Environmental Impact Assessment for a 400 MW(t) Pebble Bed Modular Reactor Demonstration Power Plant.

Specialist Study: Radioactive Waste Management
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Title: Environmental Impact Assessment for a 400 MW(t) Pebble Bed Modular Reactor Demonstration Power Plant

Specialist Study: Radiological Waste Management

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Offices

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Directors: J.J. van Blerk (Ph.D), H. Janse van Rensburg (Ph.D, MBA)
The Environmental Impact Assessment (EIA) process for the 400 MW(t) PBMR DPP can be separated into the Scoping Phase and the Impact Assessment Phase. During the Scoping Phase, several issues were identified for consideration in the Impact Assessment Phase.

The purpose of the radioactive waste management study is to address those issues identified during the Scoping Phase related to the management of the radioactive waste that will be generated during the operation and decommissioning of the PBMR DPP. The Terms of Reference for the study requires a description of the following:

- The sources, quantities, and level of radioactivity of all radiological waste (liquid, gaseous, and solid) estimated to be generated by the PBMR DPP, including a description of conventional and mixed waste, where applicable.

- The manner in which all the radiological waste (including tritium) is likely to be managed for the PBMR DPP based on the cradle to grave principle.

- How radiological waste may be processed and the potential for processing of radiological waste generated by the PBMR DPP.

- Current research being undertaken to reduce the volume of radiological waste generated by the PBMR DPP, including a comparison of the estimated volume of waste against the volume of waste generated by a generation 3 nuclear reactor technology.

- An estimate of the amount of low and intermediate level radioactive waste likely to be generated by the PBMR DPP and the source (clothing etc.) of this waste.

- The manner in which low and intermediate level radiological waste is currently transported to Vaalputs from the KNPS site.

- The manner in which low and intermediate level radiological waste from the PBMR DPP is intended to be transported to Vaalputs.

- The available capacity for low and intermediate level radiological waste disposal at Vaalputs.

- The manner in which low and intermediate radiological waste is disposed of at Vaalputs.

- International trends and policies with respect to the disposal of high-level radioactive waste;

- The South African policy and strategy on high-level radioactive waste and how
this policy compares with international policies in this respect.

— The manner in which high-level radiological waste is managed at the existing KNPS site.

— The proposed manner in which high-level radioactive waste from the PBMR DPP will be managed on-site.

— The manner in which nuclear fuel is currently transported to the KNPS site.

— The manner in which nuclear fuel is likely to be transported from the fuel manufacturing plant at Pelindaba to the proposed PBMR site.

To realise the objectives of the study and to fulfil the Terms of Reference, the report is structured as follows:

— Section 2 presents an overview of the nuclear regulatory framework governing the management of radioactive waste in South Africa, as defined by the National Policy and Strategy for Radioactive Waste Management, as well as an overview of the applicable regulations regarding safety standards and regulatory practices.

— Section 3 presents the elements of a Radioactive Waste Management Programme, as a framework for the management of radioactive waste generated at a nuclear power station. The discussion is generic and largely based on International Atomic Energy Agency (IAEA) guidelines presented in IAEA (2002a).

— Section 4 presents an overview of the characteristics of the radioactive waste that will be generated by the PBMR DPP. The discussion is divided into gaseous radioactive waste, liquid radioactive waste, solid radioactive waste, as well as other conventional and mixed PBMR DPP waste. The discussion covers the source (origin) of radioactive waste, quantity (volume) of waste, and level of radioactivity associated with the waste type, as far as possible.

— Section 5 provides an overview of the radioactive waste management practices envisaged being part of the Radioactive Waste Management Programme for the PBMR DPP, from generation to disposal. The discussion includes the management of gaseous waste and liquid waste at the PBMR DPP, as well as an overview of the management practices (e.g. storage and disposal) envisaged for low and intermediate level waste (LILW) and high-level waste (HLW). Where applicable, the discussion includes the processing (pre-treatment, treatment, or conditioning) of radioactive waste.

— Section 6 provides the international basis for the management of high-level waste. This overview serves then as basis to compare South Africa’s Radioactive Waste Management Policy and Strategy, with international trends and policies. The discussion includes an overview of the applicable articles contained in the Joint Convention on the Safety of Spent Fuel Management and the Safety of
Radioactive Waste Management (IAEA, 2006c), and some basic concepts for high level radioactive waste management from the international literature.

