An assessment model for annual worker radiation dose calculation for the 400MWth PBMR plant

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It all starts here ™

DEDICATED TO

Our heavenly Father, who created me with the ability to perform this study.

My father, Leendert van Ieperen, who believed in my potential and inspired me to develop my gifts and talents.

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ABSTRACT

Increasing demands on energy supplies have renewed interest in high temperature gas-cooled reactors such as the Pebble Bed Modular Reactor. However, public pressure due to nuclear power plant accidents, such as at Chernobyl, has highlighted radiation safety as a primary concern. In the radiation safety assessment and the safety planning of a nuclear facility, worker dose assessments play an important role. For water-cooled nuclear power plants, these assessments are based on operational experience gained and data collected during the design and operation of existing reactor designs.

Most of the research on high temperature gas-cooled reactors was stopped as far back as the 1980s. Information on safety assessments for these reactors is therefore outdated. This forces a new approach, as the 400 MWth Pebble Bed Modular Reactor is a first-of-a-kind development with design parameters set higher than previous designs. Most worker dose assessments are retrospective studies, based on historical or existing information captured in dose and maintenance records.

With the innovative Pebble Bed Modular Reactor, a new integrated dose assessment and integrated dose assessment model were developed, as operational experience and dose measurements are absent. These were developed within the processes and procedures defined for the nuclear industry. It is recognised that when operating experience or radiation monitoring data is not available, dose estimation requires extensive research, compilation of theoretical scenarios and innovative models.

A new simplified dosimetric formula was developed based on the exposure determinants that will contribute to the representative worker's dose. This formula makes it possible for the conceptual exposure scenarios to be quantified in order to arrive at an annual worker's dose. Sensitivity analyses of the key input parameters were conducted to assess their impact on the results. This formula aims to calculate a conservative estimate of the upper limit of annual worker doses received on the plant.

Implementing the integrated assessment model provided the design team with new quantitative and qualitative information. Quantifying the annual worker dose brought an improved understanding of the level of radiation hazard present on the plant. It provided a method to carry out comparative assessments for various combinations of alternative design and maintenance concepts. This is especially important during system optimisation evaluations. It also established a quantitative benchmark for comparison of design improvements.

The results of this study were biased towards upper limit values, due to a lack of safety analysis results for normal operating conditions. Most of the available safety design analyses focused on abnormal plant conditions and accident scenarios during early design phases. It is recognised that as a design matures, more information regarding normal operating plant conditions becomes available. This requires regular updating of the assessment model to ensure that the selected missions are representative of actual exposure scenarios expected.

This integrated assessment and integrated assessment model proved to be a useful engineering tool. It provided the engineers with feedback on the adequacy of the integration of safety considerations early in the design process. The results of the worker dose assessment and the insights gained from the assessment also allow for easier compilation and changes of safety programmes and procedures for the operating plant.

Key words:

Integrated assessment, safety analysis, annual worker radiation dose, high temperature gas-cooled reactors, radiation safety, design optimisation, baseline dose calculation

OPSOMMING

Die toenemende vraag na energie het tot hernude internasionale belangstelling in die ontwerp van hoë temperatuur gasverkoelde reaktors gelei. Die Korrelbed Modulêre Reaktor Kragsentrale is ’n voorbeeld van hierdie tipe reaktors. Stralingsveiligheid in die ontwerp van kernkragaanlegte is van primêre belang. Dit is kernkragongelukke soos die van Chernobyl in 1986 wat tot openbare kritiek van kernkrag gelei het. In die analise van stralingsveiligheid en die veiligheidsbeplanning van ‘n kernaanleg, speel werkerdosisontledings ’n belangrike rol. Vir waterverkoelde kernkragaanlegte is werkerdosisontledings gebaseer op operasionele ervaring en empiriese data wat versamel word tydens die ontwerp en bedryf van bestaande kernaanlegte.

Navorsing op hoë temperatuur gasverkoelde reaktors is reeds so lank terug as die 1980’s gestop. Die bestaande inligting oor stralingsveiligheidaspekte van hierdie tipe reaktors is om hierdie rede verouderd. ’n Nuwe benadering word vereis in die ontwerp van die 400 MWth Korrelbed Modulêre Reaktor Kragsentrale, wat ’n ‘eerste-van-’n-soort-ontwikkeling’ is. Die ontwerpparameters van hierdie aanleg is hoër as die van vorige ontwerpe.

Die meerderheid van beskikbare werkerdosisontledings is terugblikkende studies, gebaseer op historiese inligting van dosis- en onderhoudsrekords. Die ontwerp van die innoverende 400 MWth Korrelbed Modulêre Reaktor verg dat ’n dosisontledingsmetode gevind word, wat in die afwesigheid van operasionele ervaring en dosismetings uitgevoer kan word.

Die doel van hierdie studie is om ’n nuwe geïntegreerde dosisontledingsmetode te ontwikkel om werkerdosisontledings uit te voer tydens die ontwerp van ‘eerste-van-’n-soort’ ontwikkelings, veral waar operasionele ervaring en empiriese data afwesig is. Hierdie is binne die prosesse en prosedures van die kernindustrie ontwikkel. Dit word erken dat dosisramings uitgebreide navorsing, opstel van teoretiese scenario’s en nuwe modelle vereis wanneer operasionele ondervinding en data oor stralingsregulering afwesig is.

‘n Nuwe prosedure, gebaseer op die blootstellingsdeterminante van die dosis van ’n verteenwoordigende werker, is ontwikkel. ’n Vereenvoudigde formule maak dit moontlik om die konseptuele blootstellingscenario’s te kwantifiseer. Hierdie scenario’s is verteenwoordigend van onderhoud en inspeksies op die aanleg wat ’n virtuele werker jaarliks uitvoer.

Sensitiwiteitsanalises is gedoen om die invloed van die belangrikste invoerparameters op die resultate te evalueer. Hierdie vereenvoudigde metodologie poog om ’n konserwatiewe beraming van die boonste limiete van die jaarlikse werkerdosis te bereken.

Implimentering van die metodologie verskaf nuwe kwantitatiewe en kwalitatiewe inligting oor die stralingsvlakke op die aanleg. Die geïntegreerde model maak dit moontlik om vergelykende studies van verskillende kombinasies van alternatiewe ontwerpe en onderhoudskonsepte uit te

voer. Dit is veral belangrik tydens sisteemoptimiseringsevalusies. Dit verskaf ook ‘n kwantitatiewe basislynkriterium vir vergelyking van ontwerpverbeterings.

Die resultate van hierdie studie neig na die boonste verwagte limiet waardes van jaarlikse werkerdosisse, weens ’n gebrek aan beskikbare veiligheidsanaliseresultate vir normale operasionele toestande. Die veiligheidsanalise fokus op abnormale toestande en ongelukscenarios in die vroeë ontwerpfases van ’n aanleg. Soos meer detail tydens die ontwerpproses verkrygbaar word, moet hierdie evalueringsmodel hersien raak. Dit is om te verseker dat die geselekteerde missies verteenwoordigend van realistiese normale blootstellingscenario’s is.

Die geïntegreerde model kan as ’n nuttige ingenieurshulpmiddel gebruik word. Vroeg in die ontwerpproses, verskaf dit ’n terugvoermeganisme aan ingenieurs oor die geskiktheid van die integrasie van veiligheidsvereistes. Die resultate van die werkerdosisontleding, en die insigte verkry uit die ontleding, vergemaklik identifisering van voorstelle en veranderinge vir die radiologiese veiligheidsprogram en -prosedures vir die operasionele aanleg.

Sleutelwoorde:

geïntegreerde dosisontledingsmetode, veiligheidsanalise, werkerdosisontledings, hoë temperatuur gasverkoelde reaktors, stralingsveiligheid, sisteemoptimiseringsevalusies

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NOMENCLATURE

List of symbols

Symbols	Description
α	alpha particle
β	beta particle
γ	gamma particle
Sv	sievert
μSv	microsievert
mSv	millisievert
μSv/h	microsievert per hour
mSv/h	millisievert per hour

List of abbreviations and acronyms

Abbreviation or Acronym	Definition
AGR	Advanced Gas-cooled Reactor
AGS	Auxiliary Gas Subsystem
ALARA	As Low As Reasonably Achievable
AMD	Activity Measurement Device
AMS	Activity Measurement System
AVR	*Arbeidsgemeinschaft Versuchsreaktor*
BUMS	Burn-up Measurement System
CAD	Computer-aided Design
CBA	Conveying Block Assembly
CLIB	Charge Lock Inlet Block
CLOB	Charge Lock Outlet Block
COTS	Commercial Off-the-shelf
CUD	Core Unloading Device
DLOB	Discharge Lock Outlet Block
DPP	Demonstration Power Plant
EA	Exposure Assessment
FHSS	Fuel Handling and Storage System
FSAR	Final Safety Analysis Report
GCR	Gas-cooled Reactor
GCS	Gas Circulating Subsystem
GT-MHR	Gas Turbine Modular High-temperature Reactor
HFE	Human Factors Engineering
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor

Abbreviation or Acronym	Definition
I&C	Instrumentation and Control
IA	Integrated Assessment
IAEA	International Atomic Energy Agency
IAM	Integrated Assessment Model
ICRP	International Commission on Radiological Protection
LEU	Low Enriched Uranium
LRU	Line-replaceable Unit
MBA	Measurement Block Assembly
MCNP	Monte Carlo N-particle Transport Code
MDEP	Multinational Design Evaluation Programme
MME	Modular Maintenance Equipment
MPS	Main Power System
n/a	not applicable
NGNP	Next Generation Nuclear Plant
NIOSH	National Institute for Occupational Safety and Health
NNR	National Nuclear Regulator
No.	Number
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OECD	Organisation for Economic Cooperation and Development
PBMR	Pebble Bed Modular Reactor
PBMR (SOC) Ltd	PBMR State-owned Company Limited
PCU	Power Conversion Unit
PEA	Process Element Assembly
PM	Pebble-bed Module [China]
PWR	Pressurized Water Reactor
QA	Quality Assurance
QC	Quality Control
RP	Radiation Protection
SAR	Safety Analysis Report
SCALE	Standardised Computer Analyses for Licensing Evaluations
SCS	Sphere Circulation Subsystem
SOC	State-owned Company
SP	Shaft Penetration
SRS	Sphere Replenishment Subsystem
SSC	Structures, Systems and Components
SSS	Sphere Storage Subsystem
THTGR	Thorium High-temperature Gas Reactor
US	United States
US NRC	United States Nuclear Regulatory Commission
USA	United States of America
V&V	Verification and Validation

Abbreviation or Acronym	Definition
WADD	Worker Annual Design Dose
WANO	World Association of Nuclear Operators
WHSS	Waste Handling and Storage System

Note on list of references

In the list of references, starting on page 133, a distinction has been made between publicly available documents and PBMR internal documents, which are proprietary information. In the text, the PBMR internal documents are referenced in the format P[1], P[2], etc. where the number in square brackets is the document number in the list of PBMR internal documents on page 137.

DEFINITIONS

Term	Definition
Absorbed dose	The amount of energy deposited by ionising radiation in a unit mass of tissue expressed in joule per kilogram (J/kg), and called 'gray' (Gy) [1].
Activity (radioactivity)	The rate of decay of radioactive material expressed as the number of disintegrations per second. Activity is proportional to the original number of atoms present in the material. Activity is measured in becquerels or curies. A becquerel is one disintegration per second. It does not provide information on the type of radiation emitted during the decay [1].
Analysis	Often used interchangeably with assessment, especially in more specific terms such as 'safety analysis'. In general, however, analysis suggests the process and result of a study aimed at understanding the subject of the analysis, while assessment may also include determinations or judgements of acceptability. Analysis is also often associated with the use of a specific technique. Hence, one or more form of analysis may be used in assessment [2].
Assessment	The process and the result of analysing systematically and evaluating the hazards associated with sources and practices, and associated protection and safety measures. Assessment is often aimed at quantifying performance measures for comparison with criteria. Assessment should be distinguished from analysis. Assessment is aimed at providing information that forms the basis of a decision on whether or not something is satisfactory. Various kinds of analysis may be used as tools in doing this. Hence, an assessment may include a number of analyses [2].
Burn-up	A measure of fuel consumption in a reactor [2].
Cleaning	A term for all of the following processes: dust removal, flow-restricting indexer, dust-pocket cleaning and stopped-sphere dislodgement P[1].
Collective radiation exposure	The amount of radiation received by a group of people. It is calculated by multiplying the average effective dose received by the number of persons exposed [2].
Commercial off-the-shelf (COTS)	All products that are ready-made and are generally available from suppliers. They can be integrated into existing systems without the need for special modification.
Conditioning lines	The continuous circulation of gas through the sphere and gas lines, whereby the temperature cycles on the lines and valves are minimised. The gas system blower is set at a minimum speed and the gas flow in the lines need not be balanced P[1].
Conservative safety analysis (also related to conservative assumptions/data/results)	Analysis requiring adequate margins. This is achieved through analyses using conservative assumptions and input data without the introduction of a final margin. For such analyses, input data that is pessimistic in terms of the analytical results is used with the purpose of arriving at a set of safety analysis results that are demonstrably pessimistic in comparison with any likely result [3].
Cumulative dose	The total dose resulting from repeated or continuous exposure of the same portion of, or the whole body, to ionising radiation [1].
Deterministic effects	Effects that can be related directly to the radiation dose received. A deterministic effect typically has a threshold below which the effect will not occur. Examples of deterministic effects are cataract formation, hair loss, skin burns, nausea, etc. [1].

Term	Definition
Dose assessment (radiation)	The process of determining radiological dose and uncertainty included in the dose estimate through the use of exposure scenarios, bioassay results, monitoring data, source-term information and pathway analysis [4].
Effective dose	A dosimetric quantity useful for comparing the overall health effects of irradiation of the whole body. It takes into account the absorbed doses received by various organs and tissues and weights them according to present knowledge of the sensitivity of each organ to radiation. It accounts for the type of radiation and the potential for each type to inflict biological damage. The unit of effective dose is the Sievert (Sv) [1].
Exclusion areas	Exclusion areas are those radiologically controlled areas where access must be prevented during operation, depending on the operational mode and state of the facility, to avoid uncontrolled and over-exposure [3].
Exposure Assessment (EA)	The systematic collection and analysis of occupational hazards and exposure determinants such as work tasks, magnitude, frequency, variability, duration and route of exposure, and the linkage of the resulting exposure profiles of individuals and similarly exposed groups for the purposes of risk management and health surveillance [4].
Exposure determinant	Factors contributing to the worker dose such as type of work tasks, task frequency, task variability, task duration, magnitude of radiation exposure or dose rate and route of exposure.
Fission	Splitting of a nucleus into at least two other nuclei with the release of a large amount of energy [1].
Gamma rays	High-energy, short wavelength electromagnetic radiation emitted by most radioactive substances [1].
Genetic effects	Hereditary effects (mutations) that can be passed on through reproduction because of changes in sperm or ova [1].
Integrated Assessment (IA)	An assessment is integrated when it brings together and summarises information from diverse fields of study. It integrates knowledge from two or more domains into a single framework [5].
Integrated Assessment Model (IAM)	Integrated assessment modelling is that part of integrated assessments that relies on the use of numerical models. An IAM is a mathematical tool for conducting an integrated assessment. It is a framework to organise and structure various pieces of interdisciplinary knowledge [5], [6].
Ionising radiation	Any radiation capable of displacing or removing electrons from atoms [1].
Occupational dose (radiation)	An individual's dose due to exposure to ionising radiation (external and internal) as a result of that individual's work assignment. Occupational dose does not include planned special exposures, exposure received as a medical patient, background radiation or voluntary participation in medical research programmes [4].
Optimisation of protection (safety)	The process of determining what level of protection and safety makes exposures, and the probability and magnitude of potential exposures, 'as low as reasonably achievable, economic and social factors being taken into account' (As Low As Reasonably Achievable (ALARA)) [2].
Process Element Assembly (PEA)	Line-replaceable Units (LRUs) that perform a specific process function, but have a standardised geometry P[2].
Quantitative dose assessment	The determination of the magnitude, frequency, duration and route of exposure based on collection and quantitative analysis of data sufficient to adequately characterise exposure [5].

Term	Definition
Radionuclide (also referred to as radioisotope or radioactive isotope)	A radionuclide is an atom with an unstable nucleus, which is a nucleus characterised by excess energy. This energy is available to be imparted either to a newly created radiation particle within the nucleus or to an atomic electron; or to be emitted as an electromagnetic wave. The radionuclide undergoes radioactive decay, while emitting the excess energy. The energy emitted is called radiation. Different forms of radiation – alpha and beta particles, gamma rays, or x-rays – can be emitted. Radionuclides may occur naturally, but can also be artificially produced [2].
Safety assessment	This is the systematic process carried out throughout the lifetime of the facility or activity to ensure that all relevant safety requirements are met by the proposed (or actual) design [7].
Somatic effects	Effects of radiation that are limited to the exposed person, as distinguished from genetic effects, and which may also affect subsequent generations [1].
Stochastic effect	An effect that occurs on a random basis independent of the size of dose. The effect typically has no threshold. It is based on probabilities, with the chances of seeing the effect increasing with dose [1].
Stuck sphere	A sphere that stopped moving in the line and cannot be removed pneumatically P[1].

CHAPTER 1: INTRODUCTION AND LITERATURE REVIEW

Chapter 1 gives a brief history of how this research evolved, including an evaluation of previous and current literature.

Chapter 1: Introduction and literature review

1.1 Background

The global need for electricity is on the increase and numerous Nuclear Power Plants (NPPs) are being planned and developed [8]. High Temperature Gas-cooled Reactors (HTGRs) are one of many nuclear designs being investigated. Gas-cooled Reactors (GCRs) have a long history dating back to the very early days of the development of nuclear energy. In South Africa, the Pebble Bed Modular Reactor (PBMR) was developed.

The South African PBMR project is a joint commercial and government venture utilising HTGR technology and a 400 MWth PBMR demonstration plant was designed over the past decade. The commercial project was discontinued in 2010 due to the announcement by the South African government that it would stop funding the development of a demonstration power plant. This resulted in the retrenchment of 600 of its 800 core employees. The project is currently in a state of care and maintenance [9].

Radiation exposure is one of the health risks to which a worker is exposed while working in a nuclear facility. A substantial amount of the radiation to which workers are exposed is due to a lack of attention during design regarding the avoidance or reduction of exposure [8]. International organisations, such as the International Atomic Energy Agency (IAEA), have developed safety standards and guidance documents. These documents assist developers to improve designs in order to improve safety [10], [11].

One of the IAEA's fundamental safety requirements for the design of a nuclear facility is that workers are adequately protected against radiation exposure [3], [10], [11]. Safety assessments are an integral part of nuclear engineering analyses. They include worker dose assessments that evaluate plant radiation safety [3]. The purpose of a worker dose assessment is the radiation health surveillance of the workers on the plant. These assessments provide quantitative results, allowing for comparison with other designs and with dose limits.

Many documents that provide advice on how to perform worker dose assessments in NPPs are available from international organisations and national initiatives [8]. However, this information is largely based on experience and lessons learnt from the existing fleet of NPPs, mostly water-cooled reactors. Available information includes reports on collective radiation exposure received by workers, analyses of dose trends and individual dose distributions.

In the design of a first-of-a-kind nuclear facility, there is an absence of operational experience and measurement data. This results in unique challenges in performing a worker dose assessment during the development of new plant designs. In the absence of radiation monitoring data or exposure records, dose estimation requires extensive research, as well as the compilation of theoretical scenarios and models. The compilation of theoretical scenarios and models are applicable to retrospective and prospective studies where operational data is absent [12]. It was therefore necessary to develop novel and innovative methods to perform this assessment for the PBMR.

One of the purposes of this study was to conduct an Integrated Assessment (IA) to perform a worker dose assessment in the absence of operational experience and measurement data. This proposed IA combines methods used in public dose assessments, dose reconstruction and worker dose assessments. Therefore, the methods and techniques used in these three methods are of particular importance to this study. The IA is based on estimating the dose received by a hypothetical worker, defined as a representative worker.

This proposed IA was tested during the design of the 400 MWth PBMR plant. The IA was performed through the systematic collection and analysis of both radiation physics and engineering design variables or worker exposure determinants. Worker exposure determinants are the type of work tasks or missions; task frequency, variability and duration; magnitude of radiation exposure or dose rate; and route of exposure [13]. This information was used to identify possible exposure scenarios.

The design of a nuclear facility is performed by a multidisciplinary team and is a highly interactive, iterative and continuously evolving process. Specialist fields for this process included design engineers, human factors engineers, physicists and analysts from different fields. Information from these different disciplines had to be managed, analysed and integrated to perform the worker dose assessment during the 400 MWth PBMR plant design. This IA has to collate and summarise information from these diverse fields of study.

Conceptual exposure scenarios were developed based on available plant information from diverse fields of study. The conceptual exposure scenario was developed by integrating information on the plants' major equipment location; maintenance and surveillance task analysis and breakdown to be performed on this equipment; and the dose rates calculated for this equipment. The only dose rate analyses available for this equipment were for conditions when a fuel sphere got stuck and other spheres piled up in the equipment.

It was necessary to develop a dosimetric formula to quantify the annual dose of a worker exposed to these conceptual exposure scenarios. This formula uses the exposure determinants as input parameters.

A number of sensitivity studies were performed by varying the input parameters in the calculation. These studies give safety designers valuable insight into the contribution of different parameters on the annual dose. They also provide information on the expected upper limit values for predicted annual worker dose, by selecting maintenance tasks in high dose rate areas.

It is acknowledged that the precision or quantitative value of the dose estimate is of secondary importance [14]. This study demonstrates that this assessment is a useful nuclear engineering analysis tool to critically evaluate whether safety was adequately considered during plant development. This is because a comprehensive review of the design and safety analysis documentation was necessary to perform this assessment. It can be used for engineering decisions regarding design changes, optimisation purposes and evaluation of engineering processes [14].

The nuclear industry requires that an integrated design approach be followed in all the design phases of an NPP. This is to ensure that the design integrates radiation safety, performance, life cycle support and life cycle costs [15]. The integration of safety in the design is evaluated during a worker dose assessment, thus providing a useful nuclear engineering analysis tool to evaluate whether the integrated design process functions effectively.

The worker dose assessment provides insights to guide decisions on the control of exposures, deficiencies in the safety design of the plant and identified solutions to reduce exposures. Therefore it can be concluded that these assessments should be directly integrated with the nuclear engineering design analysis process and programme management activities involved in plant development. A worker dose assessment should not function as an add-on to the development process.

Furthermore, it is also recognised that this assessment should be performed at several stages during the development of a nuclear facility. This is to ensure that the safety design of the plant evolves and improves as the engineering design progresses. In this way, expensive design changes later in the design could be avoided by attending to the reduction of exposure to radiation.

1.2 Literature review

1.2.1 Introduction

This literature review gives an overview of the relevant and important literature applicable to this research area. It also identifies gaps in this study field. These gaps provide the justification for the work performed in this study. This paragraph discusses some of the most relevant papers used in this research, and their implications.

Firstly, an overview is provided of the historical development and importance of safety assessments in the nuclear industry. The nuclear industry established international bodies to provide requirements and guidelines for performing different types of safety assessments. In this study, the IA developed has to be performed within these requirements and guidelines. This provides credibility to the study and ensures that it makes a useful contribution to the nuclear industry.

The literature survey summarises the different methodologies used in complex systems to perform dose assessments and analyse radiation exposure scenarios. These methods have been applied successfully to other domains as well, including the mining and military weapon industries.

Dose assessments play an important role in the radiological protection programme of a nuclear facility. The aims of these assessments can be summarised as follows [3], [14]:

- Determine the dose received by individuals or groups.
- Estimate the potential health consequences of human exposure to radiation.
- Provide information on the effectiveness of engineering and procedural control measures.
- Demonstrate compliance with regulatory limits.

The majority of dose assessments are performed on operational facilities and form part of a facility's on-going regulatory obligations during the operational phase. In operational facilities, specialised equipment is used to perform extensive monitoring and surveillance to analyse radiation exposure conditions. The measurement data is captured in databases and used to compile dose records for the individuals exposed. Dose assessment methodologies for operational facilities are based on the results reported in these dose records.

The development process of new nuclear reactors is an extended and time-consuming effort. Currently, a worldwide drive towards the standardisation and harmonisation of nuclear reactor designs is advocated [16]. Most of the proposed designs that are marketed, e.g. Westinghouse's AP1000 and the European Pressurized Water Reactor (PWR), are improved designs of existing water-cooled reactors.

An advantage of improved designs of existing reactors is that safety assessments can be based on recorded personnel monitoring and surveillance data. Over the past decades that these plants have been operated, comprehensive databases of measurement data have been collected. This data can be extrapolated and adapted to justify and demonstrate design improvements and safety assessment results. In addition, the process for the Verification and Validation (V&V) of the safety assessment is straightforward and less complicated when available measurement data is used.

A further advantage of using historical data and operational experience in assessments is that system design could incorporate the lessons learnt from experience. It thus ensures that design mistakes are not repeated and that safety designs can be improved [8]. However, little information is available on how to perform worker dose assessments in other phases of engineering development, such as design and plant testing, where operational experience and measurement data are absent.

Another problem is that measurement data for water-cooled reactors is not applicable to GCR designs. Significant differences exist in the radiation environment present on these different types of reactors, due to different design concepts. High Temperature Reactors (HTRs) are generally smaller and produce less power. They are designed to use gas as a coolant rather than water. Enriched uranium is contained in ceramic, billiard ball-sized ‘pebbles’ that use graphite as a moderator, and not in rods as in water-cooled reactors [9].

The literature study also indicated that the available safety assessments performed on early GCR designs are outdated and not applicable to the 400 MWth PBMR design. The 400 MWth PBMR design is really a first-of-a-kind effort, due to the thermodynamic cycle and design parameters that differ from the other HTGR designs [17].

The literature study further emphasised the important contribution that worker dose assessments have as a nuclear engineering analysis tool. New insights into the adequacy of safety considerations in the design are gained when this assessment is performed. For instance, insights are gained into whether adequate shielding has been included; whether maintenance and surveillance times are optimised; whether testing and calibration frequencies are optimised; and whether the need for specialised equipment has been identified in high dose rate areas.

A worker dose assessment is also a useful nuclear engineering tool to evaluate whether the integrated design process is effective. In the design of nuclear facilities, the developer is required to follow an integrated design process, during which the design integrates safety, performance, life cycle support and life cycle costs. It is possible to evaluate whether this process is effective, because design information has to be reviewed, analysed and integrated in the worker dose assessment [15].

In the design of a complex facility where a number of multidisciplinary teams are involved, project management and coordination are a real challenge. For instance, in such a project the design is busy evolving while, simultaneously, safety assessments are performed. It is essential that the engineering design teams and safety analysis teams maintain thorough communication. This is necessary to ensure that safety assessments remain relevant to the design and do not become outdated.

The conclusions and results of worker dose assessments are expected to mature and change as the different iterations of the assessment are performed. However, the Integrated Assessment Model (IAM) described in this study will remain valid regardless of the level of maturity of the input data. The simplified method developed can be applied to the design of new facilities in the nuclear power industry, as well as other domains such as the mining industries and the military weapon industry.

The literature review identified a need to develop an IA and IAM to perform a worker dose assessment during the development of new reactor designs, where measurement data and operational experience are absent. The main purpose of this study is therefore the development and evaluation of such an IA and IAM.

1.2.2 Safety assessments

Historical background

In 1939, the chemists Hahn and Strassman reported that they had successfully bombarded and split the uranium atom, and in so doing created a nuclear fission reaction. On 20 December 1951, nuclear heat released from nuclear fission reactions was transformed into electrical energy for the first time. This was achieved in a small experimental breeder reactor, EBR-1, in Idaho in the United States of America (USA). By the early 1960s, demonstration power reactors were in operation in all the leading industrial countries [18].

In 1945, the nuclear bomb attacks on Hiroshima and Nagasaki had catastrophic health effects on the local populations, due to excessive radiation exposure. The international community then realised that the use of nuclear material has to be controlled in order to prevent similar nuclear disasters in the future. This led to the establishment of a number of international organisations concerned with nuclear safety and radiological protection [18].

The IAEA is of specific importance to this study, because of its contributions on requirements, standards and guidelines on the following [18]:

- Radiation Protection (RP) of workers, including development of techniques for the assessment of occupational exposure;
- RP techniques for the assessment of exposure of the public; and
- safety assessment methods and techniques for NPPs.

In 1958, the IAEA began collecting information on plant safety and regulations from its member states and from other international bodies [18]. This provided the Agency with the necessary background information to draw up its own international recommendations. The IAEA also carried out a limited number of safety inspections on operational NPPs in the early 1960s [18].

For most of the 1960s, the IAEA’s work on safety standards consisted of the drawing up of international recommendations, guides and standards. In this manner, the IAEA was laying the basis for national regulations and legislation, and the development of internationally acceptable safety standards. This work was carried out mainly at the IAEA headquarters in Vienna, Austria [18].

The 1970s witnessed the growing preference in most countries for light-water nuclear power reactors [18]. In 1974, the IAEA launched its Nuclear Safety Standards Programme.

A comprehensive series of codes and safety guides intended to ensure the safe design, siting and operation of the then current generation of nuclear power reactors, was developed. The development of these documents was mainly influenced by the experience gained from the light-water nuclear power reactors [7].

Revision of these documents began at the end of the 1980s. The purpose was to include new developments on both the technological and philosophical levels of safety assessments. A complete revised set of safety standards including safety fundamentals, requirements and guides was available in early 2000 [7]. The process of improving and updating the IAEA safety standards has been ongoing [7].

The IAM developed in this study has to be performed within safety requirements and guidelines framework of the nuclear industry. This provides credibility to the study and ensures that it is a useful contribution to the industry.

Requirements and guidelines – safety assessments

The IAEA requires in its safety standards that comprehensive safety analyses are carried out during the development of an NPP. This is to determine whether an adequate level of safety has been achieved in the design and whether the safety requirements of the facility have been fulfilled [10]. Worker dose assessments form an integral part of these assessments.

Worker dose assessments require the evaluation of radiation doses that workers could receive. The IAEA also recommends that safety assessments should be performed at various stages during the design process of an NPP [10]. This is necessary because it recognises that the design matures and information evolves as the design progresses, and safety assessment results become outdated.

The IAEA has published several standards and guidance documents to assist the nuclear industry in performing these assessments [10], [19], [20] and [21]. However, it should be recognised that for historical reasons, the safety basis for nuclear reactors is primarily tailored to water-cooled reactors [20], [22], [23]. The IAM proposed in this study has to be developed within the framework of these standards.

Recently, Chinese researchers identified the lack of enough standards, codes and guides directly applicable to GCRs, as one of the many challenges in the development of its HTGR-10 test reactor [22]. Similarly, several other developers mentioned that an urgent need exists to establish a new licensing and safety analysis framework that is applicable to the advanced reactors, such as the Generation IV GCRs [23], [24].

Requirements and guidelines – national requirements

In South Africa, the National Nuclear Regulator (NNR)[1] developed a set of regulatory documents for the PBMR, called requirement documents. In these requirement documents it is stipulated that a comprehensive safety justification, contained in a Safety Analysis Report (SAR), must be compiled [3]. Several requirements applicable to worker dose assessments are included [3], and this study had to consider and incorporate these requirements.

Most of the high-level requirements prescribed by the NNR are derived from the IAEA series of safety standards. The interpretation and implementation of these high-level requirements on the design of an HTGR were a challenge to the PBMR safety assessment groups. Methodologies used for water-cooled reactors had to be adapted or new methodologies had to be developed, in order to make assessments possible.

Requirements and guidelines – regulatory dose limit

Regulatory radiation dose limits for workers at nuclear facilities are set to restrict exposure to acceptable levels. The NNR has set a design dose limit of 20 mSv for the PBMR as the highest cumulative dose that a worker may receive during any year [3]. This dose limit is derived from the linear no-threshold model used internationally by regulators to set dose limits. In this study, the results obtained by implementing the new simplified IAM have to demonstrate that this worker dose limit is met for the PBMR design analysed.

The linear no-threshold model is a risk model, which conservatively assumes that there is a direct relationship between radiation exposure and cancer rates. Reports by the International Commission on Radiological Protection (ICRP) stated that the linear no-threshold model provides the best overall fit for RP purposes. Radiation doses at or below these limits are considered ‘safe’ in that there is no direct medical or scientific evidence to show that they cause harm [25].

Requirements and guidelines – multinational agreements

The United States Nuclear Regulatory Commission (US NRC) initiated the Multinational Design Evaluation Programme (MDEP). This programme facilitates cooperation amongst nuclear regulators responsible for reviewing designs of Next Generation Nuclear Plants (NGNPs), intended for construction worldwide. This initiative is a result of the worldwide drive towards the standardisation and harmonisation of nuclear reactor designs and regulation [24].

[1] Regulating nuclear and radiation safety remains a national responsibility. One of the basic purposes of the South African National Nuclear Regulatory Act (Act No. 47 of 1999) (NNR Act) and its associated regulatory documents is to protect the health and safety of the employees of the licensees conducting operations under these regulations [3]. The PBMR is the first nuclear plant to be designed in South Africa and licensed by the NNR.

It was therefore important to consider the US NRC's requirements related to the licensing of NGNPs. In addition, the US legislation is also of specific importance in this assessment. The US NRC provides useful guidance to perform worker dose assessments and recommends what information should be included in such an assessment [14].

These NRC guides require, for instance, that the applicant should describe how the RP design was improved by using experience from past designs and operating plants. They emphasise that measurement data collected from previous plant designs should be used to improve and optimise RP [8], [14]. Therefore, it is clear that developers of NPPs prefer to base worker dose assessments on available dose information.

The available examples of recent worker dose assessments in literature are largely performed for improved water-cooled reactors. In paragraph 1.2.4, several examples of the content of the available assessments obtained through the literature review, are discussed. These examples demonstrate the emphasis on the use of measurement data and experience obtained from operating plants to perform worker dose assessments. Use of measurement data ensures that the results of the safety assessment can be justified based on operating experience.

1.2.3 Dose assessment methods

Public dose assessment

During routine operations at nuclear facilities, limited amounts of radioactive materials are released in the environment through atmospheric and/or liquid pathways. These releases potentially result in a radiation dose commitment to people off site. The principal exposure types through which people are exposed to releases of radioactivity are [26]:

- Inhalation
- Ingestion
- Skin absorption
- External exposure

Public dose assessments are used to assess radiation doses to individuals off-site from nuclear facilities, due to the releases of radioactive material from the site. Dose to the public cannot be measured directly without considerable difficulty and costs. The methods used to perform such an assessment are based on models and calculations, or measurement data from environmental surveillance programmes, but usually on a mix of both [26].

Figure 1 is a diagrammatic representation of the possible pathways of exposure to human receptors, due to the release of radioactive material from a nuclear facility. The figure explains the route of exposure by distinguishing between the source, mode of release, collector, accumulator, pathways and exposure types. This figure is adapted from [26].

In Figure 1, the nuclear facility is the source for the release of radioactive material. The mode of release to the environment can be through atmospheric discharges or aqueous releases. The collector is the environmental media in which the radionuclides are deposited. The pathways are the manner in which the radionuclides present in the accumulators will reach the human receptor.

