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IONISING RADIATION DOSE CALCULATIONS FOR THE RELEASE OF ^{131}I DURING ACCIDENT CONDITIONS AT THE SAFARI-1 MATERIALS TEST REACTOR

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

School of Physics

Johannesburg, 2010

DECLARATION

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



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Willem Bekker

11 October 2010

ABSTRACT

This study demonstrates how the Monte Carlo N-Particle (MCNP) radiation transport code can be used as a contribution to the family of Safety Analysis tools for Research Reactors. Since Research Reactors are used worldwide over a wide range of research and commercial applications it can be justified that effort must be spent to improve on safety analysis. While the advantages of these installations to society are widely recognized, it is still necessary that safety analysis provides the necessary assurance that these installations do not cause an undue risk to society. A Materials Test Reactor (MTR) is used as an example to describe the Safety Analysis steps that need to be done, limited to the reactor and its building. A radioactive inventory in the reactor core is determined. For further evaluation of the effects of the release due to a hypothetical accident in such a reactor, the ^{131}I radioisotope is chosen to demonstrate the capabilities of the MCNP code. A ^{131}I cloud resulting from the release is simulated together with an MCNP model of the reactor building. Personnel in the proximity of the ^{131}I cloud for short times, either due to emergency actions or accidental entrapment, are also modelled. The external photon whole-body doses and β-decay (electron) skin doses are determined. The MCNP code results are also benchmarked against another method. The findings show that the method presented in the study could be used to pre-determine emergency actions that could be incorporated into emergency planning and even into design of a new research reactor. This is substantiated by the conclusion that only the external photon dose could result in unacceptable doses above 500 mSv for short exposure times. The study concludes that the MCNP code can be used effectively for Safety Analyses and leaves the opportunity open to expand the use of the code to other fields related to research reactors.

DEDICATION

Dedicated to my wife

Tersia Bekker

2010

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I would like to thank my wife, Tersia, and my daughter, Petrouw, for their support and patience with me during the time of this study. I also appreciate the support and guidance received from my parents, Hennie and Miemie Bekker, and from my mother in law, Tina Potgieter.

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ABBREVIATIONS

ALARA	As Low as Reasonably Achievable
BDBA	Beyond Design Basis Accident
BOL	Beginning of Life – refers to fuel elements that are freshly loaded
BTE	Boltzmann Transport Equation
BWR	Boiling Water Reactor
EOL	End of Life – refers to fuel elements that has undergone maximum burn-up
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
kW	kilowatt
LEU	Low Enriched Uranium
LOCA	Loss of Coolant Accident
MCNP	Monte Carlo N-Particle Transport code
MTR	Materials Test Reactor
MW	Megawatt
MWDs	Megawatt Days
MWth	Megawatt Thermal – refers to thermal power of a reactor
NECSA	South African Nuclear Energy Corporation
NPP	Nuclear Power Plant
OPAL	Open Pool Australian Light-water reactor
ORNL	Oak Ridge National Laboratory
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
RPV	Reactor Pressure Vessel
RR	Research Reactor
TRIGA	Training, Research, Isotopes, General Atomics – A type of a research reactor

GLOSSARY

Activity	The number of nuclei in a sample undergoing radioactive decay in each second expressed in becquerel (Bq) where 1 Bq is equal to 1 disintegration per second. Also see *radioactivity*.
Bremsstrahlung	Bremsstrahlung is the continuous spectrum of electromagnetic radiation generated when charged particles, e.g. electrons, are slowed down by matter. When a fast moving electron passes close to a nucleus, the strong attractive coulomb force causes the electron to deviate from its original path and loose its energy, causing electromagnetic radiation (X-ray photons) to be emitted.
Cross Section	Denoted by σ. The probability of a specific interaction of radiation with matter with units of barns per atom ($1 \text{ barn} = 10^{-24} \text{ cm}^2$). Used in Fig. 2.11. Also see *linear attenuation coefficient* and *mass attenuation coefficient*.
Deterministic health effects	Health effects that have a severity that is dependent on dose and only starting to occur above threshold level. These is effects are associated with large amounts of radiation received by a person, e.g. in an emergency or accident situation. In contrast, Stochastic health effects occur without a threshold level, has a probability proportional to the dose received and is associated with low doses.
Dose	The quantity of radiation absorbed, specifically in energy per unit mass, expressed in sievert (Sv) or gray (Gy).
Linear attenuation coefficient	Denoted by μ. The probability of a specific interaction of radiation with matter in units of cm^{-1}. To convert from σ to μ; σ in cm^2 / per atom is multiplied with the number of atoms (in atoms / cm^3) to get the linear attenuation coefficient (μ) for a given material. Also see *cross section* and *mass attenuation coefficient*.
Mass attenuation coefficient	Denoted by μ/ρ or μ_m. Obtained by dividing μ with the mass density (ρ) of the absorbing medium to get μ_m in cm^2/g. This is a convenient quantity to use since it is essentially independent of the density and physical state of the absorbing medium. Used in Fig. 2.12. Also see *cross section* and *linear attenuation coefficient*.
Negative temperature coefficient	An attribute of reactor design that causes the fission reaction to slow down as the temperature increases and stabilizes reactor operations.
Radioactivity	The spontaneous emission of radiation from a nucleus decaying to a lower energy state.
Specific Activity	The Activity (in Bq) per unit mass.

CHAPTER 1

INTRODUCTION

Research reactors are used effectively worldwide for a wide range of research and commercial applications. Careful safety analysis has to be performed to provide assurance that these installations are operated safely, and thereby not cause an undue risk to society. Modern computational techniques are used in safety analysis. For this study, a specific computational technique, the Monte Carlo N-Particle (MCNP) transport code, was evaluated and used for effective contribution to the ongoing development of safety analysis.

In this chapter, an overview of research reactors and their uses will be given and, thereafter, more details on the need for safety analysis will be provided. Thereafter, the development of modern computational techniques used in safety analysis will be discussed and the specific field where the MCNP code will be used and evaluated, will be defined. This will be followed by a brief discussion of the methodology followed.

1.1 Research Reactors

An overview on research reactors is given in this Section by elaborating on the different types thereof and their purposes and uses.

1.1.1 Definition and uses

Research reactors form part of the family of nuclear reactors, but are normally not used for the production of electricity like nuclear power plants. In nuclear power plants the energy produced by the nuclear reaction is used for generation of electricity. In research reactors the emphasis falls on utilising the neutrons produced for useful applications. Whilst some of the neutrons produced in the nuclear reaction are used to sustain the nuclear reaction taking place in the reactor core, other neutrons are used in or outside the reactor core to e.g. bring into effect the desired changes in isotopic composition of the materials exposed to the neutron flux.

The different ways and timeframes in which materials can be exposed to the neutron flux from the reactor core result in a variety of applications. Therefore, research reactors have a wide range of uses. These include analysis and testing of materials, and production of radioisotopes. Their capabilities are applied in many fields, within the nuclear industry as well as in fusion research, environmental science, advanced materials development, drug design and nuclear medicine. Since such scientific facilities are expensive, research reactors tend to be multi-purpose, and many have been operating for more than 30 years. As a result, the variety of designs and the various ways in which each type can be utilised increase the complexity of safety analysis. This will be discussed later in more detail.

In general, the purpose of nuclear research reactors is not for energy generation; the maximum power generated generally never exceeds 100 MW. They are commonly devoted to the generation of neutrons for different scientific and other purposes of societal benefit. However, high power densities are involved in the core and specific features are necessary to ensure safe utilisation of these installations [TEW08].

1.1.2 Categories and quantities

A particular research reactor can be used for many purposes and a great variety of types and designs do exist. In contrast, the majority (80%) of the world's nuclear power plants' designs are of only two similar types, Pressurised Water Reactors (PWRs) and Boiling Water Reactors (BWRs).

Research reactors are broadly classified into a few categories, according to their type of design and use. The first category is reactors that are used for research, and of which some are also used for commercial applications, namely the production of radioisotopes. About 160 reactors fall into this category. Other reactors are called Test Reactors (23), Training Facilities (37) and Prototypes (2). One research reactor produces electricity. A special type of research reactor, called a "critical assembly", also exists, of which there are 60 worldwide.

Russia has most of the research reactors (62), followed by the USA (54), Japan (18), France (15), Germany (14) and China (13). Many small and developing countries also

have research reactors, including Bangladesh, Algeria, Colombia, Ghana, Jamaica, Libya, Thailand and Vietnam. About 20 more research reactors are planned or under construction, and 361 have been shut down or decommissioned, about half of these in the USA. Many research reactors were built in the 1960s and 1970s. The peak number of reactors that operated was in 1975, with 373 in 55 countries. Thereafter an increasing number began to be shut down.

1.1.3 Research reactor types and designs

The operation of all the types of reactors will not be explained. However, to understand the model developed for this study, the operation of a few types are explained below.

Tank-in-pool type reactor

The tank-in-pool type research reactor is a vessel situated in a large pool of water. Inside the reactor vessel is the reactor core, consisting of a cluster of fuel elements. The pool water supplies radiation shielding on the top of the reactor tank, while the thick concrete walls of the pool provides shielding to the sides of the reactor. High-density concrete is normally used.

Shown in Fig. 1.1 is a simplified representation of a tank-in-pool type research reactor. The reactor vessel is also called a tank, hence the name "tank-in-pool type".

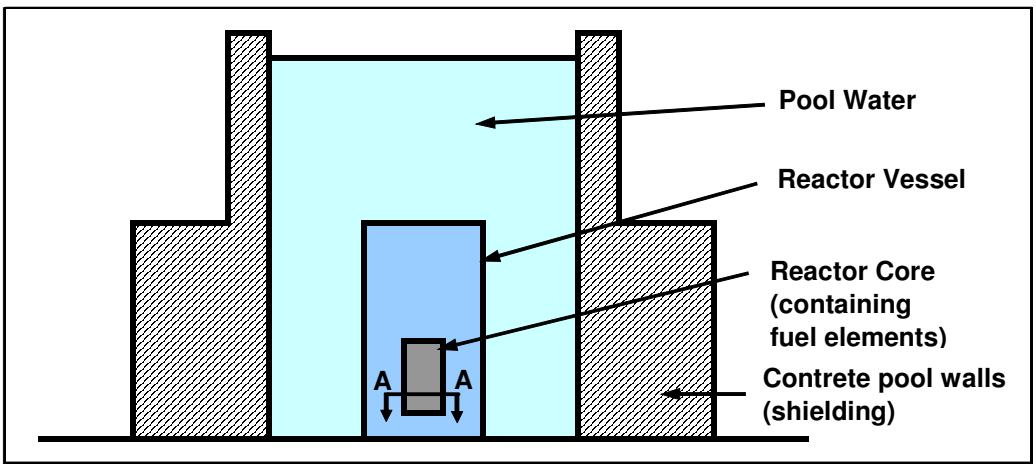

Figure 1.1: Simplified representation of a tank-in-pool type reactor.

Among the fuel elements are control rods that control the nuclear fission chain reaction. A high neutron flux exists in the core and therefore some in-core devices that utilise this high flux, are put in empty channels among the fuel elements, for example materials to undergo isotopic composition change. Both graphite and beryllium is commonly used for neutron reflectors. Shown in Fig. 1.2 is a representation of a cut through the SAFARI-1 reactor core, as an example, as indicated by Section A-A in Fig. 1.1.

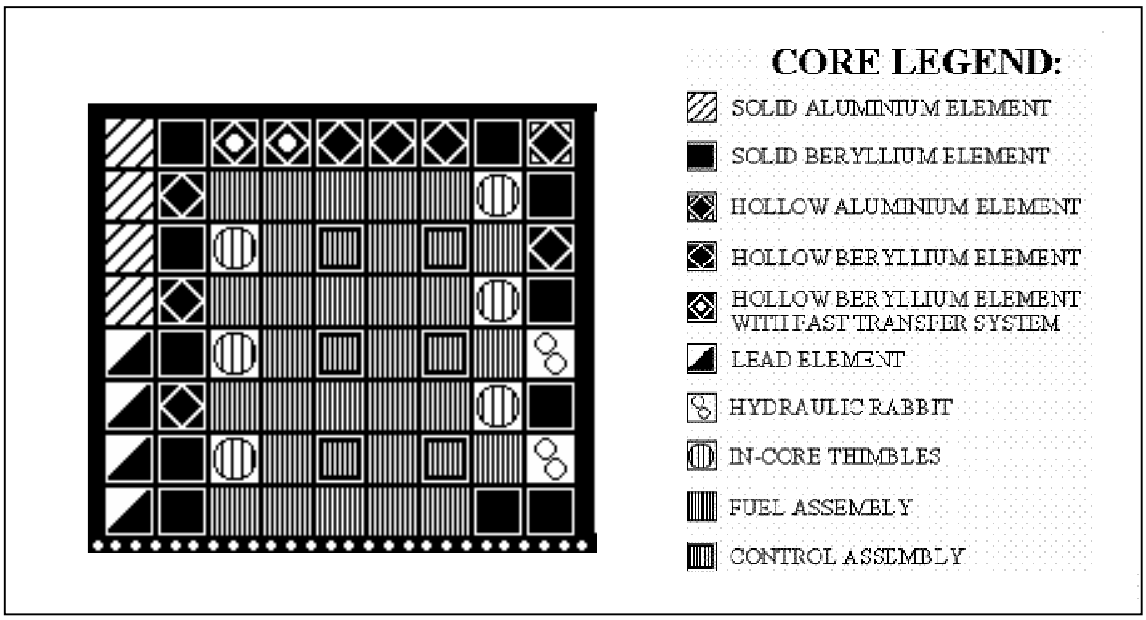

Figure 1.2: Section through the SAFARI-1 reactor core at position A-A of Fig. 1.1, showing the core components.

Each fuel element comprises several, typically between 18 and 21, curved or flat aluminium-clad fuel plates, and is shown schematically in Fig. 1.3. The fuel element assembly is rectangular and is placed vertically in the core.

In addition to the shielding function provided by the water, it also both moderates and cools the reactor. The water flowing through the reactor core goes through the fuel elements in order to ensure that heat generated in the fuel element is taken away. Efficient flow is ensured by means of a primary water loop that pumps water through the core.

Another way of utilising the neutrons is to access the core from outside the biological shield (the concrete wall of the pool). Pipes that penetrate through the wall of the pool

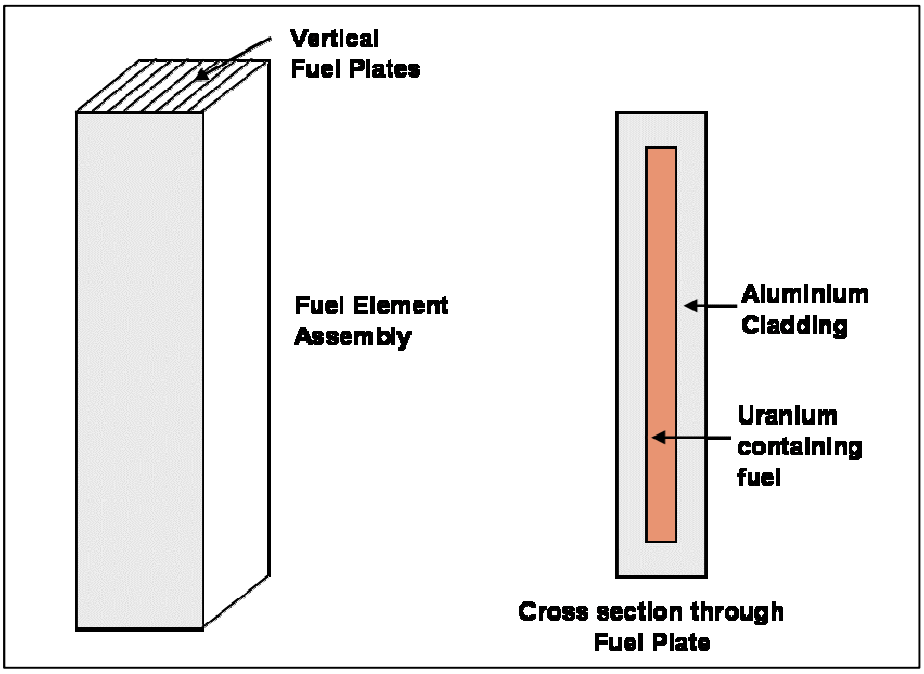

Figure 1.3: Schematic representation of vertical fuel element assembly.

provide access to the core for experimental devices. These pipes are called beam tubes. Collimators are fitted within the beam tubes and direct the neutrons to experimental devices outside the pool wall. When the beam tubes are not in use they are filled with water to provide shielding to the outside of the pool. Worldwide, there are about 32 research reactors of the tank-in-pool type design.

Pool type reactor

The pool type reactor is very similar to the tank-in-pool type design. The major difference is that the reactor core is a cluster of fuel elements sitting in the reactor pool, but without a reactor vessel. The main reason is that less cooling is needed and the cooling provided by the water in the pool is sufficient. This is possible because these reactors run at lower power, typically in the region of kWs and not in tens of MWs as for tank-in-pool type reactors. About 67 research reactors worldwide are of the pool type design.

TRIGA reactor

In another common design, the TRIGA, the reactor core consists of 60-100 cylindrical fuel elements which are about 36 mm in diameter. The fuel elements are enclosed in aluminium cladding and contain a mixture of uranium fuel and zirconium hydride

moderator. The reactor sits in a pool of water and generally uses graphite or beryllium as a reflector. This kind of reactor can safely be pulsed to very high power levels (e.g. 25 000 MW) for fractions of a second. Its fuel gives the TRIGA type reactor a very strong negative temperature coefficient, and the rapid increase in power is quickly cut short by a negative reactivity effect of the hydride moderator. About 40 research reactors worldwide are of this type of design.

Other research reactor designs

Other research reactor designs are moderated by heavy water (12 units worldwide) or graphite. A few are fast reactors, which require no moderator and can use a mixture of uranium and plutonium as fuel. Homogenous type reactors have a core comprising a solution of uranium salts as a liquid, contained in a tank about 300 mm in diameter. The simple design made them popular early on, but in 2009 only five remained in operation.

1.1.4 Materials Test Reactor (MTR) used in this study

The reactor type investigated in this study is a tank-in-pool Materials Test Reactor (MTR). MTRs were originally built to test the behaviour under irradiation of materials to be used in Nuclear Power Plants (NPPs). The flux in a research reactor is about two orders of magnitude higher than in NPPs and, therefore, accelerated tests can be done on materials to be used for construction in NPPs.

The example South African Nuclear Energy Corporation (NECSA) MTR investigated is a 20 MW research reactor and was built in the 1960s following the so-called “Oak Ridge design”. Similar reactors were built more or less at the same time in the USA at Oak Ridge, in Sweden, the Netherlands and South Africa. A modern type of reactor that is very similar is the OPAL reactor in Australia.

1.2 Need for Safety Analysis

In the introductory paragraph it was mentioned, and through Section 1.1 explained, that research reactors with all their uses are highly valued installations and hold a lot of

advantage to society. It is, therefore, worth considering that such installations continue operating or to even allow new research reactors to be built. Before the continued operation of research reactors or construction of new ones is going to be coupled to Safety Analysis, the Fundamental Safety Principles of the IAEA [IAE06] and the "system of radiological protection" will be discussed.

1.2.1 Fundamental Safety Principles

When the Fundamental Safety Principles no. SF-1 [IAE06] was approved for publication by the IAEA's Board of Governors in September 2006, the IAEA has succeeded in formulating for the first time a unified philosophy of nuclear safety and protection against ionizing radiation with a broad international consensus.

The development of the Fundamental Safety Principles was initiated in June 1995 by the IAEA Board with the aim of combining three fundamental safety texts viz. "the safety of nuclear installations", the "safety of radioactive waste management" and "protection and the safety of radiation sources". The three different areas were considered and consolidated into a coherent and consistent set of ten principles. This publication is the highest in the three tier hierarchy of IAEA safety standards which comprises in decreasing order Safety Fundamentals, Safety Requirements and Safety Guides.

The IAEA is required by its statute to promote international cooperation. This is motivated by the fact that radiation risks may transcend national borders, and that international cooperation serves to promote and enhance safety globally. International cooperation is fostered by promoting exchanging of experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

The definition of safety of facilities and activities

The Fundamental Safety Principles publication defines "safety" as the protection of people and the environment against radiation risks, and the safety of facilities and activities that give rise to radiation risks. Safety of facilities and activities includes the safety of nuclear installations, radiation safety, the safety of radioactive waste

management and safety in the transport of radioactive material. It includes both radiation risks due to normal operating conditions and to accident conditions. Protection of people, as stated above, includes protection of an occupationally exposed worker and the public.

The safety principles are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilised for peaceful purposes, and to protective actions to reduce existing radiation risks.

Fundamental safety objective and the ten safety principles

The fundamental safety objective of the IAEA reflected in this publication is to protect people and the environment from harmful effects of ionizing radiation without unduly limiting the operation of facilities or the conduct of activities that give rise to radiation risks. In order to achieve this safety objective, ten safety principles have been formulated, and they are as follows;

Principle 1: Responsibility for safety

The first fundamental principle states that the prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks. The regulatory body, established according to Principle 2, grants an authorization to an organization known as the licensee. The licensee retains the prime responsibility for safety throughout the lifetime of facilities and activities and plays an important role in the implementation of Principles 3 to 10 by means of aspects as leadership and the implementation of effective safety management systems.

Principle 2: Role of government:

The second principle states that an effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained. A proper legal and governmental framework provides for the clear assignment of responsibilities to safety. The government is responsible to adopt its national legal system to make provision for legislation related to safety that

fulfils the objectives of the fundamental principles. Programmes must be undertaken to achieve the safety of facilities and activities.

Principle 3: Leadership and management for safety

Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks. Leadership in safety matters has to be demonstrated at the highest levels in an organization. This has to be achieved by an effective management system which has to integrate all elements of management including safety, quality, security and human performance. The management system also has to ensure the promotion of a safety culture.

Principle 4: Justification of facilities and activities

"Justification" is also the first of three principles that makes up the "system of radiological protection". These principles, the other two being "Optimization" and Dose Limitation", correspond in broad terms to "Principle 5" and "Principle 6", respectively. Therefore, these three principles will be discussed in more detail in the next section (Section 1.2.2).

In summary, Justification requires that facilities and activities that give rise to radiation risks must yield an overall benefit which means that the benefits that nuclear installations or other facilities and activities yield must outweigh the radiation risks to which they give rise.

Principle 5: Optimization of protection

Protection must be optimized to provide the highest level of safety that can reasonably be achieved. The safety measures that are applied to facilities and activities that give rise to radiation risks are considered optimized if they provide the highest level of safety that can reasonably be achieved throughout the lifetime of the facility or activity, without unduly limiting its utilisation. To determine whether radiation risks are As Low as Reasonably Achievable (ALARA), all such risks, whether arising from normal operations or from abnormal or accident conditions, must be assessed prior to and, periodically, during the lifetime of

facilities and activities. As stated above, this principle will also be discussed further in Section 1.2.2.

Principle 6: Limitation of risks to individuals

Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk or harm. The application of the principles of Justification and Optimization of protection alone do not guarantee this. Consequently, doses and radiation risks must be controlled within specified limits. The principle of "Dose Limitation" is discussed in Section 1.2.2.

Principle 7: Protection of present and future generations

People and the environment, present and future, must be protected against radiation risks. The effects of radiation exposure on human health (workers and public) are better understood than these of radiation exposure to the environment and measures must be taken to limit adverse radiation exposure to the environment.

Radiation risks may also transcend national borders and may persist for long periods of time. The possible consequences, now and in the future, of current actions have to be taken into account in judging the adequacy of measures to control radiation risks. As an example, the generation of radioactive waste must be kept to the minimum and it must be managed in such a way as to avoid imposing an undue burden on future generations.

Principle 8: Prevention of accidents

The most harmful consequences arising from facilities and activities have come from the loss of control over nuclear reactor cores (e.g. Chernobyl [WIL08]), nuclear chain reactions (inadvertent criticality) and overexposure due to the misuse of a radioactive source or other source of radiation. Therefore, all practical efforts must be made to prevent and mitigate nuclear or radiation accidents and measures have to be taken to ensure that the likelihood of an accident having harmful consequences is extremely low.

The primary means of preventing and mitigating the consequences of accidents is through a concept called "defence in depth". Defence in depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. The concept of defence in depth and how it is applied play an important role in the effectiveness of the protection measures mentioned in Section 2.4.4.

Principle 9: Emergency preparedness and response

This principle requires that arrangements must be made for emergency preparedness and response for nuclear or radiation accidents. Prior assessment for the need of defined emergency actions must be done for emergency preparedness. The primary goals of emergency preparedness are:

- To ensure that arrangements are in place for an effective response to a nuclear or radiation emergency.
- To ensure that, for reasonably foreseeable incidents, radiation risks would be minor,
- For any accidents that do occur, to take practical measures to mitigate any consequences for human life and health and the environment.

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

Radiation risks may arise in situations other than in facilities and activities that are in compliance with regulatory control. In such situations, if the radiation risks are relatively high, consideration has to be given to whether protective actions can reasonably be taken to reduce radiation exposures and to remediate adverse conditions. In all of these cases, the protective actions considered each have some foreseeable economic, social and, possibly, environmental costs and may entail some radiation risks (e.g. to workers carrying out these protective actions) and must be carefully considered.

Conclusion on the discussion of Fundamental Safety Principles

The objective of the Fundamental Safety Principles publication and its ten principles provides an overview of the important concepts jointly related to nuclear safety and radiation protection. The need for Safety Analysis and the principles of applying it are reflected in the ten principles. An earlier scheme, but closely related to the above principles, the "system of radiological protection" is going to be discussed next, in view of further understanding the need for Safety Analysis.

1.2.2 System of Radiological Protection

A system of radiological protection has been defined by the International Commission on Radiological Protection (ICRP), an international organization that brings out various publications on radiation protection[1]. The "System of radiological protection" involves three fundamental principles; justification, optimization and dose limitation.

Justification

Although the newest ICRP publication in this series, ICRP 103 [ICR07], replaced the previous document, ICRP 60 [ICR90], some aspects of relevance in the latter document will first be mentioned and explained, especially in respect of justification.

ICRP 60 [ICR90] stated that "due to some human activities, an increase in overall exposure to radiation will occur". These human activities were defined as "Practices". For proposed and continued Practices, the ICRP stated that "no Practice should be introduced, unless it produces sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes", i.e. "it should aim to do more good than harm". This principle was called "Justification of a practice". ICRP 60 [ICR90] also defined "intervention" which is an action to reduce exposure in existing situations.

In ICRP 103 [ICR07], the ICRP did not indicate that any fundamental change should be introduced to the system of radiological protection as defined in ICRP 60 [ICR90]. It

[1] For the purpose of this study, and also internationally accepted, "radiation protection" and "radiological protection" are the same. Therefore, with introducing the system of radiation protection the words are used interchangeably. While ICRP refers to "radiological protection", this study refers to "radiation protection" in most cases.

should be noted though that the terms “Practices” and “intervention” have been replaced by “controllable exposure situations”, which are now defined as “planned”, “emergency” and “existing” exposure situations. Therefore, strictly speaking, it is not correct any more to speak of “Justification of a practice”, but “justification” still applies and it could be mentioned in terms of keeping an existing research reactor running. ICRP 103 [ICR07] defines “justification” as “any decision that alters the radiological exposure situation should do more good than harm”. “Justification” is frequently decided on at a political level. In the Fundamental Safety Principles [IAE06] it is stated that “in many cases, decisions relating to benefit and risk are taken at the highest levels of government, such as a decision by a State to embark on a nuclear power programme. In other cases, the regulatory body may determine whether proposed facilities and activities are justified”.

