

SAFETY ASSESSMENT FOR A RADIOACTIVE EFFLUENT TREATMENT FACILITY

Elna Fourie

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ABSTRACT

An in depth evaluation of safety is an essential component of the successful operation of nuclear facilities. The Safety Assessment Methodology is the tool applied to perform the safety evaluations of a facility in order to demonstrate satisfactory safety of such a facility to the operator, the National Nuclear Regulator (NNR) in South Africa, and to the public.

A Safety Assessment is performed and documented to demonstrate the compliance with safety objectives, principles and criteria as stipulated by national and international standards. In South Africa the Department of Mineral and Energy publishes the National Nuclear Regulatory Requirements for demonstrating safety of a Nuclear Facility. As such, this Safety Assessment provides a formal structured procedure for defining the logic of systems and assessing the consequences if failure in the control logic occurs.

This Safety Assessment demonstrates that the radioactive effluent generated at a typical nuclear facility the South African Nuclear Energy Corporation (Necsa) is managed according to national and internationally acceptable proven technology and standards. Those standards are documented in the National Nuclear Regulatory Act. Licensing documents have been issued by the Regulator to guide the holder of a nuclear installation licence to compile a Safety Assessment. International standards are provided by the International Atomic Energy Association (IAEA) through a number of technical documents and the Safety Standards Series.

This Safety Assessment aims to show that the Radioactive Effluent Treatment Facility at Necsa can be operated without endangering the lives of the operators and the public and without having adverse effects on the environment. The hazards of operating a Radioactive Effluent Treatment Facility include radiological, chemical and conventional hazards. However, this Safety Assessment only focuses on the radiological (radiation and contamination) hazards and, in particular on the radiological hazards associated with the

accidental exposure of an operator to Medium Activity Effluent.

This report demonstrates the safety condition of the Radioactive Effluent Treatment Facility at Necsa by providing:

1. a systematic evaluation of consequences in the case of accidental exposure and
2. a documented demonstration of the controls in place to ensure that the workers, members of the public and environment are protected sufficiently, with the main focus being on worker exposure.

The justification of the discharge of treated radioactive effluent into the environment is a globally accepted practice. This Safety Assessment briefly discusses environmental discharge practices but the main focus is on the analysis of the protection of the workers of the Radioactive Effluent Treatment Facility. The radioactive effluent is only discharged into the nearby Crocodile River if it is in compliance with discharge criteria as set by the National Nuclear Regulator and the Department of Water Affairs and Forestry.

DECLARATION

I, Elna Fourie, declare that this dissertation, submitted to the University of the Witwatersrand, is my own, unaided work. It has not been presented before for any degree or examination to any other University.

A handwritten signature in black ink, appearing to read 'Elna Fourie', written in a cursive style.

Elna Fourie

Signed 26 August 2009

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GLOSSARY

For the purpose of this document, the following terms shall have the meanings given in the table below:

Advanced Gas Cooled Reactor	A term used for the second generation of British Power Reactors, now operated by British Energy. The fuel used in the reactor is slightly enriched uranium oxide clad in stainless steel. The coolant is carbon dioxide and the moderator is graphite
Accident / Incident or Event (Nuclear accident)	Any unintended event, including operating errors, equipment failures or other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety. Any accident involving facilities or activities from which a release of radioactive material occurs or is likely to occur and which has resulted or may result in an international trans-boundary release that could be of radiological safety significance for another State.
ALARA	An approach to control or manage radiation exposures (both individual and collective to the workforce and the public) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit; it is a practice that has as its objective the attainment of dose levels as far below applicable limits as possible (see Appendix A).
Analysis / Assessment	A method for determining and evaluating the detailed performance of a process.

Anticipated operational occurrence	An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety nor lead to accident conditions.
Critical group	Group of people representative of those individuals in the population expected to receive the highest annual dose, which is a small enough group to be relatively homogeneous with respect to age, diet and those aspects of behaviour that affect the annual dose received. The mean dose of this group is compared with dose limits.
Deterministic effects	The radiation effect for which generally a threshold level of dose exists above which the sensitivity of the effect is greater for a higher dose.
Dose	A measure of the radiation received or absorbed by a target. The quantities termed absorbed dose, organ dose, equivalent does, effective dose, committed equivalent dose or committed effective dose are used, depending on the context.
Epidemiological studies	The studies of radiological factors affecting the health and illness of populations, and serves as the foundation and logic of interventions made in the interest of public health and preventive medicine.
Exposure pathways	The routes or more strictly environmental processes by which people are exposed to radiation.
External events	Events originating outside the nuclear installation with the potential to cause adverse conditions or even damage to safety important structures, systems or components.
Facility	Buildings, containers or equipment in which a process is conducted.

Generator	Any facility that produces waste or where the waste originates from.
HAZOP	A HAZOP (Hazard and Operability study) is a method for systematically identifying hazards and operational problems for intended new facilities or the modification to existing facilities or equipment.
Internal events	Events originating inside the nuclear installation with the potential to cause adverse conditions or even damage to safety important structures, systems or components.
Magnox Reactor	A term for the first generation of British power reactors (at Berkeley, Bradwell, Calder Hall, Chapelcross, Dungeness A, Hinkley Point A, Hunterston A, Oldbury, Sizewell A, Trawsfynydd and Wylfa) from the use of "Magnox" as the Cladding Material.
Operating Technical Specifications	A set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility.
Probabilistic Risk Assessment	Methodology provided for a formal structured procedure for defining the functional logic of the identification of initiating events and risk determination.
Process Safety	A discipline that focuses on the prevention of explosions, accidental releases or the unsafe handling of radioactive substances.
Process	The activity involving any radioactive materials, including their use, manufacturing, transportation, and storage, or the combination of such activities.
Radioactive effluent treatment facility	A facility designed to manage the radioactive liquid waste in order to discharge the treated effluent to the environment within approved release limits.
Risk	$Risk = Probability \times Consequence$

Safety	The expectation that a process will not lead to a state in which any human life or the environment is adversely affected.
Safety Assessment	An assessment of all aspects of a practice that area relevant in protection and safety; for an authorised facility, this includes siting, design and operation of the facility. A Safety Assessment is a collection of arguments and evidence in support of the safety of a practice.
Safety Assessment methodology	The methodology applied during the compilation of a Safety Assessment is the assumption of possible deviations that could occur during the operation of a facility and the associated consequences when such deviations occur. Various techniques and safety evaluation processes are applied to identify such possible deviations or initiating events and allocate a risk to such deviations based on consequences and likelihood of occurring.
Single-failure events	Events caused by the failure of a single control.
Source Term	The amount, and isotopic composition of material released (or postulated to be released) from a <i>nuclear installation</i> or <i>action</i> as well as the release characteristics and associated data required for the impact analysis.
Storage	The holding of waste in a facility that provides for its containment, with the intention of retrieval.
Stochastic effects	The health effects, the probability of occurrence of which is greater for a higher radiation dose and the severity of which (if it occurs) is independent of dose and generally occurs without a threshold.
Waste	Material in gaseous, liquid or solid form and in concentrations or forms that do not permit economic recovery and that are designated for disposal.

ABBREVIATIONS

ALARA	As Low As Reasonable Achievable
BNFL	British Nuclear Fuels
DAC	Derived Air Concentration
DOH	Department of Health
DWAF	Department of Water Affairs and Forestry
FMEA	Failure Mode Effective Analysis
HAZOP	Hazard and operability study
HL	Hazard Level
HVL	Half Value Layer
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INES	International Nuclear and Radiological Event Scale
IROFS	Items Relied on for Safety
LA	Low Activity
MA	Medium Activity
NECSA	South Africa Nuclear Energy Corporation
NNR	National Nuclear Regulator
NORM	Natural Occurring Radioactive Material
OTS	Operating Technical Specifications
PRA	Probabilistic Risk Assessment
QP	Quality Plan
QRA	Quantitative Risk Assessment
RPO	Radiation Protection Officer
SFL	Springfields Fuels Limited
SHE	Safety, Health and Environment
TLD	Thermoluminescent dosimeter
TVL	Tenth Value Layer

CHAPTER 1

INTRODUCTION AND OVERVIEW OF THE NECSA RADIOACTIVE EFFLUENT TREATMENT FACILITY AND COMPARISON WITH SIMILAR NATIONAL AND INTERNATIONAL FACILITIES

Nuclear power is a reliable source of clean energy and is largely considered as the answer to the energy threat experienced globally. There is still significant opposition to nuclear power based on safety grounds, but it is generally considered to be a much less significant contributor to global warming than many other existing means of electricity generation. From a public perception the use of nuclear power is unsafe due to a number of events and general ignorance.

The holders of Nuclear Site Licences face the challenge of creating a positive public perception associated with nuclear energy. They need to demonstrate safety through the application of various safety methodologies [1-6]. The holders of Nuclear Site Licences are obliged to have adequate capability within their own organisations and to take steps to retain this capability that enables them to understand the nuclear safety requirements of all its activities and to demonstrate adherence to these requirements on national and international levels.

The holders of Nuclear Site Licences (including the nuclear design engineers and the operators of the nuclear facilities) in conjunction with the Nuclear Regulators of each country are faced with the question as to whether a facility is acceptable from a nuclear safety perspective to allow it to be designed, constructed, commissioned, operated and decommissioned. The demonstration of adequate safety is a very challenging problem to the nuclear industry as a whole and to South Africa Nuclear Energy Corporation (Necsa) in particular. It must be demonstrated that the safety of nuclear facilities are in compliance with national and internationally acceptable standards in such a way that the South African

operator, the National Nuclear Regulator (NNR) and the public will be satisfied.

The negative perception caused by the well known Chernobyl nuclear accident and a few other smaller nuclear accidents have resulted in the global stagnation of the nuclear industry. The so called “nuclear winter” is over and the whole world is entering an exciting nuclear renaissance. Nuclear accidents must be prevented at all cost to ensure development and to prevent any interruptions in nuclear research programmes.

Public perception and acceptance of the application of nuclear technology

Nuclear technology involving radioactive material also contributes to the well-being of people through medical applications. Two different applications of nuclear technology are applicable in the medical field i.e. Radiation Therapy and Radiology.

Radiation Therapy (also known as Radiotherapy or Radiation Oncology) is the medical use of ionizing radiation as part of cancer treatment to control malignant cells. Radiotherapy may be used for palliative treatment (where cure is not possible and the aim is for local disease control or symptomatic relief acting as a painkiller) or as therapeutic treatment (where the therapy has survival benefit and it can be curative).

Radiology is the medical specialty of directing medical imaging technologies to diagnose diseases. Originally it was the aspect of medical science dealing with the medical use of electromagnetic energy emitted by X-ray machines or other such radiation producing devices for the purpose of obtaining visual information as part of medical imaging. Modern day radiological imaging is no longer limited to the use of X-rays, and now includes technology-intensive imaging with high frequency sound waves, magnetic fields and radioactivity

The use of nuclear technology through medical applications is, however, more acceptable to the public due to healing effects and the possible prevention of death caused by cancer due to its early diagnoses and treatment. A large fraction of the population receives nuclear medicine examinations every year. The part of the U.S.A. population that receives nuclear medicine examinations has been characterised by age and sex. Males received 42% of examinations while females received 58%. More than one-third of the examinations were done on persons older than 64 year of age and more than two-thirds on patients older than 45 years of age. The effective dose equivalent from nuclear medicine procedures can be compared with the 2 mSv from natural background [7].

Industrial activities or practices, in radiological as well as the chemical industries, are also associated with risk but the acceptance by the public and regulators requires a higher level of justification than for medical applications. Radiation Protection is based on the concept of trade-offs, of balancing risks against benefits. A patient undergoing radiation treatment may be subjected to relatively high doses. However, this increased risk is tolerated, because the benefits provided to the patient by the treatment are judged to outweigh any harm that might be caused by the increased risk.

Justification of Practices

Practices are more radiological specific and are defined as deliberate human activities which, as a by-product, result in increased exposure of individuals or populations. In principle, these can be designed and operated to meet requirements for radiological protection that are specified in advance. The International Commission on Radiological Protection (ICRP) defines the justification of a Practice stating that “*no Practice involving exposures to radiation should be adopted unless it produces sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes*” [8].

The system of radiological protection recommended by ICRP for proposed and continuing Practices has the following principles:

- The process of justification is required not only when a new Practice is being introduced, but also when an existing practice is being reviewed in the light of new information about its efficiency or consequences.
- If such a review indicates that a Practice no longer produces sufficient benefit to offset the detriment, withdrawal of the Practice should be considered.

The performance of a Safety Assessment is one way to determine if a Practice should be terminated or if it can continue operating.

Industrial activities envisage continuous improvements through the development of technology specifically focused on improving safety measures. As explained previously it is essential to demonstrate that the radiological risk associated with a facility is managed in such a way that it will not adversely affect the health of the workers and the public or the environment. As part of the ICRP principles it is also required to demonstrate on a regular basis that the practice is still justifiable.

Description of Necsa

Necsa, also well known as Pelindaba, [9] is located on the farm Weldaba and covers an area of approximately 24 km². As indicated in Fig. 1.1, it is located in the North West Province of South Africa. The topography of the Necsa site is hilly to mountainous with great differences in the elevation and steep surface slope gradients to the north and south.

Necsa is a Multi-facility Nuclear Site that operates the processes involved in the front end of the Nuclear Fuel Cycle and, therefore, excludes the reprocessing of spent fuel [9].

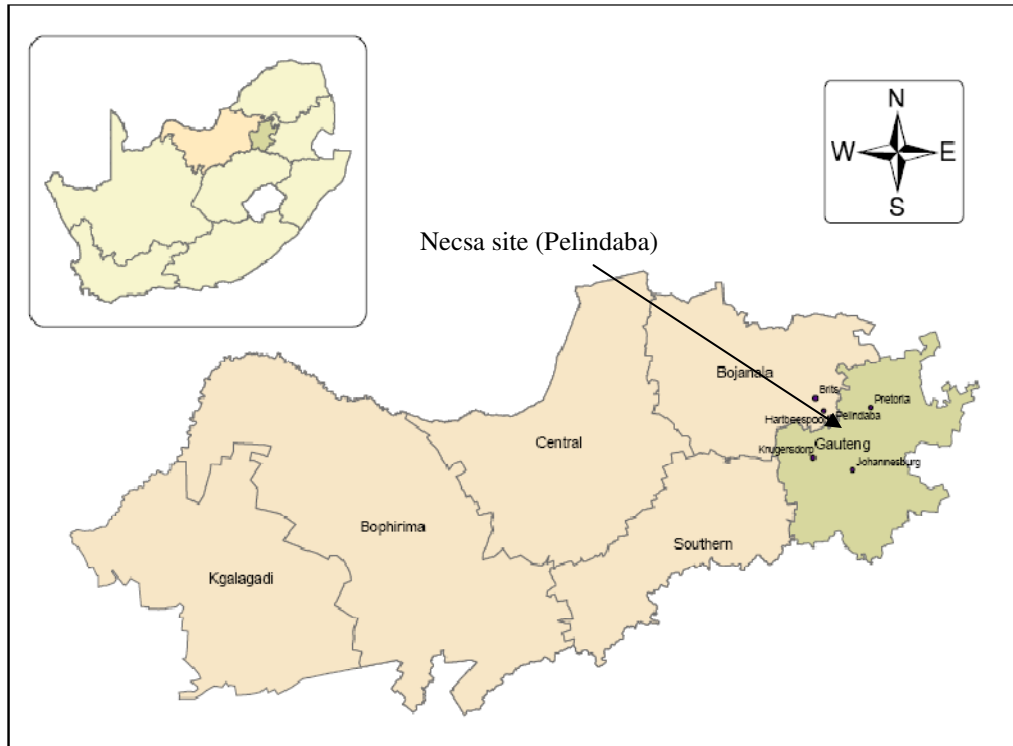


Figure 1.1: Map of North West Province and Gauteng showing Necsasite relative to Pretoria and Johannesburg.

The South African nuclear programme started in 1948 and focussed on research and development in military applications and in the Nuclear Fuel Cycle. One of the highlights in the history of Necsasite was the successful separation of uranium isotopes and the start of the Uranium Enrichment Programme [9, 39].

Figure 1.2 is a view over the eastern part of the Necsasite. The Radioactive Effluent Treatment Facility is situated in the eastern part of the Necsasite. In the early 1970s a uranium conversion plant and a uranium enrichment plant were constructed on the Necsasite. The enriched uranium was used as fuel for the Research Reactor SAFARI-1, the nuclear power plant at Koeberg and for military purposes. Various other small research projects and laboratories were initiated during that time also focussing on the Nuclear Fuel Cycle. Many of the facilities generated radioactive waste (solid, liquid and gaseous). Most of the radioactive liquid (effluent) waste was and is treated on the Necsasite in the Radioactive Effluent Treatment Facility. It should be noted that the solid and gaseous

radioactive wastes are managed through other processes and are not included in this Safety Assessment. Various chemical facilities are also operating on the Necsa site but non-radioactive chemical effluent is not treated at the Radioactive Effluent Treatment Facility. Non-radioactive chemical effluent is treated by independent, certified, waste contracting companies.



Figure 1.2: View of the eastern part of the Necsa site where the Radioactive Effluent Treatment facility is located.

The Necsa site incorporates hilly terrain and the position of the building that houses the Radioactive Effluent Treatment Facility is situated slightly lower than most of the neighbouring facilities. The hilly terrain is clearly visible in Fig. 1.3 where the light brown indicates the lower contours and the dark brown colours indicate the hilly areas. Figure 1.3 shows that the ground slopes towards the Radioactive Effluent Treatment Facility, thus allowing most of the liquid effluent to be received mainly from gravity-fed drains. Some liquid waste is, however,

pumped and some is transported by tanker to the Radioactive Effluent Treatment Facility.

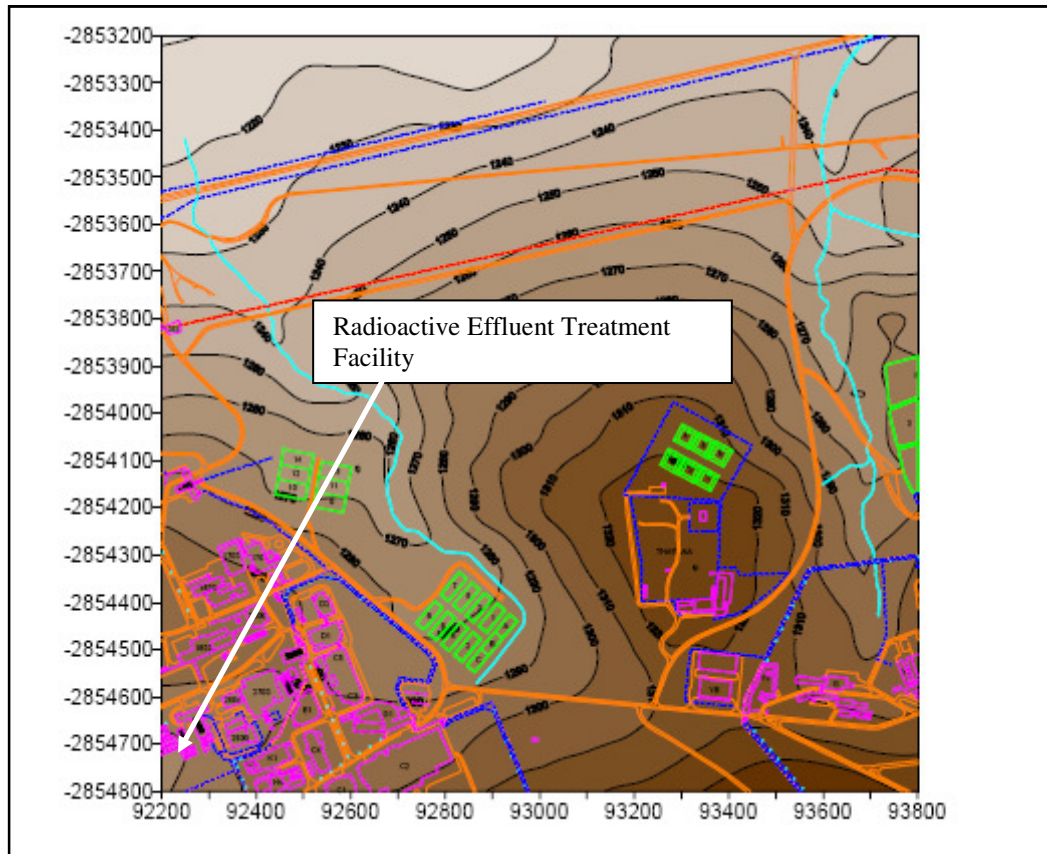


Figure 1.3: Topographical contour map of part of Necsia showing that the Radioactive Effluent Treatment Facility is situated lower than most of the other facilities.

The discharge of treated effluent to the nearby Crocodile River is also by gravity feed. The river is approximately 1 km from the facility and the nearest public road is approximately 1.3 km from the Radioactive Effluent Treatment Facility. The Necsia site was selected and justified for suitability due to its stable geological characteristics and central location near to but not too close to fast developing industries in the Gauteng area.

1.5.1 Description of the Radioactive Effluent Treatment Facility

The Radioactive Effluent Treatment Facility receives and treats two types of effluent namely:

- i. Medium Activity Effluent and
- ii. Low Activity Effluent.

The Medium Activity Effluent is defined as the radioactive effluent for which the α -activity exceeds $100 \text{ Bq}\ell^{-1}$ and the β^- -activity exceeds $4000 \text{ Bq}\ell^{-1}$ when received from the different facilities, while the Low Activity Effluent is that for which the α -activity is between 10 and $100 \text{ Bq}\ell^{-1}$ and the β^- -activity is between 40 and $4000 \text{ Bq}\ell^{-1}$. These limits are referred to as the waste acceptance criteria of the Radioactive Effluent Treatment Facility. If the radioactive effluent at the facility where it is generated does not comply with this waste acceptance criteria the effluent will not be accepted by the Radioactive Effluent Treatment Facility. Transfer is managed by the workers of the Radioactive Effluent Treatment Facility. Sometimes the generators of the radioactive effluent must store the radioactive effluent for some time to allow sufficient decay (cooling down period) to enable them to release the radioactive effluent to the Radioactive Effluent Treatment Facility. The waste acceptance criteria were determined by evaluating the decontamination effectiveness of the Radioactive Effluent Treatment Facility, the shielding design and the Regulator's agreed effluent discharged limits.

i. Medium Activity Effluent treatment process

Medium Activity Effluent is received from the various generators. The effluent is fed through pipelines from reception tanks at the facilities to the Medium Activity Effluent reception tanks. The Medium Activity Effluent facility consists of a reception system, a storage system and an evaporator system with sub systems as indicated in Fig. 1.4. Each of these systems is operated according to a specific control logic operating valves, instruments and pumps. These systems are also supported by safety systems i.e ventilation, shielding, containment systems and analytical systems.