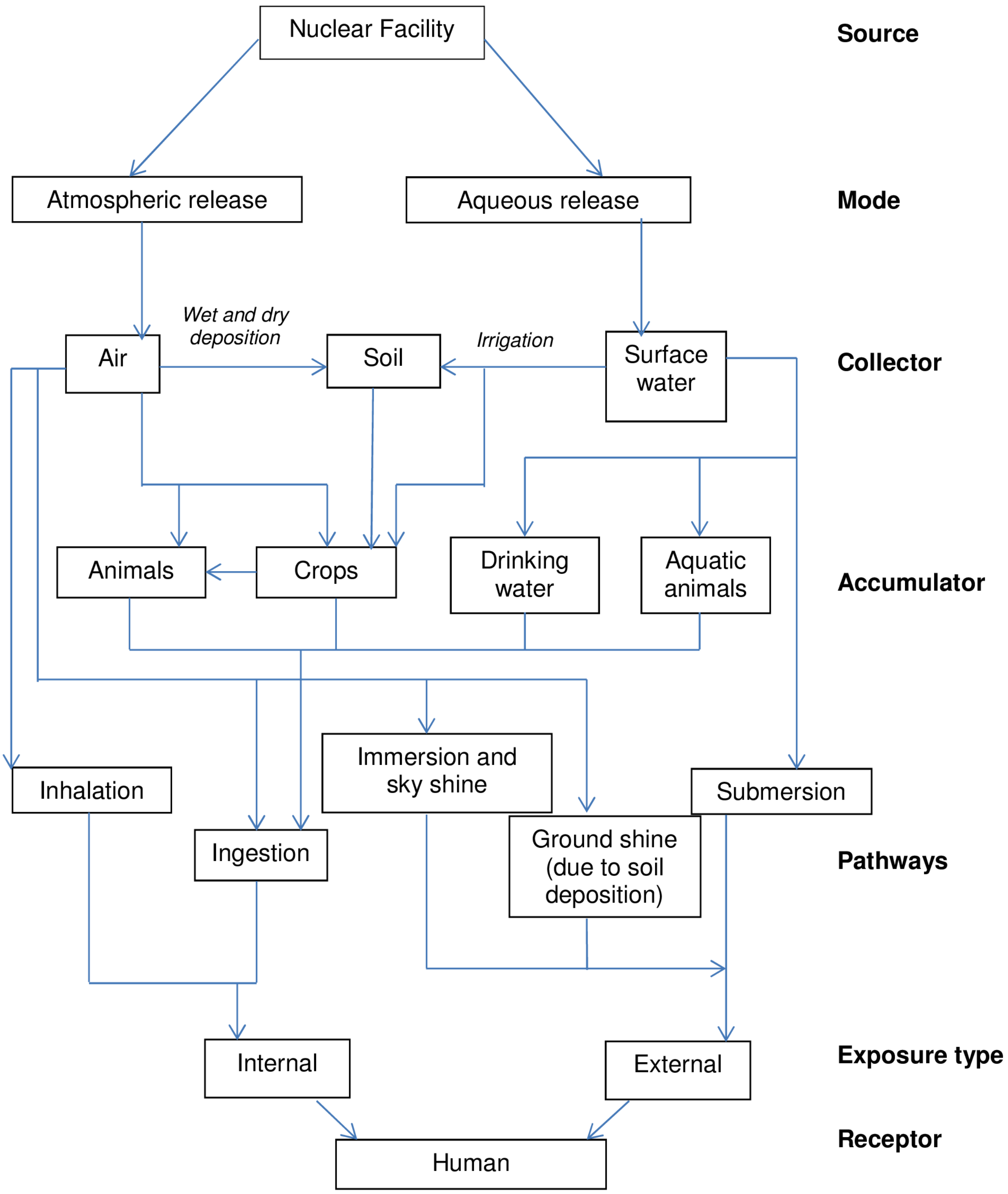

Figure 1: Exposure pathways due to release of radioactive material

The purpose of the public dose assessment is for planning, optimisation or compliance evaluation. Planning and optimisation will require evaluation of a variety of exposure circumstances. The results are used to determine where there are opportunities to incorporate further protective measures. In contrast, compliance assessments are usually designed to demonstrate that for predetermined exposure circumstances, conditions are or are not being met.

Operational nuclear facilities annually assess the potential effects of releases for compliance evaluation. The results are published in site environmental reports, which are made available to the public. Therefore many examples of public dose assessments are available in the public domain. Public dose assessments are performed for various nuclear facilities such as NPPs; for the mining and processing of minerals; as well as for the manufacturing and testing of military weapons.

A set of conceptual exposure scenarios has to be defined when a public dose assessment is performed. For each exposure scenario, dosimetric models are developed. These are expressed as a group of equations in mathematical form. More than one mathematical equation may be appropriate for a given dosimetric model [26]. For instance, for atmospheric releases, one might model the dispersion of a plume of airborne radionuclides, and another the deposition of these radionuclides on the ground.

These mathematical equations may be empirically or physically based. Furthermore, the complexity of the equation will depend on the level of detail required. The equations and their associated parameters form the basis of the mathematical formulas used to quantify dose to the public. Similarly, in this study, a simplified equation was developed, based on exposure determinants as input parameters, to quantify radiation exposure due to conceptual exposure scenarios for workers.

A public dose assessment is performed through a multistage process:

- In the first stage, information is obtained about the radiation source and the radionuclides discharged from the nuclear facility's site. Data includes [27]:
    - Type and amount of radionuclides being discharged.
    - Chemical and physical form of release.
    - Location and condition of release.
- In the second stage, information is collected on the concentrations of radionuclides in environmental media, arising from the modes of releases [26]. For instance, soil contamination occurs due to deposition of a plume of radionuclides, and water contamination due to migration of groundwater contaminated with radionuclides. In many

studies, environmental monitoring data is used at this stage to determine the radionuclide concentrations in environmental media.

- In the third stage, the concentrations of radionuclides are combined with human habit data. This is necessary to develop the exposure scenario to be assessed. At this stage, models of expected human behaviour of the exposed population are used. These models include information on physiological parameters, dietary information and residence data of the human population modelled.
- In the fourth stage, dose coefficients are used to convert the radionuclides ingested, inhaled or absorbed into dose values. Over the past years, extensive research has been performed on dose coefficient values. The results are available in published databases from various organisations such as the ICRP. In the final stage, all contributions from the different exposure scenarios, defined in the third stage, are collated. The result is a cumulative dose to the individuals of the reference group [26].

Guideline documents on public dose assessments warn analysts to avoid selecting extreme percentile values for exposure determinants used in calculations. This is to prevent excessive conservatism in the assessment results. Such results could lead to a significant and unrealistic overestimation of the dose. This will unduly burden the design of the nuclear facility, by requiring the implementation of excessive protective measures [26], [27]. For the same reason, this should also be avoided in worker dose assessments.

The most realistic method of public dose assessment is the extensive monitoring of the main exposure pathways. However, this is time-consuming and costly, and levels in the environment may be below the analytical detection limits of instruments. Typically, an assessment will involve a combination of environmental measurement data and modelled data [27].

It is not possible to model the various exposure scenarios of all the different individuals in the population. For this reason, the concept ‘reference group’ has been introduced to be used in public dose assessments. A reference group is intended to be representative of those people in the population who receive the highest dose. This concept is adopted in this study, but adapted to include selected occupancy categories.

In specifying reference groups, two broad approaches are used in literature:

- The first approach is based on carrying out surveys in the local population to determine their habits, where they live, what they eat, etc. From these surveys, the people who are receiving or who have received the highest doses are identified.
- The second approach involves using generalised data to establish generic groups of people who are likely to receive the highest doses [27].

Published data about food consumption habits and occupancy rates is available for various countries. However, it is recognised that using this information for generic groups is not ideal due to the large variation in information on [27]:

- Indoor/outdoor occupancies.
- Occupancies over inter-tidal areas and riverbanks.
- Consumption of terrestrial and aquatic foods for both average and high-rate consumers in different age groups.

Public dose assessments may be prospective or retrospective. Prospective doses are doses that might be received in the future, and retrospective doses are doses that have occurred in the past [26]. In this study, the worker dose assessment is a prospective assessment. It predicts the radiation exposure of workers on the plant to be built and operated in the future.

Some other important methods used in public dose assessments were adopted to develop the IA proposed in this study. These are:

- The need to identify conceptual exposure scenarios.
- The development of a simplified dosimetric formula and identification of its associated parameters.
- The stages for which to perform the calculation of a prospective assessment.

Dose reconstruction studies

Dose reconstruction is commonly used in occupational, environmental and medical epidemiological studies, as well as for compensation, litigation and incident assessment. It is used to estimate radiation dose received by an individual or group of individuals to evaluate historical or retrospective exposures [12], [28]. In dose reconstruction, it is important to characterise and include all significant sources of exposure in the assessment.

Many dose reconstruction studies are publicly available. The most important dose reconstruction studies have been associated with nuclear weapon testing, reactor accidents, routine releases from installations of the nuclear fuel cycle and careless disposal of industrial or medical radioactive waste.

These assessments make extensive use of historic information obtained from plant records, public records or environmental data to estimate the radiological source term. This information is often supported by direct measurements of environmental radioactivity, which are used to confirm and extend the original measurements. In most cases, the intake of the different radionuclides by the exposed individuals is calculated through the development of food chain models.

The US National Institute for Occupational Safety and Health (NIOSH) established priorities for data to be used in dose reconstruction. The top priority is assigned to individual monitoring data for the worker, followed by monitoring data for co-workers, area monitoring data and process data, such as the types and quantities of radioactive materials handled in the workplace [29].

The greatest challenge in a dose reconstruction programme is to obtain adequate data to characterise site operations and all plausible sources of exposure to radiation. Available plant data sets do not contain complete monitoring data for every worker at a given facility. However, these are usually sufficiently robust to generate statistical distributions of the exposure data for a given worker population [29].

The literature study demonstrates that the methods used to perform dose reconstruction studies are based on:

- measurement data from plant and environmental surveillance programmes;
- models and calculations; and (in the majority of studies)
- a mix of both.

The basic elements of dose reconstruction are important and were used to develop the IA proposed in this study [29]. Table 1 summarises the basic elements of the dose reconstruction process. These basic elements are also used in the development of the new IA.

Table 1: Basic elements of dose reconstruction process

Basic element	Summary description
Definition of exposure scenarios	Activities of individuals in areas where radiation exposure could occur and characteristics of radiation environment in those areas.
Identification of exposure pathways	Relevant pathways of external and internal exposure.
Development and implementation of methods of estimating dose	Data, assumptions and methods of calculation used to estimate dose from relevant exposure pathways in assumed scenarios.
Evaluation of uncertainties in estimates of dose	Evaluation of effects on estimated dose of uncertainties to obtain expression of confidence in estimated dose. This includes uncertainties in assumed exposure scenarios, models and data.
Presentation and interpretation of results	Documentation of assumptions and methods of estimating dose and discussion of results in context of purpose of dose reconstruction.
Quality assurance and quality control	Systematic and auditable documentation of dose reconstruction process and results.

The literature study demonstrates the importance of obtaining adequate data to characterise the radiological source term and habitual data of exposed individuals or populations. In dose reconstruction, the importance of the evaluation of uncertainties in the dose estimates is also emphasised. In this study, it is achieved by performing sensitivity analyses on input parameters. This is documented in Chapter 4.

Worker dose assessments

Worker dose assessments play an important role in any radiological protection programmes of operational nuclear facilities. The three main aims of worker dose assessments are [3], [14]:

- Determine the doses received by individuals or occupational category.
- Provide information on the effectiveness of engineering and procedural control measures.
- Demonstrate compliance with regulatory limits.

Many worker dose assessments are publicly available. It is standard regulatory practice to require worker dose assessments to be performed at operational nuclear facilities. These assessments are available to the stakeholders as annual dose reports, or are included in safety assessment reports.

At operational facilities, the management and assessment of occupational exposure to radiation are usually undertaken within the context of a radiation management plan. The radiation management plan contains information to allow all significant exposure determinants to be identified and recorded [4], [5].

Annual reports report the radiation exposure of each monitored individual, based on radiation exposure records. These radiation exposure records for operating plants are compiled from data obtained from personnel monitoring with specialised equipment. Each occupationally exposed worker is monitored with a personal dosimeter that provides information on the dose received by the worker while performing work in a mission area [4].

Dosimeter information captures both the ambient dose rate and the time spent in a mission area (occupancy factor) by registering the total dose received for a task. This dose information for different workers is grouped according to occupancy categories. Annual reports on worker dose provide information on exposures amongst the monitored individuals and are useful for trend analysis [10], [14]. Trend analysis provides important insights into whether radiation conditions on the plant are improving or deteriorating.

Worker dose assessments should be an integral part of safety assessments performed during the development of new nuclear facilities [19]. These reports provide information on techniques and practices employed to meet RP standards. Reports also include information on RP methods and estimated radiation exposure of personnel.

Standards and guideline documents require that the following plant information be reported and discussed in the chapter on worker dose assessment in the SAR [20], [30]:

- Radiation sources, layout and shielding.
- Material specification.
- Fuel integrity.
- Maintenance planning.
- Conceptual design of radiation monitoring systems.
- Quantitative dose estimation values.

These assessment reports should demonstrate that [20]:

- The plant components are designed to reduce the frequency of maintenance.
- The frequency of access to high dose rate areas is kept at a minimum.
- Inspections are performed in low dose-rate areas.
- Decontamination of components is possible before maintenance is performed.
- Shielding is adequately considered.

The quantitative dose estimate allows comparison of dose values with acceptable dose limits and makes comparison with other plant designs possible.

A worker’s annual dose is the sum of the doses received during all the missions they performed in a radiation environment. This is used to develop the dosimetric formula in this study. Each mission dose consists of the dose received [26]:

- when accessing the area;
- when performing the necessary functions; and
- when exiting the area.

Worker dose assessments are not only performed in the nuclear power industry, but also in other domains such as the medical, mining and military weapon industries. They are not only based on measurement data recorded in dose reports. In many of these assessments, modelling is used when information on some of the exposure determinants is absent.

In many examples, the radiation source term is determined empirically. Where measurement data of the source term is available, modelling is used to evaluate different exposure scenarios. Usually it is used to evaluate and compare the influence of different tasks and task duration on the calculated dose values. For instance, maximum worker dose associated with predetermined activities is estimated and used as the basis for comparison between alternative practices.

The following worker dose assessment methods were adopted in this study:

- the identification of the different exposure determinants necessary to compile an exposure scenario;
- the use and application of the concept of mission dose; and
- occupancy categories.

1.2.4 Dose assessment reports – water-cooled reactors

Extensive information exists on operational experience for water-cooled reactors [8]. New Generation III plant designs are examples of improved water-cooled reactors. These are either design improvements or design modifications of predecessor plants. The new designs of these plants are aimed at using the latest developments and existing operating experience to improve an existing design [8], [31].

In the early 1980s, the IAEA advised that worker dose assessments be based on information derived from the operational experience of NPPs with identical or similar designs. This is because the radiation source terms on operational plants are known, and the effectiveness of shielding analysis methods has already been tested and verified in these reactors. This then justifies further reductions in worker exposure by additional design measures [11].

The current trend in the deployment of NPPs is the emergence of multinational utilities, managing a fleet of one or two standardised designs. The deployment of standardised reactors will offer a much broader basis of experience feedback in design. Harmonisation and standardisation will also allow for the sharing of design assessments and licence application documents between regulators, vendors and operators [8], [16].

However, operational experience of water-cooled reactors is not applicable to GCR designs, due to the vast differences amongst plant designs. These differences result in significant differences in the radiation environment present on these plants. Existing operational data available from water-cooled reactors can therefore not be used to perform the worker dose assessment for HTGRs.

Examples of worker dose assessments are available publicly. These form part of the SARs submitted to regulators for licence application purposes [30], [31], [32], [33], [34]. These assessments were performed according to the guidelines provided in [14] and [35]. In all these examples, exposure data obtained from operating plants has been used. Available information from similar operating plants was extrapolated and has been used to develop detailed dose-predictive models for improved designs.

Unistar Nuclear Services submitted an SAR to the NRC in 2008 for the additional construction work performed at the Bell Bend NPP [30]. This SAR included associated models, assumptions and input parameters used to estimate the annual doses. Historical operational data was available to support these assessments. For instance, the following reports were referenced as source information: the Off-site Dose Calculation Manual, the annual Radiological Effluent Release Report, the Radiological Environmental Operating Report and the Final Safety Analysis Report (FSAR) [30].

The more recent worker dose assessment of Westinghouse's AP1000 is also based on operating experience [33]. Extensive operating information on occupational radiation exposure is readily available for American domestic plants having a Westinghouse design. In the design documents of the AP1000, the developers mention explicitly that historical data and operating experience were used to improve the assessment of worker dose [33].

Likewise, the European PWR, another example of an improved Generation III NPP, made extensive use of experience gained from operational experience. Its safety report mentions that a large effort was made to improve the plant design with respect to RP. It also mentions the experience gained during the design of former generations of PWR in France and Germany, and their current operation was used to improve the safety design [8], [31].

The above literature study demonstrates that safety assessments performed for water-cooled reactor-type NPPs make extensive use of the information obtained from operating plants. The majority of available worker dose assessments of NPPs are therefore based on extrapolation from, or adaption of, operational experience and measurement data.

1.2.5 Dose assessments – high temperature gas-cooled reactors

Much of the HTR technology, based on pebble fuel, was developed in West Germany between the 1950s and the 1980s. The design was originated by Professor Rudolf Schulten of Aachen University in Germany in the 1950s. Schulten pioneered the notion of nuclear fuel in the form of a pebble. The design was simpler than previous types of reactors, with greater safety features [9], P[3].

The development of the HTR has proceeded in two directions: (a) the pebble bed concept in West Germany and Russia; and (b) the prismatic core in the US, the United Kingdom, Japan and recently with the Gas Turbine Modular High-temperature Reactor (GT-MHR), also in Russia [36].

The following are early HTR development programmes:

- The Arbeidsgemeinschaft Versuchsreaktor (AVR) programme, conducted by Germany between 1967 and 1988 on a 15 MWe design P[3].

- The Thorium High-temperature Gas Reactor (THTGR) technology, based on thorium fuel, and developed by Germany between 1985 and 1989 for a 300 MWe plant [37], P[3].
- The 40 MWe Peach Bottom-1 Reactor (operated between 1967 and 1974) and the 330 MWe Fort St Vrain Reactor (operated between 1976 and 1988), developed in the US as part of its HTGR programme [37].

The PBMR is a high temperature, helium-cooled, graphite-moderated, and continuously fuelled pebble bed reactor. In the 400 MWth design, the Power Conversion Unit (PCU) is directly coupled to the reactor. Power turbines are driven through a direct closed-circuit helium cycle. This reactor design displays characteristics of Generation-IV reactors and has its origin in the German high-temperature nuclear reactor technology [38].

The PBMR design is a first-of-a-kind effort, due to the thermodynamic cycle and design parameters that differ from the German HTGR experience [17]. The available completed safety assessments performed on early HTGR designs are incomplete, outdated and not applicable to the 400 MWth PBMR design. In the SAR for the German AVR programme, the dose assessments were limited to area radiation level calculations.

The main purpose of these assessments was to perform radiological zoning. Cumulative annual worker dose calculations were not performed and reported in this report P[3]. This safety assessment was performed during the early years of nuclear legislation where comparison of annual worker dose to prescribed annual dose limits was not required as part of the submission of an SAR.

The most recent HTGR design in operation is a reactor designed and developed in China. The Chinese have conducted research on HTGR technology since the 1970s as part of the China High Technology Programme. China has operated the prototype HTGR-10 since the year 2000. This plant was licensed as a research reactor. It is possible to perform limited studies when licensing small research reactors.

The design of the Chinese 250 MWth HTGR Pebble-bed Module (PM) has been completed and is now under construction [39], [40]. Information to be used in the SAR of the 250 MWth HTGR PM was extrapolated from the research experience collected from the HTGR-10 prototype plant.

It is therefore recognised that new nuclear reactor concepts, such as the gas-cooled HTGR designs, face many design and licensing challenges. This is due to the lack of operational experience and safety standards for these plants. Innovative and novel methods have to be used to perform the worker dose assessment for these plants.

1.2.6 Integrated design process

The design of an NPP is performed by a multi-skilled team of design engineers, physicists and analysts [17]. Nuclear engineering analyses to be performed in support of the design, plant operation and licensing, span various disciplines. Analyses include logistic support task analyses; reactor neutronics; fuel performance; radionuclide and dust transport; and radiation shielding [38].

The coordination and communication amongst the individuals performing the design and analyses is vital to the overall success of the project. An integrated design process should be developed, implemented and maintained to ensure that this is achieved.

It is important to involve safety experts from the early design stages of an NPP. The earlier the life cycle safety requirements are identified and defined, the more effectively and efficiently the project will progress through the various phases of development. This is necessary for project management to ensure that safety project baselines, agreements and commitments are met [15].

This is also clear when the cost implications of a design change are considered. Modifying and correcting the design can become very expensive if significant safety-related problems have to be corrected late in the design process [15]. The licensing process of a nuclear plant has a significant cost implication. For this reason, developers of nuclear plants have to ensure that regulatory requirements are integrated in the design from early on in the project to meet the regulator’s expectations.

The design process for a complex facility, such as a nuclear plant, is iterative, incremental and continuously evolving. Therefore the safety design of a plant evolves and matures over time. The safety assessments and analyses performed within the design process should be used to identify unanticipated safety issues and provide feedback to improve the design.

The worker dose assessment is one of the tools used to assess the adequacy of safety considerations in the design and enables feedback to be provided the design engineers. The design information and detail progress as the design develops through the different iterations. As a result, the understanding of different design teams of the plant characteristics and behaviour also changes and matures. This allows for different levels of feedback in the integrated design process as the worker dose assessment develops and matures.

The worker dose assessment requires review, analysis and integration of information from various engineering groups, such as system design; Human Factors Engineering (HFE); maintenance design and support; and scientific and analysis groups. It requires the evaluation of the maturity of the various design aspects, such as system design, maintenance support design and safety analysis. One of the outcomes of the worker dose assessment is to determine whether the various design aspects are adequately integrated and whether the level of design maturity is aligned. A worker dose assessment is therefore a valuable engineering tool to evaluate whether the integrated design process is effective.

1.2.7 Design engineering tool

The worker dose assessment is a useful engineering tool, because it is able to identify areas for improvement in the design and provide feedback to design engineers. It also requires communication and integration of information between various design teams to ensure that safety is optimised.

The outcome of this IA enables the preliminary prioritisation of dose-significant tasks. This allows for the identification of the tasks that require further optimisation to demonstrate that dose optimisation was adequately considered. Further cost-benefit analyses can then be performed on these selected tasks to justify further design improvements.

The IAM enables the investigation of the influence of different exposure determinants through the performance of sensitivity studies. This can be used to develop specifications on: maintenance, surveillance, task frequency and duration (for maintenance design engineers); shielding requirements to obtain optimised dose rates (for Structures, Systems and Components (SSC) engineers); and required general area dose rates, etc.

Dose rates in radioactive areas are one of the inputs used in the worker dose assessment. Worker dose assessments performed for plants where no operating experience exists, require extensive safety analyses on radiation risk-significant components to be performed. These safety analyses have to calculate expected dose rates in the vicinity of the components, taking into account contributions from other sources [11].

These dose rates are influenced by the adequacy of the shielding used in the design of the various components. The dose rate information obtained from the safety analyses has to be communicated to the design teams responsible for developing the components. This allows for feedback regarding whether adequate shielding has been included in the design of the components. In this manner, the designers of the various components will receive feedback on optimising the components shielding design for radiation safety.

The maintenance and surveillance design teams are responsible for reducing radiation exposure through access restrictions. This is applicable to those cases where the design engineers cannot reduce dose rates by adding more shielding. These teams have to provide for minimum maintenance; minimum testing and calibration; minimum surveillance requirements; and maximum reliability [20].

If shielding design or access restriction cannot be further improved in some high dose rate areas, remote techniques should be included in the design to reduce to a minimum the need for personnel to enter high dose rate mission areas. Remote techniques are often dependent on purpose-made, specialised equipment [20].

There are numerous examples of safety design decisions that may affect multiple objectives, such as the installation of permanent fixtures in high dose rate areas during construction of the plant. This could lead to dose reduction during operation of the plant. In-service inspection and maintenance personnel will spend less time in high dose rate areas erecting removable scaffolding. However, permanent fixtures will increase the construction cost of the plant [8]. The IAM developed provides a quantitative method to evaluate this.

Many documents are available on operational radiological protection measures to consider during development of nuclear facilities [8], [10], [11], [19], [20]. The worker dose assessment provides an engineering tool to evaluate how these measures were included in the development of the nuclear facility.

1.3 Research objectives

Based on the literature study, a number of gaps and problems in the field of worker dose assessments have been identified for first-of-a-kind designs. From the literature review, the following research objectives have been identified:

- Establish an IA to perform a worker dose assessment.
- Develop a new quantitative assessment model based on a non-empirical approach.
- Develop an IAM based on a simplified formula to calculate the mission dose and annual worker dose by using exposure determinants as input parameters.
- Test the IAM by collecting, analysing, interpreting and integrating available plant design information and system-specific source term information.
- Compile an exposure scenario for hypothetical workers from selected occupancy categories.
- Identify unique assumptions to enable analysis of the identified exposure scenarios.
- Create a unique calculation sheet to calculate the annual worker dose.
- Perform sensitivity analyses by varying certain key input parameters to gain insights into the influence of this on the results.
- Provide a quantitative estimate of the cumulative annual worker dose during normal operation of the Fuel Handling and Storage System (FHSS) to evaluate whether prescribed dose limits are met.

- Critically evaluate the results obtained in order to provide new insights and feedback for the development of the next phase of the reactor.
- Use the worker dose assessment as an engineering tool to evaluate whether radiological protection criteria were adequately considered and integrated during the development phase.

This IA and IAM were tested by implementing them on the FHSS and testing the following hypothesis:

The engineers of the FHSS considered radiation safety measures adequately during the design of the FHSS. The outcome of this assessment must therefore quantitatively demonstrate that the most exposed worker's annual dose shall comply with the set dose constraints.

1.4 Limitations

The following limitations of this study are noted:

- This IA and IAM do not include the internal dose contribution due to the inhalation of airborne radioactive material. The models to determine radioactive leakage from the SSC, and the circulation of this airborne radioactive material through the building, were still being developed and were immature at the time of the study.
- Uncertainty exists regarding the amounts of dust generated as a result of abrasion of the fuel spheres. Abrasion of fuel spheres is due to movement through the reactor core and Sphere Circulation Subsystem (SCS). This radioactive-contaminated dust has the potential to contribute significantly to a worker's dose. It was assumed that the dust-purging system was 100% efficient in removing loose dust from the sphere circulation pipes. Besides this, it is standard operational radiological protection practice to wear full-face masks and protective clothing where a risk of inhalation of radioactive dust exists.
- This research focused only on normal operating conditions and did not include accident conditions. Special accident scenarios have to be compiled for abnormal and transient plant conditions, and will be separately analysed as part of abnormal plant conditions. This will be addressed in separate studies through other techniques such as probabilistic risk assessment.
- This study was performed in the early developmental phase of the plant. The maintenance task breakdown and analysis were not completed. The design team focused on high-risk maintenance tasks during this phase of development. The results should therefore be regarded as upper-limit dose values for normal plant operation.
- The results were validated through a benchmark exercise. Limited annual worker dose information is available for other HTGR programmes. Only average worker dose and

collective radiation exposure values are available for the Fort St Vrain programme. Su and Engholm estimated collective radiation exposure for HTGR designs [41]. At this early design phase, the validation of this IAM was only to ensure that the results were in the expected range of values reported for HTGR designs, and produced conservative estimates.

- The IAM developed in this study calculates the upper annual worker dose values for the PBMR. However, the collective radiation exposures reported for HTGR designs are based on the average worker dose levels. It is recognised that the annual worker dose on a plant can vary by an order of magnitude. However, the benchmark exercise demonstrates that the results obtained are in the expected range of values for HTGRs. This is adequate to use in the decision-making process during design assessment.
- Complete validation of the IAM for the PBMR design can only be performed after the demonstration power plant has been built. Validation will be performed by comparing the estimated worker dose values with dose measurements obtained from using specialised dosimetric equipment.

1.5 Contributions from this research

1.5.1 Provision of uniquely devised model

The IA and IAM were devised in a unique way. The literature study indicated that worker dose assessments for NPPs are mainly based on or extrapolated from measurement data in operational plants. In the absence of this data, there are unique challenges in performing such an assessment. It was necessary to devise a unique approach to perform this assessment for the PBMR.

The IA was based on a combination and integration of methods used in public dose assessments, dose reconstruction and worker dose assessments. This culminated in an IA tailored to the needs of worker safety assessment for a design of a first-of-a-kind plant.

It is based on estimating the dose received by a hypothetical worker, defined as the representative worker, in a prospective exposure scenario. These methods are adopted from public dose assessments.

The breakdown of design and safety analysis information was required to determine the worker exposure determinants used in worker dose assessments. This entailed extensive task analysis and task breakdown. This was performed by following the steps described as the basic elements of a dose reconstruction study.

A simplified IAM was developed by linking the worker exposure determinants in a simple dosimetric formula. This enabled the creation of an input-output worksheet to provide quantitative results of annual worker dose.

1.5.2 Evaluation of worker dose early in design process

Assessment results make it possible to evaluate annual worker dose early in the design process, in order to compare design changes and improvements.

The results obtained from safety assessments should be used to determine the feasibility of major safety design concepts. They are therefore an important engineering decision-making tool.

System optimisation can be achieved by comparing results obtained from different assessments for various combinations of alternative SSC engineering design concepts, maintenance concepts and operating procedures [19]. This has been discussed in paragraph 1.2.7.

1.5.3 Provision of simplified model for safety specifications and requirements

The research provides a simplified IAM to set up safety specifications and requirements for SSC.

In the design of the many different facilities on an NPP, insight into the level of radiation hazard present on the plant is required. This IA and IAM enable the design team to develop a better understanding of the level of hazards present on the plant. For instance:

- The design of the waste storage facilities and decommissioning plan depends on the extent of radiation hazards present on the plant. The levels of radiation hazards present on the plant are quantified by the analyst when this worker dose assessment is performed. This information is used to develop these specialised SSC specifications and requirements.
- Special tooling is required for the operation and maintenance of the plant, e.g. regarding the possibility of spheres being stuck in the conveying lines. This assessment determines whether the design of robotics and semi-remote tooling has been adequately addressed. Results are used to specify requirements on the design of this equipment, such as distances from the operator and shielding thicknesses of the containers.
- The level of radiation hazards on the plant will determine the radiation monitoring equipment to be purchased for the operating plant. Results will be used to specify requirements for this radiation monitoring and surveillance equipment.

New results and insights gained from a worker dose assessment provide valuable design inputs for identifying safety specifications and requirements. They also provide valuable insights into expected conditions to guide designers who have to consider safety issues. Therefore, one of the important outcomes of the worker dose assessment is the identification of safety specifications for the different design engineering teams.

1.5.4 Evaluation of effectiveness of integrated design process

This assessment provides a new tool to evaluate the effectiveness of the integrated design process. The importance of the integrated design process has already been discussed in paragraph 1.2.6. The IA provides a coherent framework for organising and assessing knowledge about ambient radiation environmental conditions on the designed plant.

The worker dose assessment requires the collection of design information from various engineering groups. During this process, it is possible to evaluate whether the necessary interfacing between the various disciplines took place.

Integration of the radiation safety aspects recommended in international guidelines has to be demonstrated in all aspects of the design. The review of the design information provides insights into whether this received adequate attention. For instance:

- Has the HFE group included adequate access servitudes in the plant layout to ensure quick entry and easy access? This should also be reflected in the building layout diagrams and specifications.
- Has remote operation capability in high-risk mission areas been identified? Was this communicated to the different engineering groups to design appropriate equipment?
- Has adequate shielding been included in the design to ensure personnel safety? What is the status of the shielding analyses to demonstrate that radiological safety has been adequately considered?
- Have insights been gained into the expected radiological conditions present on the plant? This information is used to plan radiological zoning. The locations of all the high-risk areas have to be identified on plant layout diagrams and the plant shall be zoned accordingly.
- Have all possible radiation sources been identified and evaluated? Have these results been used to rank SSC in terms of radiation hazard? Have these results been communicated to engineers to implement design improvements towards reducing dose rates in areas where routine operations are performed?
- Have Instrumentation and Control (I&C) systems been designed to monitor radiation levels? The design and selection of these instruments depend on the level of radiation present in the area. Has adequate information been provided to the designers of the I&C systems to design these systems?
- Have the duration and frequency of access been minimised? The assessment uses maintenance duration and frequency as an input. In high dose rate mission areas, the duration and frequency of access should be minimised. This is evaluated in this assessment and feedback can be provided to maintenance and logistics support engineers.

1.5.5 Identification of outstanding design information

The IA assists the design team to identify outstanding design information. It offers a systematic approach to the identification of gaps and the adequacy of integration of design between the different engineering groups:

- Several iterations of this quantitative dose assessment will be performed as the plant develops and matures. This assessment and its associated review comments will provide useful insights into determining outstanding information and establishing priorities for further assessments.
- Outstanding engineering information will be identified, e.g. task breakdown of high-level procedures.
- Insights gained from these assessments might indicate that it is necessary to implement operational controls where engineering controls cannot be improved or become too costly. This will also be identified and documented as outstanding information during the performance of this assessment.
- The different iterations of these studies need to verify key safety strategy assumptions, make technology selections, improve process operations, and identify and document safety improvements through the evaluation of the available design information.

1.5.6 Development of operating procedures and test programmes

The IA allows the development of operating procedures and test programmes for first-of-a-kind nuclear plants. Written safety documents must be compiled, implemented and maintained to explain procedures that need to be followed when working in hazardous areas [42]. These have to be completed before the plant is commissioned. For instance:

- The development of operating procedures and safety programmes is started during the basic engineering design phase. These procedures describe working methods and include information on work permits required to access mission areas, etc. The levels of radiation hazards that exist on the plant have a significant influence on the development of the content of these procedures and programmes, and must be included in these procedures and programmes.
- It is recognised that analyses and simulations are not adequate to ensure safety. Comprehensive V&V and test programmes have to be carried out on experimental facilities to demonstrate the adequacy of the analyses, simulations and models used. This assessment provides valuable insights into determining the requirements of these test programmes.

- Outstanding intervention, prevention and control measures can be identified based on the assessment results. These have to be logged in a database and included in the procedures and programmes once these are developed.
- The results of the safety assessment will influence training requirements and skills development of the staff to be employed on the operating plant.
- This IAM enables the calculation of cumulative dose during the detail design phase, when information on the staffing structures becomes available. The annually accumulated dose is a performance indicator used to compare the safety of NPPs. Cumulative dose combines individual annual dose with the number of staff employed on the plant.

1.5.7 Assessment of worker dose without operational experience and measurement data

The IA and IAM enable a developer of a nuclear facility to perform a worker dose assessment in the absence of operational experience and measurement data.

Any developer of a nuclear facility has to perform comprehensive safety analyses. This requires the identification of all sources of exposure at the facility. The radiation doses received by the workers working at the facility, due to exposure to these sources, are then estimated. This simplified IAM enables the developer of a new nuclear facility to perform such a worker dose assessment in the absence of historical data.

1.5.8 Development of formula for results for comparison with numerical targets and limits

A simplified dosimetric formula was developed that provides quantitative results necessary for comparison with prescribed numerical targets and limits.

This simplified dosimetric formula provides quantitative results. It is based on the various exposure determinants contributing to the worker's dose and is used to calculate the worker's annual accumulated dose.

The quantitative results can be used to compare projected plant performance with prescribed numerical targets and limits.

1.5.9 Calculation of baseline dose values for evaluation and comparison purposes

The IAM makes the calculation of baseline dose values possible, for evaluation of different design options and future comparison with monitoring data.

The annual worker dose values can be used as baseline or benchmark values for the comparison of design improvements or, in the future, test monitoring data. For instance:

- This IAM allows for the comparison of annual dose value results for different design baselines. Alternative design concepts can be analysed and compared. The safest alternative design, resulting in the lowest annual dose, can then be selected to optimise the design.

- The quantitative results derived from this assessment provide baseline values against which empirical data obtained in the future test or demonstration facilities can be tested. This allows for the improvement of models, simulations and analyses developed in the design phase.

1.6 Summary

The aim of this chapter was to introduce the reader to this field of study. The literature study gives the reader a brief overview of the importance and historical development of worker dose assessments in the nuclear industry. Furthermore, the literature survey summarises the different methodologies used in complex systems to perform dose assessments and analyse radiation exposure scenarios.