Optimization

ICRP 103 [ICR07] defines the “principle of optimization of protection” as ”the likelihood of incurring exposures, the number of people exposed, and the magnitude of their individual doses should be kept as low as reasonably achievable, taking into account economic and societal factors”. It states that “the level of protection should be the best under the prevailing circumstances, maximizing the margin of benefit over harm”. This normally includes careful planning of activities related to construction, operation and maintenance that must be undergone in the phases of design, operation and decommissioning of a nuclear facility. It also sometimes includes an iterative process where the benefits from further reducing the exposure is each time carefully weighed against the extra costs it introduces. Further reduction of exposure could be done by e.g. introducing a further design feature or by altering an operational procedure.

Dose Limitation

The principle of application of dose limits is defined as “the total dose to any individual from regulated sources in planned exposure situations, other than medical exposure of patients, should not exceed the appropriate limits recommended by the ICRP”. Dose limits are defined by the regulatory body of a country and are normally in line with recommendations of the ICRP. Dose limits and risk limits are defined for workers and members of the public. Dose limits normally apply to planned normal exposure

situations. Risk limits apply to accident conditions of a nuclear facility. Safety analysis is primary concerned with demonstrating that dose and risk limits will not be exceeded.

1.2.3 How Safety Analysis uses the three fundamental principles, in particular for research reactors

In the following Sections, the three fundamental principles of the "System of radiological protection" are again discussed, but in the context of research reactors. Any existing or new action, called a "Practice" in ICRP60 [ICR90] terms, must be authorized by the regulatory body of a country.

The action, in this context to be authorized by the nuclear regulator, will be to keep an existing research reactor running or to allow the construction of a new research reactor. The authorization of an action needs some basis and justification, optimization and dose limitation are used in the process to provide such a basis.

Justification (Research Reactor)

In the introductory paragraph of Section 1.2 it was stated that it is worth considering research reactors to continue operating or even allowing new research reactors to be built as some of them are highly valued installations that hold advantage to society. This already provides a good argument to say that the net benefit will be large on the one side of the scale for "Justification" according to radiation protection principles. As already explained above, the net benefit of a planned or existing exposure situation must outweigh the risks it imposes. Thus, on the other side of the scale, it must now be proven that the potential detriment to society is not larger than the benefit.

This could not be proven for many research reactors and, therefore, many of them have been shut down since the 1970s, as mentioned in Section 1.1.2. These reactors were originally built for specific research and training activities and because ongoing sustainable development of commercial applications of the reactors could not be carried out the benefit from keeping them operational decreased. On the other side, because of ageing considerations, the risk increased. One reactor that was shut down had a leaking pool which rendered the installation to become a liability rather than an asset.

However, focussing on reactors that have been modified or built to also house commercial applications a careful method must be used to quantify the risk. The method used, which is well known in the nuclear industry, is Safety Analysis.

Safety Analysis is used to quantify the detriment, i.e. the risk. If the net benefit clearly outweighs the risk, e.g. radioisotopes are produced to be used for medical diagnosis and treatment of patients, the practice of keeping a research reactor running can be justified. Thus, the first of the three radiation protection principles, Justification, is satisfied.

Optimization (Research Reactor)

In the Safety Analysis process the second principle, Optimization, is also catered for. Installations are evaluated during design and operation to assure that doses are kept As Low as Reasonably Achievable (ALARA).

Dose Limitation (Research Reactor)

Finally, the Safety Analysis must ensure that during normal operating conditions the effective dose to the worker and to the public is kept below the dose limit of 20 mSv and 1 mSv, respectively. Also, during anticipated accident conditions the potential doses must be kept below acceptable limits. If the probability of a certain accident type is expected to be very low, e.g. an accident caused by a large earthquake, the risk is expressed in terms of risk criteria and not in terms of radiation dose.

The nuclear industry is very strictly regulated in respect of nuclear safety. To authorize a Practice both the dose limitation and risk criteria, collectively called Regulatory Criteria, must be satisfied. Nuclear safety analyses are performed for new installations and periodically on existing facilities.

Other aspects of safety analysis

Safety Analysis is not only performed to demonstrate conformance to regulatory criteria, but also lately to enable risk-informed decision making.

To put Safety Analysis of research reactors into context in comparison with other nuclear installations, the following brief explanation is also given. As a result of a long operating history (in reactor-years) of Nuclear Power Plants (NPPs), and the use of

similar designs world-wide, a very consistent approach to Safety Analysis of these has been developed. This was formalized in the early 1970s and is captured in reports like WASH 1400 [WAS75]. However, for research reactors, because of the very wide range of designs and different powers generally ranging from kWs to tens of MWs, it has been more difficult to develop standardized approaches. For the sake of completeness it is stated here that various Safety Analysis and assessment techniques for other (non-reactor) nuclear installations exist, but that is beyond the scope of this study. Whereas the scope of the present study will not focus on NPPs, some of the concepts used are taken from NPP Safety Analysis methodology. It must also be said that modern Safety Analysis approaches for research reactors are largely derived from NPP techniques (but tailored for research reactors).

The further development of Safety Analysis techniques for research reactors is supported by past incidents that occurred in these installations. For example, at least three accidents involving fatalities have occurred at research reactors facilities. It is, therefore, worth expending effort in developing additional tools for the Safety Analysis of research reactors.

1.3 Rationale of Safety Analysis using modern computational techniques

Modern computational techniques are used in support of safety analysis. Advanced safety analysis and design optimizations that were not possible a few years ago can now be performed [TEW08]. This has resulted in a gradual switch to a new generation of computational tools in order to simulate more realistically complex phenomena resulting from accident progression. Generally, the challenge today is to revisit safety design features of the existing research reactors in order to verify that the regulatory criteria are still met.

Also, from time to time, regulatory criteria changes. This causes the need to revisit the existing safety analysis that was done for a research reactor. The improvement of the safety analysis, usually also using the newest tools, sometimes requires plant modification. A second factor that introduces the need for plant modification is when

new equipment has to be introduced from recent advancement of new technologies, both for better utilisation and better performance of the research reactor. To better understand the contribution of modern computational techniques to the development of safety analysis for research reactors, some light is shed on the maturity of this process for NPPs.

Initially, computational techniques provided conservative answers. In contrast to conservative methods, best estimate methods supported by uncertainty analysis techniques are now well developed for NPPs through international expertise. To support these standardized techniques, a comprehensive experimental database exists for the safety technology of NPPs. The importance of transferring NPP safety technology tools and methods to research reactor safety technology has been noted in recent international activities led by the IAEA, e.g. IAEA Safety Report 53 [IAE08a] and IAEA Safety Report 55 [IAE08b]. However, the ranges of parameters of interest to research reactors are different from those for NPPs. This is in particular applicable for fuel composition, system pressure, materials used and overall system geometric configuration.

In order to perform proper safety analysis, so-called initiating events have to be defined. Initiating events are events that can trigger an accident. Once an initiating event has occurred, it does not automatically follow that an accident will occur and the probability of an accident occurring depends on the effectiveness of the plant's systems of protection. The standardized designs and known operating history NPPs enabled the generation of standard lists of initiating events that could have been agreed upon and published. In contrast, for research reactors, the large variety of type, designs and operating modes prevented so far the achievement of systematic and detailed lists of initiating events based upon qualified studies. However, bounding and generalized lists of initiating events are available from IAEA documents and can be considered for further studies in this area.

Current efforts are well in progress to apply modern techniques to research reactors as well, but especially in the field of neutronic and thermal-hydraulic calculations [TEW08]. A substantial part of safety analysis is improved by improving these methods. However, these calculations lie in the field of "plant response" to an accident.

For NPPs, the plant's response to an initiating event that could lead to an accident is called Level 1 safety analysis or Level 1 Probabilistic Safety Analysis (PSA).

Another field of study that thus needs improvement for research reactors is the progression of accidents where the radioactive inventory of the reactor core is released from the core through the building, filters, stack and eventually leaves the reactor building. The radioactive inventory results from the radioisotopes produced in the nuclear reactor core. This field is also well advanced for NPPs and is called Level 2 PSA for analysis within the building and Level 3 PSA when the radioactive isotopes leave the building and disperse into the atmosphere. The former is also called "building response".

The challenge is thus to enhance techniques in the Level 2 and 3 space. As will be seen in Section 2.5.2 "Availability of release fractions for research reactors and the selection thereof", for research reactors, data are relatively scarce that quantifies the amount of radioactive inventory released from the core, the distribution thereof within the building and the eventual release to the atmosphere. The hazard posed by a NPP to the public and environment is far greater than that of a research reactor because of the much larger radioactive inventory, higher temperatures and higher water pressures. Therefore, Level 3 analysis is much more important for a NPP. For research reactors, it is rather important to analyse in the Level 2 space. This is motivated by the larger risk to personnel within the facility.

This study, therefore, aims to contribute to the knowledge of using modern computational methods in support of Level 2 Safety Analysis of research reactors. Modern computational techniques are available and it has been decided to test the application of a well known radiation transport code, the Monte Carlo N-Particle (MCNP) code [SHU07]. Although this code is well known and used in many fields, it is little explored in the particular application mentioned.

With this study it is thus hoped not to revise the complete safety analysis process but to contribute to the family of techniques used to enhance the ways in which safety analyses are done, to make it more effective and to improve the results.

1.4 Hypothesis and methodology

Fundamental hypothesis: The MCNP code can be used to provide good answers in the application of accident studies for research reactors, in particular where workers will be potentially exposed to external radiation because of the postulated accident. Note that "accident studies" is part of the process of safety analysis. However, for the present study, the terms "accident studies" and "safety analysis" are used interchangeably.

In order to provide a demonstration of how the MCNP can be used for accident studies of an MTR reactor and how it fit into the process of safety analysis, a representative MRT reactor is described with an example of how the MCNP code is applied. An MCNP model was developed based on a real example MTR reactor, that includes the dimensions of the reactor hall, the radioactive inventory etc. The fraction of the radioactive inventory released, and the dimensions and materials of construction present in the reactor hall also provided input into the MCNP model. Although other radioisotopes will also be released from the reactor core during accident conditions, the effect of iodine, in particular the ^{131}I isotope, was studied.

Some assumptions were made during the development of this model. Although external dose rates resulting from ^{131}I were calculated with the model, calculated external doses are also presented assuming that a worker, e.g. an emergency worker performing life saving actions or protecting actions, spends 10 minutes in the reactor hall. Therefore, the effects of radioactive decay and the effect of iodine plate-out against the walls, floor and roof could be ignored. Subsequently, it was also assumed that the ventilation system was removing fission gases at a very slow rate from the reactor hall. Thus, the analysis was regarded as taking a "static" sample in a short period of time. One of the underlying assumptions in such a consideration is that the emergency ventilation, which prevents gaseous radioisotopes to leave the building during an accident, had come into operation as one of the protecting actions designed into the reactor system for accident constitutions.

It is shown how the radioactive inventory of the fuel, the operational history resulting in a certain amount of burn-up, the type of accident, the protecting measures, the

characteristics of the reactor building etc. play a role in the radioactive isotopes released in a reactor building during an accident.

However, before the results of the calculations done by this model are presented, the dose received by a worker standing in the centre of a hemispherical cloud of ^{131}I were calculated with a simple MCNP model and compared with a benchmark analytical method, solved numerically. The result of this comparison provides further confidence in the complete MCNP model.

1.5 Research Report structure and chapter outline

According to the methodology described above, the layout of the study is diagrammatically presented in Fig. 1.4.

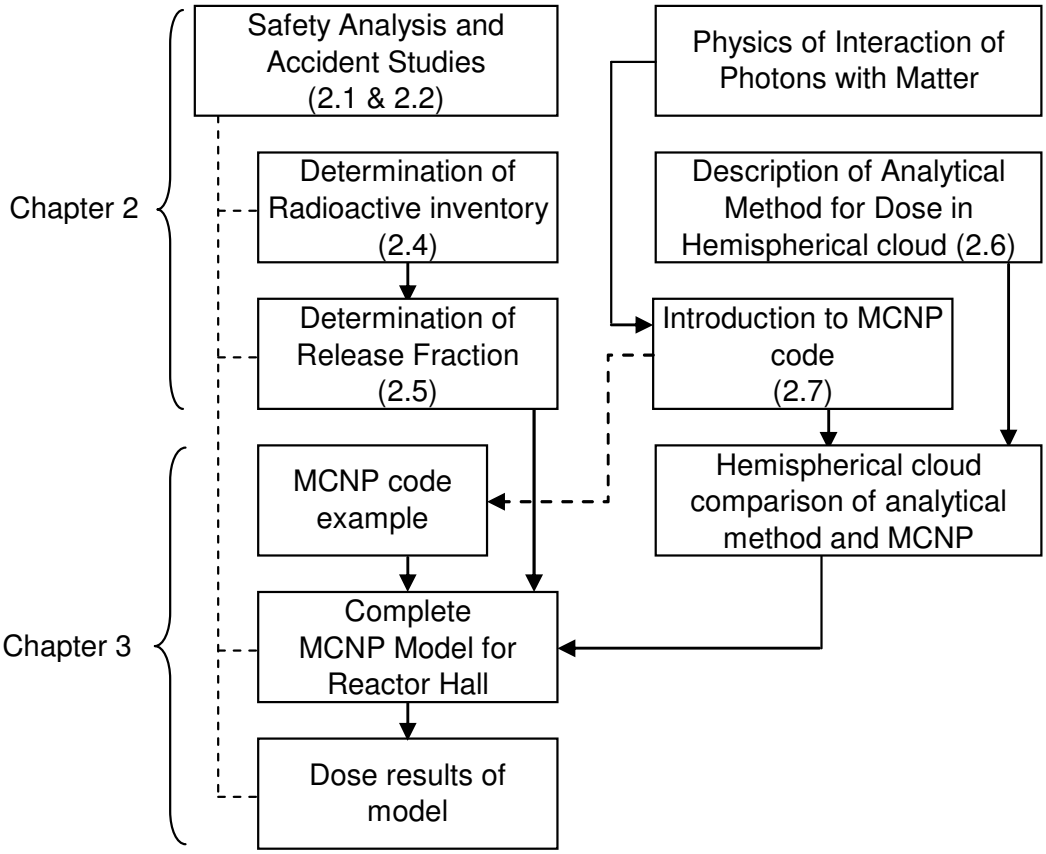

Figure 1.4: Diagrammatic presentation of Chapter 2 and 3 layouts of the present Research Report.

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

Chapter Two – Theory part consisting of:

- More theory on safety analysis is described. The risk of an accident to the worker is weighed against that to a member of the public. This is done for research reactors and NPPs to strengthen the emphasis of analysis in the Level 2 space, where the worker is involved.
- Radioactive inventory – the mechanisms are described of arriving at a set of radioisotopes available in the core at the time of an accident.
- Release fraction – the release fraction determines what percentage of radioisotopes will be released to the building for Level 2 analysis. The choice of ^{131}I to study as a representative gas being released during accident conditions is justified and it is explained how and on what basis a release fraction for ^{131}I was chosen.

Chapter Three – Results consisting of:

- A hemispherical shaped cloud of ^{131}I release is used to compare the analytical method with the MCNP method.
- A simple MCNP example applicable to the complete MCNP model is discussed.
- More detail is presented on the MCNP model constructed specifically for the reactor building.
- A presentation of results is provided. The results consist of the external whole-body dose due to γ-radiation from the ^{131}I cloud and the external skin dose due to β^- (electron) radiation of the ^{131}I cloud.

Chapter Four – Conclusions and recommendations are presented.

CHAPTER 2

THEORETICAL CONSIDERATIONS RELEVANT TO SAFETY ANALYSIS

In this chapter more theory on safety analysis is described. This includes detail on the levels of safety analysis, the type of accident chosen and the assumptions made with respect the amount of fuel damaged. The probabilities and consequences of accidents affecting the worker and the public are also discussed.

Thereafter, to better understand the assumptions made for the safety analysis and the results of the MCNP code runs, some physics of basic interactions of photons is discussed. The formation of radioactive inventory as an input parameter to the safety analysis is also discussed and some physics of interaction of the neutrons with matter, in particular fission, is discussed.

Some details are provided on the radiotoxicity and other attributes of iodine. The release fraction and the derivation of source term is also discussed here and it is explained that only ^{131}I is further studied, although a complete accident study would involve the study of other radioisotopes as well, in particular these that are more volatile.

Before an introduction of the MCNP code is provided, the analytical method to determine the dose to a person in the centre of a hemispherical cloud of ^{131}I is discussed. The MCNP code is then introduced before further progressing to Chapter 3.

2.1 Safety analysis

2.1.1 More detail on the levels of safety analysis

Safety Analysis of a reactor is concerned with three levels, irrespective of whether a deterministic or probabilistic safety analysis is done.

The **first level** of safety analysis concerns analysing the plant response to an initiating event that could develop into an accident sequence. For Probabilistic Safety Analysis (PSA), this level of analysis corresponds to PSA Level 1. The types of accident initiators (called initiating events) that can lead to plant damage in the context of research reactors are briefly discussed in this study as a precursor to the next level of safety analysis.

The **second level** of safety analysis addresses the building response to an accident that has progressed far enough that reactor core damage has occurred and that a subsequent release of radioactive material can be expected from the core into the reactor building. This corresponds, for probabilistic analysis, to PSA Level 2. The present study describes the accident progression on this level for the specific application in brief terms and then focuses on one specific element, namely the radiation from a radioactive "cloud" that has formed due to release as a gas of the more volatile radioactive isotopes, and in particular the isotope ^{131}I of iodine.

The **third level** of safety analysis is concerned with the spreading of radioactive gases once the accident has progressed in such a way that it has gone beyond the "barrier" provided by the reactor building. Atmospheric release models are used to determine doses to the public. The description of this level of analysis is beyond the scope of this study.

2.1.2 Type of accident chosen for the present study

The type of accident plays a major role in determining the quantity of fission products released and the timescale for the release. A certain amount of fission products is released into the reactor hall, and, depending on the efficiency of the outlet ventilation filters and the characteristics of the building, a percentage of this is eventually released out of the ventilation stack into the environment. Based on various selection criteria, the scope being beyond the present study, the type of accident chosen for the present study is a Loss of Coolant Accident (LOCA).

Before elaborating on a LOCA, a simplified diagram (see Fig. 2.1) will be discussed that shows how the primary coolant inlet and outlet piping is connected to the reactor

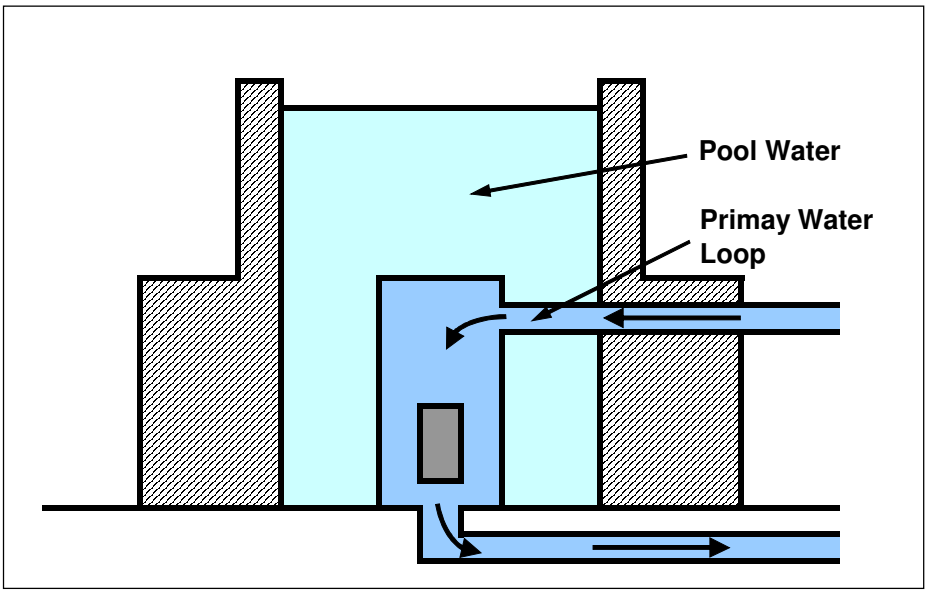

Figure 2.1: Connection of primary water loop to reactor vessel.

vessel. During normal operation of the research reactor, the residual heat is adequately removed by pumping the primary water through the core. The primary water loop is cooled, through a heat exchanger, by a secondary water loop [CAR65]. Under normal conditions, the contamination of water with radioisotopes is confined to the primary water loop, but the secondary loop is monitored for radioactivity to detect any leak from the primary loop. The secondary loop is cooled by a tertiary water loop, which is in turn cooled by the use of cooling towers. Specially engineered safety systems are normally incorporated into the design to prevent or mitigate a severe postulated accident. An example of this is an emergency core cooling water system that provides cooling to the core in the case of a LOCA.

A LOCA is an accident of concern in a research reactor [TEW05]. This type of accident can be caused by the rupture of a pipe in the primary cooling system or a break of an experimental beam tube [IAE92]. Figure 2.2 shows how a break in the inlet pipe to the reactor vessel can cause the core to be partially or completely uncovered and exposed to air, thus decreasing the heat removal capacity.

2.1.3 Amount of fuel damaged

In any reactor accident, an important measure is to know whether any fuel damage will occur during the accident. As soon as fuel damage has occurred, fission products from

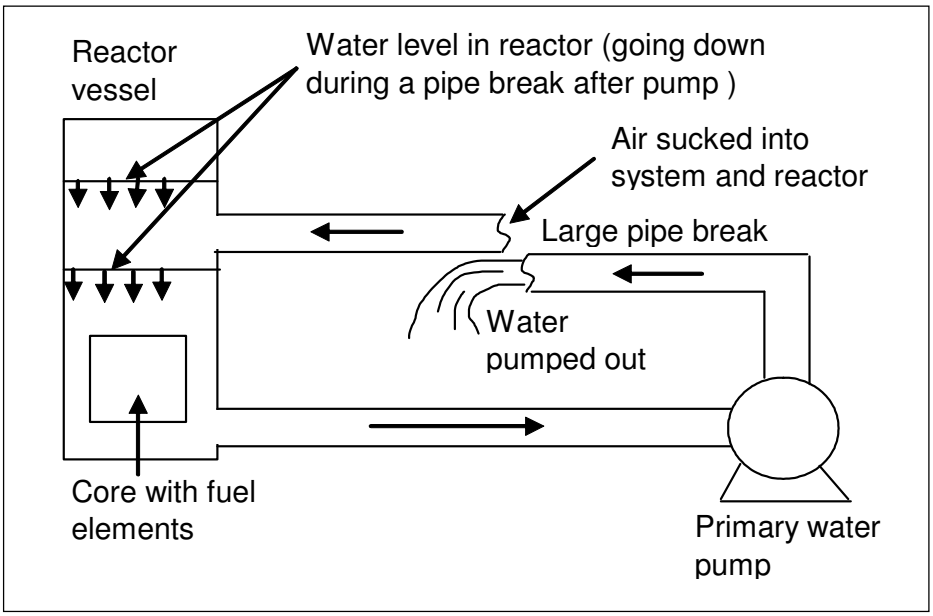

Figure 2.2: Most severe type of LOCA: a large pipe breaks downstream of the primary water pump.

the radioactive inventory can be released. The amount of fission products released will depend on various parameters like the temperature of the fuel plate, thickness and material of cladding and the composition of fission products that have been formed during burn-up. It is of great importance that fuel damage is avoided at all times. Under particularly severe core damage accidents, the total core melts down and forms a molten metal mass that flows to the bottom of the reactor vessel. This particular case is more applicable to Nuclear Power Plants (NPPs) and will not be considered for the purpose of the present study.

2.1.4 The decreasing probability of severe fuel damage because of a LOCA

Normally, on a new fuel type, experiments are performed to test behaviour with respect to fuel damage and release of fission products under simulated accident conditions in e.g. a laboratory setup. In this way the behaviour of the new fuel type is qualified. Because these experiments on accident conditions in the fuel cannot be representatively performed with a great degree of accuracy for the fuel design and core configuration of a particular research reactor, theoretical calculations with thermal-hydraulic and neutronic codes must be done to verify the use of the specific fuel type. An example of

a thermal-hydraulic code is RELAP5/Mod 3.2 [TEW05], which is used to simulate a LOCA or other accident scenarios.

According to Sharp and McCracken [SHA03], a hypothetical severe fuel damage event is seen as a Beyond Design Basis Accident (BDBA), i.e. severe fuel damage, where a total core melt-down is regarded as only happening with an annual frequency lower than 1×10^{-6} per reactor-year. In other words, this type of accident is also a "non-credible" accident.

2.1.5 Basis for choice of damage to fuel due to representative accident

In light of the above, a total core meltdown is not considered as the "representative" accident for this study. Whereas total core meltdown is often modelled[1] for NPPs, because of the greater probability thereof, this is not the case for the representative MTR reactor. However, it is assumed that a substantial amount of fuel damage occur affecting all 26 fuel elements. The fuel damage considered includes the formation of blisters or partial melting of the fuel cladding that allows the more volatile fission products to escape. Subsequently, the more volatile fission products i.e. the more volatile gases are released into the reactor confinement building.

2.2 Risk of accidents to the worker and to the public – a comparison with NPPs

NPPs have a single function, namely to produce electricity. In general, the tasks performed by the personnel in the plant are of a routine nature. In contrast, at research reactors a more frequent alteration in tasks take place. Some operating parameters may change. Research reactors also generally have more operating modes than NPPs. This results in the probability and variety of accidents to the worker being greater for research reactors. On the other hand, NPPs pose a greater risk to the public.

[1] This includes the effects of damage to the Reactor Pressure Vessel (RPV) in a NPP because of the molten metal core, the subsequent flow of molten metal to the bottom of the reactor containment building and the built-up of pressure in the containment building due to hydrogen that is formed and fission products that are released.

2.2.1 Risk of accidents at research reactors

The risk of accidents is more to the worker in research reactors

The argument was put to the Research Reactor Review [RRR93] that worker accidents are more common in multipurpose research reactors than in NPPs because there are more frequent start-ups, shut-downs, fuel and rig movements, and more opportunities for human error.

The Research Reactor Review [RRR93] referred to a 1980 report by the US Oak Ridge National Laboratory (ORNL) which listed 9 serious accidents involving prototype power reactors or experimental reactors, and a further 3 involving multipurpose research reactors [BER80].

The risk of accidents is less to the Public for research reactors

The Research Reactor Review [RRR93] claims that: "Excluding experimental and prototype reactors, only four accidents designated as serious have occurred in multi-purpose research reactors. No off-site consequences were identified in any of the four cases.". Thus, as stated these serious accidents did not even lead to off-site effects. Off-site effects are defined as projected or actual exposure to off-site members of the public due to the large release of radioactive material.

The available radioactive inventory is also much smaller, i.e. approximately two to three orders of magnitude lower, for research reactors than for NPPs. Therefore, it can be concluded that the potential risk to the public from research reactors is less than from NPPs.

2.2.2 Risk of accidents at NPPs

The risk of accidents is less to the worker in NPPs

At NPPs, on the other hand, because of less frequent start-ups, shut-downs, fuel movements and experimental and other activities around the reactor while running, worker accidents are less common. There are also fewer opportunities for human error that will cause exposure to the worker.

The risk of accidents is more to the public in NPPs

As argued in Section 2.2.1 there is no dispute that, in general, accidents involving NPPs pose a far greater risk to the general public because of the far greater volumes of fissile material used as fuel in NPPs, which entails higher inventories of fission products.

2.3 Physics of basic interactions of ionizing photons with matter

In this section, an overview of basic interactions of ionizing photons with matter is provided with the aim in mind to obtain more insight into the calculations done to determine dose to a worker from a radioactive cloud of ^{131}I. This is valid for the traditional method used to calculate the dose from a large amount of discrete volumes to a receptor as explained in Section 2.6 and to get insight into the photon interactions of absorption and scattering that is accounted for in the built-in physics and operation of the Monte Carlo N-Particle (MCNP) code used.

To better understand the concepts of interaction of radiation with matter, some basic atomic physics will first be introduced in Sections 2.3.1 and 2.3.2. Section 2.3.1 also provides some input into understanding Section 2.4 but the material is included here for the completeness of this section. Thereafter, the principles of absorption and scattering of photons will be discussed in Section 2.3.3.