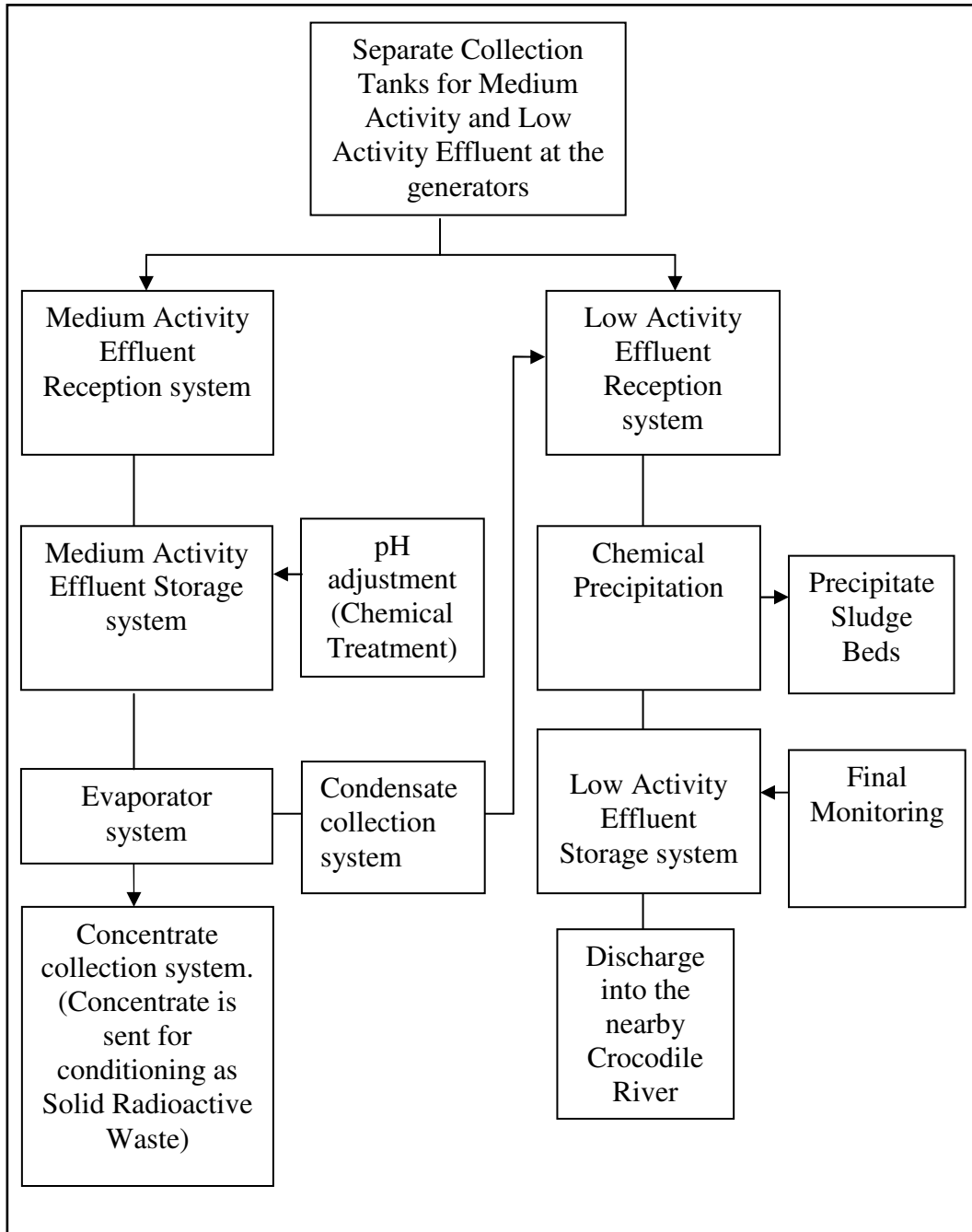


Figure 1.4: Block diagram showing basic treatment steps and integration of the Medium Activity and Low Activity Effluent treatment facilities.

The pH of the received Medium Activity Effluent is measured and adjusted by the addition of acidic or alkaline reagents into the storage system. The pH is adjusted in order to ensure effective decontamination in the evaporator. The function of the evaporator is to decontaminate the Medium Activity Effluent. The evaporation process results in the activity being concentrated at the bottom of the evaporator and cleaner condensate is captured at the top of the evaporator. Condensate from the evaporator is then pumped to the Medium Activity Effluent condensate monitoring tanks and the concentrate is pumped to the concentrate collection tank after each evaporation cycle. The condensate from the evaporator is treated as Low Activity Effluent and the concentrate from the evaporator is treated and conditioned as solid radioactive waste which is sent to another facility and managed as solid radioactive waste.

The radiation protection programme for the occupational exposure of workers under normal operation of the Medium Activity Effluent facility will be described to confirm compliance with the international standards as stipulated in Refs [4, 5, 6]. Worker exposure due to an event at the Medium Activity Effluent process is analysed in detail in this Safety Assessment.

ii. Low Activity Effluent treatment process

Low Activity Effluent is also received through pipelines from the different facilities on the Necsa site. If the Low Activity Effluent from the generator does not conform to the Waste Acceptance Criteria it will be treated as Medium Activity Effluent. The Low Activity Effluent is treated through a chemical precipitation process. The precipitate generated in the Low Activity Effluent facility is flushed into radioactive sludge drying beds. The precipitate is treated as solid radioactive waste. The Low Activity Effluent is fed to holding tanks where it is sampled and discharged into the Crocodile River if it conforms to discharge criteria as authorised by the Regulator and the Department of Water Affairs and Forestry (DWAF) according to internationally accepted annual discharge limits based on ICRP recommendations [8].

Decontamination of the Low Activity Effluent is done at the Radioactive Effluent Treatment Facility:

- to ensure that there is a single point of controlled and monitored discharge of radioactive effluent into the environment and
- as an ALARA principle (see Appendix A) application because most of the Low Activity Effluent received at the Radioactive Effluent Treatment Facility is already in compliance with the authorised discharge criteria.

Although specific limits are authorised by the NNR, Necsa is obliged to investigate the ALARA principles [5] and implement methods and measures to ensure the continual reduction in discharge of radioactive material into the environment.

In general, the aim of ALARA is to minimize the risk of radioactive exposure or amount of dose while keeping in mind that some exposure may be acceptable in order to continue with the process actions. Radiology and other practices that involve the use of radiation bring benefits to population, so reducing radiation exposure can reduce the effectiveness of a medical practice. The economic cost, for example, of adding a barrier against radiation must also be considered when applying the ALARA principle. However, the obligation to apply ALARA at Necsa and other nuclear facilities is higher, especially when considering dose to the public. The compromising of ALARA in the medical industry as well as the application in the nuclear industry with regard to the workers can be justified because exposure is agreed on by the receiver where as exposure to the public is not agreed on.

This compromise is well illustrated in radiology. The application of radiation can aid the patient by providing doctors and other health-care professionals with a medical diagnosis, but the exposure should be reasonably low enough to keep the statistical probability of cancers or stochastic effects below an acceptable level, and to eliminate deterministic effects.

The obligation at Necsa is dictated by the ALARA policy that is based on the principle that any amount of radiation exposure, no matter how small, can increase the chance of negative biological effects such as cancer, though perhaps by a negligible amount. It is also based on the principle that the probability of the occurrence of negative effects of radiation exposure increases with cumulative lifetime dose.

Description of a Similar National Waste Management Process

Currently, there are various sites or facilities in South Africa that are under specific authorisation of the NNR and Department of Health (e.g. iThemba LABS, hospitals, mines etc.) that generate radioactive effluent. In this report the Koeberg Nuclear Power Station will be used as comparison for Necsa practices [10].



Figure 1.5: Koeberg Nuclear Power Station on the Cape West Coast.

Koeberg Nuclear Power Station is situated on the Cape West Coast, approximately 30 km north from Cape Town and is operated under NNR licence. The Koeberg Power Station as illustrated in Fig. 1.5, was visited in January 2008 by the author to compare the Necsa Safety Assessment methodology with the Koeberg Safety Assessment methodology.

Recently, Koeberg undertook a site-specific study applying the principles, methodologies and techniques developed internationally (UK specifically) in the field of Public Dose Assessments [11, 12]. Koeberg identified a Critical Group that would be affected by the radioactive sources and exposure pathways from the Koeberg Nuclear Power Station. The identification of a Critical Groups was done in compliance with international and national Critical Group requirements. Control of radioactive waste disposal to the environment, based on the recommendations of the ICRP, necessitates an identification of the Critical Group of members of the public exposed from a given practice. Criteria for identification of Critical Groups were based mainly on ICRP recommendations but newer techniques were also applied to recent surveys of fish and shellfish consumption in the coastal area of the North-East Irish Sea during the Koeberg site-specific study [13].

Koeberg also did a comprehensive Safety Assessment [14] of all the activities at Koeberg by applying a similar methodology as described in this report. As such, this Safety Assessment is reviewed and revised in accordance with a schedule agreed on between Koeberg and the NNR. It should be noted that Necsa has the SAFARI-1 Research Reactor with various other research facilities but the overall risk factor associated with a research facility is less than the risk factor allocated to a Nuclear Power Station [1].

Koeberg has a well established worker monitoring programme that is in line with the ICRP recommendations as stated in Ref [15, 16]. In addition, Koeberg has a Whole Body Counter (WBC) on site and their occupationally exposed workers are subjected to a Medical Surveillance Programme similar to the Necsa programme.

There are also excellent worker training programmes in place and a simulation centre where operators are continually trained and retrained under normal and abnormal operating conditions. These quality assurance procedures are in place to ensure that the Safety Assessment does not become obsolete. More detail of what is required in such quality assurance procedures is described in this report.

In comparison, Necsa is a multi-facility radiological site with diverse processes that generates various radioactive waste streams whereas Koeberg has well defined waste streams with limited deviations generated by the single Practice. Deviations in the radioactive effluent waste streams at Koeberg are only foreseen during abnormal situations.

Koeberg discharges radioactive effluent through two main streams:

- i. batch discharges and
- ii. continuous discharges.

i. Batch discharges

Each batch is analysed prior to discharge for all possible nuclides as identified by Koeberg and the NNR for normal as well as abnormal operation. If all the radionuclides are within the required limits the batch is isolated from any additional inflow and released to the Atlantic Ocean.

Operating technical specifications are applicable to allow release (e.g. minimum operation of two circulation pumps to ensure sufficient dilution). Koeberg's release limits were determined in a comprehensive study done on the marine life and the testing for a wide range of radionuclides. The limits set for release are still much lower than the detection limits of any of these radionuclides in the marine life. The measured values of the actual released effluent [14] are again much lower than the permitted values.

ii. Continuous discharges

The continuous discharge stream is typical effluent generated in non-process areas or at secondary loops that are not directly into contact with the process side. It is

very unlikely that this stream will have high levels of contamination. The continuous discharge stream is, however, analysed with inline monitors that give readouts every 8 minutes. Should an inline monitor detects values out of range it will automatically redirect the discharge stream to holding tanks.

Description of a Similar International Waste Management Process

The Springfields Nuclear Fuel Cycle facility in the UK was visited in March 2008 by the author in order to become familiar with their practices and more specific their Effluent Waste Management Programme. Springfields site is located between Preston and Blackpool, approximately 8 km west of Preston, in Lancashire and has an area of approximately 1 km² (83 hectares). Springfields manufactures nuclear fuel products for the UK's nuclear power stations and for other international customers. Fuel manufacture is scheduled to continue until 2023. In addition to fuel manufacturing, Springfields is also undertaking decommissioning activities and has decommissioned various nuclear facilities and to date 87 buildings have been fully demolished [17].

The operations at Springfields UK are similar to those of Necsa although Springfields operations are currently not as diverse due to limited research and development work being performed there. The site dates from the Second World War and has been owned respectively by the United Kingdom Atomic Energy Association (UKAEA), British Nuclear Fuel (BNFL) and the Nuclear Decommissioning Authority (NDA). The plant is currently operated by Springfields Fuel Limited and managed by Westinghouse Electrical UK limited on behalf of the NDA.

Due to the war effort, the Springfields site became involved in the early stages of the United Kingdom military nuclear programmes. A civil nuclear programme was run in parallel with the military nuclear programme and the first fuel elements for both the Windscale Piles and the Calder Hall Magnox reactors were produced.

In the early 2000s the Springfields site was the sole manufacturer of nuclear fuel for the UK's Advanced Gas-cooled Reactor and Magnox reactors. Springfields also supplied fuel for the Light Water Reactor at Sizewell B as well as for export. Following the closure of the Magnox reactors the Springfield site embarked on an "oxide only" contract and started with the decommissioning of the rest of the facility.

Unlike Necsa, the Springfields operations do not include the operation of a Research Reactor. The Springfields operations briefly comprise of the following where the process steps shown in *italics* generate radioactive effluent:

- *uranium extraction from the uranium ore (U_3O_8) by adding nitric acid forming a uranyl nitrate $UO_2(NO_3)_2$ liquid solution,*
- *uranium extraction from uranyl nitrate solution by addition of organic solvent and heat to obtain uranium trioxide (UO_3),*
- reduction of uranium trioxide with hydrogen gas to form uranium dioxide (UO_2) by calcinations in a hydrogen environment,
- *formation of uranium tetrafluoride (UF_4) by reaction with hydrofluoric acid and*
- formation of natural uranium hexafluoride (UF_6) by the reaction with fluorine gas.

The natural uranium hexafluoride is sent off site to be enriched by Urenco at Capenhurst in the UK. The enriched uranium is then returned to Springfields for further processing, including the following:

- *the 5% enriched uranium hexafluoride (UF_6) is hydrolysed and oxidised with steam and hydrogen at high temperatures to form uranium dioxide (UO_2) powder and*
- the uranium dioxide (UO_2) is pelletized and sintered and the pellets are loaded into stainless steel canisters.

The main difference between Necsa's radioactive effluent treatment process and the Springfields process is that at Springfields the radioactive effluent treatment is treated up to release criteria at the source (at the process that generates the waste) whereas Necsa treats all the radioactive effluent at a central facility, the Radioactive Effluent Treatment Facility, and not at the source.

The Springfields Fuels Limited discharge of radioactive effluent into the sea is managed at a single discharge point but there are no treatment steps at this point. Radioactive effluent can only be redirected to holding tanks and then returned to the process where it was generated if it is not in compliance. It cannot be rectified to enable discharge at this point.

For England and Wales, the Authorisations are issued and regulated by the Environment Agency and are reviewed at regular intervals. The Radioactive Substances Act 1993 (RSA93) requires operators to have authorisation to dispose of radioactive waste from their sites.

Springfields Fuels Limited was granted a certificate of Authorisation for the disposal of radioactive waste from their site, after the identification and implementation of a number of Improvement and Additional Information Requirements (IAIR's). Springfields Fuels Limited had to submit a full report of a comprehensive review of the means used to assess the activity of radionuclides in disposals and determine compliance with the authorisation including consideration of national and international developments in best practice.

Springfields Fuels Limited has an Environment, Health, Safety and Quality policy briefly stating that the health and safety of their employees, the local community and the environment is Springfields Fuels Limited's is very important. They have implemented stringent conditions to ensure adherence to the Environment, Health, Safety and Quality policy and regulatory requirements as reflected in their site specific authorisation. In addition Springfields has been striving in recent years to achieve and maintain 'world-class' environmental performance. This has been

recognised by the site being commended for its excellence in environmental management in RoSPA's Dilmun International Environmental Award. The site's discharge Authorisation under the Radioactive Substances Act 1993 was reissued in 2005, coming into force on 2005/04/01 with the transfer to the NDA of site assets and the creation of a new company, Springfields Fuels Ltd (SFL). This Authorisation places a duty on SFL to minimise all discharges and emissions using Best Practicable Means (BPM) [18]. Springfields Fuels Limited produces an Annual Report to provide full details of their Environment, Health, Safety and Quality activities. The latest (2007/08) Annual Report can be viewed for more information [17, 18].

Intention of a Safety Assessment

The intention of any Safety Assessment is to deliver a documented demonstration to the National Nuclear Regulator, Necsa and the South African Public that the safety of the workers and the public is important and that exposure is managed. The Safety Assessment aims to provide a systematic evaluation of consequences of normal and abnormal operations. It, therefore, also provides a management tool to ensure that the safety goals are met.

This Safety Assessment covers the following aspects in order to demonstrate the safe operation of the Radioactive Effluent Treatment Facility:

- Chapter 1 aims to introduce the reader to the Necsa Radioactive Effluent Treatment Facility as well as providing information on similar national and international facilities that were compared to the Necsa facility.
- Chapter 2 aims to supply the reader with the theoretical considerations applicable to the Radioactive Effluent Treatment Facility. Chapter 2 explains the relevance of a Critical Group, the decay process of nuclides and the safety principle followed at the Radioactive Effluent treatment facility during normal operating conditions. Chapter 2 gives a detailed description of worker protection and monitoring programme as required by the ICRP [15] that are implemented at Necsa.

- Chapter 3 aims to introduce the reader to the different steps followed when conducting a Safety Assessment and allocating a hazard levels to a nuclear facility. Chapter 3 also gives a description of the Safety Assessment methodology.
- Chapter 4 discusses the results of the Safety Assessment and presents conclusions.

It should be noted that conventional and chemical hazards are not addressed in this Safety Assessment. The Occupation Health and Safety Act of 1993, regulation 9 for Hazardous Chemical Substances [19], however, requires that an Occupation Health Risk Assessment must be performed on facilities and must address conventional and chemical hazards and the control program. This Occupation Health Risk Assessment is a separate study done by an Occupational Health specialist.

CHAPTER 2

THEORETICAL CONSIDERATIONS

2.1 Normal operating conditions of Radioactive Effluent Treatment Facilities

An inevitable part of any industrial process and in particular a nuclear process is the generation of waste. Radioactive solid, liquid and gaseous waste is generated and treated by various processes that are agreed upon at an international level and are continually evaluated to ensure continuous improvement and development of future treatment processes.

It is an accepted international practice to release treated radioactive effluent into the environment under controlled conditions [20]. The release criteria are site specific and the limits are determined by means of an assessment that is done on the impact that the discharged effluent will have on the environment and the public (Critical Group) of that area. There are guidelines given by the IAEA [20] set as a highest proposed value but the identified Critical Group might have a significant influence on the site specific values. Much lower values could be applicable to a specific site due to the specific behaviour habits of the Critical Group. The characteristics of the discharged effluent, the geological/meteorological behaviour of the area and Critical Group are unique to each area. It must also be clearly noted that the routine discharge limits are calculated and do not accommodate accidental discharge scenarios.

2.2 Critical Group

A Critical Group is defined as a small group of people being reasonably homogeneous with respect to its exposure for a given radiation source and given exposure pathway. The identification of a Critical Group is required to enable the demonstration of compliance to the set national and international requirements [20]. An area around a nuclear facility is identified and within that area a

radiological habit survey is conducted. Data with regard to eating habits, activities such as farming and recreational habits (e.g. housing, fishing, boating, skiing etc.) is collected and potential pathways of exposure are identified. A pathway habit-profile is compiled and a cut-off limit to consumption and exposure time is set to the profile. The individuals with the highest consumption rates or occupancy exposure time is then identified as the Critical Group all within the cut-off limit set on the pathway habit-profile.

Man-made radiation that includes radioactive elements called radionuclides which emit alpha, beta, or gamma radiation, or a combination of these types of radiation produced at nuclear facilities has the potential to reach the public through different pathways as shown in Fig. 2.1. A pathway outlines the route which radionuclide contaminants may follow to reach the public. Not all the pathways indicated in Fig. 2.1 are applicable to the Necsa site. The pathways applicable to the Necsa site are indicated in red. Radionuclides may enter the local environment by air or water. The Public and more specifically the Critical Group could inhale radioactive elements from the air and from water vapour. People could also absorb radiation from air or water through the skin or by drinking contaminated water. Radionuclides released into the air or water can also pass through the soil, plants, or wildlife and reach the public through ingestion of crops and game animals.

The primary potential pathways of radiation exposure to the Public include:

- Air and wind transport via re-suspension of surface soil contamination from current or historic nuclear activities.
- Discharge of radioactive effluent from nuclear facilities into rivers or the sea.
- Movement through groundwater from near surface disposal, deep geological disposal or nuclear incidents that lead to the contamination of groundwater
- Ingestion of contaminated game animals exposed to contaminated soils and plants on the nuclear site.

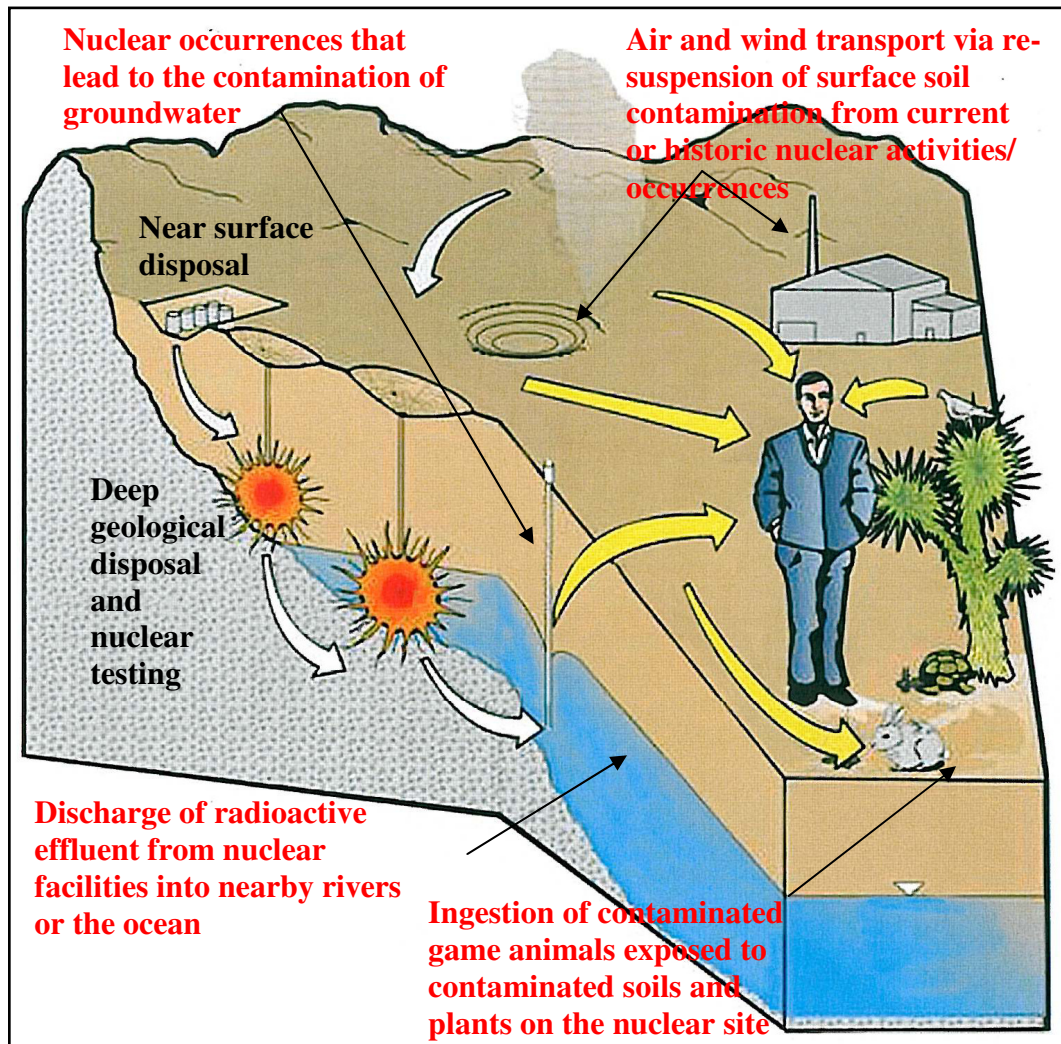


Figure 2.1: Different pathways of radionuclides reaching the Public.

2.3 Relevant nuclides at the Necsca site

Medium Activity Effluent is mainly generated at the SAFARI-1 Research Reactor and the medical isotope manufacturing facilities. The Medium Activity Effluent is produced in the primary cooling loop in the SAFARI-1 Research Reactor through the fission of uranium. At the medical isotope manufacturing facilities nuclides are generated during the dissolving of the irradiated uranium target plates.

When ^{235}U undergoes fission (absorbs neutrons) inside the SAFARI-1 Research

Reactor during a controlled criticality cascade various fission nuclides or fragments are formed. The average of the fragment mass is about 118, but very few fragments are found near 118. The fission process leads typically to break up into two unequal mass fragments with the most probable fragment masses being around mass 95 and 137 as indicated in the fragment mass distribution shown in Fig. 2.2 [21].