— Section 7 provides an overview of the manner in which nuclear fuel is currently transported to the KNPS, and the manner in which nuclear fuel is likely to be transported to the proposed PBMR DPP.

The main conclusions drawn from the study are:

— The PBMR DPP generates liquid, gaseous, and solid radioactive waste as by-products of operational conditions and decommissioning activities. The solid radioactive waste is divided further into compactable waste, non-compactable waste, abnormal waste and spent fuel. Waste other than radiological waste that will be generated can be divided into conventional and hazardous waste.

— Radioactive waste management practices envisaged for the PBMR DPP is consistent with the IAEA guidelines for a Radioactive Waste Management Programme for nuclear power stations, from generation to disposal.

— The PBMR DPP strives to minimize production of all solid, liquid, and gaseous radioactive waste, both in terms of volume and activity content, as required for new reactor designs. This is being done through appropriate processing, conditioning, handling, and storage systems. In addition, production of radioactive waste is minimized by applying good practices for radiological zoning, provision of active drainage and ventilation, appropriate finishes, and the use of current best practices for the handling of solid radioactive waste. Where possible, the PBMR DPP reuses or recycles materials. For example, slightly contaminated processed water rather than fresh water is used to minimize water consumption.

— Processing of gaseous and liquid waste is aimed at reducing activity levels in the reactor building and in effluent generated as part of operational conditions. It also ensures that radiation doses to members of the public due to discharges to the environment (i.e., controlled discharges) do not exceed a fraction of the dose limit for the public (dose constraint). For this purpose, Authorised Discharge Quantities (AADQ) are defined for these waste streams. Compliance monitoring will be done at the source and in the environment. Processing of solid waste is aimed at reducing the volume of waste (e.g., compaction), containing dispersible activity (e.g., immobilisation), or reducing the activity of abnormal waste (e.g. decontamination). The proposed processing and conditioning of solid waste are conducive to safe storage and consistent with the Vaalputs waste acceptance criteria.

— Systems are designed store processed solid radioactive waste for a period of up to three years within the facility. The storage containers are consistent with the requirements for the disposal of solid waste at the radioactive waste disposal facility at Vaalputs. The waste unsuitable for disposal at Vaalputs will be stored on site until a suitable facility is available.
The transfer and associated transport of the waste to Vaalputs will be done in conjunction with waste shipments from the KNPS. This will be done according to the appropriate provisions of the IAEA Regulations for the Safe Transport of Radioactive Material, subject to a graded approach. The objective of the Regulations is to protect persons, property, and the environment from the effects of radiation during the transport of radioactive material. In terms of the Regulations, the transport process is subject to radiation protection, emergency response, quality assurance, and compliance assurance programmes.

The concept for the disposal of solid waste at Vaalputs consists of near-surface trenches using metal containers for low-level waste, and concrete containers for intermediate level waste. The long-term safety of the facility, which complies with international best practices for the disposal of low and intermediate level waste, has been demonstrated for a national inventory of radioactive waste. The inventory derived for this purpose, included waste of 10 potential future PBMRs. Vaalputs therefore has more than enough capacity to dispose of the solid waste estimated to be generated by the PBMR DPP.

Processing of PBMR DPP spent fuel involves making the graphite more resistant to degradation due to radiation and thereby reducing the specific waste production rate. In addition, the intention is to reduce the volume of spent fuel by removing the graphite matrix and outer pyrolytic graphite of the coated particles. It is also the intention to remove radioactive contaminants and C-14 isotopes from the graphite to reuse the cleaned graphite for the production of fresh fuel.

The current ratio between the PBMR DPP and a Generation 3 nuclear reactor (PWR) of spent fuel storage space required (m$^3$) per unit of electrical energy produced, is in the order of 20 (i.e. for every 1 m$^3$ of storage space which a PWR like the KNPS requires, the PBMR DPP will require 20 m$^3$). If the volume minimization programme is successful, then the ratio PBMR:PWR would be reduced to 1 cubic metre per unit of electrical energy produced (PBMR, 2007).

