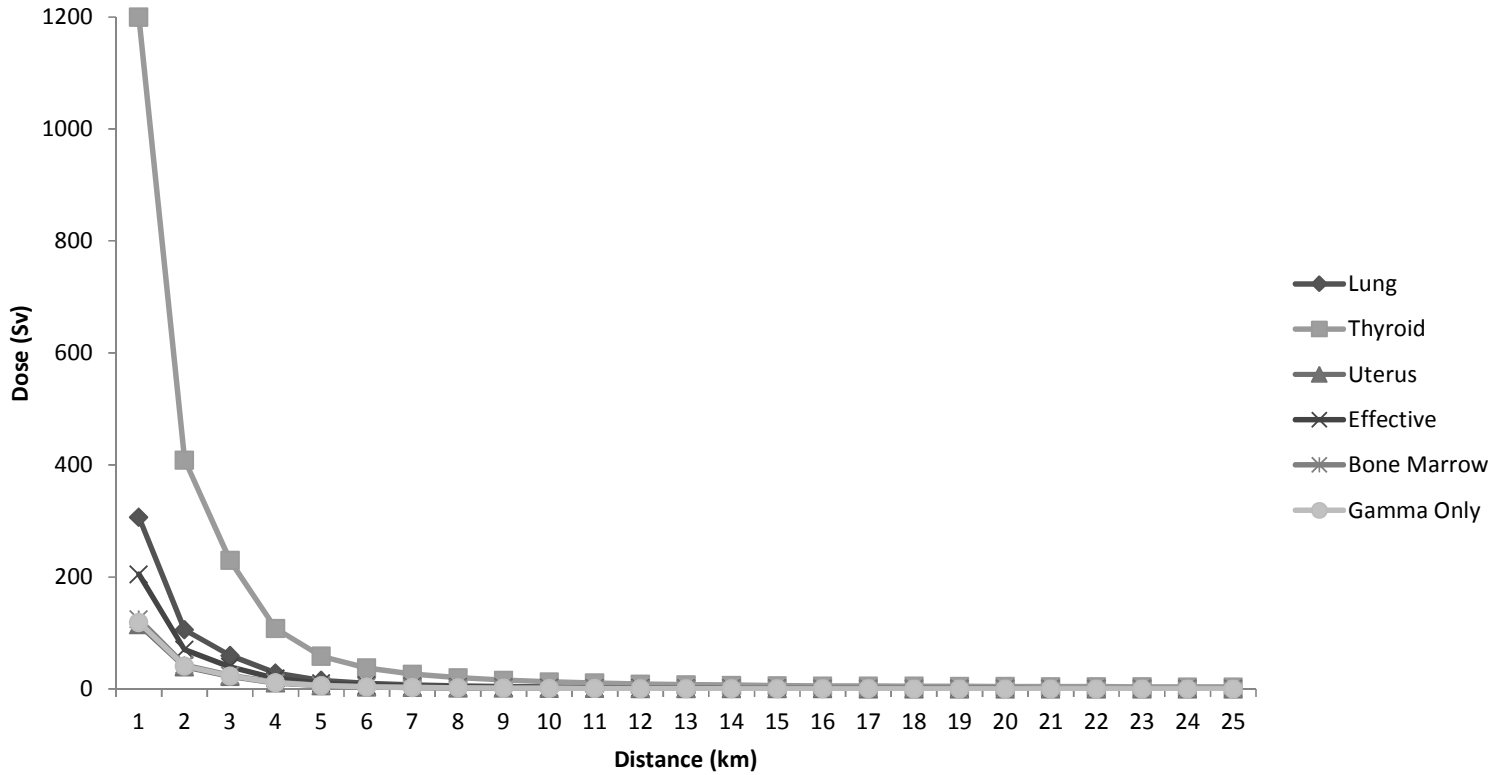

Figure 4.8: 30 Day Dose (Sv) – D Dispersion Category

Table 4.9: 30 Day Dose (Sv) - F Dispersion Category

Distance	Lung	Thyroid	Uterus	Effective	Bone Marrow	Gamma Only
0.35 km	770	2790	257	472	278	266
0.80 km	343	1240	117	213	127	121
1.20 km	213	763	73.5	133	79.6	76.2
2 km	110	386	38	68.1	41.2	39.4
3 km	62.3	215	21.5	38.4	23.3	22.3
4 km	41.1	139	14.1	25.2	15.3	14.7
5 km	29.6	98.8	10.2	18.0	11.0	10.6
6 km	22.5	74.3	7.70	13.7	8.36	8.00
7 km	19.0	62.0	6.48	11.5	7.04	6.73
8 km	17.0	54.8	5.76	10.2	6.26	5.99
9 km	15.4	49.0	5.18	9.18	5.63	5.38
10 km	14.1	44.3	4.72	8.36	5.13	4.90
11 km	13.0	40.4	4.32	7.65	4.70	4.49
12 km	12.0	37	3.97	7.04	4.32	4.13
13 km	11.2	34.1	3.69	6.54	4.01	3.83
14 km	10.5	31.7	3.43	6.09	3.73	3.56
15 km	9.86	29.4	3.20	5.68	3.49	3.32
16 km	9.33	27.5	3.01	5.35	3.28	3.12
17 km	8.83	25.8	2.83	5.03	3.08	2.93
18 km	8.38	24.2	2.67	4.75	2.91	2.77
19 km	7.98	22.8	2.53	4.50	2.76	2.62
20 km	7.62	21.5	2.40	4.27	2.62	2.49
21 km	7.35	20.4	2.32	4.11	2.53	2.41
22 km	7.03	19.4	2.21	3.91	2.41	2.30
23 km	6.76	18.4	2.12	3.74	2.31	2.20