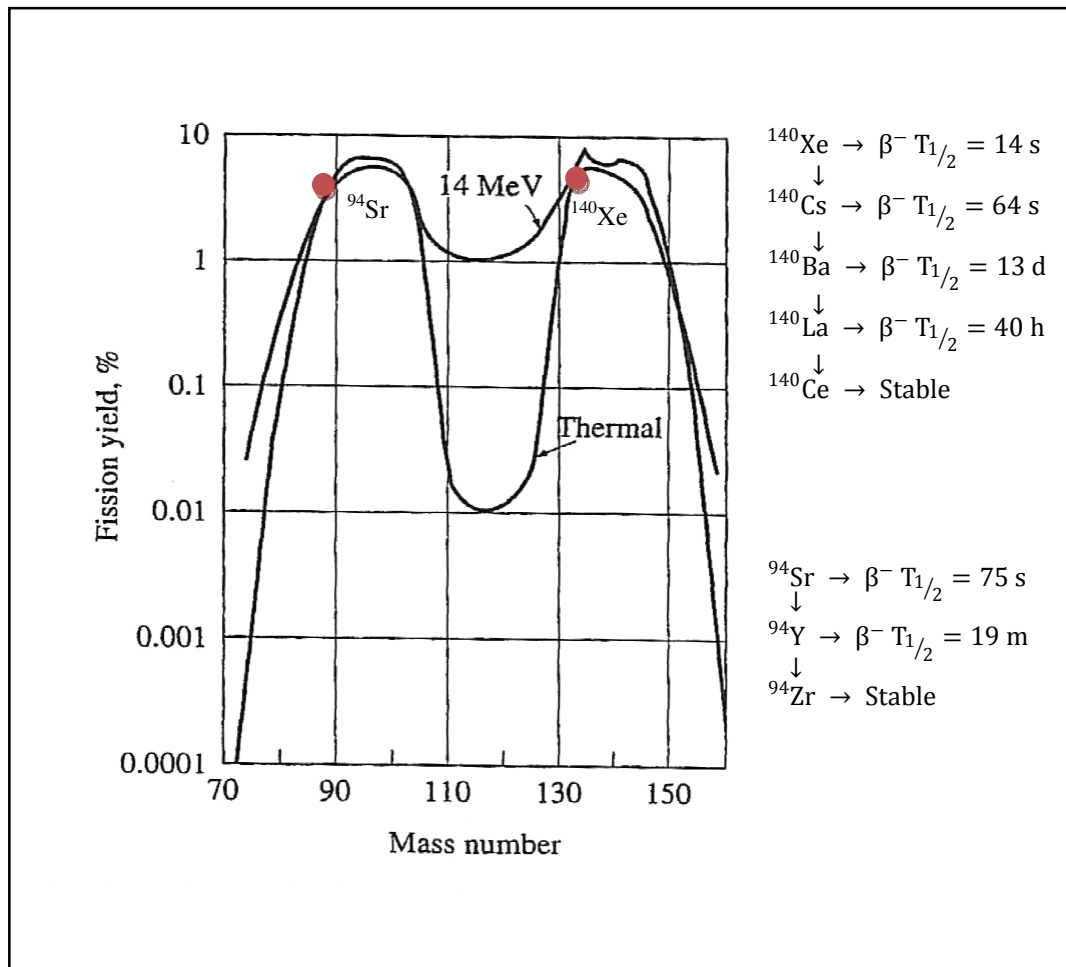
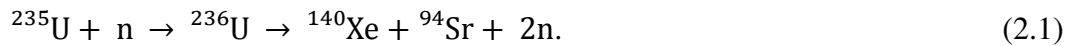


Figure 2.2: Fission-product yields for thermal and 14 MeV fission neutrons in ^{235}U indicating a common pair of fragments and subsequent decay (taken from [21]).

By way of example, a common pair of fragments from ^{235}U fission is ^{140}Xe and ^{94}Sr formed in the following process:



These are indicated in Fig 2.2, but are not relevant to this Safety Assessment because of their short half-lives and the short half lives of their decay products and as such these fragments and their decay products never enter the Radioactive Effluent Treatment Facility.

This particular set of fragments from ^{235}U fission undergoes a series of β^- decays to form stable end products. Most of the fission fragments are highly unstable as shown in Fig 2.2. ^{140}Xe is highly radioactive, and decays with a half-life of 14 s and finally produces the stable isotope ^{140}Ce . ^{94}Sr decays with a half-life of 75 s, finally producing the stable isotope ^{94}Zr . The many other fragment pairs produced in the fission process follow the same pattern of decay as in the example above and thus also do not enter the Radioactive Effluent Treatment Facility. ^{131}I may give a higher initial dose, but its short half-life of 8 d ensures that it will soon disappear. ^{131}I is one of the particularly relevant nuclides when considering medical radioactive effluent but not a relevant nuclide for Necsra radioactive effluent.

2.5.1 2.3.1 Properties of ^{137}Cs

Produced in quantity and of particular concern is ^{137}Cs , being the focus of this Research Report. Caesium is a soft, malleable, silvery white metal and is one of only three metals that is a liquid at room temperature. ^{137}Cs is the most dangerous radioisotope to the environment in terms of its long-term effects; it is water-soluble and extremely toxic in minute amounts. ^{137}Cs has an intermediate half-life of 30.32 y suggesting that it is not only highly radioactive but that it has a long enough half-life to be around for hundreds of years. Besides its persistence and high activity, ^{137}Cs has the further insidious property of being mistaken for potassium by living organisms and taken up as part of the fluid electrolytes. This

means that it is passed on up the food chain and re-concentrated from the environment by that process. The main health concern is associated with the increased likelihood for inducing cancer. While in the body, caesium poses a health hazard both from beta and gamma radiation. ^{137}Cs behaves in a manner similar to potassium and distributes uniformly throughout the body. Gastrointestinal absorption from food or water is the principal source of internally deposited ^{137}Cs in the general population. Essentially, all ^{137}Cs ingested is absorbed into the bloodstream through the intestines. ^{137}Cs tends to concentrate in muscle because of its relatively large mass. Like potassium, caesium is excreted from the body fairly quickly. In an adult, 10% is excreted with a biological half-life of 2 d, and the rest leaves the body with a biological half-life of 110 d. Clearance from the body is somewhat quicker for children and adolescents. This means that if someone is exposed to radioactive caesium and the source of exposure is removed, much of the caesium will readily clear the body along the normal pathways for potassium excretion within several months [22].

Various factors have been offered to explain and/or predict the wide variation in the retention of Cs among humans. The biological half-life of Cs has been expressed as an increasing function of body mass and also as an increasing function of age throughout life. Some early investigators attempted to describe accumulation of Cs in the body in terms of discrimination factors between Cs and the chemically similar element K, but such factors were found to oversimplify the relation between Cs and K and they were soon abandoned. K may however be a useful index, for predicting the retention patterns of Cs in individuals. A predictive model for individuals is constructed, using observed relationships between the mass of total-body K (K_t) and the parameters in the standard two-exponential retention model for Cs. There is still a lot of uncertainty and assumptions are still made conservatively until more certainty is obtained [22]. Due to the nuclear weapons testing programs and nuclear power stations, ^{137}Cs is of more concern in the United States of America (USA) than in the rest of the world. The USA has sponsored several studies to determine the health effects of ^{137}Cs [22].

As mentioned, the isotope ^{137}Cs has a half-life of 30.32 y and undergoes only β^- decays and has two branches. In the one β^- process (5% branching ratio) a β^- particle is emitted with an endpoint energy of 1.170 MeV, producing ^{137}Ba in its stable ground state. In the other β^- process (95% branching ratio) a β^- particle with an endpoint energy of 0.510 MeV is emitted to produce $^{137\text{m}}\text{Ba}$ in a metastable state. The $^{137\text{m}}\text{Ba}$ in the metastable state then decays via isomeric transition with the emission of a γ -rays of energy 0.662 MeV. $^{137\text{m}}\text{Ba}$ in a metastable state has a half-life of 2.55 minutes and is responsible for the emission of γ -rays. The decay scheme of ^{137}Ba is shown in Fig. 2.3.

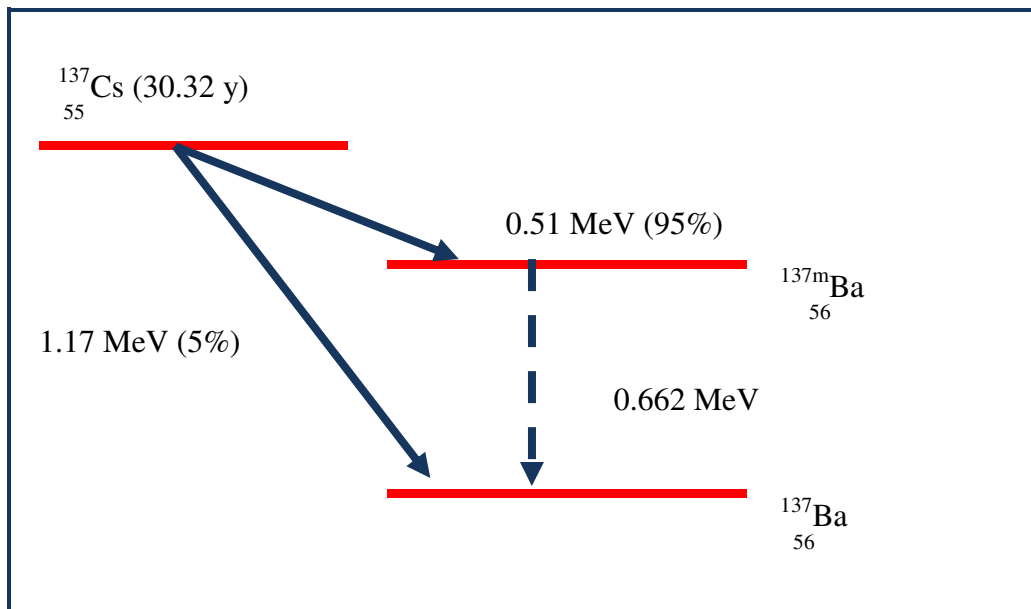


Figure 2.3: Illustration of the decay scheme of ^{137}Cs .

The photon energy of ^{137}Ba can be used in food irradiation, or in radiotherapy of cancers. ^{137}Cs is also one of the most common radioisotopes used in the industry.

Thousands of devices use ^{137}Cs , for example:

- moisture-density gauges, widely used in the construction industry,
- leveling gauges, used in industries to detect liquid flow in pipes and tanks,
- thickness gauges, for measuring thickness of sheet metal, paper, film and many other products and
- well-logging devices in the drilling industry to help characterize rock

strata.

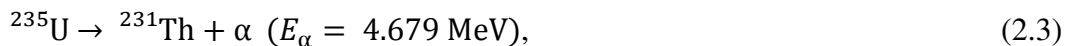
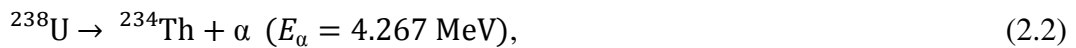
2.4 Basic decay processes and biological significance

3.5.1 2.4.1 Natural α -decay of uranium

At this stage it must be clearly noted that the Medium Activity Effluent is mainly generated through the fission of ^{235}U and not through the natural decay of uranium. Naturally occurring uranium contains two main isotopes, ^{238}U (99.275%) and ^{235}U (0.72%). Both isotopes are radioactive and, respectively, have the following half-lives, 4.468×10^9 y and 7.038×10^8 y [23].

There is however a third isotope of insignificant quantity ^{234}U (0.005%), with a half-life of 2.48×10^5 y, much shorter than the age of the earth (4.5×10^9 y), and has therefore almost all decayed away.

Both ^{235}U and ^{238}U form naturally occurring decay series starting with the α -decay of the parent nucleus to the corresponding daughter nucleus:



where E_{α} corresponds to the energy of the α -particle emitted. The resulting daughter nuclei of the two decay series are also unstable and for ^{235}U the series ends with stable ^{207}Pb , while for ^{238}U the series ends with the stable ^{206}Pb . This is achieved by α -decay or β^{-} emission until stability is reached.

The pre-formed α -particles are released from the unstable parent nucleus by a process of quantum mechanical tunneling through the Coulomb barrier. Lower energy α -particles ($E_{\alpha} \approx 4$ MeV) are associated with very long half-lives ($T_{1/2} \approx 10^9$ y) while higher energy α -particles ($E_{\alpha} \approx 9$ MeV) are associated with very short half-lives ($T_{1/2} \approx 1$ ms). In natural uranium, these decay chains

generally are in secular equilibrium. This means that in 1 g of natural uranium each nuclide of the ^{238}U series has an activity of 12 356 Bq and each nuclide of the ^{235}U series an activity of 568 Bq. However, the SAFARI-1 research reactor is loaded with enriched uranium fuel ^{235}U (90%) and the target plates used in the medical isotope manufacturing are also made of enriched uranium ^{235}U (46%). Figure 2.4 shows the ^{238}U decay process.

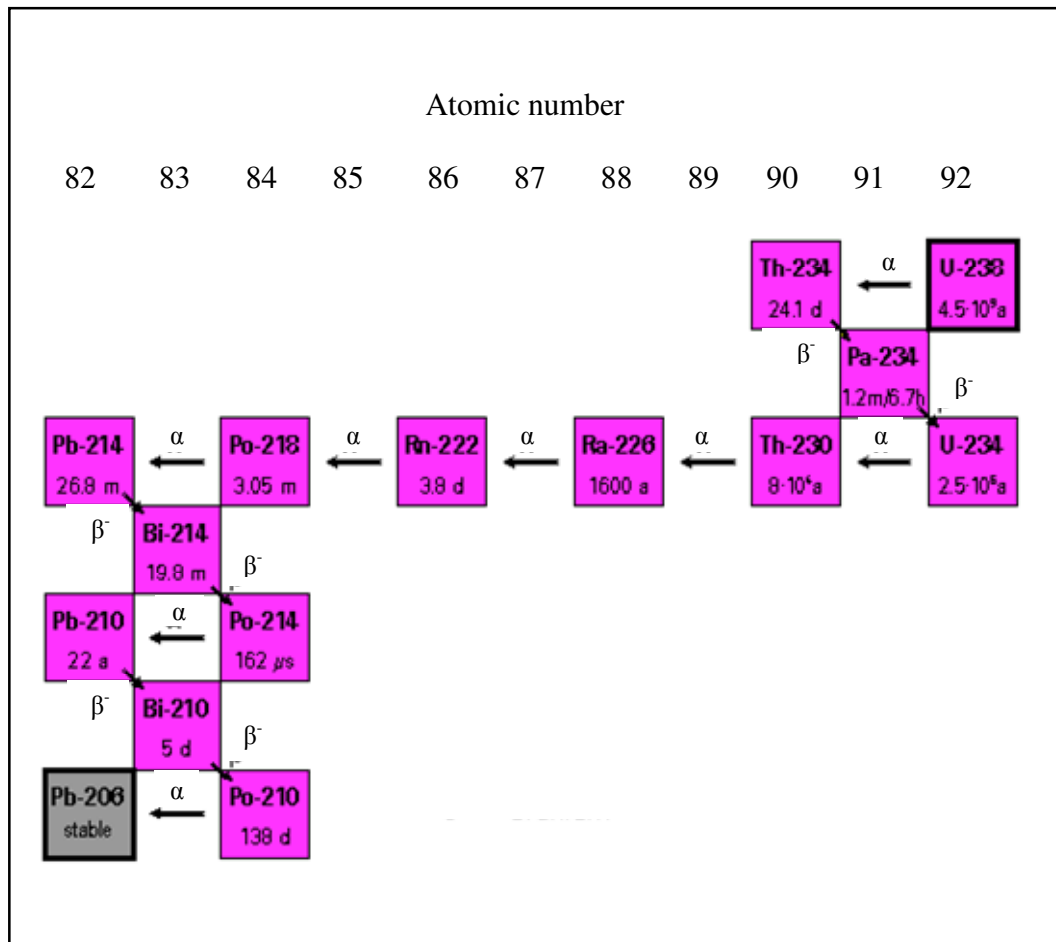


Figure 2.4: ^{238}U decay series (times shown indicate the half-life of each daughter nucleus).

An α -particle (^4He nucleus) has a double positive charge and is highly ionizing as it slows down in matter. On living tissue an α -particle can have significant health effect; when interacting with living matter substantial destruction to molecules can be caused while slowing down to become a neutral helium atom. The

concentration of α -emitters specified in the waste acceptance requirements is very low, because uranium containing effluent is not allowed in the Medium Activity Effluent facility due to the possible concentration of the ^{235}U isotope in the evaporator that could lead to criticality. The presence of uranium can be easily detected from its α -emission during sampling of the effluent prior to discharge to the Radioactive Effluent Treatment Facility.

Health effects of α -emitters are not a major concern in the operation of the Radioactive Effluent Treatment Facility due to the very low probability for the presence of uranium that will result in a low probability of inhalation and ingestion. As also previously mentioned, the concentration of α -emitters is limited due to a criticality hazard. The concentration of other α -emitting nuclides are minute and are not of concern to the workers health although the medical surveillance program for the workers of the Radioactive Effluent Treatment Facility is such that it will detect exposure to α -particles. The medical surveillance program worker monitoring is explained in more detail at a later stage in this report.

4.5.1 2.4.2 β -decay processes

There are three common forms of beta decay:

- (a) β^- - decay: (Electron emission, a neutron turns into a proton)
- (b) β^+ - decay: (Positron emission, a proton turns into a neutron)
- (c) Electron Capture.

(a) β^- -decay

Certain nuclei which have an excess of neutrons may attempt to reach stability by converting a neutron into a proton with the emission of an electron and an anti-electron neutrino, $\bar{\nu}_e$:



The electron is called a β^- -particle. The electron does not exist inside the nucleus

but is spontaneously produced in the decay and is ejected. The energy of the emitted electron is of concern to radiation protection.

β^- -decay occurs in ^{131}I which decays into ^{131}Xe . The Mass Number in the case of electron emission remains the same and the Atomic Number increases by one which is characteristic of this type of decay. The decay of ^{131}I is not relevant to this Safety Assessment due to the short half live of ^{131}I ($T_{1/2} = 8$ d), but it is a very relevant nuclide when considered in the medical treatment and diagnose process. The decay of ^{131}I is another example of β^- -decay and is illustrated as follows:



Due to the three-body nature of the decay a continuous distribution of electron energy is produced up to the end point energy (typically ≈ 1 MeV), the Q-value of the reaction.

(b) β^+ -decay

When the number of protons in a nucleus is too large for the nucleus to be stable it may attempt to reach stability by converting a proton into a neutron with the emission of a positively-charged electron and an electron neutrino. The positron is the β^+ particle. The Mass Number remains the same and the Atomic Number decreases by 1:



A typical example of this type of decay occurs when ^{22}Na which decays into ^{22}Ne :



Again as with β^- -decay, due to the three body nature of the decay a continuous distribution of positrons is produced upto the end point (typically ≈ 1 MeV), the

Q-value of the reaction. It should be noted that the positron quickly annihilates with an electron to produce two γ -rays with an energy of 511 keV, the rest energy of the electron:



(c) Electron Capture

In this third form of beta decay an inner orbiting electron is attracted into an unstable nucleus where it combines with a proton to form a neutron. This process is also known as **K-capture** since the electron is often attracted from the K-orbit of the nucleus. The Mass Number is unchanged in this form of decay and the Atomic Number is decreased by 1:



A typical example of this type of radioactive decay occurs when ^{55}Fe decays into ^{55}Mn :



5.5.1 2.4.3 γ -decay

The gamma decay process is the most prominent decay process of interest to the Radioactive Effluent Treatment Facility. Gamma decay involves the emission of energy from an unstable nucleus in the form of electromagnetic radiation. The radiation can be characterised in terms of its frequency, its wavelength and its energy. Very low energy electromagnetic radiation called radio waves, infra-red radiation at a slightly higher energy, visible light at a higher energy still, then ultra-violet radiation and the higher energy forms of this radiation are called X-rays and γ -rays. These radiations form what is called the Electromagnetic Spectrum. The high energy radiated by X-rays and γ -rays is illustrated in Fig. 2.5. The energy, frequency and wavelength of the radiation make up the

Electromagnetic Spectrum. The X- and γ -ray window has been used for diagnostic and therapeutic medical applications for over one hundred years. The Nuclear Magnetic Resonance (NMR) window has more recently found application in medicine and uses radio waves in an imaging technique called Magnetic Resonance Imaging (MRI).

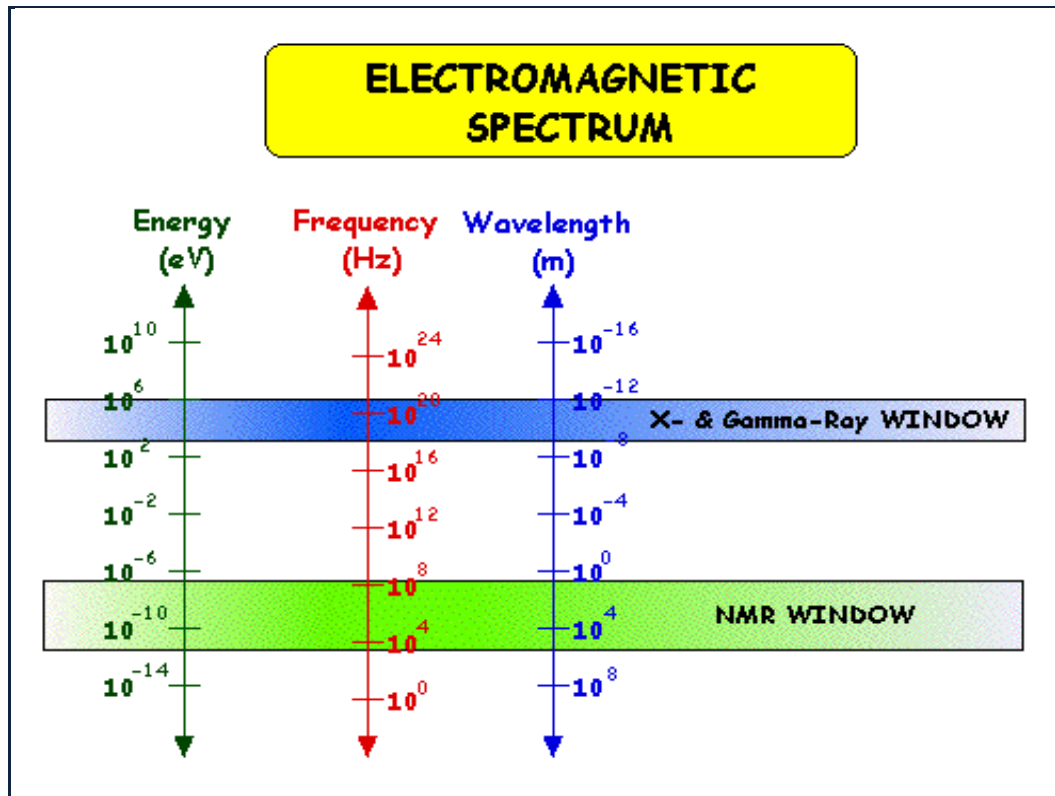


Figure 2.5: Electromagnetic spectrum for radiation.

All radioactive materials decay and eventually become stable, losing their harmful radiation characteristics. However, the rate at which different forms of radioactive material decay varies enormously. When radioactive materials enter the body, some will become harmless very quickly, whereas others may take so long to decay that they can remain active for the rest of a person's life. The type and energy of the radiation that it emits must be taken into account, along with the probability of the radioactive material being excreted from the body. In the case of internal and external exposure, the fraction of radiation energy absorbed in the

body/organs must also be taken into account. This concept is generally referred to as the effective half-life. The effective half-life for a radionuclide within the body is the period of time needed for the compound to be decreased by 50%. This occurs due to the combined effect of the biological half-life and the physical half-life for the compound:

- The biological half-life is the time required for half of the material to be eliminated from body through excretion in urine, stool, sweat or other routes.
- The physical half-life is the time required for the compound to lose half of its activity through radioactive decay.

2.5 Safety principles followed at the Necsa Radioactive Effluent Treatment Facility

The radiological safety challenges of the Necsa Radioactive Effluent Treatment Facility are the management of radiation and airborne contamination hazards of the Medium Activity Effluent facility. This Safety Assessment evaluates design safety principles as well as administrative safety principles applicable to the Radioactive Effluent Treatment Facility.

The safety principle hierarchy followed for the Radioactive Effluent Treatment Facility to address the hazards was extended as far as practicable to the following:

- selection of processes to eliminate hazards (this, however, was not entirely possible since the Radioactive Effluent Treatment Facility does not have control over the generation of Low Activity and Medium Activity Effluent),
- passive control through design by the application of engineered features (e.g. shielding),
- active control through design by the application of engineered features (e.g. ventilation) and
- administrative controls by means of procedures although engineering controls provide better control than administrative controls it is not

always cost effective and practical to implement.