2.3.1 Overview of some concepts in atomic physics

An atom consists of a nucleus and a "cloud" of negatively charged electrons moving in different orbits at different energy levels. The atomic nucleus further consists of protons and neutrons, the former being positively charged. Other particles also exist in an atom, but the major particles of concern to understand the physics below are the ones mentioned. Protons and neutrons are also collectively called nucleons.

Ninety-two naturally occurring elements exists in nature with uranium having the highest atomic number (92). Several heavier elements have been made up artificially e.g. plutonium with an atomic number of 94. Of the same element, stable isotopes and unstable radioactive isotopes exist.

Instability and radioactivity

Of the elements with small atomic numbers, large percentages of the naturally occurring isotopes are stable. However, small amounts of radioactive naturally occurring isotopes of these elements exist in nature, e.g. ^{3}H (Tritium) and ^{14}C.

All isotopes of elements with an atomic number larger than 83 are radioactive. To better understand this phenomena a short explanation of the influence of the size of the nucleus and the forces therein are explained.

Two main types of forces play a role between the nucleons in a nucleus, the attractive strong nuclear force and the Coulomb repulsive force which acts between the positively charged protons. When the atomic number is low, the repulsive force among the protons is small and a neutron to proton ratio of close to unity is appropriate for stability. With increasing atomic number the repulsive force increases proportional to the square of the atomic number (i.e. to Z^2). To compensate for the increasing repulsive force the ratio of neutrons to protons is larger in a stable isotope in order to maintain stability. However, a limit exists where the excess amount of neutrons over protons cannot compensate to maintain stability.

Unstable isotopes undergo radioactive decay, which is spontaneous change at a definite rate, and emit α or β-particles. The α-particles emitted are the same as helium nuclei and consists of two protons and two neutrons. The β-particles are electrons that are not originating from the electron cloud but rather from the unstable nucleus, where a neutron is converted to a proton with the emission of a β^--particle or vice versa with the emission of a positron (β^+-particle), which is a positive electron.

Beta decay can be better explained by means of the following formulae:

$$^{60}\text{Co} \rightarrow {}^{60}\text{Ni} + \beta^- + \bar{\nu}_e, \tag{2.1}$$

where ^{60}Co is the parent nucleus, ^{60}Ni is the daughter nucleus, β^- the β^--particle and $\bar{\nu}_e$ the neutral anti-neutrino. Similarly, the decay of ^{22}Na can be shown below, yielding a β^+-particle and neutrino instead of a β^--particle and anti-neutrino, respectively.

$$^{22}\text{Na} \rightarrow {}^{22}\text{Ne} + \beta^{+} + \nu_e. \qquad (2.2)$$

Because either a neutrino or an antineutrino, which shares the reaction energy of the decay, is emitted as well, there is an energy spectrum, ranging from 0 to the maximal available energy. The shape of the energy curve of the β-particles depends upon what fraction of the balance of the reaction energy or missing energy is carried by the neutrino or an antineutrino. The average energy of the β-particles emitted is approximately equal to a third of the maximum energy.

The energy curve of β^{+}-particles also differs from these of β^{-}-particles. This can be explained by the fact that the negatively charged β^{-}-particle is attracted to the positively charged nucleus, whereas the positively charged β^{+}-particles are repelled from the positively charged nucleus. Shown in Fig. 2.3 are the continuous energy spectra for β^{-}-particles and β^{+}-particles, respectively.

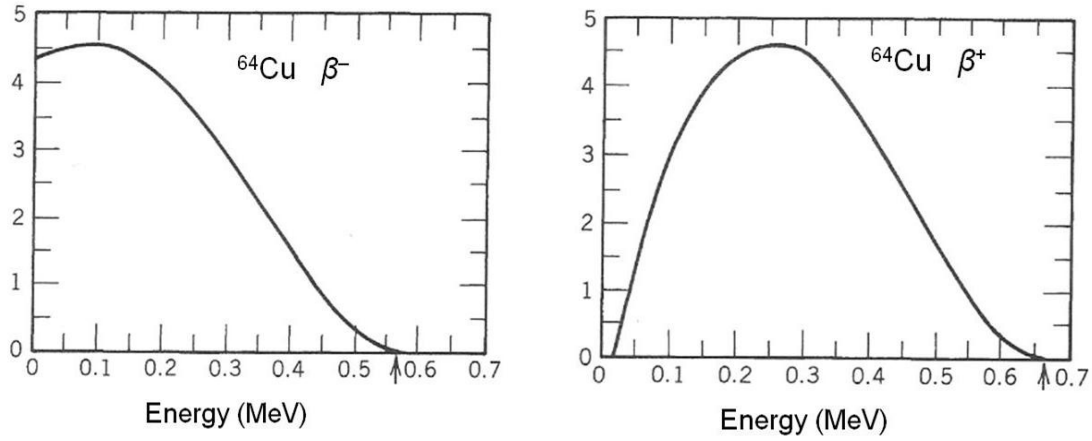

Figure 2.3: Continuous energy spectra for β^{-}-particles and β^{+}-particles of ^{64}Cu [KRA88].

As will be elaborated on in Section 2.3.3 under "the emission of photons", both α and β radioactive decay is usually accompanied by the emission of γ-ray photons. The decay schemes for ^{60}Co and ^{22}Na are shown in Figs. 2.4 and 2.5, indicating the β-decay and corresponding γ-ray photons. Although the interactions of γ-ray photons with matter are only explained in Section 2.3.3, the gamma spectra of ^{60}Co and ^{22}Na, which provide better insight into the decay schemes and γ-ray photons emitted, are also shown in Figs 2.6 and 2.7, respectively. The Compton Scattering region and the photo peaks arising

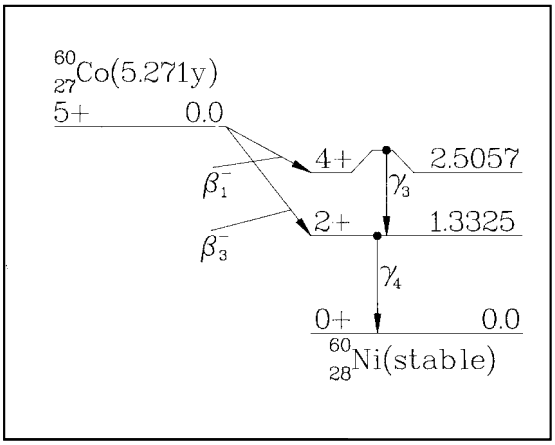

Figure 2.4: ^{60}Co decay scheme [ICR83].

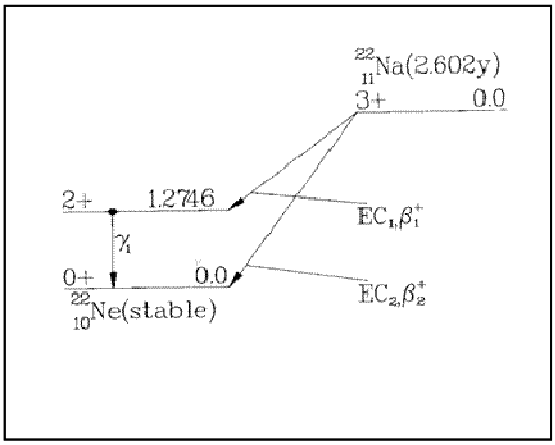

Figure 2.5: ^{22}Na decay scheme [ICR83].

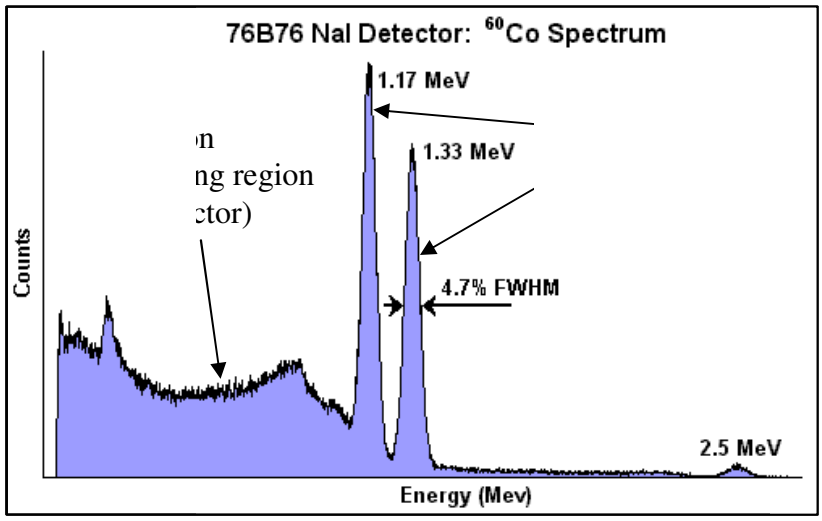

Figure 2.6: Gamma spectrum for ^{60}Co.

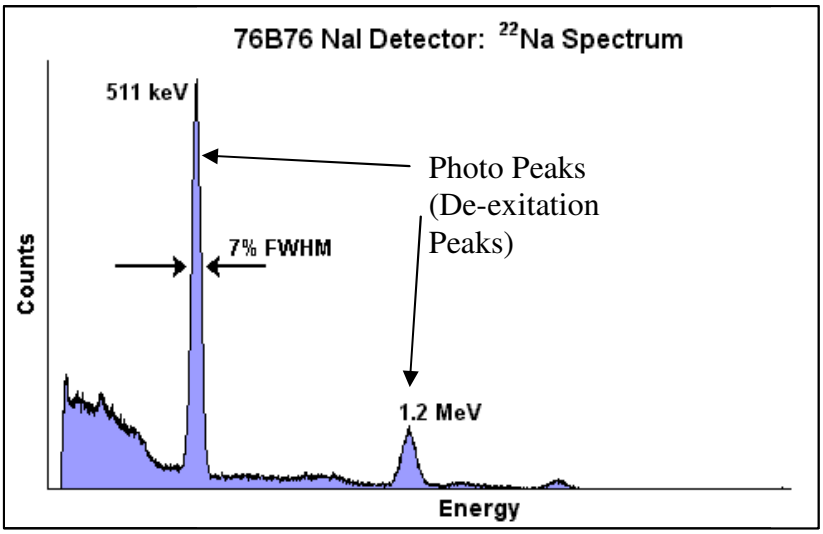

Figure 2.7: Gamma spectrum for ^{22}Na.

from de-exitation of the parent nuclides can be seen in Figs. 2.6 and 2.7. Bremsstrahlung, originating from the Photo-electric Effect (also explained in Section 2.3.3), occur as a result of interactions of the electrons emitted in the detector by the γ-ray photon interactions. However, since the energy spectrum of the Bremsstrahlung is continuous and the energy of Bremsstrahlung photons may take any value between "zero" and the maximum kinetic energy of incident electrons, a distinct peak of Bremsstrahlung emitted, cannot be seen in Figs 2.6 and 2.7.

β^--particles are primarily the concern of the present study. This can be justified by the fact that reactions taking place in a nuclear reactor results in a neutron rich environment and generally produces radioisotopes that are rich in neutrons. Hence these isotopes, when decaying with β-decay, emit β^--particles. These β^--particles are referred to in the results chapter (Chapter 3) as electrons when considering radiation transport calculations with MCNP.

Radioactive decay series

To better understand how various radioactive isotopes can result in other radioactive isotopes eventually forming a series, brief mention is made of the four distinct radioactive decay series. Although as previously stated, uranium is the heaviest element found in nature, the heaviest stable elements found in nature are several isotopes of lead (atomic number 82) and one of bismuth (atomic number 83). These isotopes of lead are ^{204}Pb, ^{206}Pb, ^{207}Pb, and ^{208}Pb found at 1.4%, 24.1%, 22.1%, and 52.1%, respectively, and the isotope of bismuth, ^{209}Bi. Except for the first natural isotope of lead (^{204}Pb), all of these isotopes are at least partially of radiogenic origin. As seen in Table 2.1, three radioactive series end in ^{206}Pb, ^{207}Pb, and ^{208}Pb and the fourth one in ^{209}Bi. In Table 2.1, the four series are shown by indicating the series' name, its first isotope, the first isotope's half-life and its last isotope.

Three of these radioactive decay series are named after the isotope with the longest half-life. The actinium series is named after one its members, in order to avoid having two series with identical names. Only the first three series in Table 2.1 are presently found in nature since the Neptunium series half-life is much shorter than the age of the earth [GRE02]. An example is provided in Table 2.2 of the full radioactive decay chain

Table 2.1: Four naturally occurring radioactive decay series. The half-life is that of the longest living member.

Series Name	First isotope	Half-life [years]	Last isotope
Uranium	^{238}U	4.49×10^{9}	^{206}Pb
Actinium	^{235}U	7.10×10^{8}	^{207}Pb
Thorium	^{232}Th	1.39×10^{10}	^{208}Pb
Neptunium	^{237}Np	2.14×10^{6}	^{209}Bi

Table 2.2: The Actinium decay series.

Isotope	Half-life	Decay Mode	Branching Ratio	Decay Product
^{235}U	7.04×10^{8} y	α	100%	^{231}Th
^{231}Th	25.52 h	β^-	100%	^{231}Pa
^{231}Pa	3.28×10^{4} y	α	100%	^{227}Ac
^{227}Ac	21.78 y	β^-	98.62%	^{227}Th
		α	1.38%	^{223}Fr
^{227}Th	18.68 d	α	100%	^{223}Ra
^{223}Fr	22.00 m	β^-	99.994%	^{223}Ra
		α	0.006%	^{219}At
^{223}Ra	11.43 d	α	100%	^{219}Rn
^{219}At	56 s	α	97.00%	^{215}Bi
		β^-	3.00%	^{219}Rn
^{219}Rn	3.96 s	α	100%	^{215}Po
^{215}Bi	7.60 m	β^-	100%	^{215}At
^{215}Po	1.78 ms	α	99.99977%	^{211}Pb
		β^-	0.00023%	^{215}At
^{215}At	0.10 ms	α	100%	^{211}Bi
^{211}Pb	36.10 m	β^-	100%	^{211}Bi
^{211}Bi	2.14 m	α	99.724%	^{207}Tl
		β^-	0.276%	^{211}Po
^{211}Po	516 ms	α	100%	^{207}Pb
^{207}Tl	4.77 m	β^-	100%	^{207}Pb
^{207}Pb	stable			

of the Actinium series. Radioactive decay plays an active role in the built-up of and constant change of composition of radioisotopes in the core of a nuclear reactor. This will be put into context in Section 2.4.

2.3.2 Exited states of the Atom and Nucleus

Exited states of the Electrons in an Atom

Some of the electrons moving in orbits are more tightly bound in an atom than others. More energy is needed to remove an electron from the inner most orbit (K shell), than what is needed to remove it from an outer orbit. For example, for lead, 88 keV is needed to remove an electron from the K shell and only 7.38 keV is needed to remove an electron from the outer orbit [LAM01]. The process of removing electrons from its orbits is called ionization. The energy that is needed to remove an electron from its orbit is called the ionization energy or, alternatively, is called the "binding energy" of the electron.

With respect to the energy states of the electrons in an atom, there are various energy states in which the electrons can be. The lowest energy state is called the *ground state*. When the atom is in a higher energy state, it is called to be in the *excited* state. The different energy states can be depicted on an energy-level diagram. When an electron(s) is (are) completely removed from the atom, the atom is ionized. Thus, the energy added to an atom by means of incoming radiation (i.e. neutrons, photons or charged particles) can either cause its electrons to be in particular excited states or ionize the atom.

When an electron is in one of the excited states it tends to move towards a lower energy state and eventually the ground state. In this process, in turn it emits a photon. The atom thus "decays" to a lower energy state.

Exited states of the Nucleus

Similar to the scheme in which the electrons are at different energy levels, the nucleons can also be at different energy levels. The state of lowest energy is also called the ground state. Except for very light nuclei, all other nuclei have excited states, generally with many more states than for electrons and also at much higher energies. The nucleus can also decay by emitting a photon.

Emission of photons by an excited nucleus

Although there are many types of photons and at various energies, we are here concerned with ionizing photons. These are photons that have the capability of causing radiation damage. Two types of ionizing photons are of concern; X-ray photons and γ-ray photons.

Because nuclear excited states are at much higher energies than those of electrons in atoms, the photons emitted from the nucleus are emitted at much higher energies than the photons that are emitted from the electron cloud around the nucleus, e.g. when the electrons move to a lower energy state. Hence, a distinction is made between photons originating from the nucleus and photons originating from the electron cloud. In the former case γ-ray photons are produced and in the latter X-ray photons with energies in the order of MeVs and keVs, respectively.

γ-ray photons are produced usually alongside other forms of radiation such as α or β-decay. When a nucleus emits an α or β-particle, the daughter nucleus is sometimes left in an excited state. It can then jump down to a lower level by emitting a γ-ray photon.

2.3.3 Basic interactions of ionizing photons with matter

As stated in the introduction to Section 2.3, the basic interactions of ionizing photons are discussed with the explanation of the MCNP code in mind. The application of the MCNP code for the present study is treated in Section 2.7. Since we are primarily concerned with γ-ray photons for the present study, when referring to "ionizing photons" or just "photons", it is implied that it is γ-ray photons, unless otherwise stated.

Of particular interest are the ways in which ionizing photons can deposit energy into matter, especially the human body. The amount of energy deposited is called the absorbed dose and is the measure that will be used for the MCNP results.

When ionizing photons pass through a medium, the photon can:

- penetrate through the section of matter without interacting;
- interact with the matter and be completely absorbed by depositing its energy, called the Photo-electric Effect; or

- interact and be scattered or deflected from its original direction and deposit part of its energy, called the Compton Effect.

A third interaction, which does not strictly falls into the categories above is Pair Production where an electron-positron pair are formed, usually followed by the annihilation of the positron with another electron, resulting in another two photons ($E\gamma$ = 0.511 MeV) being produced.

Each of the three interactions, the Photo-electric Effect, the Compton Effect and Pair Production are discussed in more detail below. These are considered as the main types of interactions ionizing photons can undergo [KRA88].

Photo-electric Effect

An ionizing photon interacts with and transfers its energy to an atomic electron, ejecting that electron from the atom. The kinetic energy of the resulting electron is equal to the energy of the incident γ-ray photon minus the binding energy of the electron [KRA88]. The Photo-electric Effect is more dominant at lower energies, generally in the region of 50 keV and lower.

As can be seen, the transfer of energy takes place from the incident photon to the resulting electron. The resulting electron, if passing near another nucleus, emits Bremsstralung. Also, a "gap" is left in the original atom's orbit, caused by the resulting electron that had been ejected. An electron from an higher energy orbit decays to the vacancy and emits a characteristic X-ray photon [GLA81]. It can thus be seen that all the energy of the incident photon is transferred to the matter through which it passes. Figure 2.8 shows a representation of the Photo-electric Effect.

Compton Effect

This is an interaction in which an incident ionizing photon loses enough energy to an atomic electron to cause its ejection, with the remainder of the original photon's energy being emitted as a new, lower energy ionizing photon with an emission direction different from that of the incident photon [GLA81]. Compton scattering is thought to be the principal absorption mechanism for γ-ray photons in the wide energy range from 100 keV to 10 MeV, although it can also occur at lower energies. The probability

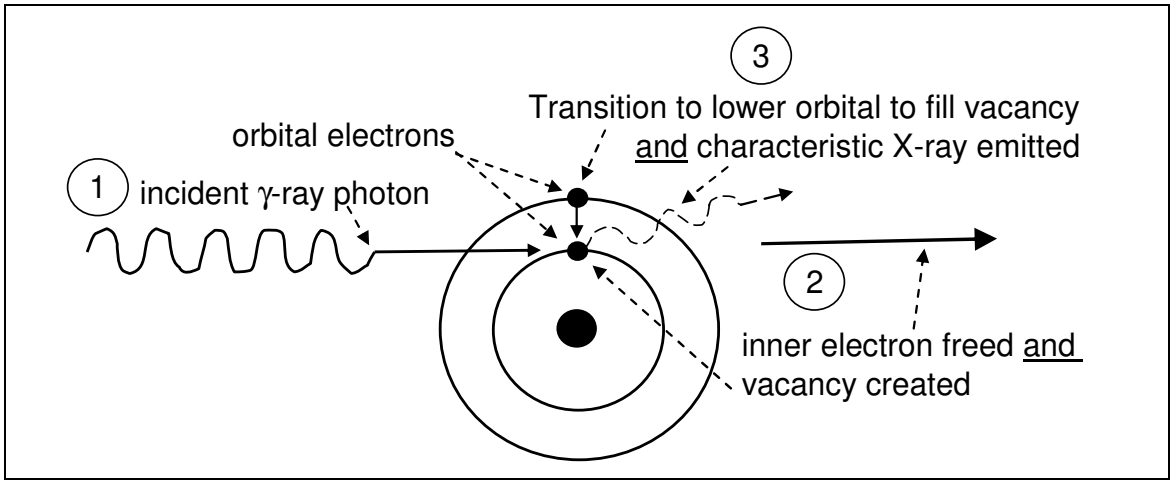

Figure 2.8: Schematic representation of the Photo-electric Effect.

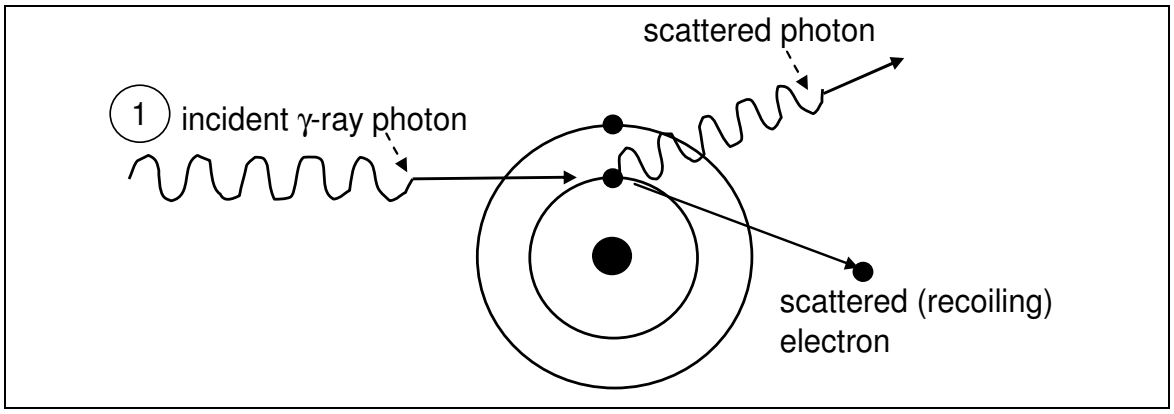

Figure 2.9: Schematic representation of the Compton Effect.

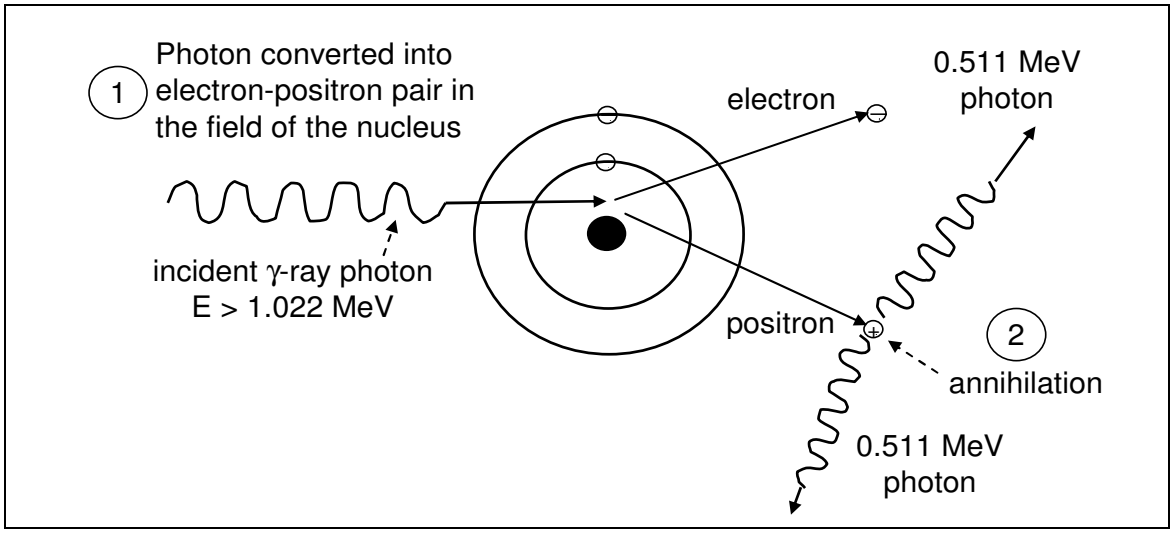

Figure 2.10: Schematic representation of Pair Production.

of the Compton Effect taking place at a given energy versus the probability other interactions, i.e. the Photo-electric Effect and Pair Production, will be discussed under the subheading "The Photo-electric Effect versus the Compton Effect" below. For example, it will be made clear that in the region of 50 keV to 100 keV, either the Photo-electric Effect or the Compton Effect can occur. Figure 2.9 shows a representation of the Compton Effect.

Pair Production

Pair Production is a photon-matter interaction that can occur only when photons have energy at an excess of 1.022 MeV.

By interaction with the electric field of a nucleus, the energy of the incident photon is converted into the matter. The interaction produces a pair of particles, an electron and a positron. When the positron meets a free electron, the two particles are converted to two 0.511 MeV or higher energy photons. This process is called annihilation. [CEM96]. Pair production is more dominant at higher photon energies, generally in the region of above 10 MeV. It is important to note that the two 0.511 MeV photons produced (equivalent to the rest mass of the electron) can interact in their turn either by Compton Scatteringor the Photo-electric Effect. One or both photons can also leave the medium. It can thus be seen that Pair Production is also an energy absorption mechanism. Figure 2.10 shows a representation of Pair Production.

Photon cross-section data

The probability that each the above interactions can take place depends on the energy of the incident photon and the absorbing medium in which the interaction takes place. The probability of interaction is sometimes called the cross-section (σ) when expressed in barns / atom. Figure 2.11 shows the relative importance of the three main interactions as a function of the Z number of the absorbing medium and the energy of the incident photon travelling at the speed of light. The energy of the incident photon, can also be expressed as $h\upsilon$ where h is Planck's constant and υ the frequency of the photon.

Figure 2.12 shows the probability of photon interactions as a function of energy for air expressed as *mass attenuation coefficients*. Also shown are the corresponding *mass absorption coefficients*. The mass attenuation coefficients is representing the case

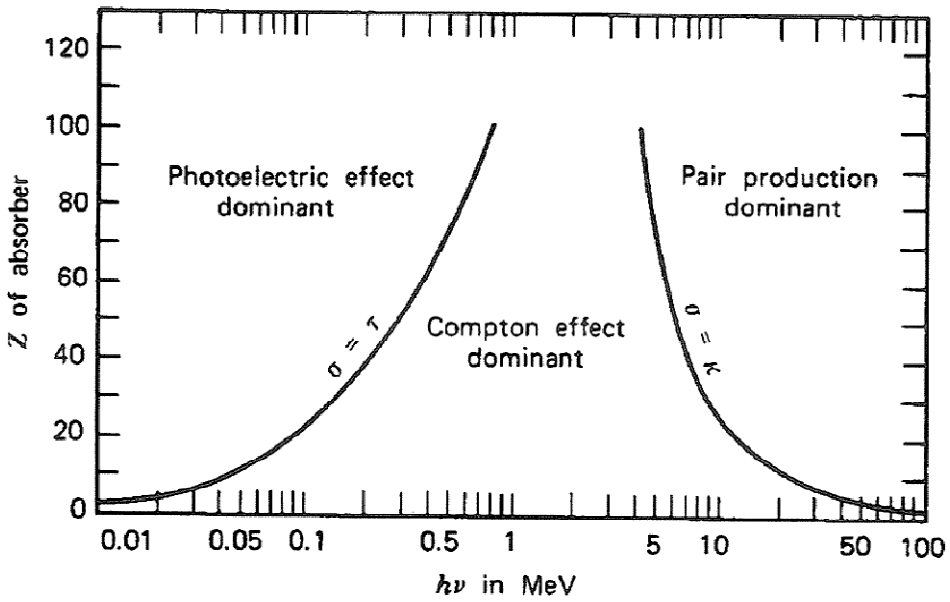

Figure 2.11: The relative importance of the three major types of interactions of ionizing photons [KNO99].