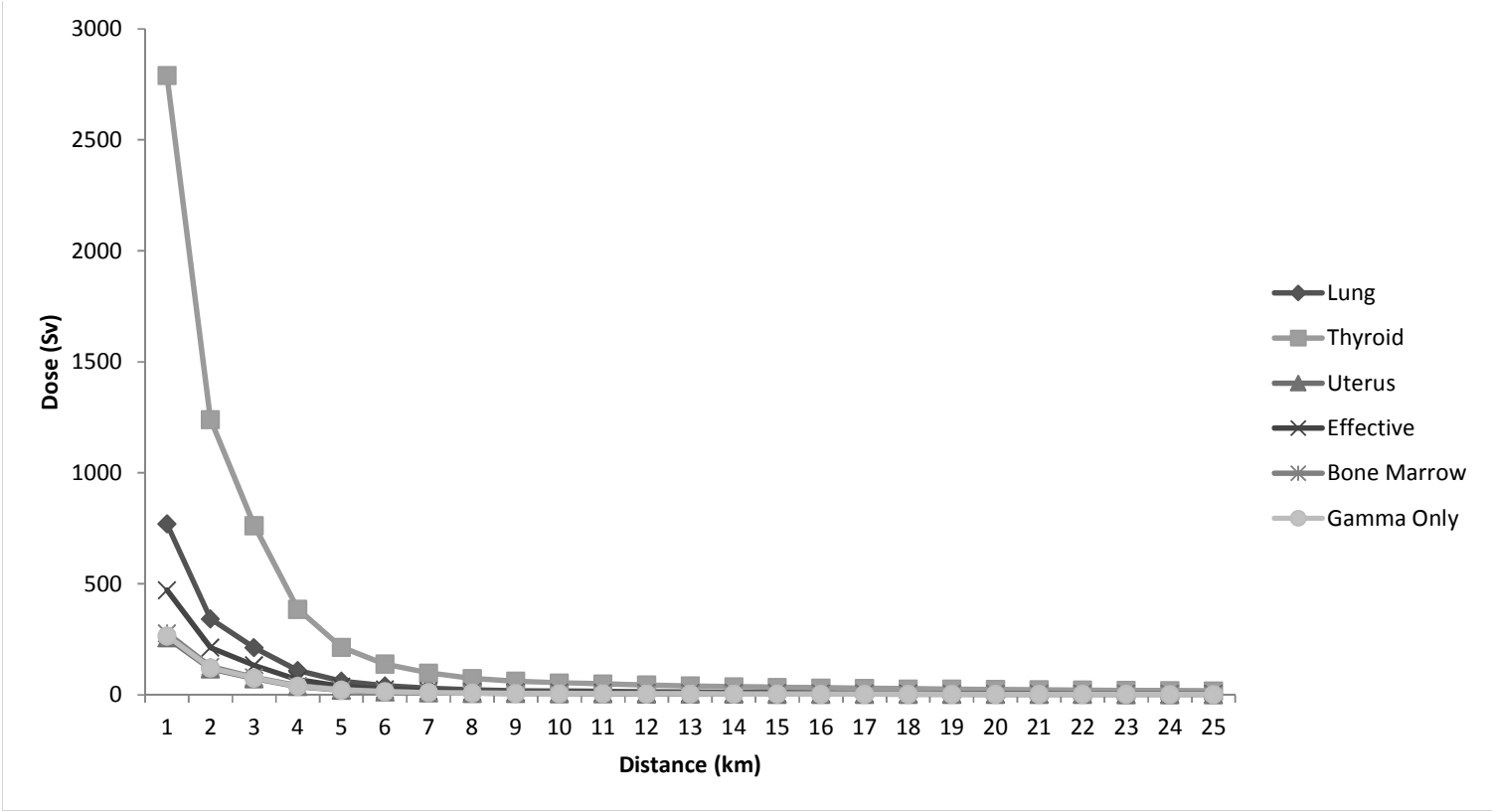

Figure 4.9: 30 Day Dose (Sv) – F Dispersion Category

Table 4.10: 50 Year Dose (Sv) - D Dispersion Category

Distance	Lung	Thyroid	Effective	Bone Marrow	Gamma Only
0.35 km	744	1520	464	366	342
0.80 km	256	516	160	126	118
1.20 km	145	290	90.3	71.6	66.9
2 km	69.2	137	43.0	34.1	31.9
3 km	38.0	74.4	23.6	18.7	17.5
4 km	24.7	48.1	15.3	12.2	11.4
5 km	17.7	34.2	11.0	8.72	8.16
6 km	13.4	25.8	8.31	6.63	6.20
7 km	10.6	20.3	6.58	5.25	4.91
8 km	8.70	16.5	5.37	4.29	4.01
9 km	7.28	13.8	4.49	3.58	3.35
10 km	6.20	11.7	3.82	3.05	2.86
11 km	5.37	10.1	3.30	2.64	2.47
12 km	4.71	8.82	2.90	2.32	2.17
13 km	4.20	7.83	2.58	2.06	1.93
14 km	3.82	7.11	2.35	1.88	1.76
15 km	3.63	6.74	2.23	1.78	1.67
16 km	3.41	6.31	2.09	1.67	1.57
17 km	3.27	6.04	2.00	1.60	1.50
18 km	3.13	5.76	1.92	1.54	1.43
19 km	2.99	5.48	1.83	1.47	1.37
20 km	2.87	5.24	1.75	1.40	1.31
21 km	2.76	5.02	1.68	1.35	1.26
22 km	2.65	4.82	1.62	1.30	1.21
23 km	2.56	4.63	1.56	1.25	1.17

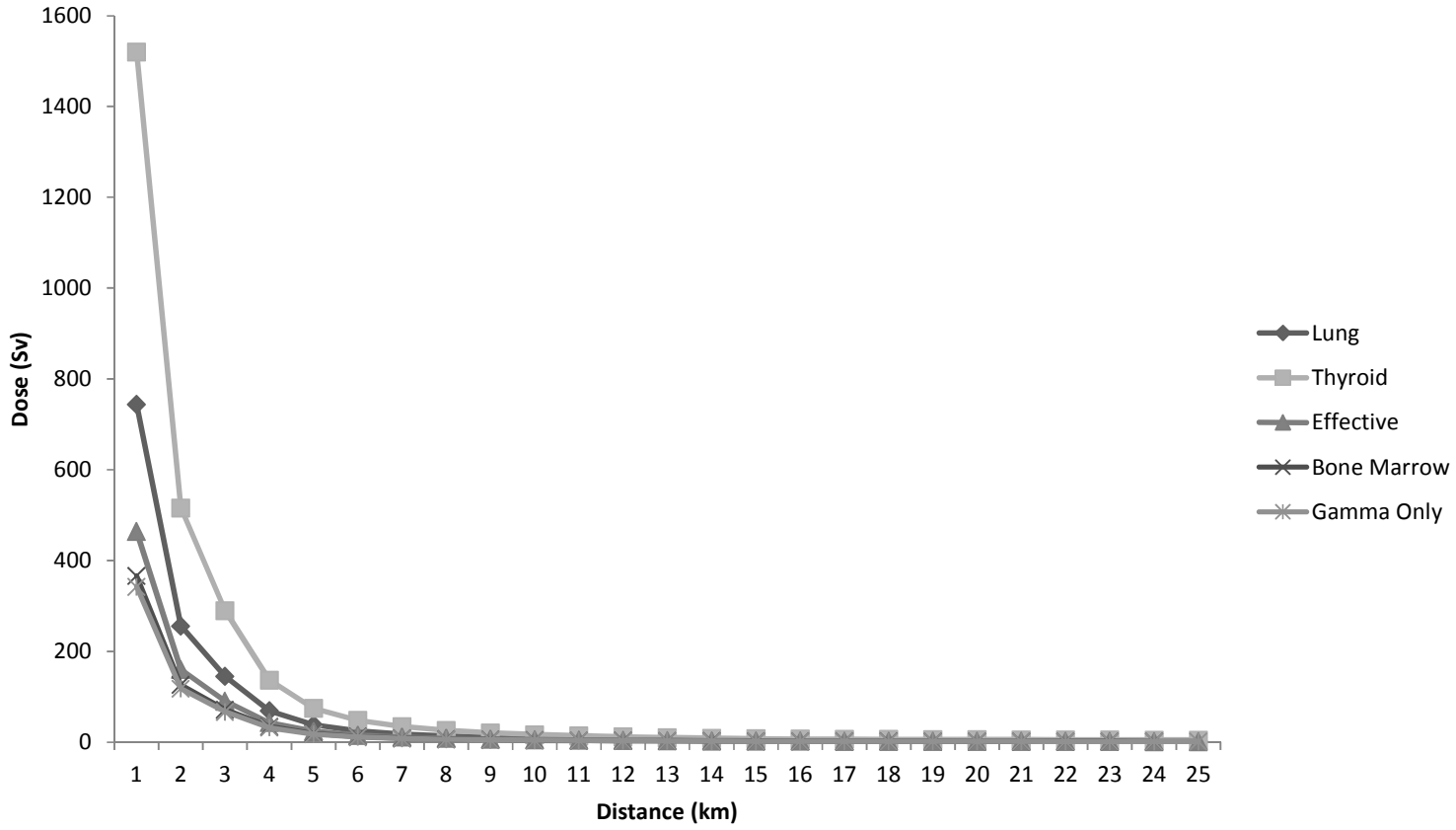

Figure 4.10: 50 Year Dose (Sv) – D Dispersion Category

Table 4.11: 50 Year Dose (Sv) - F Dispersion Category

Distance	Lung	Thyroid	Effective	Bone Marrow	Gamma Only
0.35 km	1940	3460	1050	797	729
0.80 km	857	1550	477	367	337
1.20 km	532	958	299	231	213
2 km	273	487	155	120	111
3 km	155	273	88.1	68.8	63.6
4 km	103	178	58.1	45.5	42.1
5 km	74.0	126	41.9	32.9	30.4
6 km	56.4	95.3	31.9	25.1	23.2
7 km	47.8	79.8	26.9	21.2	19.6
8 km	42.7	70.7	24.1	19.0	17.5
9 km	38.8	63.4	21.8	17.2	15.9
10 km	35.5	57.5	19.9	15.7	14.5
11 km	32.8	52.5	18.3	14.5	13.4
12 km	30.5	48.3	17.0	13.4	12.4
13 km	28.5	44.7	15.8	12.6	11.6
14 km	26.8	41.6	14.9	11.8	10.9
15 km	25.2	38.8	13.9	11.1	10.2
16 km	23.9	36.4	13.2	10.5	9.67
17 km	22.7	34.2	12.5	9.93	9.16
18 km	21.6	32.2	11.9	9.44	8.70
19 km	20.6	30.4	11.3	9.00	8.29
20 km	19.7	28.8	10.8	8.6	7.92
21 km	19.0	27.4	10.4	8.29	7.64
22 km	18.2	26.1	9.93	7.94	7.31
23 km	17.5	24.9	9.55	7.65	7.04