There are four major ways to reduce radiation exposure to workers or to public:

- **Shielding:** Use proper barriers to block or reduce ionizing radiation.
- **Time:** Spend less time in radiation fields.
- **Distance:** Increase distance between radioactive sources and workers or population.
- **Amount:** Reduce the quantity of radioactive material for a practice.

Of these four ways shielding as an engineering control is implemented and as administrative control the time spend and reducing quantities are implemented at the Radioactive Effluent Treatment Facility.

6.5.1 2.5.1 Passive controls applicable in the Radioactive Effluent Treatment Facility

Shielding is a passive engineering control measure and is more effective than administrative controls. All equipment containing Medium Activity Effluent are housed in cells providing sufficient shielding to ensure that operators will not be exposed to radiation levels (dose) higher than $1 \mu\text{Svh}^{-1}$ as stipulated by the Regulator outside the cells under normal operating conditions.

Figure 2.6 shows the different shielding required for the protection against different radiation rays. Shielding is only required for processes that emits γ -rays.

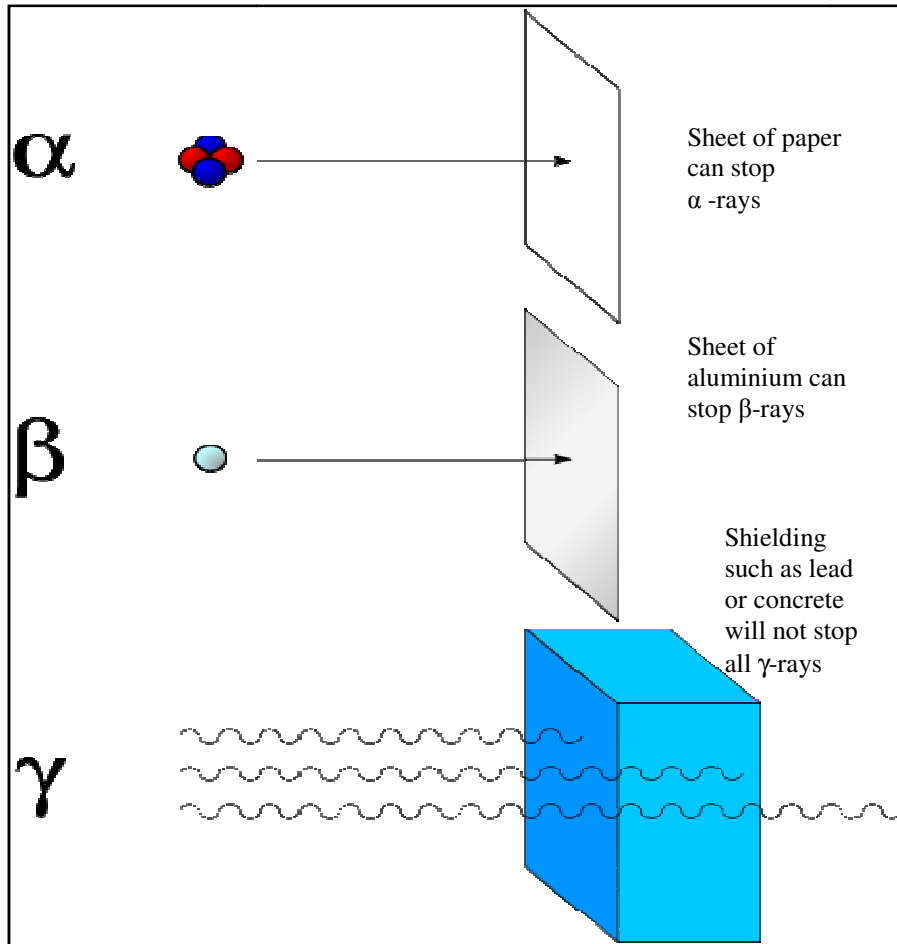


Figure 2.6: Demonstration of different Shielding requirements.

Under conditions of good geometry, the attenuation of a beam of γ radiation is given by:

$$I = I_0 e^{-\mu t}, \quad (2.11)$$

where t is the thickness of material and μ is the linear attenuation coefficient [24]. I_0 is the intensity of radiation on the source side and I is the intensity of radiation that gets through (in the case of the Radioactive Effluent Treatment Facility it is limited to $1 \mu\text{Svh}^{-1}$). The value of $1 \mu\text{Svh}^{-1}$ is set as an ALARA objective [40].

The limit for white classified radiological areas are less than $2.5 \mu\text{Svh}^{-1}$ [34]. The areas outside the cells are classified as white with no radiological control measures required with regard to occupational exposed workers.

Shielding reduces the intensity of radiation exponentially depending on the thickness of the shielding material and the type of material. The effectiveness of γ -ray shielding is frequently described in terms of the half value layer (HVL) or the tenth value layer (TVL). The value layers are the thicknesses of an absorber that will reduce the γ radiation to half, and one tenth of its intensity respectively.

The most effective gamma shields are materials which have a high density and high atomic number, such as lead, tungsten, and uranium among others. Generally speaking these materials are expensive, so, in situations where space is not a constraint and where structural strength is required, concrete is used even though it is a less effective shielding material. Lead shields are frequently used where space is limited or where only a small area of absorber is required. When added thicknesses are used, the shielding multiplies. The effectiveness of a shielding material in general increases with its density [24]. Approximate half-value layer (HVL) and tenth-value layer (TVL) for lead and concrete as attenuation material against radiation from ^{137}Cs are:

- HVL in lead for ^{137}Cs is 6.5 mm;
- TVL in lead for ^{137}Cs is 21 mm;
- HVL in concrete for ^{137}Cs is 48 mm;
- TVL in concrete for ^{137}Cs is 157 mm [24].

There are a few pipelines, which form part of the Medium Activity Effluent process, that are not shielded and access is limited by administrative controls. The implementation of engineering control is investigated as part of continuous improvement actions and the application of ALARA. Currently, the layout of unshielded pipelines inside the facility is such that it will not be possible to shield the pipelines with concrete in order to obtain sufficient shielding due to limited space. It could, however, be possible to obtain sufficient shielding by using lead

as shielding material. Shielding is required to protect the workers against the γ -rays that are emitted by ^{137}Cs in the Medium Activity Effluent.

A typical recommendation from this Safety Assessment could be to register a project to calculate the shielding required on the Medium Activity Effluent pipelines as a passive engineering control and to investigate implementation.

7.5.1 2.5.2 Administrative control applicable in the Radioactive Effluent Treatment Facility

The different areas inside the Radioactive Effluent Treatment Facility are zoned by permanent structures (walls with security controlled doors or step over benches) clearly indicating the boundaries of the different zones with the entrance and exit of each zone signposted. Access to areas with high radiation levels (inside the cells) are not part of normal operation and access is only rendered under strict administrative controls. Access is only required during the performance of ad hoc tasks and maintenance. Control is such that events with a probability of 10^{-2} are covered under normal operations.

Most of the areas inside the Radioactive Effluent Treatment Facility that contain Medium Activity Effluent have a radiation dose rate of $25 \mu\text{Svh}^{-1}$. Access to these areas is restricted to Radioactive Effluent Treatment Facility workers only and they are under the specific medical surveillance programme. Ad hoc entry by non-radiological workers is only allowed under controlled conditions. Should maintenance on equipment be required in the restricted areas the equipment is conditioned (flushed and separately shielded) to lower radiation levels. A radiological survey is performed prior to allowing maintenance workers in such areas. After the radiological survey is performed, an ALARA review and dose budgeting are performed to ensure that the task will not result in exceeding dose constraints. These tasks are continuously monitored by a facility Radiation Protection Officer (RPO).

Under normal operating conditions the contamination levels are controlled by ventilation and administrative controls. The normal contamination levels are well controlled to below 0.4 Bqcm^{-2} for α and to below 4 Bqcm^{-2} for β and γ contamination. Historically, the highest contamination levels found in any cell were 200 Bqcm^{-2} for α and 2000 Bqcm^{-2} for β/γ activity. The areas with high contamination levels are decontaminated under the control of a Radiological Work Permit. Areas are decontaminated as quick as possible after spillages or other causes of contamination above normal operating levels.

The principles of ALARA for the workers are applied by limiting time spent in classified areas, reducing radiation levels prior to entering an area, monitoring worker exposure etc. The ALARA objective for the workers of Necsa is that the average annual effective dose to the occupationally exposed workforce of Necsa shall not exceed 4 mSvy^{-1} [5]. The exposure dose level set by the ICRP is 50 mSvy^{-1} [8] and the value set by the NNR is 20 mSvy^{-1} .

Necsa management and employees are fully aware of the need to avoid environmental damage, especially to the beautiful surroundings at Pelindaba. They are committed to manage the environmental effects of their nuclear, chemical and related activities so as to ensure sustainable development, and to protect the health and safety of their workers, and that of the public, by developing and operating waste management processes. The Necsa website contains the Safety Health and Environment policy [9] and is available to the public.

2.6 Medical surveillance and Worker Monitoring Program at the Necsa site

The operators of the Radioactive Effluent Treatment Facility are all registered as occupational exposed workers and are subject to annual medical examination. The annual medical examination for radiation workers on the Necsa site differs from person to person. Each radiological worker has a Person-Job Specification

to enable the Necsca Occupational Physician to determine the most appropriate medical examination programme as recommended in Refs. [8, 15].

The estimation of internal dose needs to be inferred from the measurement of activity in the individual (by in-vivo or in-vitro methods) or in the workplace environment [15]. Various methods exist that can be used to perform individual monitoring for intakes of radionuclides i.e. whole body counts, excreta monitoring, air sampling with personal air samplers or any combination of these techniques [25, 26]. The choice of a measurement technique is determined by several factors:

- Radiation emitted by the radionuclide.
- Metabolic behaviour of the contaminant.
- Retention of the radionuclide in the body, taking into account, both biological clearance and radioactive decay.
- Required frequency of measurements.
- Sensitivity, availability and convenience of the appropriate measurement facilities.



Figure 2.7: Whole body counter used for internal monitoring.

The whole body counters are used in the nuclear industry as part of the medical surveillance programs. Figure 2.8 is a photo of a whole body counter very similar to the one used by Necsa. The advanced body counters can provide accurate and reliable measurements of uranium, plutonium and americium in the lungs and it can provide fast, accurate measurements of fission and activation products in the body.

The whole body counters are installed in a well shielded box shielding equivalent to a minimum of 10 cm of low background steel. The shields reduce spectral background due to bremsstrahlung interactions from the naturally occurring ^{40}K within the workers body. The reduction or limiting of background helps to improve the detection capabilities of the detectors and reduce count times. The standard whole body counters include a lung counting detector positioning mechanism, a subject bed and a scanning whole body counting mechanism. The lung counting detector positioning mechanisms features six degrees of positioning freedom to allow accurate placement of the detectors over the lungs for improved detection limits. The lung counting detector positioning mechanism accommodates sodium iodide energy detector crystals

During normal or routine counting situations the worker will be lying on the bed. The system operator will position the detector array over the workers lungs and in uniform contact with the workers upper chest. The shield door of the box will be closed and the count can then start. The whole body counter is controlled by software that will turn on the high voltage for the sodium iodide detectors; it controls the scanning mechanism for the sodium iodide detector and begins the data acquisition and spectral displays. When the count is done, the spectral data is stored, analyzed, the results displayed and/or printed, and any appropriate warning messages are generated to the system operator. This data will be store on the workers medical surveillance file and if any warning messages were generated an investigation will be launched and if necessary a nuclear occurrence will be registered.

The medical examination programme for the Radioactive Effluent Treatment Facility exposed personnel includes a whole body count, together with blood and urine sampling. The combination as an annual examination is regarded as sufficient for the workers of the Radioactive Effluent Treatment Facility. If exposure is suggested an ad hoc examination will be performed that will include a whole body count. The main radionuclide of concern at the Radioactive Effluent Treatment Facility is ^{137}Cs as explained previously. Inhalation and ingestion of ^{137}Cs can be easily detected by the whole body counter, even if small amounts have entered the body.

Different methods of adequate sensitivity are available and the general order of preference in terms of accuracy of interpretation is:

- body activity measurements,
- excreta analysis and
- personal air sampling.

Direct measurement of the body or organ (done by the whole body counter) provides a quick and convenient estimate of activity in the body/organ [16]. The whole body count is appropriate for the surveillance of the Radioactive Effluent Treatment Facility workers. The whole body counter is an effective instrument for the detection of radionuclides that emit radiation that can penetrate through the body and be detected by the sodium iodide detectors. In principle, this technique can be used for radionuclides that emit X-or γ -radiation or radionuclides that emit positrons, since they can be detected by measuring the annihilation photons mentioned in Section 2.4.2 (b), energetic β^- -particles that can be detected by measuring the bremsstrahlung and some α -emitters that can be detected by measuring their characteristic X-ray radiation that accompany the α -decay [16].

^{137}Cs present in the Radioactive Effluent Treatment Facility, if inhaled and ingested will emit sufficient γ -radiation (see Section 2.3.1 and Fig. 2.3) that will be detected by the whole body counter even at intakes that are small compared to annual limits. Necsa has a new whole body counter that can also detect long-lived

α -emitters well within the annual limits on intake, depending on the time lapse after intake and examination.

As an additional precaution and application of ALARA, each operator working on the Radioactive Effluent Treatment Facility is issued with a Thermoluminescent dosimeter (TLD), see Fig. 2.8, which is replaced on a monthly basis. TLDs are ideal for the use at the Radioactive Effluent Treatment Facility because the TLDs have the required sensitivity to detect the radiation. TLDs can be re-used after analysis. The TLD also provides a very suitable measurement of total accumulated dose over a period of exposure [27].



Figure 2.8: Thermoluminescent dosimeter (TLD).

TLD detectors utilize the electron trapping process. The material is selected so that electrons trapped as a result of exposure are stable at normal temperatures. In the TLD mechanism, as illustrated in Fig 2.9 and Fig 2.10, there are insulators whose electrons are bound in the valence band at normal temperature. During exposure to radiation, free electrons in the TLD crystals become trapped in lattice

imperfections. Electrons are elevated from the valence to the conduction band, but are then captured at one of the trapping centres. If the trap energy level is significantly below the conduction band, the probability per unit time that the electron will escape back to the conduction band is small at ordinary room temperature. Exposure to a continuous source of radiation will only increase the number of trapped electrons. The TLD will be activated from a state of equilibrium into a meta-stable state but on heating the TLD (typically to 300° C) it will be relaxed into the equilibrium state again. During this relaxation the electrons escape the traps and release visible light which can then be measured using a photomultiplier. The amount of light detected is related to the radiation dose received by the TLD detector [27].

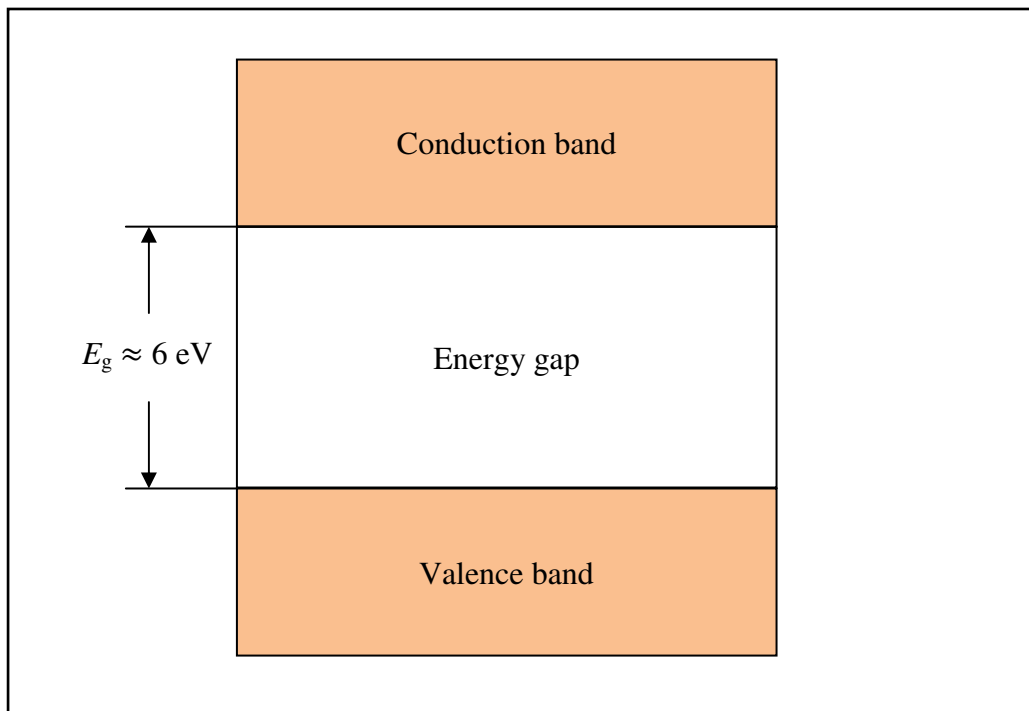


Figure 2.9: Energy band structure of electron trapping process in TLD detectors.

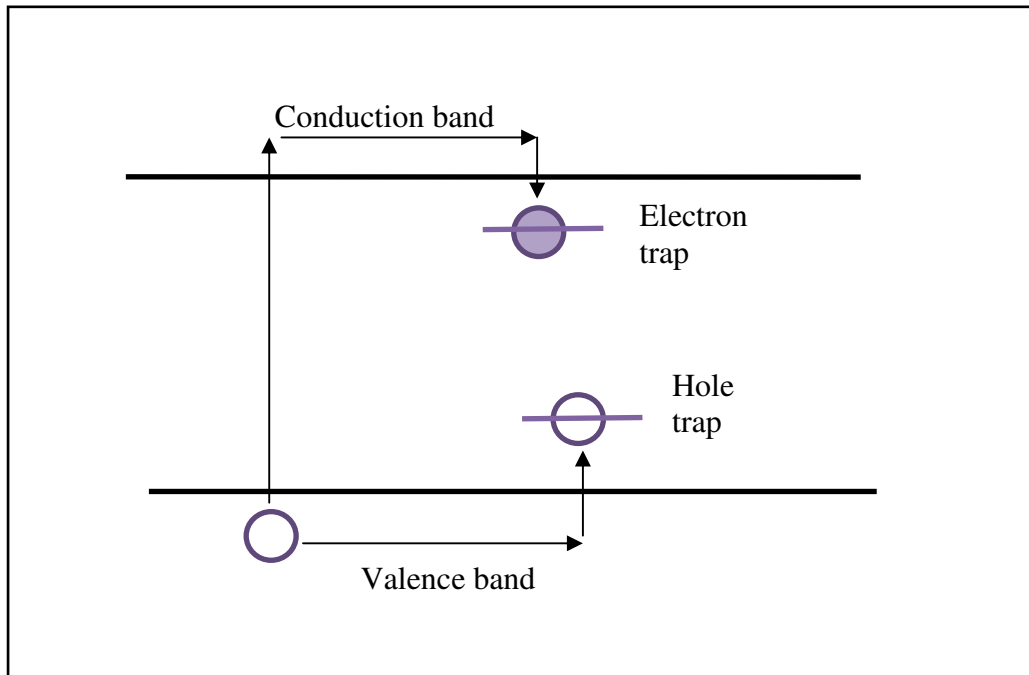


Figure 2.10: Electron trapping used in TLD mechanism.

Radiation detection equipment has gone through an amazing evolution in size reduction, sensitivity and power [28]. Since the turn of the century, digital signal processing technology has been steadily adopted in radiation monitoring equipment in order to make more accurate and precise measurements of radiation dose. Figure 2.11 is a photograph of a detector functioning by digital signal processing. Silicon semiconductor detectors and dosimeters, which can detect α , β^- and γ radiation as well as neutrons, are replacing the conventional radiation detectors in radiation monitoring systems.



Figure 2.11: Example of an electronic radiation monitoring device

The main purpose of a personal dosimeter is to monitor the radiation dose of personnel on a day-to-day basis. Besides routine monitoring, detectors are also used in personnel dosimetry for the determination of absorbed doses as a result of accidental exposure to radiation. The cost of internal dosimetry monitoring programmes can be significantly greater than those for external dosimetry. It is thus also important to implement the most suitable and effective routine monitoring programmes to ensure optimisation of cost. It is also important to select the correct radiation monitoring equipment.

2.7 Exposure pathways at the Necsa site

Workers at the Radioactive Effluent Treatment Facility can be exposed to ionising radiation through inhalation and ingestion of radioactive particles causing internal irradiation of the body or external irradiation through nuclides emitting high-energy photons that can cause damage to the organs in the body by penetrating the body. The Public can be exposed to ionising radiation from different sources and in different ways when radioactive material is discharged into the environment. Radionuclides can be present in drinking water and people can be exposed by direct intake, or the water can be used to irrigate crop and people can be exposed indirectly by eating the food that was irrigated by the contaminated water. The Public can even be exposed to external radiation when exposed the sources of radiation that emits high-energy photons for example from stolen sources. These different ways or routes of exposure are referred to as exposure pathways.

Mathematical models of radionuclide behaviour can be used to estimate the dose rates to affected people for accidental releases. For a given release to the environment or into a facility, the first step in assessing radiation doses to people is usually to identify the exposure pathways by which those people are exposed to radiation. It is preferable, and usually more accurate, to base estimates of doses on measured dose rates or on measurements of radionuclide concentrations in samples taken from environmental materials. Mathematical models should only be used if no data are available or the available data is not sufficiently comprehensive. Models are also to be used when a possible future practice that may release radionuclides into the environment, is being evaluated. Furthermore, doses to the Public (collective doses) invariably have to be calculated using mathematical models since the doses are often delivered over large temporal and spatial scales.

Radioactive material may be released to either the atmospheric or the aquatic environment and models are required to describe the transfer of radionuclides through the respective parts of the biosphere to people. Radionuclides discharged to

the atmosphere are dispersed by normal atmospheric mixing processes such as movement of air due to temperature differences. Airborne radionuclides can give rise to exposure by two principal routes (exposure pathways): external radiation by photons and electrons emitted as a result of the radioactive decay process and internal radiation following the intake of radionuclides. Aerosol particles as a primary contamination is not addressed since aerosol and solid waste are excluded from this assessment.

In the case of the Radioactive Effluent Treatment Facility, exposure to radionuclides due to airborne contamination is due to secondary exposure as a result of liquid effluent that dries producing aerosol particles that re-suspend to become airborne. Radionuclides that become airborne can be re-deposited from the atmosphere by impacting with the underlying surface or by rainfall. This transfer onto land surfaces may lead to further exposure of people by the following three important routes:

- external radiation from deposited activity,
- internal radiation from inhalation of re-suspended activity or
- internal radiation from ingestion of contaminated food or inadvertent ingestion of soil.

The relative importance of these pathways depends on the radionuclide and the nature of the surface onto which the deposition occurs. Appropriate dosimetric models and habit data are also required to determine individual and collective doses. In order to estimate collective doses, spatial distributions of population and agricultural production are required for the region of interest.

Liquid radioactive effluents as in the case of the Radioactive Effluent Treatment Facility may be discharged to freshwater (particularly rivers e.g. the nearby Crocodile River), estuaries or the marine environment. Radionuclides discharged to rivers are dispersed due to general water movements and sedimentation processes. In the case of the Radioactive Effluent Treatment Facility, the discharge limits were calculated on the dispersion taking the most conservative dilution factor into account over the past 20 years.

The principal routes leading to the exposure of people are:

- external radiation from sediments,
- ingestion of food obtained from the river,
- drinking water taken from the river and
- water used for irrigation of crops and pasture.

By way of example, in the case of Koeberg Nuclear Power Station the large sea-water volumes and the fact that two pumps are required to be in operation during discharge ensures a constant dilution factor and no conservative value has to be assumed and justified. The local features of the environment at Koeberg Nuclear Power Station, in particular tidal currents and the degree of sedimentation, initially determine the dispersion of radionuclides discharged into the marine environment. General water movement and sedimentation processes in the larger sea and ocean masses influence subsequent dispersion. There are again a number of pathways leading to the exposure of people including:

- ingestion of marine foodstuffs,
- external radiation from activity on beaches and
- inhalation of sea-spray.

None of these last pathways are relevant to the Necsa site.