The literature review has demonstrated that the methods used to perform worker dose assessments of nuclear facilities are mainly based on existing operational experience and measurement data. There is therefore a gap in the literature regarding how to perform these assessments during the design of a first-of-a-kind plant. A different approach is required to perform a worker dose assessment in the absence of operating experience and measurement data.

This chapter has also described the research objectives, the limitations of this study and the contributions made by this study.

CHAPTER 2: PBMR PLANT INFORMATION

Chapter 2 gives a brief overview of available PBMR plant information to test the integrated assessment and integrated assessment model.

Chapter 2: PBMR plant information

2.1 Introduction

One of the research objectives of this study is to test the IA and IAM developed. This is discussed in Chapter 4 of this document. A worker dose assessment is possible only after an in-depth review of available plant information, when operational experience and measurement data are absent.

The primary objective of this chapter is to present those aspects of system design and interrelationships that affect plant radiological conditions. The function and purpose of various systems are presented, along with their associated radiological hazards.

It is not necessary for RP personnel to have the in-depth working knowledge of system operational-related parameters that is required of plant operators. Consequently, the intricate details of system design and functions comparable to the level of knowledge required of plant operators are not covered. However, it is essential to have sufficient knowledge of plant systems to adequately address the radiological requirements for activities performed either on or in the vicinity of plant systems.

Available PBMR plant information was reviewed to identify the possible exposure determinants on the plant. Exposure determinants include the type of work tasks; task frequency; task variability and task duration; dose rate; and route of exposure [4]. These exposure determinants are used to develop conceptual exposure scenarios for the PBMR.

The IA and IAM were tested on the FHSS, which was selected for the following reasons:

- This system will contribute significantly to the radiation dose received by workers on the plant. Radioactive material (such as fuel spheres) will accumulate in the SSC and is the source of radiation exposure of the workers.
- The system's routine operations are comprehensive and justify a dedicated specialised team of workers.
- This system is a major support system of the plant.
- The design of this system is at a level of maturity where significant changes to its basic design are not expected.

The FHSS transports, measures and stores fuel spheres; graphite spheres; damaged and scrapped fuel spheres; and contaminated graphite dust. Chapter 2 provides an overview of the functions of the FHSS and its SSC. This information is the background information required to understand the expected maintenance and surveillance tasks identified, and the safety analysis results performed on these SSC.

The available maintenance and surveillance procedures provide the information on the type of work tasks, task frequency, task variability and task duration. Safety analysis results provide the information on the magnitude of radiation exposure (dose rate) and route of exposure. These exposure determinants are the input parameters used in the dosimetric formula developed. The dosimetric formula includes the input parameters necessary to quantify the exposures resulting from the conceptual exposure scenarios.

2.2 Fuel handling and storage system functions and overview

2.2.1 Preamble

The primary purpose of the FHSS is to circulate the spherical fuel elements through the reactor core while the reactor is operating at power. Figure 2 shows the location of the FHSS inside the reactor building. Sphere circulation is achieved by means of a combination of gravitational flow and pneumatic conveying. This system uses helium at Main Power System (MPS) operating pressure to circulate spheres. Fuel circulation is done according to a ‘multi-pass’ fuelling scheme. This means that fuel spheres are moved through the core several times before they are removed as spent fuel.

The burn-up level of the fuel is monitored continuously. Fuel is removed from the circulation loop when the predetermined fuel burn-up level is reached. These spent fuel spheres are then stored in intermediate storage tanks. Spent fuel is replaced by fresh fuel, which is introduced into the circulation loop. Provision is also made for: sampling fuel to verify fuel quality; identifying scrap fuel; and storing accumulated scrap fuel and dust P[4].

The following are the most important mechanical subsystems of the FHSS. Most of the maintenance and surveillance work will be performed on these systems P[4]:

- Sphere Circulation Subsystem (SCS)
- Gas Circulating Subsystem (GCS)
- Sphere Replenishment Subsystem (SRS)
- Auxiliary Gas Subsystem (AGS)
- Sphere Storage Subsystem (SSS)
- Instrumentation equipment
- Scrap sphere sampling

These subsystems are connected to form a pressurized system consisting of valve blocks, valve inserts, vessels, filters, blowers, instrumentation equipment, other Commercial Off-the-shelf (COTS) items, heat exchangers and unloading devices. The conveying lines comprise many components including pipes, flanges, supports, gaskets, bolts, valves, strainers and expansion joints P[4], P[5].

As shown in Figure 2, the three major building compartments associated with the PBMR design are the FHSS, PCU and generator buildings. Figure 2 also shows the position of the reactor vessel in the building and the major maintenance locations, which are magnified in Figure 3 to

Figure 8.

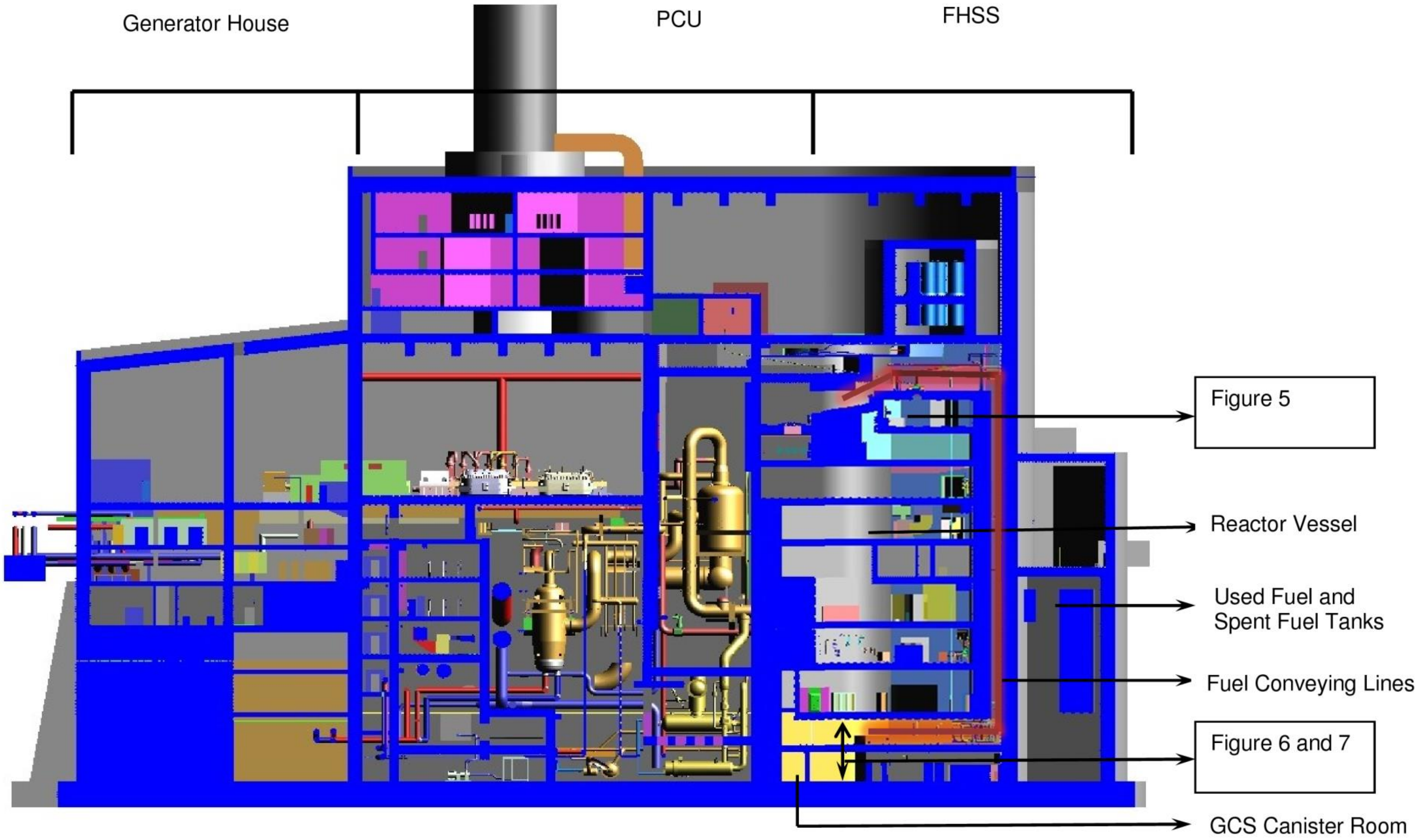

Figure 2: PBMR building layout

2.2.2 Fuel handling and storage system vessels

Vessels (also referred to as tanks or canisters) are cylindrical, structural envelopes that are used to hold and/or store solids, liquids or gases under various internal pressures. They are positioned vertically to reduce footprint space requirements. The FHSS vessels have to store radioactive material such as contaminated helium, radioactive graphite dust, fuel spheres and radioactive graphite spheres.

The following vessels form part of the FHSS:

- Graphite dust storage tank (low pressure)
- Spent fuel storage tank (low pressure)
- Graphite sphere storage tank (low pressure)
- Scrap sphere storage canisters (low pressure)
- FHSS filter blow back accumulator (high pressure)
- FHSS filter blow back booster vessels (high pressure)
- FHSS filter vessel

Figure 3 shows the design and location of the graphite dust storage tank and dust filter, which are part of the GCS. As can be seen in the figure, the building design includes dedicated rooms for the tank and dust filter to ensure that the walls and floors provide adequate shielding to reduce exposure to radiation.

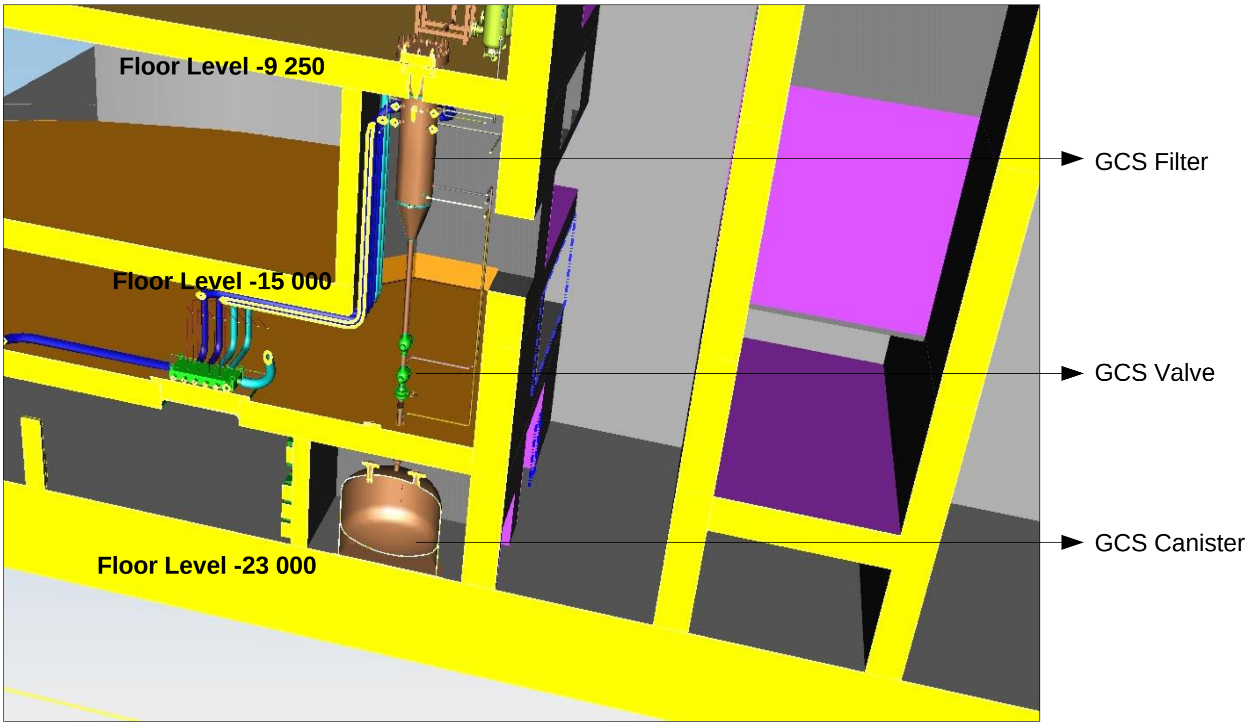

Figure 3: Dust filter, valve and canister located on levels -9 250, -15 000 and -23 000

Figure 2 shows the orientation of the graphite dust storage tank or GCS canister room in the building. This room is located on the lowest level in the building.

These vessels are located in rooms classified as exclusion areas, because of the high dose rates expected around them. Exclusion areas are not included in this assessment. These areas will not be accessed routinely during normal plant operation. They will be included in the assessment of plant shutdown states, which is outside the scope of this study.

2.2.3 Sphere circulation subsystem

Description

The SCS consists of the main sphere-circulating transport loop. Figure 4 shows the reactor vessel coupled to the SCS subsystem and lines. The SCS lines are illustrated with yellow, blue and red lines. This system is coupled to the reactor vessel. A slight over-pressure is created in the SCS to prevent gas and dust from flowing from the core into the SCS. Spheres enter or exit the loop via the various charge and discharge locks. Pneumatic conveying is done by utilising the circulating gas provided by the GCS.

On the opposite side of the reactor vessel, the core inlet and outlet pipes couple the PCU to the reactor. Figure 4 shows the reactor vessel coupled to the FHSS and PCU.

Figure 2 shows the position of the fuel conveying line in the FHSS building relative to the reactor vessel. The fuel conveying lines will be located inside a dedicated building servitude.

The SCS contains high levels of radioactive material, mainly in the fuel spheres. Work performed on this subsystem was included in the conceptual exposure scenarios identified in the worker dose assessment performed. Dose rates elevated above background values will be present in the areas accessed to maintain this system.

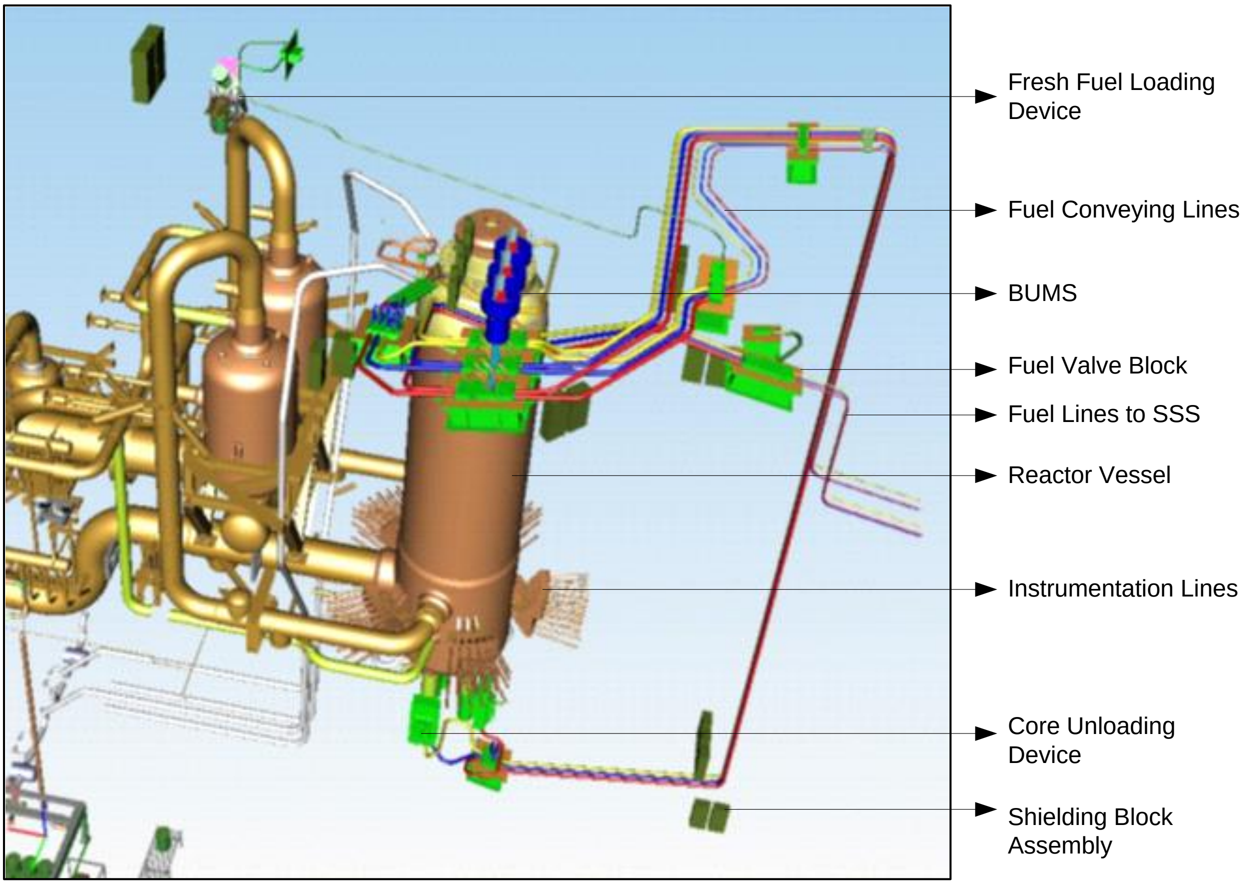

Figure 4: Reactor core coupled to pipes and main components of structures, systems and components

Valve blocks

Valve blocks are important components inside the SCS. Each valve block is a uniquely machined metal forging. The metal provides shielding against radiation emitted from radioactive material inside the valve block. No repairs on valve block bodies are planned during the operating life of the plant. These blocks will be located in the floors as separate units P[4], P[5].

Inside the valve blocks there are different borings. The functional units such as the diverters, indexers, isolation valves and other components are located inside the borings. These functional units are called Process Element Assemblies (PEAs). The function of the valve block is accomplished by grouping different types of PEAs in a predetermined manner P[4].

Valve blocks are connected by pipelines. All valve blocks are angled between 10° and 15°, with the lowest side indicating the direction of sphere flow. This design feature allows for natural gravitational flow of the fuel spheres to avoid spheres from becoming stuck in the valve block.

The concrete floors in the building will provide shielding of the pipelines routing the radioactive fuel spheres P[4].

Valve blocks are positioned either in the ceiling, facing down; or vertically in the walls, facing forward. Servitudes of at least 1,5 m between the valve blocks and the floors are provided for in the design. This is to ensure that sufficient space is allowed for the maintenance of the valve block. It is important to provide unobstructed areas for maintenance work on the valve blocks to ensure ease of maintenance P[6].

The following 13 valve block units are included in the FHSS design P[4]:

- Core Unloading Device (CUD)
- Cleaning block unit
- Conveying Block Assembly (CBA)
- Burn-up Measurement System (BUMS)
- Charge lock outlet unit
- Discharge lock outlet unit
- Gas supply block unit
- Gas return block unit
- Charge lock inlet unit
- Discharge lock inlet unit
- Filter block unit
- Isolation block unit
- Sensor block unit

It is recognised that the bearings and sealing elements will require regular surveillance. Shaft seals form part of the PEA and require frequent inspections to monitor the level in the pressurized oil accumulator. Low oil levels have to be replenished to ensure shaft seals are pressurized for effective sealing of helium gas. Figure 5 shows the location of the BUMS unit and charge lock outlet valve block units in the floor above the reactor core, and the units' access route. Figure 2 shows the location of Figure 5 relative to the reactor vessel.

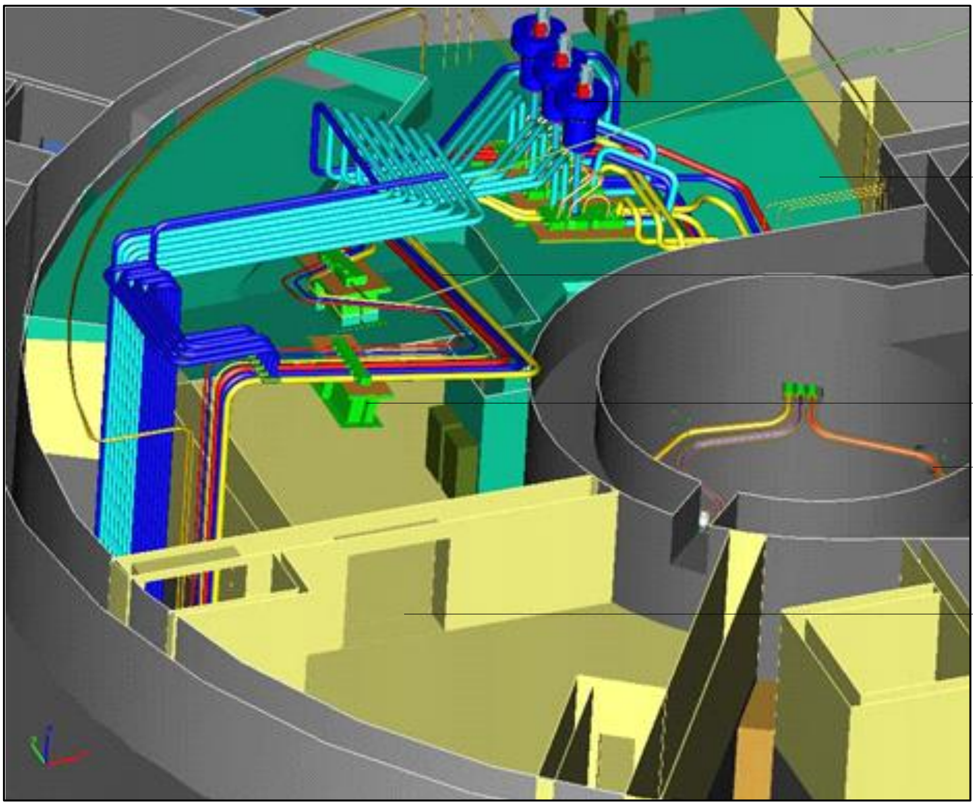

Figure 5: Burn-up measurement system and charge lock outlet valve blocks

Fuel spheres can only pile up at four of these valve block locations due to a sphere becoming stuck. This is due to the elevation of the piping. These locations are:

- CBA
- BUMS assembly
- Isolation block assembly
- Discharge Lock Outlet Block (DLOB) assembly

Work performed on these valve blocks was included in the conceptual exposure scenarios identified in the worker dose assessment. In this iteration of the assessment, the influence on the results of a pile-up of fuel spheres is investigated. This is the worst-case scenario for the removal of a PEA, and results in upper limit values for the expected dose received. Dose rates elevated above background values will be present in the areas where these valve blocks are located.

2.2.4 Gas circulating subsystem

The GCS is coupled to the MPS. It provides the transport gas that moves the spheres through the pipelines. It also performs the pressure management and secondary conveying line-purging functions. Frequent purging of pipelines is necessary to clean radioactive dust and stuck spheres from the lines. Radioactive dust originates from fuel abrasion due to the circulation of fuel spheres through pipelines and the reactor core.

Contaminated dust within the pressure boundary is continuously removed by the GCS gas filter. The dust falls from the filter into the graphite dust storage tank. This ensures that internal surfaces of the pipes remain clean. The dedicated graphite dust storage tank forms part of the system and is designed for lifetime storage of radioactive dust.

Component surfaces are designed to be smooth, without sharp edges and without unnecessary pockets. This helps to limit radioactive graphite dust deposition and accumulation inside the pipelines. The GCS maintains the circulating gas temperature at a selected value and conditions the SCS piping.

Figure 6 shows the major components of the GCS, located underneath the reactor vessel. The GCS pipelines are the dark blue lines in the figure. Figure 2 shows the location of Figure 6 relative to the reactor vessel.

This subsystem will be contaminated, as it contains radioactive material. Work performed on this subsystem was included in the conceptual exposure scenarios identified in the worker dose assessment. However, maintenance on the GCS dust filter and GCS dust storage tank was excluded, because these are located in areas classified as exclusion areas.

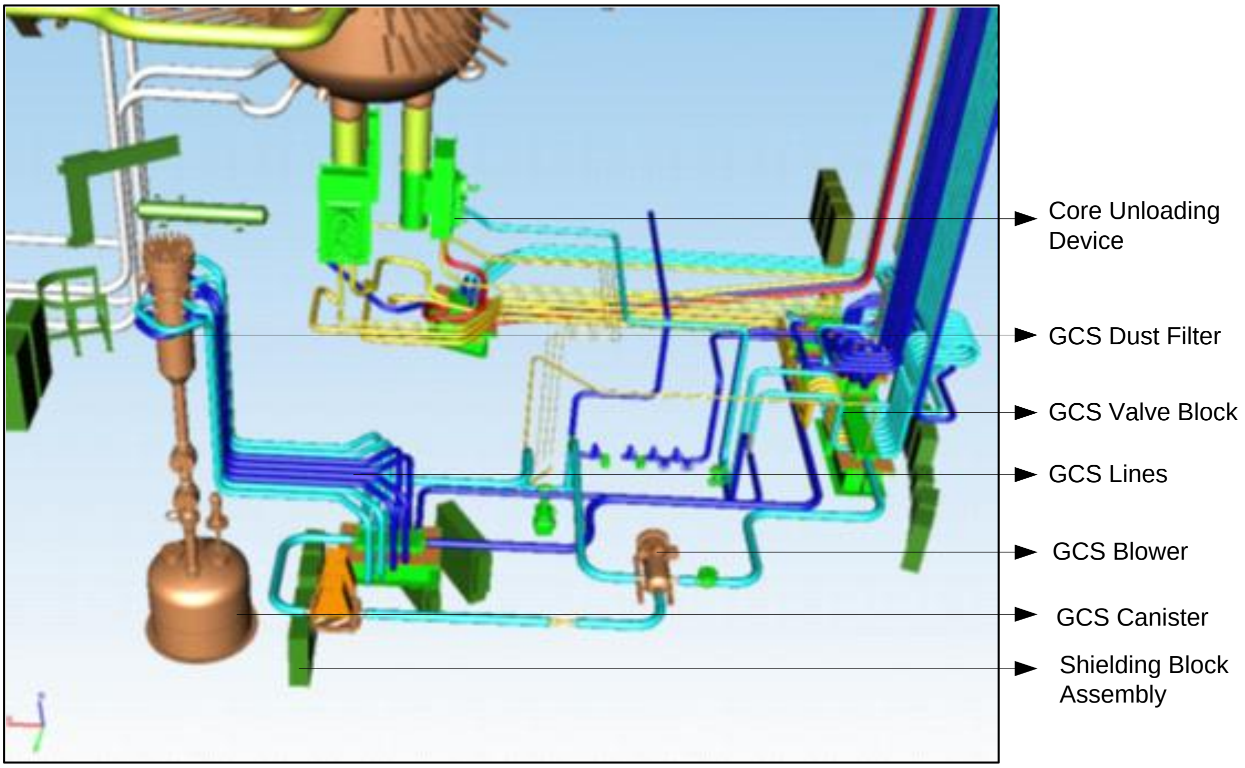

Figure 6: Gas circulating subsystem and main components

2.2.5 Sphere replenishment subsystem

This SRS stores the fresh fuel canisters and introduces fresh fuel spheres into the SCS conveying lines. These fuel spheres have not been circulated or exposed to the neutron flux inside the reactor core. The dose rate on the fresh fuel spheres is very low (contact dose rate is 3 μSv/h) P[7].

The neutron flux inside the reactor core causes the uranium isotopes inside the pebble fuel spheres to undergo fission reactions. These fission reactions result in elevated levels of radioactive material inside the fuel spheres. The radioactive fuel spheres are the source of most of the radiation exposure in the FHSS.

The SRS does not require significant shielding, due to low dose rates present in the vicinity of fresh fuel. Furthermore, little time is expected to be spent in this area, therefore a worker working here will not receive a significant dose. For this reason, work performed on this subsystem was not included in the conceptual exposure scenarios identified in the worker dose assessment.

2.2.6 Auxiliary gas subsystem

The AGS is a support system to the FHSS. It supplies, distributes and collects the helium gas inventory, and ensures purity, minimum leakage and pressure tightness. This system is isolated from the GCS and filled with uncontaminated gas. All static pressure boundary penetrations or flanges are subject to manual leak monitoring to identify streaming.

Dynamic penetrations (such as shaft penetrations of the PEA) are serviced by the AGS as part of the leak management condition monitoring function. This ensures pre-warning of seal wear-out or other malfunctions, the correction of which can then be planned. The AGS valves require maintenance in a radiation environment. Work performed on this subsystem was included in the conceptual exposure scenarios identified in the worker dose assessment.

2.2.7 Sphere storage subsystem

The SSS stores the spent fuel, used fuel and graphite spheres. The BUMS measures the fuel to determine if the required burn-up level of the fuel has been reached. If the burn-up level is reached, the fuel sphere will automatically be sent to the spent fuel tank. Loading and unloading will be from the top of the tanks [43].

A preselected vessel will be connected to the FHSS by setting up a route manually when the tank is still empty. The spent fuel sphere will roll down the 10°-sloped sphere pipe into the mechanical brake, due to gravitation P[8]. The filling of a spent fuel tank will take roughly a year after the reactor has been under full power conditions. When the vessel is filled to capacity, the helium will be replaced with nitrogen and the vessel will be sealed [43].

Only the routine operations carried out in the SSS service hall, located above the SSS tanks, are included in the conceptual exposure scenarios. Various tests and checks have to be performed on assemblies located in this area. The thickness of the service hall floor is designed to reduce dose rates from the fuel tanks to background levels.

Figure 7 shows the orientation of the used fuel and spent fuel tanks related to the CUD. The reactor vessel, which is located directly above the CUD, is not indicated in Figure 7. Figure 2 shows the position of these structures in the FHSS building relative to the reactor vessel.

Figure 7: Major structures, systems and components on levels -15 000 and -23 000

2.2.8 Scrap sphere sampling

Scrap sphere canisters will be located in the shielding floor under the CUD compartment. These canisters will be removed every six months to allow for analysis of the scrap spheres. Specialised tools will be used to remove the canisters from underneath the CUD compartment. These tools can be handled remotely to ensure that adequate shielding and distance are provided between the radiation source and the worker.

One of the design specifications is that adequate shielding be provided to ensure low dose rates outside the canisters when installed. The design specification requires that a dose rate value of 2 μSv/h should not be exceeded outside these canisters. A maintenance tool, which removes the bolts, will be used to remove these canisters. This will allow for adequate distance between the maintenance worker and the radioactive source.

The sphere canister will then be lowered remotely into a dedicated transport cask P[9], P[10].

Work performed on this subsystem was included in the conceptual exposure scenarios identified in the worker dose assessment.

2.2.9 Instrumentation equipment

This paragraph explains the maintenance aspects of the instrumentation equipment that supports the SCS, GCS, SRS, AGS and SSS.

Instrumentation equipment sends, receives and controls signals to equipment. Instrumentation will be located in purpose-built cabinets or attached to equipment. This equipment does not contain radioactive material. It will be installed in low dose rate areas inside the FHSS building, and outside high radiation zones P[4].

The placement and spacing of cabinets is done in such a manner as to facilitate access. The design requirement is that the frontal face of the cabinets should be unobstructed. This is to ensure that enough space is provided in front of the cabinets for workers to do maintenance work. Sufficient space between the back of the cabinets and the walls will be provided for access to attached cable connections P[4].

Since the instrumentation will be inspected frequently, access to the FHSS building is necessary. It is expected that testing of I&C equipment will be a major part of the maintenance technician’s work. Work performed on this subsystem was included in the conceptual exposure scenarios identified in the worker dose assessment.

Maintenance tasks on instrumentation equipment are included to ensure that the conceptual exposure scenarios that have been developed represent realistic exposure conditions. This is to prevent excessive conservatism in the assessment results. Maintenance tasks, such as the removal of a PEA when a pile-up of fuel spheres is present, are extreme exposure conditions. Excessive conservatism could lead to an unrealistic overestimation of the dose. This would unduly burden the design by requiring excessive protective measures to be implemented [26], [27].

2.3 Fuel handling and storage system operating and maintenance overview

2.3.1 Fuel handling and storage system operation

The FHSS operates under varied thermo-hydraulic conditions, ranging from high pressure to atmospheric pressure. It is necessary for the FHSS to be isolatable from the reactor vessel, MPS and other support systems. This is to allow for maintenance and surveillance of the system, while other systems continue operation. To achieve this, the fuel feed system is shut down, separated from the main helium system by double isolation valves, depressurised and purged with air.

Pressure will be maintained in the FHSS pipes during most maintenance activities. This is to condition the pipes. In very special cases, e.g. replacement of inserts, the pressure will be lowered to atmospheric conditions. Some SSC are part of the reactor pressure boundary and are not isolatable. These SSC can only be maintained when the reactor is depressurised. An example of such an SSC is the CUD with its different components. For this reason, maintenance on the CUD will occur only during planned plant shutdown periods.

Figure 8 shows the primary SSC components, its functions and their relationships. This engineering diagram illustrates the location of the different SSC inside the FHSS building. It is an aid for the reader to understand the orientation of the various components P[4].

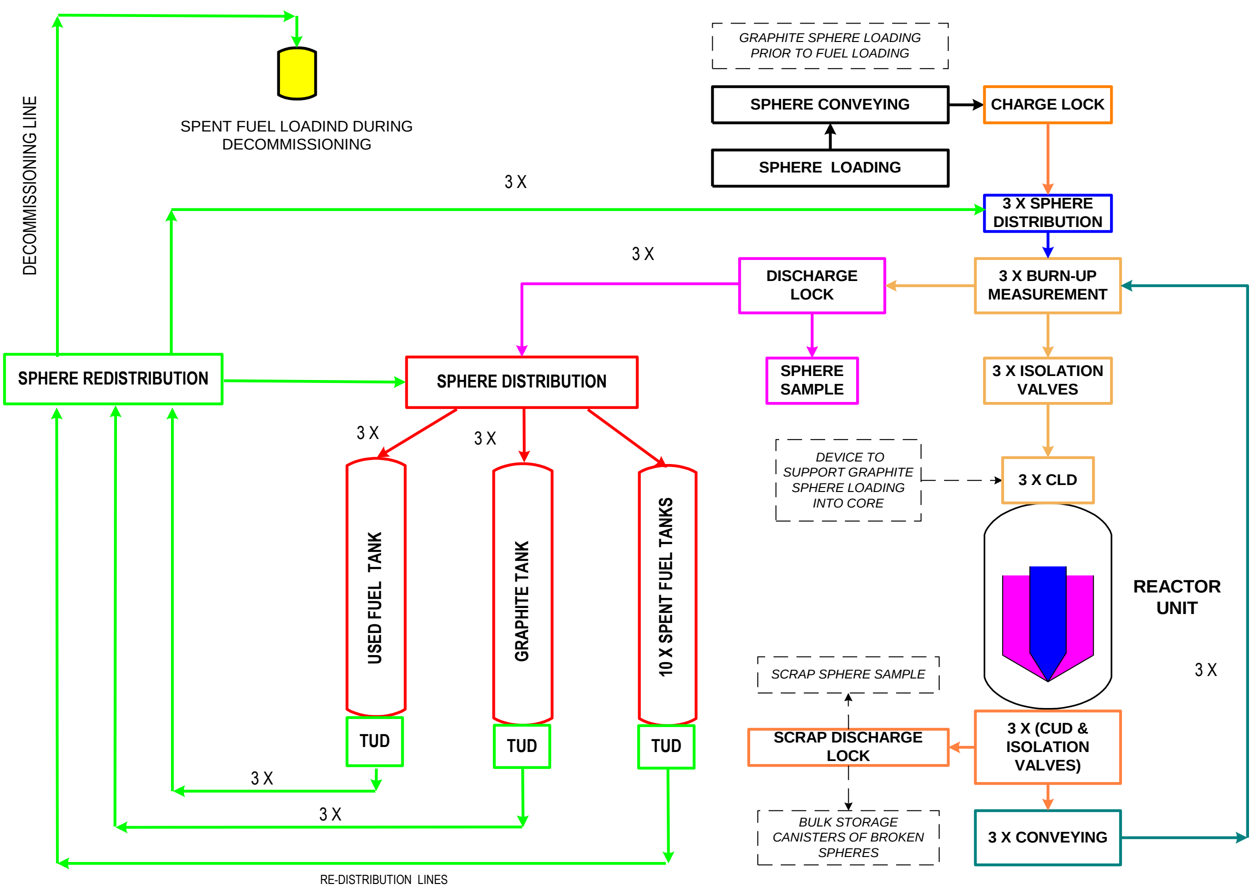

Figure 8: Engineering diagram of fuel handling and storage system basic functions and relationships

Maintenance and surveillance are planned for periods when the FHSS is not operating and spheres are not circulated. This is important to ensure that low ambient dose rates exist when workers access the plant. High dose rates will exist when large numbers of fuel spheres are present in the pipelines during fuel circulation. The following are the most important FHSS activities P[11], P[12]:

- The primary activity of the FHSS is sphere circulation. The expected time allocated for sphere circulation is 8 h to 12 h per day. Daily sphere circulation is concluded by emptying all buffers.
- Preventative closed maintenance will not breach the pressure boundary. A dedicated period of 2 h per day is allocated for preventative closed maintenance.
- Corrective maintenance might require a breach of the pressure boundary. This will depend on the task to be performed. There are 12 h to 16 h a day available for this, when the FHSS is isolated and not operating.
- Fuel burn-up measurement occurs when fuel circulation takes place.
- BUMS calibration cannot occur when the spheres are circulated. Calibration is expected to occur at a frequency of six months.
- Fresh fuel sphere loading can occur when the spheres are circulating.
- Sphere-conveying lines will be purged daily to remove loose contaminated dust and spheres before access to the compartments is approved.