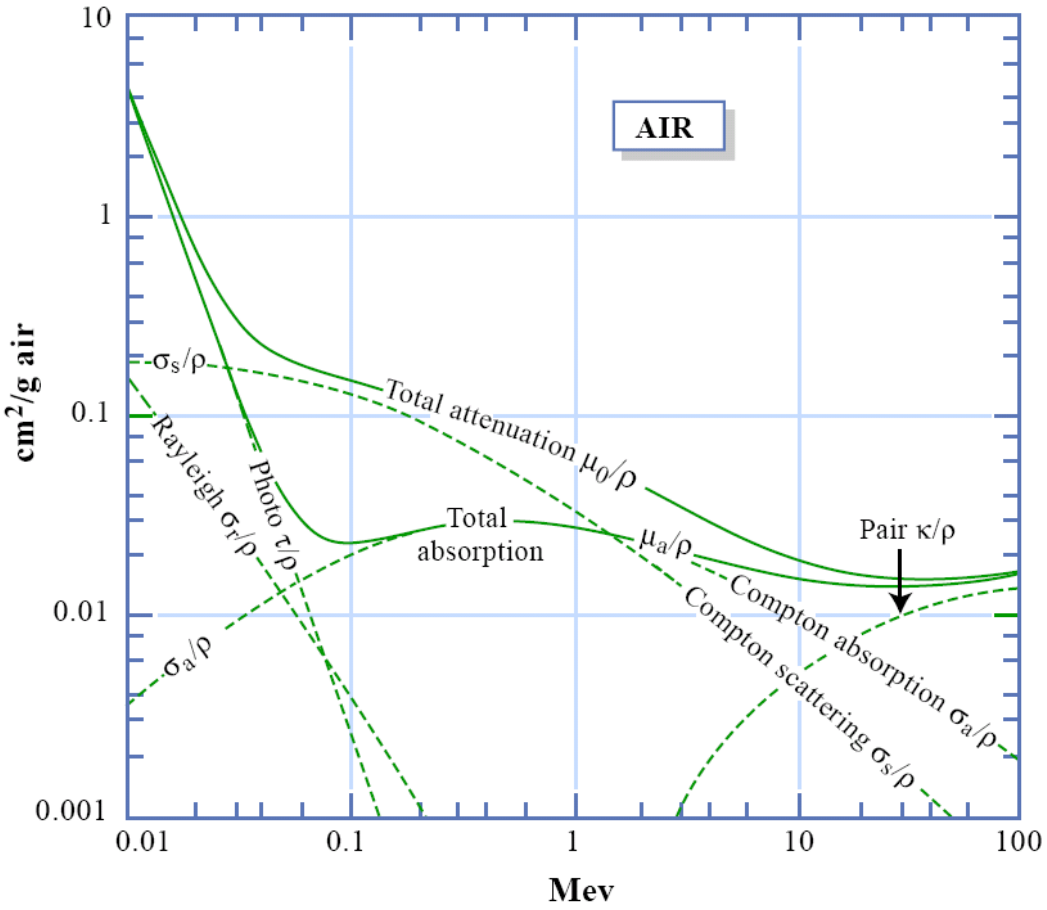

Figure 2.12: Mass absorption and attenuation coefficients as function of energy of incident photon [EVA55].

where it is assumed that all photons are absorbed, i.e. that the energy carried away by the scattering photon in Compton Effect, and the energy carried away by annihilated radiation (photon produced) in Pair Production is not included. If these effects are included, less energy is absorbed in the medium, and this is represented by the mass absorption coefficients shown Fig. 2.12. Figure 2.13 shows an example of mass absorption coefficients for different materials and incident photons of different energies.

The Photo-electric Effect versus the Compton Effect

From Fig. 2.11 it can be seen that the Photo-electric Effect is dominant at low incident photon energies and in absorbing mediums of high Z numbers. At approximately 1 MeV, the Compton Effect is the dominant effect for all Z numbers. From the descriptions of the Photo-electric Effect and the Compton Effect, and the energy ranges mentioned for these interactions (below 50 keV and above 100 keV, respectively), it may seem that there is an apparent gap in which it is not clear what interaction takes place. However, as now can be seen, both of these interactions can occur in this energy range, depending on the Z number of the absorbing medium. Also, considering the specific example of air (Fig 2.12), it can be seen that both interactions takes place up to 100 keV, but below 40 keV, the Photo-electric Effect is dominant.

Compton Effect versus Pair Production

From Figs. 2.11 it can be seen that the Compton scattering is the dominant effect in the intermediate energy ranges (~ 1 – 5 Mev) and that the photon cross-section for Pair Production remains low until the energy of the incident photon approaches several MeVs. In this energy region, in turn, this is the dominant mechanism. For the example of air (Fig. 2.12), Pair Production only becomes dominant above ~ 10 MeV.

Most probable interaction for ^{131}I

The majority of the photons emitted by ^{131}I have an energy that exceeds 0.1 MeV. Also, none of the photon energies exceeds 1 MeV. For example, the photon with the highest abundance emits 0.3645 MeV with a branching fraction of 81.2%. Therefore, according to Fig. 2.12, it is expected that the majority of the photons emitted by the ^{131}I gas cloud in the reactor hall will undergo the Compton Effect.

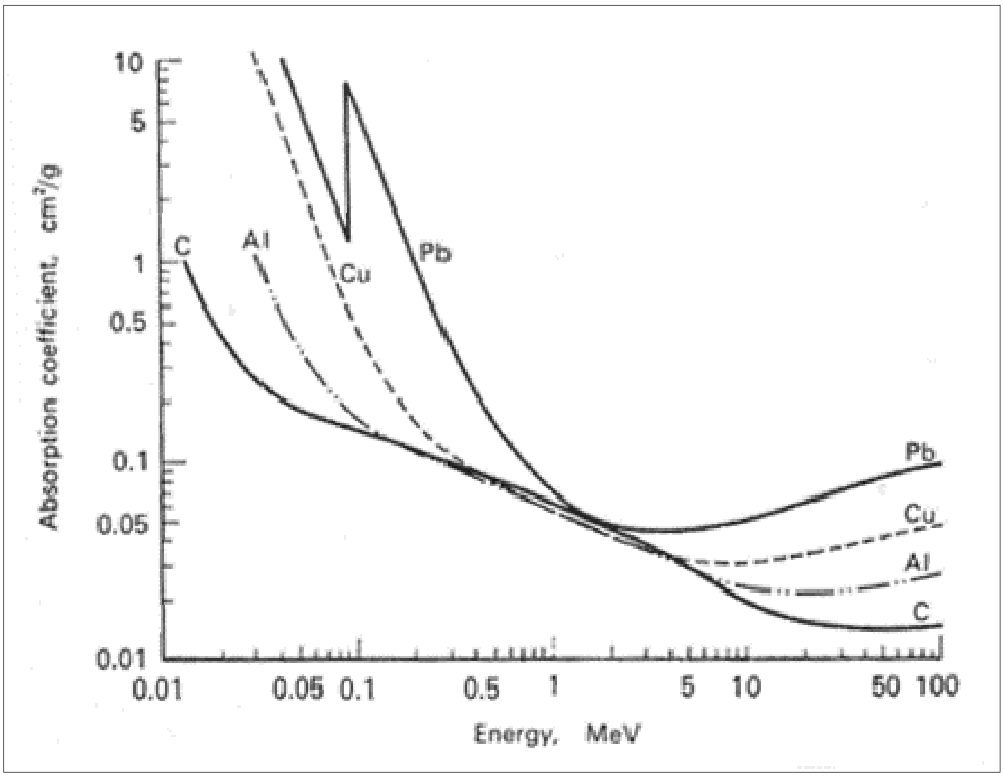

Figure 2.13: Mass absorption coefficients as function of energy of incident photon for different materials [CEM96].

2.4 Radioactive Inventory in a reactor core

In this section, the mechanism of formation of a collection of radioisotopes in a reactor core during operation is discussed. This is an important input for accident studies. However, before more detail will be given to the assumptions made and method used to arrive at the collection of radioisotopes in the reactor core, called Radioactive Inventory, some basic physics concepts of neutron interaction with matter will be explained.

2.4.1 Physics of basic interactions of neutrons producing a radioactive inventory

Various neutron reactions are considered for reactor-core operational performance calculations amongst which are elastic scattering, inelastic scattering, fission and neutron capture [KES83]. The latter two reactions are of importance to the production of a radioactive inventory and are discussed in more detail in sections below under subheadings “Fission” and “Neutron capture and radioactive decay”, respectively.

Fission

Nuclear fission is a process by which a nucleus splits into two or more lighter isotopes either spontaneously or after the original nucleus has absorbed various particles, in most cases an incoming slow neutron, or γ-ray photons. Fission is the main source of nuclear energy in a nuclear reactor.

The fission process was first discovered by Hahn and Strassman in 1939 [ROY67]. After the discovery of fission it was quickly perceived by Meitner and Frisch (1939) that the phenomenon could be explained as a consequence of the Liquid Drop Model of the nucleus [LYN89].

Liquid Drop Model

The Liquid Drop Model can be used to envisage how a nucleus breaks apart. An incoming neutron is absorbed by a large nucleus and causes it to become unstable. The unstable nucleus can be compared with a liquid drop that is about to split into two separate drops. Some energy is needed to deform the liquid drop enough to split. A heavy nucleus is more easily deformed. Also, certain factors like the strong nuclear force and surface tension of the "liquid drop" play an important role in the fission process. The Liquid Drop Model has five components i.e. Volume, Surface, Coulomb, Asymmetry and Pairing Energy terms.

Fission barrier, "fissile" and " fissionable"

When an incident neutron is absorbed by a nucleus, the binding energy of the neutron is added to the nucleus. The minimum excitation energy required for nuclear fission to take place is called the fission barrier energy. If the binding energy added by the incident neutron is enough to overcome the fission barrier energy, fission takes. This is only possible in fissile materials, where fission takes place due to incident slow or thermal neutrons. The fissile isotopes are ^{233}U, ^{235}U and ^{239}Pu.

Other materials are fissionable. In these cases, however, the kinetic energy of the incident neutron is also needed to overcome the fission barrier energy. Therefore, in fissionable materials, fission can be caused by fast neutrons.

Thermal neutrons

A thermal neutron is a free neutron with a kinetic energy of approximately 0.025 eV which is the energy corresponding to the most probable velocity at a temperature of 290 K. After a number of collisions with nuclei (scattering) in a neutron moderator medium at this temperature (most effectively protons of a light water moderator), neutrons arrive at approximately this energy, provided that they are not absorbed.

Fast neutrons

A fast neutron is a free neutron with a kinetic energy close to 1 MeV. Fast neutrons are produced by nuclear processes such as nuclear fission. Fast neutrons can be changed into thermal neutrons by means of the moderation process. In reactors, typically heavy water, light water or graphite are used to moderate neutrons.

Fission yield curve and asymmetric fission products

Many different isotopes may be produced by fission. This is illustrated by Fig 2.14 which shows the distribution by mass number of binary fission fragments induced by thermal neutrons and fast neutrons.

It is seen that symmetrical fission, which is the break-up into two nuclei of equal mass, is quite rare, especially for thermal neutrons. It can also be seen in the upper curve that with increasing neutron energy, i.e. in the range of fast neutrons, the probability of symmetrical fission increases. The mass numbers of the isotopes produced vary generally between 70 and 170. Typical examples of radioactive fission products are ^{85}Kr, ^{90}Sr, ^{131}I and ^{137}Cs.

Neutron capture and radioactive decay

As mentioned in the introductory paragraph of this section (Section 2.4.1), neutron-nucleus reactions can be classified into three categories, i.e. fission, scattering and neutron capture. Except for most instances of elastic scattering, the first stage in a neutron-nucleus reaction is usually the absorption of the neutron by the nucleus to form a "compound nucleus" in an excited stage [GLA81]. In the instance of inelastic

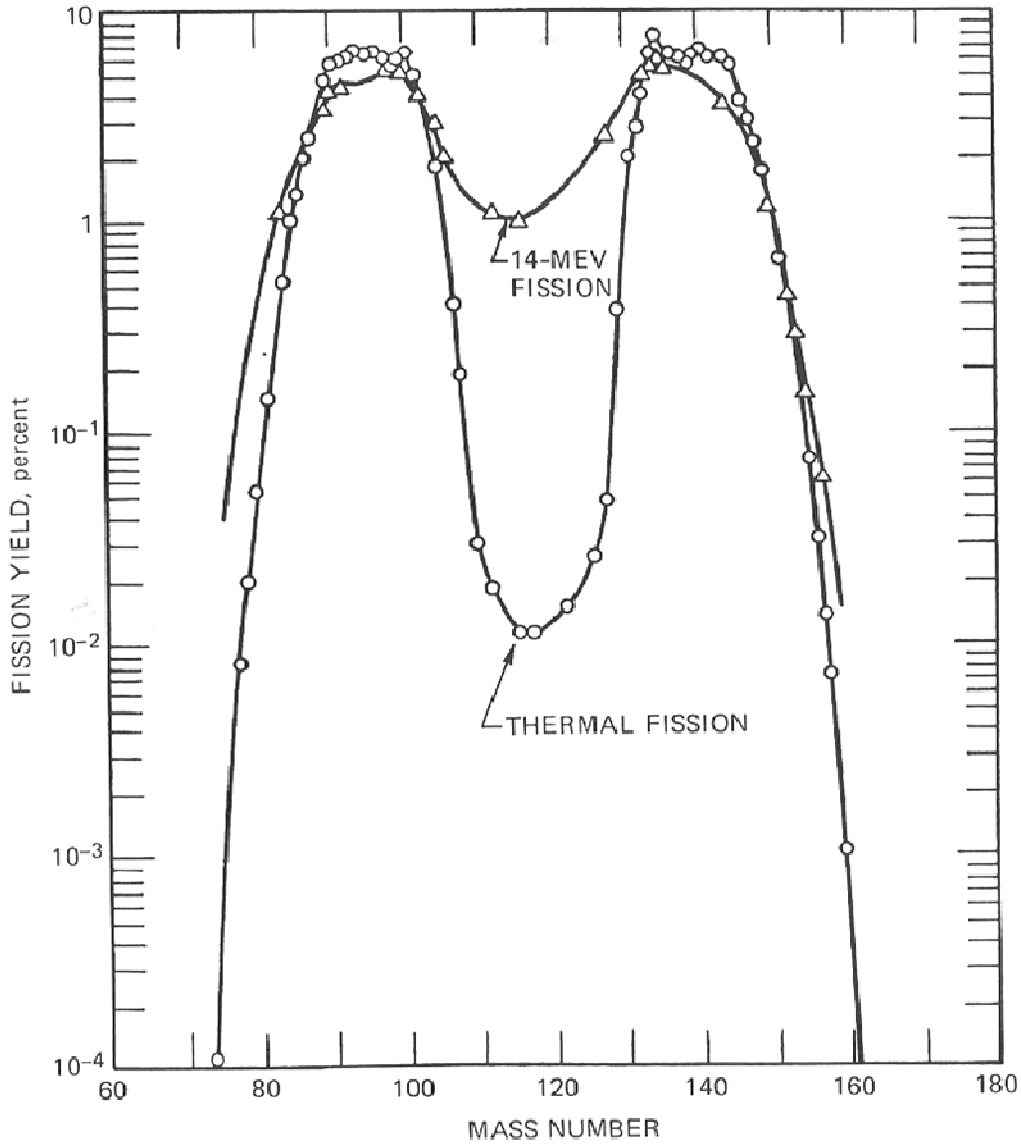

Figure 2.14: Fission yield as a function of mass number of the fission [GLA81].

scattering, another neutron is almost immediately expelled from the compound nucleus. However, the compound nucleus can also change in undergo change in other ways, e.g. by emitting a γ-ray photon. This process is called radiative capture, denoted by the symbol (n,γ). Another isotope, with one more neutron is thus formed. Radiative capture occurs more readily with slow than fast neutrons. Followed by the (n,γ) reaction are often a series of reactions where β^--particles are emitted as well.

Similarly, other processes e.g. (n, 2n) reactions takes place. Together with fission, these nuclear processes that take place in the uranium fuel change the concentration of the

various isotopes in the reactor core during operation. The radioactive inventory, therefore, constantly changes.

Classification of isotopes that makes the radioactive inventory

The isotopes produced by the mechanisms described in subheadings "Fission" and "Neutron capture and radioactive decay" above are classified into three groups, fission products, activation products and actinides. The production of fission products has already been described. Activation products form due to neutron capture of fission products and produce radioactive isotopes with similar mass numbers.

The last group, called "actinides", are formed through the same mechanisms but from high mass number elements i.e. from uranium and higher. These elements are also called the trans-uranic elements. Typical examples of actinides that are formed in a reactor are ^{234}U, ^{235}U, ^{236}U and ^{238}U, although the amounts and ratio of all three (except ^{236}U) are also a function of the enrichment process that was used during the manufacturing of the fuel.

2.4.2 Radioactive Inventory at Time of Accident

As a point of departure for accident analysis, the radioactive inventory in the reactor core, at the time when the accident sequence started, must be known. This is determined by means of theoretical calculations, done by a computer code. For the present study, the computer code ORIGIN-S, which is part of the modular code system SCALE [SCA05], was used to calculate the radioactive inventory.

Before more details of the ORIGIN-S calculation are provided in Section 2.4.3, some aspects influencing the calculation and of relevance to the output of the calculation will be discussed.

Systematic built-up of radioactive inventory during an operational cycle

As already stated, it is important to know the radioactive inventory in the fuel. However, of more importance is to know which part of the radioactive inventory can be relatively easily released and under what conditions. In this regard, the original content and irradiation history of the fuel is important.

Un-irradiated fuel versus irradiated fuel in relation to hazard imposed by radioactive inventory

When the fuel is manufactured, it is clad. For the MTR type reactor the uranium fuel is clad in a sandwich pattern, between aluminium plates, as already explained in Section 1.1.3 and depicted in Fig. 1.3. The uranium in the fuel is enriched, which means that the percentage of ^{235}U is more than that of naturally occurring uranium, which is approximately 0.72% with ^{238}U making up over 99%.

When the fuel is loaded in the reactor, it called "fresh fuel". It has not yet been irradiated, which means that the fissile material, the ^{235}U in this case, has not undergone any fission yet. Fuel is manufactured in such a way that the uranium is in a stable matrix form. The only radioactive isotopes in the fuel at this stage are these of uranium itself. As seen from Table 2.1, the half-lifes of ^{235}U and ^{238}U are very long (7.10×10^8 y and 4.49×10^9 y, respectively). Hence the specific activity of the uranium is relatively low. The uranium is also in a solid form. The chances of releasing this radioactive uranium to outside the cladding, not to even mention outside the pool, are very remote. That is why accident studies are not much concerned with the radioactive inventory of the fuel at this stage.

However, the moment when the fission reaction commences, a build-up of a wide spectrum of other radioactive isotopes starts. Some of these isotopes are volatile or semi-volatile and can thus more easily escape from the cladding when it is damaged due to accident conditions, e.g. because of excessive heat in the fuel.

Burn-up

Burn-up, which is the decrease of fissile ^{235}U present in the fuel, is taken as a measure of how much nuclear fuel has been used and is a function of the amount of fission that has taken place. Burn-up can thus be seen as a decrease of fissile ^{235}U present in the fuel. Burn-up can also be defined as a function of the total number of fissions that has taken place i.e. the fissions due to ^{235}U being present as well as due to the fissions of ^{239}Pu. Some ^{239}Pu forms in the fuel during operation and can contribute to the fission process. The burn-up percentage is calculated as

the amount of ^{235}U that has undergone fission over the total amount of ^{235}U that was present before start-up.

As will be explained below, for the purpose of the study, all the fuel elements are assumed to be at a burn-up of 80%. The most hazardous fission products are generally found at the maximum burn-up i.e. when most of the ^{235}U has been used, which is at the so called End of Life (EOL) of a fuel element. In the case of a research reactor, at this percentage of burn-up, it is difficult to operate the reactor, i.e. to obtain criticality or to keep it critical at the desired power level because of the lack of sufficient fissile material.

In practice, for research reactors, all the fuel elements will not be simultaneously at EOL, as this would make operation of a nuclear reactor rather difficult. Therefore, an operational core will consist of some "fresh" fuel elements which are at Beginning of Life (BOL), some semi-burned-up or semi-depleted fuel elements and some fuel elements near EOL.

The choice of 80% burn-up for all the fuel elements is a conservative assumption and is often made for the purpose of accident analysis. This imply that each of the fuel elements contain the maximum amount of fission products. Such conservative assumptions can be refined to give a more realistic (called "best estimate") picture of the combination of fresh, semi-burned and fully burned fuel elements but this is not the purpose of the present study.

The history of a fuel element, which can be directly related to burn-up, is expressed in Megawatt Days (MWDs). For example, if it is known that a core has been running at 20 MW for 20 days, the reactor core and fuel elements have been running for 20 x 20 = 400 MWDs. To ensure that the actual burn-up corresponds to theoretical calculations, measurements are also taken with, for example, copper flux wires.

A typical cycle is about 4 weeks long which ends in a burn-up of 20 MW / 26 Fuel Elements and with 28 days gives 560 / 26 ≈ 22 MWDs per cycle for one fuel element assembly. A fuel element is used for approximately 6 cycles, which

results in a burn-up of approximately 132 MWDs per fuel element. This is far less than the calculation that is done for 180 MWDs or 80% burn-up. The present investigation is, therefore, conservative because it is an accident study.

2.4.3 Calculation of radioactive inventory in a fuel element

The ORIGIN-S code [SCA05] is able to calculate the spectrum, quantity and activity of isotopes in a fuel element at any given time, knowing the amount of burn-up of a fuel element. This code incorporates the effects of the mechanisms discussed in Section 2.4.1.

For the purpose of this study, a calculation with the ORIGIN-S code was done with 26 fuel elements, and, as explained in Section 2.4.2, a burn-up of 80% in each fuel element. An output file was generated which contained over 40 fission products, activation products and actinide elements and over 180 isotopes of these elements. The masses, and more important for the study, the activities (in Bq) of each of these isotopes are also generated. The list of isotopes and there corresponding activities are depicted in Table C.1 of Appendix C.

Theoretically, each of these isotopes can be now be analysed by hand, in view of the photons and particles it gives off, and the energy of each of these, and a long list of photons and particles and corresponding energies and branching fractions could be generated, using a source like ICRP 38 [ICR83]. In our case, only the ^{131}I isotope's energies and branching fractions were analysed and the data used for the MCNP model.

2.4.4 Protection measures during time of accident

The type of protection and mitigating measures also play a major role in the amount fission products released and the time scale in which they are released. The accident sequence and the outcome in terms of severity of the radiological consequence is to a great extent determined by the protection and mitigating measures in place and the effectiveness thereof. Protection and mitigating measures consist of hardware systems and human actions like work procedures for operating the reactor and for emergency situations. The main systems that play a role during an accident sequence are the

reactor protection system that contains the control-rod scram system and the emergency ventilation system that forms part of the radioactive confinement system. The former system is a protection system that prevents an accident and the latter system is a mitigating system that limits the radiological effects outside the building.

In the case of the "representative" accident (see Section 2.1.2), it is assumed that the emergency ventilation still works. This implies that the integrity of the reactor hall is still good enough that the emergency ventilation provides a negative pressure and that the activated charcoal filters capture the majority of the iodine. Some events, namely, internally originated events like fire or externally originated events like air craft crashes, can damage a building in such a way that the emergency ventilation does not function, but this is not considered for the representative accident.

2.4.5 Characteristics of the reactor building

The reactor, as described in Section 1.1.3, is situated in a large hall serviced by an emergency ventilation system. The reactor hall serves as "confinement" structure. This means that the reactor is not designed, as in the case of most NPPs, with a "containment" building, that is designed for internal pressure in the building due to the release of fission products and other phenomena that cause a pressure built-up in the building. It is assumed that the ventilation can remove the fission products at a sufficient rate to prevent an overpressure (see Section 2.4.4). Since the reactor vessel is situated underneath the water, and during an accident the more volatile fission product gases will migrate through the water to the surface and will then be released in the building, it is expected that a radioactive cloud, consisting of the radioisotopes of the noble gases xenon and krypton and some halogens, namely iodine, will form above the reactor pool. During a LOCA, as can be seen from Fig. 2.2, excess amounts of water may be released from the pool and primary water system, which may result in steam forming above the core due to the excessive heat in the core. This will enhance the release of the volatile noble gases, xenon and krypton, and the halogen iodine. The behaviour of the fission products and complete radioactive inventory is better explained in Section 2.5, in particular Section 2.5.3, where release fractions are explained.

2.5 Release Fractions used for Source Term

Another important part of the calculation that needs to be considered for the accident study is the determination of release fractions. For a Level 2 analysis, this is the fraction of the fission products that will escape out of the pool into the reactor hall.

2.5.1 Definition of Source Term

Source Term is generally defined as the amount of radioactivity, given by the different radioisotopes and corresponding activities, that is "available" during an exposure scenario such as an accident to cause radiological damage to an exposed person, the environment or material. Therefore, the whole Radioactive Inventory can not be considered as comprising the Source Term since all of these nuclides will not leave the reactor core, vessel and pool to the reactor hall. For the purpose of this investigation, the fraction of fission gases, in particular iodine, that is released to the building are of importance. The term "Release Fraction" is used to define the ratio of activity of a particular isotope that leaves the reactor pool to the reactor hall over the amount of activity of that particular isotope that was originally in the fuel at the time when the release from the fuel started. The Release Fraction can thus be understood as the percentage of a particular isotope that leaves the fuel.

For Level 3 Analysis, the Source Term is defined as the activity that leaves the reactor and that serves as input for atmospheric dispersion modelling. In this case, release fractions are defined as the ratio of a certain radionuclide that leaves the building over the original activity in the fuel.

Since a Level 3 Analysis is outside the scope of the present study, it can thus be seen that the Source Term is the activity that is present in the building. The release fraction, as will be elaborated on in the next section, is used to derive the Source Term from the original radioactive inventory.

2.5.2 Availability of release fractions for research reactors and the selection thereof

Whereas for NPPs the behaviour of fuel under accident conditions have been well studied, information of release fractions for research reactors is relatively scarce [STA92].

The usefulness of information from NPP fuels is limited due to the differences in fuel type, power level and thermal-hydraulic conditions. A first investigation indicates that using data from NPP fuel leads to an overestimation of the Source Terms. Further research on this subject could be very useful for the research reactor community, in order to define more realistic Source Terms and to improve the emergency preparedness.

This is also true for the plate-type fuel of the MTR reactor under observation. It is mainly due to the uncertainty and variation in the physical core parameters during the envelope of the specific accident and implies that release fractions that are needed to arrive at source terms must be derived from literature on experiments done.

2.5.3 Choice of ^{131}I and release fraction for this study

Although the heading of Section 2.5 refers to "Release Fractions", for the present study the author has decided to investigate only one radioisotope, ^{131}I. In a full Level 2 analysis each radioisotope would be assigned a release fraction. Radioisotopes from the same element, e.g. ^{129}I and ^{131}I would be assigned the same release fraction, because of the exact similar chemical behaviour and physical behaviour as far as it is concerned with vapour pressure, volatility etc. Therefore, different chemical elements are categorized according to volatility. Experimental work done by e.g. Margeanu *et al*. [MAR07] and ORNL support the classification scheme that is done according to volatility.