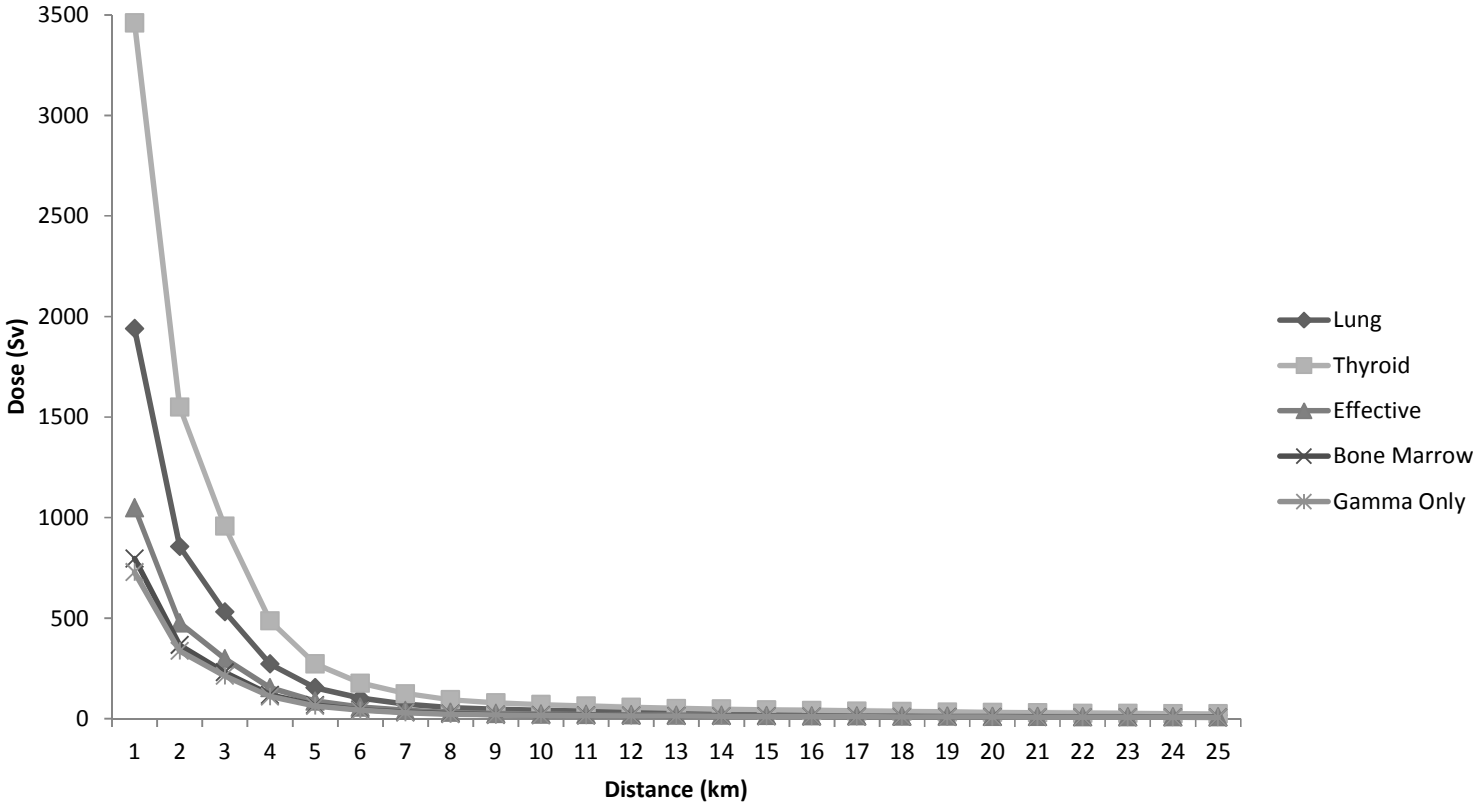

Figure 4.11: 50 Year Dose (Sv) – F Dispersion Category

Table 4.12: 1 Day - D Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	31.8	744	1520	23.40	47.80
0.80 km	11.5	256	516	22.26	44.87
1.20 km	6.75	145	290	21.48	42.96
2 km	3.34	69.2	137	20.72	41.02
3 km	1.87	38.0	74.4	20.32	39.79
4 km	1.22	24.7	48.1	20.25	39.43
5 km	0.892	17.7	34.2	19.84	38.34
6 km	0.679	13.4	25.8	19.73	38.00
7 km	0.537	10.6	20.3	19.74	37.80
8 km	0.437	8.70	16.5	19.91	37.76
9 km	0.365	7.28	13.8	19.95	37.81
10 km	0.312	6.20	11.7	19.87	37.50
11 km	0.268	5.37	10.1	20.04	37.69
12 km	0.234	4.71	8.82	20.13	37.69
13 km	0.207	4.20	7.83	20.29	37.83
14 km	0.188	3.82	7.11	20.32	37.82
15 km	0.178	3.63	6.74	20.39	37.87
16 km	0.167	3.41	6.31	20.42	37.78
17 km	0.16	3.27	6.04	20.44	37.75
18 km	0.153	3.13	5.76	20.46	37.65
19 km	0.143	2.99	5.48	20.91	38.32
20 km	0.136	2.87	5.24	21.10	38.53
21 km	0.132	2.76	5.02	20.91	38.03
22 km	0.126	2.65	4.82	21.03	38.25
23 km	0.121	2.56	4.63	21.16	38.26

Table 4.13: 2 Days - D Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	43.9	744	1520	16.95	34.62
0.80 km	15.6	256	516	16.41	33.08
1.20 km	9.08	145	290	15.97	31.94
2 km	4.43	69.2	137	15.62	30.93
3 km	2.46	38	74.4	15.45	30.24
4 km	1.61	24.7	48.1	15.34	29.88
5 km	1.17	17.7	34.2	15.13	29.23
6 km	0.885	13.4	25.8	15.14	29.15
7 km	0.699	10.6	20.3	15.16	29.04
8 km	0.569	8.70	16.5	15.29	29.00
9 km	0.475	7.28	13.8	15.33	29.05
10 km	0.405	6.20	11.7	15.31	28.89
11 km	0.349	5.37	10.1	15.39	28.94
12 km	0.304	4.71	8.82	15.49	29.01
13 km	0.27	4.20	7.83	15.56	29.00
14 km	0.245	3.82	7.11	15.59	29.02
15 km	0.232	3.63	6.74	15.65	29.05
16 km	0.217	3.41	6.31	15.71	29.08
17 km	0.208	3.27	6.04	15.72	29.04
18 km	0.199	3.13	5.76	15.73	28.94
19 km	0.187	2.99	5.48	15.99	29.30
20 km	0.178	2.87	5.24	16.12	29.44
21 km	0.172	2.76	5.02	16.05	29.19
22 km	0.165	2.65	4.82	16.06	29.21
23 km	0.158	2.56	4.63	16.20	29.30

Table 4.14: 7 Days - D Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	75.2	744	1520	9.89	20.21
0.80 km	26.3	256	516	9.73	19.62
1.20 km	15.1	145	290	9.60	19.21
2 km	7.27	69.2	137	9.52	18.84
3 km	4.00	38.0	74.4	9.50	18.60
4 km	2.60	24.7	48.1	9.50	18.50
5 km	1.87	17.7	34.2	9.47	18.29
6 km	1.42	13.4	25.8	9.44	18.17
7 km	1.12	10.6	20.3	9.46	18.13
8 km	0.914	8.70	16.5	9.52	18.05
9 km	0.763	7.28	13.8	9.54	18.09
10 km	0.65	6.20	11.7	9.54	18.00
11 km	0.56	5.37	10.1	9.59	18.04
12 km	0.489	4.71	8.82	9.63	18.04
13 km	0.434	4.20	7.83	9.68	18.04
14 km	0.394	3.82	7.11	9.70	18.05
15 km	0.373	3.63	6.74	9.73	18.07
16 km	0.35	3.41	6.31	9.74	18.03
17 km	0.335	3.27	6.04	9.76	18.03
18 km	0.32	3.13	5.76	9.78	18.00
19 km	0.302	2.99	5.48	9.90	18.15
20 km	0.289	2.87	5.24	9.93	18.13
21 km	0.278	2.76	5.02	9.93	18.06
22 km	0.267	2.65	4.82	9.93	18.05
23 km	0.256	2.56	4.63	10.00	18.09

Table 4.15: 30 Days - D Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	119	744	1520	6.25	12.77
0.80 km	41.3	256	516	6.20	12.49
1.20 km	23.5	145	290	6.17	12.34
2 km	11.3	69.2	137	6.12	12.12
3 km	6.18	38.0	74.4	6.15	12.04
4 km	4.01	24.7	48.1	6.16	12.00
5 km	2.88	17.7	34.2	6.15	11.88
6 km	2.18	13.4	25.8	6.15	11.83
7 km	1.72	10.6	20.3	6.16	11.80
8 km	1.40	8.70	16.5	6.21	11.79
9 km	1.17	7.28	13.8	6.22	11.79
10 km	0.996	6.2	11.7	6.22	11.75
11 km	0.859	5.37	10.1	6.25	11.76
12 km	0.751	4.71	8.82	6.27	11.74
13 km	0.667	4.2	7.83	6.30	11.74
14 km	0.606	3.82	7.11	6.30	11.73
15 km	0.574	3.63	6.74	6.32	11.74
16 km	0.539	3.41	6.31	6.33	11.71
17 km	0.515	3.27	6.04	6.35	11.73
18 km	0.493	3.13	5.76	6.35	11.68
19 km	0.467	2.99	5.48	6.40	11.73
20 km	0.446	2.87	5.24	6.43	11.75
21 km	0.429	2.76	5.02	6.43	11.70
22 km	0.412	2.65	4.82	6.43	11.70
23 km	0.396	2.56	4.63	6.46	11.69