2.3.2 Maintenance design principles

The maintenance strategy is that the major components will follow a strategy called ‘repaired by replacement’. The remove-and-replace maintenance strategy allows PEAs to be removed as units P[13]. Major components will be removed with special tools and immediately replaced by a spare component.

The removed component will be maintained in a dedicated maintenance area, where additional temporary shielding can be provided when necessary. This is also to ensure that FHSS operation is not interrupted due to long periods of maintenance. A target of 8 h to 12 h per day is set for sphere circulation.

Another PBMR design principle is that equipment will be located in accessible areas with appropriate servitudes. Adequate servitudes ensure maintainability and easy access into and egress from these areas. It is known that enclosed areas with small servitudes result in longer access and egress times on the plant. Small servitudes cause an increase in the time a worker spends in a mission area.

The expected maintenance tasks were grouped in maintenance similarity types. Table 2 lists the equipment maintenance types associated with maintenance on the FHSS major equipment. This table has been used to identify expected missions and tasks for work to be performed by the selected representative worker. It was an important input into creating the conceptual exposure scenarios.

Table 2: Maintenance similarity grouping

Major equipment	Equipment maintenance types							
	Valve blocks	Skids	Pipework	Vessels	Instrumentation	COTS items	Devices/ machines	Structures and fixtures
Valve inserts	x							
Blower units		x			x			
Filter units	x				x			x
Tank unloading units				x				
BUMS and Activity Measurement Device (AMD)	x				x		x	x
Piping systems			x					
Tanks				x				
Core loading system (counters and brakes)	x		x		x			
Cleaning block unit	x				x			x
CBA	x				x			x
Measurement block unit	x				x			x
Charge lock outlet unit	x				x			x
Discharge lock outlet unit	x				x			x
Heat exchangers		x		x	x			
Cables					x			
Instrumentation cupboards					x			
Core loading brake system						x		
Connectors			x		x			
Controllers and instrumentation					x			
Calibration sphere insert	x							
Gas supply block unit	x				x			x
Gas return block unit	x				x			x
Anti-surge valves						x		
FHSS pressurization valve						x		

Figure 9 is a schematic maintenance layout engineering diagram that explains the high-level major maintenance tasks and access requirements P[11]. This engineering diagram assists the reader to understand the location of the various SSC relative to each other, and the position of the SSC inside the FHSS section of the building.

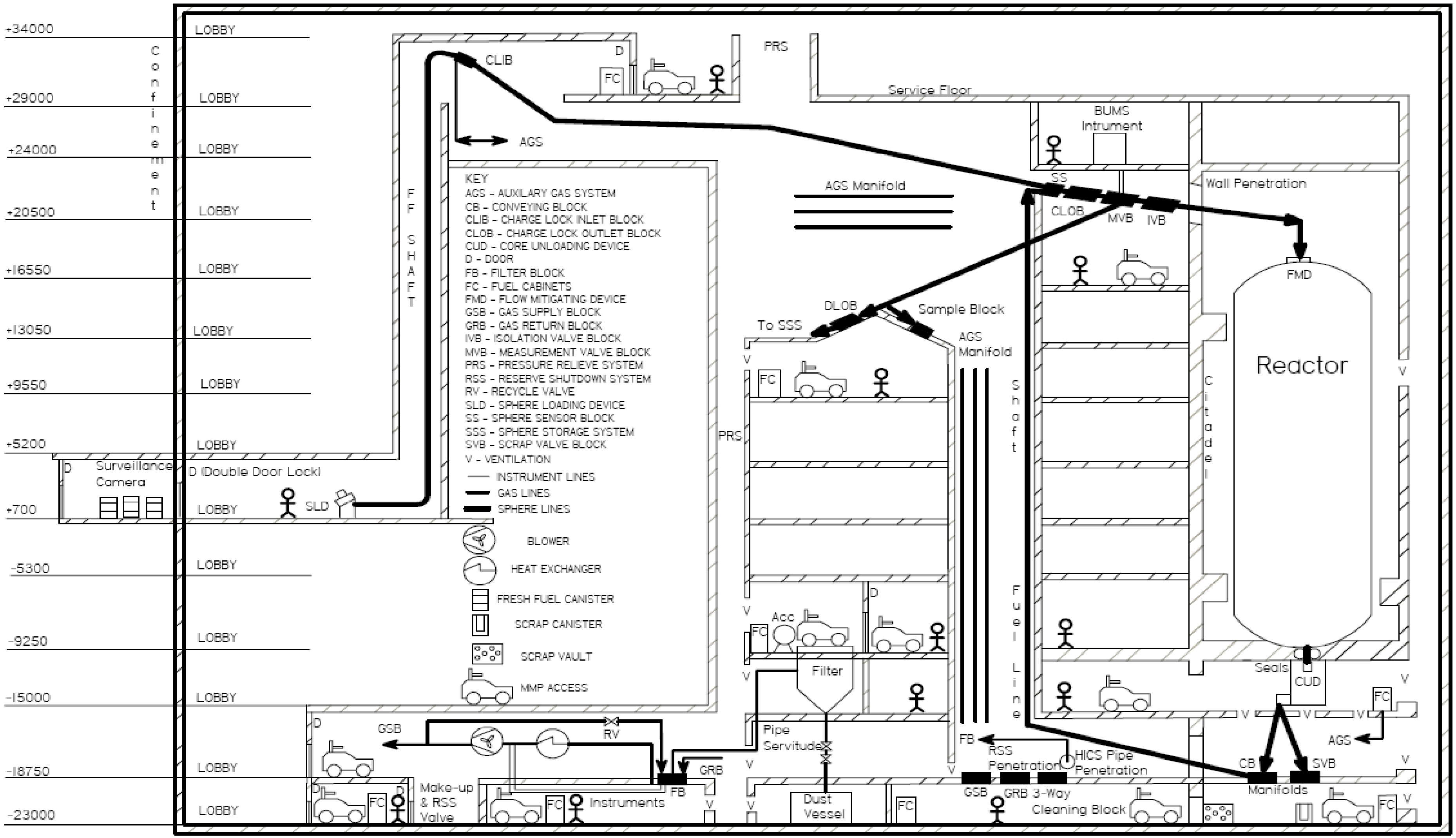

Figure 9: Fuel handling and storage system engineering diagram of maintenance layout

2.4 Fuel handling and storage system – radiation safety analysis information

2.4.1 Overview

Different radiological sources will be present in the PBMR's FHSS. Radiological sources originate from the build-up and spread of radioisotopes present in the core; fuel spheres; dust arising from the abrasion of fuel spheres and activation of construction material; coolant gas; and air. Workers may be exposed to radiation through the following exposure pathways during maintenance and surveillance activities:

- External penetrating gamma radiation (to the whole body).
- External non-penetrating beta radiation (only to exposed parts of the skin) P[14].
- Inadvertent ingestion of dust, containing alpha and beta emitters, settled on lips and nasal tissue. (Ingestion will not result in significant exposure. The wearing of personal protective equipment, specifically gloves and dust masks, is a prerequisite to entering the building.)
- Possible inhalation of and exposure to airborne noble gases and their short-lived decay products.

In operating plants, a worker's dose is measured with special radiation monitors. Personal dosimeters are used to record the dose to a worker, or radiation dose rate in which he or she works. Each radiation worker wears a dosimeter, which is read out at regular time intervals. The accrued dose is recorded in a dose record. The dose records are then used to calculate the worker's annual dose.

In the absence of dose records, a worker's exposure is estimated through the systematic collection and analysis of exposure determinants. The purpose of radiation safety analyses is to calculate (a) the magnitude and variability of radiation exposure around the different SSC that contain radioactive material, and (b) the worker's annual dose.

The PBMR nuclear engineering analyses are performed using various code systems, many of them legacy software obtained from the German HTR programme. The main software being used for PBMR nuclear engineering design and safety assessments is [38]:

- VSOP99 (neutronics calculation which includes thermal hydraulics)
- TINTE (core thermal hydraulics, transients and group kinetics)
- GETTER (metallic fission product release)
- NOBLEG (noble gas fission product release)
- SCALE (fuel depletion and ex-core criticality)
- FISPACT (material activation)

- RADAX (radionuclide and dust transport, and plate-out)
- MCNP (Monte Carlo N-particle transport code for reactor neutronics, ex-core criticality, radiation transport and shielding)
- MicroShield (radiation shielding)

The PBMR radiation source term analysis process is a complex analysis chain. It consists of various different software packages and calculation models that interactively model and analyse the source term.

Figure 10 explains the different high-level elements of the process used to analyse all the radiation sources on the plant, including those in the FHSS.

In this chapter, it is not possible to provide a comprehensive overview of all the software programmes used. Emphasis will be placed on that software used for dose rate calculations: MCNP and MicroShield. In addition to this software, only the VSOP99 code system and Origen, used to calculate the core neutronics and intermediate source term, will be briefly discussed.

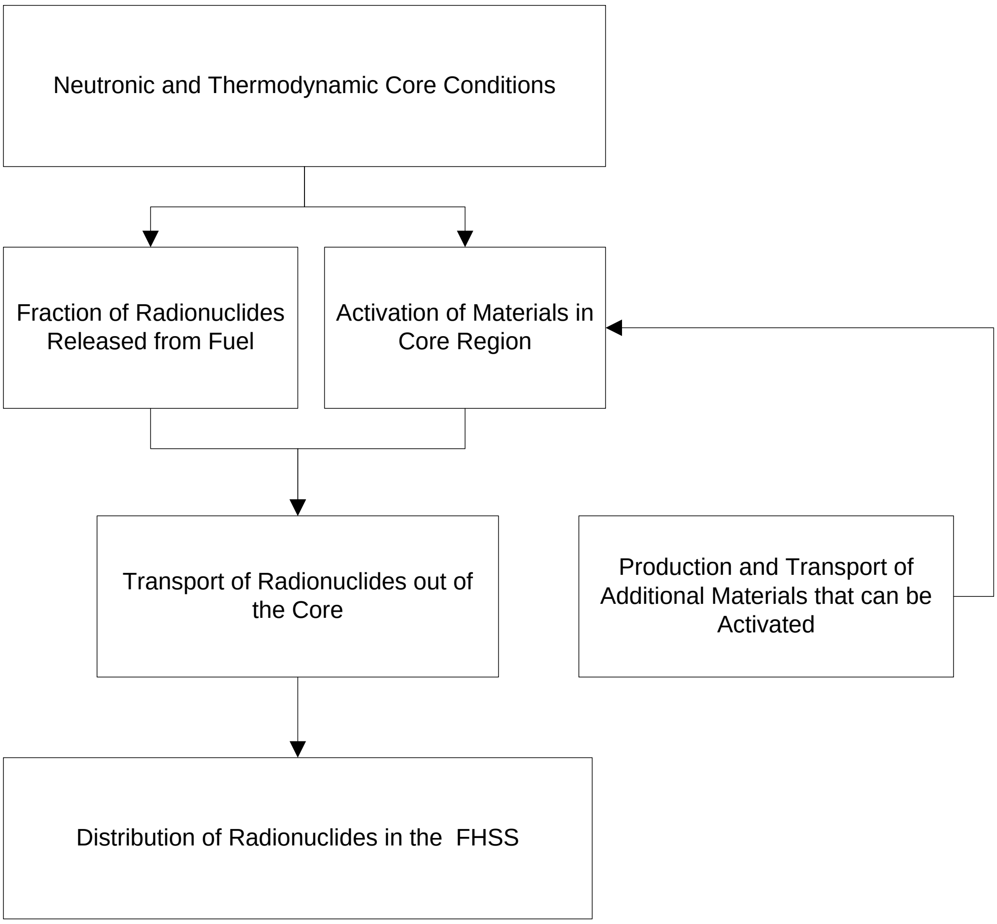

Figure 10: Overview of process for calculating radiological source term

Radiation source dose rate values are a key input parameter in the dosimetric formulas used to quantify the conceptual exposure scenarios. For this reason, more emphasis is placed on the results of the dose rate calculations. MicroShield is used to perform dose rate calculations for an SSC, which acts as a single radioactive source. Dose-rate distributions in compartments with several SSC – acting as multiple sources – will normally be performed with MCNP.

The first step was to calculate the radionuclide inventories in the fuel spheres. A study was performed to determine which radionuclides contribute to the majority of the dose. Due to time constraints, it is not possible to study the contribution of all the radionuclides. Only selected radionuclides were included to determine the dose rates P[15], P[16].

Dose rates in the vicinity of SSC have been analysed by the PBMR's safety analysis teams. Reports on these analyses document the assumptions used in calculations and calculate the associated dose rates due to radiation emitted from SSC.

2.4.2 Software used

VSOP99 code system

VSOP99 is a computer code system for the comprehensive numerical simulation of the physics of thermal reactors. Its entails the set-up of the reactor and of the fuel element; processing of cross sections; neutron spectrum evaluation neutron diffusion calculations; fuel burn-up; fuel shuffling; reactor control; and thermal hydraulics and fuel cycle cost calculations [44].

The code can simulate the reactor operation from the initial core towards the equilibrium core. The code system contains some important features required for pebble bed reactor analysis, such as treatment of double heterogeneous fuel, the different fuelling mode simulations, and two-dimensional thermal-hydraulic capabilities including the pebble bed thermal hydraulic correlations [44].

VSOP99 is a suite of codes developed over many years at the Research Centre Jülich. The VSOP99 code is used to generate the reference isotopic distribution and realistic equilibrium core conditions, i.e. temperatures and control rod positions [45].

Origen

The ORIGEN-S software was used to determine the radionuclide inventories for the fuel spheres after passing through the equilibrium 400 MWth core [46]. This is the starting point for deriving the fuel source term. This analysis included the contribution from radioactive impurities in the matrix graphite.

MicroShield

MicroShield was designed to analyse shielding and estimate gamma radiation exposure. It is a one-dimensional radiation transport code, based on geometrical input models of the radiation source. By specifying user input parameters, MicroShield can model a variety of scenarios such as P[17]:

- Source configuration
- Distance and orientation between source and receptor
- Dimensions and density of source
- Orientations of intervening shields
- Material density of shield
- Selection of build-up factor

MCNP

The MCNP code is extensively used at PBMR to quantify radiation originating from a variety of sources. The code is used to transport particles in three dimensions. The MCNP code is widely used in the nuclear industry, and is verified and validated extensively. This code was applied during PBMR design in safety analysis calculations of both radiation shielding and neutron activation of exposed material.

Calculation models are developed using the MCNP code. The MCNP code allows the use of continuous energy libraries of nuclear data and a detailed geometry description of the system modelled. The cross-section data provided with the MCNP standard distribution was used P[18].

Various radial and axial boundary sources are used, where necessary, as input for dose rate and shielding calculations. For instance, a radial boundary source at the outer wall of the reactor pressure vessel is used as the source term in dose and shielding analysis sideways of the reactor. As a result, the calculation model developed is less complex and saves calculation run-time.

2.5 Dose rate analysis results

2.5.1 Dose rates from fuel

The PBMR reactor fuel consists of low enriched uranium triple-coated isotropic fuel spheres that were developed in Germany from 1969 to 1988. The PBMR core contains approximately 451 000 fuel spheres in an annular fuel region. The reactor is continuously fuelled through loading tubes and discharge tubes. The fuel spheres are circulated several times through the core until they reach target burn-up [38].

Fresh fuel that has not been circulated through the core contains no fission products. The contribution of dose rates from fresh fuel to the radiation exposure of workers is therefore insignificant. Fresh, uncirculated fuel has been modelled using MCNP. A dose rate of 3 μSv/h is calculated on the fresh fuel spheres P[7].

The pebble fuel contains large quantities of fission products after it undergoes neutronic fission reactions when passing through the reactor core. The fuel spheres are considered to be the major radiation source in the FHSS, specifically when a sphere (or spheres) is stuck in a PEA and needs to be removed.

The half-life of many of the fission products in the fuel is < 10 days. Table 3 indicates that these short-lived isotopes have a significant influence on the plant dose rates. These dose rates decrease with an order of magnitude within the first 20 days after plant shutdown P[19]. However, it is not practical to shut down the plant and wait 20 days to perform maintenance and surveillance activities. Therefore, dose rate values used in the assessment are based on a radioactive source term that does not consider radioactive decay.

Table 3 lists the dose rate per kernel in the fuel sphere. Each fuel sphere contains a large number of these fuel kernels. The dose rate from a fuel sphere where no shielding is present will result in conditions that are not maintainable. Shielding will be provided by the concrete of the FHSS floor and wall structures.

Table 3: Calculated dose rates on kernel

Distance from source (cm)	Point source dose rate (mSv/h)		
	Initial	30 days	60 days
1	2 310	867	411
20	5,76	2,17	1,03
50	0,922	0,346	0,164
100	0,230	0,087	0,041
200	0,058	0,022	0,010
300	0,026	0,010	0,005

Dose rate calculations were performed for different numbers of spheres in SCS pipes with and without shielding P[17]. Table 4 summarises the estimated dose rates due to a different number of fuel spheres in a fuel line, without shielding. At 100 cm from an SCS fuel line, the dose rate due to one sphere is as high as 0,50 Sv/h (500 mSv/h). Maintenance in such a high dose rate area will result in unacceptably high worker dose.

Table 4: Dose rates from spheres in structure, system and component pipes without shielding

Description	Dose rates (Sv/h) at different receptor locations (cm) from pipe(s)		
	1 cm	100 cm	300 cm
One fuel sphere from one pipe	259	0,50	0,06
One fuel sphere in each of three pipes	778	1,51	0,18
Ten fuel spheres from one pipe	483	4,89	0,58
Ten fuel spheres in each of three pipes	1 450	14,70	1,74
Fifty fuel spheres from one pipe	502	16,00	2,69
Fifty fuel spheres in each of three pipes	1 510	48,00	8,06
One hundred fuel spheres from one pipe	470	19,00	4,46
One hundred fuel spheres in each of three pipes	1 410	57,90	13,40

For this reason, the specification for the thickness of the FHSS building floor is at least 1 m concrete. The concrete floor has to provide the shielding for the radiation emitted from the pile-up of spheres. Workers have to maintain the PEAs located in the valve blocks, which contain the radioactive material.

In the assessment, as discussed in Chapter 4, the dose rate due to a pile-up of 50 spheres is investigated.

2.5.2 General area dose rate levels

The dose rates on the outside of the reactor wall are calculated. Table 5 lists these dose rates P[20]. The reactor citadel wall is designed to be 2,5 m thick and reduces the dose rate from the reactor pressure vessel side to the FHSS side of the building.

Table 5: Calculated dose rates behind citadel wall (400 MWth)

Parameter	Dose rate (μSv/h)
Neutron	0,002
Gamma	8,70
Total	**8,70**

The calculated dose rate directly outside the citadel wall that shields the reactor core is 8,70 μSv/h. A decrease of the radiation dose rate due to distance from the citadel wall, and additional attenuation provided by other structural walls and SSC, were not credited in this assessment. In the operating plant, additional shielding by other walls will significantly reduce the general area dose rates.

In the assessment, different sensitivity analysis cases were performed to investigate the variation in area dose rate values. In sensitivity analysis case 1 – paragraph 4.4.2, local area dose rates are also set equal to general area dose rate values. This enables the evaluation of the effect on the results of varying only general area dose rate values by 2,00 µSv/h, 5,00 µSv/h and 10,0 µSv/h.

In the reference case values listed in Table 6, a 5,00 µSv/h general area dose rate is used. A design specification of 2,00 µSv/h is identified in Chapter 4 for the general area dose rate. However, to introduce conservatism in the reference case, a 5,00 µSv/h dose rate is used in the calculation.

The local area dose rate level should be distinguished from general area dose rates. The general area dose rate should be distinguished from background dose rates, which usually refer to the background level of radiation in the natural environment which surrounds us at all times.

2.5.3 Fuel handling and storage system – dose rates of structures, systems and components

Valve blocks

Valve blocks contain the PEAs that distribute, divert, collect and measure burn-up, and route spheres and helium gas. Different analyses were performed to estimate the dose rates in these compartments. Valve blocks were modelled using MCNP and MicroShield P[21], P[22], P[23]. The worst-case scenario is used, which assumes that a stuck sphere in a PEA results in a pile-up of fuel spheres in the pipeline.

Valve block openings will exist in the floors of the building. It is assumed that a pile-up consists of a maximum of 50 of these spheres. These might not be adequately shielded in the opening between the valve block body and shielding floor. This analysis is only applicable to the following four valve blocks, where a pile-up of fuel spheres can occur. A pile-up of fuel at other valve blocks is not possible due to the inclination of the pipelines.

- CBA
- Measurement Block Assembly (MBA) or BUMS
- Isolation block assembly
- DLOB assembly

The maximum value (80,0 µSv/h) calculated at the bottom of the heavy concrete block is used for the purpose of calculating the PEA replacement in the reference case. Annexure B provides details on how the calculation is performed.

It is assumed that this value represents the dose rate to which a worker will be exposed when replacing a PEA. A value of 80,0 µSv/h will be used to represent dose rates in the whole area.

Gas circulating subsystem sources

The possible sources of radiation in the GCS are the filters, filter valves and dust canister. It is expected that several kilograms of dust will be generated in the GCS and that this dust will be significantly contaminated with radionuclides. The GCS filter and GCS canister are located in compartments that will be classified as exclusion areas. These will not be accessed during routine maintenance activities.

The GCS contains the following valve blocks: the gas return block, cleaning block and gas supply block. Although they will be maintained routinely, significant contamination is not expected in these valves blocks. All the valve blocks are also designed to provide the required shielding, should contamination occur.

In the assessment, the dose rate in this area is assumed to be the same as the general area dose rate.

Sphere replenishment subsystem sources

The SRS handles and stores the fresh fuel canisters and introduces the fuel spheres into the SCS conveying lines. These spheres have not been circulated and irradiated in the core, and therefore do not contribute to exposure of workers. The low dose rates on these fuel spheres (3 µSv/h) will be sufficiently shielded by the fresh fuel canisters. The dose rate in this area is assumed to be the same as the general area dose rate.

Auxiliary gas subsystem sources

The AGS is isolated from the GCS and filled with uncontaminated gas. Mixed gas/air volumes are extracted via the AGS and discharged via the heating, ventilation and air-conditioning system. The dose rate in this area is assumed to be the same as the general area dose rate.

Waste handling and storage system sources

The Waste Handling and Storage System (WHSS) is a subsystem of the FHSS that removes damaged fuel spheres, samples and contaminated dust from the SCS and GCS. The sphere scrap is accumulated in batch canisters within the pressure boundary and unloaded under atmospheric conditions.

The scrap canisters are mechanically handled and stored in an intermediate storage vault under shielded and contained atmospheric conditions. Dust from the filter is discharged after FHSS isolation as a process operation under atmospheric pressure, and stored in a life capacity container, also under shielded and contained atmospheric conditions.

All waste canisters are designed to ensure low dose rates on the surface in accordance with the transport regulations. If these canisters are temporarily stored in the FHSS building, they will be shielded behind temporary shields. Temporary shielding will be provided to ensure that local dose rates are < 5,00 μSv/h.

The waste storage area will be classified as an exclusion area and routine access to the waste storage area will not be required. This source is not included in the conceptual exposure scenarios of the worker dose assessment.

Sphere storage subsystem sources

The SSS of the FHSS provides a facility for the storage of spent fuel, used fuel and graphite spheres. It is a separate facility from the WHSS. This facility is contained within the module building and has sufficient capacity to store all the used and spent fuel produced throughout the life of the plant.

The SSS allows for the management of irradiated fuel and graphite sphere inventories. SSS tank areas will be classified as exclusion areas. Access to the compartments containing the SSS tanks will not be allowed during routine operations at normal power operations.

However, access will be required to the SSS service hall located above the SSS tank areas and separated by a shielding floor from the tank areas. A number of analyses have been performed on radiological conditions in the service hall area. It is clear from the results of these analyses that the shielding floor provides adequate shielding. Even without any water around the spent fuel tanks, the dose rate in the service hall above the SSS tanks is estimated to be < 0,300 μSv/h above the general area dose rate radiation levels P[24], P[25].

The dose rate in this area is assumed to be the same as the general area dose rate in the assessment.

Summary of dose rate values used in worker dose assessment

Chapter 4 discusses the worker dose assessment performed on the FHSS. Chapter 4 also contains several tables used to calculate the dose due to a worker's exposure to the conceptual exposure scenarios. Table 6 summarises the dose rate values used for maintenance and surveillance of different SSC in the FHSS (reference case).

These values are obtained from the safety analyses discussed in this chapter. This information helps the reader to see which dose rate values were used in the calculation sheets to estimate the annual worker dose. Table 6 includes the Computer-aided Design (CAD) drawing room compartment numbers, indicated per floor level inside the FHSS building. The room description and associated contents of the compartment number are also provided.

Furthermore, the last column in Table 6 indicates whether or not this compartment will be an exclusion area. This helps the reader to know if access into this compartment will be required during normal plant operation.

Table 6: Summary of dose rate per fuel handling and storage system compartment (reference case)

Floor level	CAD compartment number	Compartment name	Identified content	Dose rate (µSv/h)	Comment
-23 000	069725	Dust storage tank compartment	Dust storage tanks	n/a	Exclusion area.
-23 000	069702	Scrap fuel storage area	Scrap fuel loading equipment	n/a	Exclusion area.
-23 000	087774	GCS valves	GCS valves and lines	5,00	General area dose rate used in calculation.
-23 000	069724	Gas supply unit	Gas supply unit Cleaning unit Gas return block AGS CBA Gas blower unit Filter block unit	5,00 5,00 5,00 5,00 80,0 5,00 5,00	General area dose rate used in calculation, except for location at conveying block.
-23 000	102830	Spent fuel storage (wet) tank area	Spent fuel tanks	n/a	Exclusion area.
-23 000	069699	Spent fuel storage (dry) tank area	Spent fuel tanks	n/a	Exclusion area.
-23 000	069700	Spent fuel storage (dry) tank area	Spent fuel tanks	n/a	Exclusion area.
-18 800	069723	GCS valves and fuel lines	AGS SCS sphere line 1 + 2 + 3 GCS gas pipes Gas return block unit Gas blower unit Cleaning block unit Filter block unit Gas filter unit Gas supply block unit	5,00	General area dose rate used in calculation.

Floor level	CAD compartment number	Compartment name	Identified content	Dose rate (µSv/h)	Comment
-18 800	071851	CUD, CBA and fuel lines	HLWHS – high-level waste handling system SCS sphere line 1 + 2 + 3 GCS gas pipes CUD CBA	n/a	Exclusion area.
-18 800	071844	Dust filter and storage valves compartment	Filter and dust storage connection pipe	n/a	Exclusion area.
-15 000	069721	FHSS and CUD maintenance area	CBA and lines	n/a	Exclusion area.
-15 000	071848	CUD maintenance access area	CUD maintenance access	5,00	General area dose rate used in calculation.
-15 000	071517	GCS filter compartment	Filter and pulsed dust level inside filter	n/a	Exclusion area.
-9 250	069720	FHSS SSC	Gas filter unit Sphere replenishment system AGS GCS gas pipes SCS sphere pipes	5,00	General area dose rate used in calculation.
-9 250	069958	Reactor cavity	Reactor pressure vessel	n/a	Exclusion area.
-9 250	102835	Spent fuel storage (wet) – high	Spent fuel tanks	n/a	Exclusion area.
-9 250	102835	Spent fuel storage (wet) – low	Spent fuel tanks	n/a	Exclusion area.
+9 550	069831	Valve maintenance area	Gas filter unit	5,00	General area dose rate used in calculation.
+9 550	091974	SFS maintenance area	Spent fuel tank valve maintenance area	5,00	General area dose rate used in calculation.
+9 550	071617	Spiral floor	Maintain discharge outlook block from bottom	80,0	Used in calculation.
+14 400	069716	Valve maintenance area	Discharge lock outlet unit AGS	5,00	General area dose rate used in calculation.

Floor level	CAD compartment number	Compartment name	Identified content	Dose rate (µSv/h)	Comment
+14 400	071617	Spiral floor	Maintain charge lock from bottom on this floor	80,0	Used in calculation.
+18 200	090178	FHSS valve maintenance area	BUMS Isolation valve block SCS sphere pipes GCS gas pipes AGS	80,0 80,0 5,00 5,00	Used in calculation. Used in calculation. General area dose rate used in calculation.
+18 200	071617	Spiral floor	Maintain BUM valve block from bottom SCS sphere pipes GCS gas pipes AGS	80,0 5,00 5,00 5,00	Used in calculation. General area dose rate used in calculation.
+24 400	069710	Burn-up measuring system	BUMS	80,0	Used in calculation.
+29 000	069711	Fresh fuel sphere replenish	Sphere loading device	5,00	General area dose rate used in calculation.

2.6 Conclusion

Chapter 2 has identified and summarised the most important SSC expected to be maintained and surveyed. A group of maintenance tasks has been identified to develop the conceptual exposure scenarios. For each task, the annual frequency and task duration have been determined. This is based on available plant and maintenance design information P[4], P[5].

The selected exposure scenarios to be analysed are from maintenance and surveillance tasks on the following FHSS subsystems:

- SCS
- GCS
- SRS
- SSS
- Instrumentation equipment
- Scrap sphere sampling

This chapter has summarised the dose rate results of the calculations performed on the various SSCs in the FHSS. These quantitative dose rate results are an important input into the dose formula used in this worker dose assessment.

Justification has also been provided for the classification and identification of compartments as exclusion areas. These compartments will not be included in the dose assessment, because they will not be routinely accessed. During operation, special access control measures will be implemented in these exclusion areas to ensure minimised worker dose.

It is necessary to make a number of assumptions in order to implement the IAM. This is to ensure that the results of the assessment are binding. Therefore conservative assumptions are made related to the radiation source term and occupancy factors. These assumptions are described in detail in Chapter 3.

CHAPTER 3: RESEARCH METHODOLOGY

Chapter 3 discusses the development of the integrated assessment and new simplified integrated assessment model.

Chapter 3: Research methodology

3.1 Introduction

Chapter 3 explains how the IA and IAM were developed to perform a worker dose assessment during the design of an NPP. This IA combines methods used in performing public dose assessments, dose reconstruction and worker dose assessments. The IA is based on estimating the dose received by a hypothetical worker, defined as the representative worker.

The development of the IAM is based on the knowledge that a worker's annual dose is the sum of the doses received during all the missions performed on the plant, for a year. A worker's dose will be accrued due to the time they spend in a radiation field (also referred to as occupancy factors) and the dose rate resulting from the radiation field in which they work.

The dose received during each mission that a worker has to perform on the plant consists of:

- the dose received when accessing the area;
- the dose received when performing the necessary functions; and
- the dose received by exiting the area.

This dosimetric formula includes the exposure determinants as parameters. The results obtained from the dose formula are a quantitative estimation of the annual dose a worker will receive on the operating plant.

An exposure scenario is developed for each representative worker. These scenarios are developed by identifying a set of missions associated with each representative worker. These are considered to be a representation of the exposure scenarios that are expected to exist on the operating plant. The exposure scenarios are compiled by reviewing the available maintenance design information and the radiation safety analyses of the plant SSC.

An overestimation of the doses will create a design burden by requiring more shielding than necessary for the plant, or more RP controls, which will increase the operating cost of the plant. An underestimation might result in workers being unnecessarily exposed, which could have detrimental health effects. Different exposure scenarios have to be developed for each occupancy category. This could only be achieved by making a number of both qualitative and quantitative assumptions.

These assumptions consider engineering design characteristics, maintenance design characteristics, source term characteristics, radiation safety analyses and mathematical modelling. Assumptions such as maintenance duration; building layout and servitudes; travel time; access to exclusion areas; removal of stuck spheres; and the selected risk-dominant occupational categories are described further in detail.

3.2 Dose assessment methods

3.2.1 Methods used from public dose assessments

Defining representative worker

The analysis of annual dose to PBMR workers corresponds to prospective dose assessments [26]. It estimates the potential exposure of a hypothetical worker on the future operating plant. In these assessments, assumptions are made about the future operating plant, based on hypothetical individuals and exposure scenarios. In public dose assessments the assessment of the dose to a representative worker relates to the fact that it is impossible to obtain habit details for each member of the public [27]. Similar, the purpose of defining a representative worker is to allow for design and maintenance uncertainties between different occupational groups. This hypothetical or representative worker is allocated a selection of representative missions in order to develop the exposure scenario to be analysed for a specific occupational group.

A hypothetical worker is therefore defined and the corresponding radiation dose is estimated, based on theoretical exposure scenarios. The estimated doses calculated for the hypothetical workers are considered to be representative and realistic of the expected dose that workers on the operating plant will receive.

In selecting the exposure scenario for the representative person, the analyst must ensure reasonableness. Reasonableness implies that the exposure scenario is realistic and is not outside the range that is encountered in everyday life. One of the challenges in this study was to achieve a fair balance between realism and simplicity [27]. For this reason, a large number of assumptions were made.

Establishing reference groups

Another important concept adopted from public dose assessments is that of reference groups in the population. Reference groups are intended to be representative of those groups of people in the population who receive the highest doses [27].

In specifying reference groups for public dose assessment, two broad approaches are possible:

- The first approach involves carrying out surveys of the local population to determine their habits, where they live, etc. From these surveys, the people who are receiving the highest doses can be identified.
- The second approach involves using more generalised data to establish generic groups of people who are likely to receive the highest doses [27].

In operational NPPs, workers are generally grouped according to occupancy category when dose assessments are performed or reported. Similarly, in this study, the representative worker will be an individual from the occupancy categories expected to receive the highest annual dose [26].

Provisionally, the following occupancy categories (reference group) have been proposed for the PBMR plant:

a. Routine operations (reactor operations, RP surveillance and fuel operations).

b. Maintenance (preventative and corrective).

c. In-service inspection.

d. Waste processing.

e. Technical surveillance and tests.

f. Post-commissioning, modifications and upgrades.

Only three of these categories have been selected for assessment purposes. Two occupancy categories expected to result in the highest radiological exposures were selected and analysed. These two cases will represent the upper limit annual worker dose. One occupancy category was selected to represent a more realistic annual worker dose.

3.2.2 Methods used from dose reconstruction methodology

This required that a systematic collection and analysis of plant design information be performed to determine the expected radiological conditions on the plant. Information on occupancy factors and potential radiation sources has to be collected. Dose-significant tasks are determined by the dose rate and time associated while performing the task. This has to be considered for each of the selected occupancy categories [14].

Occupancy factors are determined by evaluating the expected tasks associated with maintenance and surveillance missions on the plant. Only those tasks that are potentially dose-significant are analysed in detail. This information is collected from the design documents and maintenance support concepts are provided by design engineers. The information is obtained by combining and integrating design and analysis information from diverse fields of study.