To continue the scenario where a full Level 2 analysis would be done, within the spectrum of radioisotopes that has formed (see Appendix C) each would be assigned a release fraction varying almost from 0%, in case of the actinide group (Group 8, "NpO_2 and PuO_2"), to 100%, in the case of the highly volatile group (Group 1, "Noble gas" and

Group 2, "I_2, CsI, HBr"). Each radioisotope's activity (Bq), according to Table C.1 must then be multiplied with its own release fraction to get a Total Source Term, should a complete Level 2 accident analysis be required. Since the release fractions of Group 1 and 2 are high and are also called collectively the highly volatile group [PAR73], the release of the noble gases and iodine compared to the rest of the radioisotopes dominate by far. This implies that even during a severe accident like a LOCA, the less volatile radioisotopes would remain, to a large extent, in the core.

Because the focus of the present study is to demonstrate how the MCNP code can be applied, it is more appropriate to choose one significant radioisotope. Therefore, ^{131}I which falls under the highly volatile group and is a large contributor to the dose of an exposed person during an accident has been chosen for the study. Since the aim of the present study is not to do work to refine the release fractions based on previous data, the release fraction that was taken for ^{131}I is 100% for the release into the reactor hall. This is backed up by Margeanu *et al*. [MAR07] and ORNL data.

The statements on the volatility of ^{131}I, its contribution to dose during an accident and why in particular the ^{131}I isotope of iodine, is further backed up by the following overview on iodine.

General overview on iodine and justification for choice of ^{131}I

Iodine occurs in nature in various chemical forms, however not in the pure form, to a major extent in seawater and to lesser extent in rock. Under standard conditions it is a bluish black solid [BRO09]. However, it needs to be heated slightly above room temperature in order to sublime. Iodine in its gaseous form is violet-pink and has an irritating odour. It forms compounds with many elements, but is less reactive than the other halogens. There are 37 isotopes of iodine of which only one, the ^{127}I isotope is stable. Iodine is produced by extracting sodium iodate and sodium iodide from sodium nitrate or by purifying it from brine. It is used for a variety of applications, including diagnostic and medical treatment, photography and dyes used to perform colouring of substances. Some overview is further given in this section on the medical uses of iodine, the radiotoxicity, chemical toxicology, volatility, plate-out and decay mode and a final conclusion is provided on the choice of ^{131}I.

Iodine used in medical treatment

Examples of where iodine is used for medical treatment are ^{125}I which is used in biological assays and in radiation therapy to treat prostate cancer and brain tumours. This isotope is produced in a nuclear reactor by irradiation and consequent neutron activation of ^{124}Xe. Another isotope of iodine, ^{123}I, is used in nuclear medicine imaging using it as tracer or to study the high linear energy transfer characteristics of Auger electrons [SLA98]. The latter isotope is produced in a cyclotron. On the other hand, the neutron rich ^{131}I isotope, that is produced in a nuclear reactor by fission, has both diagnostic and therapy uses. The diagnostic uses include imaging and the therapy uses include procedures for thyroid disorders (e.g. where the thyroid is producing excessive hormones or when thyroid cancer is present). The relatively short half-lives of these three isotopes are 59.4 days, 13 hours and 8.02 days for ^{125}I, ^{123}I and ^{131}I, respectively, which makes them suitable for medical use.

Radiotoxicity of iodine

It is well known that iodine can be concentrated in the thyroid and can cause thyroid diseases. Within the first hours after occurrence of a nuclear or radiation accident the radioactive ^{131}I enters into the body mainly through inhalation and is rapidly taken up into the blood. There are other pathways that are not of interest because they occur over the longer term, i.e. in Level 3 analysis where the iodine once released into the environment gets concentrated like in milk, for example. Also, in the present study, the external and skin radiation is investigated, rather than the internal dose.

The most important radioactive isotope of the iodine from radiation protection point of view is ^{131}I to which a special attention is paid in case of nuclear or radiation accidents. Other radioactive isotopes as ^{133}I and ^{135}I also pose a health hazard but have shorter half-lives and for the internal exposure of the individual their contribution is smaller in comparison with ^{131}I. In addition, ^{129}I is also formed but has a very long half-life (15.7 million years) and hence a low specific activity. It should be noted that ^{131}I was also a significant contributor to the health effects of the Chernobyl disaster [WIL08] and was also shown to be the case during weapons testing e.g. at the Hanford Site in the USA.

Iodine Prophylaxis, normally in the form of potassium iodine (KI) tablets, is a

preventive protection measure aimed at avoiding the health damage to individuals resulting from the accumulation of radioactive iodine in the thyroid. The tablets, containing stable (not radioactive) iodine are taken prior to or up to two hours after exposure during an accident in order to saturate the thyroid with stable iodine thus avoiding the incorporation of radioactive iodine into the gland.

^{131}I, with a half-life of 8.02 days, emits a few high energy γ-photons as can be seen from Table 2.3. This is of importance to the external whole body dose received by an exposed person. The distance travelled in air by photons increases with the energy. Therefore, it can be expected that a person standing in or within a distance of approximately 1 m of a ^{131}I cloud would receive a large external dose, compared to a person standing several metre away. The latter person would only be affected by the very high energy photons. This will be shown by the results in Chapter 3.

Chemical Toxicity

Elemental iodine is an oxidising irritant. When the iodine is in crystal form, direct contact with the skin can cause lesions. Contact with solutions of iodine can cause tissue damage at prolonged contact. If ingested, a few grams can be lethal, Iodine vapour is very irritating to the eye, to mucous membranes and in the respiratory tract. Limits exist of allowable iodine concentration in the air. A so-called "iodine sensitivity" can be developed by some people. Therefore, also out of a pure chemical toxicology perspective, iodine is a noticeable hazard.

Volatility and plate-out

Iodine suffers from low solubility in water and its volatility makes it difficult to handle directly. According to the classification scheme proposed by Parker and Barton [PAR73], iodine is in the high volatility group. Therefore, it is expected to stay in the form of a gas cloud for a relatively long time while in the reactor hall. After a while, however, the iodine plate-out against the building surfaces, e.g. walls. The plate-out of iodine is a well studied subject as shown by e.g. Clough and Hood [CLO85, NEE97] and the time it takes to plate-out may vary between a few hours to a few days, depending on the physical phenomena resulting in varying surface temperatures during and after the accident.

Decay mode and scheme for ^{131}I

^{131}I decays in the following manner:

$$^{131}\text{I} \rightarrow {}^{131}\text{Xe} + \beta^- + \bar{\nu}_e. \tag{2.3}$$

The decay scheme, showing the modes of decay for photons and β^--particles and the accompanying energies when emitted is shown in Fig. 2.15.

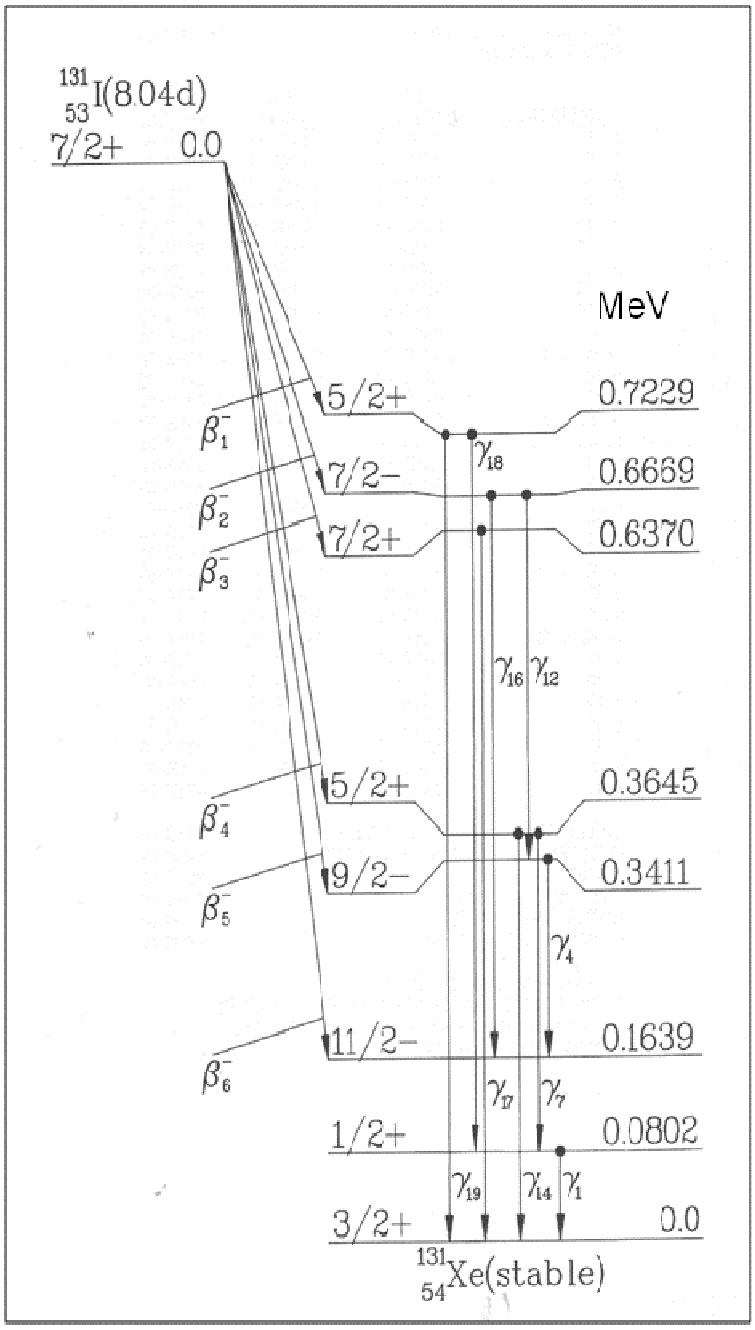

Figure 2.15: The decay scheme of ^{131}I [ICR83].

Radiation emitted by ^{131}I

The results of the decay of ^{131}I are shown in Tables 2.3 and 2.4, respectively. Here, in Table 2.3, γ_1 represents the transition depicted in Fig. 2.15 labelled γ_1 having an energy $E\gamma_1 = 0.0802$ MeV. Similary, K_{α_1} and K_{α_2} represent the x-rays that are produced due to electrons that move from a higher orbit (in this case to the K-shell) and follows the Siegbahn notation.

In Table 2.4, the energies and branching fractions of the electrons emitted by ^{131}I are shown. β_1^- to β_6^- shows the average energies of the β^--particles emitted as indicated in Fig. 2.15. Also, for example, CE_K, γ_1 represents a conversion electron originating from the K-shell and is competing with γ-transition γ_1 as depicted in Fig. 2.15, resulting in an electron of energy $E_{CE_K}, \gamma_1 = 0.04562$ MeV with binding energy of the K-shell electron having been taken into account. The conversion electrons, in contrast to the continuous energy distribution of the β^--particles, have a sharp energy peak.

The energies and branching fractions for photons and electrons emitted by ^{131}I as shown in Tables 2.3 and 2.4, respectively, were obtained from ICRP Publication 38 [ICR83]. The sums of the branching fractions add up to 1.0404 and 1.0596 for Tables 2.3 and 2.4, respectively, and not to exactly one. This due to experimental uncertainties in measurements of the branching fractions [ICR83] and is accounted for in the MCNP calculations by normalizing it to one.

Summary of choice of ^{131}I

Although a variety of isotopes are available due to the fission process, ^{131}I was chosen for a variety of reasons. As said previously, the ^{131}I isotope is well known to contribute to a large percentage of dose during an accident. Although other isotopes are more radiotoxic and have a longer half-life, e.g. ^{92}Sr, the release fractions of these isotopes are considerably lower. Of the noble gases only the external radiation of these is of importance. Because they are quite volatile, they are also expected to leave the building sooner than iodine and, therefore, are not considered such a great risk to an emergency worker than iodine. Iodine can stay in the reactor hall for a longer time and can plate-out against the surfaces, e.g. walls, of the reactor hall. It is, thus, more probable that an emergency worker would be exposed to iodine than to the noble gases.

Table 2.3: Photon energies and branching fractions emitted during the radioactive transformation of ^{131}I [ICR83], see also Fig. 2.15.

Origin of Radiation	Photon Energy (MeV)	Branching Fraction (no units)
γ_1	8.018×10^{-2}	2.62×10^{-2}
γ_4	1.772×10^{-1}	2.65×10^{-3}
γ_7	2.843×10^{-1}	6.06×10^{-2}
γ_{12}	3.258×10^{-1}	2.51×10^{-3}
γ_{14}	3.645×10^{-1}	8.12×10^{-1}
γ_{16}	5.030×10^{-1}	3.61×10^{-3}
γ_{17}	6.370×10^{-1}	7.27×10^{-2}
γ_{18}	6.427×10^{-1}	2.20×10^{-3}
γ_{19}	7.229×10^{-1}	1.80×10^{-2}
K_{α_1}	2.978×10^{-2}	2.59×10^{-2}
K_{α_2}	2.946×10^{-2}	1.40×10^{-2}

Table 2.4: Electron energies and branching fractions emitted during the radioactive transformation of ^{131}I [ICR83], see also Fig. 2.15.

Origin of Radiation	Electron Energy (MeV)	Branching fraction (no units)
β_1^-	0.06935	0.0213
β_2^-	0.08693	0.0062
β_3^-	0.09660	0.0736
β_4^-	0.19150	0.8940
β_6^-	0.28320	0.0042
CE_K, γ_1	0.04562	0.0363
CE_{L_1}, γ_1	0.07473	0.0043
CE_K, γ_7	0.24970	0.0025
CE_K, γ_{14}	0.32290	0.0155
CE_{L_1}, γ_{14}	0.35900	0.0017

2.6 An analytical method derived for comparison with MCNP

Whilst MCNP was used as the main method for calculating the external photon and electron doses to a worker, another method was used to show the adequacy of the MCNP code results.

This method uses a hemispherically shaped cloud of ^{131}I in an open space of air. A dose receptor is placed at the origin of the hemispherical cloud. An analytical equation for calculating the dose to the receptor was developed by Van Rooyen [ROO03]. The analytical equation was solved numerically using the computer programme MathCad [MAT05].

The contribution of photon radiation of each discrete volume *dV* of gas to the person (receptor) standing in the centre of the cloud is added. This method is very similar to the so-called "point kernel method". The analytical method is explained below.

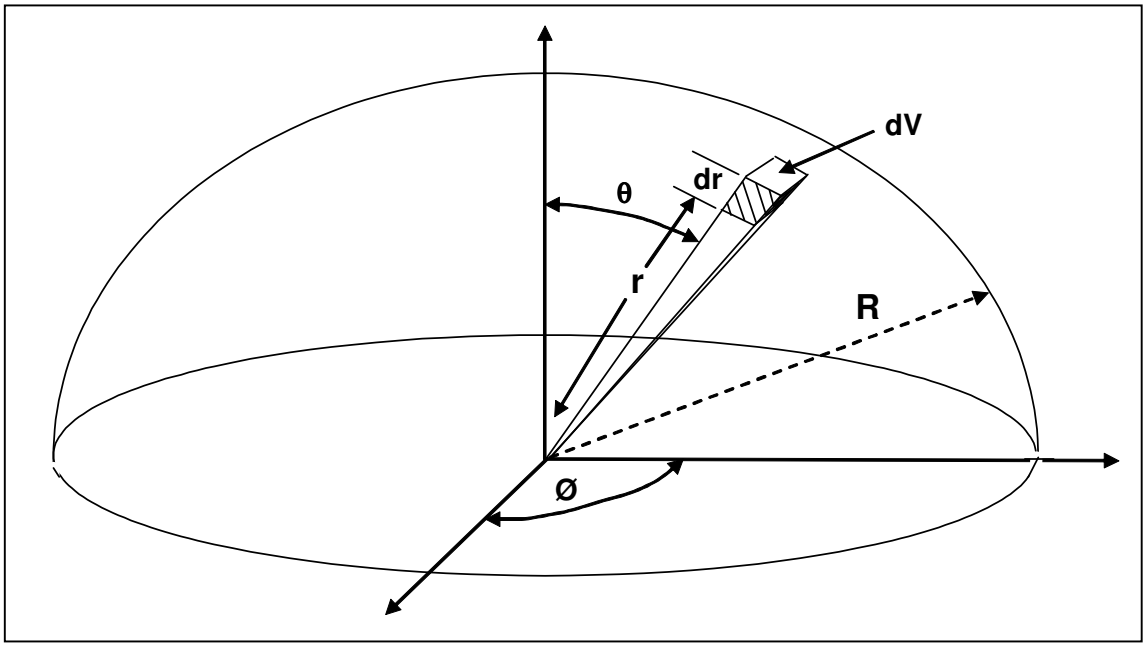

Figure 2.16: A schematic representation of parameters involved in the analytical method derived for comparison with MCNP for a Hemispherical Cloud of ^{131}I.

Figure 2.16 represent the hemispherical cloud of ^{131}I and the space parameters involved. A small discrete volume *dV* is shown, made up by three dimensions, which are

functions of the radial distance *r*, the polar angel θ, and the azimuthal angle $\emptyset$. The radioactivity of ^{131}I in each *dV* results in defined dose rate $d\dot{D}$ at the receptor at the origin. The $d\dot{D}$'s are integrated to get the total dose rate $\dot{D}$ received by the receptor.

To calculate $\dot{D}$, a method is first derived to calculate $d\dot{D}$. The starting point of the derivation will be to obtain the dose *dD* per decay disintegration, which will then be related to $d\dot{D}$. To simplify the calculation, *dD* is given for a vacuum, after which the effect of the attenuating medium (air) is brought into effect.

In Fig. 2.16, the volume of infinitesimal *dV* is given by:

$$dV = dr \cdot r d\theta \cdot (r \sin\theta \cdot d\emptyset). \tag{2.4}$$

Different energy photons emitted in the decay of ^{131}I have a different resultant dose at the receptor. Therefore, the energies of the photons and the corresponding branching fraction, as given in Table 2.3, must be taken into account. The flux at a distance *r* from the point source *dV* is inversely proportional to r^2. Due to the inverse square law, the dose due to a particular energy photon dD_i can be calculated as:

$$dD = \frac{(\Re(E_i) \cdot Y_i)}{4\pi r^2}, \tag{2.5}$$

where $\Re(E_i)$ is the flux-to-dose conversion factor as a function of photon energy, obtained from [ANS91a] and Y_i the branching fraction of the specific photon energy.

Then, the total dose *dD* from the activity in *dV* can then be given as:

$$dD = \frac{\sum_{i=1}^{n} (\Re(E_i) \cdot Y_i)}{4\pi r^2}, \tag{2.6}$$

where *n* is the total number of distinct energy photons emitted as given in Table 2.3.

If the same case is now to be viewed bringing into account the effect of the attenuating medium and, in particular, the effect of absorption of photons (see Section 2.3.3), Eq. (2.6) can be adjusted to:

$$dD = \frac{\sum_{i=1}^{n} \left(\mathcal{R}(E_i) \cdot Y_i \cdot e^{(-\mu_i \cdot r)} \right)}{4\pi r^2} \quad , \tag{2.7}$$

where μ_i is the linear attenuation coefficient of air for photon energy *i*.

If the effect of build-up, due to scattering (as explained in Section 2.3.3) is to be included, the Build-up Factor (*B)*, as a function of E_i, the radius (*r*) and the mean free path (mfp_i) of the specific energy photon, can be included in Eq. (2.7) as follows:

$$dD = \frac{\sum_{i=1}^{n} \left(\left(\mathcal{R}(E_i) \cdot Y_i \cdot e^{(-\mu_i \cdot r)} \right) \cdot B\left(E_i, \frac{r}{mfp_i} \right) \right)}{4\pi r^2} \quad , \tag{2.8}$$

where

$$mfp_i = \frac{1}{\mu_i} \quad . \tag{2.9}$$

In order to define the dose rate, the activity (in Bq) in *dV* must be known. The activity concentration in the cloud (A_c), calculated as if homogenously distributed in the cloud, based on the total activity is given by:

$$A_c = \frac{A_{Total}}{V} \quad , \tag{2.10}$$

where

A_{Total} = the total activity in the hemispherical cloud and

V = the volume of an hemisphere with radius R (Fig. 2.16).

The activity (dA_c) in *dV* is the activity concentration (A_c) in the hemisphere multiplied by *dV*:

$$dA_c = A_c \, x \, dV \, . \tag{2.11}$$

As already indicated, *dD* is the dose (expressed in Sv) per disintegration. The number of disintegrations in *dV* is given by dA_c (units in Bq or s^{-1}, which imply disintegrations per second) and therefore $d\dot{D}$ can be written as:

$$d\dot{D} = dA_c \, x \, dD \, . \tag{2.12}$$

Combining Eq. (2.8) and Eq. (2.12) gives:

$$d\dot{D} = dA_c \cdot \frac{\sum_{i=1}^{n}\left(\left(\mathcal{R}(E_i)\cdot Y_i \cdot e^{(-\mu_i \cdot r)}\right)\cdot B\left(E_i, \frac{r}{\mathrm{mfp_i}}\right)\right)}{4\pi r^2} . \quad (2.13)$$

Substituting Eq. (2.4) and Eq. (2.11) into Eq. (2.13) gives:

$$d\dot{D} = A_c \cdot \frac{dr \cdot r d\theta \cdot (r \sin\theta \cdot d\emptyset)}{4\pi r^2} \cdot \sum_{i=1}^{n}\left(\left(\mathcal{R}(E_i)\cdot Y_i \cdot e^{(-\mu_i \cdot r)}\right)\cdot B\left(E_i, \frac{r}{\mathrm{mfp_i}}\right)\right). \quad (2.14)$$

To obtain the total contribution ($d\dot{D}$) of each dV to the receptor at the origin, integration has to take place over the radius (r), $d\theta$ and $d\emptyset$. Taking into account that the integral of $d\emptyset$ from 0 to 2π gives 2π, and that the integral of $\sin\theta \cdot d\theta$ from 0 to $\pi/2$ gives 1, the following integral is obtained to calculate $\dot{D}$:

$$d\dot{D} = \frac{1}{2} A_c \cdot \int_0^R \left[\sum_{i=1}^{n}\left(\left(\mathcal{R}(E_i)\cdot Y_i \cdot e^{(-\mu_i \cdot r)}\right)\cdot B\left(E_i, \frac{r}{\mathrm{mfp_i}}\right)\right)\right] dr. \quad (2.15)$$

The total dose rate to a receptor at the origin or to a person standing in the middle of a hemispherical cloud is therefore determined. The results of the numerical solution of Eq. (2.15) are presented in Chapter 3. An exact representation of such a case was also duplicated, for comparison, in the MCNP code and the result is also presented in Chapter 3.

2.7 The MCNP Code

The Monte Carlo N-Particle (MCNP) code is an internationally recognized code for analysing the transport of photons and neutrons. The code is developed and maintained by the Los Alamos National Laboratory [SHU07]. In addition to analysing the transport of photons and neutrons, the transport of secondary γ-ray photons resulting from neutron interactions are treated. The transport of electrons can also be treated, both as primary particles and secondary particles resulting from photon interactions. Protons can also be handled.

The code can be used for many applications e.g. accelerator applications (neutron and photon transport), detector design, radiation protection, nuclear medicine, radiochemistry, reactor core calculations (mainly neutron transport), criticality calculations, tomography and radioactive waste handling and disposal. In the case of this study, the code was not used for reactor-core neutron transport calculations but for photon and electron transport resulting from the ^{131}I that was released from the reactor core due to the accident defined.

The programme can be used for calculating dose rates, fluence rates, criticality parameters, energy deposition and radiation damage. Dose rates were calculated in the model prepared for the present study.

2.7.1 A brief description of the Monte Carlo radiation transport method applied by the MCNP code

MCNP is a probabilistic radiation transport code that simulates real particle transport by following track lengths (with probabilistic length distribution), reaction type distribution (probability dictated by real branching ratios) and scattering direction (distributed as per reaction type specifics) at each vertex, for many particles. This includes primary and secondary particles created at the vertices. Real cross-section data from extensively tabulated and up-dated files provides realistic radiation related quantities such as dose rate, provided that a sufficient number of particle tracks have been followed.

To explain the principle of tracing particles through a system, a simple diagram, shown in Fig. 2.16, has been set up. The diagram represents a concrete building where photons originated from a gaseous source that is present inside the air volume. The movements of these are tracked individually. Some photons interact with the air. Other photons interact with the concrete. Other photons leave the building to the "Outer world" outside the building where it may not be necessary to track them anymore. The interactions that can take place for each photon in the materials, air and concrete, are absorption and scattering as explained in Section 2.3.3. Cross-section data, which is the probability of a certain interaction at a given photon energy, are used by the MCNP code to determine the interaction history of each particle. The history for each particle is recorded.

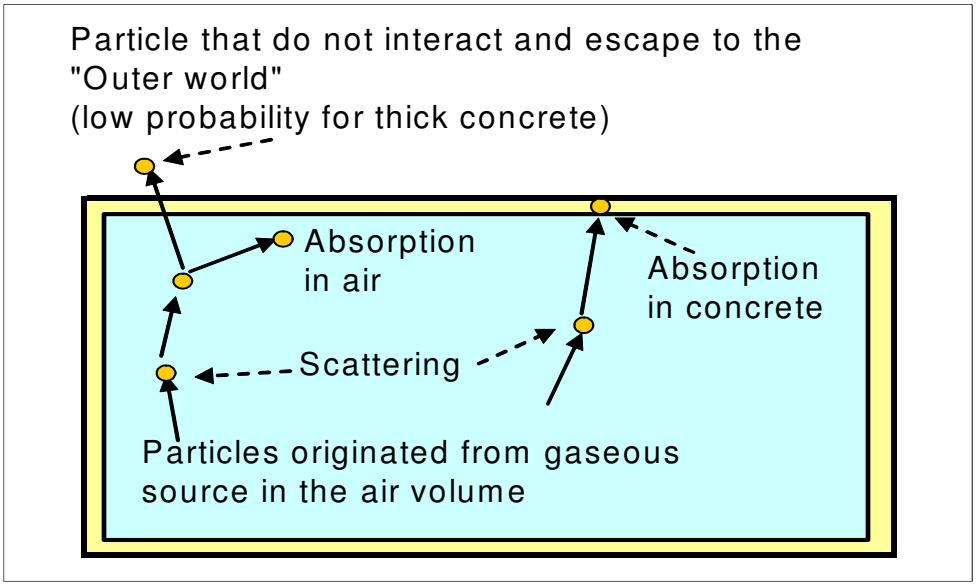

Figure 2.17: Diagram to show the principle of tracing particles through a system.

In a sense, MCNP is the closest thing to a real-life particle random-walk simulation. It gives very realistic results that include extremely complicated scattering behaviour such as build-up effects, hot-spots, spectral characteristics, distribution of radiation types, etc. As such, the code has the ability to follow millions of individual particle tracks over a region with almost arbitrary complex composition and geometry and the dose (for primary or secondary radiation) can then be tallied up where needed. The disadvantage lies in the fact that an extreme number of starting particles need to be followed to provide enough statistics for an accurate dose calculation in certain areas, such as areas behind high attenuation shielding. There exist enhancement techniques to increase statistics in MCNP to accelerate calculations in such cases. This is called “Variance reduction”.

If one could determine the exact path each particle makes and energies it assumes while passing through a medium in its random-walk fashion, one could, in principle, calculate many useful quantities by averaging over a large number of individual particle histories so as to minimise the stochastic effects of the individual particle interactions. For example, the probability that a particle in a certain energy range will be absorbed in a certain volume could be estimated by computing the proportion of all particles, in the specific energy group, that terminates in the specified volume. This concept of using a

large number of particle histories, each of which is random in nature, to estimate some average particle behaviour, is the essential feature of the Monte Carlo method.