Table 4.16: 50 Years - D Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	342	744	1520	2.18	4.44
0.80 km	118	256	516	2.17	4.37
1.20 km	66.9	145	290	2.17	4.33
2 km	31.9	69.2	137	2.17	4.29
3 km	17.5	38.0	74.4	2.17	4.25
4 km	11.4	24.7	48.1	2.17	4.22
5 km	8.16	17.7	34.2	2.17	4.19
6 km	6.20	13.4	25.8	2.16	4.16
7 km	4.91	10.6	20.3	2.16	4.13
8 km	4.01	8.70	16.5	2.17	4.11
9 km	3.35	7.28	13.8	2.17	4.12
10 km	2.86	6.20	11.7	2.17	4.09
11 km	2.47	5.37	10.1	2.17	4.09
12 km	2.17	4.71	8.82	2.17	4.06
13 km	1.93	4.20	7.83	2.18	4.06
14 km	1.76	3.82	7.11	2.17	4.04
15 km	1.67	3.63	6.74	2.17	4.04
16 km	1.57	3.41	6.31	2.17	4.02
17 km	1.50	3.27	6.04	2.18	4.03
18 km	1.43	3.13	5.76	2.19	4.03
19 km	1.37	2.99	5.48	2.18	4.00
20 km	1.31	2.87	5.24	2.19	4.00
21 km	1.26	2.76	5.02	2.19	3.98
22 km	1.21	2.65	4.82	2.19	3.98
23 km	1.17	2.56	4.63	2.19	3.96