Radiation sources are identified and characterised in terms of SSC location. The source term values of these sources are used to calculate local dose rates expected in the vicinity of the source. Dose rates were calculated separately for different SSC located on the plant and are discussed later in this study. The emphasis was on the maintenance of the PEAs located in the valve blocks.

PBMR plant information is discussed in more detail in Chapter 2.

3.2.3 Methods used for worker dose assessment

The simplified dosimetric formula developed and its associated parameters form the basis of the dose calculations in this study.

The annual dose to the worker is expressed through the following simplified formula. The formula demonstrates the relationship between the variables dose, dose rate and time.

$$\text{worker dose} = \text{dose rate} \times \text{time exposed in radiation field} \quad (1)$$

For the purpose of this study, the SSC dose rates are assumed to be equal to the local area dose rate in which this SSC is installed.

The other input variable is the expected time spent in each radiological area accessed, or occupancy factor. This is calculated from the frequency and time to perform the maintenance and surveillance tasks. This was obtained from analysing the predicted maintenance and surveillance tasks.

The proposed dosimetric formula to quantify the representative Worker Annual Design Dose (WADD) is:

$$WADD = \sum_{i=1}^{N} FQ_i \times [(TT_i \times DT_i) + (ST_i \times DA_i)] \quad (2)$$

Where:

WADD = worker annual design dose (mSv/a)

N = number of missions performed per annum

FQ_i = annual frequency of access to perform mission i (number/a)

TT_i = travel time to the compartment where mission i will be performed (h)

DT_i = maximum dose rate incurred during travel to the compartment (mSv/h)

ST_i = stay time in the compartment to perform mission i (h)

DA_i = dose rate in the compartment when mission i is performed (mSv/h)

This formula calculates the annual cumulated dose received by a worker belonging to a specific occupational category. For the purpose of this study, three occupational categories are selected. This selection is discussed in paragraph 3.3.5. This formula includes the dose a worker receives from travelling to and from the compartment where the task is to be performed, and the dose received while performing the task.

The dose received while travelling is captured in the following parameters:

- Travel time includes travel to and from area (TT).
- Dose rate during travelling (DT).

The dose received while performing the task is captured in:

- Stay time parameter (ST).
- Dose rate in the compartment (DA).

The assessment is based on expected tasks to be routinely performed during a normal power operational year. The parameter (FQ) accounts for the expected annual frequency of this task that will be routinely performed.

Figure 11 is the process diagram used to implement the IA and IAM, collect the required input data and evaluate the results.

The selected tasks will vary for the different occupancy categories selected. Dose rate information is provided from dose rate analysis studies on the different SSC and maintenance concept evaluations.

Figure 11 illustrates the optimisation by the iterative loop that follows the derivation of the WADD.

The proposed IAM allows for a theoretical, quantitative estimation of the cumulative annual dose to a representative worker. The results obtained from implementing this formula are applicable to the specific iteration of the design baseline analysed.

It is further recognised that when demonstrating compliance with the design targets, conservative assumptions should be made for the location and duration of the exposure [20]. The identification of design target values to optimise the design will be based on dose constraint values, discussed later in this chapter. This is necessary when using the results of the worker dose assessment as justification that the plant design meets international requirements on dose limits.

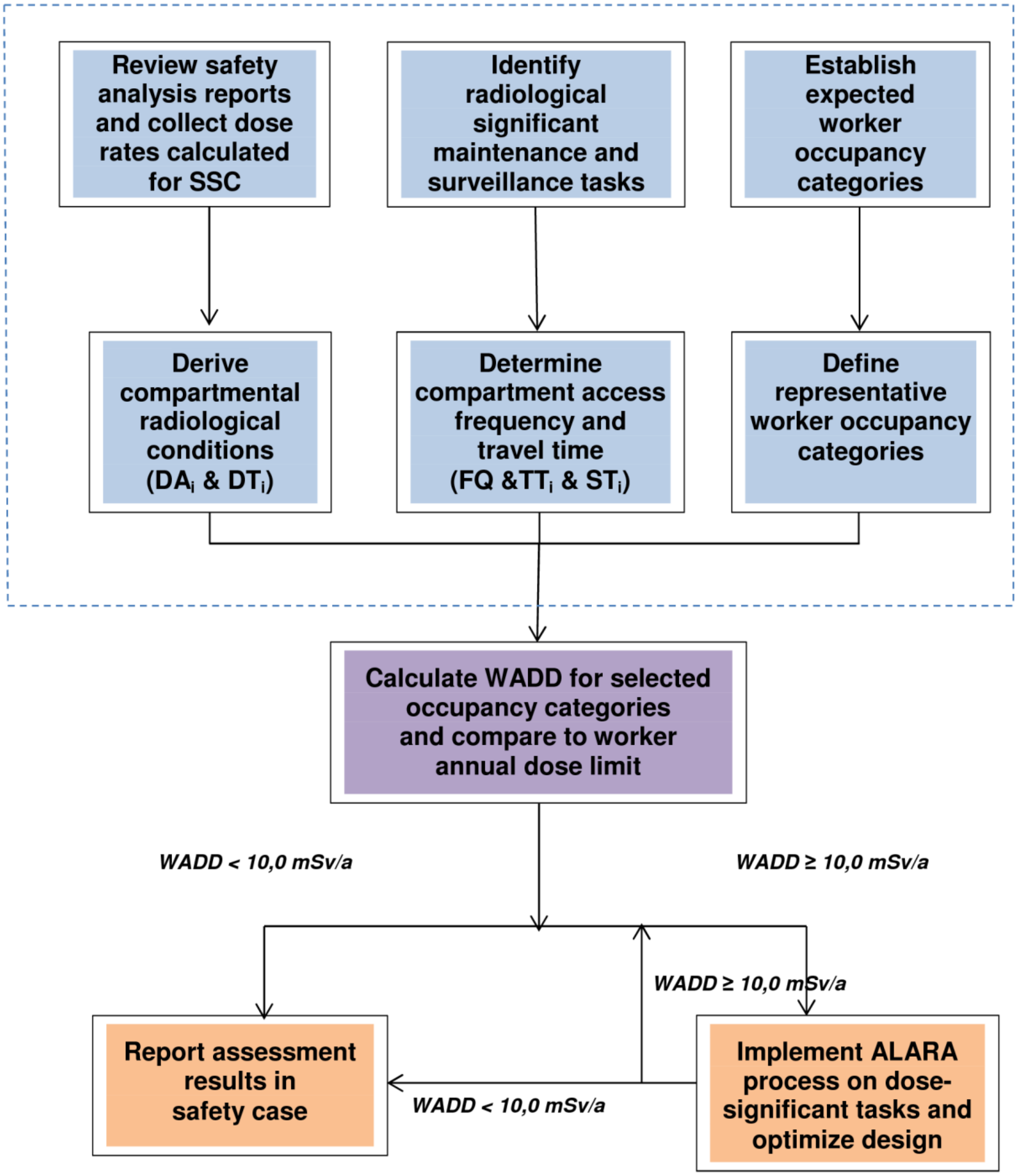

Figure 11: Process diagram of integrated assessment and integrated assessment model used to calculate worker annual design dose

3.3 Assumptions

3.3.1 Limited scope – field handling and storage system

This assessment is limited to only the FHSS building during normal plant operation. The reasons for this are listed in paragraph 2.1.

Only the exposure of plant personnel during normal power operation is considered. Maintenance periods scheduled during plant shutdown are excluded from this assessment, since the radiological environment will change significantly during open maintenance compared to power operating conditions.

During maintenance, the reactor will be shut down, as it will be necessary to open the systems to access component internals. Access to exclusion areas will be required during these periods. This is discussed in more detail in paragraphs 3.3.3 and 3.3.4. Analyses of exposure during maintenance periods or during abnormal or accident conditions will be treated separately in a different assessment. However, the IA and IAM proposed in this study will also remain valid for assessment of maintenance periods.

The documents referenced were compiled based on the 400 MWth demonstration power plant design. The 400 MWth source term analysis was the most comprehensive source term analysis information available when the assessment was performed. It should be recognised that the source term calculations are design specific. In the case where the plant design changes to a different power level, the source term has to be recalculated. This will have an impact on all dose rate results calculated.

3.3.2 Exposure scenario development – task selection

Table 2 has been used to identify expected missions and tasks for work to be performed by the selected representative worker and was used to develop the conceptual exposure scenarios.

Maintenance of the following major equipment was included in the conceptual exposure scenarios:

- Valve inserts
- BUMS
- CBA
- Measurement block unit
- Charge and discharge lock unit
- Instrumentation cabinets
- Gas supply and return block unit

3.3.3 Exclusion of dust component

Most of the dust originates in the reactor core during fuel circulation due to abrasion of the fuel spheres and graphite reflectors. The dust will contain high levels of radioactive contamination, as it will act as a sink for many of the radioisotopes. It will also absorb many of the released activation and fission products that exist in various chemical forms. The GCS and AGS will be used to purge the FHSS pipes of the loose dust from the system.

At the time of this assessment, the amount of dust generated was not quantified. Extensive research on how to quantify the amount of dust generated is still ongoing. In the design of the FHSS, specific features were included to remove loose dust (called purging). One of the design requirements of the FHSS is that at least 90% of the graphite dust with particle size > 0,1 µm will be removed from the FHSS circulation during normal operations P[4]. For this reason, effective purging is credited in the development of the normal operation scenarios.

However, the accumulation of dust in scrap sphere canisters and filters will present a significant radiological hazard. Access to these areas will therefore be restricted and they will be classified as exclusion zones. This will have to be addressed in the assessment of plant shutdown conditions.

The FHSS will have the capability to remove stuck or broken spheres, fuel kernels and dust from the conveying lines in the SCS. This function will be performed by the GCS. This is achieved through automated high-velocity gas-flow release of stuck spheres. Reverse gas flow will also be able to dislodge a stuck sphere and remove loose dust.

The SCS conveying lines will be operated at a slightly higher pressure than the pressure in the core and the MPS. This 'over-pressure' will prevent a flow of gas and dust into the FHSS from the core. This process will reduce the risk of dust and particle contamination in the SCS conveying lines.

The FHSS maintenance is also designed to introduce a dust-tight, remove-and-replace philosophy. This is to ensure that loose radioactive graphite dust remains confined, to prevent contamination, contamination propagation and the inhalation thereof by personnel when components are removed P[2].

It is also expected that the contribution of radiological exposure due to dust accumulation will be negligible compared to the dose rates resulting from a pile-up of fuel spheres. In the assessment of normal plant operating conditions, the contribution of dust to the total accumulated dose is therefore excluded.

3.3.4 Exclusion areas

Only the compartments that require access during normal power operations will be considered in this dose assessment. It is reasonable to assume that areas classified as exclusion areas will only be accessed during plant shutdown states. Therefore the dose received due to access to exclusion areas was not considered in the calculation. Examples of exclusion areas are the CUD inside the SSS storage tank area and the dust filter canister compartments. Also refer to Table 6, which identifies the location of exclusion areas in the FHSS building.

In operational plants, additional RP requirements are implemented before access to exclusion areas is allowed.

3.3.5 Selected occupational categories

The process of selecting the three occupational categories is based on a task analysis. Three occupational categories have been selected:

- Maintenance mechanical technician: valves.
- Maintenance electrical technician: I&C.
- Routine operations: RP.

Workers from the two occupational categories – maintenance mechanical technician: valves, and routine operations: RP – are expected to spend the longest time periods on the plant in the most stringent radiological conditions. Therefore these two occupational categories were selected as bounding cases representing the highest worker annual dose. Annual worker dose results for these two categories will be considered as upper bound values.

Workers from the maintenance electrical technician: I&C group are expected to be significantly less exposed. This group was selected to be representative of a more realistic exposure scenario. It includes access to a large number of low dose rate areas.

It is recognised that high annual worker dose is associated with in-service inspection workers performing non-destructive testing. PBMR's strategy is to perform most of these non-destructive tests using remotely operated equipment on the FHSS side of the building. This is necessary due to difficult access and the number of pipes located in the pressure relief system servitudes where pipes will be located. It is therefore expected that in the case of the PBMR, this group will not necessarily be the most exposed.

Maintenance mechanical technician: valves

It is expected that this occupational category will represent an upper limit dose value, due to the responsibility of removing and replacing PEAs and possible stuck spheres. This is a time-consuming operation while high dose rates are present. Therefore, both the occupancy time and local area dose rate of this task are considered to be upper limit conditions. This worker will also be responsible for routine inspection and the maintenance required for mechanical components throughout the operation of the power plant.

This group of workers will be also spend time in areas where radiation is not present. This is discussed in paragraph 3.3.

Maintenance electrical technician: instrumentation and control

This occupational category will be representative of workers performing routine inspection and maintenance work in high- and low-risk radiological environments. Routine inspection and maintenance are required for electrical components throughout the operation of a power plant. This group includes I&C support to the PEA maintenance missions.

It is expected that these workers will receive a lower dose than the maintenance mechanical technician: valves occupancy group. More time will be spent on performing maintenance on I&C equipment, located in low dose rate areas, which corresponds to general area dose rates. The FHSS layout is designed in such a manner that I&C components will be installed in low dose rate areas.

This group of workers will be also spend time in areas where radiation is not present. This is discussed in paragraph 3.3.6.

Routine operations: radiation protection

This occupational category represents the routine area radiation surveillance workers. Routine surveillance has to be performed in all accessible areas. This is to determine work area radiological conditions with appropriate radiation detection survey meters. The surveillance includes pre-job surveys and routine area surveys.

Changing plant conditions and movement of possible radioactive sources, such as dust, could result in increased radiation fields. In the FHSS it will be possible for hot particles (which could include kernels from failed spheres) to be deposited and to change the existing radiation fields. Therefore all radiation survey information has to be frequently confirmed and updated. This will require a high frequency of access to all FHSS plant areas.

This group of workers will be the only workers that are expected to spend all their working hours in areas where radiation is present.

3.3.6 Occupancy factors

Time spent in mission areas (excluding travel time)

In this assessment, it is assumed that workers will spend 2 000 h during the year at work. This is calculated by assuming 40 h spent at work per week and a 50-week working year. Furthermore, it is assumed workers spend two-thirds of their time performing work.

The other third of their time will be spent on activities such as daily plant feedback meetings, eating, drinking and dressing in the RP facilities. From this it can be calculated how many hours per annum (h/a) are spent performing work:

$$2/3 \times 2\ 000 \text{ h/a} = 1\ 333 \text{ h/a} \tag{3}$$

It should be recognised that the different occupational categories as identified in paragraph 3.3.5 will not spend 1 333 h/a carrying out work in compartments where radiation is present. It is reasonable to assume that the 1 333 h/a will be divided into approximately 50% radiological work and 50% non-radiological work.

It can be justified from operating experience on conventional NPPs that a significant portion of work will be performed on support equipment such as buffer and cooling circuits, where no radioactive material is present. This equipment will be located in the auxiliary buildings. Work performed on this support equipment does not contribute to the annual dose of the worker.

It is therefore assumed that a representative worker will spend approximately 670 h/a performing work on dose-significant SSC, and the other 660 h/a in areas where dose rates are similar to the background outside the plant. This 660 h/a will be allocated to maintenance mechanical technician: valves and maintenance electrical technician: I&C.

It is assumed that radiation surveillance workers will spend all their working time in mission areas. The total of 1 330 h allocated to this worker represents work performed in areas where elevated radiation levels are present. This work will consist of job coverage and routine area surveys. Routine area surveys are necessary to monitor if radiological conditions on the plant remain the same.

Building layout

Safety measures have been specified and integrated into the design and layout of the FHSS building. It is assumed that these safety measures will be implemented correctly when the plant is constructed.

For this reason, the chosen method of calculation of travel speed and expected maintenance time per task was similar to that used in other NPPs. Restrictions due to a small space in which to walk or work were not considered to impact these values.

These safety measures can be summarised as follows:

- Additional space is allocated for maintenance areas beneath the valve block units, which are embedded in the ceiling of the corresponding floor levels. These block units are specifically designed for shielding purposes.
- All sphere and high-pressure gas lines are routed within the pressure relief system servitudes of the building. This provides radiation shielding and serves as a channel that routes high-pressure radioactive gases for safe pressure relief.
- The building structure and block units provide the bulk of the radiation shielding. Consideration was given to the prevention of hot spots in the design of the valve blocks.
- The storage areas are separated from the rest of the FHSS. Special additional shielding was provided in canister storage areas.
- Additional maintenance laydown areas are included in the building design. Maintenance on PEA elements with stuck spheres will be performed remotely in these areas.

Access routes and estimated travel times

One of the organisational units within the plant system's engineering group is the Human Factors Engineering (HFE) group. This group provides requirements for access routes, access servitudes, maintenance servitudes, maintenance areas and accessibility of SSC. If these ergonomic factors are properly considered, the tasks performed by personnel can be performed effectively, efficiently and safely.

The HFE group was also responsible for ensuring that the specific guidance provided by the IAEA was considered in the design. More specifically, one of the IAEA requirements is that the length of personnel routes through plant areas should be minimised. This is to reduce the time spent in transit through these areas. In addition to this, access routes to areas of lower radiation should not pass through areas of high radiation [11].

In the evaluation of the FHSS design, it was clear that the HFE group ensured sufficient access servitudes to the various maintenance locations, storage areas and operational support locations. Sufficient space was allocated for the movement of equipment and personnel. An equipment and personnel hoist within a concrete shaft to provide shielding against radiation will provide access to the different floors.

The HFE group made assumptions to estimate the distance and time travelled by personnel for each of the access routes to the FHSS compartments P[26]. This was used as input to estimate travel times in the building to different maintenance locations. The example provided in Annexure A explains how travel time was estimated in this research. The travel times range from 1,39 min to a maximum of 3,15 min. These calculations include travel to the SSC and back to the non-radiological area in the building P[26].

For the purpose of simplifying the calculation, a maximum travelling time of 3,15 min is used in all calculations. Table 7 summarises the calculated travel times to SSC.

Table 7: Summary of calculated travel times to structures, systems and components

Level	Compartment number	Distance travelled (m)	Total travel time to SSC (min)
-23 000	069702	44,1	3,02
-23 000	087774	21,9	2,28
-23 000	069724	16,4	2,09
-18 800	069723	20,2	2,02
-18 800	071851	54,6	3,15
-15 000	071844	55,0	2,96
-15 000	069721	18,7	1,75
-15 000	071848	55,0	2,96
-9250	069720	18,7	1,45
9 550	069831	18,7	1,39
700	091974	51,6	2,51
Spiral floor	071671	34,2	2,16
14 400	069716	15,7	1,55
18 200	090178	22,0	1,95
24 400	069710	20,5	2,23
29 000	069711	26,6	2,67
		Maximum travel time (min)	**3,15**

3.3.7 Dose rate – mission – process element assembly maintenance

At the time of this assessment, only the dose rate analyses of a pile-up of spheres present during PEA maintenance missions were available. Radiation safety analyses of the normal operational conditions were scheduled to be performed at a later stage in the plant development. The inclusion of this mission in the exposure scenario ensures that the results represent upper limit values.

However, PEA removal when a stuck sphere is present will be classified as a corrective maintenance action and not as routine maintenance performed during normal plant operation. It has been mentioned in Chapter 2 that the sphere lines are designed to ensure that the probability of stuck fuel spheres is low. The reader should consider PEA removal when a stuck sphere is present as a worst-case maintenance scenario.

The performance of tasks associated with PEA maintenance, while a pile-up of spheres is present, introduces high area dose rates (up to 80 µSv/h on contact) into the calculation P[21]. It is assumed that the maximum dose rate reported in the analyses is present in the whole compartment where this task will be performed. Annexure B provides detail on this calculation.

In Chapter 4, a sensitivity analysis is performed that demonstrates the effect of a variation in local area dose rate for the three sensitivity cases: 20,0 μSv/h, 40,0 μSv/h and 80,0 μSv/h.

Number of fuel spheres stuck in fuel line

It is assumed that up to 50 spheres can be present in a fuel pipeline at a valve block when a fuel sphere is stuck in a PEA. This is the maximum number of fuel spheres that will not be adequately shielded by the gap between the valve block and where the shielding floor starts.

Shielding of local hot spots

It is routine RP practice to perform extensive monitoring of the building during the start-up of the plant when fuel is circulated for the first time. The task of the RP group is to identify localised areas of streaming called hot spots. If such localised hot spots occur, they will be identified. These hot spots are then plugged by inserting appropriate shielding material.

3.4 Ensuring conservative assessment results

One of the most important contributions of a worker dose assessment is the input to the licensing process. Regulators require conservative safety analyses to be performed to demonstrate compliance with design dose limits. Selected parameters and inputs are therefore conservative values. (It is expected that the calculated dose rates will be higher than what is anticipated for the operating facility.) To be conservative, the highest calculated dose rates are selected from the corresponding SSC dose rate analyses.

The PEA maintenance is selected as the critical maintenance mission. Paragraph 3.3.7 explains why this task selection will ensure upper limit values in the results obtained.

3.5 Optimisation – dose constraint

It is international practice that regulators prescribe dose limits to which workers might be exposed. In South Africa, the NNR set a dose limit of 20,0 mSv per year as limit to which workers may be exposed during normal operation of a PBMR plant [3]. However, the NNR requires that radiation exposures of workers should be optimised. For the purpose of this study, in order to determine whether the design is acceptable, a dose constraint is introduced.

The use of the dose constraint is usually associated with prospective dose assessments, and is appropriate to use in this study. From [47]:

> 'The doses to be compared with the dose constraint or reference levels are usually prospective doses, i.e., doses that may be received in the future, as it is only those doses that can be influenced by decisions on protective actions.'

For the purpose of this work, a dose constraint is introduced as a decision-making tool to determine whether or not a calculated worker dose is acceptable. The proposed dose constraint value is that value below which it is planned to keep the prospective annual worker dose.

If the calculated annual worker dose exceeds the dose constraint value, the assessment must be reviewed to determine if it is realistic and representative of the expected operating plant conditions. In several countries, 10,0 mSv/a is used as a dose constraint value [47]. This value is also used in this study.

3.6 Conclusion

The collection of information to perform the assessment is a multi-stage process. It can be briefly summarised as follows:

- Obtain information on the radioactive source that includes information on the types and quantities of radiation emitted and the associated dose rates around these sources.
- Collect information on the occupancy factors of the relevant exposed groups, i.e. the type of work tasks, task frequency, task variability and task duration.
- Collate the contributions from all expected missions.
- Recognise that dose assessment is an iterative process.

A number of assumptions were made to enable identification and compilation of exposure scenarios to be assessed. In this chapter, the assumptions used in the dose assessment have been discussed. It was expected that a radiation worker (other than the radiation surveillance worker) would not spend more than 670 h in areas with elevated levels of radiation.

The FHSS design includes special features to allow for dust removal through purging. Dust removal is designed to be done remotely. The contribution of dust to the external exposure of the workforce was therefore not included.

Building layout and allowance for adequate servitudes are very important in the accessibility of the SSC. They ensure that time is not wasted in restricted spaces when accessing equipment. In this assessment, it is assumed that they were adequately considered in the design.

The access routes were analysed and corresponding travel times for the different routes were estimated. The longest travel time will be used in the assessment to simplify calculations.

The dose received by workers accessing exclusion areas will not be included in the dose assessment. It is planned that access to these areas will only be required during plant shutdown states. Furthermore, special RP requirements exist to allow access to these areas.

Reasons were provided for including the PEA maintenance in the mission selection. This also allows for a conservative worker dose assessment, which is an important prerequisite for the regulator. This chapter concludes by identifying the occupancy categories expected to result in the highest annual exposures.

CHAPTER 4: WORKER DOSE ASSESSMENT

Chapter 4 describes the application and testing of the new integrated worker dose assessment model with, inter alia, various sensitivity cases.

Chapter 4: Worker dose assessment

4.1 Introduction

Chapter 3 described the development of the IA and IAM to perform a worker dose assessment. Chapter 4 describes the implementation and testing of the newly developed IAM during PBMR plant development.

This chapter must be read in conjunction with Chapter 3, which explains in detail the underlying assumptions used to compile the conceptual exposure scenarios and perform the assessment. Chapter 3 also explained the reasons why workers from the following three occupancy categories were selected as representative workers:

- Maintenance mechanical technician: valves.
- Maintenance electrical technician: I&C.
- Routine operations: RP.

Conservative results are achieved through analysis using conservative assumptions and input data. The purpose of conservative assessments is to arrive at a set of safety analysis results that are demonstrably conservative, in comparison with any likely result that will exist on the operating plant [3].

However, a fair balance between conservative assumptions, realism and simplicity must be achieved in the assessment [27]. Realism must be introduced to ensure that the design will not be unduly burdened by unrealistic safety requirements.

A sensitivity study is included to demonstrate the influence of varying the input parameters, or exposure determinants, on the IAM results. The method used to perform the sensitivity analyses is individual parameter variation. This sensitivity study provides insights into the role of the uncertain parameters and initial values in the IAM runs.

The tables in Chapter 4 show the various values allocated to the exposure determinants required as inputs in the formula. The results presented in the tables show the dose associated per mission as well as the cumulative total annual worker dose for each occupancy category.

4.2 Implementation of integrated assessment and integrated assessment model

The Delphi technique was used to predict and analyse the expected maintenance and surveillance work to be performed on the plant SSC. Use of the Delphi method for forecasting, issue identification and prioritisation is proven to be valuable in the early stages of design [48]. For this reason, an expert group of maintenance engineers was used to perform mission and task analysis required to estimate occupancy factors in mission areas.

This group of maintenance engineers was also expected to allocate an annual frequency and estimated time to each mission and task identified. These frequencies and estimated time values enabled the calculation of the occupancy factors required as input in the formula used to quantify annual worker dose. The team was required to follow a conservative approach in allocating parameter values. However, individuals were asked to avoid over-conservatism that would unduly burden the design.

To demonstrate the method, Table 8 gives an example of the results obtained from the PEA maintenance mission analysis. The applicable maintenance items were identified from the breakdown of PEA maintenance mission tasks (only tasks performed in mission areas). Table 8 indicates that the estimated time to perform PEA maintenance is 7.67 h.

Table 8: Task analysis of process element assembly maintenance mission

Mission description: Replace a process element assembly Example: Charge Lock Inlet Block (CLIB) – worst-case task				
Task	**Task description**	**Assumptions**	**Time (h)**	**Time (min)**
Verify system conditions.	Communication with control room to verify that actual system conditions correspond to control room instructions.		0,250	15,0
Verify maintenance environment.	Access the area closest to the intended maintenance area, with all material, data, and people.	Maintainers have not yet been in an adjacent area.	0,167	10,0
Position the Modular Maintenance Equipment (MME) near maintenance station.	Sphere line is cleaned with helium to remove dust and debris.	Operator task: sphere line is clear of dust and debris.	0,167	10,0
Engage (electronic) interlock.			0,167	10,0
Notify control room: ready to commence repair.	Test for FHSS temperature, pressure, radiation at the maintenance area.	Operator task: leaks are NOT present. Cleaning of the sphere lifting-line is not active (for maintenance on CLIB only).	0,167	10,0
Wait for approval to commence.	Confirm that there is no sphere in the valve block. Verify environmental conditions for maintenance area. Test for temperature, radiation levels, helium leakage.	Check for sphere(s) in the valve block. Hold point.	0,167	10,0
Position specialised maintenance equipment (Modular Maintenance Equipment (MME) with PEA at maintenance station.	Attach the interface equipment with specific maintenance equipment for next subtask.	Refer to the specialised maintenance equipment (MME) specification for detail.	0,250	15,0
Lift personnel.	Dock the interface equipment to the ‘valve block assembly’.	MME is positioned and aligned. Attach special equipment to connect with the valve block at the exact specified position.	0,250	15,0

Mission description: Replace a process element assembly Example: Charge Lock Inlet Block (CLIB) – worst-case task				
Task	**Task description**	**Assumptions**	**Time (h)**	**Time (min)**
Remove the actuator. Decontaminate if necessary.	Assure retention of radiation level to expected values for maintenance.	Hold point.	0,250	15,0
Store the actuator.		The AGS provides the internal ventilation, to limit dust egress in the case of a breach of MME containment.	0,333	20,0
Attach containment gate.	Lock MME and open the gate.	Containment is assured.	0,083	5,00
Reconfigure and dock MME and lock.			0,500	30,0
Lock MME.	Temporarily store the PEA in its canister at the laydown area.	The worst-case analysis caters for restoring the function as soon as possible before removal of extracted items.	0,083	5,00
Establish ventilation airflow into the valve block.	Dock and open, to assure containment.		0,083	5,00
Open the containment gate.	Install decontaminated PEA.	There is no dust in the valve block, i.e. no jamming of process elements upon installation. Worst-case concept: used PEAs are replaced with new or refurbished ones.	0,250	15,0
Remove the PEA.	Manually draw vacuum and use a leak-detection device.	SP = Shaft Penetration (part of the PEA).	0,083	5,0
Close the containment gate.	Isolate the system.	This is only done once no leaks are detected.	0,500	30,0
Store the PEA at the laydown area.	Manually verify the functionality of replaced parts.	The design of the PEA prevents incorrect installation.	0,083	5,00
Reconfigure the MME.	Use MME for this task.	Refer to the MME specification for detail.	0,167	10,0
Open the containment gate (mechanical actuation).	Reinstall used actuator if adequate life expectancy. If not, install a new actuator.	Worst-case: new actuator required.	0,500	30,0

Mission description: Replace a process element assembly Example: Charge Lock Inlet Block (CLIB) – worst-case task				
Task	**Task description**	**Assumptions**	**Time (h)**	**Time (min)**
Install the PEA.	Inspect all connectors and couplings and reconnect I&C interfaces, i.e. ‘hook-ups’, including I&C harnesses and pipes, and the AGS services.	Spares determined during detail design.	0,500	30,0
Verify leak-tightness at the shaft penetration.	Verify all relevant PEA-related functioning unit characteristics.	The functional unit commissioning procedure is applied locally at the maintenance area.	0,500	30,0
Stop the ventilation blower.		Only the particular line worked on.	0,083	5,00
Verify functionality of the replaced PEA.	Remove all MME, COTS, consumables, waste and personnel from the maintenance area.	All components are packed, labelled, safe, etc.	0,250	15,0
Remove the containment gate.	Sign off permit to work.		0,500	30,0
Install the actuator.			0,500	30,0
Verify PEA’s functionality.	Purge as described in the fuel handling operating descriptions.	Filtered by helium inventory control system.	0,250	30,0
Declare the PEA replacement functional.	Pressurize up step-wise and check for leaks according to procedure.	A total of three cycles is adequate to assure adequate purity of helium.	0,083	5,00
Exit the maintenance area.	Pressurize system and test for FHSS pressure, pressure boundary integrity (no helium leaks).	Leaks are not expected. Cleaning of the sphere lifting-line is not active. (If a leak is detected, repeat ‘purge the FHSS’ subtask.)	0,250	15,0
		Expected total time for maintenance mechanical technician: valves	**7.67***	**460**

Note: * In the calculation of the dose of the maintenance mechanical technician: valves for the reference case, a value of 8 h is allocated to this mission to simplify the calculation and introduce further conservatism. A sensitivity case is performed to investigate the variation in occupancy factors, by allocating 7 h to this task.

Similar to the preceding process, other missions and associated tasks were identified and analysed for each representative worker. The total amount of time used to perform these missions must equal the estimated time a worker will spend annually in mission areas. (Refer to paragraph 3.3, which explains the assumptions made to determine the estimated time a worker will spend annually in a mission area.)

Table 6 summarises the expected dose rate per FHSS area. This table also provides the information on the location of the various SSC in the FHSS building, where the identified missions will take place. The table enables the analyst to link the mission or task with the local area dose rate.

Conceptual exposure scenarios for each representative worker are developed by combining the information in Table 6 with the occupancy factors estimated from the mission and task analyses. The dosimetric formula for each conceptual exposure scenario is derived in paragraph 3.2.3.

Different spreadsheets, based on equation (2) but using different exposure determinants, were created to quantify the conceptual exposure scenarios. Annexure C contains spreadsheets of examples of the sensitivity cases.

In paragraph 4.3, the dose to the representative worker is calculated for a reference case. The dose a worker will accrue due to travel to and from the workplace is based on the longest travel time of 3,15 min or 0,05 h for a mission. Paragraph 3.3.6 explains how this was calculated.

4.3 Worker annual design dose calculation – reference case

A reference case is used to demonstrate how the IAM is implemented during plant design. In the reference case, a general area dose rate of 5,00 µSv/h is used. An additional 80,0 µSv/h contact dose rate on the valve block is used for compartments with elevated radiation levels (local area dose rate 85,0 µSv/h). Elevated local area dose rates are based on the results of a pile-up of spheres while replacing a PEA. It is assumed that a pile-up of not more than 50 spheres will occur. Paragraph 2.5.3 explains what dose rates will be allocated to local areas with elevated radiation levels.

At the following valve block locations, a pile-up of fuel spheres can accumulate due to a stuck sphere. This is due to the elevation of the piping. These locations are:

- CBA
- BUMS assembly
- Isolation block assembly
- DLOB assembly

The dose rate of 85,0 µSv/h (general area dose rate plus elevated area dose rate) used for the highest local area dose rate is a contact dose rate calculated on the surface of the valve block. Several figures in Annexure B show how the valve blocks were modelled and the detector locations where the dose rate is measured.

At those valve block locations where a pile-up of spheres cannot accumulate due to the elevation of the piping, local area dose rates are set to general area values.

4.3.1 Dose – maintenance electrical technician: instrumentation and control

Table 9 shows the dose assessment for the representative worker in the occupancy category – maintenance electrical technician: I&C.

Missions include those performed on the GCS, AGS, CBA, BUMS, FHSS, I&C and DLOB assembly. These missions are included in the conceptual exposure scenario of this representative worker.

Maintenance work is assumed at the following three valve block areas, where a fuel sphere could be stuck:

- Conveying block
- BUMS
- Charge Lock Outlet Block (CLOB)

Table 9 demonstrates that the estimated annual dose for a representative worker – maintenance technician: I&C – will be 4,04 mSv/a for the reference case.

Table 9: Annual dose assessment for maintenance electrical technician: I&C

Task description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location compartment number	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace insert	4,00	4,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	GCS	Gas return block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	GCS	Cleaning block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	GCS	Diverter block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	CBA	SCS	-18 800	069723	85,0	340
Replace Insert	4,00	4,00	1,00	GCS	Gas filter block unit	-18 800	069723	5,00	20,0
Replace pump	4,00	4,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	20,0
Replace helium service unit LRU	4,00	4,00	1,00	AGS	Helium service unit	-15 000	069723	5,00	20,0
Replace gas ventilation LRU	4,00	4,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	5,00	20,0
Replace gas unit LRU	4,00	4,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	20,0
Calibration test	12,0	4,00	3,00	BUMS	BUMS isolation valve block	18 200	090178	5,00	60
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 13	-22 500	071617	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	5,00	260

Task description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location compartment number	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 8	-18 800	069723	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 10	-15 000	071848	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 1	-9 250	069720	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	700		5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 32	5 200	069831	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	13 050	069716	5,00	260
Inspect Instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 9	16 550	090178	5,00	260
Replace insert	4,00	4,00	1,00	DLOB	Spiral floor	14 400	071617	85,0	340
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 4	24 400	069710	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	29 000	069711	5,00	260
Total per annum	**680**		**638**						**4 040**

4.3.2 Dose – routine operations: radiation protection surveillance

Table 10 shows the dose assessment for the representative worker routine operations: radiation protection surveillance. These workers will perform radiation support services for the maintenance workers. They will also perform routine area surveys.

It is not expected that RP surveillance will take longer than 1 h during PEA maintenance missions. These workers can leave the mission area when the maintenance technicians perform their work. Table 10 includes a breakdown of the support tasks, routine tasks and their associated exposures.

Maintenance work is assumed at the following three valve block areas, where a fuel sphere could be stuck:

- Conveying block
- BUMS
- CLOB

It is assumed that this worker will provide maintenance support to three PEA maintenance missions. In normal operational circumstances, dose management is performed. The radiation safety manager will ensure that the rotation of personnel takes place. This is to ensure that high radiation dose tasks are shared amongst workers.