In Monte Carlo calculations, the particle tracks or histories are generated by simulating the random nature of the particle interactions with the medium. To do this, one requires mathematical expressions for the probability relationships which govern the track length of an individual particle between interaction points, the choice of an interaction type at each such point, the choice of a new energy and a new direction if the interaction is of a scattering type, and the possible production of additional particles. These are all stochastic variables. One needs a complete understanding of the physics of the various processes a particle undergoes in its lifetime from the time it is born in the source until it is either absorbed or leaves the system under consideration, in order to make selections of the specific values for these variables.

In some cases, there are equations that adequately describe the behaviour of such systems and that can be solved either analytically or numerically. The question is asked why Monte Carlo methods are used.

Comparison with deterministic methods

The fundamental advantage of Monte Carlo techniques over deterministic techniques (i.e. numerical solutions to the Boltzmann Transport Equation (BTE)), is that Monte Carlo techniques more accurately represent the geometry and the nuclear data than deterministic techniques. Deterministic techniques require reasonably simple geometries for the numerical technique to work, and use the multigroup approximation to cross-section data. Monte Carlo techniques can handle complex geometries and continuous cross-section data (as well as simple geometry and multigroup data). The disadvantage of the Monte Carlo technique is that it is statistical in nature and does not provide an exact solution to the problem. All results represent estimates with associated uncertainties. Also, Monte Carlo techniques can be quite time-consuming on a computer if small uncertainties are required.

The relationship between Monte Carlo techniques and deterministic techniques is: deterministic techniques provide a highly exact solution to a significantly simplified

approximation of the problem, while Monte Carlo techniques provide an approximate solution to a highly exact representation of the problem.

The Monte Carlo code MCNP operates very differently from deterministic transport methods and codes. Deterministic methods, the most common of which is the Discrete Ordinate Method [SHU07], solve the Boltzmann transport equation for the average particle behaviour. In contrast, MCNP does not solve an explicit equation, but rather obtains answers by simulating individual particles histories and recording some aspects (tallies) of their behaviour. The average behaviour of particles in the physical system is then inferred using the central limit theorem from the average behaviour of the simulated particles. MCNP supplies information only about specific tallies requested by the user.

CHAPTER 3

APPLICATION OF THE MCNP CODE TO ^{131}I ACCIDENT CONDITIONS, RESULTS AND DISCUSSION

The main aim of this chapter is to provide results that were obtained from calculation with the complete MCNP model, representing the reactor hall.

The comparison of the calculation of the dose to a receptor situated at the origin of a hemispherical cloud will be, however, presented first. This includes the results of both the numerical solution of the analytical method presented in Section 2.6 and those of a simple MCNP model representing only a hemispherical ^{131}I cloud with a receptor at the origin. The results of these are presented in Section 3.1. Thereafter, a simple MCNP example that contains the necessary principles is given in Section 3.2. The complete MCNP model will then be presented in Section 3.3.

External photon dose calculations using the complete MCNP model will be presented in Section 3.4. For these calculations, it was important to optimize the running time. Therefore, a run to establish the optimum running time was first performed. Thereafter, runs with varying radii of the ^{131}I cloud originated from above the reactor pool were performed. The dose rates to eight persons at different positions in the reactor hall, called phantoms in the calculations, were obtained and compared for varying radii of the ^{131}I cloud. The results were also compared with the dose rates and doses to eight persons at the same position, but assuming that the complete reactor hall is filled with the same amount of ^{131}I activity, i.e. that the ^{131}I has dispersed to fill the entire reactor hall. The photon calculations represent the external photon dose obtained by persons in the reactor hall during the time of the accident.

Finally, in Section 3.5, electron β^--decay calculations representing the skin dose to persons at the same positions are presented. This was done for a ^{131}I cloud of fixed radii and compared with the results when the complete reactor hall is filled with ^{131}I.

3.1 Comparison of analytical method with MCNP using a hemispherical cloud

3.1.1 Solving the analytical method

As stated in Section 2.6, the analytical method (Eq. 2.15) developed for calculating the dose rate at a receptor placed at the origin of a hemispherical ^{131}I cloud was solved numerically with MathCad. The input parameters are summarized in Table 3.1.

The total activity (A_{Total}) of the ^{131}I cloud used is these of a 26 fuel element reactor core and was taken from Table C.1 of Appendix C as 1.04 x 10^{16} Bq. Therefore, the activity concentration is given by Eq. 2.10 as:

$$dD = \frac{1.04 \times 10^{16}\ [\mathrm{Bq}]}{\frac{4\pi \cdot 20^3}{3}[\mathrm{m}^3]} = 3.10 \times 10^{11}\ [\mathrm{Bq\ m^{-3}}]\ . \qquad (2.5)$$

Table 3.1: Input parameters for the MathCad model of the analytical method.

Symbol	Parameter	Value	Units
R	Radius of Hemisphere	20	m
A_{Total}	Total activity in Hemisphere	1.04 x 10^{16}	Bq
A_c	Activity Concentration in Cloud	3.10 x 10^{11}	Bq m^{-3}
E_i	Energies of Photons emitted by ^{131}I	Obtained from Table 2.3	MeV
Y_i	Branching Fractions of Photons	Obtained from Table 2.3	no units
$\mathcal{R}(E_i)$	Flux-to-Dose conversions factors	Obtained from [ANS91a]	Sv cm^2
B	Built-up factor	Obtained from [ANS91b]	no units
μ_i	Linear attenuation coefficient	Obtained from [BER98]	m^{-1}

In the numerical solution of Eq. (2.15), compensation was made in the MathCad model for the consistent treatment of units. Also, a tolerance of 10^{-12} was set for the results of two consecutive runs to fall within. The programme starts to calculate the integral with initial values set. It then does another run and compares the answer with the previous answer to be within the tolerance. Continuing runs are performed until the difference in consecutive runs is within the tolerance. As can be seen, the tolerance is rather small and the accuracy of the answer of a high degree. The dose rate that was obtained is 4.405 Sv h^{-1}.

3.1.2 Results of simple MCNP model representing same geometry

As also stated in Section 2.6, an exact representation of the hemispherical ^{131}I cloud with a receptor at the origin was duplicated in the MCNP code for comparison. The results of a calculation done in one minute computer running time yielded a dose rate of 4.443 Sv h^{-1} with an error of 0.69%.

As can be seen from the above two calculated dose rates, the results of the two independent methods compare very favourably. This mean that although the MCNP code are widely applied and could therefore be applied with a large amount of confidence, the MCNP code can be regarded as suitable for this study.

3.2 A short explanation of how the MCNP code works with an example

In order to better explain the operation of the code and in preparation for understanding the actual model of the reactor and building, an example has been chosen that is relatively simple, but contains the necessary principles.

A box representing a building has been defined. Inside the building is a phantom, representing a human being. Shown in Fig. 3.1 is a representation of a vertical cut through the building and phantom.

Defining the Cell and Surface cards

The "Surfaces" of the model need to be defined in order to establish the boundaries of the bodies, called "Cells". To define the boundaries of the concrete walls, floor and roof of the building, an inner and outer box, corresponding to surfaces 2 and 3, respectively, have been defined. A cylindrical surface (Surface 1), defining the outside boundary of the phantom, has been placed right at the middle of the floor. A fourth surface (Surface 4) defines the outside boundary of the surrounding air. The information pertaining to surfaces can be found in the MCNP input data set generated as an example and is called "Surface Cards".

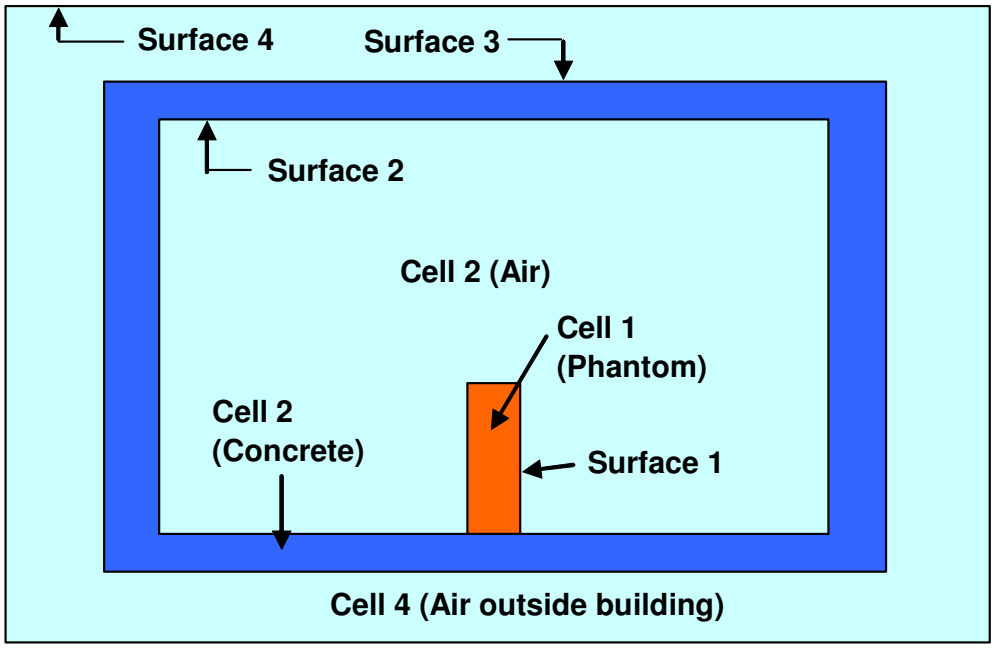

Figure 3.1: Sketch of Box MCNP Example.

The inner box's length is 1000 cm, the width is 600 cm and the height 300 cm. The MCNP code requires dimensions in centimetres as input data, while densities are in g cm^{-3}. The phantom is 170 cm high with a diameter of 23 cm.

The dimensions of the surfaces are defined in the MCNP input file according to a Cartesian coordinate system. The origin of this axial system can be chosen by the user. For this example, the origin is chosen at point C as indicated in Fig. 3.2. It would have been easier to set the origin at e.g. point A or point B. In the case of the origin being chosen at point A, the X dimensions would have been: X_{min} = 0 cm and X_{max} = 600 cm. The Y and Z dimensions would have been Y_{min} = 0 cm, Y_{max} = 1000 cm and Z_{min} = 0 cm, Z_{max} = 300 cm.

However, for compatibility with other radiation codes, e.g. with the point-kernel code QAD-GCCP, it is common to set the origin of the Cartesian coordinate system at a place like point C, which is at the bottom centre of the cylinder. This would have been needed, if the cylinder had defined the outside boundaries of a radioactive source, which is often the case.

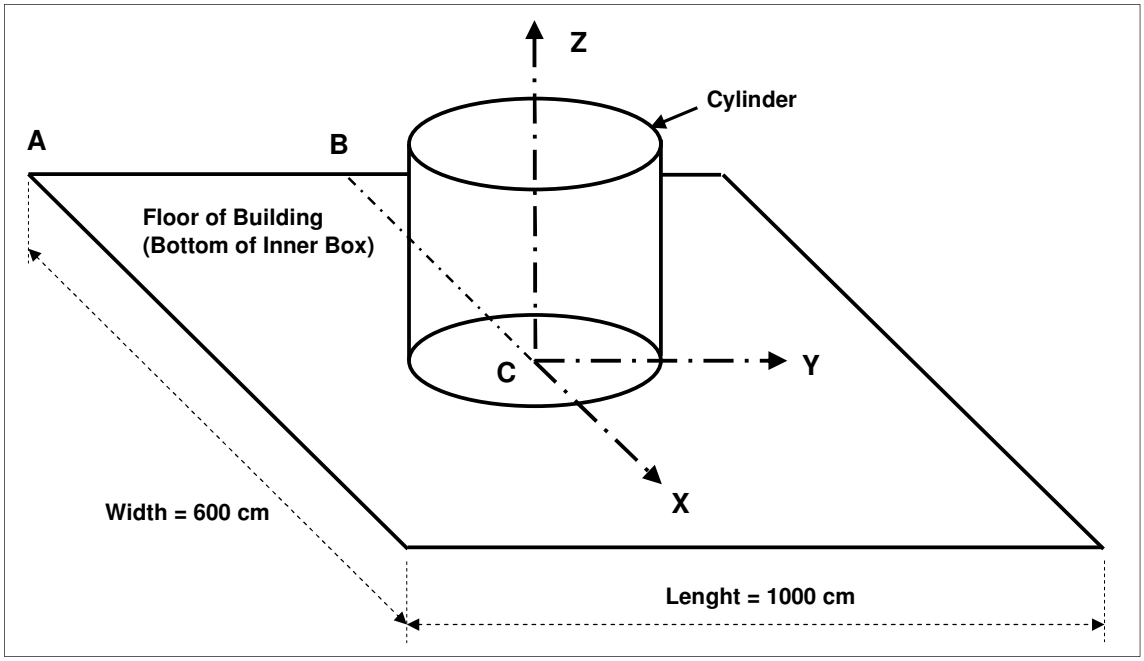

Figure 3.2: Box MCNP Example – Cartesian system.

The next step in developing an MCNP input model, after defining the Surface "Cards", is to define the "Cells" (called "Cell Cards"). All space is composed of contiguous volumes or cells [SHU07]. The surfaces are only the dimensions or constraints set as a framework to define bodies (cells) within. The cells in this example are the building walls (Cell 3, constrained by the inner and outer boxes) and the cylindrical phantom (Cell 1, constrained by the outer surface of the cylinder). Another cell (Cell 2) is defined as the air within the inner box but outside the cylinder. Cell 2 is filled with $^{131}1$ from where the cylindrical phantom receives its dose. The last two cells (Cell 4 and 99) are defined as the air outside the building but within Surface 4 and the "outside world" outside Surface 4, where the MCNP input data set defines the importance of tracing of particle movement as zero.

At this point, the part of input data set containing the Cell and Surface cards is shown in Table 3.2. Because of the history of the development of the MCNP code, the Cell cards appear first, before the Surface cards, and not the other way around, which would have been more logical.

The first line contains the title, "MCNP Example". The second line contains a comment for the user of the MCNP programme to remember that Cell cards follow. All text

Table 3.2: MCNP Input Data Set – Cell and Surface cards.

```
MCNP Example
c Cell Cards
01  1  -1          -1        imp:p=1  $  Phanton
02  2  -1.21E-3    +1  -2    imp:p=1  $  Air in Room (I-131 filled)
03  3  -2.35       +2  -3    imp:p=1  $  Concrete Wall
04  2  -1.21E-3    +3  -4    imp:p=1  $  Outer Air
99  0              +4        imp:p=0  $  Umwelt (Outer world)

c Surface cards
01  RCC  0 0 0   0 0 170  11.5          $  Phanton
02  RPP  -300 300  -500 500    0  300   $  Inner wall
03  RPP  -330 330  -530 530   -30 330   $  Outer wall
04  RPP  -400 400  -600 600  -100 400   $  Outer air border
```

following the letter "c" in a line is regarded as comments and is not used by the programme.

The third line contains the first Cell card, the fourth line the second Cell card and so on. The numbers of the Cell card in the third line are explained as an example.

- If virtual columns are assigned to the numbers in this line, the number 01 is in the first "column". This is the number of the cell, given arbitrary, but in sequential order. It does not make a difference whether it is captured as 1 or 01 in the input data set.
- The number in the second "column" is the material number that "fills" the cell, in this case material number one, which is tissue equivalent material. The materials are later defined in the input data set.
- The number in the third "column", in this case 1, indicates the density of the material and the negative sign means that the units are in g cm^{-3}.
- The fourth "column" is reserved for reference to the surface numbers that forms the boundaries of the cell. A negative sign means that the cell is inside the applicable surface and a positive sign that the cell is outside the surface. The -1 in this case thus means that Cell 01 is inside Surface 1, which is a cylinder (called an RCC).
- The importance of tracing particles in the particular cell is set in the fifth "column". Everything after the $ sign is also regarded as comments.

The Surface cards follow the Cell cards. Again, the Surface number follows first. Thereafter the type of macrobody is defined. An RPP is a rectangular box. In the case of an RPP, the x, y and z coordinates are defined in the order x_{min}, x_{max}, y_{min}, y_{max} and z_{min}, z_{max}. In the case of an RCC surface, the x, y, and z coordinates of the bottom centre are defined, thereafter the x, y and z coordinates relative to the origin and thereafter the radius.

Defining the Data cards

After the Cell and Surface cards, the Data cards follow. These contains radioactive source information, material information, the way in which the photons are counted or tallied etc. The first part of the Data cards region, i.e. the source definition, is presented in Table 3.3. The mode of transport is set to p (photons). The "sdef" cards contain the summary of the source definition. The particle to be tracked is photons (par=p). The energies of the photons and emitted by the ^{131}I source and its corresponding branching fractions (abundances) are provided in distribution 4 (erg=d4). The energies in MeV are found after "si4 L" and the corresponding branching fractions after "sp4". Note that the photon energies and corresponding branching fractions for ^{131}I were obtained from Table 2.3 and are also used for the complete model described in Section 3.3. Sampling, i.e. tracing of particles originating from the source volume, is restricted to distributions 1, 2 and 3 (x = d1, y = d2, z = d3) for the x, y and z dimensions, respectively.

The remainder of the Data cards, namely the Material and Tally cards, are shown in Table 3.4. Each material is specified by means of defining the atomic numbers and mass or atomic fraction of each element it is made up of. For example, material 1 (m1) contains oxygen, with atomic number 8 represented by 8000. The mass fraction, indicated by the negative sign, is 6.143×10^{-1}. A positive sign would have meant an atomic fraction.

The text following the "fc16" is the Tally cards. This defines the way in which the particles arriving at the counting position, in this case defined by the phantom which is Cell 1, must be counted. The 1 in "fc16" means that it is Tally number 1 (more than one tally can be defined) and the 6 means that it is a tally of type F6, which implies that energy deposition due to photon interaction is measured. The MCNP codes provide, for

Table 3.3: MCNP Input Data Set – Data cards – Source definition.

```
c Data Cards
c Mode: transport photons
mode p
sdef par=p erg=d4 x=d1 y=d2 z=d3
si1   -300 300
sp1   0   1
si2   -500 500
sp2   0   1
si3   0  300
sp3   0   1
si4 L  2.946E-02  2.978E-02  8.018E-02  1.772E-01  2.843E-01
       3.258E-01  3.645E-01  5.030E-01  6.370E-01  6.427E-01
       7.229E-01
sp4    1.400E-02  2.590E-02  2.620E-02  2.650E-03  6.060E-02
       2.510E-03  8.120E-01  3.610E-03  7.270E-02  2.200E-03
       1.800E-02
```

type F6 tally, the units of energy position in MeV/g. If it is further assumed that the source emits particles at a certain tempo per second, corresponding to its activity in Bq, the MCNP units become MeV/g/s. The answer must, therefore, be multiplied by the Tally Multiplication Factor of 2.2201 x 10^4 to arrive at Gy/h which the code does. Gy/h is equivalent to Sv/h, and can be used as a measure of radiation damage due to energy deposition of the ionizing radiation.

The running time is defined in the input data set and is in this case 10 minutes (indicated by "ctme 10"). By running the code, an output data set is generated.

3.3 Setting up the complete MCNP model

After some explanation of the MCNP code by way of a simple example applicable to the field it was applied to in this study, the complete model will now be introduced. The input data set for ^{131}I external radiation photon tracking is presented in Appendix B. Here reference was made to old drawings on which the dimensions were still indicated in inches – since the reactor was built in the 1960s – obtained from the NECSA Nuclear Facilities' Drawing Office. The main dimensions of the "reactor hall", the large room in which the reactor pool has been built, have been identified. To make provision for the proper simulation of particles with MCNP, the concrete shielding and outside walls, floor and roof of the reactor building had to be defined. For each of these parts of the reactor hall an inner and outer surface card had to be defined; the inner surface card

Table 3.4: MCNP Input Data Set – Remainder of Data cards.

```
c Material Data
m1      8000    -6.143E-01      $ TEM = Tissue-Equivalent Material
        6000    -2.286E-01      $ TEM
        1000    -1.000E-01      $ TEM
        7000    -2.571E-02      $ TEM
       20000    -1.429E-02      $ TEM
       15000    -1.114E-02      $ TEM
       19000    -2.000E-03      $ TEM
       16000    -2.000E-03      $ TEM
       11000    -1.429E-03      $ TEM
       17000    -1.357E-03      $ TEM
       12000    -2.714E-04      $ TEM
       26000    -6.000E-05      $ TEM
m2      6000    -1.24E-4        $ Air, dry.
        7000    -0.755267       $ Air, dry.
        8000    -0.231781       $ Air, dry.
       18000    -0.012827       $ Air, dry.
m3      1000    -0.013          $ Type 04 Ordinary Concrete
        8000    -1.165          $ Type 04 Ordinary Concrete
       11000    -0.040          $ Type 04 Ordinary Concrete
       12000    -0.010          $ Type 04 Ordinary Concrete
       13000    -0.108          $ Type 04 Ordinary Concrete
       14000    -0.740          $ Type 04 Ordinary Concrete
       16000    -0.003          $ Type 04 Ordinary Concrete
       19000    -0.045          $ Type 04 Ordinary Concrete
       20000    -0.196          $ Type 04 Ordinary Concrete
       26000    -0.030          $ Type 04 Ordinary Concrete
c
fc16 Photon Absorbed Dose Rate inside the Phantoms
+f16  1    $ Tally 1
fm16  2.2201E4  $For comparison to 3.7E10 Bq = 1 Ci
c
c ==============
PRINT 10 40 50 60 72 100 110 120 170 200
c ==============
ctme 10
```

being the inside surface of the wall and the outer surface card being the outside surface of the wall. The walls of the reactor hall are 30 cm thick. Therefore, a much larger number of cells and surfaces were now defined.

A "source" cell, containing the radioactivity is also defined but is, in this case, either a hemispherical cloud of ^{131}I above the reactor pool or the whole reactor hall filled with ^{131}I gas. The materials of construction, mainly high density concrete of the building structure and aluminium for the reactor vessel were included in the definitions of the cells.

A vertical cross section of the reactor hall, as would be generated by the MCNP geometrical plotter, is shown in Fig. 3.3.

Eight "phantoms" consisting of tissue equivalent material, representing the workers that would potentially perform an emergency task or be accidentally trapped during an accident, were placed in the reactor hall as shown in Fig. 3. 4.

Additional drawings, with cross sections in other locations in the reactor hall are shown in Appendix A.

3.3.1 Source Term input part of the MCNP model

The available radioactive inventory of ^{131}I was taken, for a 26 fuel element reactor core, from Table C.1 of Appendix C as 1.04 x 10^{16} Bq. The release fraction was selected in Section 2.5.3 as 1. The radioactive inventory multiplied by the release fraction therefore provides a source term of 1.04 x 10^{16} Bq of ^{131}I. The energies and branching fractions for photons and electrons emitted by ^{131}I were obtained from Tables 2.3 and 2.4, respectively. A tally multiplication factor of 6.24 x 10^{9} s^{-1} was derived for type F6 tallies.

3.4 External photon dose results obtained with complete MCNP model

The complete MCNP model presented in Section 3.3 developed to determine photon and electron doses was executed using various parameters. In a real accident scenario, where radiological exposure from ^{131}I is of importance, a large part of the ^{131}I dose accrued by an exposed person is internal dose, specifically, thyroid dose. It has been, however, assumed that for the emergency scenarios investigated in the present study, that the emergency worker wears appropriate protecting clothing and, in particular, breathing apparatus that reduces thyroid intake by, typically, factors of a 100 to 10 000. It is, therefore, appropriate to evaluate the external dose rate for an emergency scenario where doses to the emergency workers are of importance.

As mentioned in the introductory paragraph of Chapter 3, the first step in evaluating the external doses was to optimize the running time. Therefore, a run to establish the optimum running time was first performed. Because of the complexity of the model

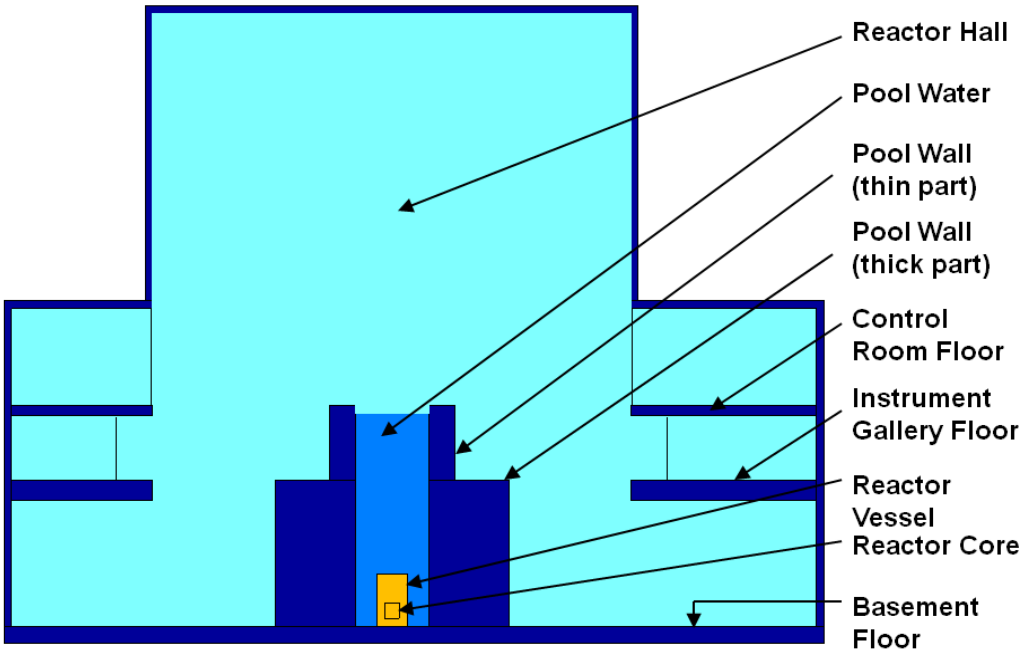

Figure 3.3: A vertical cross Section of the reactor hall generated by the MCNP geometrical plotter.

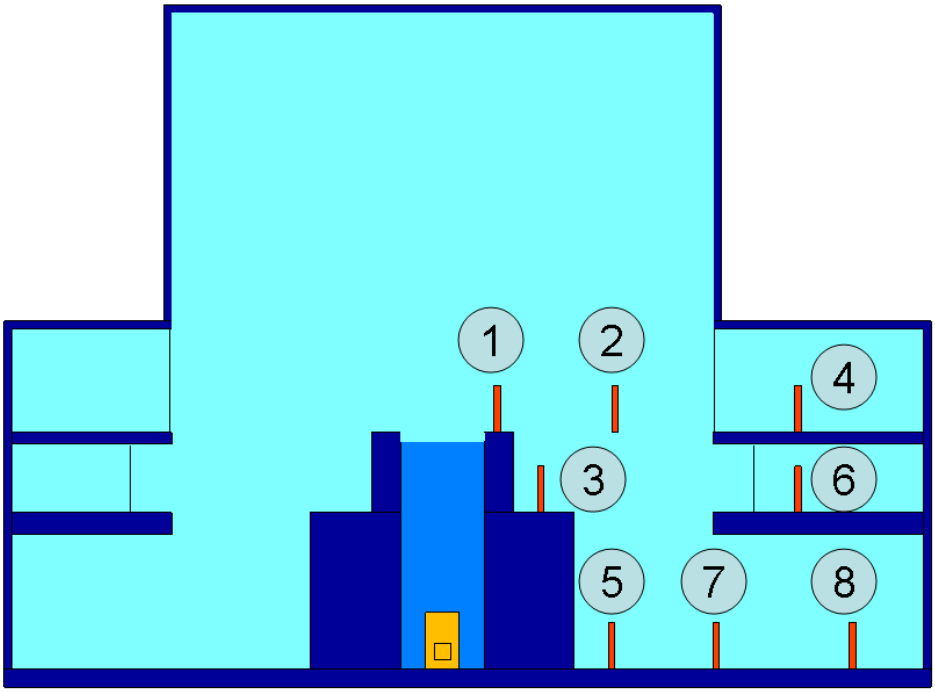

Figure 3.4: Eight phantoms representing workers placed at different positions in the reactor hall.

and the amount of concrete for shielding and structural purposes, the running times to arrive at an acceptable error had to be investigated in this step and is presented in the following section.