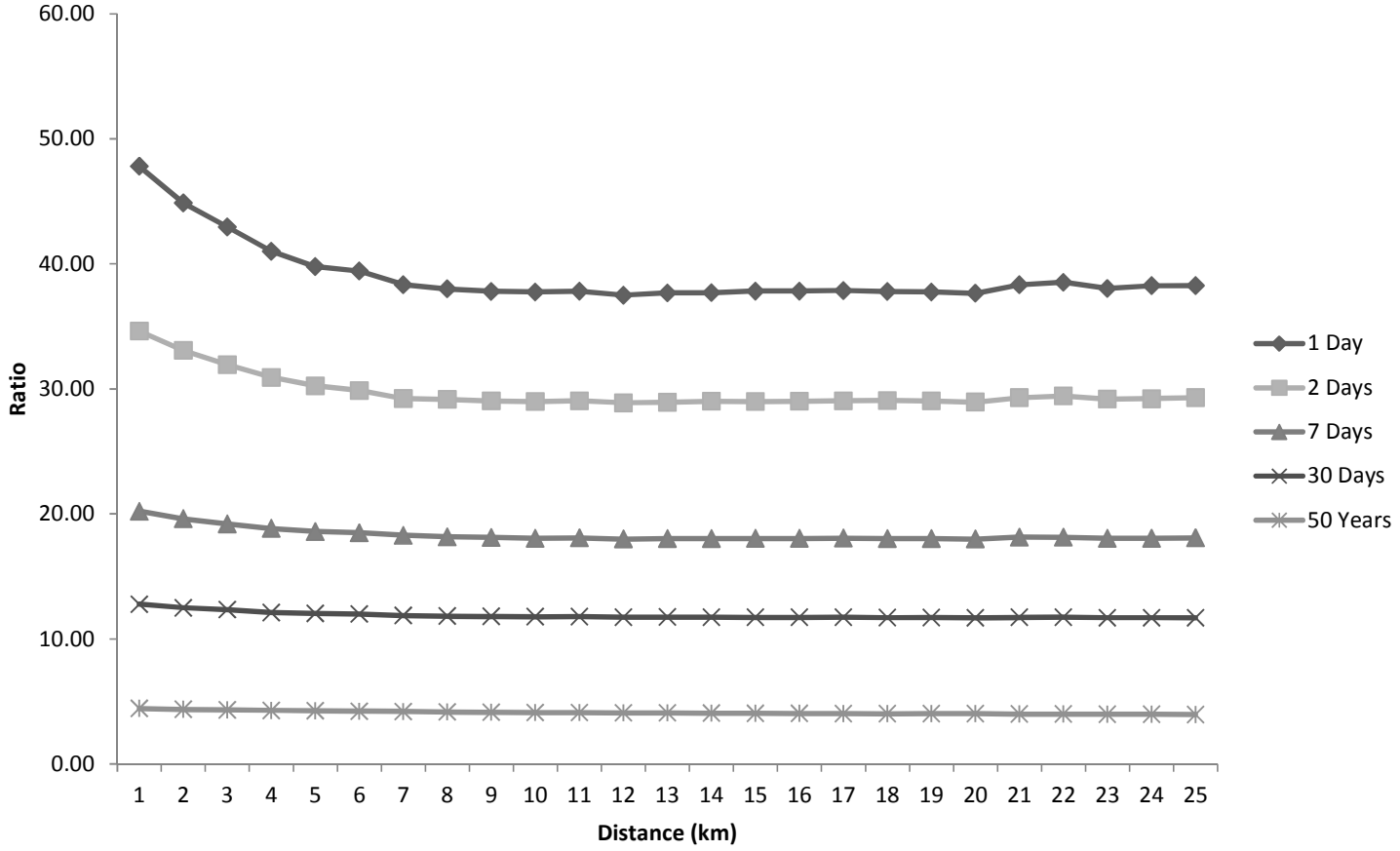

Figure 4.12: D Dispersion Category – Thyroid Dose / Gamma Dose Ratios

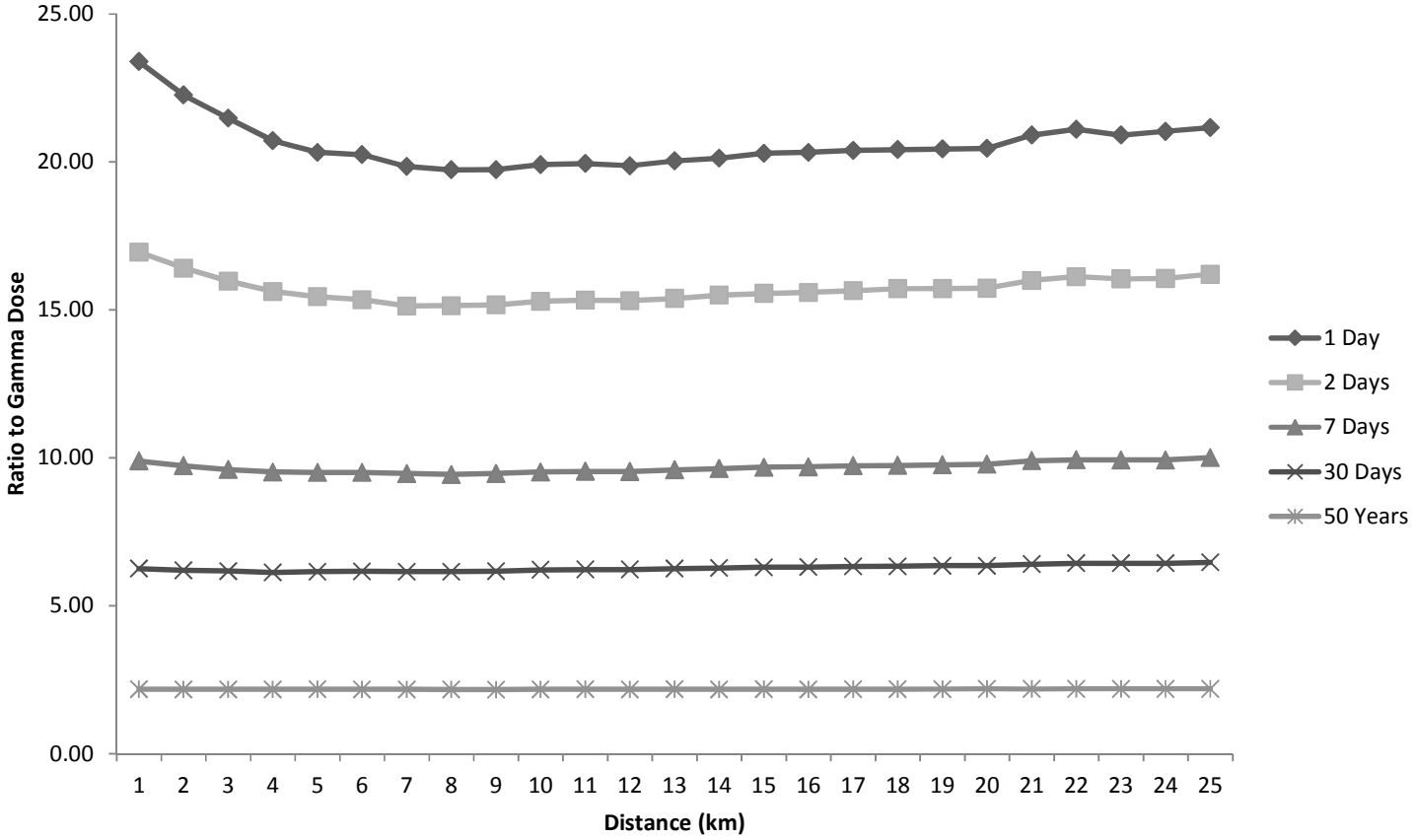

Figure 4.13: D Dispersion Category – Lung Dose / Gamma Dose Ratios

Table 4.17: 1 Day - F Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	67.5	1940	3460	28.74	51.26
0.80 km	32.1	857	1550	26.70	48.29
1.20 km	20.9	532	958	25.45	45.84
2 km	11.1	273	487	24.59	43.87
3 km	6.45	155	273	24.03	42.33
4 km	4.27	103	178	24.12	41.69
5 km	3.13	74.0	126	23.64	40.26
6 km	2.38	56.4	95.3	23.70	40.04
7 km	2.02	47.8	79.8	23.66	39.50
8 km	1.80	42.7	70.7	23.72	39.28
9 km	1.61	38.8	63.4	24.10	39.38
10 km	1.48	35.5	57.5	23.99	38.85
11 km	1.34	32.8	52.5	24.48	39.18
12 km	1.23	30.5	48.3	24.80	39.27
13 km	1.13	28.5	44.7	25.22	39.56
14 km	1.05	26.8	41.6	25.52	39.62
15 km	0.973	25.2	38.8	25.90	39.88
16 km	0.913	23.9	36.4	26.18	39.87
17 km	0.85	22.7	34.2	26.71	40.24
18 km	0.799	21.6	32.2	27.03	40.30
19 km	0.752	20.6	30.4	27.39	40.43
20 km	0.711	19.7	28.8	27.71	40.51
21 km	0.721	19.0	27.4	26.35	38.00
22 km	0.683	18.2	26.1	26.65	38.21
23 km	0.652	17.5	24.9	26.84	38.19

Table 4.18: 2 Days - F Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	94.6	1940	3460	20.51	36.58
0.80 km	44.3	857	1550	19.35	34.99
1.20 km	28.4	532	958	18.73	33.73
2 km	15.0	273	487	18.20	32.47
3 km	8.61	155	273	18.00	31.71
4 km	5.68	103	178	18.13	31.34
5 km	4.13	74.0	126	17.92	30.51
6 km	3.14	56.4	95.3	17.96	30.35
7 km	2.65	47.8	79.8	18.04	30.11
8 km	2.36	42.7	70.7	18.09	29.96
9 km	2.12	38.8	63.4	18.30	29.91
10 km	1.93	35.5	57.5	18.39	29.79
11 km	1.76	32.8	52.5	18.64	29.83
12 km	1.61	30.5	48.3	18.94	30.00
13 km	1.49	28.5	44.7	19.13	30.00
14 km	1.38	26.8	41.6	19.42	30.14
15 km	1.28	25.2	38.8	19.69	30.31
16 km	1.20	23.9	36.4	19.92	30.33
17 km	1.12	22.7	34.2	20.27	30.54
18 km	1.06	21.6	32.2	20.38	30.38
19 km	0.997	20.6	30.4	20.66	30.49
20 km	0.943	19.7	28.8	20.89	30.54
21 km	0.942	19.0	27.4	20.17	29.09
22 km	0.893	18.2	26.1	20.38	29.23
23 km	0.853	17.5	24.9	20.52	29.19

Table 4.19: 7 Days - F Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	165	1940	3460	11.76	20.97
0.80 km	75.8	857	1550	11.31	20.45
1.20 km	48.0	532	958	11.08	19.96
2 km	25.0	273	487	10.92	19.48
3 km	14.3	155	273	10.84	19.09
4 km	9.37	103	178	10.99	19.00
5 km	6.77	74.0	126	10.93	18.61
6 km	5.13	56.4	95.3	10.99	18.58
7 km	4.33	47.8	79.8	11.04	18.43
8 km	3.85	42.7	70.7	11.09	18.36
9 km	3.46	38.8	63.4	11.21	18.32
10 km	3.15	35.5	57.5	11.27	18.25
11 km	2.88	32.8	52.5	11.39	18.23
12 km	2.64	30.5	48.3	11.55	18.30
13 km	2.44	28.5	44.7	11.68	18.32
14 km	2.27	26.8	41.6	11.81	18.33
15 km	2.12	25.2	38.8	11.89	18.30
16 km	1.99	23.9	36.4	12.01	18.29
17 km	1.86	22.7	34.2	12.20	18.39
18 km	1.75	21.6	32.2	12.34	18.40
19 km	1.66	20.6	30.4	12.41	18.31
20 km	1.57	19.7	28.8	12.55	18.34
21 km	1.54	19.0	27.4	12.34	17.79
22 km	1.46	18.2	26.1	12.47	17.88
23 km	1.40	17.5	24.9	12.50	17.79

Table 4.20: 7 Days - F Dispersion Category Gamma Ratios

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	266	1940	3460	7.29	13.01
0.80 km	121	857	1550	7.08	12.81
1.20 km	76.2	532	958	6.98	12.57
2 km	39.4	273	487	6.93	12.36
3 km	22.3	155	273	6.95	12.24
4 km	14.7	103	178	7.01	12.11
5 km	10.6	74.0	126	6.98	11.89
6 km	8.00	56.4	95.3	7.05	11.91
7 km	6.73	47.8	79.8	7.10	11.86
8 km	5.99	42.7	70.7	7.13	11.80
9 km	5.38	38.8	63.4	7.21	11.78
10 km	4.90	35.5	57.5	7.24	11.73
11 km	4.49	32.8	52.5	7.31	11.69
12 km	4.13	30.5	48.3	7.38	11.69
13 km	3.83	28.5	44.7	7.44	11.67
14 km	3.56	26.8	41.6	7.53	11.69
15 km	3.32	25.2	38.8	7.59	11.69
16 km	3.12	23.9	36.4	7.66	11.67
17 km	2.93	22.7	34.2	7.75	11.67
18 km	2.77	21.6	32.2	7.80	11.62
19 km	2.62	20.6	30.4	7.86	11.60
20 km	2.49	19.7	28.8	7.91	11.57
21 km	2.41	19.0	27.4	7.88	11.37
22 km	2.30	18.2	26.1	7.91	11.35
23 km	2.20	17.5	24.9	7.95	11.32

Table 4.21: 50 Years - F Dispersion Category

	Initial Data (Sv)			Ratio to Gamma Dose	
Distance	**Gamma**	**Lung**	**Thyroid**	**Lung**	**Thyroid**
0.35 km	729	1940	3460	2.66	4.75
0.80 km	337	857	1550	2.54	4.60
1.20 km	213	532	958	2.50	4.50
2 km	111	273	487	2.46	4.39
3 km	63.6	155	273	2.44	4.29
4 km	42.1	103	178	2.45	4.23
5 km	30.4	74	126	2.43	4.14
6 km	23.2	56.4	95.3	2.43	4.11
7 km	19.6	47.8	79.8	2.44	4.07
8 km	17.5	42.7	70.7	2.44	4.04
9 km	15.9	38.8	63.4	2.44	3.99
10 km	14.5	35.5	57.5	2.45	3.97
11 km	13.4	32.8	52.5	2.45	3.92
12 km	12.4	30.5	48.3	2.46	3.90
13 km	11.6	28.5	44.7	2.46	3.85
14 km	10.9	26.8	41.6	2.46	3.82
15 km	10.2	25.2	38.8	2.47	3.80
16 km	9.67	23.9	36.4	2.47	3.76
17 km	9.16	22.7	34.2	2.48	3.73
18 km	8.70	21.6	32.2	2.48	3.70
19 km	8.29	20.6	30.4	2.48	3.67
20 km	7.92	19.7	28.8	2.49	3.64
21 km	7.64	19.0	27.4	2.49	3.59
22 km	7.31	18.2	26.1	2.49	3.57
23 km	7.04	17.5	24.9	2.49	3.54