2.8 Dose constraints

The NNR is responsible for specifying the value of dose constraints, although registrants or licensees may additionally specify them in their internal rules or systems as long as it does not exceed the levels set by the regulator. When establishing the dose constraints for the Radioactive Effluent Treatment Facility the relevant sources were clearly described. The ICRP recommendation [8] for an upper value for the dose constraint for waste management activities for members of the public of $300 \mu\text{Svy}^{-1}$ was not adopted by Necsa but a more conservative limit of $250 \mu\text{Svy}^{-1}$ [5]. This lower dose constraint for Necsa was set by the NNR. It is general practice to set lower values for the dose constraint by regulatory

authorities for particular situations or types of installation and as an application of ALARA. The lower dose constraint for Necsa also influences the choice of the Radioactive Effluent Treatment Facility value. The lower value for a dose constraint reflects the need to ensure that the dose to the Critical Group now and in the future is unlikely to exceed the dose limit. Doses expected to be incurred by the Critical Group from all other activities or sources on the site, to which they are also exposed, was taken into account.

In setting values for the liquid and gaseous dose constraints, allowance has to be made for the relative contributions to the Public's dose from the two discharge modes (based on historical precedent), as well as for dose estimation uncertainties and the ALARA principle. The combined liquid plus gaseous discharge dose constraint is fixed ($250 \mu\text{Sv}^{-1}$) as explained previously but the ratio of these doses is subject to review on a routine basis. The values of these annual constraints are currently as follows [35]:

- radioactive effluent discharges: 150 μSv
- radioactive gaseous discharges: 50 μSv
- dose margin, ALARA and uncertainties: 50 μSv .

CHAPTER 3

DETERMINING THE PARAMETERS, EXECUTION AND COMPLETION OF THE SAFETY ASSESSMENT

3.1 Introduction to Safety Assessment process steps

A Safety Assessment consists of different process steps in order to evaluate the hazards and the various controls in place to manage the risk associated with these hazards for normal operation of a facility as well as accident conditions.

The Safety Assessment for normal operation should address all the facility conditions under which systems and equipment are being operated as expected. This includes all the phases of operation for which a facility was designed to operate in the course of normal operation over the lifespan of the facility. The effects of variations in the inputs (feed material, source material, etc.) on normal operations should also be considered.

The objective of the Safety Assessment for normal operation should be to assess whether normal operation of the facility can be carried out safely, that is, whether radiological doses to workers and members of the public, and planned discharges of radioactive material from the facility, are within acceptable limits. In addition, the Safety Assessment should establish any appropriate requirements and limitations for safe operation of the facility.

The secondary objective of the Safety Assessment for normal operation is to identify the sources which could give rise to exposure to workers so that it can be included in the normal routine personnel monitoring programme. The worker exposure for the Radioactive Effluent Treatment Facility is calculated as part of the Safety Assessment for normal operation.

The impact of an event such as the loss of containment is calculated as part of the Safety Assessment to determine if it is safe to operate the facility and that enough controls are in place to manage the risks. Risks are conceived to be managed if the consequences of an event are acceptable and if not, the possibility of occurring has an acceptably low frequency as a result of the implementation of control measures ranging from engineering controls to administrative controls.

As a general rule to comply with radiological risk requirements, the nuclear safety requirements (design and operational) would be more stringent for facilities posing a higher risk than for facilities posing a lower risk. Facilities such as Nuclear Power Plants pose a higher risk than a radioactive waste management facility. A Radioactive Effluent Treatment Facility poses a much lower risk to the workers or the Public than a Nuclear Power Station.

Defence-in-Depth or different levels of protection requirements to illustrate that the possibility of a nuclear occurrence is very low or that the consequences will be managed to acceptable levels will not be the same for a Nuclear Power Plant and a small Laboratory set-up. This approach is referred to as a hazard graded approach and was applied when performing this Safety Assessment.

A hazard graded approach implies the initial determination of the hazard level of a facility or project. When determining the hazard level there are a number of variables that will be used in the calculation, many of which are unique to a facility or specific site where the facility is housed [29]. An important variable, for example, is the meteorological conditions at the site that will have an influence on the dispersion calculations. The meteorological data, e.g. the rainfall that affects the river flow pattern, have a significant impact on public exposure levels in the case of accidental and routine discharges. During the calculation of the allowable discharge limits from the Necsa site, the nearby river flow pattern associated with the lowest rain fall over a period of five years was used to calculate the most conservative dilution factor.

Nuclear Facilities such as the Necsa Radioactive Effluent Treatment Facility conduct a wide variety of activities. The activities of the Radioactive Effluent Treatment Facility range from simple reception of waste and analysis prior to discharge to the operation of a complex evaporator system. Not all these activities will significantly contribute to a nuclear event. It is necessary to avoid placing excessive requirements on simple or even trivial operations. The facility must be divided into segments, defining the segments according to the hazards present in each segment. If an event in one segment has no significant consequences and will not initiate an event in another segment it should not be included in the overall risk assessment. The concept of independent Nuclear Facility segments is applied only when it is certain that there is no harmful interaction between the different segments during an event. If a common initiating event can lead to the harmful interaction between the different segments the risk assessment should include the evaluation of all segments and the interaction between the segments.

In the case of the Radioactive Effluent Treatment Facility at Necsa an analysis or risk assessment was performed on the Medium Activity Effluent process because it was identified as a segment that could give rise to a nuclear event with significant consequences but the consequences would not have harmful effects on the Low Activity Effluent process or segment.

The Safety Assessment comprises of a dose assessment for the normal operation of the facility as well as the dose assessment for developed accident scenarios. The process steps for a Safety Assessment are shown in Fig 3.1.

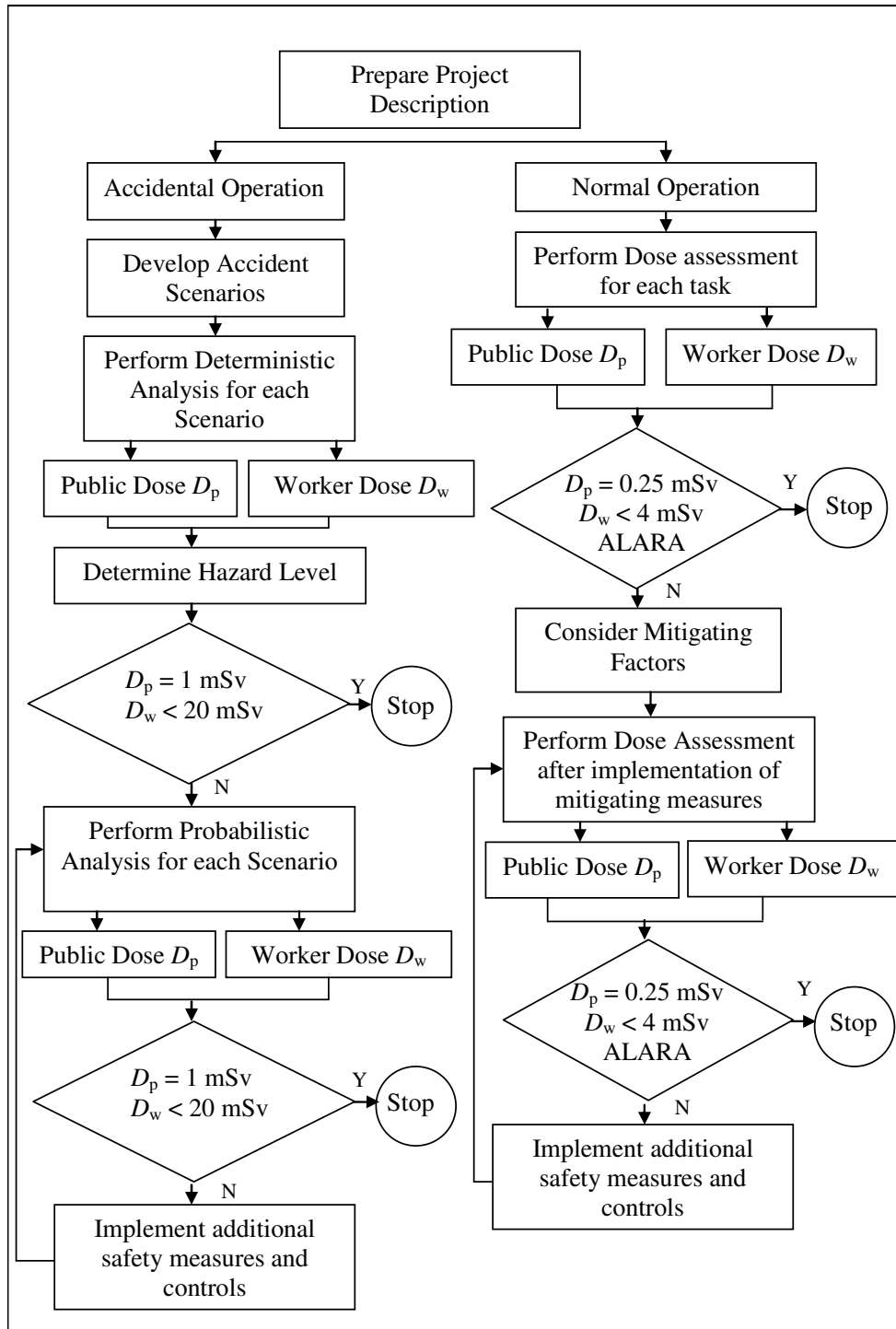


Figure 3.1: Flow diagram of process steps for a Safety Assessment.

The flow logic of a Safety Assessment can be summarised as follows:

- **Facility Description:** a detail description of the facility to be assessed must be compiled. The boundaries of the Safety Assessment must be determined as described later in this Chapter in section 3.3.1
- **Plant conditions:** distinguish between normal operations and accidents.
- **Normal Conditions:** define the various process actions (Compile a work breakdown).
 - Perform a Dose Assessment for the different actions and determine the total dose to the Public and to a worker from all the actions. If the dose to the public is less than $250 \mu\text{Svy}^{-1}$ and is less than 4mSvy^{-1} for the worker the facility can be operated. No additional dose assessment is required. If the dose to the Public is more than $150 \mu\text{Svy}^{-1}$ and more than 4mSvy^{-1} for the worker the facility cannot be operated. The dose assessment must be repeated and mitigating factors (e.g. Engineering controls and Administrative controls) must be implemented. This is an iterative process and must be repeated until the dose is acceptable. The implementation of engineering controls is preferable above administrative controls.
- **Develop Accident Scenarios:** realistic accident scenarios should be developed to determine the magnitude of the accident.
 - Perform a Determinist Analysis for the various accident scenarios and determine the dose to the Public and the workers. If the dose to the public is less than 1 mSv for the event and less than 20 mSv for the worker the facility can be graded as a **Hazard Level 0** facility and no additional analysis is required. If the dose to the public is more than 1 mSv and more than 20 mSv for the worker the facility is graded as a **Hazard Level 1-3** depending on the results of the analysis. A detailed probabilistic analysis should be performed and if required additional controls should be implemented should the outcome of the probabilistic analysis be not satisfactory

3.2 Hazard level determination

The hazard level of a facility can be determined by developing a scenario of the serious accident or event that could be possible in a facility with the highest or most serious consequences. A loss of inventory was assumed for the Radioactive Effluent Treatment Facility at Necsa. The event can be categorised to determine the seriousness of such an event. An international event rating scale has been developed based on past events experienced in the nuclear industry globally.

The International Nuclear and Radiological Event Scale (INES) [30], as adopted by the IAEA is a worldwide tool for communicating to the public in a consistent way the safety significance of a nuclear event. The INES explains the significance of events from a range of activities, including industrial and medical use of radiation sources, operations at nuclear facilities and transport of radioactive materials.

The events are classified on the seven point scale as shown in Fig. 3.2. Levels 1-3 are called *incidents*; Levels 4-7 are called *accidents*. The scale is designed so that the severity of an event is about ten times greater for each increase in level of the scale. Events without safety significance are called *deviations* and are classified below scale or Level 0. Table 3.1 and Table 3.2 give a summary of typical nuclear events in each category [30].

INES classifies a nuclear event at nuclear facilities as an incident or accidents by considering the three areas of impact as shown in Table 3.1:

- **People and the Environment** covers the radiation doses to people close to the location of the event and the widespread, unplanned release of radioactive material from an installation.
- **Radiological Barriers and Control** covers events without any direct impact on people or the environment and only applies inside major facilities. It covers unplanned high radiation levels and spread of significant quantities of radioactive materials confined within the installation.

- **Defence-in-Depth** also covers events without any direct impact on people or the environment, but for which the range of measures put in place to prevent accidents that did not function as intended.

INES classifies nuclear events that involve radiation sources and transport by considering only two of the areas of impact (**People and the Environment** and **Defence-in-Depth**) as shown in Table 3.2.

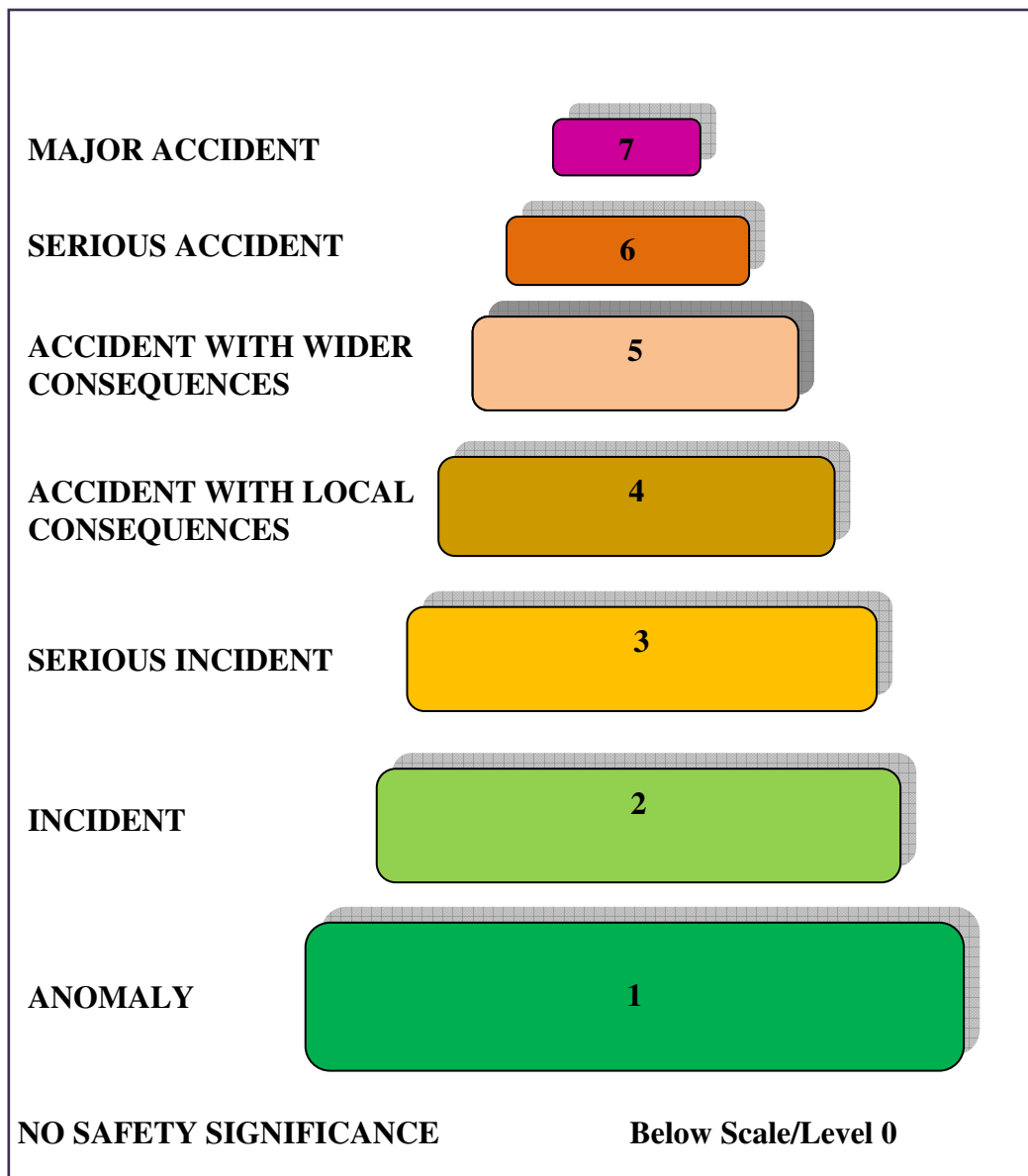


Figure 3.2: The International Nuclear and Radiological Event Scale.

Table 3.1: Examples of Events at Nuclear Facilities.

INES Scale	People and Environment	Radiological Barriers and Control	Defence-in-Depth
7	Chernobyl, 1986 – Widespread health and environmental effect External release of a significant fraction of reactor core inventory.		
6	Kyshlym, Russia, 1957 – Significant release of radioactive material to the environment from explosion or a high activity waste tank.		
5	Windscale Pile, UK, 1957 – Release of radioactive material to the environment following a fire in a reactor core.	Three Mile Island, USA, 1979 – Severe damage to the reactor core.	
4	Tokaimura, Japan, 1999 – Fatal overexposures of workers following a criticality event at a nuclear facility.	Saint Laurent des Eaux, France, 1980 – Melting of one channel of fuel in the reactor with no release outside the site.	
3	No example available	Sellafield, UK, 2005 – Release of large quantities of radioactive material, contained within the installation.	Vandellors, Spain, 1989 – Near accident caused by fire resulting in loss of safety systems at the nuclear power station.
2	Atucha, Argentina, 2005 – Overexposure of a worker at a power reactor exceeding the annual limit.	Cadarache, France, 1993 – Spread of contamination to an area not expected by design.	Forsmark, Sweden, 2006 – Degraded safety functions for common cause failure in the emergency power supply system at nuclear power plant.
1			Breach of operating limits at a nuclear facility.

Table 3.2: Examples of Events Involving Radiation Sources and Transport.

INES Scale	People and Environment	Defence-in-Depth
7		
6		
5	Goiania Brazil, 1987 – Four people died and six received doses of a few Gy from an abandoned and ruptured highly radioactive ^{137}Cs source.	
4	Fleurus, Belgium, 2006 – Severe health effects for a worker at a commercial irradiation facility as a result of high doses of radiation.	
3	Yanango, Peru, 1999 – Incident with radiography source resulting in severe radiation burns.	Ikitelli, Turkey, 1999 – Loss of a highly radioactive ^{60}Co source.
2	USA, 2005 – Overexposure of a radiographer exceeding the annual limit for radiation workers.	France, 1995 – Failure of access control systems at accelerator facility.
1		Theft of a moisture-density gauge.

8.5.1 3.2.1 Hazard level determination of the Radioactive Effluent Treatment Facility at Necsa

The IAEA event rating was used as a basis to derive various hazard levels for the Radioactive Effluent Treatment Facility. The INES, however, consists out of 7 levels and is not practical to use when performing a Safety Assessment. The 7 levels can, however, be reduced to the 3 or 4 levels as described in this Safety Assessment only to classify the Nuclear Facility with regard to risk. The accidents and events are still classified using the INES. The hazard level of the various segments of the Radioactive Effluent Treatment Facility will be determined by applying the following criteria [37]:

Hazard Level 0 (HL 0): Segments where a nuclear event (incident or accident as defined in this document) is not possible or an unmitigated nuclear event cannot lead to a significant radiological dose that can be classified as a nuclear incident or accident. Limited qualitative hazard analysis is required. The emphasis is on safety management programmes that are required to be in place in accordance with the current status of the installation or project. This includes ALARA.

Hazard Level 1 (HL 1): Segments where an unmitigated nuclear event is estimated to have only localised consequences at the facility itself, e.g. a worker who is present at the process where the event occurs, receives a dose that exceeds 20 mSv but less than 50 mSv; no workers in other facilities on site are affected. There are no off-site consequences and no member of the public is exposed to annual normal operation dose constraint of 0.25 mSv. A hazard analysis is performed that includes likelihood estimates of nuclear events.

Hazard Level 2 (HL 2): Segments where an unmitigated nuclear event is estimated to result in worker exposure above 50 mSv and/or public exposure above 0.25 mSv but less than 1 mSv. Comprehensive but appropriate hazard and accident analyses are performed.

Hazard Level 3 (HL 3): Segments where an unmitigated nuclear event is estimated to result in worker exposure above 50 mSv and/or public exposure above 1 mSv in a year as a result of the event. Comprehensive hazard and accident analyses are required that includes a Probabilistic Risk Assessment.

3.3 Safety Assessment Methodology

Safety Assessments are performed to demonstrate the compliance with safety objectives, principles and criteria as stipulated by national and international standards in order to protect workers and members of the public. This concept has been explained in detail in the previous sections but there are various acceptable Safety Assessment methodologies. Two different approaches i.e. *deterministic approach* (no mitigation taken into account) and *probabilistic approach* will be discussed in this Safety Assessment.

If the calculated dose due to accidental exposure (*following deterministic approach*) is not in compliance with the international and national standards it will be recalculated taking into account the preventative and mitigating measure (*following probabilistic approach*) and these measures will then be implemented to ensure compliance. When taking preventative and mitigating measure into account while doing a probabilistic risk assessment, specific reliability factors are allocated to the different control measures. The different control measures have different reliability factors due to specific failure rates of equipment. These failure rates have been determined over years of operating specific equipment. Due to lack of operating history not all equipment has failure rates. Engineering measures, passive, active and human intervention measures all bear different credibility.

The probabilistic risk assessment approach is carried out through a systematic approach. The two factors that determine the risk of an operation or facility are the probability of a failure occurring and the consequences should the failure occur. The determination or calculation of the probability of failures of a facility is subject to engineering judgment, historical failure data and the number of

protection barriers in place to prevent a failure. Various methods such as event trees and fault trees can be used to determine the probability of failure. The consequence of a failure can usually be determined very accurately by the calculation of the Source Term. The magnitude of the consequences can be changed by the implementation of mitigating engineering or administrative measures.

9.5.1 3.3.1 Determining boundaries of the Safety Assessment

Two different boundaries need to be determined during the performance of a Safety Assessment. Firstly, the boundary of the facility that could be affected or the areas on which the facility could have an effect must be identified. The second boundary of the Safety Assessment that needs to be identified is the boundary of uncertainty that will be subject to judgement. The radius of impact is established by consequences judged to be significant.

Boundary of the facility

It must be determined whether a single, all-encompassing analysis should be made or if the system should be subdivided with a number of standalone analyses performed on the respective sections. Large, single analyses are complex and bulky but include all interactions that can occur among systems. Allowing subdivision into small independent studies reduces the complexity; however, it increases the possibility of excluding system interactions and common-cause effects or failures. The recommended approach is to perform several independent analyses, but ensuring that both output and input of materials and energies that can affect each analysis are properly considered (correct identification of boundary of the Safety Assessment). Material or energy can be released during an accident in a manner that can adversely affect people, equipment, processes, or the environment and if boundaries are unclear some of the effects or affected could be missed. An overall boundary analysis is presented in Fig. 3.3.