Table 10 demonstrates that the estimated annual dose for a routine operations – RP surveillance – will be 6,820 mSv/a for the reference case.

Table 10: Annual dose assessment routine operations – RP surveillance

Task description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace insert	1,00	1,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	GCS	Gas return block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	GCS	Cleaning block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	GCS	Diverter block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	CBA	SCS	-18 800	069723	85,0	85,0
Replace insert	1,00	1,00	1,00	GCS	Gas filter block unit	-18 800	069723	5,00	5,00
Replace pump	1,00	1,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	5,00
Replace helium service unit LRU	1,00	1,00	1,00	AGS	Helium service unit	-15 000	069723	5,00	5,00
Replace gas ventilation unit LRU	1,00	1,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	5,00	5,00
Replace gas unit LRU	1,00	1,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	5,00
Calibration test	12,0	1,00	12,0	BUMS	BUMS isolation valve block	18 200	90178	5,00	60,0
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 13	-2 2500	071617	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	5,00	260

Task description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 8	-1 8800	069723	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 10	-15 000	071848	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 1	-9 250	069720	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 3	700		5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 32	5 200	069831	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 3	13 050	069716	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 9	16 550	090178	5,00	260
Replace insert	1,00	1,00	1,00	CLOB	PEA	14 400	071617	85,0	80
Inspect instrumentation	104	2,00	52,0	FHSS instrumentation	Cabinet × 4	24 400	071617	5,00	260
Inspect instrumentation	104	2,00	52,0	FHSS instrumentation	Cabinet × 3	29 000	069711	5,00	260
Routine cleaning and surveillance	60,0	5,00	12,0	Plant wide				5,00	300
Total per annum	**1 331**		**659**						**6 820**

4.3.3 Dose – maintenance mechanical technician: valves

Table 11 demonstrates that the estimated annual worker dose for a representative worker – maintenance technician: valves – will be 9,67 mSv/a, for the reference case.

Maintenance work is assumed at the following four valve block areas, where a fuel sphere could be stuck:

- CBA
- BUMS assembly
- Isolation block assembly
- DLOB assembly

Furthermore, it is assumed that this worker will be responsible for 14 PEA maintenance missions when a fuel sphere is stuck. This implies that a pile-up of fuel spheres will present with more than a monthly frequency. This is a very conservative scenario.

In normal operational circumstances, dose management is performed. The radiation safety manager will ensure that the rotation of personnel takes place. This is to ensure high radiation dose tasks are shared among workers.

Table 11: Annual dose assessment: maintenance mechanical technician – valves

Task description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location compartment number	Gamma dose rate (µSv/h)	Total task dose/ annum (µSv)
Replace insert	8,00	8,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Gas supply block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas supply block unit	-23 000	069724	5,00	10,0
Replace insert	8,00	8,00	1,00	GCS	Gas return block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Gas return block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,5	4,00	GCS	Gas return block unit	-23 000	069724	5,00	10,0
Replace insert	8,00	8,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Cleaning block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Cleaning block unit	-23 000	069724	5,00	10,0
Replace insert	8,00	8,00	1,00	GCS	Diverter block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Diverter block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Diverter block unit	-23 000	069724	5,00	10,0
Replace insert	8,00	8,00	1,00	CBA	CBA	-18 800	071848	85,0	680
Maintenance on PEA	16,0	8,00	2,00	CBA	CBA	-18 800	071848	5,00	80,0

Task description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location compartment number	Gamma dose rate (µSv/h)	Total task dose/ annum (µSv)
Bearing oil and seal check	2,00	0,50	4,00	CBA	CBA	-18 800	071848	85,0	170
Replace insert	8,00	8,00	1,00	GCS	Gas filter block unit	-18 800	069723	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Gas filter block unit	-18 800	069723	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas filter block unit	-18 800	069723	5,00	10,0
Replace pump	8,00	8,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	40,0
Maintenance on pump	16,0	8,00	2,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	10,0
Replace helium service unit LRU	8,00	8,00	1,00	AGS	Helium service unit	-15 000	069723	5,00	40,0
Maintenance on LRU	16,0	8,00	2,00	AGS	Helium service unit	-15 000	069723	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Helium service unit	-15 000	069723	5,00	10,0
Replace gas ventilation unit LRU	8,00	8,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	5,00	40,0
Maintenance on LRU	16,0	8,00	2,00	AGS	Ventilation and vacuum service	Maintenance laydown	069720	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Ventilation and vacuum service	-9 250	069720	5,00	10,0
Replace gas unit LRU	8,00	8,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	40,0

Task description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location compartment number	Gamma dose rate (μSv/h)	Total task dose/ annum (μSv)
Maintenance on LRU	16,0	8,00	2,00	AGS	Auxiliary gas SSS unit	Maintenance laydown	069720	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	10,0
Replace canister LRU	4,00	4,00	1,00	Canister storage	Canister	700		5,00	20,0
Maintenance on LRU	16,0	8,00	2,00			Maintenance laydown		5,00	80,0
Calibration and test	16,0	4,00	4,00	BUMS	BUMS isolation valve block	18 200	90178	5,00	90,0
Calibration and test	16,0	4,00	4,00	AMS	AMS	18 200	90178	5,00	90,0
Visual inspection valve	15,0	0,250	60,0	Gas supply manifold	Valve × 15	-23 000	069724	5,00	75,0
Visual inspection valve	10,0	0,250	40,0	Gas return manifold	Valve × 10	-23 000	069724	5,00	50,0
Visual inspection valve	3,00	0,250	12,0	Gas cleaning block	Valve × 3	-23 000	069724	5,00	15,0
Visual inspection valve	7,00	0,250	28,0	MBA 1	Valve × 7	+18 200	90178	85,0	595
Visual inspection valve	7,00	0,250	28,0	MBA 2	Valve × 7	+18 200	90178	85,0	595
Visual inspection valve	7,00	0,250	28,0	MBA 3	Valve × 7	+18 200	90178	85,0	595
Visual inspection valve	9,00	0,250	36,0	Isolation block assembly	Valve × 9	+18 200	90178	85,0	765
Visual inspection valve	5,00	0,250	20,0	CLOB assembly	Valve × 5	+14 400	69716	85,0	425
Visual inspection valve	8,00	0,250	32,0	DLOB assembly	Valve × 8	+14 400	69716	85,0	680

Task description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location compartment number	Gamma dose rate (µSv/h)	Total task dose/ annum (µSv)
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	85,0	510
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	85,0	510
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	85,0	510
Visual inspection valve	6,00	0,250	24,0	CBA	Valve × 6	-15 000	71851	85,0	510
Scrap sphere samples	2,00	1,00	2,00	Room below CUD		-23 000	69702	45,0	90,0
GCS hex inspect and test	8,00	2,00	4,00	GCS		-9 250	69720	5,00	40,0
AGS pump repair	4,00	4,00	1,00	Maintenance laydown	Pump	-23 000	69724	5,00	20,0
Pool water test and replenish	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
Storage tanks monitor	12,0	1,00	12,0	SSS service hall	Monitor	+9 550	91974	5,00	60,0
Visual piping inspection	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
Pool water cooling assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
Pool conditioning and cleaning assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
SSS ventilation assembly	24,0	2,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	120
SSS blower assembly	24,0	2,00	12,0	SSS service hall	Blower	+9 550	91974	5,00	120
Sphere unloading machines	24,0	2,00	12,0	SSS service hall	Unloading device	+9 550	91974	5,00	120

Task description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location compartment number	Gamma dose rate (µSv/h)	Total task dose/ annum (µSv)
Sphere replenishment system valves check	4,00	1,00	4,00	Sphere replenishment	Valves	+29 000	69711	5,00	20,0
Sphere replenishment counters check	4,00	1,00	4,00	Sphere replenishment	Counters	+29 000	69711	5,00	20,0
Other	120							5,00	600
Total per annum	**681 h**		**575**						**9 665**

4.3.4 Summary – reference case results

Table 12 summarises the results obtained for the reference case. The reference case was used as the benchmark for the conceptual exposure scenarios analysed, and was applied to the following three occupancy categories:

- Maintenance mechanical technician: valves.
- Maintenance electrical technician: I&C.
- Routine operations: RP.

Table 12 also gives information on the total amount of time each representative worker will spend annually in a mission area, the number of missions performed during a year and the cumulative annual dose to each worker.

The dose due to travel is calculated by multiplying the travel time (3,15 min or 0,05 h) by the annual access frequency and a general area dose rate of 5,00 µSv/h.

Table 12: Summary of results for reference case

Occupancy category	Annual time in mission area (h)	Number of missions per annum	Dose due to travel (mSv/a)	Worker annual design dose (mSv/a)	
				Excluding dose due to travel	Including dose due to travel
Maintenance electrical technician: I&C	684	638	0,170	4,04	4,21
Routine operations: RP	1 331	659	0,170	6,82	6,99
Mechanical technician: valves	681	575	0,150	9,67	9,82

The contribution of the dose due to travel (maximum 0,17 mSv/a) to the total annual worker dose is considered insignificant. If the contribution of travel to the WADD can be ignored, equation (2) can be simplified to be equation (4). This is discussed in paragraph 4.4.

In Chapter 3, a dose constraint of 10,0 mSv/a is introduced as a decision-making tool to determine whether or not a calculated worker dose is acceptable. The proposed dose constraint value is the upper dose value considered to be acceptable for the calculated annual worker dose. The results in Table 12 demonstrate that the worker annual dose for each representative worker, using the reference case as conceptual exposure scenario, will not exceed the dose constraint value specified: 10,0 mSv/a.

It should be noted that the estimated annual dose of 9,82 mSv/a of the maintenance mechanical technician: valves is close to 10,0 mSv/a. In the exposure scenario composed for the reference case of this worker, all 14 of the tasks performed on the following components assume a pile-up of spheres and a local area dose rate of 80,0 µSv/h:

- CBA
- BUMS assembly
- Isolation block assembly
- DLOB assembly

The sensitivity cases discussed in the following paragraphs will demonstrate that a comfortable margin within the dose constraint can be obtained by reducing general area dose rates and local area dose rates.

4.4 Sensitivity study

4.4.1 Preamble

Various sensitivity analyses were performed for design and annual worker dose evaluations. This enabled an increased understanding of the relationship between input variables or exposure determinants, and model results. This is to determine which variables have a significant effect on the model output, as well as their relative importance.

4.4.2 Cases

Three cases are briefly presented and discussed:

- Case 1 – investigating the effect of variation in general area dose rate level. This case investigates the variation in general area radiation dose rate values only, while local area dose rates are set equal to these general area values. General area dose rate values of 2,00 µSv/h, 5,00 µSv/h and 10,0 µSv/h are used.
- Case 2 – investigating the effect of variation in local area dose rate. The sensitivity cases are for elevated local area dose rates of 20,0 µSv/h, 40,0 µSv/h and 80,0 µSv/h. General area dose rates are fixed at 5,00 µSv/h.
- Case 3 – investigating the effect of variation in occupancy factor. A sensitivity case is performed to investigate the effect on the results when an occupancy factor of 7,00 h is allocated to the PEA replacement mission for the maintenance technician: valves.

Case 1 – variation in general area radiation levels

In this sensitivity case, set general area dose rate values of 2,00 µSv/h, 5,00 µSv/h and 10,0 µSv/h are used in the different exposure scenarios of each occupancy groups. The local area dose rate is fixed at these dose rates. No elevation above general area dose rate in local area dose rate is considered.

Annexure B contains the spreadsheets for the sensitivity case for a 2,00 µSv/h general area dose rate. The spreadsheets for the 5,00 µSv/h and 10,00 µSv/h are similar; only the dose rate is set to 5,00 µSv/h and 10,00 µSv/h. Table 13 summarises the results for the different sensitivity analyses for variation in general area dose rates values.

The 5,00 µSv/h was selected to introduce conservatism. The design specification of 2,00 µSv/h was determined as a result of this study from one of the sensitivity cases.

Table 13: Case 1 – variation in general area dose rates

Occupancy category	Annual time in mission area (h)	Worker annual design dose (mSv/a)		
		General area radiation level 2 µSv/h	General area radiation level 5 µSv/h	General area radiation level 10 µSv/h
Maintenance electrical technician: I&C	684	1,37	3,42	6,84
Routine operations: RP	1 331	2,66	6,66	13,3
Maintenance mechanical technician: valves	681	1,36	3,41	6,81

Table 13 demonstrates that general area radiation dose rate levels have a significant effect on the annual worker dose. The annual worker dose for the representative worker routine operations: radiation protection increases by 4,00 mSv/a when general area levels increase from 2,00 µSv/h to 5,00 µSv/h. A further increase of 6,65 mSv/a results when the general area dose rate levels increase from 5,00 µSv/h to 10,0 µSv/h. This sensitivity case demonstrates that a design specification of 2,00 µSv/h should be set for the general area dose rate levels in the building.

Figure 12 compares the results of sensitivity set: Case 1 with the dose constraint and the reference case results. It illustrates that the representative worker from the group routine operations: RP will accumulate the highest annual dose when only general area dose rate levels are considered.

Figure 12 shows the important influence that general area dose rate levels have on WADD. A general area dose rate of 10,0 µSv/h will result in a WADD even higher than that for the reference case for maintenance technician: valves.

This is explained by the amount of time the routine operations: RP worker will spend in mission areas. This worker will spend approximately double the time that the other workers spend in mission areas.

Figure 12 also shows how results of the reference case and sensitivity case 1 compare to the set dose constraint of 10,0 mSv/a. The result for the maintenance mechanical technician: valves is close to the dose constraint value. The routine operations: RP worker exceeds the set dose constraint value, which is unacceptable. For this reason, the general area dose rate level of 10,0 µSv/h is unacceptably high.

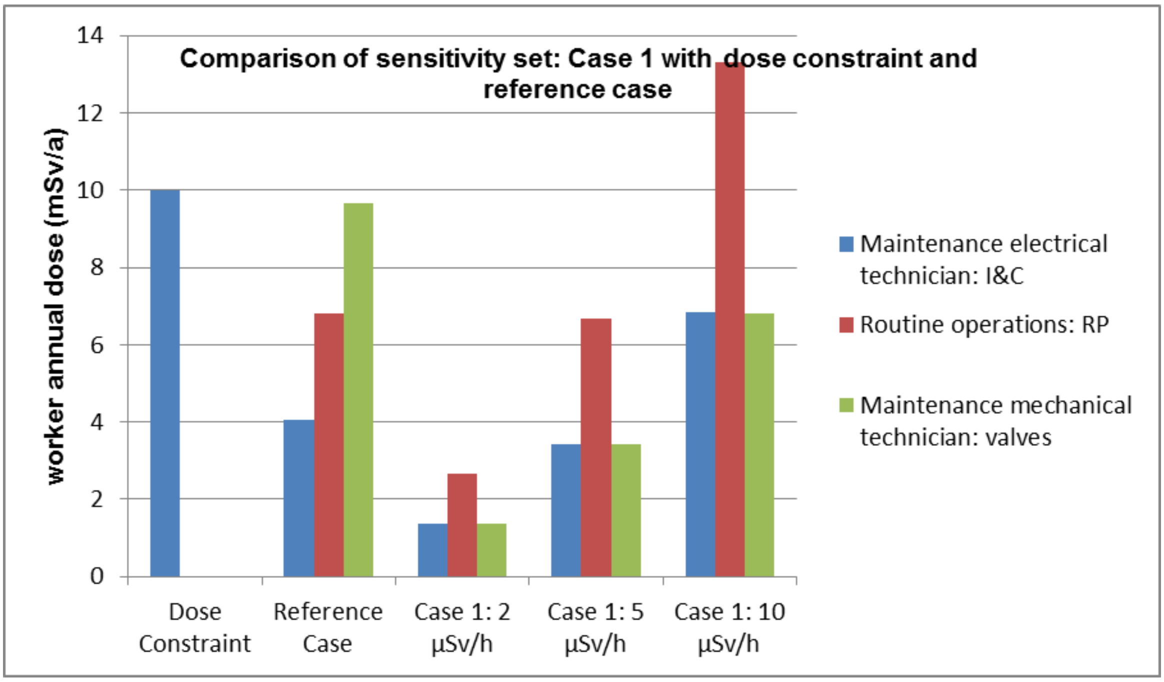

Figure 12: Influence of variation in general area dose rate level

If the design achieves a low general area dose rate (< 2,00 µSv/h), the dose due to travelling can be ignored and equation (2) can be simplified.

The dosimetric formula to quantify the representative WADD reduces to:

$$WADD = \sum_{i=1}^{N} FQ_i \times ST_i \times DA_i \qquad (4)$$

Where:

WADD = worker annual design dose (mSv/a)

N = number of missions performed per annum

FQ_i = annual frequency of access to perform mission i (number/a)

ST_i = stay time in the compartment to perform mission i (h)

DA_i = dose rate in the compartment when mission i is performed (mSv/h)

Under these conditions, a linear relationship exists between WADD, stay time and dose rate. Equation (4) allows the design team to follow the rule of thumb that in the same proportion as the local area dose rate or stay times reduce, the WADD reduces. This simplification of equation (2) enables safety analysts to estimate the reduction in WADD due to a design change by only analysing the reduction of dose rate and stay time.

Case 2 – variation in local area dose rates

Another sensitivity analysis was performed to investigate the effect of a variation in local area dose rate on the annual worker dose. The sensitivity cases were for elevated local area dose rates of 20,0 µSv/h, 40,0 µSv/h and 80,0 µSv/h. This is based on the mission to replace a PEA.

In all these sensitivity cases, a general area dose rate level of 5,00 µSv/h was used, instead of the proposed specification of 2,00 µSv/h. This is to ensure conservatism in the results.

Table 14 summarises the results of the sensitivity analyses of the variation in local area dose rate. This table includes columns for the total annual worker dose per occupancy category. Annexure B contains the sensitivity case for the 20,0 µSv/h local area dose rate.

Table 14: Case 2 – variation in local area dose rate (mSv/a)

Worker description	Worker annual design dose (mSv/a)		
	Local area dose rate 20,0 µSv/h	Local area dose rate 40,0 µSv/h	Local area dose rate 80,0 µSv/h
Maintenance technician: I&C	3,64	3,88	4,04
Routine operations: RP surveillance	6,70	6,74	6,82
Maintenance technician: valves	5,04	6,85	9,67

Figure 13 shows these results as a graph and includes the dose constraint for the purpose of comparison. Case 2: 80,0 µSv/h and the reference case have the same dose rate parameters, therefore the reference case is not included in Figure 13.

An occupancy time of 8,00 h is allocated to the maintenance technician: valves in the localised high dose rate area, when a PEA is maintained. A corresponding 4,00 h are allocated to the maintenance technician I&C for work performed in the same area. Figure 13 illustrates the increase in annual worker dose, as local area dose rate increases for the high dose rate mission.

From the results in Table 14, it can be concluded that the design team should ensure that dose rates in localised areas are < 40,0 µSv/h. In this way, the dose constraint of 10,0 mSv/a can be achieved within a large error margin.

It is further proposed that shielding should be such that the dose rate in those areas where maintenance will be performed are optimised to be 20,0 µSv/h. This will not have a significant effect on the WADD for the routine operations: RP, but will have a significant reduction on the WADD for the other two maintenance groups.

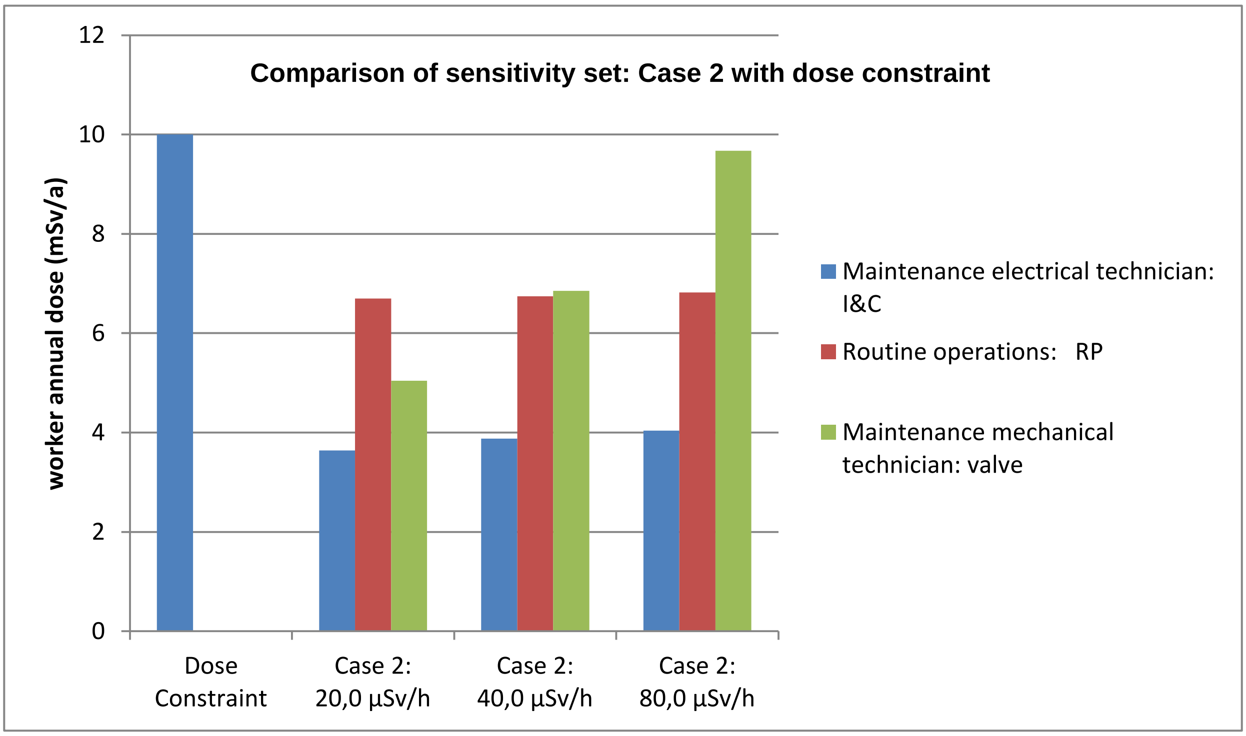

Figure 13: Influence of variation in local area dose rate level

Case 3 – variation in occupancy factor

In the reference case, the maintenance technician: valves receives the highest dose (9,67 mSv/a) when PEA maintenance is performed in an area dose rate of 80,0 µSv/h. In the reference case, 8,00 h are allocated to perform this task. A sensitivity case is performed to investigate the effect if an occupancy factor of 7,00 h is allocated to this task.

The sensitivity case assumes less time will be spent in areas where high general area dose rates exist. Table 15 provides for changing the occupancy factor of PEA replacement and maintenance from 8,00 h to 7,00 h (also refer to Table C.7).

Table 15: Dose assessment for maintenance mission time – 7,00 h (mSv/a)

Occupational description	Reference case (8 h)	Calculated annual dose in mSv/a (general area level 5,00 µSv/h) (7 h)
Maintenance technician: valves	9,67	9,50

This sensitivity case demonstrates that a decrease in occupancy factor in high dose rate areas reduces the annual worker dose. Refer to Table 14.

4.5 Discussion of results

The results of the sensitivity analyses demonstrate that a general area dose rate value of 10,0 µSv/h will result in unacceptably high annual worker dose values. The representative worker from the group route operations: RP will exceed the dose constraint of 10,0 mSv/a.

From the results of the sensitivity case, the proposed target general area dose rate will be 2,00 µSv/h. Therefore it will be required that the thickness of the FHSS wall and floor will be such that it will provide adequate shielding to reduce these dose rates in the building to < 2,0 µSv/h.

Under these conditions, a linear relationship exists between WADD, stay time and dose rate. The assessment demonstrates that the dose due to travel is insignificant compared to the dose accrued during performance of the missions. It can then be derived from the formulation of equation (2) that a linear relationship exists between a mission dose and the three variables FQ, ST and DA. The dose for a single task will hence vary linearly with the variation of each of these parameters, presenting the same sensitivity for all.

Table 12 shows that the annual worker dose for the maintenance technician: valves is 9,67 mSv/a for the reference case. This is if all 14 local area dose rates for the PEA maintenance mission are 80,0 µSv/h and general area dose rates are 5,00 µSv/h. This value does not exceed the 10,0 mSv/a set for the dose constraint.

The annual worker dose for the occupancy category maintenance technician: I&C for the same conditions is only 4,04 mSv/a. This demonstrates that the difference can be explained by the maintenance technician: I&C assisting only three PEA maintenance missions in high dose rate areas compared to the 14 missions in elevated areas performed by the maintenance technician: valves.

The value of 80,0 µSv/h used in the reference case is calculated as a contact dose rate on the valve block. Workers will be at least at arm’s length from the PEA when it is removed from the valve block (approximately 0,50 m from contact dose rate).

Furthermore, the above demonstrates the effect of assuming that all the maintenance work performed on valve blocks will be in conditions where a fuel sphere is stuck and a pile-up of not more than 50 fuel spheres will exist.

In addition, varying the occupancy time of the maintenance technician: valves from 8,00 h to 7,00 h for PEA replacement and maintenance in the reference case, reduces the annual worker dose by only 0,15 mSv/a. Ensuring that the general area dose rate is 2,00 µSv/h instead of 5,00 µSv/h, results in significant dose reductions.

Table 13 demonstrates that if a general area dose rate of < 2,00 µSv/h is achieved, a further reduction of at least 2,00 mSv/a in annual worker dose is expected. If the design of the building is such that general area dose rates are < 2,00 µSv/h, radiation zoning of areas inside the building will be simplified. Limited access requirements into controlled areas will be necessary.

The results in this chapter demonstrate that the major factors contributing to low occupational radiation exposure include low general area dose rates and adequate shielding to ensure elevated localised dose rates are < 40,0 µSv/h.

4.6 Input data limitations

As previously mentioned, several iterations of this quantitative dose assessment will be performed during the design. It is expected that as the safety assessment matures, a better understanding of the radiation risks in the facility will develop. This will influence the scope and level of detail of the assessment. New iterations will also be required where design modifications take place.

Dose rate calculations for the various SSC were performed while the design was still changing and maturing. It is recognised that this could result in dose rate analysis becoming outdated while input data to the calculation is still changing. Therefore, project management has to be in place to ensure control of the input variables to this assessment.

A limited number of dose rate analysis results were available for SSC inside the FHSS, for normal operating conditions. Due to the limited design information available, these safety analyses were aimed at estimating upper bound annual dose values, such as PEA maintenance when a pile-up of spheres is present.

It is important to note that these results are valid for a fuel sphere burn-up of 90 000 MWd/t [46]. For the type of fuel to be used in the demonstration plant, a burn-up of at least 100 000 MWd/t is proven technology. The expectation is that this can be increased to 200 000 MWd/t if an enrichment of 20% is achieved [49]. The drive in reactor development is to increase burn-up and reduce the operating and maintenance costs of the facility. However, this will increase the radiation emitted from the spheres in and around the valve blocks. The increase of source term due to increase in fuel burn-up is outside the scope of this study.

V&V processes play an important role in ensuring that the results of this assessment can be justified. Furthermore, effective quality control and management of data are essential due to the large amount of information that has to be integrated into these dose assessments.

CHAPTER 5: VERIFICATION AND VALIDATION

Chapter 5 explains the verification and validation methods used to provide confidence in the results obtained with the integrated assessment and integrated assessment model.

Chapter 5: Verification and validation

5.1 Introduction

Safety is the number one priority in the nuclear industry. For this reason, safety analyses conducted for a nuclear reactor are subjected to high levels of scrutiny. Safety forms an integral part of the development of the PBMR. Specific procedures and guidelines have been compiled at the PBMR for Quality Assurance (QA), and Verification and Validation (V&V).

QA and V&V work together to ensure confidence in the safety analyses performed. This is to ensure that the new design is safe and that the results of the safety analyses are accurate enough to confirm this. The results obtained from this study should be conservative, as per international standards P[27].

V&V builds confidence in the use of the IAM and the suitability for its intended application. Verification investigates whether a certain tool is doing what it is supposed to do. An example of this would be to verify whether a code is solving the equations correctly. Verification is mostly a mathematical exercise.

Verification of the assessment in this study is performed according to model and code verification. Model and code verification is the process of determining that a computational or mathematical model correctly implements the intended conceptual model, and focuses on the mathematics and software engineering [10]. This ensures that the controlling physical data has been correctly translated in the calculation.

Validation, on the other hand, tries to ascertain the accuracy of the results. The aim, therefore, is to check that the correct equations are being used in the code or model and that the results compare well with empirical studies. Validation tests the performance of the model.

Model and code validation is the process of determining whether a mathematical model is an adequate representation of the real system being modelled and focuses on the physics used [10]. This is achieved by comparing the results of the model with observations of the real system or with experimental data.

The V&V of the IAM developed in this study is partially achieved through benchmarking. Benchmarking in the nuclear industry is a well-established concept [50]. Code, model and data benchmarking are important within the nuclear industry, as analyses used in safety-related and design studies are increasingly reliant on computer software and software applications [50].

In the benchmark study, annual worker dose of GCRs and other HTGR designs, obtained through a literature review, are used as the standard to benchmark these IAM results. Results calculated with the IAM are compared to values reported for:

- GCRs;
- Fort St Vrain; and
- other HTGR designs.

The limitations of this benchmark exercise are discussed in paragraph 1.4. The benchmark study demonstrates that the values obtained are in the range and of the same order of magnitude reported for GCRs and other HTGR designs. This provides confidence that the numerical values obtained with this IAM are in the expected range of results for similar technologies.

The V&V process then focuses on the software, models and codes used to implement the IAM to ensure that the safety codes being used for analyses are well verified and validated. This provides confidence in the instruments used to perform the calculations.

Figure 14 is a flow diagram that summarises the different elements of the V&V process followed in this study. This is to provide confidence in the results obtained with the simplified IAM developed in this study.

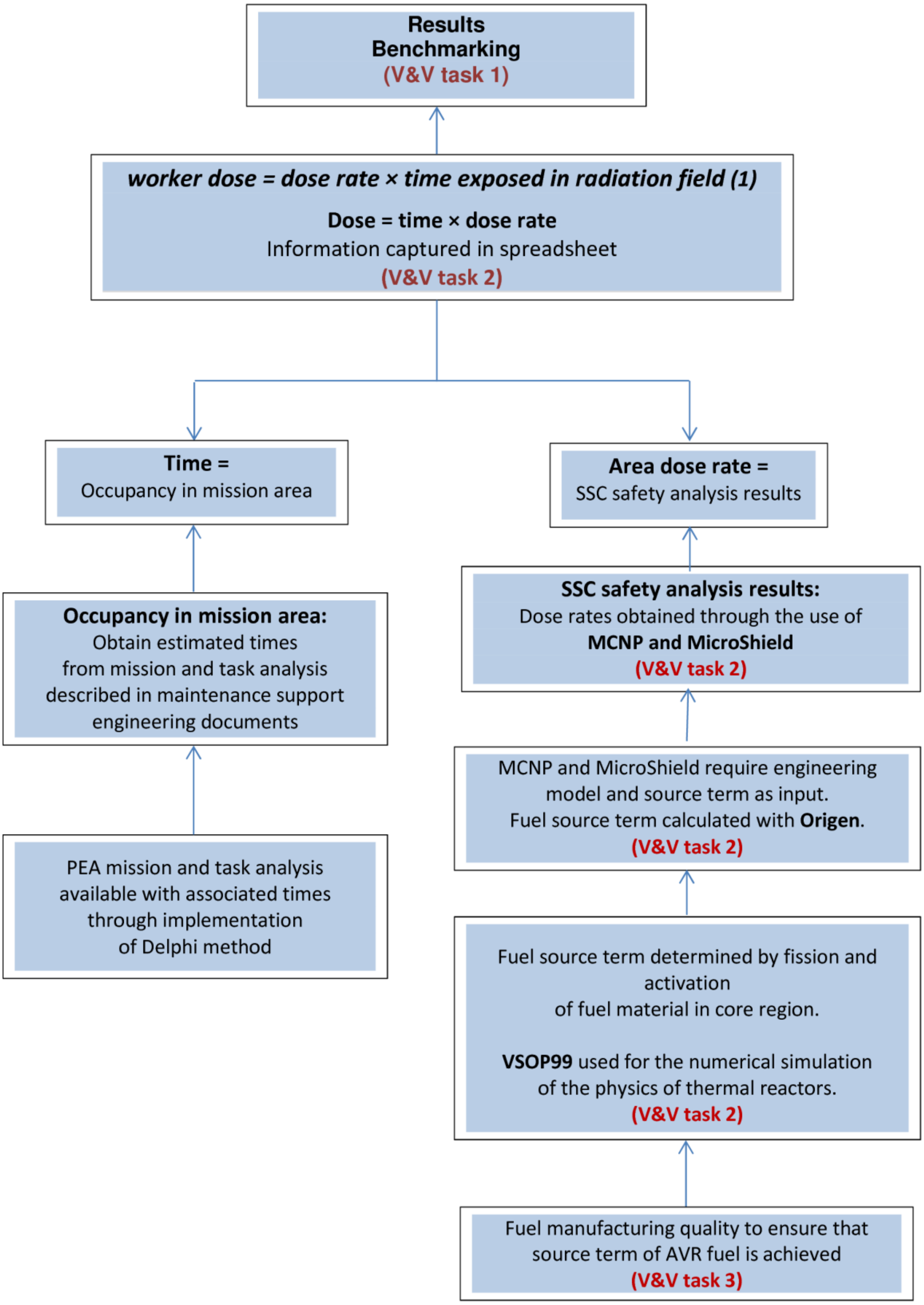

Figure 14: Process diagram for verification and validation of software, models and codes

5.2 V&V task 1 – benchmarking

5.2.1 Preamble

Uncertainty in the results obtained with the IAM is largely due to a lack of knowledge. This is due to a lack of reported annual worker dose values for other HTGR programmes and lack of safety analysis results performed for normal operating conditions during the PBMR design.

In this early design phase, the validation for this IAM was to ensure that the results obtained are in the expected range of values reported for GCR and HTGR designs, and produced conservative estimates.

In the literature review in Chapter 1, different types of HTGR designs were identified. Those applicable to this study can be summarised as follows [37]:

- In Germany, the AVR programme (a 15 MWe design), which was based on pebble fuel and conducted between 1967 and 1988 (no publicly available annual worker dose results).
- In the US, the 40 MWe Peach Bottom-1 Reactor (operated in the US between 1967 and 1974) and the 330 MWe Fort St Vrain Reactor (operated between 1976 and 1988), which were based on prismatic fuel.

There is a lack of information on annual worker dose reported for these designs. Only average annual worker dose and collective radiation exposure are available to use in the benchmark exercise.

5.2.2 Average annual worker dose

Table 13 and Table 14 in Chapter 4 give the results of the different sensitivity cases. Table 16 summarises this information. It is recognised that these values are WADD for specific tasks allocated to representative workers.

Table 16: Summary of calculated worker annual design dose for different sensitivity cases (mSv/a)

Worker description	Case 1: 2 µSv/h	Case 1: 5 µSv/h	Case 2: 20 µSv/h	Case 2: 40 µSv/h	Case 2: 80 µSv/h
Maintenance technician: valves	1,37	3,42	5,04	6,85	9,67
Routine operations: RP	2,66	6,66	6,70	6,74	6,82
Maintenance technician: I & C	1,37	3,41	3,64	3,88	4,04

The range of WADD values is 1,37 mSv/a to 9,67 mSv/a. This is due to design factors and maintenance factors used in this study. It is recognised that task-specific dose can vary by an order of magnitude.

Table 17 gives the operating history of Fort St Vrain and indicates the years in which the plant was in normal operation and shutdown states [51].