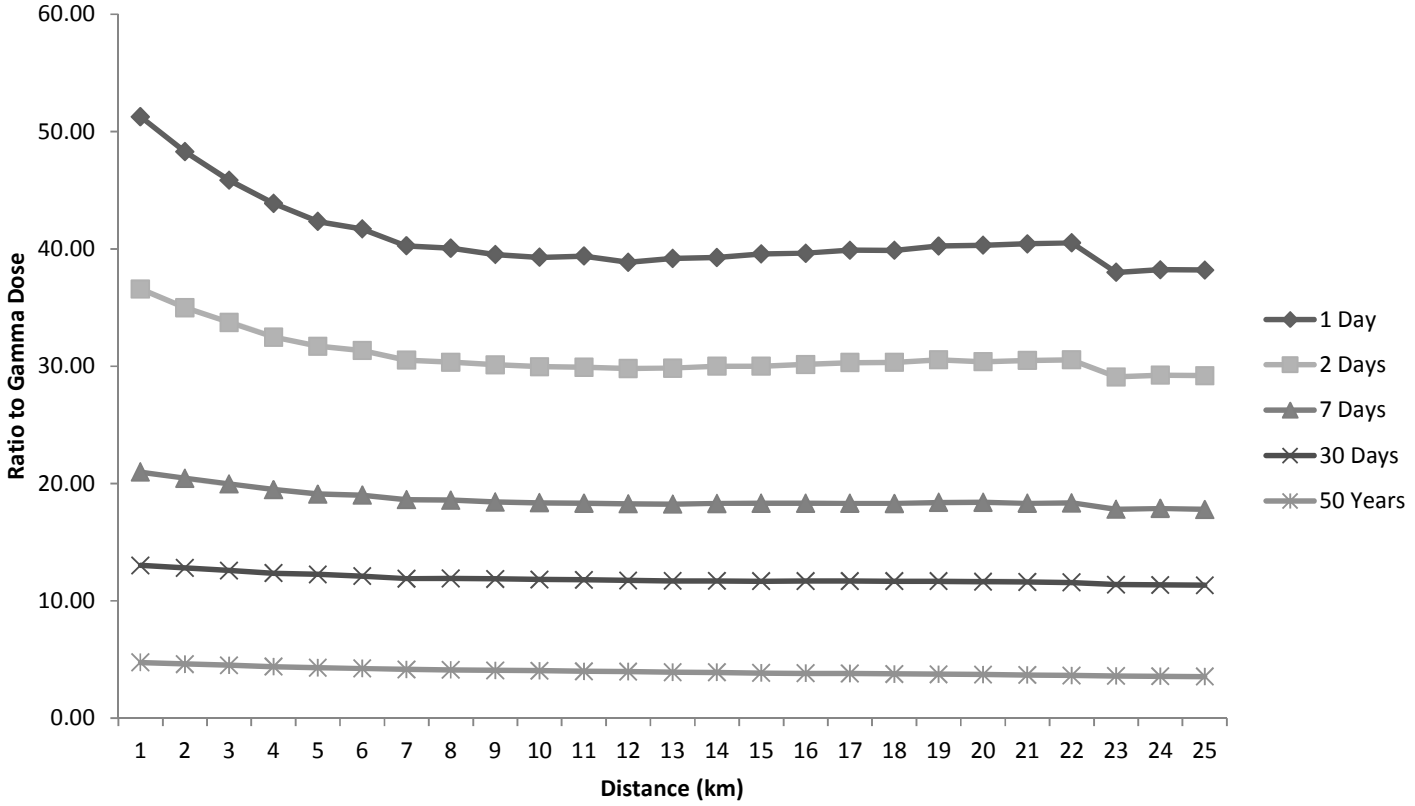

Figure 4.14: F Dispersion Category – Thyroid Dose / Gamma Dose Ratios

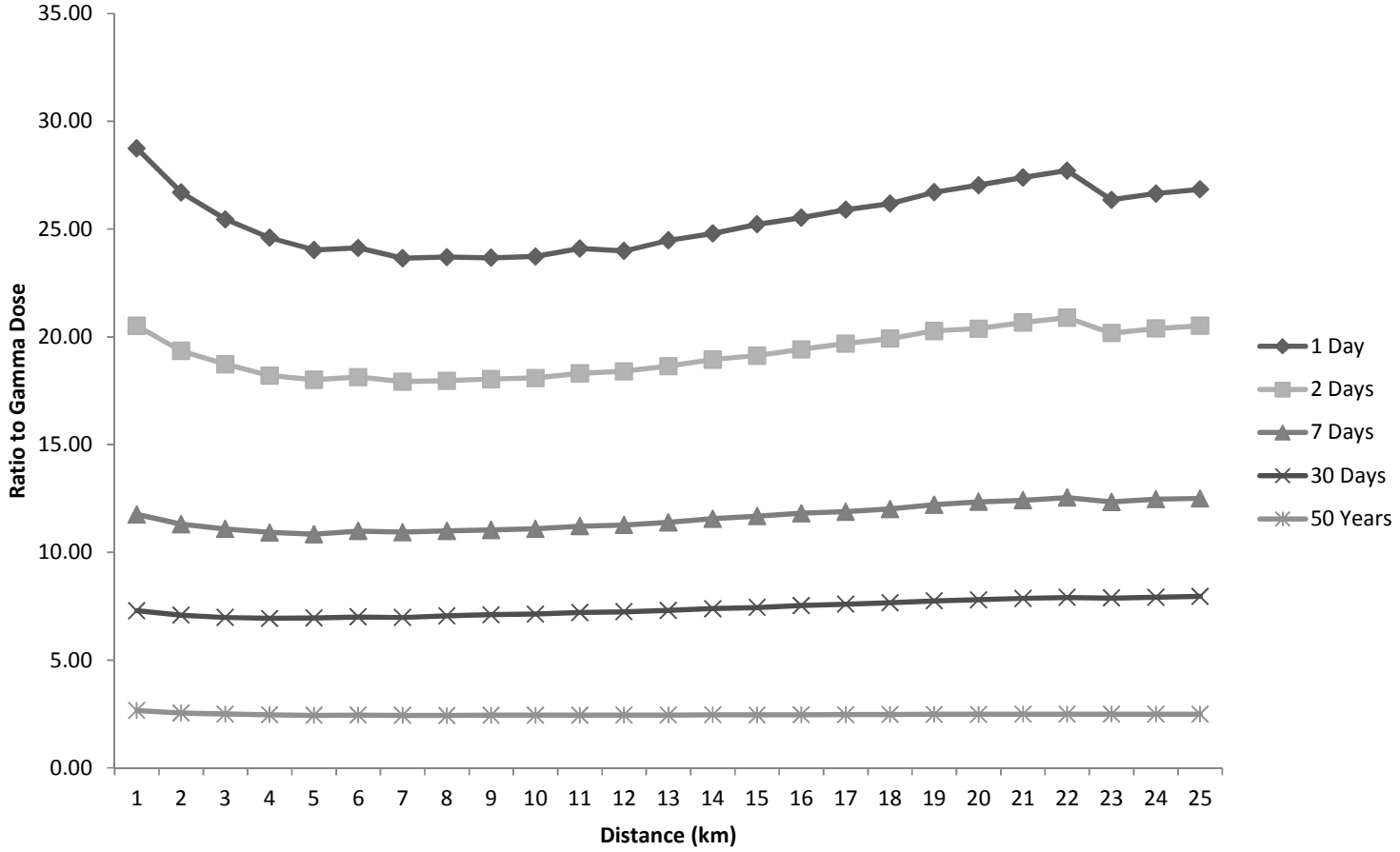

Figure 4.15: F Dispersion Category – Lung / Gamma Dose Ratios

4.2 Analysis

To meet the purpose of this project, the results will require that the limits as recommended by the IAEA as per Table 2.6 and Table 2.7 are not exceeded. The 2 km, 5 km, 10 km and 16 km data will be used. The justification for the use of these distances is as follows, the 2 km distance is the boundary of the plant and as such the beginning of the public zone, the 5 km zone is the end of the PAZ and the 16 km the end of the UPZ. The 10 km distance is an in-house distance to split the UPZ into 5 km to 10 km and then 10 km to 16 km. The reason for the split is that the majority of the public are currently residing between 10 km and 16 km, so as it may be reasonably easy to evacuate up to 10 km, it would be a lot more difficult from 10 km up to 16 km.

Table 4.22: 1 Day Ratios- D Dispersion Category

Distance	Gamma Dose (Sv)	Lung Dose (Sv)	Thyroid Dose (Sv)	Lung/Gamma Ratio	Thyroid/Gamma Ratio
2 km	3.34	69.2	137	20.72	41.02
5 km	0.892	17.7	34.2	19.84	38.34
10 km	0.312	6.20	11.7	19.87	37.50
16 km	0.167	3.41	6.31	20.42	37.78

Table 4.23: 7 Day Ratios- D Dispersion Category

Distance	Gamma Dose (Sv)	Lung Dose (Sv)	Thyroid Dose (Sv)	Lung/Gamma Ratio	Thyroid/Gamma Ratio
2 km	7.27	69.2	137	9.52	18.84
5 km	1.87	17.7	34.2	9.47	18.29
10 km	0.65	6.20	11.7	9.54	18.00
16 km	0.35	3.41	6.31	9.74	18.03

Although the dose rates would decrease reasonably quickly, an average 24 hour dose rate of 1 mSv/h for a gamma dose rate will equate to a 20.72 mSv/h lung dose rate and a 41.02 mSv/h thyroid dose rate at 2 km.

For deterministic effects, a maximum limit for the foetus of 100 mSv is recommended in a 10 hour period. This would mean if a gamma OIL of 10 mSv/h is used for 10 hours, this would result in an equivalent dose of 207.2 mSv for lung and 410.2 mSv thyroid dose. Using the 30 day dose limit for the thyroid, which is 2 Sv, it would be acceptable to use this OIL for a 10 hour exposure.