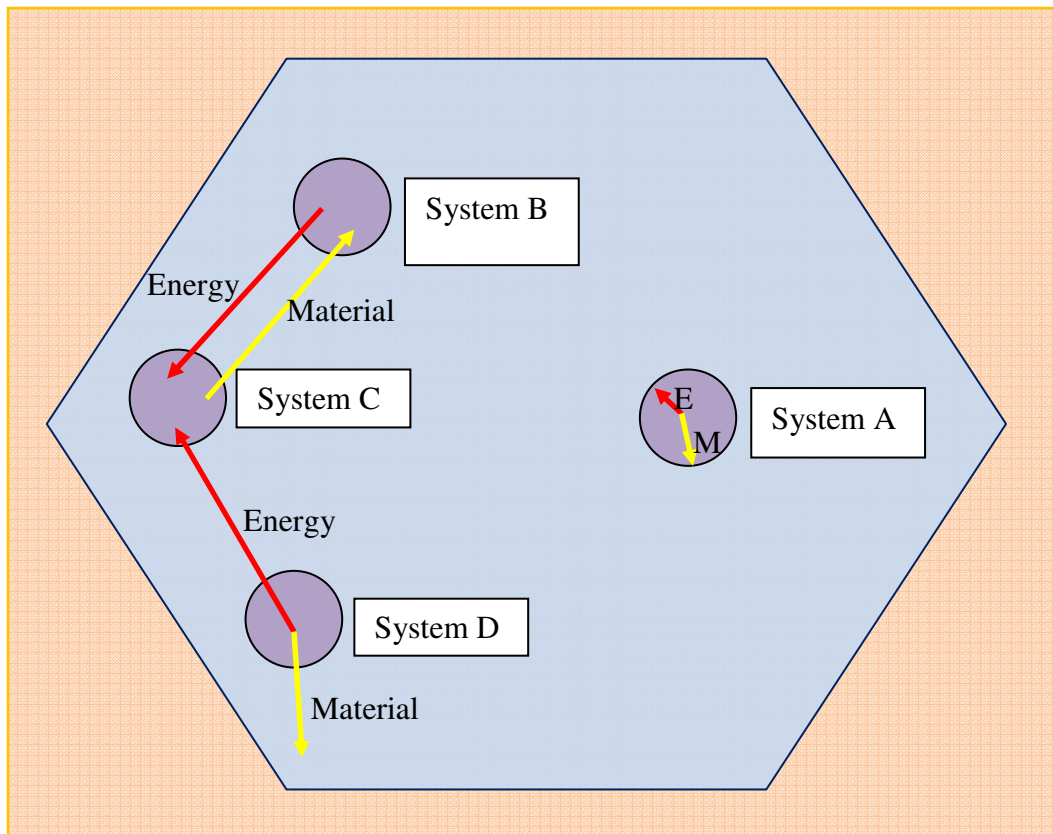


Figure 3.3: Illustration of an Overall Boundary Analysis.

The different systems shown in Fig. 3.3 can be identified as follows:

System A: The energy or material released by an accident does not have an impact beyond the system boundary. No other systems or facilities (outside the System A boundary) will have a direct effect on System A during an accident. The materials released from System A will not impact other systems, but do contribute to the impact on the overall analysis. An independent analysis could be performed on System A with confidence.

System B: The energy released during an accident adversely impacts System C, and the materials released from System C adversely impact System B. An example of such an occurrence is where the energy from a fire in system B is released to System C causing runaway chemical reaction that in return releases toxic material into System B. Both Systems B and C should be included in the

same analyses to avoid omitting possible common-cause effects that the interactions might have.

System C: The material released from System C during an accident adversely impacts System B. An example of such an occurrence is where the material from System C (toxic vapour from uncontrolled chemical release) is entrained in System B's ventilation system causing the spread of toxic vapour into System B. Both Systems B and C should be included in the same analyses to avoid omitting possible common-cause effects that the interactions might have.

System D: The energy released during an accident adversely impacts System C. The materials released do not impact other systems, but do contribute to the impact on the overall analysis. The effects of the energy released from this system define the envelope of the overall analysis. System B is unaffected by the release from System D, however, the energy impact from System D to System C must be considered in the analysis of Systems C and B as a whole.

Boundary of uncertainty

It must be remembered that the judgement of scientists and engineers will always be a key attribute during the performance of any risk assessment process. It is not always possible to demonstrate how far an overall plant design or an individual item will perform in extreme circumstances. Judgement is then required. Judgement is also called upon for possible contingency that was left out during the estimates or calculations. Judgement should always be on the conservative side and should be used to determine additional safety features to be built into the design to cater for the unexpected. The inputs, judgement figures or assumptions used in the risk assessment process are used so that the actual probability of installation failure is generally lower than what emerges from the calculations. This principle is called conservatism in input data and assumptions. Any assumptions made in performing the Safety Assessment should be explicitly documented and examined for reasonableness and relevancy. It is, however, necessary to guard against over conservatism. Quality control over documented

assumptions is easier and it will be easier to recognize any future changes that invalidate the assumptions and thus require modification to the Safety Assessment. If specific initiating events are deemed not to be credible, e.g. below a defined cut-off frequency of 10^{-6} [37] per annum, are left out the assumptions made to justify the decision should also be explicitly documented.

Additional safety features should specifically include greater robustness in design or provision of equipment or aim at redundancy in the design or provision of back-up equipment. These additional safety features or the provision of independent means of protection is the principle known as Defence-in-Depth. Defence-in-Depth will be described in more detail later on.

10.5.1 3.3.2 Deterministic and Probabilistic Approaches

The first step of a deterministic approach is to determine the Source Term of a facility. The Source Term is the amount and isotopic composition of material released (or postulated to be released) from a facility. The Source Term is determined on an unmitigated basis during an accidental release and exposure.

Once the Source Term has been calculated the result must be used to calculate the potential dose to an operator or member of the public. This dose will be an initial indication of the hazard level of the facility. Should this result indicate significantly high levels, a comprehensive Safety Assessment should be performed on the facility to evaluate the probability of such exposure by taking the preventative measures and mitigating measures that are implemented in the facility in the form of engineering controls and administrative controls, into account.

In order to do the initial hazard grading a deterministic approach is followed and the Source Term is calculated and assumptions are made that the total inventory Material at Risk is released or a most conservative assumption is made when allocating a value to the Material at Risk in order to determine a conservative value for the Source Term. Should the calculated Source Term indicate a

consequence or dose impact, as described in the previous paragraphs to turn out to be a Hazard Level 0 no further analysis is required and the basic ALARA and Defence-in-Depth principles will be applied. Should the Hazard Level be higher than a Hazard Level 0, extended analysis should be done in correlation with the magnitude of the outcome of the calculation. Various methodologies can then be applied to identify release scenarios. A probabilistic approach is then followed when the detriment (consequence) of an event is not acceptable and it now requires to demonstrate that the frequency or probability of such an event occurring is within acceptable limits and sufficient mitigation and control measures are in place to manage the risk.

The advantages of both approaches/techniques can be summarised as follows:

- **Main Advantages of a Deterministic Approach**
 - Minimum reliance on probabilities.
 - Use of conservative analysis methodologies.
 - Insistence on Defence-in-Depth (e.g. single failure criterion).
 - Easier to apply than probabilistic risk approach when doing initial hazard grading (less detail of equipment reliability needed).

- **Main Advantages of a Probabilistic Risk Approach**
 - Takes equipment reliability fully into account.
 - Considers much wider scope of events (e.g. multiple failures beyond single failure).
 - Addresses the question of “*how safe is safe enough?*”.
 - Gives a balanced, realistic assessment of safety.
 - Provides deep insight into nuclear safety.

Both processes will be applied during the performance of the Radioactive Effluent Treatment Facility Safety Assessment. This Safety Assessment process will follow a “bottom-up” approach involving identification of sources of radioactivity and potential release/exposure pathways which could pose a hazard. The magnitude of these sources will be calculated following an unmitigated approach.

If the unmitigated consequence is too high the probabilistic approach will be applied. Barriers and means to mitigate events will be identified and if sufficient barriers are not present, additional barriers will be recommended. Conservative, probabilistic analysis and decision-making will be applied accordingly to determine the risk of operating the facility.

While performing a Safety Assessment the following principles must be applied:

- Hazards that could result in major events (serious consequences) should be managed such that the likelihood of occurring is very small. Smaller hazards, with a limited consequence must be more or less inversely proportional to the probability of occurring. This principle focuses more attention on main risk contributors, aiming to balance the risks.
- The likelihood of accidents, challenging the first safety barrier, should be extremely low.
- No single credible accident damages the first safety barrier (beyond a certain level of tolerance) or exceeds any safety goal.
- Consideration must be given to the cumulative consequences of events, including extremely unlikely events.

When performing a probabilistic Safety Assessment the above principles need to be applied and that also requires the categorisation of events as follows [30]:

- **Intended by design:** Anticipated normal operation $>1y^{-1}$ (e.g. small spillages).
- **Possible abnormal conditions:** Upset conditions (practically expected during the lifetime of the facility) $10^{-2} y^{-1}$.
- **Less probable but credible:** Accident conditions (possible to improbable) $10^{-2} - 10^{-6} y^{-1}$.
- **Extremely unlikely/incredible:** Accident conditions (extremely unlikely e.g. criticality) $< 10^{-6} y^{-1}$.

To enable the management of risk to comply with the above categorization of events specific engineering and administrative controls are implemented and are known as Items Relied on for Safety (IROFS).

Administrative controls are generally not considered to be as reliable as engineered controls since human error usually occurs more frequently than equipment failure [31]. Engineered controls may be categorized as either “passive” or “active” controls. A typical passive engineering control is the use of shielded rooms. Active controls for example include equipment such as switches that will terminate an action or stop the operation of a piece of equipment when for instance unauthorised access has been gained to an open source. Active engineering control often used is the purging of equipment with an inert gas such as nitrogen to prevent any explosions or fires by the creation of an inert atmosphere. In general, passive controls are considered to be less prone to failure than active controls. A passive control such as shielding is fixed while a nitrogen supply could fail.

11.5.1 3.3.3 Defence-in-Depth

The concept of Defence-in-Depth, which concerns the protection of both the Public and the workers, is fundamental to the safety of nuclear installations [3, 5]. The concept of Defence-in-Depth, as applied to all safety activities, whether organizational, behavioural or design related, ensures that they are subject to overlapping provisions, so that if a failure were to occur it would be detected and compensated for or corrected by appropriate measures. This idea of multiple levels of protection, or lines of defence, is the central feature of Defence-in-Depth.

The strategy for Defence-in-Depth is twofold: first, to prevent accidents and second if prevention fails to limit (or mitigate) their potential consequences and prevent any evolution to more serious conditions. Accident prevention is, however, the first priority.

It is important to understand the principles of Defence-in-Depth and to understand the levels of defence applied in the facility that are assessed.

Defence-in-Depth is generally structured in five levels:

- Level 1:** The aim of the first level of defence is to prevent deviations from normal operation and to prevent systems failures.
- Level 2:** The aim of the second level of defence is to detect and intercept deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions.
- Level 3:** The aim of this level of defence is to control the accident (should it develop to that stage) within the plant design basis.
- Level 4:** The aim of the fourth level of defence is the control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents.
- Level 5:** The fifth and final level of defence is aimed at mitigation of the radiological consequences of potential releases of radioactive material that may result from accident conditions.

It cannot be emphasised enough that the choice of methodology and processes applied during the performance of a Safety Assessment is difficult and crucial to ensure optimum results. There are a number of Safety Assessment evaluation methods that may be used to analyse process hazards. Safety Assessment techniques focus either on an inductive (bottom-up) or a deductive (top-down) analysis approach. If the system is complex and the boundaries of uncertainty are big it is recommended to apply more than one methodology to decrease the level of uncertainty and gain confidence in the results of the analysis.

During the application of the Inductive approach the possible sequence of failures or deviations during normal operation that must occur to result in an event are identified. Normally, it is recommended to apply the Inductive methods for analysing single-failure events. The Deductive method again is applied to

identifying combinations of equipment failures and human error that can result in an accident (i.e. multi-failure events). In other words, the Inductive approach is useful in identifying a broad range of potential accidents. The Deductive approach, on the other hand, provides a deeper understanding of the mechanism by which a particular accident might occur and help to identify the possible combinations of failures and root causes that could lead to an accident. By identifying the root causes, the Deductive approach can provide assurance that common-mode failures are understood and are properly addressed. For a complex system with a high probability of failure, it is recommended to apply both methods and to compare the results. It is very assuring and useful to combine two types of techniques, using the Inductive approach (e.g. Hazard and Operability study (HAZOP) and Failure Mode and Effective Analysis (FMEA)) to identify the broad range of potential accidents and the Deductive approach (qualitative Fault-Tree) to analyse the most significant of those accidents (or any other that are postulated) in detail.

The various Safety Assessment methodologies are well developed processes and are well documented in IAEA guideline documents as well as engineering design literature studies. The hazard level or risk of a facility will dictate the use of various methodologies instead of just applying one method to identify hazards and evaluate these hazards.

When the hazard level or hazard grading of a facility is initially determined low, the Safety Assessment methodologies applied to evaluate the hazards could be limited to an elementary safety review consisting of hazard identification checklists and a ranking of the identified hazards. A facility initially rated as a Hazard Level 1 dictates the use of more intense methods of risk assessment and renders the compilation of a Preliminary Hazard Analysis by performing a What-If Analysis, HAZOP and FMEA. The following methods:

- Safety Review
- Checklist Analysis
- Relative Ranking

- Preliminary Hazard Analysis
- What-If Analysis

are considered to be particularly useful when a broad identification and overview of hazards are required.

The application of the complete Safety Assessment Methodologies is required to justify the safety of a facility evaluated as a Hazard Level 2 or 3. The methods applied for the evaluation of a Hazard Level 1 facility should be used and in addition the following methods or part thereof should be applied:

- System and component reliability
- Availability and maintainability analyses
- Fault Tree Analysis
- Event Tree Analysis
- Cause-Consequence Analysis
- Human Reliability Analysis
- Performance of a Quantitative Risk Assessment (QRA).

Methods such as the What-If Analysis, HAZOP and FMEA are more suitable for performing detailed analyses of a wide range of hazards to identify potential accident sequences. These methods lay the foundation to enable the performance of more detailed accident analysis as required for the evaluation of facilities with higher hazard classifications. The Fault Tree Analysis, Event Tree Analysis, Cause-Consequence Analysis and Human Reliability Analysis are applied to provide an in-depth analysis of specific accidents that have been identified through the use of the other methods mentioned above. In general, their use requires a higher degree of analyst expertise and increased time and effort.

These methods are all considered “qualitative” methods in the sense that they can provide important insights useful for reducing risk without requiring a quantitative estimation of risk. QRA, which is most often used when the consequences of an accident are severe e.g. nuclear criticality accidents, is a technique that provides quantitative estimates of the risk of accidents. In addition to providing

information useful for prioritising measures for reducing risk, QRA can also be used to demonstrate that the frequency of occurrence of a severe accident is acceptably small when measured against regulatory criteria.

Once the applicable methodologies have been utilised to evaluate a facility the results are specifically used to identify the critical items that are relied on to manage these hazards, the so-called Items Relied On For Safety (IROFS). In addition to the identification of the specific IROFS the Safety Assessment must evaluate whether the system of IROFS in place in a process will make the identified accidents sufficiently unlikely in order to meet the risk criteria as specified in national and international standards and as mentioned in the accident classification in [30].

Administrative controls are generally not considered as very reliable IROFS in comparison with engineered measures since human error usually occurs more frequently than equipment failures. It should, however, be noted that engineering measures, especially active engineered measures, are as reliable as the Quality Plan (QP) that ensure the functioning of these controls. An Operating Technical Specification (OTS) for any facility should be written specifying which IROFS should be functioning in order to allow operation of a facility. The functioning of the IROFS described in the OTS are again maintained and inspected according to a Quality Plan that includes an inspection and maintenance plan. Again it must be emphasised that the Safety Assessment process will identify hazards of a facility and quantify the associated risk of the facility but additional measures should be implemented to ensure that the processes in place to manage the risks are also maintained. The maintenance of these processes is not a function of the Safety Assessment. Credit is taken for these processes during the compilation of the Safety Assessment and if these processes are not maintained the Safety Assessment assumptions become invalid.

In summary, to provide reasonable assurance that a particular accident sequence will not occur, the Safety Assessment should not only identify the controls that

have been implemented, but also reference the specific features of its safety program (i.e. training, quality assurance, inspection, maintenance, calibration and surveillance) that ensure the reliability of those controls.

The Safety Assessment takes credit for a QP or for quality assurance measures that ensure that the IROFS identified are implemented to satisfy the design criteria. The QP will also confirm that personnel are trained and that the equipment and IROFS are maintained and inspected to ensure functionality when called on during an accident. Audits and inspections should be part of the quality plan and should be conducted to determine whether standard operating procedures are being followed and if the inspections and maintenance are executed to ensure that the Safety Assessment does not become invalid.

During the operation of the Radioactive Effluent Treatment Facility the sources of radioactive material are limited to the Medium Activity and Low Activity Effluent received from the various generators. These sources differ from batch to batch of varying quantities. For the analysis process the most conservative quantity and activity were used in determining the possible exposure.

3.4 Analysis and Results of the Safety Assessment of the Necsa Radioactive Effluent Treatment Facility

Health effects have been studied since the invention and application of X-rays by Wilhelm Conrad Röntgen in 1895. The results of these studies and related research [32] are periodically updated and published as the recommendations of the International Commission for Radiological Protection (ICRP) [8].

The Necsa Radioactive Effluent Treatment Facility has a well-established Radiation Protection program [40] and the work areas are surveyed continuously. The work areas are zoned according to the prevailing radiation and contamination levels. Workers do not normally enter the rooms where the Medium Activity Effluent storage tanks are located. These rooms, also referred to as cells are constructed from reinforced concrete as shielding material. Entry to the cells is

controlled and access is only allowed with a Radiation Protection Officer present. Since no entry is required during normal operation, a radiological work permit will be issued to describe the actions required to be performed inside the cell and the personal protective equipment required as well as the additional radiation protection equipment. This is an example of an administrative control measure implemented in the Radioactive Effluent Treatment Facility, to control the risk associated with radiological hazards. The radiation outside the cells is measured to be $1 \mu\text{Sv h}^{-1}$. The contamination level is less than 4 Bq cm^{-2} for β^- emitters. These levels are maintained by the ventilation system. According to the survey history of the facility, the highest surface contamination inside the cells was $20\,000 \text{ kBq m}^{-2}$ ($2\,000 \text{ Bq cm}^{-2}$) after a spillage. The operation of the ventilation system is described in the Safety Assessment and the Operating Technical Specifications was derived from the Safety Assessment. The ventilation of the Radioactive Effluent Treatment Facility is one of the major Items Relied On For Safety (IROFS) that is an active engineering measure and is part of a formal maintenance and inspection programme. Shielding is also an important IROFS in the Radioactive Effluent Treatment Facility and is part of the passive engineering measures.

The equations published in the IAEA Safety Series [6] were used to calculate external and internal dose to a worker during normal and accidental exposure. The conversion factors for ^{137}Cs were used [6]. Exposure to the Public is not applicable for the Medium Activity Facility because no Medium Activity Effluent is discharged into the nearby river and all sources of radiation generated from Medium Activity Effluent are contained within the building.

The following equations were be used to calculate the dose to a worker due to normal operation of the facility. These same equations were used to calculate the dose to a worker due to an accidental exposure.

12.5.1 3.4.1 Dose calculation for internal radiation (Inhalation)
(Equation was adapted for required application) [33]

The general equation for estimating the committed dose by inhalation received by the individuals defined in the operation of the Radioactive Effluent Treatment facility exposure scenario for radionuclide i (^{137}Cs) is:

$$H_{\text{INH},i} = VtDF_{\text{INH},i}(C_d C_{w,i} + C_{s,i}RF), \quad (3.1)$$

where the different terms are defined as follows:

$H_{\text{INH},i}$	The committed effective dose equivalent from one year's intake of radionuclide i by inhalation (Svy^{-1})
V	Ventilation/Breathing rate of the worker (m^3h^{-1}) (taken as $1.2 \text{ m}^3\text{h}^{-1}$ in this analysis)
t	Duration of exposure for the individual (hy^{-1})
$DF_{\text{INH},i}$	Committed effective dose equivalent per unit intake: (SvBq^{-1})
C_d	The concentration of Respirable dust in air (gm^{-3}) (The level depends on the dust generation potential by the operational actions)
$C_{w,i}$	The non-diluted concentration of radionuclide i in material at risk (Bqg^{-1})
$C_{s,i}$	The concentration of radionuclide i present as surface contamination (Bqcm^{-2})
RF	The re-suspension factor for surface activity (m^{-1}).

Input data used in the calculation of dose due to Inhalation are given in Table 3.3.

Table 3.3: Input data used in the calculation of dose due to Inhalation

Value	Legend/Meaning
1.2	Ventilation/Breathing rate of the worker (m^3h^{-1}) (taken as the inhalation rate for light activity in this analysis)
2×10^3	Duration of exposure for the individual (hy^{-1}). This is taken as 8 hour shifts for 250 days per annum working on 2000 hours.
4.8×10^{-9}	Committed effective dose equivalent per unit intake: (SvBq^{-1}) <i>Table II.III: Committed Effective Dose per unit intake via inhalation for ^{137}Cs (SvBq^{-1}) (p 126)[6]</i> <i>It must be noted that ^{137}Cs has a very high conversion factor compared to most of the other nuclides ^{131}I, ^{60}Co etc. that could also be present in the Medium Activity Effluent. Allocating all the activity to ^{137}Cs is very conservative.</i>
1×10^3	The concentration of Respirable dust in air (gm^{-3}) (The level depends on the dust generation potential by the operational actions). This value is taken from the Radioactive Effluent Treatment Facility survey results as real measured values. The surveys are performed according to an approved QP and Radiation Protection Programme.
1	The non-diluted concentration of radionuclide i in material at risk (Bqg^{-1}). The radioactive effluent received from the client is contained throughout the Radioactive Effluent Treatment Facility processes and the workers are not exposed to direct contact with the radioactive effluent.
4×10^4	The concentration of radionuclide i present as surface contamination (Bqm^{-2}) Taken as average real measured values in facility for Beta contamination (4 Bqcm^{-2}).
1×10^{-6}	The re-suspension factor for surface activity (m^{-1}). The radioactive effluent received from the client is contained throughout the Radioactive Effluent Treatment Facility processes and the workers are not exposed to direct contact with the radioactive effluent. During loss of containment the cleaning of the area is done under the control measures of a radiological work permit and respiratory protection will be worn.

Evaluation of the component internal radiation (Inhalation) using Eq. (3.1) yields the following value:

$$H_{\text{INH},i} = 4.72 \times 10^{-7} \text{ Svy}^{-1}.$$

13.5.1 3.4.2 Dose calculation for internal radiation (Ingestion)
(Equation adapted for required application) [33].

The general equation for estimating the committed dose by ingestion received by the individuals defined in the operation of the Radioactive Effluent Treatment facility exposure scenario for radionuclide i (^{137}Cs) is:

$$H_{\text{ING},i} = tDF_{\text{ING},i}(IC_{\text{ING},i} + I_2TF_{\text{ING},i}C_{s,i}), \quad (3.2)$$

where the different terms are defined as follows:

$H_{\text{ING},i}$	The committed effective dose equivalent from one year's intake of radionuclide i by ingestion (Svy^{-1})
t	Duration of exposure for the individual (hy^{-1})
$DF_{\text{ING},i}$	The committed effective dose equivalent for ingestion of 1 Bq of radionuclide i (SvBq^{-1})
I	Rate of secondary ingestion of removable surface contamination (g/h^{-1})
$C_{\text{ING},i}$	The concentration of radionuclide i in the removable dust (Bqg^{-1})
I_2	Rate of secondary ingestion of removable surface contamination (m^2h^{-1})
$TF_{\text{ING},i}$	The transfer fraction for ingestion of surface activity
$C_{s,i}$	Concentration of radionuclide i present as surface contamination (Bqm^{-2}).

Input data used in the calculation of dose due to Ingestion are given in Table 3.4.

Table 3.4: Input data used in the calculations of dose due to Ingestion

Value	Legend/Meaning
2×10^3	Duration of exposure for the individual (hy^{-1})
1.3×10^{-8}	The committed effective dose equivalent for ingestion of 1 Bq of radionuclide i (SvBq^{-1}). <i>Table II.III: Committed Effective Dose per unit intake via ingestion for ^{137}Cs (SvBq^{-1}) (p 126)[6]</i>
4.17×10^{-3}	Rate of secondary ingestion of removable surface contamination (gh^{-1}). This value is for children which by implication would be less for adults. Value is 0.1 gd^{-1} by a mouthing frequency of 10 times a day. Value is taken and divided by 24 to obtain an hourly rate.
1	The concentration of radionuclide i in the removable dust (Bqg^{-1})
1×10^{-4}	Rate of secondary ingestion of removable surface contamination (m^2h^{-1}). <i>Note: Taken from the information used in [6] and [33].</i>
1×10^{-2}	The transfer fraction for ingestion of surface activity. This value is assumed conservatively because the radioactive effluent received from the client is contained throughout the Radioactive Effluent Treatment Facility processes and the workers are not exposed to direct contact with the radioactive effluent.
4×10^4	Concentration of radionuclide i present as surface contamination (Bqm^{-2}) Taken as average measured in facility for Beta contamination (4 Bqcm^{-2})

Evaluation of the component internal radiation (Ingestion) using Eq. (3.2) yields the following value:

$$H_{\text{ING},i} = 4.59 \times 10^{-9} \text{ Svyr}^{-1}.$$

14.5.1 3.4.3 External radiation
(Equation adapted for required application) [33].