3.4.1 Determination of optimum MCNP model running time with a hemispherical cloud of radius 250 cm

A ^{131}I cloud with a radius of 250 cm was chosen to perform runs to determine the optimum running time. A representation of a cross section through the reactor hall with 250 cm radius ^{131}I cloud arising from the pool is shown in Fig. 3.5. In this chapter, where a hemispherical cloud is considered, it is always assumed that the activity of 1.04×10^{16} Bq as obtained in Chapter 2, is distributed homogeneously throughout the cloud. The external photon dose rates obtained with consequent MCNP runs are shown in Table 3.5. The associated errors are shown in Table 3.6.

From Table 3.5 it can be seen that the dose rates for all eight phantoms are already in the correct order of magnitude when compared to the results of longer processor running times. However, from Table 3.6, it can be seen that after a 2 minute run, large errors (in the region of 30%) still exist for the phantoms further away from the source.

In contrast, it can be seen that the error associated with Phantom 1, partially submerged into the cloud, is very small (1.5%) already after a short run of 2 minutes. This can be explained by the fact that many photons have "hit" the phantom in this time due to the short distance and have been absorbed into the phantom. Considering 4% as a maximum approximate acceptable error, it was decided to use 120 minutes as a benchmark processor run time for further runs determining photon dose rates from a hemispherical ^{131}I cloud of varying radius.

3.4.2 External photon dose Rates with a hemispherical cloud of varying radii

The purpose of these runs was to determine the doses to workers in the different positions as the size of the hemispherical ^{131}I cloud emerging from the pool increases with time as postulated during the accident.

Various runs starting with a radius of 150 cm and ending with a radius of 950 cm were performed. Here, for a hemispherical cloud, it is always assumed that the activity of 1.04×10^{16} Bq as obtained in Chapter 2 is distributed homogeneously throughout the cloud, regardless of the cloud radius chosen. To get a visual picture of when the radius of the cloud reaches 950 cm, it can be seen in Fig. 3.6 that both Phantoms 1 and 2 are completely submerged into the cloud. It is also of interest to note that at this stage some time has passed since the accident sequence has started. The results of these runs are presented in Table 3.7. The associated errors were all acceptable and are not presented.

As can be seen from Fig. 3.7, the dose rate at the position of Phantom 1 reaches its maximum when the radius of the hemispherical is approximately is approximately 250 cm. This can be explained in the following manner. When the cloud's radius is 150 cm, Phantom 1 is not yet submerged into the ^{131}I cloud. Only the external radiation from the cloud nearby has an influence on Phantom 1, resulting in a very high dose rate. As the radius of the cloud increases, the radiation emitted has an effect on other structures and phantoms. However, the dose rate at the position of Phantom 2 never reaches the high values of Phantom 1, even when submerged into the cloud. This can be attributed to the average concentration in the cloud decreasing as the cloud expands.

It is also of interest to note from Fig. 3.8 that the dose rate in the position of Phantom 3 undergoes a steep increase between radii of approximately 220 and 500 cm. From Fig. 3.4 it can be seen that Phantom 3 is positioned right underneath the cloud on the same level as the instrument gallery floor. However, when the radius of the cloud is still small as shown in Fig. 3.5, the thin part of the reactor pool shielding wall (as indicated in Fig. 3.3) provides effective shielding to Phantom 3. Phantom 4, always standing outside the cloud, is receiving a more or less constant dose, which can be explained by the long range of ionizing γ-photons in air.

3.4.3 External photon dose to eight phantoms due to complete reactor hall filled with ^{131}I

A run was also performed for the complete reactor hall being filled with a ^{131}I gas cloud. As can be seen from Table 3.8, the dose rates at the positions of Phantom 1 and 2 are 4.36 and 4.37 Sv/h, respectively whereas the dose of all the other phantoms are in the

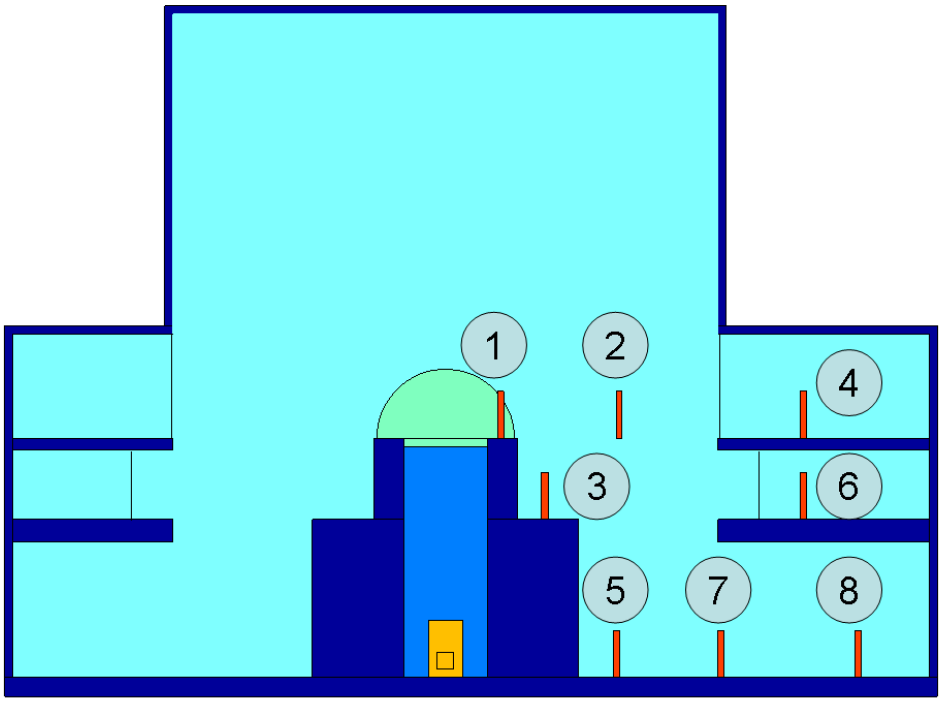

Figure 3.5: A cross section of the reactor hall with a 250 cm radius ^{131}I cloud arising from the pool.

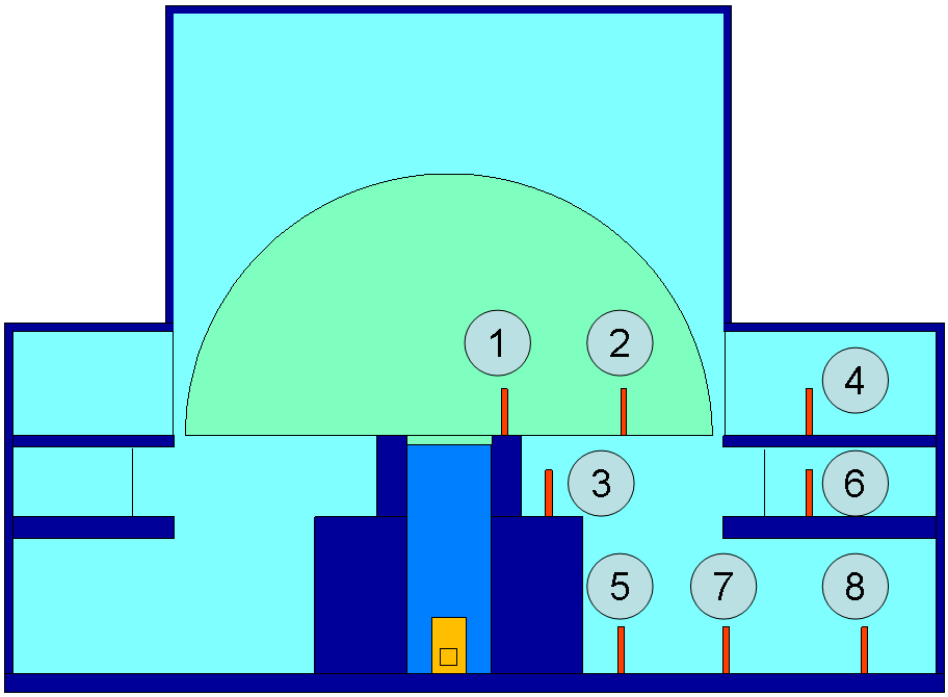

Figure 3.6: A cross section of the reactor hall with a 950 cm radius ^{131}I cloud arising from the pool.

Table 3.5: External Dose rates for different computer running times for a ^{131}I cloud radius of 250 cm.

	Dose Rates (Sv/h) obtained for Processor Running Times			
Phantom No.	**2 min**	**60 min**	**120 min**	**240 min**
1	1.62×10^{2}	1.67×10^{2}	1.67×10^{2}	1.67×10^{2}
2	1.09×10^{1}	1.16×10^{1}	1.16×10^{1}	1.15×10^{1}
3	7.48×10^{-1}	8.39×10^{-1}	8.46×10^{-1}	8.76×10^{-1}
4	3.26×10^{0}	2.60×10^{0}	2.67×10^{0}	2.67×10^{0}
5	1.70×10^{-1}	1.44×10^{-1}	1.46×10^{-1}	1.41×10^{-1}
6	1.41×10^{0}	1.61×10^{0}	1.55×10^{0}	1.58×10^{0}
7	2.36×10^{-1}	3.13×10^{-1}	3.25×10^{-1}	3.26×10^{-1}
8	3.10×10^{-1}	3.70×10^{-1}	3.85×10^{-1}	3.84×10^{-1}

Table 3.6: Errors for different computer running times for radius = 250 cm.

	Errors obtained for different Processor Running Times			
Phantom No.	**2 min**	**60 min**	**120 min**	**240 min**
1	1.5%	0.3%	0.2%	0.1%
2	5.4%	1.0%	0.7%	0.5%
3	15.4%	3.3%	2.2%	1.6%
4	9.8%	2.0%	1.3%	0.9%
5	29.8%	5.7%	3.8%	2.7%
6	13.9%	2.6%	1.8%	1.2%
7	29.1%	5.4%	3.6%	2.6%
8	28.1%	5.2%	3.6%	2.5%

order of 2 – 3 mSv. The dose due to a very short exposure time (10 minutes) was also calculated and it can be seen that it will be enough to result in deterministic health effects, considering that such effects start to take place at approximately 500 mSv. Indeed, for emergency planning, this may be compared to standards that exist for allowing emergency workers to perform certain life saving or protecting actions.

When comparing the hemispherical scenarios results (Table 3.7) with the full dispersion scenario results (Table 3.8) it is evident that, in the first case, the dose rates between Phantom 1 and 2 are significantly different whereas in the latter case it is almost the same. These two phantoms are positioned near each other. This phenomenon can be explained as follows. The exposure geometry of the hemispherical cloud is quite different from the full dispersion scenario. In the case of the hemispherical scenario of maximum radius (950 cm), the radioactive cloud is primarily above Phantoms 1 and 2 (see Fig. 3.6), although both phantoms are submerged in the cloud. Phantoms 1, however, is nearer to the centre of the cloud and the contribution from external γ-radiation from all sides is substantially larger than for Phantom 2, which only receives a large amount of γ-radiation from one side and less, due to the geometry, from another side.

In contrast, in the full dispersion scenario, the exposure geometry of both Phantoms 1 and 2 significantly increase, since the radiation cloud is now below as well as above the phantoms. However, Phantom 1, because of the reactor wall being directly underneath, is shielded partially from underneath. In comparison, Phantom 2, which does receive radiation from underneath, receives mainly radiation from one side. This results in the approximate same dose rates to these phantoms.

As already mentioned, all the other phantoms (Phantoms 3 – 8) receive doses in the order of 2 – 3 mSv. They are placed in more restricted volumes of the radiation cloud with all of these phantoms receiving approximately the same radiation from the main volume of the radiation cloud above and to one side.

Table 3.7: Dose rates to Phantoms 1 to 8 due to a hemispherical ^{131}I cloud of varying radius.

Radius of hemispherical ^{131}I cloud [cm]	Dose Rate [Sv/h] for Phantom No.							
	1	**2**	**3**	**4**	**5**	**6**	**7**	**8**
150	110.1	9.3	0.3	2.0	0.1	0.8	0.1	0.1
200	144.5	10.5	0.4	2.4	0.1	1.2	0.1	0.2
250	166.8	11.6	0.8	2.7	0.1	1.5	0.3	0.4
300	147.5	12.4	2.3	2.8	0.2	1.8	0.6	0.6
500	105.2	12.9	4.1	3.0	0.2	1.9	0.9	0.6
800	105.2	12.9	4.1	3.0	0.2	1.9	0.9	0.7
950	105.2	12.9	4.1	3.0	0.2	1.9	0.9	0.6

Table 3.8: External photon dose rate to phantoms in eight positions (see Fig. 3.4) for a ^{131}I gas cloud dispersed throughout the entire reactor hall.

Phantom No.	Dose Rate (Sv/h)	Error	Dose accrued in 10 minutes (mSv)
1	4.36	1.48%	727 ± 10.8
2	4.37	1.48%	728 ± 10.8
3	2.93	1.79%	488 ± 8.7
4	2.59	1.88%	432 ± 8.1
5	2.61	1.85%	435 ± 8.0
6	1.99	2.10%	332 ± 7.0
7	2.98	1.77%	497 ± 8.8
8	2.27	1.99%	378 ± 7.5

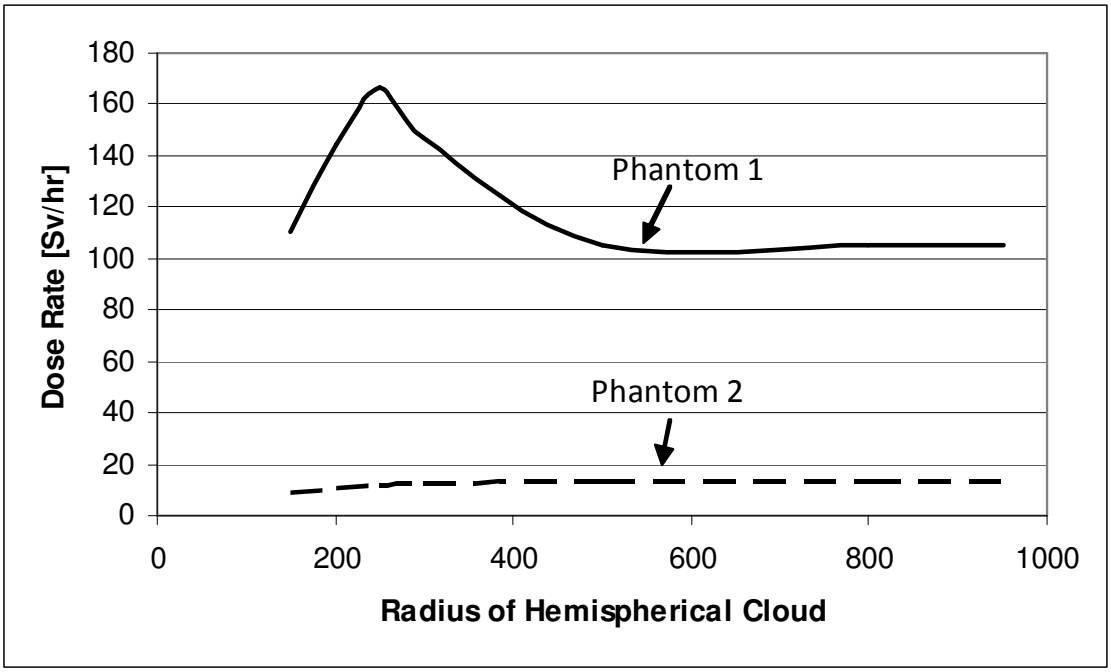

Figure 3.7: Dose rates to phantoms in positions 1 and 2 due to a hemispherical ^{131}I cloud of varying radius.

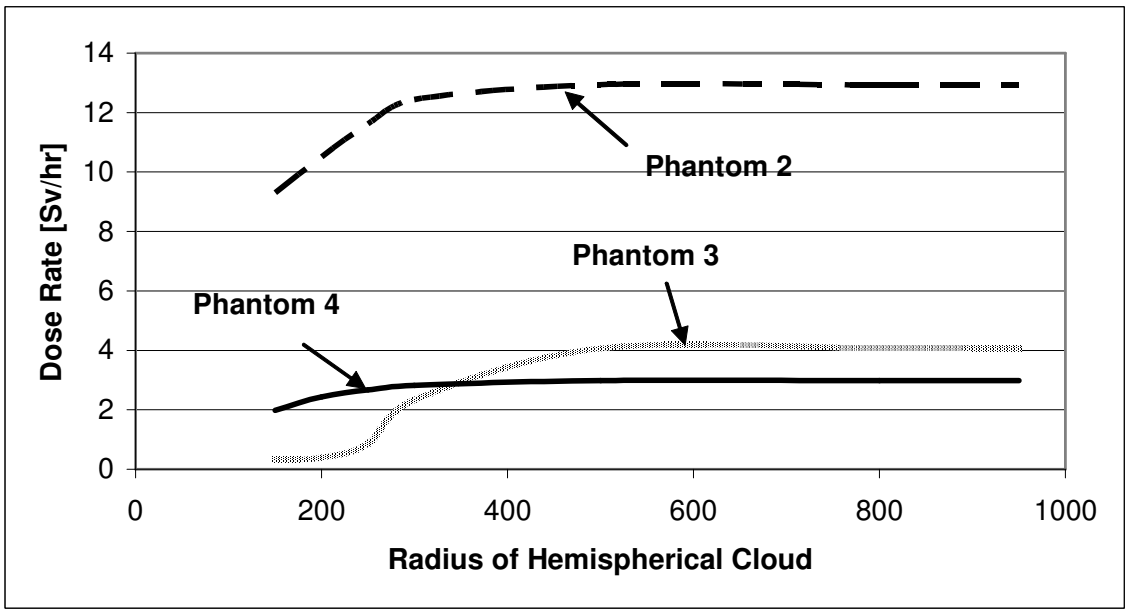

Figure 3.8: Dose Rates to phantoms in positions 2, 3 and 4 due to a hemispherical ^{131}I cloud of varying radius.

3.5 Skin dose results obtained with complete MCNP model

In this section, β^--exposure representing the skin dose to persons at the same eight phantom positions are presented. This was done for a ^{131}I cloud of fixed radii and compared with the results when the complete reactor hall is filled with ^{131}I.

3.5.1 Skin dose to eight phantoms due to β^--exposure from a hemispherical ^{131}I cloud of a fixed diameter of 250 cm

Two runs, one of 480 minutes and one of 3000 minutes were performed, using the same geometry as indicated in Fig. 3.4. The purpose was again to establish an optimum running time. However, in contrast to the runs performed for photons in Section 3.2.1, it was found that the results of the phantoms near the cloud rendered good statistics. On the other hand, the phantoms far away from the cloud rendered large errors, even despite of the relatively long running time of 3000 minutes. This can be easily explained by the travel distance of electrons in air being about one metre, on average. Therefore, enough electrons did not "hit" the phantoms far away to provide good statistics. The skin dose rates for the two runs are provided in Table 3.9.

It can be seen that very large skin dose rates are present for Phantom 1 partially submerged into the cloud, with a corresponding low error.

3.5.2 Skin dose to eight phantoms due to complete reactor hall filled with ^{131}I

Finally, results are presented of a 4000 minute run that was performed for the complete reactor hall being filled with a ^{131}I gas cloud. The results are shown in Table 3.10. Again, in order to get an indication of the skin dose to a person spending 10 minutes at the specified positions, the dose rates were multiplied by the exposure time and are presented in the last column.

As can be seen from the results in Table 3.10, for the short time exposed the β^- skin dose received in a single event will exceed the yearly skin dose limit of 500 mSv. It can also been seen from Table 3.10 that, due to the ^{131}I cloud that has dispersed throughout the reactor building resulting in a more homogeneous distribution of the electrons emitted from the ^{131}I, the skin dose accrued by the phantoms in all eight positions are all

Table 3.9: Skin dose to eight phantoms due to a hemispherical ^{131}I cloud of fixed diameter of 250 cm for two different running times.

	480 minute run		3000 minute run	
Phantom No.	**Dose Rate (Sv/h)**	**Error**	**Dose Rate (Sv/h)**	**Error**
1	1.32×10^{3}	0.4%	1.33×10^{3}	0.2%
2	3.65×10^{-2}	29.5%	2.84×10^{-2}	14.5%
3	2.44×10^{-3}	100.0%	2.15×10^{-3}	40.8%
4	0	0.0%	1.47×10^{-3}	52.7%
5	0	0.0%	6.69×10^{-4}	100.0%
6	4.00×10^{-3}	100.0%	1.83×10^{-3}	51.5%
7	0	0.0%	6.74×10^{-4}	100.0%
8	0	0.0%	4.98×10^{-4}	100.0%

Table 3.10: External β^- skin dose rate to phantoms in the specified eight positions for a ^{131}I gas cloud dispersed throughout the entire reactor hall (see Fig. 3.4).

Phantom No.	**Dose Rate (Sv/h)**	**Error**	**Dose accrued in 10 minutes (mSv)**
1	4.10	2.09%	684 ± 14.3
2	4.30	2.03%	717 ± 14.6
3	4.08	2.08%	679 ± 14.1
4	4.08	2.11%	679 ± 14.3
5	4.20	2.07%	700 ± 14.5
6	4.26	2.06%	710 ± 14.6
7	4.01	2.10%	668 ± 14.0
8	4.19	2.07%	698 ± 14.5

approximately the same. Because of the assumption that the cloud disperses uniformly throughout the hall, the short travel distance of electrons does not have such a large influence as with the runs performed in Table 3.9 and approximately equal electrons "hit" the phantoms in the radiation transport performed by the MCNP code.

It should also be noted that the external photon radiation will cause a radiation dose in excess of 500 mSv/h, which, on its own, will cause deterministic health effects. Although the electron skin dose is comparable with the external photon dose, it is not in the range of deterministic health effects, but only close to the yearly skin dose limit of 500 mSv/h.

CHAPTER 4

CONCLUSIONS AND RECOMMENDATIONS

The conclusions and recommendations resulting from the present study are presented in this chapter. Finally, a conclusion on the fundamental hypothesis formulated in Chapter 1 is drawn.

4.1 Conclusions from the present study

4.1.1 Aspects related to the Safety Analysis of the MTR Reactor

In order to properly demonstrate the use of the MCNP code, a substantial amount of background information, relevant to safety analysis and the physics of formation of radioactive inventory and the interaction of photons with matter had to be provided, and a real case example MTR was used to benchmark the code against. The steps that need to be performed, typical data that need to be obtained and the assumptions made during the safety analysis process had to be shown, which includes:

a. Obtaining the radioactive inventory

The manner of obtaining the radioactive inventory is explained in Sections 2.4.2 and 2.4.3 and is shown in Appendix C.1. The assumption that was made, related to the radioactive inventory during the time of the accident, was that all the fuel elements are at a burn-up of 80% which is, as stated in the text, a conservative assumption.

b. Release fractions and amount of fuel damaged

The choice of a release fraction and the consideration related to that was explained in Section 2.5.3. Definite assumptions, without going deeper in this field of study, were made in the choice of a 100% for ^{131}I. However, if one should choose to refine this part of a similar study, the influence of these assumptions could be done as part of an uncertainty and sensitively analysis. Such a uncertainty and sensitively analysis would include the release fraction chosen as well as the percentage of the reactor core, i.e. the amount of fuel, that was damaged with subsequent release of radioactive isotopes. Uncertainty and sensitively analysis is often performed for NPPs.

c. The effect of emergency ventilation and the effect on different categories of workers in the reactor hall

It was assumed that the normal extraction ventilation, which removes radioactive gases out of the building at a relatively fast rate, stops and that the emergency ventilation, which extracts the radioactive gases at a much slower rate come into operation. The electronic instrumentation and instrumentation system is programmed in this way for the example MTR. However, this protection measure, which is an engineered safety feature, is aimed at protecting primarily the potentially affected persons outside the reactor building, which includes the public, and does not necessarily take into account the potential exposure, which was shown to be quite high, to workers in the reactor hall. Among the workers in the reactor hall are emergency workers and users of the Experimental Beam Tube facilities on the basement floor. However, a distinction can be made with regards to former and the latter categories of persons that would be potentially present during the time of an accident:

i. Emergency worker

If immediate emergency actions, to be performed by emergency workers, are needed, e.g. the manual coupling of emergency core cooling water to keep cooling the exposed reactor core due to the LOCA taking place, the potential exposure in such cases will be quite high, as shown by the calculations. Although appropriate personnel protective equipment, e.g. suitable breathing apparatus, would be provided to such an emergency worker to avoid the inhalation of ^{131}I, such a person would still be exposed to external photon radiation due to the penetrating power through protective clothing of the energetic γ-ray photons of ^{131}I.

In a real emergency situation resulting from a LOCA, a direct shine path will develop from the reactor core with its remaining radioactive inventory if the emergency core cooling system could not be put into operation in time. This will add to the radiation exposure of the emergency worker as the reactor pool water acts as a biological shield against radiation.

ii. Experimental Beam Tube user

It was seen in the study that the noble gases, xenon and krypton will escape fast from the reactor core, through the building, to the environment due to its high volatility. In addition, the ^{131}I cloud will progress slowly, from the assumption of a small hemisphere, where the persons standing in the immediate vicinity (i.e. Phantoms 1 and 2), to the eventual scenario where the whole reactor hall is uniformly filled with ^{131}I gas. This will, therefore, allow enough time for an Experimental Beam Tube user to leave the building, if not trapped in the building due to other phenomena causing or accompanying the nuclear accident, e.g. in the case of a severe earthquake that caused the LOCA. As shown by the calculations in Chapter 3, the Experimental Beam Tube user will have enough time to evacuate the reactor hall until such time that the hemispherical cloud has grown to its full size (radius 950 cm). Due to ventilation extraction, it is estimated that such a cloud will take 30 minutes to reach size.

d. Opinion on the safety of the reactor

Although it was stated in the text and proven with various thermal-hydraulic calculations, e.g. as stated by Sharp and McCracken [SHA03] (See Section 2.1.4), that a hypothetical severe fuel damage event is seen as a Beyond Design Basis Accident (BDBA), i.e. the probability of severe fuel damage as considered in this study due to a LOCA, emergency planning is still needed. Emergency planning is the fifth level of defence in depth (the concept that was introduced under Principle 8 of the Safety Fundamentals introduction in Section 1.2.1) and also explicitly required under Principle 10.

Therefore, emergency planning actions need to be performed under accident conditions. For an existing MTR, retrospective assessments can be justified to ensure that no emergency action that are foreseen to be performed during a severe accident will require a worker to spent longer than a few minutes in the reactor hall. If the principle of optimization of protection, i.e. ALARA, is followed, it can follow that protecting systems are designed in such a way that no access to the reactor hall is required during a severe accident.

4.1.2 Appropriateness of the MCNP code

Although being widely applied as discussed in the introduction, the MCNP code can also be used effectively as part of the safety analysis process as shown in this study. Some conditions and considerations for code to be effectively applied are discussed below.

Comparison of the MCNP code with the Analytical Solution

As explained in Section 3.1 the results of the two independent methods of MCNP and the Analytical Solution compare quite good. This provides more confidence in the use of the MCNP code for the present study.

Running time of MCNP code

The MCNP code running time can be relatively long, for example as in the case for electron transport resulting from β^--radiation as presented in Section 3.5.2 where the ^{131}I gas has uniformly dispersed into the reactor hall. Therefore, the setting up of the MCNP model, performed by generating the MCNP input file, must be done with the aspects that influence running time in mind (e.g. the source volume, model volume, number of "radiation" tracks to be followed). The runs must also be carefully selected as to obtain the maximum insight per run.