The limit for stochastic risk is 50 mSv to the thyroid, 100 mSv to the foetus as well as an effective dose of 100 mSv over a period of 7 days. Using the 7 day ratios, 1 mSv/h gamma dose rate will equate to 18.84 mSv/h thyroid dose rate.

In one week (168 hours), an average gamma dose rate is calculated as follows;

$$\text{Average Dose rate} = 100\,\frac{\text{mSv}}{168\ \text{h}} \qquad (4.1)$$

resulting in 0.6 mSv/h average gamma dose rate limit as an OIL for gamma only

Using the calculated ratios, this means that there will be 11.3 mSv/h equivalent dose rate to the thyroid.

Thyroid prophylaxis is recommended to be used at a dose of 50 mSv which will equate to 4.42 hours. In this scenario, it would mean that the OIL can only be used if potassium iodate tablets are issued.

If iodine prophylaxis cannot be implemented, then an OIL of 0.2 mSv/h will need to be used to ensure that the thyroid dose limits are not exceeded.

Using the same approach as above, the calculations have been performed and are shown in Tables 4.24 to Table 4.27. To calculate the equivalent doses to the lung and thyroid, a gamma dose rate of 1 mSv/h is assumed as the starting point to calculate the end results.

The deterministic 10 hour limits as prescribed in Table 2.5 have been used in Table 4.24 and Table 4.26, and the stochastic risk limits prescribed in Table 2.6 were used in Table 4.25 and Table 4.27.

Table 4.24: Recommended OILs with D Dispersion Category using Deterministic Limits

	10 Hours after Release based on 1 Day Ratios			10 hour Limit	Lung Dose Limit	Thyroid Dose Limit	OIL Gamma Dose Rate	Resultant Lung Dose	Resultant Thyroid Dose
Distance	**Gamma Dose Rate (mSv/h)**	**Lung Dose Rate (mSv/h)**	**Thyroid Dose Rate (mSv/h)**	**Based on Foetus Dose (mSv)**	**Based on 30 Day Dose (mSv)**	**Based on 30 Day Dose (mSv)**	**Based on 10 hours (mSv/h)**	**Based on 10 hours (mSv)**	**Based on 10 hours (mSv)**
2 km	1	20.72	41.02	100	30 000	2000	10	207.2	410.2
5 km	1	19.84	38.34	100	30 000	2000	10	198.4	383.4
10 km	1	19.87	37.50	100	30 000	2000	10	198.7	375.0
16 km	1	20.42	37.78	100	30 000	2000	10	204.2	377.8

Table 4.25: Recommended OILs with D Dispersion Category using Stochastic Risk Limits

	7 Days after Release based on 7 Day Ratios			7 Day Limit Gamma	7 Day Thyroid Limit	OIL Gamma Dose Rate	Resultant Lung Dose	Resultant Thyroid Dose	Calculated OIL Gamma Dose Rate
Distance	**Gamma dose Rate (mSv/h)**	**Lung Dose Rate (mSv/h)**	**Thyroid Dose Rate (mSv/h)**	**Based on Foetus Dose (mSv)**	**(mSv)**	**7 Day Limit to Foetus (mSv/h)**	**1 week exposure (mSv)**	**1 week exposure (mSv)**	**No Iodine Protection (mSv)**
2 km	1	9.52	18.84	100	50	0.60	952	1884	0.02
5 km	1	9.47	18.29	100	50	0.60	947	1829	0.02
10 km	1	9.54	18.00	100	50	0.60	954	1800	0.02
16 km	1	9.74	18.03	100	50	0.60	974	1803	0.02

Table 4.26: Recommended OILs with F Dispersion Category using Deterministic Limits

	10 Hours after Release based on 1 Day Ratios			10 hour Limit	Lung Dose Limit	Thyroid Dose Limit	OIL Gamma Dose Rate	Resultant Lung Dose	Resultant Thyroid Dose
Distance	**Gamma mSv/h**	**Lung Dose Rate (mSv/h)**	**Thyroid Dose Rate (mSv/h)**	**Based on Foetus Dose (mSv)**	**Based on 30 Day Dose (mSv)**	**Based on 30 Day Dose (mSv)**	**Based on 10 hours (mSv)**	**Based on 10 hours (mSv)**	**Based on 10 hours (mSv)**
2 km	1	24.59	43.87	100	30 000	2000	10	245.95	438.74
5 km	1	23.64	40.26	100	30 000	2000	10	236.42	402.56
10 km	1	23.99	38.85	100	30 000	2000	10	239.86	388.51
16 km	1	26.18	39.87	100	30 000	2000	10	261.77	398.69

Table 4.27: Recommended OILs with F Dispersion Category using Stochastic Risk Limits

	7 Days after Release based on 7 Day Ratios			7 Day Limit Gamma	7 Day Thyroid Limit	OIL Gamma Dose Rate	Resultant Lung Dose	Resultant Thyroid Dose	Calculated OIL Gamma Dose Rate
Distance	**Gamma Dose Rate mSv/h**	**Lung Dose Rate (mSv/h)**	**Thyroid Dose Rate (mSv/h)**	**Based on Foetus Dose (mSv)**	**(mSv)**	**7 Day Limit to Foetus (mSv)**	**1 week exposure (mSv)**	**1 week exposure (mSv)**	**No Iodine Protection (mSv/h)**
2 km	1	10.92	19.48	100	50	0.60	1092.00	1948.00	0.02
5 km	1	10.93	18.61	100	50	0.60	1093.06	1861.15	0.02
10 km	1	11.27	18.25	100	50	0.60	1126.98	1825.40	0.02
16 km	1	12.01	18.29	100	50	0.60	1201.01	1829.15	0.02

CHAPTER FIVE – GENERAL DISCUSSION AND CONCLUSION

5.1 Discussion

The intent of this research project was met, however, the choice of this accident type was challenging due to the exceptionally high doses obtained. This postulated accident type assumes multiple failures, of multiple safety systems, all at the same time with no filtering or reduction of the release quantities. The code assumes that the fraction of core inventory as stated in Table 3.3 is released to the environment within 24 hours. This type of accident is seen not as a credible accident, but rather an incredible accident.

Based on the deterministic effect limit for the foetus of 100 mSv in 10 hours, this would be the most limiting factor, however, there is also a stochastic limit of 100 mSv in seven days. If it is possible to guarantee that someone would be removed from the area within 10 hours, a dose rate OIL of 10 mSv/h could be used. This in itself, however, is not a practical approach.

The latest recommendations in terms of dose limits, both stochastic and deterministic have resulted in the thyroid dose being the most limiting risk. After the Chernobyl accident it has become evident that thyroid cancers of the population increased when they were exposed as children and teenagers. This realisation (ICRP, 2007) has resulted in substantial differences in limits recommended in earlier years, with the thyroid limits being the most impacted.

Using the results obtained in this project, without thyroid blocking, a maximum gamma dose rate of 0.02 mSv/h can be used as an OIL to ensure that a seven day thyroid equivalent dose of 50 mSv will not be exceeded. In the event of an accident, which has fuel damage, but the containment spray systems and the containment building maintains its integrity as per the design, the resultant iodine impact may be less limiting with respect to public protection.

The results of this project have furthermore highlighted operational concerns related to our current methodology of stockpiling potassium iodate tablets for public distribution in the event of an accident. Currently the plan does not cater for the distribution of potassium iodate to the public unless evacuation of the affected public is ordered. Integrated working groups consisting of Koeberg and government officials are continuously working towards world's best practices and are currently investigating the pre-distribution of potassium iodate tablets to the public for the 5 km (PAZ) area. This however, is not without its own challenges, such as storage containers, storage conditions at homes, transient population etc. This project has highlighted the need that the Koeberg plan should possibly be in place to extend the area of pre-distribution of potassium iodate tablets to include the UPZ (up to 16 km) to achieve the maximum protection of the public.

It is also important to note that this modelling of an accident using PC COSYMA uses ICRP-60 [ICRP, 1991] data for a reference adult. For the purpose of this report the assumption is that only adults are affected in the surrounding areas. This is obviously not completely accurate and will result in ineffective protection of the foetus and children alike. For implementation of OILs for Koeberg this data should be further analysed using age and gender specific dose conversion factors.

After looking at the multitude of possible changing parameters, a method of using OILs for early phase decision making is really challenging for the following reasons:

- Release Duration: The duration of a release will ultimately determine the length of time that someone would be exposed to a passing plume. It would be difficult to predict this in some cases, especially if release pathways were as a result of a large explosion or if there was a large break in the structure with inadequate cooling of the core after the core melt accident. If this is not known, it is extremely challenging to accurately choose applicable OIL. It may, depending on the extent of the damage be appropriate to look at previous accidents such as the Chernobyl as well as the Fukushima incidents to use as a

basis for release duration. It is also important to realise that the results of this project show that the release rate of the various isotopes during the release also varies with time. The assumption of modelling for this project was that the total release inventory was released to the environment within 24 hours.