Table 17: Fort St Vrain operating history

Year	Annual time online (h)	Operation factor (%)	Energy availability factor (%)		Load factor (%)	
			Annual	Cumulative	Annual	Cumulative
1976	0	0	0	0	0	0
1977	2 930	0	0	0	0	0
1978	4 505	0	0	0	0	0
1979	1 640	22,2	8,5	8,5	8,5	8,5
1980	4 707	53,6	54,0	38,7	23,3	18,3
1981	4 210	48,1	48,9	42,8	26,1	21,4
1982	3 263	37,2	37,6	41,3	20,4	21,1
1983	4 626	52,8	53,2	44,0	26,3	22,3
1984	659	7,5	7,5	37,3	2,9	18,8
1985	0	0,0	0,0	31,6	0,0	15,9
1986	1 086	12,4	12,4	29,0	2,6	14,1
1987	2 028	23,2	23,2	28,4	6,5	13,2
1988	3 486	39,7	39,7	29,5	23,2	14,3
1989	2 703	46,4	46,4	30,6	28,1	15,2

Note: When the energy availability factor is 0, the plant is in shutdown states.

NUREG-0713, Volume 16 [52], reports the occupational exposure data for the Fort St Vrain operations. Table 18 reports the average annual worker dose adapted from these results. It is important to note that values reported in Table 18 for Fort St Vrain operations include the worker dose measured during pre-operational start-up (1976 to1978), maintenance shutdown (1985) and final shutdown operations (1990 to 1994).

It is known that doses accumulated are higher during shutdown states than that during normal operation. During shutdown states, access is required to high dose rate areas and exclusion areas not accessed during normal operation. These high values reported in Table 18 for Fort St Vrain during shutdown conditions are penalising the results.

For the purpose of this benchmark study, only the values reported for normal operating years will be considered. This will be the periods 1979 to 1984 and 1986 to 1989. This is in agreement of the scope of this study, which is limited to normal plant operation. The reported range of values for these periods is from 0,200 mSv/a to 0,500 mSv/a.

Only the result of Case 1 – 2 µSv/h (1,37 mSv/a), for the maintenance technician: I&C is similar to the Fort St Vrain operations results. The assumptions made in this study and discussed in paragraph 3.3.5 are that it is expected that the two occupational groups, maintenance technician: valves; and routine operations: RP will result in upper limit values for WADD.

The results for occupational group maintenance technician: I&C will be more representative of low dose missions.

5.2.3 Collective radiation exposure

In 1989, the World Association of Nuclear Operators (WANO) was created. The WANO Performance Indicator Programme supports the exchange of information between nuclear plant operators by collecting, trending and disseminating nuclear plant performance data. These indicators enable nuclear operating organisations to monitor their own performance and progress, and to benchmark safety performance between different plants [53].

WANO has identified 10 performance indicators used for benchmarking between organisations and different designs. It uses collective radiation exposure as a performance indicator to measure the effectiveness of personnel dose controls. The collective radiation exposure refers to the amount of radiation received by a group of people. The collective dose is expressed in person-sievert (person-Sv) and can be expressed numerically by equation (5) [54]:

Annual collective effective dose = Sum of all the measured worker dose

$$S = \sum_{i}^{N} E_i \tag{5}$$

Where:

E_i is the annual effective dose received by the ith worker

N is the total number of worker.

The WANO report contains several graphs of collective radiation exposure for different reactor designs [53].

Figure 15 from [53] shows that the collective radiation exposure for Advanced Gas-cooled Reactors (AGRs) and GCRs reported for the period 1990 to 2012 ranges between 16 person-mSv and 570 person-mSv. Over the past decade, large reductions in collective radiation exposure have been achieved due to the implementation of As Low As Reasonably Achievable (ALARA) programmes and old high dose designs being shut down.

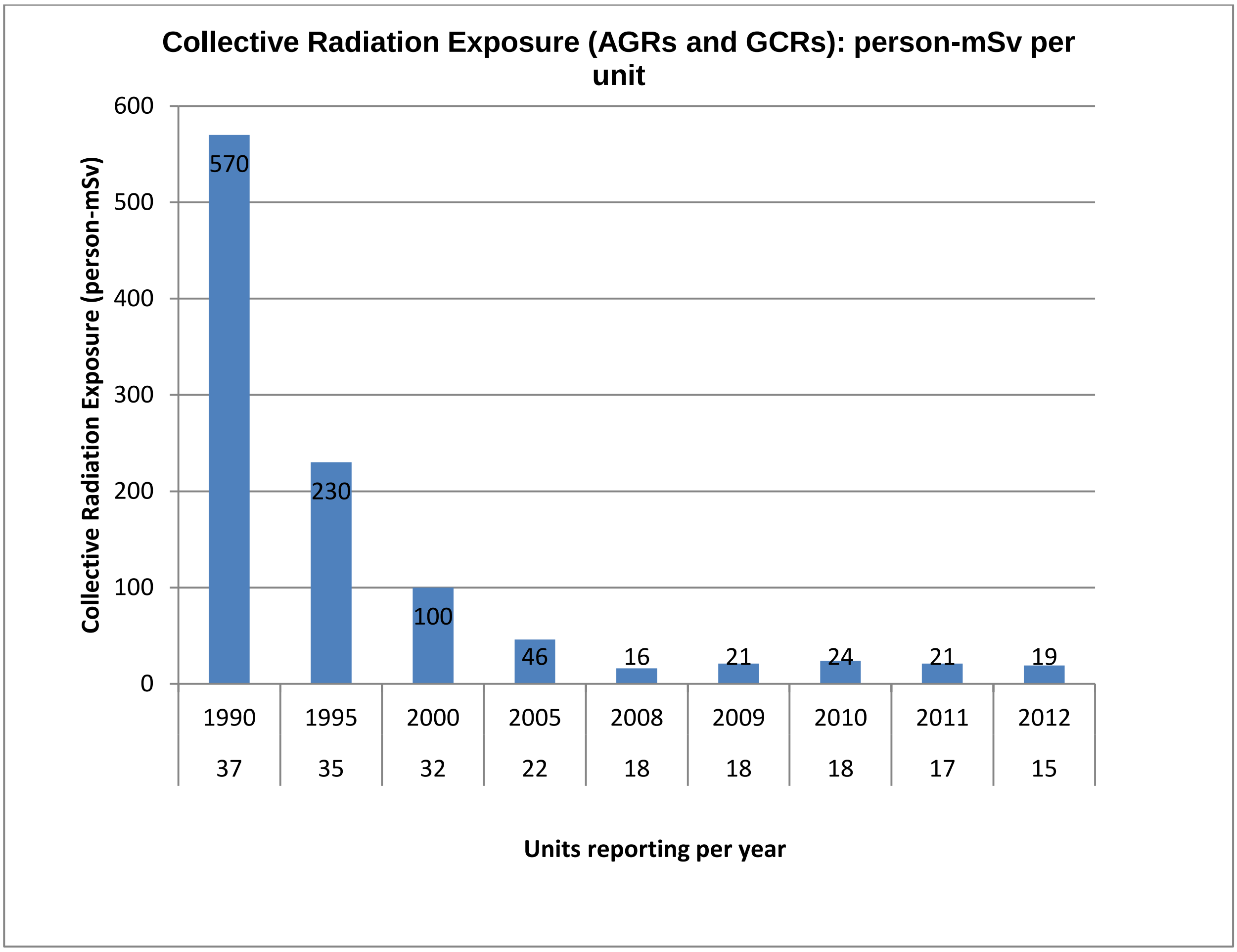

Figure 15: Collective dose comparison for workers in advanced gas-cooled reactors and gas-cooled reactors

NUREG-0713, Volume 16 reports the occupational exposure data for the Fort St Vrain operations [52]. Table 18 summarises these results. For the purpose of this benchmark study, only the values reported for normal operating years will be considered. This will be for 1979 to 1984 and 1986 to 1989.

Table 18: Occupational exposure summary of Fort St Vrain

Year	Number of monitored individuals	No measurable dose	Number of dosimeters with measurable reading	Annual collective radiation exposure (person-mSv)	Average dose (mSv/a)
1974	1 661	1 597	64	33,0	0,500
1975	1 263	1 263	0	0,000	0,000
1976	1 387	1 362	25	13,0	0,500
1977	1 002	946	56	29,0	0,500
1978	930	896	34	17,0	0,500
1979	1 271	1 149	122	64,0	0,500
1980	960	902	58	30,0	0,500
1981	1 271	1 096	175	10,0	0,300

Year	Number of monitored individuals	No measurable dose	Number of dosimeters with measurable reading	Annual collective radiation exposure (person-mSv)	Average dose (mSv/a)
1982	1 000	978	22	4,00	0,200
1983	1 013	965	48	10,0	0,200
1984	1 686	1 616	70	30,0	0,400
1985	2 372	1 929	443	350	0,800
1986	291	221	70	18,0	0,300
1987	209	155	54	12,0	0,200
1988	262	238	24	7,00	0,300
1989	371	316	55	27,0	0,500
1990	256	226	30	6,00	0,200
1991	601	525	76	54,0	0,700
1992	734	520	214	254	1,20
1993	823	657	166	752	4,50
1994	591	390	201	780	3,90

Table 18 further shows that the worker annual collective radiation exposure during normal plant operations ranges at Fort St Vrain ranges from 4 person-mSv to 64 person-mSv.

These results are supported by work performed by Su and Engholm [21]. They reviewed results on occupational radiation exposures in HTGR plants. They estimated the expected rate of collective radiation exposure accumulation for a large HTGR steam cycle unit to be 0,700 person-mSv/MWe year [21]. This value is also used to derive a benchmark collective radiation exposure value for the PBMR design.

Therefore, 0,70 person-mSv/MWe can be converted to the corresponding collective dose for a 400 MWth PBMR. For the 400 MWth PBMR design, the corresponding electrical output is 165 MWe and the corresponding collective radiation exposure is 116 person-mSv.

The values in Table 16 can be adapted to estimate values for collective radiation exposure and used to compare this to the benchmark value of 116 person-mSv. Only the dose calculated for the occupancy category maintenance technician: I&C is used to benchmark collective radiation exposures. The other two occupancy categories are considered worst-case scenarios providing upper dose limit values.

It is expected that the actual collective radiation exposure will be in the range of values reported in Table 19 for the maintenance technician: I&C, or lower. WADD values for the maintenance technician: I&C range between 137 person-mSv and 388 person-mSv.

Table 19 demonstrates that the results obtained from the sensitivity set: Case 1 – 2 µSv/h (137 person-mSv) is similar to the value calculated by Su and Engholm (116 person-mSv). The range of values is of the same order of magnitude expected for HTGR designs.

Table 19: Benchmark of collective radiation exposure values (person-mSv)

Worker description	Collective* radiation exposure (person-mSv)						
	Fort St Vrain	HTGR-type reactors	Case 1: 2 µSv/h	Case 1: 5 µSv/h	Case 2: 20 µSv/h	Case 2: 40 µSv/h	Case 2: 80 µSv/h
Maintenance electrical technician: I&C	64	116	137	342	364	388	404

Note: * Collective dose = average annual worker dose × number of measured radiation workers. Adapted from Table 16. The projected number of PBMR employees is 100 workers P[28].

It can be concluded that the adapted WADD results provide collective radiation exposure values of the same order of magnitude as those estimated for HTGR designs. The annual worker dose calculated with the simplified IAM for the FHSS was skewed by high dose tasks, such as PEA replacement when fuel spheres are stuck and a pile-up of spheres is present. Actual results can be expected to be lower if task selection includes more representative low dose rate tasks.

5.3 V&V task 2 – software verification and validation process

5.3.1 Preamble

The focus is on the software to ensure that the safety codes being used for analyses are well verified and validated. For this reason, the V&V effort was subdivided into the following topics:

- Software verification
- Software validation

5.3.2 PBMR verification and validation approach for safety analyses

Background

The safety analyses used in this study are performed using a variety of analytical software tools. Some of these tools are legacy software obtained from the German HTR programme. A formal process has been developed to perform reverse engineering and V&V [55].

- VSOP99 and the ORIGEN code system were used to quantify the fuel sphere source terms. They are used to determine the radionuclide inventories for the fuel spheres and their associated source term. The source term is used as input to the codes used to calculate dose rates [46].
- The MCNP and MicroShield codes are used extensively in different types of engineering and physics investigations, and are well verified and validated [56].

- The MicroShield codes are used extensively in RP. PBMR purchased the V&V package from Grove Software, Inc. Internal software V&V was performed.
- The worker annual dose was calculated using Excel spreadsheets.

Codes used for source term calculation – VSOP99 and ORIGEN-S

a. VSOP

V&V of VSOP99 was done by performing benchmarking tests. A rigorous benchmarking of a pebble bed reactor equilibrium cycle analysis is currently impossible, given that no pebble bed reactor has ever operated in an equilibrium cycle. Full code-to-code comparison is difficult because the simulation of these transport processes involves numerous methods, assumptions and data sources, each of which is a source of uncertainty in the final result [57].

A comparison was performed of burn-up and pebble mixing algorithms used in two pebble bed reactor fuel management codes – VSOP and PEBBED. Algorithms used approximated results by discretisation of the burn-up profile in the core. Differences in these algorithms and averaging processes resulted in differences in discharge burn-up and the concentrations of individual nuclides [57].

This study reveals the sensitivity of pebble bed equilibrium cycle calculations to differences in depletion and pebble mixing algorithms. It demonstrates that results from these algorithms must be reported with appropriate statements of uncertainty [57].

b. ORIGEN-S

The ORIGEN-S code is a module of the larger Standardised Computer Analyses for Licensing Evaluations (SCALE) computational system, developed and maintained under a configuration management plan by Oak Ridge National Laboratory [50].

The code has been extensively verified and validated by the code developers and through years of international experience with the code in routine applications; analyses of measurements and benchmarks; and code comparison studies [50]. The verification studies involve comparisons of the ORIGEN-S results with a wide array of other codes that use both similar numerical methods and codes that use independent methods including analytical solutions.

Validation is performed with benchmarks involving validated standards, experimental measurements or other validated codes designed to perform similar types of analyses. These V&V studies demonstrated that ORIGEN-S will accurately predict results over a wide range of applications [50].

Codes used for dose rate calculation – MCNP code and MicroShield code

MCNP and MicroShield are used extensively in various nuclear engineering and radiation analysis applications. MCNP has been developed and supported by the Monte Carlo team at Los Alamos National Laboratories for 25 years [56].

Four sets of verification problems were used to ensure MCNP5 code correctness [56]:

- A suite of 42 regression tests.
- A suite of 26 criticality benchmark problems.
- A suite of 10 analytic benchmarks for criticality.
- A suite of 19 radiation shielding validation problems.

In nearly all problems, MCNP5's results exactly match the previous version of MCNP. The few that differ agree well within statistics. It is concluded that MCNP5 is verified to be as reliable and accurate as previous versions, and that all previously existing capabilities have been preserved [56].

MicroShield is widely used in the nuclear industry and is well-suited for analysing simple gamma radiation shielding problems. V&V of MicroShield 6.20 is publicly available and is published in the MicroShield User's Manual.

Microsoft Excel spreadsheet design

A single user compiled the spreadsheets in a single workbook file. The worksheets are relatively simple (discussed in Chapter 3 and Chapter 4). This workbook does not contain macros, cell protection or instruction worksheets. The spreadsheet is intended for personal use and the developer was responsible for identifying the raw data, cell locations of formulas and data-entry cells.

During the developmental stage, the creator of the workbook performed in-process verification of the formula used, by manual calculations using a hand calculator. Examples of the completed worksheet are included in this study and were independently reviewed by a qualified nuclear physicist. These are considered documented evidence of the content and results obtained with the worksheets.

The input data used in the spreadsheets was validated by inspecting all collected data for completeness and reasonableness. Examples of these spreadsheets are provided in Chapter 4 and Annexure C.

5.3.3 Software installation tests

After the software code has been approved for use, each release or update still has to pass further installation tests that compare results with known validation cases. These tests were performed to confirm that the installation was done correctly. All the analyses that were conducted during the course of this study were done with software that was subject to the preceding V&V process [55].

5.4 V&V task 3 – fuel qualification

The safety analysis results could be partially validated by validation of the calculations for the radionuclide inventory of the fuel. The radionuclide inventory is the source term used to calculate the dose rate. Dose rate is an important input parameter used in the WADD formula. The validation of the source term is therefore of primary importance in this study.

A pebble bed HTR analysis of the radionuclide inventory depends on [58]:

- reactor analyses, which determine the fuel sphere inventory and temperature dependence as the fuel sphere circulates through the core;
- fuel performance analyses determining the time-dependent fission product release; and
- nuclide activation analyses.

The calculated source term has a significant contribution in the results of the annual worker dose assessment. Dose rates, one of the exposure determinants and input in the WADD formula, are based on the estimated source term. The validation of the source term will be achieved by [46]:

- validating that the fuel manufactured is of the same or better quality than that manufactured for the German programme (HOBERG/NUKEM); and
- providing assurance that analytical models used in estimating source terms are representative with an adequate margin.

The PBMR approach to fuel manufacturing is to ensure that the manufacturing process is equivalent to or better than that used for manufacturing of German fuel. Validation results are available from the many experimental facilities from the German HTR reactor design programme [38]. German Low Enriched Uranium (LEU) UO_2 fuel element test results have been published [59].

The first fuel kernels have been manufactured and Quality Control (QC) tests have been performed on these fuel kernels. The following conclusion was made on the equivalence tests between PBMR and HOBERG/NUKEM-manufactured fuel:

> 'It is clear from the above results that the QC test methods for coated particles have been established and are equivalent to the HOBERG/NUKEM method.' P[29].

Furthermore, PBMR-coated particles were irradiated at Idaho National Laboratory in the US. A paper was presented on this at the Physor 2012 conference. This experiment demonstrated that the fuel particles produced in South Africa are of high quality [55], [59].

However, some additional fuel performance data will be gathered in specific experimental facilities to confirm that equivalence to the German fuel has been achieved. The data that becomes available from ongoing tests will be used to develop and improve the technical basis of the design and its associated operational documents for the foreseen commercial plants. Test results obtained during the design at experimental facilities must continue to confirm the adequacy of assumptions made during scoping calculations of the source term and dose rates.

5.5 Ongoing validation on demonstration power plant

It is recognised that practical limitations to verify the model exist. This is due to a lack of adequate benchmarking information for similar designs.

Experimental validation will continue when the plant becomes operational. This will require collecting large amounts of experimental data to verify the results of this quantitative assessment. Comparison with operational plant data will be achieved by extensive monitoring of the built plant.

The verification of this model is not only dependent on the actual annual worker dose calculated, but on all the exposure determinant parameters contributing to this dose. Empirical data has to be collected on all exposure determinants before this model can be adequately verified.

5.6 Peer and independent verification reviews

The engineering process requires adequate quality through a rigorous process of peer and independent review. Peer and independent verification reviews of the worker dose assessment were performed by several experts inside the PBMR in 2009.

These peer and independent verification reviews were performed by suitably qualified and experienced individuals. The verification reviews allowed for the identification of any contributions to radiation risks that had not been taken into account in the assessment.

They also ensured that the models and data that were used were independently verified to be accurate representations of the design. The feedback that was provided was incorporated into the assessment. This provided a high level of confidence that the IAM is acceptable for the purpose of the assessment.

5.7 Conclusion and future development

The results obtained with the simplified IAM were of the same order of magnitude as the benchmark values used. Confidence in the IAM developed could also be achieved through rigorous QA and V&V processes.

Various international HTR benchmark efforts are still ongoing, some of which were reported on at international conferences. Validation results are also used from many experimental facilities from the German HTR reactor design programme, as well as from PBMR-specific experimental facilities [55].

The development of an integrated code system for the calculation of HTR reactors is currently being planned. The main reasons for this effort are [55]:

- To increase analysis accuracies using more modern and advanced methods by which reactor and plant designs can be optimised.
- To reduce errors by having integrated data interfaces between the different modules.
- To improve user interface with internally defined checks and balances for erroneous input.
- To have a modern modular code system, being developed according to modern software development requirements, that can be improved and debugged more easily.

There is a need to put more effort into the validation of this IAM. Assessment validation can only be partially completed during the design of a new plant. Assessment results will only be fully confirmed once it is possible to compare results from this assessment with actual measurement data from a test facility or demonstration plant. However, this is outside the scope of this study.

At this stage of PBMR development, the precision or actual value of the estimate of the annual worker dose was of secondary importance. However, this IA and IAM provide useful results to enable engineering judgements to be made regarding design improvements and changes.

CHAPTER 6: CONCLUSIONS AND RECOMMENDATIONS

Chapter 6 discusses the outcomes and the contributions of this study to the development of the PBMR project, lists the lessons learnt and gives recommendations for further research.

Chapter 6: Conclusions and recommendations

6.1 Summary and overview

Worker dose assessments are an important part of the safety assessment of an NPP. This study has presented an IA and IAM for worker dose assessment in the early design stages of a nuclear facility. The study has demonstrated that the IA and IAM are valuable tools in the decision-making process of design evaluation and design improvements.

The study is structured as follows:

- Chapter 1 provides background information about the environment within which this research was conducted.
- Chapter 2 gives background information on the available plant design and safety analysis information required to perform the assessment.
- Chapter 3 discusses the research methodology. This explains the manner in which the IA and IAM were devised. It includes the assumptions used to implement the IAM.
- Chapter 4 provides the results of the assessment for different scenarios.
- Chapter 5 explains the V&V approach. The discussion also identifies the outstanding V&V work related to this study.

6.2 Outcomes from this study

The results obtained from implementing this integrated worker dose assessment model provided insight into the cumulative annual dose accrued by workers on the plant. It was the first time during the development of the PBMR that annual worker doses were quantified. Annual worker dose is an indication of the adequacy of shielding and maintenance planning in the design.

In this study, a unique integration is performed of all the engineering design information and safety analyses related to radiation safety. Performing this assessment requires close liaison with all the different design engineering teams; radioactive waste storage and management teams; and HFE group. It also requires the close examination of fault and reliability studies; reactor and fuel analyses. This study enables project management to determine if the engineering design and SSC safety analyses are on the same level of maturity at a specific time in the project.

The IA provides early warning of many of the possible problem areas that may exist. This allows for redesign before significant expenses are incurred. The assessment also provides new insights into and requirements for V&V and test programmes for radiation safety purposes.

This study has provided invaluable knowledge regarding the collection of information on the missions and tasks to be performed on the operating plant. It enabled the identification and prioritisation of risk-significant tasks that contribute to the worker's annual dose. As a result of this, work for the optimisation of the design can be prioritised.

The assessment provides valuable inputs into RP. Maximum annual worker dose equivalents were estimated and used as the basis for comparison with prescribed dose limits. The compilation RP programmes will be based on this information.

6.3 Conclusions on annual dose results

The strength of an IAM is its ability to calculate the consequences of different assumptions and, simultaneously, interrelate many factors. However, it is recognised that an IAM is constrained by the quality and character of the assumptions and data that underlie the model. This IAM provides a very useful framework for organising and assessing available information on the design.

Assumptions with associated bounding parameters were used extensively. These were discussed in Chapter 3. These assumptions are necessary to develop the different scenarios analysed in the assessment. Thus, it is expected that due to the conservative assumptions made, large margins will exist between the results of these calculations and actual measurements on the operational plant.

One of the objectives of this study was to develop an IAM to enable the developer of a nuclear facility to perform a worker dose assessment in the absence of empirical data. Comparison with benchmark values published for GCR and HTGR designs shows that the proposed IAM gives results within the expected range of values for this type of design. This demonstrates that the IAM developed provided credible results.

The IA and IAM developed provide a framework for understanding the ambient radiation conditions on the designed plant. They also enable the design team to make informed judgements about the relative value of different contributors to worker exposure conditions expected. Perhaps the most useful general insights obtained from the IA are a better understanding of the radiation conditions on the plant, its complexities and uncertainties; and the interactions between ambient radiation levels in the building and maintenance support systems in contributing to the annual worker dose.

The NRC states that the precision or actual value of the estimate in a worker dose assessment is of secondary importance. However, it recognises that such a study is of significant value in making engineering judgements regarding design changes for design optimisation purposes [10]. This study further demonstrates that valuable insights are gained for RP purposes by implementing this IA and IAM.

The results of the various sensitivity cases are dependent on the exposure scenarios developed. In this study, exposure scenarios associated with work performed by the representative workers belonging to the following three occupancy categories, were analysed:

- Maintenance electrical technician: I&C.
- Routine operations: RP.
- Maintenance mechanical technician: valves.

These three exposure scenarios provided insight into the expected annual dose of the exposed individuals, resulting in realistic and upper bound worker dose values. Comparison of the results obtained from the maintenance technician: I&C with those of the maintenance technician: valves, gives the safety analyst insight into the effect of allocating higher dose rate missions in the exposure scenario.

The sensitivity cases demonstrated that allowing a general area dose rate of 10,0 µSv/h resulted in unacceptably high annual worker dose results. Therefore a design specification of 2,00 µSv/h is set as the maximum general area dose rate for the operational plant. This can be achieved by ensuring adequate shielding by including appropriate wall and floor thicknesses in the building design.

It was also demonstrated that the maintenance technician: valves will receive an unacceptably high annual worker dose if local area dose rates of 80,0 µSv/h exist. This study demonstrated that a target of 40,0 µSv/h should be set for elevated localised area dose rates, to ensure annual worker dose is within dose constraint values.

The inclusion of the task to maintain and replace a PEA is a significant portion of the resultant dose of the maintenance worker: valves. The 80,0 µSv/h in the valve block rooms was derived from a calculation where it was assumed a stuck sphere resulted in a pile-up of at least 50 spheres in the line. The assumption that this dose rate will exist in the whole compartment is very conservative.

The GCS is specifically designed to provide the capability to dislodge spheres and purge the circulating lines from dust and other forms of loose contamination. It is therefore a conservative assumption to assume that for the maintenance worker: valves, 14 PEA maintenance missions per annum are performed where a sphere is stuck and a pile-up of 50 spheres exists due to a stuck sphere.

The results summarised in Table 16 for these conditions can be summarised as follows:

- Maintenance technician: I&C 3,88 mSv/a
- Routine operations: RP 6,74 mSv/a
- Maintenance technician: valves 6,85 mSv/a

An upper bound of expected annual worker dose is 6,85 mSv/a. This is within the set dose constraint value of 10,0 mSv/a. Table 13 indicates that if a general area dose rate of < 2,00 μSv/h is achieved, a further reduction of at least 2,00 mSv/a is expected.

It must be recognised that actual exposure conditions in areas on an operating plant will consist of a mix of high and low dose rates. In operational NPPs, shielding is added in locations where streaming of radiation occurs. This is an RP protection management measure that ensures that the high dose rates are reduced to acceptable low levels. In this assessment, no credit was given for RP management measures.

The assessment is based on the exposure of a worker to an external radiation field. It is expected that this will be the dominant cause of exposure. As the design matures, more detail will become available. This will necessitate updates of the preliminary assessments. The results obtained from different iterations of the dose assessment will be used to determine if outstanding, or more realistic, information is needed. This will also provide important feedback for engineering design purposes.

The most difficult challenge of this assessment is to integrate all the available design information and determine assumptions to perform the assessment. These assumptions must be conservative, but also ensure credible and realistic estimates of the expected annual worker dose on the operating plant.

Not only was this objective achieved, but the results from this research have provided an improved understanding of the radiation environment of the FHSS during operation. The high level of detail that was included in this research can improve the understanding of design engineers of the environment within which systems and components would be operating.

Furthermore, the importance of this study to PBMR development is explained. The technical acceptability of an NPP design is judged on the basis of the results of the safety assessments performed. These safety assessments should provide reasonable assurance that the NPP design will meet the design objectives, performance standards and regulatory criteria [10].

It can be concluded that worker dose assessments should be directly integrated with the engineering process and programme management activities involved in plant development. Safety assessments should not function as an add-on to the development process.

6.4 Contributions to integrated design process

The assessment allows the review of the design baseline to determine if the design engineers adequately considered radiation safety measures. During the implementation of the IA and IAM on the FHSS design, it was confirmed that the following radiation safety measures were integrated in the design [11].

6.4.1 Fuel design to minimise fundamental radiation sources

A discussion of the PBMR's pebble fuel design is outside the scope of this study. The most important radiation safety feature of the PBMR is the fuel's capability to contain fission products.

The South African pebble fuel manufacturing programme results demonstrate that the same quality of fuel that was manufactured in the German programmes, can be achieved. Models used to calculate the PBMR's source term used in this study, made extensive use of information obtained from the German AVR programme [55], [59].

6.4.2 Radiation source reduction through purification and filtration systems

Review of the FHSS design documentation demonstrated that the purification and filtration systems for dust removal are considered a high priority in the FHSS design. One of the outstanding issues identified was the development of specifications for the FHSS purification systems. This has to be addressed in the next iteration of the design and safety assessment.

6.4.3 Adequacy of shielding

Shielding is provided in bulk by using concrete and large quantities of metal. This is demonstrated in the design of the floors, and valve and shielding blocks. The required levels of shielding for the different SSC have to be optimised based on feedback from radiation safety analyses results.

It is clear that the FHSS valve blocks have been designed to provide adequate shielding. However, localised hot spots were identified in the analyses of some of the valve blocks. This has to be investigated to determine if this is due to analysis deficiencies. Otherwise, it should be ensured that these hot spots are shielded adequately to reduce dose rates further in the operating plant.

Workers are also shielded by additional shielding blocks at the following locations:

- Fuel lines from the spiral floor (FHSS above the reactor cavity) to the spent fuel tank.
- Fuel lines on the spiral floor above the reactor.
- The CUD below the reactor.
- Valves and counters in valve blocks.

Fuel lines transporting fuel from below the reactor to above the reactor will be installed in a dedicated and shielded servitude.

Likewise, shielding is provided between the spent fuel tanks and the access floor for channelling the spent fuel to designated tanks, and access below the tanks is shielded.

All items on the spiral floor (higher levels of the module building) are maintainable from the floors below the components. The spiral floor provides shielding for maintenance from below and ample space has been included in the plant layout design.

Special Line-replaceable Unit (LRU) maintenance areas are allowed for in the design of the plant layout. Additional shielding will be provided in these areas. The special tools are designed in such a manner as to shield components that are removed and replaced from the spiral floor.

6.4.4 Handling distance between radiation source and worker

Review of the engineering design documents indicated that special manual and remotely operable tools have been included in the design. This is to ensure that high dose rate items will be handled at a distance from the worker. The LRU itself is a radiation source if a sphere is stuck inside. The engineering design of the special tools included shielding and special tools to create distance and shield personnel from radiation sources.

6.4.5 Ease of maintenance and equipment removal

The design engineers were aware of the importance of the ease of removal of components and this was verified in the review of the design documents. Ease of maintenance and equipment removal reduces occupancy times.

Where maintenance on LRUs outside the six-yearly outages is planned in areas with high levels of radiation, methods have been identified to lower radiation and contamination levels. The proposed handling of LRUs is to remove the LRU into a special tool that simultaneously provides shielding from and containment of contamination, and a similar replacement approach.

Handling of contaminated replacement parts is addressed. The most significant of these are LRUs, for which remote tools and shielded transfer casks will be provided. The LRUs are placed in special containers and moved to a specially equipped maintenance area until they can be removed and moved to a low dose rate area on a specially provided pump removal cart.

6.4.6 Adequate access servitudes to areas

Servitudes are allocated to special maintenance areas. These will be located in low dose rate areas. The HFE group specified adequate servitudes for the movement of decontamination equipment. Decontamination equipment and space for most equipment are provided as part of the special maintenance laydown areas.

6.4.7 Overall simplification of plant

Special equipment has been designed to ensure that high-risk work could be performed, such as the removal of stuck, damaged and broken spheres. The discussion of the details of this specialised equipment is outside the scope of this study. Appropriate operating procedures have to be developed to explain in detail how these high-risk tasks, such as sphere removal, will be performed.

6.5 Lessons learnt on project planning

Many other lessons were learnt on how to perform such an assessment in a dynamic and changing engineering design environment. The lessons related to project planning are as follows:

- It was clear that the radiation safety analyses of SSC lagged engineering design by at least six months. Lack of coordination and integration resulted in the radiation safety analyses of many SSC being outdated.
- Dose rate calculations were performed for the various SSC while the design was still changing and maturing. It is recognised that this could result in dose rate analyses becoming outdated while input data to the calculation is still maturing P[30].

This research has established the IA and IAM to be used for future analyses and has also been able to highlight many of the possible problem areas. This will greatly assist future assessment of annual worker dose on the plant. This study therefore provides a solid foundation for future evaluation of the design baseline.

6.6 Recommendations for further research

The following items were identified as problem areas that needed further investigation. These items, which are briefly summarised, are actually complicated and comprehensive topics:

- In this study, the design evolved at the same time as the safety assessments were performed. It is essential that the engineering design teams and safety analysis teams maintain thorough communication regarding the current state of the design and the status of the safety assessments at all times.
- Limited safety analysis results on calculated dose rates were available for normal operating plant conditions. SSC dose rate analyses were biased towards accident and upset conditions, such as a pile-up of spheres in a pipeline. The results of this study were therefore biased towards upper-limit values of annual worker dose. The scope of this study was to perform dose assessments for normal operational plant states.
- The review of the reports on the safety analyses of SSC dose rates indicated that dose rates to workers at a distance of 50 cm and 1 m from the SSC were not routinely calculated. It is normal RP practice to use values calculated at these distances in dose assessments. This is a deficiency that needs to be addressed by the radiation safety analysis groups.

- As more detail becomes available with the maturing of the design, the assessment model should be reviewed to ensure that the selected missions are representative of actual exposure scenarios expected. Decisions concerning the frequency of updating of this assessment should be based on professional judgement, depending on the maturity of the available design information and the design changes implemented.

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ANNEXURES

The annexures contain additional information.

ANNEXURE A: TRAVEL TIME CALCULATION

Travel times were calculated by using the following human factor speed parameters P[26]:

- Human walking speed = 1,00 m/s (60,0 m/min)
- Personnel elevator speed = 0,64 m/s (38,4 m/min)
- Equipment elevator speed = 5,00 m/min (300 m/min)

Table A.1 provides an example for estimating travel times and shows the breakdown to calculate travel time to access the SSC in compartment 069723 on level -18 800 P[26].

Table A.1: Example of time allocation for access to compartment 069723 on level -18 800

Function	Description	Time (min)	Comment
1	The radiation worker enters the building at level 700 and proceeds to the personnel elevator.	0,17	d = 10,0 m v = 60,0 m/min
2	Travels with elevator to level -18 800.	0,51	d =18,8 m + 0,7 m = 19,5 m v = 38,4 m/min
3	Walks from elevator to compartment 069723.	0,33	d = 20,3 m v = 60,0 m/min
Maintenance worker performs work in compartment 069723			
4	Returns to personnel elevator.	0,33	d = 20,3 m v = 60,0 m/min
5	Goes to level 700.	0,51	d = 19,5 m v = 38,4 m/min
6	Proceeds out of the building.	0,17	d = 10,0 m v = 60,0 m/min
	Total travel time	**2,02 min**	

ANNEXURE B: DOSE RATE CALCULATIONS

B.1 Introduction

MicroShield has been used to calculate the dose rate on the valve block surface of the BUMS P[21]. The BUMS was modelled extensively, as information regarding shielding specifications of the BUMS detector was required. BUMS design specifications were required by the manufacturer of this system.

The other valve blocks were not modelled in such detail. Measurement locations were not provided in the same level of detail as that provided for the BUMS. Figure B.1 shows the source-shield geometrical arrangement used to model the BUMS valve block.