Keeping the above in mind, the code can both be used for screening analyses in order to see where the highest expected doses are going to be and, once that is established, for further specific detailed calculations among which the analysis of expected emergency protective action doses falls.

The limitations of β^--exposure calculations

As seen from the present study, MCNP the code can effectively be used for external doses resulting from γ- radiation due to ionizing photons and can include personnel in most parts of the reactor hall. However, it is only meaningful to calculate skin doses due to β^--exposure if the worker is immersed partly or completely in the radioactive cloud or if the person is within approximately one metre of the radioactive cloud.

4.2 Further recommendations

Further recommendations are divided between aspects that are directly related to accident studies related to the present study that involves release scenarios and other related aspects, where the MCNP code can be applied.

4.2.1 Aspects directly related to accident studies involving release scenarios of research reactors

The following consideration should be kept in mind when planning to apply the MCNP code, together with a similar safety analysis method than presented in this study, in a wider manner:

The modeling of the external dose rates from the release of other fission gases from the reactor core

The volatile or semi-volatile gases expected to leak from the core during an accident, other than ^{131}I can also be modelled in the same manner used for the present study. This could provide additional insight to the total contribution to external radiation due to each isotope, and can be used as a type of a screening and sensitivity analysis. The isotopes mentioned could include some of the noble gas radioisotopes of xenon and krypton and other halogens.

The calculation of the complete dose

As done in the present study, the calculations are for external radiation dose only. The internal radiation dose was not determined but can be easily calculated using standardized methods and dose conversion factors for inhalation from, for example, Table II-III in the IAEA Basic Safety Series 115 [BSS96]. The code should, therefore, be used where external radiation dose is expected, i.e. not where the main part of the dose is expected to come from radioisotopes that pose an internal radiation-dose hazard when dispersed. As already stated in Section 4.1.1, Subsection c, Bullet i, the inhalation of radioactive can be largely prevented by using suitable breathing apparatus.

Selection of Accident Scenario

The particular accident scenario should be carefully selected. As is often the practice in safety analysis, the "bounding scenario", representing less-severe consequence accidents as well should be selected. The physical phenomena during such an accident should be well studied. This should then provide relevant input data in respect of source term for Level 2 analysis where the code can be applied. This will also provide relevant data regarding the distribution of radioactive gas in the reactor hall that varies e.g. as a result of ventilation patterns that is a function of ventilation rate and ventilation design.

Modeling in MCNP of important components that will have a larger influence on radiation transport

The important components of the reactor hall geometry should be identified. For example, thick shielding and other concrete walls is more important to model than small objects situated or standing around in the reactor hall and commensurate effort should be spent. For example, the reactor pool walls and floors are more important to model than loose trolleys or overhead cranes.

4.2.2 Other related aspects

Other aspects such as design, standard databases and the development of visual aided tools are briefly described below.

Because of the flexibility of the code, the code can be used for reactor design considerations. For example, the MCNP code can be used for the selection of reactor shielding materials in conjunction with or as a substitution for traditional methods as described in e.g. [ROC82l which entails the use of various graphs, nomograms and tables.

Standard geometries different types and designs of research reactors can be programmed into the MCNP computer code and put into databases for input into design changes and safety analyses when required.

The methodology developed in this study can be used wider and more effectively for different designs of research reactors by using visual aided MCNP code applications.

4.3 Conclusion on fundamental hypothesis

The methodology developed during the study and the MCNP model provides an illustration of the effective use of the MCNP code for Safety Analysis. With considerations provided in this chapter it can be concluded that the MCNP code can be added to the existing family of modern computational techniques that are used in support of safety analysis of research reactors. The technique presented in the study can, therefore, assist in performing advanced safety analysis and design optimizations that were not possible a few years ago as mentioned by Tewfik *et al*. [TEW08].

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APPENDICES

APPENDIX A – MCNP Model sketches

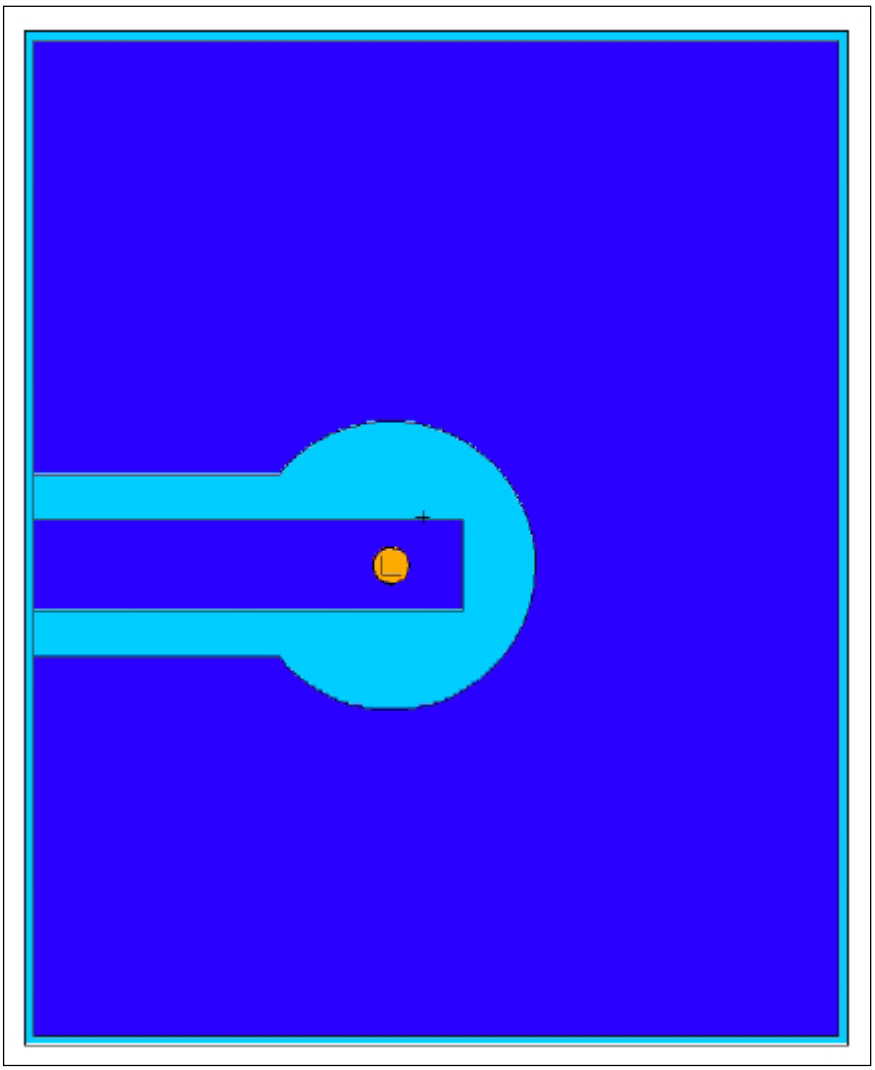

Figure A.1: A top cross-section view at the reactor level of the biological shield of SAFARI-1 as generated by the MCNP geometrical plotter.

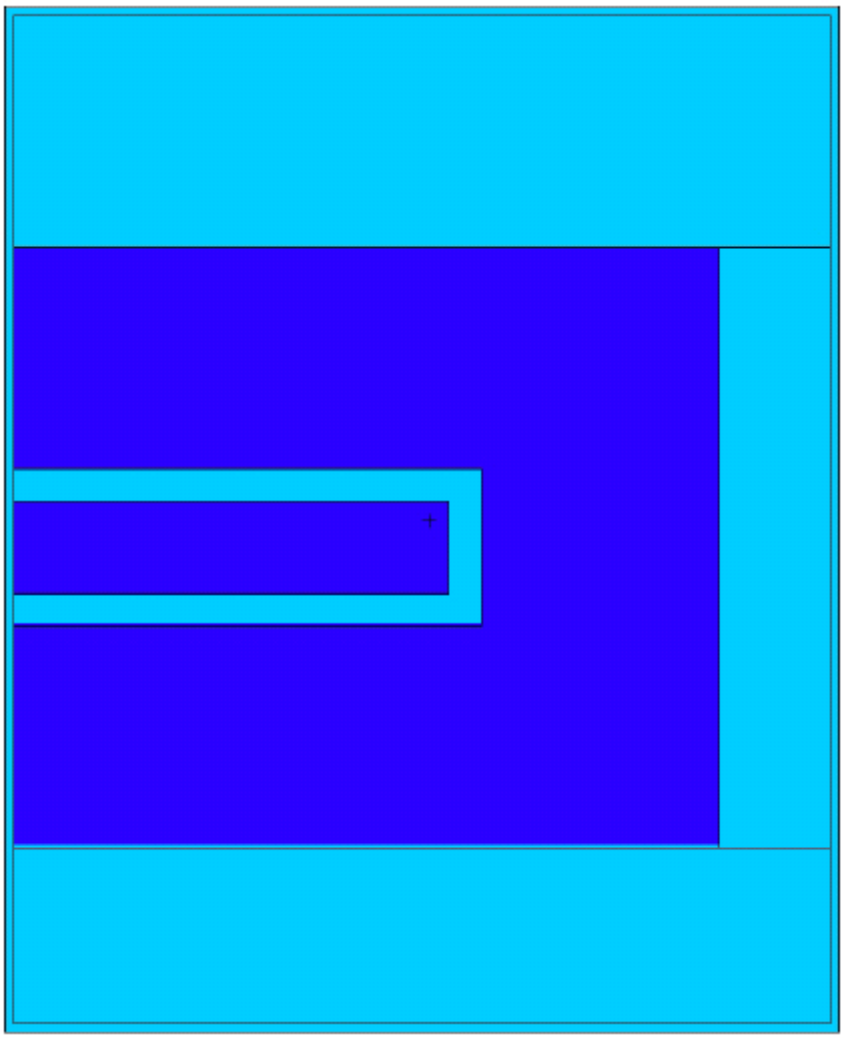

Figure A.2: A top cross-section view at the level of the SAFARI-1 control room floor.

APPENDIX B – MCNP Input File – Photon Dose – Hemisphere Radius = 250 cm

```
c Cell Cards
01   2   -1.21E-3    +5 +6 +7 +8 +9 +10          $ Air in Room (I-131 filled)
                     +11 +12 +13 +14 +15         $ Air in Room (I-131 filled)
                     +16 +17 +51 +52 +53         $ Air in Room (I-131 filled)
                     +54 +55 +56 +57 +58         $ Air in Room (I-131 filled)
                     (-1:-3:-35) (-73:72)        $ Air in Room (I-131 filled)
                                      imp:p=1    $ Air in Room (I-131 filled)
02   3   -2.35       +1 -2 +35        imp:p=1    $  Concrete Wall Bottom
03   3   -2.35       +3 -4            imp:p=1    $  Concrete Wall Top
05   3   -2.35       (-6 +5):(-5 +7)  imp:p=1    $ Lower Reactor CRT Shield
06   3   -2.35       -8 +7            imp:p=1    $ Upper Reactor CRT Shield
07   3   -2.35       -9               imp:p=1    $ IGF, E
08   3   -2.35       -10              imp:p=1    $ IGF, W
09   3   -2.35       -11              imp:p=1    $ IGF, S
10   3   -2.35       -12              imp:p=1    $ CRF, E
11   3   -2.35       -13              imp:p=1    $ CRF, W
12   3   -2.35       -14              imp:p=1    $ CRF, S
13   5   -0.9        -15              imp:p=1    $ Dry Wall, IGF, E
14   5   -0.9        -16              imp:p=1    $ Dry Wall, IGF, W
15   5   -0.9        -17              imp:p=1    $ Dry Wall, IGF, S
16   6   -2.0        -18              imp:p=1    $ Glass Pane, CRF, E
17   6   -2.0        -19              imp:p=1    $ Glass Pane, CRF, W
18   6   -2.0        -20              imp:p=1    $ Glass Pane, CRF, S
19   7   -2.5        -21              imp:p=1    $ Core Content
20   7   -1.0        -22 +21          imp:p=1    $ Core Vessel Content
21   8   -2.7        -23 +22          imp:p=1    $ Core Vessel Wall
30   3   -2.35       -30              imp:p=1    $  Floor
31   7   -1.0        -31 -7 +23       imp:p=1    $ Pool water
51   1   -1.0        -51              imp:p=1    $ Phantom 1
52   1   -1.0        -52              imp:p=1    $ Phantom 2
53   1   -1.0        -53              imp:p=1    $ Phantom 3
54   1   -1.0        -54              imp:p=1    $ Phantom 4
55   1   -1.0        -55              imp:p=1    $ Phantom 5
56   1   -1.0        -56              imp:p=1    $ Phantom 6
57   1   -1.0        -57              imp:p=1    $ Phantom 7
```

```
58  1  -1.0        -58                          imp:p=1   $ Phantom 8
72  2  -1.21E-3   (+73 -72 +55):(-7 +31)   imp:p=1   $ HemiSphere
80  2  -1.21E-3   +2 +4 +30 -80             imp:p=1   $ Outer Air
99  0             +80                          imp:p=0   $ Umwelt (Outer world)

c Surface cards
01  RPP  -1196.7  1497.0  -1572.0  1756.4   -31.5  1207.9      $  Inner Box Bottom
02  RPP  -1226.7  1527.0  -1602.0  1786.4   -31.5  1237.9      $  Outer Box Bottom
03  RPP  -1196.7  1131.3   -989.0   989.0  1237.9  2361.1      $  Inner Box Top
04  RPP  -1226.7  1161.3  -1019.0  1019.0  1237.9  2391.1      $  Outer Box Top
05  RPP  -1196.7   243.7   -304.6   304.6   -31.5   537.4      $ Thick, Lower Reactor Concrete Shield, Outer
06  RCC   0.0  0.0 -31.5     0.0     0.0  +568.9    483.4      $ Round, Thick Lower Reactor Shield, Outer
07  RPP  -1196.7   238.7   -152.4   152.4   -31.5   830.7      $ Inner wall, Reactor Pool
08  RPP  -1196.7   345.4   -259.1   259.1   537.4   830.7      $ Thin, Upper Reactor Concrete Shield, Outer
09  RPP  -1196.7  1497.0    989.0  1756.4   458.7   537.4      $ Instrument Gallery Floor, East
10  RPP  -1196.7  1497.0  -1572.0  -989.0   458.7   537.4      $ Instrument Gallery Floor, West
11  RPP   1131.3  1497.0   -989.0  +989.0   458.7   537.4      $ Instrument Gallery Floor, South
12  RPP  -1196.7  1497.0    989.0  1756.4   790.1   830.7      $ Contol Room Floor, East
13  RPP  -1196.7  1497.0  -1572.0  -989.0   790.1   830.7      $ Contol Room Floor, West
14  RPP   1131.3  1497.0   -989.0  +989.0   790.1   830.7      $ Contol Room Floor, South
15  RPP  -1196.7  1282.2   1139.0  1140.0   537.4   790.1      $ Dry Wall for Instrument Gallery, East
16  RPP  -1196.7  1282.2  -1140.0 -1139.0   537.4   790.1      $ Dry Wall for Instrument Gallery, West
17  RPP   1281.2  1282.2  -1139.0  1139.0   537.4   790.1      $ Dry Wall for Instrument Gallery, South
18  RPP  -1196.7  1132.3    989.0   990.0   830.7  1207.9      $ GlassPane, Control Room Floor, East
19  RPP  -1196.7  1132.3   -990.0  -989.0   830.7  1207.9      $ GlassPane, Control Room Floor, West
20  RPP   1131.3  1132.3   -989.0   989.0   830.7  1207.9      $ GlassPane, Control Room Floor, South
21  RPP    -30.0    30.0    -30.0    30.0     0.0    60.0      $ CoreBox - Outer Dimensions Only
22  RCC   0.0 0.0  -29.5      0.0     0.0   200.0    60.0      $ Core Vessel
23  RCC   0.0 0.0  -31.5      0.0     0.0   205.08   62.5      $ Core Vessel Outer
30  RPP  -1226.7  1527.0  -1602.0  1786.4   -100    -31.5      $  Floor
31  PZ    800.0                                                $ Water upper level
35  RPP  -1196.7  1131.3   -989.0   989.0  1207.9  1237.9      $  Air space to fill gap
c ===========================================================================
51  RCC    0.0   620.0   -31.0       0.0   0.0  170.0    11.5  $ Phantom 1
52  RCC    0.0  1000.0   -31.0       0.0   0.0  170.0    11.5  $ Phantom 2
53  RCC    0.0  1500.0   -31.0       0.0   0.0  170.0    11.5  $ Phantom 3
54  RCC    0.0   360.0   540.0       0.0   0.0  170.0    11.5  $ Phantom 4
```

```
55  RCC    0.0   200.0   832.0       0.0   0.0  170.0    11.5     $ Phantom 5
56  RCC    0.0   630.0   832.0       0.0   0.0  170.0    11.5     $ Phantom 6
57  RCC    0.0  1300.0   832.0       0.0   0.0  170.0    11.5     $ Phantom 7
58  RCC    0.0  1300.0   540.0       0.0   0.0  170.0    11.5     $ Phantom 8
c ===============================================================================
72  S    0  0  830.7  250                                          $  Sphere
73  PZ   830.7                                                     $  Bottom plane Hemisphere
80  RPP  -1326.7  1627.0  -1702.0  1886.4   -200     2491.1       $  Border of outer air space

c Data Cards
c Mode: transport photons
mode p
sdef par=p erg=d4 x=d1 y=d2 z=d3 cel=72
si1   -300 300  $ Sampling of source point along X-coordinate: sample from X.min to X.max
sp1   0   1     $ Probability of sampling of source point along X-coordinate
si2   -300 300  $ Sampling of source point along Y-coordinate: sample from Y.min to Y.max
sp2   0   1     $ Probability of sampling of source point along Y-coordinate
si3   +800  1130.7   $ Sampling of source point along Z-coordinate: sample from Z.min to Z.max
sp3   0   1     $ Probability of sampling of source point along Z-coordinate
si4 L  2.946E-02  2.978E-02  8.018E-02  1.772E-01  2.843E-01  3.258E-01  &
       3.645E-01  5.030E-01  6.370E-01  6.427E-01  7.229E-01
sp4    1.400E-02  2.590E-02  2.620E-02  2.650E-03  6.060E-02  2.510E-03  &
       8.120E-01  3.610E-03  7.270E-02  2.200E-03  1.800E-02
c Material Data
m1   8000    -6.143E-01     &      $ TEM = Tissue-Equivalent Material
     6000    -2.286E-01     &      $ TEM
     1000    -1.000E-01     &      $ TEM
     7000    -2.571E-02     &      $ TEM
    20000    -1.429E-02     &      $ TEM
    15000    -1.114E-02     &      $ TEM
    19000    -2.000E-03     &      $ TEM
    16000    -2.000E-03     &      $ TEM
    11000    -1.429E-03     &      $ TEM
    17000    -1.357E-03     &      $ TEM
    12000    -2.714E-04     &      $ TEM
    26000    -6.000E-05             $ TEM
m2   6000    -1.24E-4       &      $ Air, dry.  Density = 1.205E-03 g/cc
```

```
     7000    -0.755267      &       $ Air, dry.
     8000    -0.231781      &       $ Air, dry.
    18000    -0.012827              $ Air, dry.
m3   1000    -0.013         &       $ Type 04 Ordinary Concrete
     8000    -1.165         &       $ Type 04 Ordinary Concrete
    11000    -0.040         &       $ Type 04 Ordinary Concrete
    12000    -0.010         &       $ Type 04 Ordinary Concrete
    13000    -0.108         &       $ Type 04 Ordinary Concrete
    14000    -0.740         &       $ Type 04 Ordinary Concrete
    16000    -0.003         &       $ Type 04 Ordinary Concrete
    19000    -0.045         &       $ Type 04 Ordinary Concrete
    20000    -0.196         &       $ Type 04 Ordinary Concrete
    26000    -0.030                 $ Type 04 Ordinary Concrete
m4  26000    +1.0                   $ Iron
m5   6000    +1                     $ Wood (DryWall)
     1000    +2                     $ Wood (DryWall)
m6  14000    +1                     $ Glass
     8000    +2                     $ Glass
m7   1000    +2                     $ Water
     8000    +1                     $ Water
m8  13000    -1.0                   $ Aluminium
c
c ===================================
c **** Physics Table *****
phys:p  20   &   $  Emax
         0   &   $  If 0, generation on; (DEFAULT); If 1 generation is off.
         0   &   $  If 1, coherent scattering is turned off.
         0   &   $  If 0,  photonuclear particle production is turned off (DEFAULT).
         1       $  If 1, Doppler energy broadening is turned off (DEFAULT).
phys:e  20   &   $  Emax
         0   &   $  0 always
         0   &   $  0 always
         0   &   $  0 always
         0   &   $  0 always
         1   &   $  1 always
         1   &   $  1 always
         1   &   $  1 always
```

```
         1   &   $  1 always
         0       $  0 always (Bremsstrahlung production treatment; 0 is more accurate; 1 is faster.)
c =============
c
fc16 Photon Absorbed Dose Rate inside the Phantoms
+f16  51  52  53  54  55  56  57  58  $ was 2 spaces
fm16  6.2409E9
c
c ==============
PRINT  10  40  50  60  72  100  110  120  170  200
c ==============
ctme 240


Comments
Sum of photon yields for I-131 = 1.0404 (ICRP-38)
```

APPENDIX C – Radioactive inventory – ORIGEN-S run output

Table C.1: ORIGEN-S run output data set calculating radioactive inventory in the fuel elements of SAFARI-1.

Fuel type	LEU-Silicide fuel (U_3Si_2)
Enrichment	19,75% ^{235}U
Burn-up	80% or 180 MWD
Burn-up rate	20 MW

Name of group related to chemical compounds that form	**Nuclide**	**26 LEU Fuel Elements Inventory (Bq)**
1. Noble gas	^{83m}Kr	1.60×10^{15}
	^{85}Kr	6.97×10^{13}
	^{85m}Kr	3.82×10^{15}
	^{87}Kr	7.31×10^{15}
	^{88}Kr	1.03×10^{16}
	^{131m}Xe	1.20×10^{14}
	^{133}Xe	2.23×10^{16}
	^{133m}Xe	6.92×10^{14}
	^{135}Xe	2.86×10^{15}
	^{135m}Xe	4.24×10^{15}
2. I_2, CsI, HBr	^{82}Br	2.50×10^{13}
	^{83}Br	1.60×10^{15}
	^{84}Br	2.83×10^{15}
	^{128}I	4.55×10^{13}
	^{129}I	1.36×10^{8}
	^{130}I	5.04×10^{12}
	^{131}I	1.04×10^{16}
	^{132}I	1.52×10^{16}
	^{133}I	2.29×10^{16}
	^{134}I	2.52×10^{16}
	^{135}I	2.13×10^{16}
3. CsOH	^{134}Cs	5.07×10^{14}
	^{134m}Cs	3.20×10^{14}
	^{135}Cs	1.09×10^{9}
	^{135m}Cs	2.14×10^{14}
	^{136}Cs	1.95×10^{14}
	^{137}Cs	6.06×10^{14}
	^{138}Cs	2.18×10^{16}
	^{24}Na	1.79×10^{14}
	^{28}Al	2.33×10^{16}
	^{31}Si	1.28×10^{13}
	^{32}Si	6.40×10^{3}
	^{32}P	2.76×10^{9}
	^{33}P	3.17×10^{4}

Name of group related to chemical compounds that form	Nuclide	26 LEU Fuel Elements Inventory (Bq)
4. TeO_2 and Sb_2O_3	^{125m}Te	6.45×10^{12}
	^{127}Te	7.07×10^{14}
	^{127m}Te	9.41×10^{13}
	^{129}Te	2.76×10^{15}
	^{129m}Te	4.42×10^{14}
	^{131}Te	9.07×10^{15}
	^{131m}Te	1.44×10^{15}
	^{132}Te	1.50×10^{16}
	^{133}Te	1.28×10^{16}
	^{133m}Te	9.65×10^{15}
	^{134}Te	2.05×10^{16}
	^{122}Sb	3.72×10^{12}
	^{124}Sb	1.46×10^{12}
	^{125}Sb	3.69×10^{13}
	^{126}Sb	5.46×10^{12}
	^{126m}Sb	3.59×10^{12}
	^{127}Sb	7.12×10^{14}
	^{128}Sb	9.98×10^{13}
	^{128m}Sb	1.45×10^{15}
	^{129}Sb	2.76×10^{15}
	^{130}Sb	9.46×10^{14}
	^{131}Sb	8.71×10^{15}
5. SrO and BaO	^{87m}Sr	1.70×10^{10}
	^{89}Sr	1.68×10^{16}
	^{90}Sr	5.49×10^{14}
	^{91}Sr	1.73×10^{16}
	^{92}Sr	1.77×10^{16}
	^{135m}Ba	3.72×10^{10}
	^{136m}Ba	3.22×10^{13}
	^{137m}Ba	5.75×10^{14}
	^{139}Ba	2.12×10^{16}
	^{140}Ba	2.15×10^{16}
6. RuO_4 and MoO_3	^{99}Mo	2.06×10^{16}
	^{103}Ru	1.47×10^{16}
	^{105}Ru	6.86×10^{15}
	^{106}Ru	1.75×10^{15}
	^{99}Tc	7.90×10^{10}
	^{99m}Tc	1.81×10^{16}
7. La_2O_3 and CeO_2	^{140}La	2.25×10^{16}
	^{141}La	1.94×10^{16}
	^{142}La	1.92×10^{16}
	^{151}Sm	6.84×10^{12}
	^{153}Sm	3.20×10^{15}
	^{155}Sm	2.24×10^{14}
	^{156}Sm	1.27×10^{14}

Name of group related to chemical compounds that form	Nuclide	26 LEU Fuel Elements Inventory (Bq)
7. La_2O_3 and CeO_2 (Cont.)	^{158}Sm	3.64×10^{13}
	^{147}Pm	1.21×10^{15}
	^{148}Pm	2.68×10^{15}
	^{148m}Pm	3.35×10^{14}
	^{149}Pm	7.38×10^{15}
	^{150}Pm	1.61×10^{14}
	^{151}Pm	1.72×10^{15}
	^{142}Pr	4.52×10^{14}
	^{143}Pr	1.91×10^{16}
	^{144}Pr	1.34×10^{16}
	^{144m}Pr	1.59×10^{14}
	^{145}Pr	1.26×10^{16}
	^{146}Pr	9.72×10^{15}
	^{90}Y	5.82×10^{14}
	^{90m}Y	1.39×10^{11}
	^{91}Y	2.11×10^{16}
	^{91m}Y	1.00×10^{16}
	^{92}Y	1.78×10^{16}
	^{93}Y	1.95×10^{16}
	^{147}Nd	7.96×10^{15}
	^{149}Nd	3.87×10^{15}
	^{141}Ce	2.15×10^{16}
	^{142}Ce	1.64×10^{5}
	^{143}Ce	1.90×10^{16}
	^{144}Ce	1.32×10^{16}
	^{95}Zr	2.46×10^{16}
	^{97}Zr	1.96×10^{16}
	^{95}Nb	2.63×10^{16}
	^{95m}Nb	1.75×10^{14}
	^{96}Nb	3.59×10^{13}
	^{97}Nb	1.98×10^{16}
	^{97m}Nb	1.86×10^{16}
	^{98m}Nb	1.49×10^{14}
8. NpO_2 and PuO_2	^{238}Np	1.37×10^{15}
	^{239}Np	1.07×10^{17}
	^{240}Np	2.26×10^{14}
	^{238}Pu	2.83×10^{12}
	^{239}Pu	1.05×10^{12}
	^{240}Pu	9.33×10^{11}
	^{241}Pu	2.50×10^{14}
	^{242}Pu	1.98×10^{9}
	^{243}Pu	3.43×10^{14}
	^{237}U	6.71×10^{15}
	^{239}U	1.07×10^{17}