The ratios of OILs would also change based on this information, making it more important for the decision makers to have solid technical support when projecting using the perceived release time. However, this situation will be extremely difficult to estimate, especially in early stages after the release;

- Accident Type: From the information in this report, there are eight different accident types, which are further subdivided into about seven different plant conditions for each accident type. This will in future be increased to 13 accident types as a result of external events, such as earthquakes and tornados, which are greater than the design criteria for the station. This type of modelling will need to be performed for each accident type and then OIL's calculated for each type. The intention is then to select the appropriate accident and make use of the calculated OIL's. The criteria for selection and tools provided for the decision makers must be easy to understand and implement under stressful conditions;

- Changing Concentration Ratios with Distance: The results show that there is a significant difference in the ratios up to about six km from the plant. This will mean that OILs chosen will differ for the areas nearer the plant, which could once again result in challenges in using the correct OILs;

- Changing Concentration Ratios with Time: This is probably one of the larger variables, which make the use of OIL's in the first week very difficult. This is largely a result of a very wide range of radiological half-lives of a substantial portion of the contaminants. In the first day this changes significantly from hour to hour and continues to drop almost exponentially within the first week. As a result of this, the ratios will also change dramatically, making decision

making difficult, but more importantly not really scientifically justified unless you know exactly what isotopes have been released;

- Topography of the Area: In this research project, no impact has been calculated based on the topography of the area around Koeberg. This type of modelling assumes a plume travelling in a straight line without obstacles such as buildings and mountains. In the event of mountains, forest areas, valleys, etc., the movement of air does not travel in a straight line but will work its way around objects, or against objects such as buildings. This will also result in areas where the build-up of surface contamination will differ tremendously.

All of the above variables will result in different OILs being derived using modelling codes. Depending on the accident type, there is, however, from several hours to several days to result in a release of radioactive material to the public.

After about seven days following a release, it would then be sensible to make use of OILs based on the actual isotopes present in the environment and direct measurements to calculate the actual impact on the population. From this, local government officials can perform interventions such as decontamination of the infrastructure, feeding animals with stored feed, or movement of animals outside of the affected area. This would buy time to ensure that conditions have returned to acceptable levels before the public may return to their properties, if at all.

5.2 Improvements

1. Age and gender specific dose conversion factors need to be used when performing dose assessments for a population to ensure that more accurate modelling is performed.
2. Performing the same runs with several release times without assuming that the whole release fraction for this accident type is released. This would result in decision makers being able to take account of the possible release duration.
3. The runs need to be performed for the full suite of accident types and not only worst case accident types.

4. The accident computer modelling code needs to take into account the topography of the affected area around the site.

5.3 Concluding Remarks

For an incident resulting in a release at a Nuclear Power Station, it is not deemed feasible to wait until radioactivity has been indeed released before implementing protective actions. The use of more effective plant emergency action levels is recommended, such as pre-determined plant sensor readings, or indications of sharp increases in pressures inside the reactor building which would then be utilised for the derivation of urgent protective actions. It is extremely important to be in a position to take actions in the absence of critical plant data or indicators. This requires decision makers who are authorised and trained to predict possible plant failures, such as the Fukushima accident which was due to loss of all power.

During exercises held at Koeberg and the fact that the decision makers do not permanently work in the field of emergency preparedness, it is unreasonable to assume that response personnel will be in position to use a scientific approach to derive OILs. It is however reasonable to expect that a physicist working in this environment should be called upon to assist in this regard, especially if the release time could be excess of a couple of hours.

The OILs should be used with caution during the early phase of an accident, also, as there are usually multiple activities taking place at the same time while an emergency response team is attempting to understand the extent of the incident. One thing is however certain; the incorrect use of OILs during this early phase may result in incorrect derivation of public protective actions.

Emergency plans require specific, easy to use criteria to assist with decision making during the early stages of an incident. Ultimately it seems as if after Fukushima, that an early orderly evacuation of a larger area is preferred, long before the start of a release. This is the driver behind having plant specific

emergency action levels that could initiate the start of an orderly pre-emptive evacuation.

The use of OILs will, however, become extremely useful when assessing areas for possible infrastructure decontamination, resettlement of people or in assessing food activity once conditions have stabilised and the release has terminated. This would typically be anything from about seven days from the termination of a release (IAEA, 1997).

The protection of children and teenagers from radioactive iodine, even at low doses needs to take priority. This will have far reaching impacts into Koeberg's current emergency plan, and significant work will be required to ensure that potassium iodate can and will be distributed before or shortly after the start of a release. Any gamma dose rate greater than 20 μSv/h will necessitate evacuation of the public. Koeberg at least can now anticipate the new challenges prior to the promulgation of the new regulation.

This project has been extremely useful to the author and Koeberg alike. It has highlighted by making use of a scientific analysis, the difficulties faced by a Nuclear Power Operator when attempting to derive plant specific OILs. The expected outcome was to derive OILs for the early phase of an accident, and although the resultant outcome was not achieved, it has highlighted the fact that Koeberg requires more work in deriving more specific Emergency Action Levels to assist in classifying an accident to be in a position to start initiating urgent protective actions long before the start of a release. It was also enlightening to have discovered the technical basis for either pre-distributing potassium iodate or to significantly change the current methodology of stockpiling potassium iodate tablets.

These results express the importance, in terms of public dose to ensure that the release of radioactive material, in the event of an accident is kept to an absolute minimum.

REFERENCES

1. **Eskom** A Report for PSA Analysis [Report]. - Cape Town : Eskom, 2009. - PSA-R-T16-01.
2. **Eskom** Emergency Preparedness and Response Requirements for Eskom Nuclear Installations [Report]. - Cape Town : Eskom, 2012. - 238-53.
3. **Eskom** Re-assessment of Koeberg Emergency Planning Technical Basis and Emergency Planning Zone Sizes [Report]. - Cape Town : Eskom, 2011. - PSA-R-T16-17.
4. **Eskom** The Integrated Koeberg Nuclear Emergency Plan [Report]. - Cape Town : Eskom, 2011. - KAA-811.
5. **Herman Cember Thomas E. Johnson** Introduction to Health Physics [Book]. - United States of America : McGraw-Hill Companies, Inc., 2009. - ISBN 978-0-07-142308-3.
6. **IAEA** Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [Report]. - Vienna : IAEA, 2011. - GSG-2.
7. **IAEA** Development of an Extended Framework for Emergency Response Criteria [Report]. - Vienna : IAEA, 2005. - IAEA-TECDOC-1432.
8. **IAEA** Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident [Report]. - Vienna : IAEA, 1997. - IAEA-TEC DOC-955.
9. **IAEA** Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency [Report]. - Vienna : IAEA, 2003. - EPR-METHOD 2003.
10. **IAEA** Preparedness and Response for a Nuclear or Radiological Emergency [Report]. - Vienna : IAEA, 2002. - GS-R-2.
11. **IAEA** Terminology used in Nuclear Safety and Radiation Protection [Report]. - Vienna : International Atomic Energy Agency, 2007. - IAEA Safety Glossary.
12. **ICRP** Principles for Intervention for Protection of the Public in a Radiological Emergency [Report]. - Oxford : Pergamon Press, 1991. - ICRP Publication 63.
13. **ICRP** The 2007 Recommendations of the International Commission on Radiological Protection [Report]. - Elsevier, 2007. - ICRP 103.

14. **INPO** Special Report on the Accident at the Fukushima Daiichi Nuclear Power Station [Report]. - Atlanta : INPO, 2011. - 11-05.
15. **Jones J.A. Mansfield P.A., Haywood S.M., Hasemann I.,Steinhauer C., Ehrhardt J., Faude D.** PC Cosyma (Version 2), An Accident Consequence Assessment Package for Use on a PC [Report]. - National Radiological Protection Board & Forchungszentrum Karlsruhe GmbH Institut für Neutronenphysik und Reaktortechnik, 1996. - EUR 16239 EN.
16. **Knoll Glenn F.** Radiation Detection and Measurement [Book]. - Hoboken, United States of America : John Wiley & Sons, Inc., 2010. - Fourth Edition.
17. **Minister of Minerals and Energy** National Nuclear Regulator Act, No 47 of 1999. Government Gazette. - 1999. - 20760.
18. **Minister of Provincial and Local Government** Disaster Management Act. No 57, 2002 . Government Gazette. - 2004. - 26228.
19. **National Nuclear Regulator** Emergency Planning Technical Basis for New Nuclear Installation [Report]. - Pretoria : National Nuclear Regulator, 2012. - PP-0015.
20. **RSIC** Modular Code for Performing Standardised Computer Analysis for Licensing Evaluations. [Report]. - Oak Ridge : Oak Ridge National Laboratory, 1993. - Volume 1.2, CCC/545.