The general equation for estimating the effective dose equivalent received by the individuals defined in the operation of the Radioactive Effluent Treatment facility exposure scenario for radionuclide i (^{137}Cs) is:

$$H_{\text{EXT},i,s} = tC_iDF_{\text{EXT},i,s} , \quad (3.3)$$

where the different terms are defined as follows:

- $H_{\text{EXT},i,s}$ The effective dose equivalent from one year's external exposure to radionuclide i (or to photons of energy i) in source category s (Svy^{-1})
- t Duration of exposure h to the individual (hy^{-1})
- C_i The initial concentration of radionuclide i (or to photons of energy i) in the material handled Bqg^{-1}
- $DF_{\text{EXT},i,s}$ The effective dose equivalent from external exposures to radionuclide i (or to photons of energy i) in source category s (Svh^{-1} per Bqg^{-1}).

Input data used in the calculation of dose due to External Radiation are given in Table 3.5.

Table 3.5: Input data used in the calculations of dose due to External Radiation

Value	Legend/Meaning
2 000	Duration of exposure h to the individual (hy^{-1})
1×10^{-6}	Activity A_i (Real measured value in the facility outside the cells already converted to Svh^{-1}) ($2.5 \times 10^{-5} \text{ Svh}^{-1}$ is the average value inside the cells but no worker will spend $2\,000 \text{ hy}^{-1}$ inside a cell. The average value outside the cells was used because this is also conservative seeing that the workers will not be inside the facility for 2000 hy^{-1} but spend most of the time inside the control room.)

Equation (3.3) was adapted because real measured values are available and are used knowing the activity A_i in the facility outside the cells:

$$H_{\text{EXT},i,s} = tA_{i,s} \quad (3.4)$$

Evaluation of the component External Radiation using Eq. (3.4) yields the following value:

$$H_{\text{EXT},i,s} = 2.00 \times 10^{-3} \text{ Sv.y}^{-1}.$$

15.5.1 3.4.4 Total dose to an operator

The dose (D_{TOT}) for the total task must be such that the operator will not exceed the ALARA limit of 4 mSv. Additional dose will be allowed to emergency workers but on a controlled basis, this will not be discussed in this Safety Assessment. Thus D_{TOT} is given by [6]:

$$D_{\text{TOT}} = H_{\text{INH},i} + H_{\text{ING},i} + H_{\text{EXT},i,s} \quad (3.5)$$

where the different terms are defined as follows:

D_{TOT}	Total accumulated dose (Sv.y^{-1})
$H_{\text{INH},i}$	The committed effective dose equivalent from one year's intake of radionuclide i by inhalation (Sv.y^{-1}) (Eq. (3.1))
$H_{\text{ING},i}$	The committed effective dose equivalent from one year's intake of radionuclide i by ingestion (Sv.y^{-1}) (Eq. (3.2))
$H_{\text{EXT},i,s}$	The effective dose equivalent from one year's external exposure to radionuclide i (or to photons of energy i) in source category s (Sv.y^{-1}) (Eq. (3.3) or Eq. (3.4)).

Thus the total dose to operator during normal operations using Eq. (3.5) yields the following value:

$$D_{\text{TOT}} = 4.72 \times 10^{-7} + 4.59 \times 10^{-9} + 2.00 \times 10^{-3}$$

$$D_{\text{TOT}} = 2.00 \pm 0.1 \times 10^{-3} \text{ Sv.y}^{-1}.$$

The External Radiation component is the only significant contributing factor to the annual dose to a worker due to normal operations. The External Radiation

outside the cells is measured during daily surveys with an electronic radiation monitor. The instrumentation used in the Radioactive Effluent Treatment Facility at Necsa is calibrated yearly. An error of not more than 10% is guaranteed by the suppliers and this is confirmed during the calibration of all instruments. The lower the concentration of the specific nuclides present the higher the error in the readings of the instruments will be. The detection efficiency of the instrument increase with increase in concentration of nuclides.

3.5 Operator exposure due to abnormal operations and accidents

Scenario: Operator exposed to internal and external radiation from radioactive effluent material being released due to a broken transfer line.

An assumption is made that the Medium Activity Effluent transfer line ruptures during the transfer of radioactive effluent from the client/generator to the storage tanks inside the Radioactive Effluent Treatment Facility. A release of 8 m³ (8 000 litres) is assumed before the operators manage to close the valves. The Medium Activity Effluent evaporates leaving contamination on the floor that becomes airborne. The operator spends one shift cleaning up the contamination without wearing PPE. He is exposed to internal and external radiation.

The highest activity of Medium Activity Effluent ever recorded at the Radioactive Effluent Treatment Facility was $9.8 \times 10^7 \text{ Bq}\ell^{-1}$. This value will be used during the simulation of the accident scenario. This is a conservative approach by assuming the type of failure of equipment and by using the highest activity of radioactive effluent received at the Radioactive Effluent Treatment Facility.

Tables 3.6, 3.7 and 3.8 list values used for the calculation of internal radiation (Inhalation) Eq. (3.1), internal radiation (Ingestion) Eq. (3.2) and external radiation Eq. (3.3), respectively.

Table 3.6: Input data used in the calculations of dose due to Inhalation from an accidental exposure for Eq. (3.1).

Value	Legend/Meaning
1.2	Ventilation/Breathing rate of the worker (m^3/h) (taken as the inhalation rate for light activity in this analysis)
8	Duration of exposure for the individual (h). This is taken as one 8 hour shift.
4.8×10^{-9}	Committed effective dose equivalent per unit intake: (Sv/Bq)
1×10^{-3}	<p>The concentration of respirable dust in air ($\text{g}\cdot\text{m}^{-3}$) (The level depends on the dust generation potential by the operational actions). The activity involves cleaning of surfaces and the most conservative value of 1×10^{-3} was taken from Table II.2 [33].</p> <p>With mobile radionuclides such as ^{99}Tc and ^{137}Cs, the activity is expected to volatilize and exhaust through the off-gas system.</p>
9.8×10^4	<p>The non-diluted concentration of radionuclide i in material at risk (Bqg^{-1}).</p> <p>Taking activity as $9.8 \times 10^7 \text{ Bql}^{-1}$ from highest measured values in Radioactive Effluent Treatment Facility.</p> <p>Converting Bql^{-1} to Bqm^{-3}: $9.8 \times 10^7 (\text{Bq/l}) = 9.8 \times 10^{10} \text{ Bqm}^{-3}$</p> <p>Converting Bql^{-1} to Bqg^{-1}: Taking density of water as 1000 kgm^{-3} Activity $9.8 \times 10^7 \text{ Bqkg}^{-1} = 9.8 \times 10^4 \text{ Bqg}^{-1}$</p>
2×10^7	<p>The concentration of radionuclide i present as surface contamination (Bqm^{-2}).</p> <p>The value used here is the highest value measured in the past during an event (lost of containment) in the cells.</p> <p>Concentration of radionuclide i present as surface contamination.</p>
1×10^{-5}	<p>The re-suspension factor for surface activity (m^{-1}).</p> <p>The re-suspension factor referenced in [33] is used.</p> <p>When a relatively insoluble contaminant has been deposited on a surface, it may again be re-suspended into the air and thus become a significant contributor to the inhalation and ingestion pathways to persons at some point in time following deposition of the contaminant.</p> <p>The type of actions and the wind characteristics influences the re-suspension factor. The nuclear event is contained in the building and the meteorological conditions will thus not influence the re-suspension factor.</p>

Evaluation of the component internal radiation (Inhalation) caused by exposure as described in the accident scenario using Eq. (3.1) yields the following value:

$$H_{INH,i} = 1.37 \times 10^{-5} \text{ Sv.}$$

Input data used in the calculation of dose due to Ingestion from an accidental exposure are given in Table 3.7.

Table 3.7: Input data used in the calculations of dose due to Ingestion from an accidental exposure for Eq. (3.2).

Value	Legend/Meaning
8	Duration of exposure for the individual (h). This is taken as one 8 hour shift.
1.3×10^{-8}	The committed effective dose equivalent for ingestion of 1 Bq of radionuclide i (SvBq^{-1}).
4.17×10^{-3}	Rate of secondary ingestion of removable surface contamination (gh^{-1}). This value is for children which by implication would be less for adults. Value is 0.1 gd^{-1} by a mouthing frequency of 10 times a day. Value is taken and divided by 24 to obtain an hourly rate.
9.8×10^4	The concentration of radionuclide i in the removable dust (Bqg^{-1})
1×10^{-4}	Rate of secondary ingestion of removable surface contamination (m^2h^{-1}).
0.01	The transfer fraction for ingestion of surface activity was taken from the example in [33], there is no specific data available for this scenario.
2×10^7	Concentration of radionuclide i present as surface contamination (Bqm^{-2}) The value used here is a value measured in the past during an event (lost of containment) in the cells. Concentration of radionuclide i present as surface contamination. $2\ 000 \text{ Bqcm}^{-2}$

Evaluation of the component internal radiation (Ingestion) caused by exposure as described in the accident scenario using Eq. (3.2) yields the following value:

$$H_{ING,i} = 4.46 \times 10^{-5} \text{ Sv.}$$

Input data used in the calculation of dose due to External Radiation from an

accidental exposure are given in Table 3.8.

Table 3.8: Input data used in the calculations of dose due to External Radiation from an accidental exposure for Eq. (3.3).

Value	Legend/Meaning
8	Duration of exposure h to the individual (hy^{-1})
9.8×10^4	The initial concentration of radionuclide i (or to photons of energy i) in the material handled Bqg^{-1}
2.1×10^{-8}	The effective dose equivalent from external exposures to radionuclide i (or to photons of energy i) in source category s (Svh^{-1} per Bqg^{-1}).

Evaluation of the component external radiation caused by exposure as described in the accident scenario using Eq. (3.3) yields the following value:

$$H_{EXT,i,s} = 1.65 \times 10^{-2} \text{ Sv} .$$

Thus the total dose to operator during accidental scenario using Eq. (3.5) is calculated as:

$$D_{TOT} = 1.37 \times 10^{-7} + 4.46 \times 10^{-5} + 1.65 \times 10^{-2} \text{ Sv}$$

$$D_{TOT} = 1.65 \pm 0.1 \times 10^{-2} \text{ Sv}$$

The External Radiation component is the only significant contributing factor to in the case of accidental exposure. The same instruments that are used during normal operation surveys are also used in the case of accidental releases and nuclear occurrences. An error of not more than 10% is guaranteed by the suppliers and this is confirmed during the calibration of the instruments. The detection efficiency of the instrument increase with increase in concentration of nuclides. The readings on the instrumentation used during an unexpected high release of radioactive material will be very accurate.

The total dose to an operator during accidental scenario is well within the Hazard Level 1 objectives as described in Section 3.2.1. The consequences of the accident is localised at the facility. Due to this low value of the total dose no

further hazard analysis is required to determine the probability of such an accident occurring. However if the potential total dose was above 20 mSv for the accidental scenario, the probability of such an accident occurring would have to be determined. Two well known methods used to determine the probability of an accident occurring are Fault Tree Analysis or Event Tree Analysis.

16.5.1 3.5.1 Accident Fault Tree analysis

Fault Tree Analysis attempts to model and analyse failure processes of engineering systems. It is basically composed of logic diagrams, as shown in Fig. 3.4 that display the state of the system and is constructed using graphical design techniques. It can also be defined as another part or technique of reliability engineering and is applied where errors are intolerable (e.g. operation of nuclear reactors). It is used as a design tool to identify potential accidents and to ensure that these potential accidents are engineered out or the probability of occurring is decreased to an acceptable level. It is thus used in safety engineering and during the execution of Safety Assessments.

During the performance of a Fault Tree Analysis an undesired effect/accident is taken as the root (Top Event) of the tree logic. There should always be only one Top Event and all concerns must tree down from the one Top Event. Each situation that could cause that effect is added to the tree as a series of the logic expression. Each effect can be labelled with an actual number if it is available, (in practice values are not normally available due to lack of historical trend and because of the expense of testing). If no values are available the Fault Tree Analysis is only used as a tool to identify critical equipment/items that would require defence in depth mechanisms to protect against failure or to mitigate the consequences of failure.

Figure 3.4 is included to demonstrate the application of the Fault Tree Analysis methodology for the various actions that need to take place to lead to the exposure of an operator to Medium Activity Effluent at Necsa.

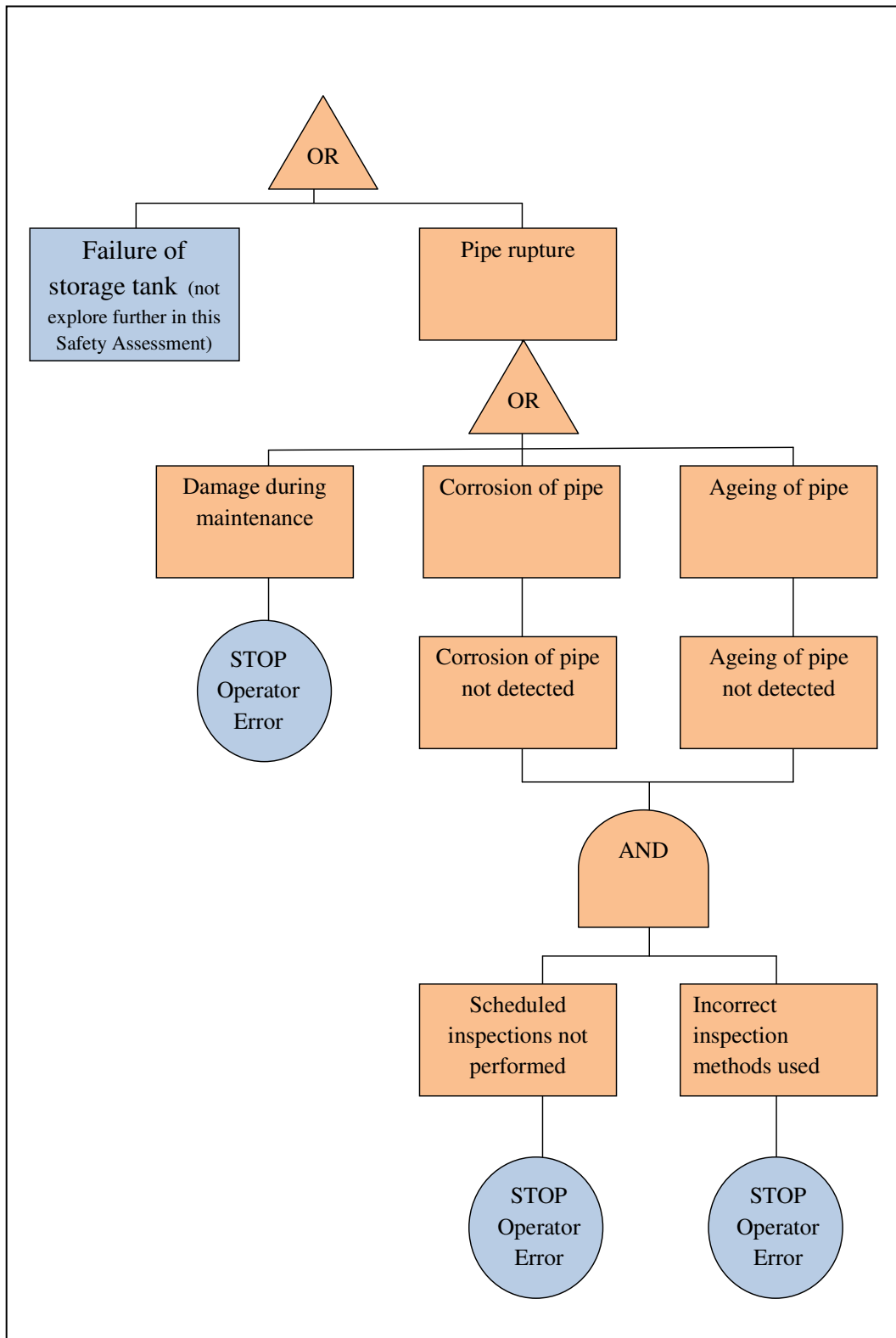


Figure 3.4: Simplified Fault Tree of the Accident Scenario

The Fault Tree Analysis uses conventional logic gate symbols (AND/OR gate symbols). If more than one failure needs to result in an effect, the “AND” gate symbol is used. When only one failure in a series of potential failures will result in an effect, the “OR” gate is used.

The Fault Tree Analysis involves five basic steps:

- define the undesired event to study,
- obtain an understanding of the systems that are to be analysed,
- construct the Fault Tree,
- evaluate the Fault Tree and
- control the hazards (failure modes) identified.

An Event Tree Analysis starts from an undesired initiator (e.g. loss of a critical component) and follows possible further system events through to a series of final consequences. As each new event is considered, a new node on the event tree is added and the probability of each branch of the tree is evaluated. It is not needed to apply both methodologies during the compilation of a Safety Assessment. There are no definite advantages or disadvantages between applying the Fault Tree Analysis or the Event Tree Analysis, it is a matter of personal preference.

CHAPTER 4

DISCUSSION AND CONCLUSIONS

The aim of this Safety Assessment was to demonstrate the safe operation of the Radioactive Effluent Treatment Facility at Necsa and the management of risk associated with the operation of a Radioactive Effluent Treatment Facility.

Table 4.1 and Table 4.2 give a summary of the various Safety Assessment steps followed in this Safety Assessment and the results found are given in Table 4.3. This report covers the methodology of the Safety Assessment process in detail but not all the steps as demonstrated in Fig. 3.1 in Chapter 3 were addressed in detail. Table 4.1 and Table 4.2 discuss the reasoning followed during the execution of the Safety Assessment according to the methodology demonstrated in Fig. 3.1.

As can be seen from Table 4.3, the workers of the Radioactive Effluent Treatment Facility's occupational exposure dose due to normal operation is well within the IAEA standards [6] and the ICRP recommendations on exposure of 50 mSv^{-1} [8], and within the Necsa ALARA objective set at 4 mSv/a [5].

The management of the safe operation of the Radioactive Effluent Treatment Facility at Necsa according to national and international best practise is achieved in the specific areas that were evaluated in this Safety Assessment.

The principles of ALARA for the workers are applied by limiting time spent in classified areas, reducing radiation levels prior to entering an area, monitoring worker exposure etc. The ALARA objective for the workers of Necsa is that the average annual effective dose to the occupationally exposed workforce of Necsa shall not exceed 4 mSv^{-1} . The exposure dose level set by the ICRP is 50 mSv^{-1} [8] and the value set by the NNR is 20 mSv^{-1} [5]. Only as an application of the ALARA principle it is recommended that a project must be registered to investigate the shielding of the Medium Activity Effluent pipes that are currently unshielded.

Table 4.1: Summary of the Safety Assessment process steps followed during the evaluation of normal operating conditions of the Radioactive Effluent Treatment Facility.

Description of Steps of Safety Assessment as illustrated in Fig. 3.1.	Comments
Facility description	
<p>A detail description of the facility to be assessed was compiled and discussed in Chapter 1, Section 1.3. The boundaries of the Safety Assessment were limited to inside the facilities and no influence from or on neighbouring facilities were analysed. The Radioactive Effluent Treatment Facility at Necsa is located downhill from the neighbouring facilities and loss of containment will be contained within the facility.</p>	<p>Facility description should include a description of neighbouring facilities in order to demonstrate that the facility to be analysed will have an influence or not on the neighbouring facilities.</p>
Normal Operation	
<i>Public Exposure</i>	
<p>The public exposure due to possible releases that are part of the normal operations must be determined for each action but due to the nearest member of the public being located more than 1 km from the Radioactive Effluent Treatment Facility at Necsa and all action with regard to liquid effluent is contained within the building, the Public exposure was not determined.</p>	<p>The exposure due to the discharge of radioactive effluent was excluded from this report.</p>
<i>Worker Exposure</i>	
<p>The different actions (work breakdown) for the Radioactive Effluent Treatment Facility at Necsa were not done for this Safety Assessment. It was decided to work conservatively on an average radiation level measured during the operation of the Radioactive Effluent Treatment Facility. This average value included maintenance actions. The average annual dose was calculated for the workers in the Radioactive Effluent Treatment Facility. This annual dose was well within the ALARA objective.</p>	<p>If the Safety Assessment resulted in high average doses, the dose associated with each task would have been calculated. This would then have been used as a guideline to group specific actions together and allow operators only to perform specific combinations of actions in order to limit exposure. Actions that could lead to high doses such as maintenance tasks would then be managed separately to reduce the exposure.</p>

Table 4.2: Summary of the Safety Assessment process steps followed during the evaluation of an accidental scenario occurred at the Radioactive Effluent Treatment Facility.

Description of Steps of Safety Assessment as illustrated in Fig. 3.1.	Comments
Accidental Operation	
Public Exposure	
An accident scenario assumed that there will be no loss of containment from the facility.	The realistic accident scenario resulted in the Radioactive Effluent Treatment Facility at Necsca being classified as a Hazard Level 1 Facility, in other words there would be no off-site consequences.
Worker Exposure	
An accident scenario of the loss of 8 m ³ of Medium Activity Effluent was assumed. Various other scenarios could have been evaluated but this scenario was seen as a realistic scenario to analyse. The outcome of the exposure due to the loss of containment was used to determine the hazard level of the facility. Because the outcome of the possible exposure dose was well within the classification of a Hazard Level 1 Facility no additional assessment was done. The result of the analysis is illustrated in Table 4.3.	A probabilistic analysis of the failure of a pipeline was not done. An example of an fault tree that could lead to operator exposure was included but no failure factors or probability was allocated to the fault tree. The deterministic analysis that followed resulted in low doses and it is thus not necessary to demonstrate that barriers are in place to prevent operator exposure.

Table 4.3: Summary of the Results of the Safety Assessment in demonstration of the Safe Operation of the Radioactive Effluent Treatment Facility

Normal operation		
Dose due to	Svy ⁻¹	Comment
Inhalation	4.72 x 10 ⁻⁷	Dose Limit for normal operation. ALARA objective 4 mSv (4 x 10 ⁻³ Sv).
Ingestion	4.59 x 10 ⁻⁹	
External Radiation	2.00 x 10 ⁻³	
Total	2.00 ± 0.1 x 10 ⁻³	
Accident Exposure		
Dose due to	Sv	Comment
Inhalation	1.37 x 10 ⁻⁵	The dose for a Hazard level 1 is less than 20 mSv (2 x 10 ⁻² Sv).
Ingestion	4.46 x 10 ⁻⁵	
External Radiation	1.65 x 10 ⁻²	
Total	1.65 ± 0.1 x 10 ⁻²	

The exposure of the workers is very low and the annual medical surveillance of the workers of the Radioactive Effluent Treatment Facility at Necsa presented acceptable results [36].

According to the hazard-level rating criteria described in Section 3.2, the Radioactive Effluent Treatment Facility at Necsa is classed as a Hazard Level 1 Facility. Segments where an unmitigated nuclear event is estimated to have only localised consequences at the Facility itself, e.g. a worker who is present at the process where the event occurs, receives a dose that exceeds 20 mSv but less than 50 mSv; no workers in other facilities on site are affected. There are no off-site consequences and no member of the public is exposed to annual normal operation dose constraint of 0.25 mSv. The result of the analysis of the accident scenario is given in Table 4.3 to demonstrate that the Radioactive Effluent Treatment Facility at Necsa is classed as a Hazard Level 1 Facility.

The hazard level rating was derived from the one accident scenario evaluated in this report. The accident scenario is seen as the most conservative scenario but small spillages are more likely to occur at a higher frequency. Despite this conservative assumption of an accidental scenario it was still clear that a catastrophic release of Medium Activity Effluent will have no off-site consequences because it will be contained inside the building and the operator will not encounter a dose exceeding 20 mSv for the single event.