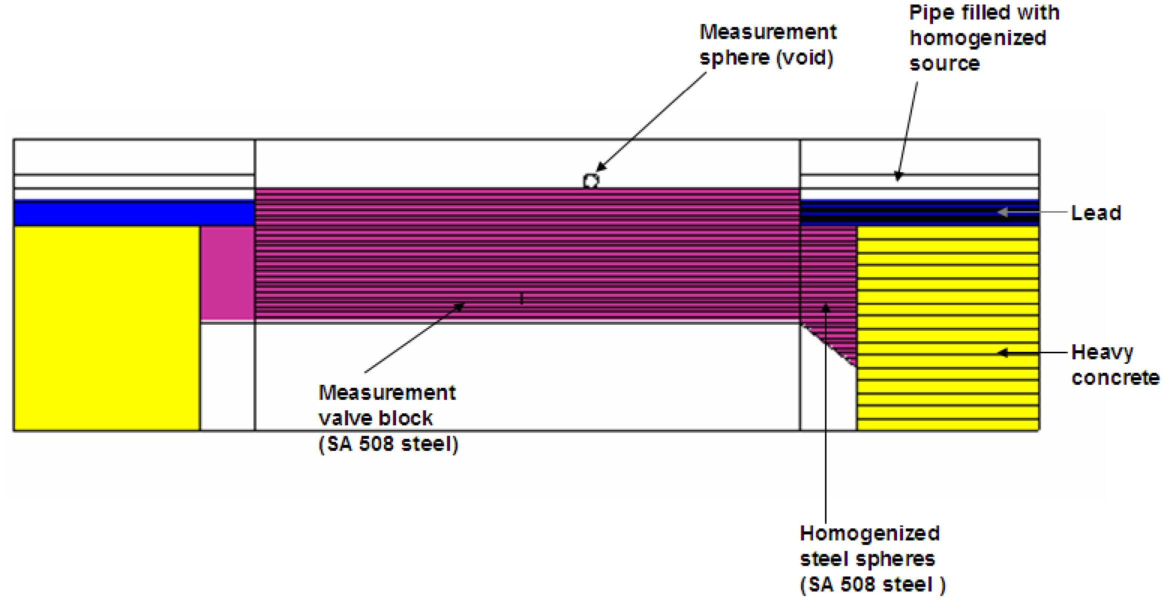

Figure B.1: Geometrical arrangement of source-shield for burn-up measurement system valve block

A 300 cm gamma source (50 fuel spheres) was used to calculate the dose rate values at the bottom of the heavy concrete block. Figure B.2 shows the source-shield geometrical arrangement of the heavy concrete block.

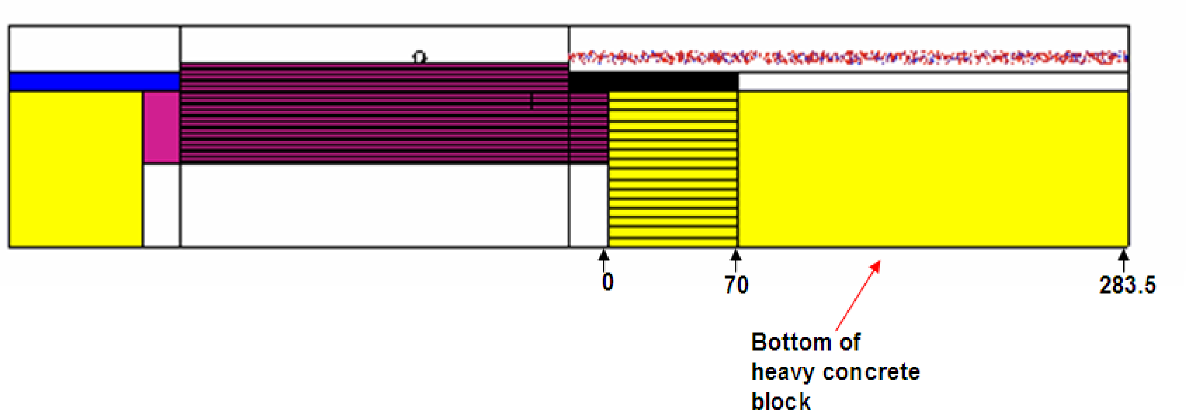

Figure B.2: Source-shield geometrical arrangement of heavy concrete block

B.2 Case 1

As shown in Figure B.3, at the bottom of the heavy concrete block from measurement locations 0,00 cm to 70,0 cm, the gammas are contributing straight down from the source covered with lead (Contribution 1) and sideways from the source, which is not covered with lead (Contribution 2).

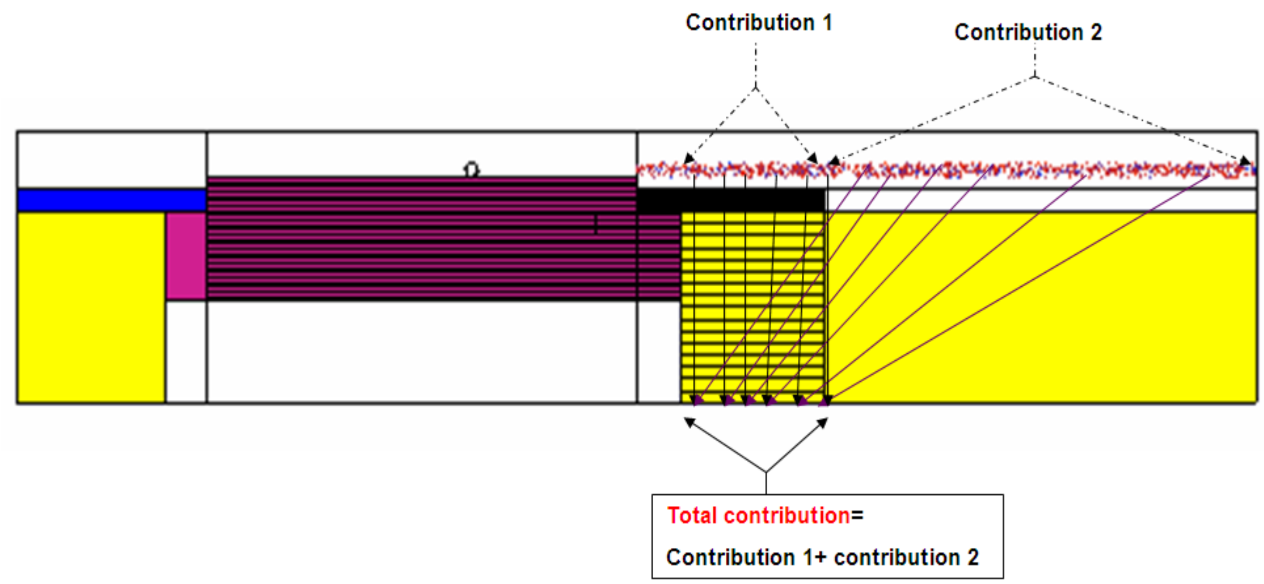

Figure B.3: Case 1 – measurement locations 0,00 mm to 70,0 cm

Table B.1 gives the dose rate values for measurement locations 0,00 mm to 70,0 cm.

Table B.1: Dose rate values for measurement locations 0,00 cm to 70,0 cm

Position of detector (cm)	Contribution 1 (µSv/h)	Contribution 2 (µSv/h)	Total dose rate (µSv/h)
0	0,31	0,86	1,2
10	0,41	1,80	2,2
20	0,48	3,62	4,1
30	0,53	6,81	7,3
40	0,53	12,00	12,5
50	0,48	19,50	20,0
60	0,41	29,60	30,0
70	0,31	40,20	40,5

B.3 Case 2

The dose rate is then calculated separately from detector locations 70,0 cm to 283.5 cm. At these locations, the gammas contribute straight down from the source, which is not covered with lead (Contribution 3) and sideways from the source, which is covered with lead (Contribution 4). Figure B.4 shows Contributions 3 and 4 of the source.

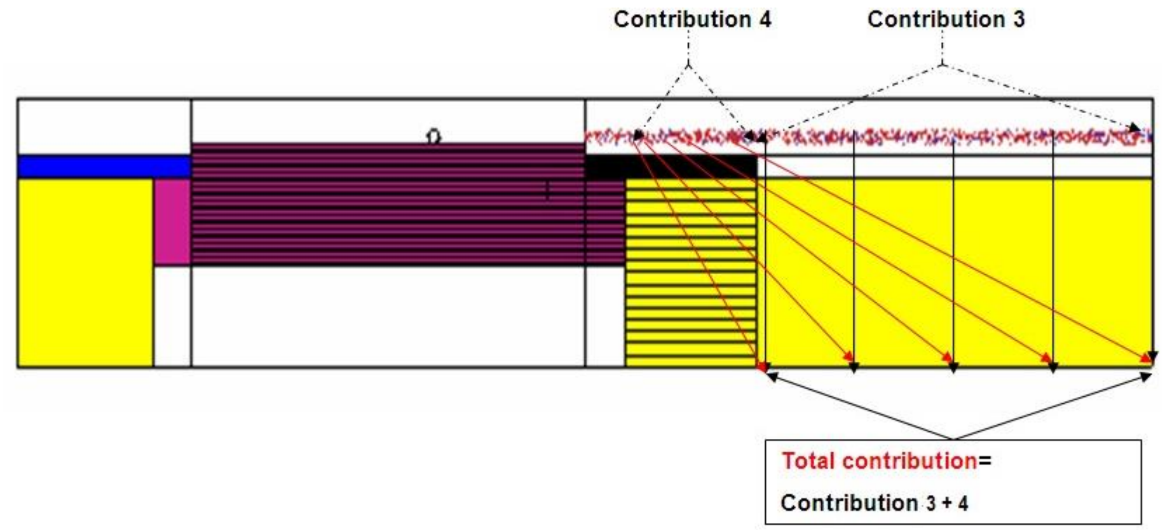

Figure B.4: Case 2 – measurement locations 71,0 cm to 203,3 cm

Table B.2 contains the values calculated for the different measurement locations.

Table B.2: Dose rate at bottom of heavy concrete block

Position of detector (cm)	Contribution 3 from 203,3 cm source (µSv/h)	Contribution 4 from 70 cm source (µSv/h)	Total dose rate (µSv/h)
71	41,3	0,300	41,6
111	73,5	0,030	73,5
151	79,9	0,007	79,9
191	80,0	0,000	80,0
231	74,7	0,000	74,7
274	40,2	0,000	40,2

Table B.2 indicates that a maximum dose rate of 80,0 µSv/h is reached at measurement location 151 cm to 191 cm. This value was used as representative of the local area dose rate, where a pile-up of spheres occurs. The general area dose rate will be added to this value when calculations are performed.

ANNEXURE C: SENSITIVITY CASES

Table C.1: Dose assessment maintenance technician: I&C – general area level 2,00 µSv/h

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace insert	4,00	4,00	1,00	GCS	Gas supply block unit	-23 000	069724	2,00	8,00
Replace insert	4,00	4,00	1,00	GCS	Gas return block unit	-23 000	069724	2,00	8,00
Replace insert	4,00	4,00	1,00	GCS	Cleaning block unit	-23 000	069724	2,00	8,00
Replace insert	4,00	4,00	1,00	GCS	Diverter block unit	-23 000	069724	2,00	8,00
Replace insert	4,00	4,00	1,00	CBA	SCS	-18 800	069723	2,00	8,00
Replace insert	4,00	4,00	1,00	GCS	Gas filter block unit	-18 800	069723	2,00	8,00
Replace pump	4,00	4,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	2,00	8,00
Replace helium service unit LRU	4,00	4,00	1,00	AGS	Helium service unit	-15 000	069723	2,00	8,00
Replace gas ventilation LRU	4,00	4,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	2,00	8,00
Replace gas unit LRU	4,00	4,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	2,00	8,00
Calibration test	12,0	4,00	3,00	BUMS	BUMS isolation valve block	18 200	090178	2,00	24,0
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 13	-22 500	087774	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 12	-22 500	087774	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 12	-22 500	087774	2,00	104

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 8	-18 800	069723	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 10	-15 000	071848	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 1	-9 250	069720	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	700		2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 32	5 200	069831	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	13 050	069716	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 9	16 550	090178	2,00	104
Replace insert	8,00	8,00	1,00	DLOB	Spiral floor	14 400	071617	2,00	16,0
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 4	24 400	069710	2,00	104
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	29 000	069711	2,00	104
Total per annum	**684**	**64,0**							**1 368**

Table C.2: Dose assessment routine operations – radiation protection general area level 2,00 μSv/h

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Replace insert	1,00	1,00	1,00	GCS	Gas supply block unit	-23 000	069724	2,00	2,00
Replace insert	1,00	1,00	1,00	GCS	Gas return block unit	-23 000	069724	2,00	2,00
Replace insert	1,00	1,00	1,00	GCS	Cleaning block unit	-23 000	069724	2,00	2,00
Replace insert	1,00	1,00	1,00		Diverter block unit	-23 000	069724	2,00	2,00
Replace insert	1,00	1,00	1,00	CBA	SCS	-18 800	069723	2,00	2,00
Replace insert	1,00	1,00	1,00	GCS	Gas filter block unit	-18 800	069723	2,00	2,00
Replace pump	1,00	1,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	2,00	2,00
Replace helium service unit LRU	1,00	1,00	1,00	AGS	Helium service unit	-15 000	069723	2,00	2,00
Replace gas ventilation unit LRU	1,00	1,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	2,00	2,00
Replace gas unit LRU	1,00	1,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	2,00	2,00
Calibration test	12,0	1,00	12,0	BUMS	BUMS isolation valve block	18 200	90178	2,00	24,0
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 13	-22 500	071617	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	2,00	208

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 8	-18 800	069723	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 10	-15 000	071848	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet x 1	-9 250	069720	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 3	700		2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 32	5 200	069831	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 3	13 050	069716	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 9	16 550	090178	2,00	208
Replace insert	1,00	1,00	1,00	CLOB		14 400	071617	2,00	2,00
Inspect instrumentation	104	2,00	52,0	FHSS instrumentation	Cabinet × 4	24 400	071617	2,00	208
Inspect instrumentation	104	2,00	52,0	FHSS instrumentation	Cabinet × 3	29 000	069711	2,00	208
Routine cleaning and surveillance	60	5,00	12,0	Plant wide				2,00	120
Total per annum	**1 331**	**41,0**							**2 662**

Table C.3: Dose maintenance technician – valves: general area level 2,00 µSv/h

Mission description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total task dose/ annum (µSv)
Replace insert	8,00	8,00	1,00	GCS	Gas supply block unit	-23 000	069724	2,00	16,0
Maintenance on PEA	16,00	8,00	2,00	GCS	Gas supply block unit	Maintenance laydown	069724	2,00	32,0
Bearing oil and seal check	2,00	0.500	4,00	GCS	Gas supply block unit	-23 000	069724	2,00	4,00
Replace insert	8,00	8,00	1,00	GCS	Gas return block unit	-23 000	069724	2,00	16,0
Maintenance on PEA	16,00	8,00	2,00	GCS	Gas return block unit	Maintenance laydown	069724	2,00	32,0
Bearing oil and seal check	2,00	0.500	4,00	GCS	Gas return block unit	-23 000	069724	2,00	4,00
Replace insert	8,00	8,00	1,00	GCS	Gas supply block unit	-23 000	069724	2,00	16,0
Maintenance on PEA	16,00	8,00	2,00	GCS	Cleaning block unit	Maintenance laydown	069724	2,00	32,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Cleaning block unit	-23 000	069724	2,00	4,00
Replace insert	8,00	8,00	1,00	GCS	Diverter block unit	-23 000	069724	2,00	16,0
Maintenance on PEA	16,0	8,00	2,00		Diverter block unit	Maintenance laydown	069724	2,00	32,0
Bearing oil and seal check	2,00	0,500	4,00		Diverter block unit	-23 000	069724	2,00	4,00
Replace insert	8,00	8,00	1,00	CBA	CBA	-18 800	071848	2,00	16,0

Mission description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total task dose/ annum (μSv)
Maintenance on PEA	16,0	8,00	2,00	CBA	CBA	-18 800	071848	2,00	32,0
Bearing oil and seal check	2,00	0,500	4,00	CBA	CBA	-18 800	071848	2,00	4,00
Replace insert	8,00	8,00	1,00	GCS	Gas filter block unit	-18 800	069723	2,00	16,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Gas filter block unit	-18 800	069723	2,00	32,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas filter block unit	-18 800	069723	2,00	4,00
Replace pump	8,00	8,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	2,00	16,0
Maintenance on pump	16,0	8,00	2,00	AGS	Auxiliary gas pump unit	-15 000	069723	2,00	32,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas pump unit	-15 000	069723	2,00	4,00
Replace helium service unit LRU	8,00	8,00	1,00	AGS	Helium service unit	-15 000	069723	2,00	16,0
Maintenance on LRU	16,0	8,00	2,00	AGS	Helium service unit	-15 000	069723	2,00	32,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Helium service unit		069723	2,00	4,00
Replace gas ventilation Unit LRU	8,00	8,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	2,00	16,0
Maintenance on LRU	16,0	8,00	2,00	AGS	Ventilation and vacuum service	Maintenance laydown	069720	2,00	32,0

Mission description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total task dose/ annum (μSv)
Bearing oil and seal check	2,00	0,500	4,00	AGS	Ventilation and vacuum service		069720	2,00	4,00
Replace gas unit LRU	8,00	8,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	2,00	16,0
Maintenance on LRU	16,0	8,00	2,00	AGS	Auxiliary gas SSS unit	Maintenance laydown	069720	2,00	32,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas SSS unit	-9 250	069720	2,00	4,00
Replace canister LRU	4,00	4,00	1,00	Canister storage	Canister	700		2,00	8,00
Maintenance on LRU	16,0	8,00	2,00			Maintenance laydown		2,00	32,0
Calibration and test	16,0	4,00	4,00	BUMS	BUMS isolation valve block	18 200	90178	2,00	32,0
Calibration & test	16,0	4,00	4,00	AMS	AMS	18 200	90178	2,00	32,0
Visual inspection valve	15,0	0,250	60,0	Gas supply manifold	Valves × 15	-23 000	069724	2,00	30,0
Visual inspection valve	10,0	0,250	40,0	Gas return manifold	Valves × 10	-23 000	069724	2,00	20,0
Visual inspection valve	3,00	0.250	12,0	Gas cleaning block	Valves × 3	-23 000	069724	2,00	6,00
Visual inspection valve	7,00	0,250	28,0	MBA 1	Valves × 7	+18 200	90178	2,00	14,0
Visual inspection valve	7,00	0,250	28,0	MBA 2	Valves × 7	+18 200	90178	2,00	14,0
Visual inspection valve	7,00	0,250	28,0	MBA 3	Valves × 7	+18 200	90178	2,00	14,0

Mission description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total task dose/ annum (µSv)
Visual inspection valve	9,00	0,250	36,0	Isolation block assembly	Valves × 9	+18 200	90178	2,00	18,0
Visual inspection valve	5,00	0,250	20,0	CLOB assembly	Valves × 5	+14 400	69716	2,00	10,0
Visual inspection valve	8,00	0,250	32,0	DLOB assembly	Valves × 8	+14 400	69716	2,00	16,0
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valves × 6	-18 800	071851	2,00	12,0
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valves × 6	-18 800	071851	2,00	12,0
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valves × 6	-18 800	071851	2,00	12,0
Visual inspection valve	6,00	0,250	24,0	CBA	Valves × 6	-15 000	71851	2,00	12,0
Scrap sphere samples	2,00	1,00	2,00	Room below CUD		-23 000	69702	2,00	4,00
GCS Hex inspect and test	8,00	2,00	4,00	GCS		-9 250	69720	2,00	16,00
AGS pump repair	4,00	4,00	1,00	Maintenance laydown	Pump	-23 000	69724	2,00	8,00
Pool water test and replenish	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	2,00	24,00
Storage tanks monitor	12,0	1,00	12,0	SSS service hall	Monitor	+9 550	91974	2,00	24,00
Visual piping inspection	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	2,00	24,00
Pool water cooling assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	2,00	24,00

Mission description	Total time/ annum (h)	Time (h) ST	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total task dose/ annum (µSv)
Pool conditioning and cleaning assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	2,00	24,00
SSS ventilation Assembly	24,0	2,00	12,0	SSS service hall	Valves	+9 550	91974	2,00	48,0
SSS blower Assembly	24,0	2,00	12,0	SSS service hall	Blower	+9 550	91974	2,00	48,0
Sphere unloading machines	24,0	2,00	12,0	SSS service hall	Unloading device	+9 550	91974	2,00	48,0
Sphere replenishment system valves check	4,00	1,00	4,00	Sphere replenishment	Valves	+29 000	69711	2,00	8,00
Sphere replenishment counters heck	4,00	1,00	4,00	Sphere replenishment	Counters	+29 000	69711	2,00	8,00
Other	120	40,0	3,00					2,00	240
Total per annum	**681**	**248**							**1 362**

Table C.4: Dose maintenance technician: I&C area dose rate 20,0 µSv/h

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace insert	4,00	4,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	GCS	Gas return block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	GCS	Cleaning block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	GCS	Diverter block unit	-23 000	069724	5,00	20,0
Replace insert	4,00	4,00	1,00	CBA	SCS	-18 800	069723	25,0	100,0
Replace insert	4,00	4,00	1,00	GCS	Gas filter block unit	-18 800	069723	5,00	20,0
Replace pump	4,00	4,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	20,0
Replace helium service unit LRU	4,00	4,00	1,00	AGS	Helium service unit	-15 000	069723	5,00	20,0
Replace gas ventilation LRU	4,00	4,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	5,00	20,0
Replace gas unit LRU	4,00	4,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	20,0
Calibration test	12,0	4,00	3,00	BUMS	BUMS isolation valve block	18 200	090178	5,00	60,0
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 13	-22 500	087774	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 12	-22 500	087774	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 12	-22 500	087774	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 8	-18 800	069723	5,00	260

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 10	-15 000	071848	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 1	-9 250	069720	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	700		5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 32	5 200	069831	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	13 050	069716	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 9	16 550	090178	5,00	260
Replace insert	8,00	8,00	1,00	DLOB	Spiral floor	14 400	071617	25,0	200
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 4	24 400	069710	5,00	260
Inspect instrumentation	52,0	1,00	52,0	FHSS I&C	Cabinet × 3	29 000	069711	5,00	260
Total per annum	**684**	**64,0**	**638**						**3 660**

Table C.5: Dose routine operations – radiation protection surveillance: area dose rate 20,0 µSv/h

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace insert	1,00	1,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	GCS	Gas return block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	GCS	Cleaning block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	GCS	Diverter block unit	-23 000	069724	5,00	5,00
Replace insert	1,00	1,00	1,00	CBA	SCS	-18 800	069723	25,0	25,0
Replace insert	1,00	1,00	1,00	GCS	Gas filter block unit	-18 800	069723	5,00	5,00
Replace pump	1,00	1,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	5,00
Replace helium service unit LRU	1,00	1,00	1,00	AGS	Helium service unit	-15 000	069723	5,00	5,00
Replace gas ventilation unit LRU	1,00	1,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	5,00	5,00
Replace gas unit LRU	1,00	1,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	5,00
Calibration test	12,0	1,00	12,0	BUMS	BUMS isolation valve block	18 200	90178	5,00	60,0
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 13	-22 500	071617	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	5,00	520

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 12	-22 500	071617	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 8	-18 800	069723	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 10	-15 000	071848	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 1	-9 250	069720	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 3	700		5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 32	5 200	069831	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 3	13 050	069716	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS I&C	Cabinet × 9	16 550	090178	5,00	520
Replace Insert	1,00	1,00	1,00	CLOB		14 400	071617	25,0	25,0
Inspect instrumentation	104	2,00	52,0	FHSS instrumentation	Cabinet × 4	24 400	071617	5,00	520
Inspect instrumentation	104	2,00	52,0	FHSS instrumentation	Cabinet × 3	29 000	069711	5,00	520
Routine cleaning and surveillance	60,0	5,00	12	Plant wide				5,00	300
Total	**1 331**	**41,0**	**659**						**6 695**

Table C.6: Dose maintenance technician – valves: area dose rate 20,0 μSv/h

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Replace insert	8,00	8,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Gas supply block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas supply block unit	-23 000	069724	5,00	10,0
Replace insert	8,00	8,00	1,00	GCS	Gas return block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Gas return block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas return block unit	-23 000	069724	5,00	10,0
Replace insert	8,00	8,00	1,00	GCS	Gas supply block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Cleaning block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Cleaning block unit	-23 000	069724	5,00	10,0
Replace insert	8,00	8,00	1,00	GCS	Diverter block unit	-23 000	069724	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Diverter block unit	Maintenance laydown	069724	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Diverter block unit	-23 000	069724	5,00	10,0
Burn replace insert	8,00	8,00	1,00	CBA	CBA	-18 800	071848	25,0	200

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Maintenance on PEA	16,0	8,00	2,00	CBA	CBA	-18 800	071848	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	CBA	CBA	-18 800	071848	25,0	50,0
Replace insert	8,00	8,00	1,00	GCS	Gas filter block unit	-18 800	069723	5,00	40,0
Maintenance on PEA	16,0	8,00	2,00	GCS	Gas filter block unit	-18 800	069723	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas filter block unit	-18 800	069723	5,00	10,0
Replace pump	8,00	8,00	1,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	40,0
Maintenance on pump	16,0	8,00	2,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas pump unit	-15 000	069723	5,00	10,0
Replace helium service unit LRU	8,00	8,00	1,00	AGS	Helium service unit	-15 000	069723	5,00	40,0
Maintenance on LRU	16,0	8,00	2,00	AGS	Helium service unit	-15 000	069723	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Helium service unit		069723	5,00	10,0
Replace gas ventilation unit LRU	8,00	8,00	1,00	AGS	Ventilation and vacuum service	-9 250	069720	5,00	40,0
Maintenance on LRU	16,0	8,00	2,00		Ventilation and vacuum service	Maintenance laydown	069720	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00				069720	5,00	10,0

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace gas unit LRU	8,00	8,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	40,0
Maintenance on LRU	16,0	8,00	2,00	AGS	Auxiliary gas SSS unit	Maintenance laydown	069720	5,00	80,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas SSS unit	-9 250	069720	5,00	10,0
Replace canister LRU	4,00	4,00	1,00	Canister storage	Canister	700		5,00	20,0
Maintenance on LRU	16,0	8,00	2,00			Maintenance laydown		5,00	80,0
Calibration and test	16,0	4,00	4,00	BUMS	BUMS isolation valve block	18 200	90178	5,00	90,0
Calibration and test	16,0	4,00	4,00	AMS	AMS	18 200	90178	5,00	90,0
Visual inspection valve	15,0	0,250	60,0	Gas supply manifold	Valve × 15	-23 000	069724	5,00	75,0
Visual inspection valve	10,0	0,250	40,0	Gas return manifold	Valve × 10	-23 000	069724	5,00	50,0
Visual inspection valve	3,00	0,250	12,0	Gas cleaning block	Valve × 3	-23 000	069724	5,00	15,0
Visual inspection valve	7,00	0,250	28,0	MBA 1	Valve × 7	+18 200	90178	25,0	175
Visual inspection valve	7,00	0,250	28,0	MBA 2	Valve × 7	+18 200	90178	25,0	175
Visual inspection valve	7,00	0,250	28,0	MBA 3	Valve × 7	+18 200	90178	25,0	175
Visual inspection valve	9,00	0,250	36,0	Isolation block assembly	Valve × 9	+18 200	90178	25,0	220

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (μSv/h)	Total individual primary task dose (μSv)
Visual inspection valve	5,00	0,250	20,0	CLOB assembly	Valve × 5	+14 400	69716	25,0	125
Visual inspection valve	8,00	0,250	32,0	DLOB assembly	Valve × 8	+14 400	69716	25,0	200
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	25,0	150
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	25,0	150
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	25,0	150
Visual inspection valve	6,00	0,250	24,0	CBA	Valve × 6	-15 000	71851	25,0	150
Scrap sphere samples	2,00	1,00	2,00	Room below CUD		-23 000	69702	45,0	90,0
GCS hex inspect and test	8,00	2,00	4,00	GCS		-9 250	69720	5,00	40,0
AGS pump repair	4,00	4,00	1,00	Maintenance laydown	Pump	-23 000	69724	5,00	20,0
Pool water test and replenish	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
Storage tanks monitor	12,0	1,00	12,0	SSS service hall	Monitor	+9 550	91974	5,00	60,0
Visual piping inspection	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
Pool water cooling assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Pool conditioning and cleaning assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
SSS ventilation assembly	24,00	2,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	120
SSS blower assembly	24,00	2,00	12,0	SSS service hall	Blower	+9 550	91974	5,00	120
Sphere unloading machines	24,00	2,00	12,0	SSS service hall	Unloading device	+9 550	91974	5,00	120
Sphere replenishment system valves check	4,00	1,00	4,00	Sphere replenishment	Valves	+29 000	69711	5,00	20,0
Sphere replenishment counters check	4,00	1,00	4,00	Sphere replenishment	Counters	+2 9000	69711	5,00	20,0
Other	120,0	40,0	3,00					5,00	600
Total per annum	**681**	**248,25**	**575**						**5 040**

Table C.7: Dose maintenance technician – valves: general area dose rate 5,00 µSv/h, area dose rate 80,0 µSv/h and task time 7,00 h

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace insert	7,00	7,00	1,00	GCS	Gas supply block unit	-23 000	A 0U JA01 RM 030 (069724)	5,00	35,0
Maintenance on PEA	14,0	7,00	2,00	GCS	Gas supply block unit	Maintenance laydown	A 0U JA01 RM 030 (069724)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas supply block unit	-23 000	A 0U JA01 RM 030 (069724)	5,00	10,0
Replace insert	7,00	7,00	1,00	GCS	Gas return block unit	-23 000	A 0U JA01 RM 030 (069724)	5,00	35,0
Maintenance on PEA	14,0	7,00	2,00	GCS	Gas return block unit	Maintenance laydown	A 0U JA01 RM 030 (069724)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas return block unit	-23 000	A 0U JA01 RM 030 (069724)	5,00	10,00
Replace insert	7,00	7,00	1,00	GCS	Gas supply block unit	-2 3000	A 0U JA01 RM 030 (069724)	5,00	35,0
Maintenance on PEA	14,0	7,00	2,00	GCS	Cleaning block unit	Maintenance laydown	A 0U JA01 RM 030 (069724)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Cleaning block unit	-23 000	A 0U JA01 RM 030 (069724)	5,00	10,0
Replace insert	7,00	7,00	1,00	GCS	Diverter block unit	-23 000	A 0U JA01 RM 030 (069724)	5,00	35,0
Maintenance on PEA	14,0	7,00	2,00	GCS	Diverter block unit	Maintenance laydown	A 0U JA01 RM 030 (069724)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Diverter block unit	-23 000	A 0U JA01 RM 030 (069724)	5,00	10,0

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Replace insert	7,00	7,00	1,00	CBA	CBA	-18 800	A 0U JA02 RM 021 (071851)	85,0	595
Maintenance on PEA	14,0	7,00	2,00	CBA	CBA	-18 800	A 0U JA02 RM 021 (071851)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	CBA	CBA	-18 800	A 0U JA02 RM 021 (071851)	85,0	170
Replace insert	7,00	7,00	1,00	GCS	Gas filter block unit	-18 800	A 0U JA02 RM 028 (069723)	5,00	35,0
Maintenance on PEA	14,0	7,00	2,00	GCS	Gas filter block unit	-18 800	A 0U JA02 RM 028 (069723)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	GCS	Gas filter block unit	-18 800	A 0U JA02 RM 028 (069723)	5,00	10,0
Replace pump	7,00	7,00	1,00	AGS	Auxiliary gas pump unit	-15 000	A 0U JA03 RM 127 (071517)	5,00	35,0
Maintenance on pump	14,0	7,00	2,00	AGS	Auxiliary gas pump unit	-15 000	A 0U JA03 RM 127 (071517)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas pump unit	-15 000	A 0U JA03 RM 127 (071517)	5,00	10,0
Replace helium service unit LRU	7,00	7,00	1,00	AGS	Helium service unit	-15 000	A 0U JA03 RM 127 (071517)	5,00	35,0
Maintenance on LRU	14,0	7,00	2,00	AGS	Helium service unit	-15 000	A 0U JA03 RM 127 (071517)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Helium service unit		A 0U JA03 RM 127 (071517)	5,00	10,0
Replace gas ventilation unit LRU	7,00	7,00	1,00	AGS	Ventilation and vacuum service	-9 250	A 0U JA04 RM 024 (069720)	5,00	35,0

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Maintenance on LRU	14,0	7,00	2,00	AGS	Ventilation and vacuum service	Maintenance laydown		5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Ventilation and vacuum service	-9 250	A 0U JA04 RM 024 (069720)	5,00	10,0
Replace gas unit LRU	7,00	7,00	1,00	AGS	Auxiliary gas SSS unit	-9 250	A 0U JA04 RM 024 (069720)	5,00	35,0
Maintenance on LRU	14,0	7,00	2,00	AGS	Auxiliary gas SSS unit	Maintenance laydown	A 0U JA04 RM 024 (069720)	5,00	70,0
Bearing oil and seal check	2,00	0,500	4,00	AGS	Auxiliary gas SSS unit	-9 250	A 0U JA04 RM 024 (069720)	5,00	10,0
Replace canister LRU	7,00	7,00	1,00	Canister storage	Canister	700	A OU JA06 RM002	5,00	20,0
Maintenance on LRU	14,0	7,00	2,00			Maintenance laydown	A OU JA06 RM002	5,00	70,0
Calibration and test	16,0	4,00	4,00	BUMS	BUMS isolation valve block	18 200	90178	5,00	80,0
Calibration and test	16,0	4,00	4,00	AMS	AMS	18 200	90178	5,00	80,0
Visual inspection valve	15,0	0,250	60,0	Gas supply manifold	Valve × 15	-23 000	A 0U JA01 RM 030 (069724)	5,00	75,0
Visual inspection valve	10,0	0,250	40,0	Gas return manifold	Valve × 10	-23 000	A 0U JA01 RM 030 (069724)	5,00	50,0
Visual inspection valve	3,00	0,250	12,0	Gas cleaning block	Valve × 3	-23 000	A 0U JA01 RM 030 (069724)	5,00	15,0
Visual inspection valve	7,00	0,250	28,0	MBA 1	Valve × 7	+18 200	90178	85,0	595

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Visual inspection valve	7,00	0,250	28,0	MBA 2	Valve × 7	+18 200	90178	85,0	595
Visual inspection valve	7,00	0,250	28,0	MBA 3	Valve × 7	+18 200	90178	85,0	595
Visual inspection valve	9,00	0,250	36,0	Isolation block assembly	Valve × 9	+18 200	90178	85,0	765
Visual inspection valve	5,00	0,250	20,0	CLOB assembly	Valve × 5	+14 400	69716	85,0	425
Visual inspection valve	8,00	0,250	32,0	DLOB assembly	Valve × 8	+14 400	69716	85,0	680
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	85,0	510
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	85,0	510
Visual inspection valve	6,00	0,250	24,0	Room next to CUD	Valve × 6	-18 800	071851	85,0	510
Visual inspection valve	6,00	0,250	24,0	CBA	Valve × 6	-15 000	71851	85,0	510
Scrap sphere samples	2,00	1,00	2,00	Room below CUD		-23 000	69702	45,0	90,0
GCS hex inspection and test	8,00	2,00	4,00	GCS		-9 250	69720	5,00	40,0
AGS pump repair	4,00	4,00	1,00	Maintenance laydown	Pump	-23 000	69724	5,00	20,0
Pool water test and replenish	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0

Mission description	Total time/ annum (h)	Time (h) (ST)	Frequency/ annum (FQ)	SSC	Component	Area description	Plant location (compartment number)	Gamma dose rate (µSv/h)	Total individual primary task dose (µSv)
Storage tanks monitor	12,0	1,00	12,0	SSS service hall	Monitor	+9 550	91974	5,00	60,0
Visual piping inspection	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
Pool water cooling assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
Pool conditioning and cleaning assembly	12,0	1,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	60,0
SSS ventilation assembly	24,0	2,00	12,0	SSS service hall	Valves	+9 550	91974	5,00	120
SSS blower assembly	24,0	2,00	12,0	SSS service hall	Blower	+9 550	91974	5,00	120
Sphere unloading machines	24,0	2,00	12,0	SSS service hall	Unloading device	+9 550	91974	5,00	120
Sphere replenishment system valves check	4,00	1,00	4,00	Sphere replenishment	Valves	+29 000	69711	5,00	20,0
Sphere replenishment counters check	4,00	1,00	4,00	Sphere replenishment	Counters	+29 000	69711	5,00	20,0
Other	138							5,00	690
Total per annum	**667**	**187**	**572**						**9 495**