The requirements for occupational exposure are [6]:

The occupational exposure of any worker shall be so controlled that the following limits are not exceeded:

- (a) An effective dose of 20 mSv per year averaged over five consecutive years (*The start of the averaging period shall be coincident with the first day of the relevant annual period after the date of entry into force of the Standards, with no retroactive averaging*).
- (b) An effective dose of 50 mSv in any single year.

According to the process steps of the Safety Assessment as discussed in Chapter 3, Section 3.1 and illustrated in Fig. 3.1 no further analyses were required after the initial results were obtained for the public and worker exposure from normal operating conditions and accidental exposure scenario.

This Safety Assessment confirms that there are sufficient measures (e.g. shielding, containment, personnel monitoring and personal protection etc.) in place to prevent the workers from being exposed above ALARA objectives.

Table 4.3 gives a summary of the Results of the Safety Assessment in demonstration of the Safe operation of the Radioactive Effluent Treatment Facility at Necsa.

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APPENDIX A: ALARA PRINCIPLES

Radiation protection can be divided into three categories:

- occupational radiation protection, which is the protection of workers.
- medical radiation protection, which is the protection of patients.
- public radiation protection, which is the protection of individual members of the public, and of the population as a whole.

The types of exposure, as well as government regulations and legal exposure limits are different for each of these groups. Radiation protection must be considered separately for each group.

Despite the regulatory limits set for each exposure group the aim of any practice must be to lower the doses As Low As Reasonably Achievable (ALARA). ALARA is not a dose limit; it is a practice that has as its objective the attainment of dose levels as far below applicable limits as possible resulting in a small but acceptable level of risk.

ALARA is not simply a phrase, but a work principle, a mindset, a culture of professional excellence. In an ideal world, one could reduce a person's exposure to hazardous materials to zero. In reality, reducing an exposure to zero is not always possible and an approach to control or manage radiation exposures (both individual and collective to the workforce and the public) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit, was taken instead.

The ALARA approach was developed because it is assumed that any exposure to ionizing radiation carries some risk. The risk is assumed to be linear, so as a person's exposure increases, so does the risk of adverse health effects. Instead of operating at or just below permissible exposure limits, one must stay as far below the exposure limits as possible. This affords a wider margin of error should a control fail or malfunction - one's exposure level may rise but still stay below the acceptable limit.

ALARA practices apply to virtually any substance, but are mandated for radiation workers by the National Nuclear Regulator. One can never have "zero" radiation exposure because of the naturally occurring radioactivity that surrounds us - cosmic rays, natural isotopes in our body etc. Therefore, the best that can be done is to add nothing to this background dose, i.e. to keep the exposure ALARA.

In some cases ALARA must be compromised and this is acceptable in the application of medical treatment but not in medical diagnosis.

APPENDIX B: EFFECTS OF RADIATION ON BIOLOGICAL SYSTEMS

Whether the source of radiation is natural or man-made, whether it is a small dose of radiation or a large dose, there will be some biological effects. Most adverse health effects on radiation exposure may be grouped in two general categories:

- Deterministic effects (harmful tissue reactions) due to a large extent to the killing/malfunctioning of cells following high doses; and
- Stochastic effects, i.e. cancer and heritable effects involving either cancer development in exposed individuals owing to mutation of somatic cells or heritable disease in their offspring owing to mutation of reproductive (germ) cells.

Deterministic effects: The induction of tissue reactions is generally characterised by a threshold dose. The reason for the presence of this threshold dose is that radiation damage (serious malfunction or death) of a critical population of cells in a given tissue needs to be sustained before injury is expressed in a clinically relevant form. Above the threshold dose the severity of the injury, including impairment of the capacity for tissue recovery, increases with dose.

Although we tend to think of biological effect in terms of the effect of radiation on living cells, in actuality, ionizing radiation, by definition, interacts only with atoms by a process called ionization. All biological damage effects begin with the consequence of radiation interactions with the atoms forming the cells. As a result, radiation effects on humans proceed from the lowest to the highest level. Summarised: Radiation causes ionization of atoms, which may affect molecules which may affect cells which may affect tissue which may affect organs which may affect the whole body.

Even though all subsequent biological effects can be traced back to the interaction of radiation with atoms there are two mechanisms by which radiation ultimately

affects cells. The two mechanisms are commonly called direct and indirect effects. Cellular damage is illustrated in Figure B1.

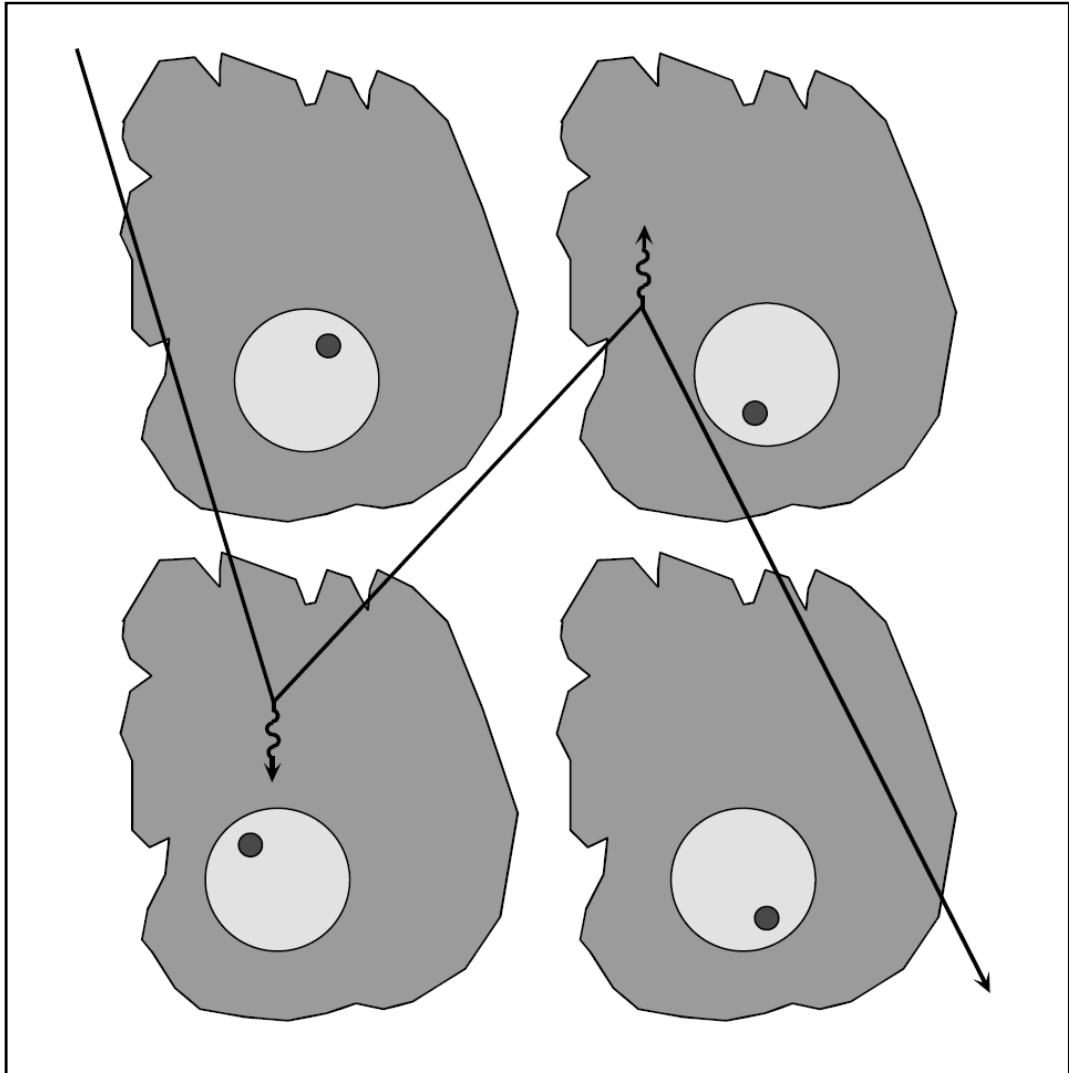


Figure B1: Illustration of Cellular damage due to radiation.

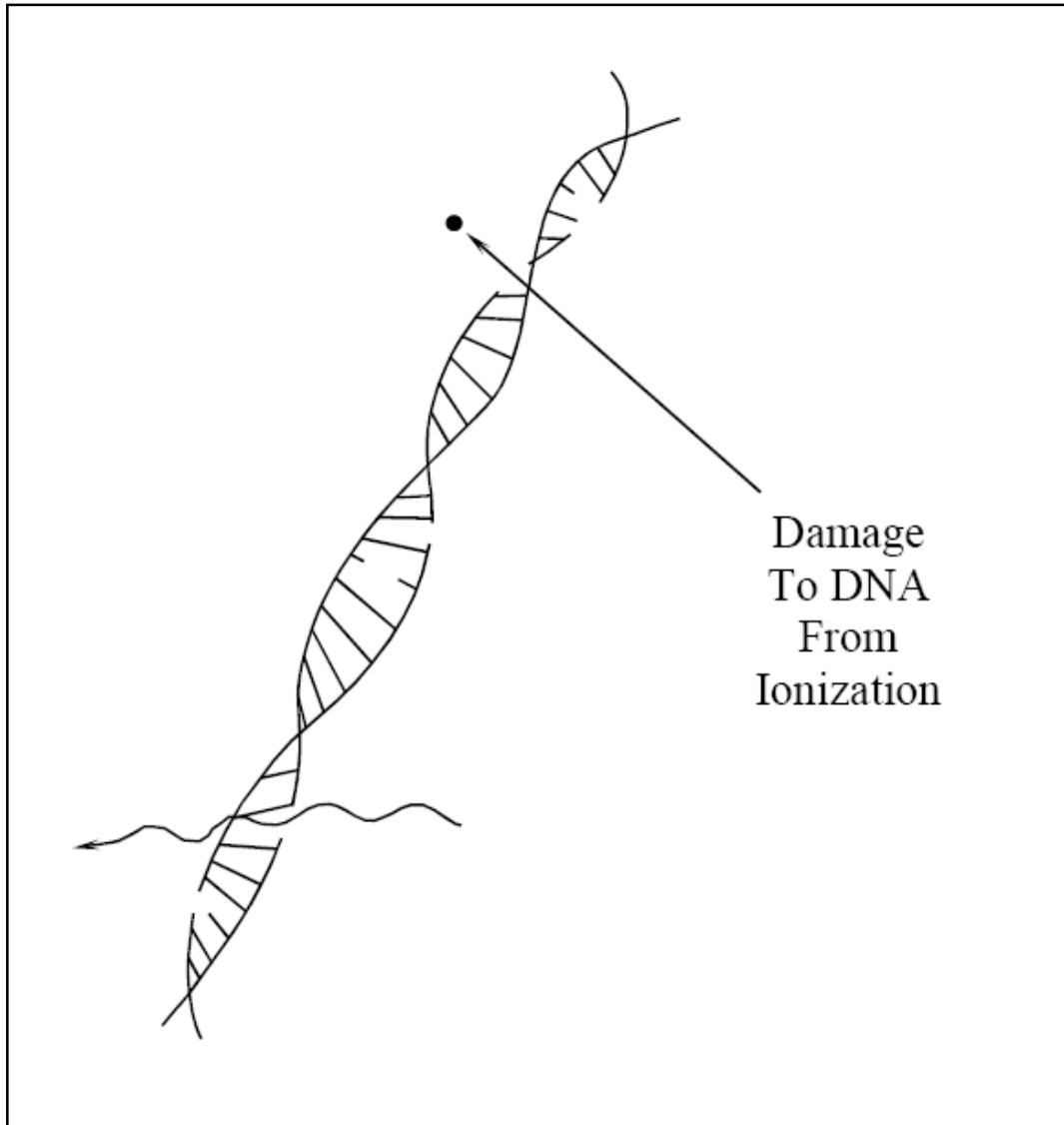


Figure B2: Direct effect of radiation causing damage to DNA in cells.

If radiation interacts with the atoms of the DNA molecule, or some other cellular component critical to the survival of the cell, it is referred to as a direct effect. Such an interaction may affect the ability of the cell to reproduce and thus survive. If enough atoms are affected such that the chromosomes do not replicate properly, or if there is significant alteration in the information carried by the DNA molecule, then the cell may be destroyed by “direct” interference with its life-sustaining system as indicated in Figure B2.

Direct Interference

If a cell is exposed to radiation, the probability of the radiation interacting with the DNA molecule is very small since these critical components make up such a small part of the cell. However, each cell, just as is the case for the human body, is mostly water. Therefore, there is a much higher probability of radiation interacting with the water that makes up most of the cell’s volume.

Indirect Interference

When radiation interacts with water, it may break the bonds that hold the water molecules together, producing fragments such as hydrogen (H) and hydroxyls (OH). These fragments may recombine or may interact with other fragments or ions to form compounds, such as water, which would not harm the cell. However, they could combine to form toxic substances, such as hydrogen peroxide (H₂O₂) which can contribute to the destruction of the cell. The radiolytic decomposition of water in a cell is illustrated in Figure B3

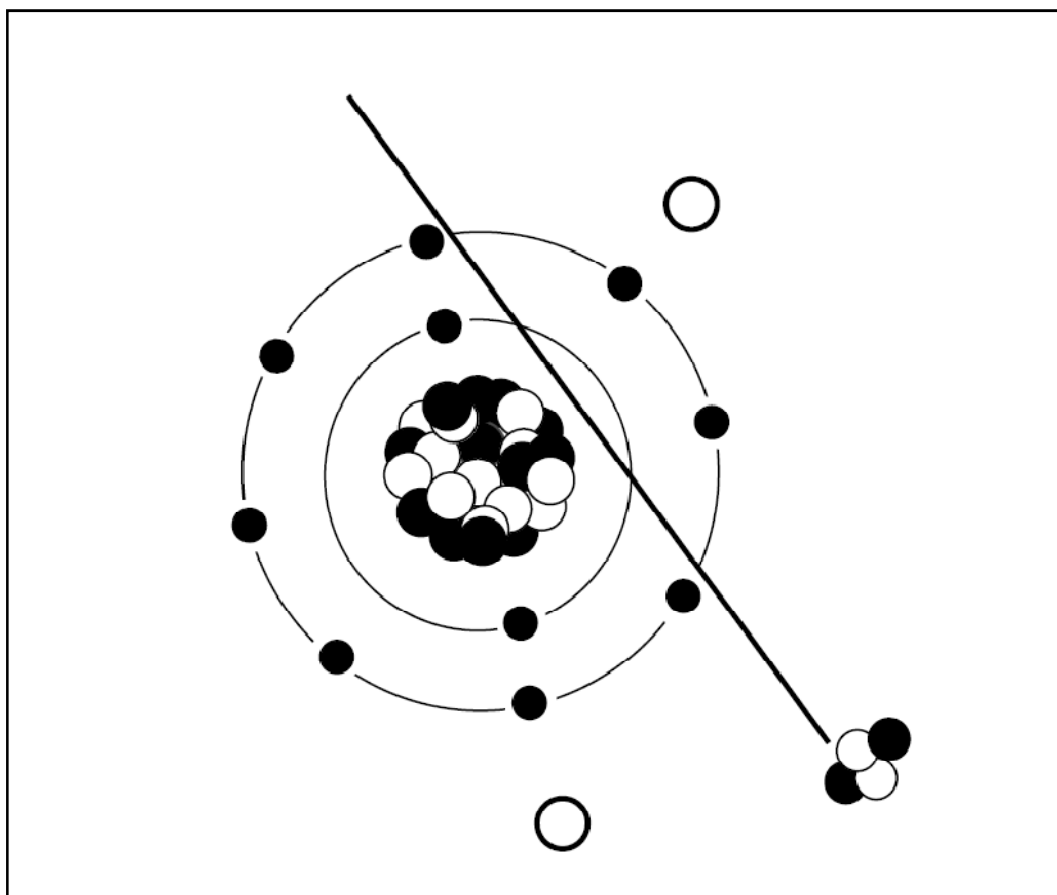


Figure B3: Radiolytic Decomposition of water in a cell.

Not all living cells are equally sensitive to radiation. Those cells which are actively reproducing are more sensitive to radiation. This is due to the dividing cells that require correct DNA information in order for the cell's off springs to survive. A direct interaction of radiation with an active cell could result in the death or mutation of the cell, whereas a direct interaction with the DNA of a dormant cell would have less of an effect.

As a result, living cells can be classified according to their rate of reproduction, which also indicates their relative sensitivity to radiation. This means that different cell systems have different sensitivities. Lymphocytes (white blood cells) and cells which produce blood are constantly regenerating, and are, therefore, the most sensitive. Reproductive and gastrointestinal cells are not regenerating as quickly and are less sensitive. The nerve and muscle cells are the slowest to regenerate and are the least sensitive cells.

Cells like those in the human body have a tremendous ability to repair damage. As a result, not all radiation effects are irreversible. In many instances, the cells are able to completely repair any damage and function normally. The ability to completely repair is illustrated in Figure B4.

If the damage is severe enough, the affected cell dies as illustrated in Figure B5. In some instances, the cell is damaged but is still able to reproduce. The daughter cells, however, may be lacking in some critical life-sustaining component, and they die as illustrated in Figure B6.

The other possible result of radiation exposure is that the cell is affected in such a way that it does not die but is simply mutated. This mutated cell reproduces and thus perpetuates the mutation. This could be the beginning of a malignant tumour as illustrated in Figure B7.

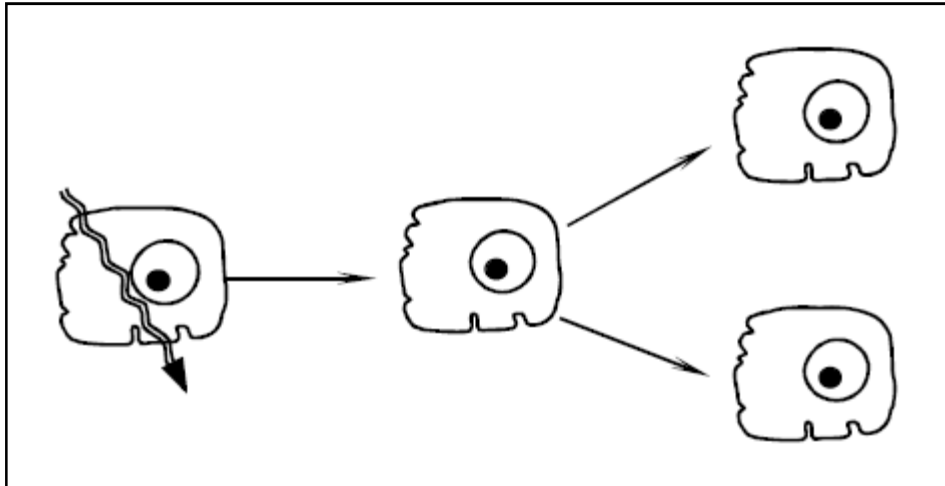


Figure B4: Normal repair of damage.

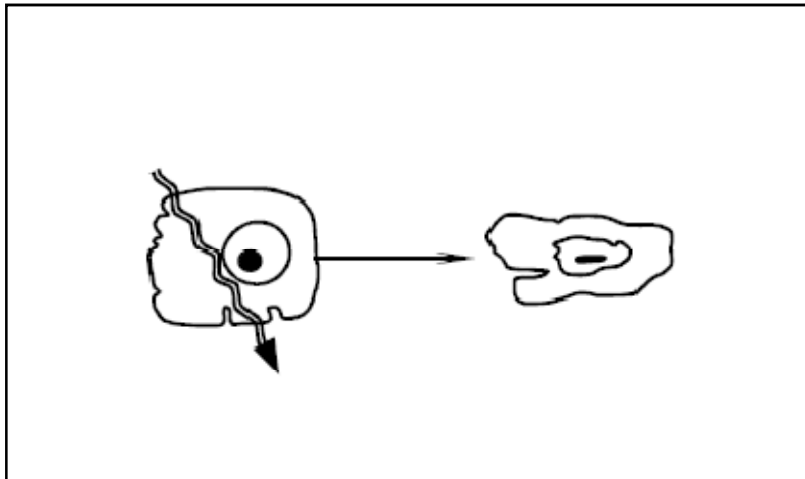


Figure B5: Cell dies from damage.

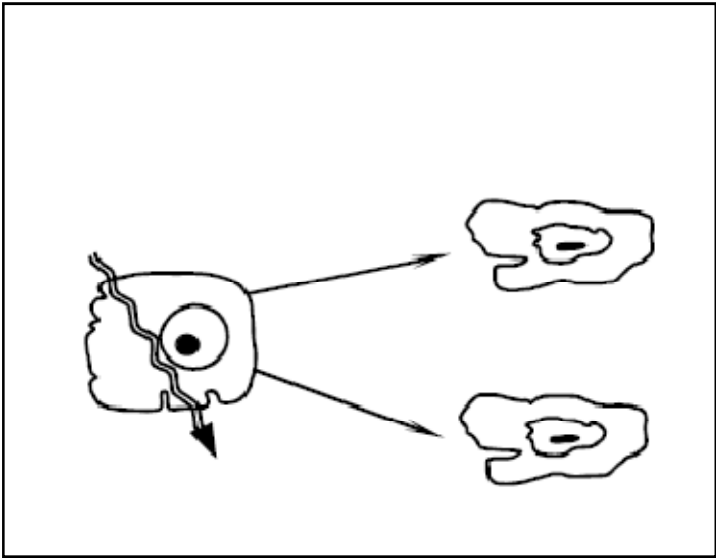


Figure B6: Daughter cells die.

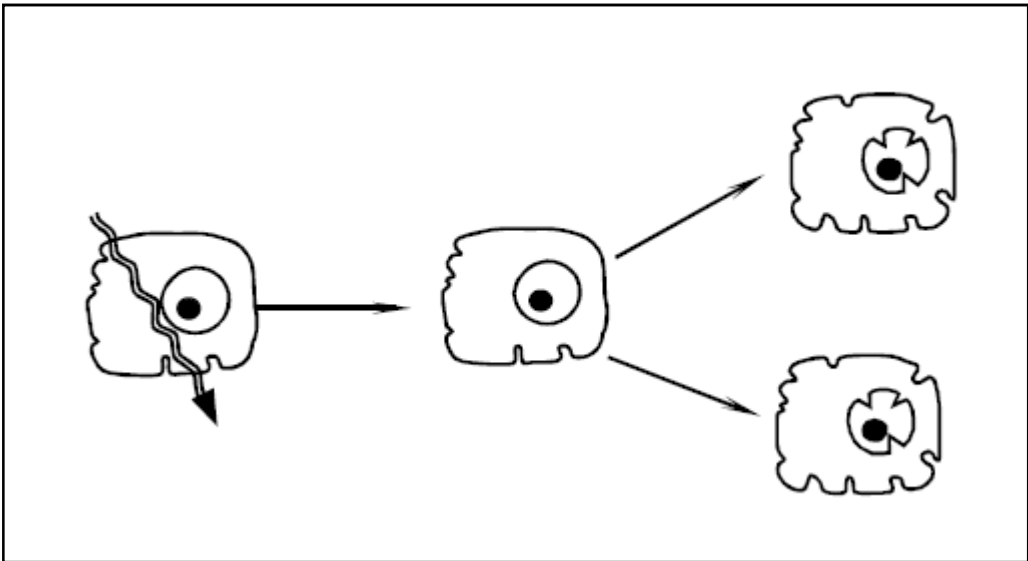


Figure B7: No repair or non-identical repair before reproduction.

The biological effects on the whole body from exposure to radiation will depend upon factors such as:

- total dose,
- type of cell,
- type of radiation,
- age of individual,
- stage of cell division,
- part of body exposed,
- general state of health,
- tissue volume exposed,
- time interval over which dose is received.

Besides death, there are several other possible effects of a high radiation dose such as skin include erythema (reddening like sunburn), dry desquamation (peeling), and moist desquamation (blistering), associated with low energy gamma and X-ray radiation. Hair loss, also called epilation, is similar to skin effects and can occur after acute high doses. Sterility can also occur and depending on the dose it can be temporary or permanent. Cataracts can occur especially with neutron radiation due to the high water content in the eye, which is particularly effective in stopping neutrons.

Low doses spread over a long period of time rather than an acute dose gives rise to non-lethal mutations, with the greatest concern being the induction of cancer.