The Fuel Handling and Storage System manage the storage of PBMR DPP spent fuel. This facility has sufficient capacity to safely store all the spent fuel produced throughout the life of the plant, and to store the spent fuel for a further 40 years after decommissioning if needed. It is thus only after 80 years that the storage facility on site (or elsewhere) will have to be upgraded to store and manage spent fuel. This should provide sufficient time to define and develop a long-term management strategy for the PBMR DPP spent fuel, e.g. a geological disposal facility or an alternative.

While reprocessing of spent fuel is not excluded as an option for spent fuel management, there is no intention to reprocess the PBMR DPP spent fuel at present. The main reason being the very high burn-up of uranium in the fuel sphere and the very high cost associated with spent fuel reprocessing.

The existing transport of fresh nuclear fuel to the KNPS and from Pelindaba to the
PBMR DPP site is subject to the provisions of the IAEA Regulations for the Safe Transport of Radioactive Material, subject to a graded approach.

— International trends and policies with respect to spent fuel and high-level waste management is based on the provisions of the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management. Internationally, this waste is currently being stored (usually above ground), awaiting the development of geological repositories. While the arrangements for storage have proved to be satisfactory and have been operated without problems, it is generally agreed that these arrangements are interim, and do not represent a final solution.

— The two basic challenges in perfecting a system of radioactive waste isolation is choosing an appropriate geological barrier (host medium), and designing an effective engineered barrier. Underground research laboratories made a very positive contribution to waste isolation research, while public acceptance of radioactive waste isolation projects remains one of the major challenges.

— The National Radioactive Waste Management Policy and Strategy is consistent with international practice for the management of high-level waste. However, additional, more detailed regulations are needed on specific issues relevant to long-term management and geological disposal of high-level waste. A summary of internationally accepted requirements for geological disposal have recently been established (IAEA, 2006b). These requirements should be supplemented from the experiences of several national programs that are within a decade of operating a geological repository for high-level waste and spent fuel, notably Finland, Sweden, and the USA.
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<td>Annual Authorized Discharge Quantities</td>
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<td>ACR</td>
<td>Authorisation Change Request</td>
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<td>ALARA</td>
<td>As Low As Reasonably Achievable</td>
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<td>AOO</td>
<td>Anticipated Operational Occurrences</td>
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<td>BATNEEC</td>
<td>Best Available Technology Not Entailing Excessive Cost</td>
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<td>PB</td>
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<td>Power Conversion Unit</td>
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<tr>
<td>SAR</td>
<td>safety analysis report</td>
</tr>
<tr>
<td>SSC</td>
<td>Structures, Systems, and Components</td>
</tr>
<tr>
<td>SSE</td>
<td>Safety Shutdown Earthquake</td>
</tr>
<tr>
<td>SSS</td>
<td>Sphere Storage Subsystem</td>
</tr>
<tr>
<td>Sv</td>
<td>Sieverts</td>
</tr>
<tr>
<td>SWS</td>
<td>Solid Waste System</td>
</tr>
<tr>
<td>WHS</td>
<td>Waste Handling System</td>
</tr>
<tr>
<td>WIPP</td>
<td>Waste Isolation Pilot Plant</td>
</tr>
</tbody>
</table>
1 INTRODUCTION

1.1 Background

The Eskom Conversion Act, 2001 (Act No. 13 of 2001) established Eskom Holdings Limited (Eskom) as a State Owned Enterprise, with the Government of South Africa as the only shareholder, represented by the Minister of Public Enterprises. According to the Memorandum of Association required by the Eskom Conversion Act and the Companies Act, 1973 (Act No. 61 of 1973), Eskom’s main objective is to "provide energy and related services including the generation, transmission, distribution and supply of electricity, and to hold interests in other entities." In accordance with this mandate, Eskom proposes to construct, commission, operate and decommission a 400 MW(t) Pebble Bed Modular Reactor Demonstration Power Plant (PBMR DPP) at the Koeberg Nuclear Power Station (KNPS) site in the Western Cape Province of South Africa (ARCUS GIBB, 2007).

The PBMR DPP includes a number of activities identified in terms of Section 21 (1) of the Environment Conservation Act, 73 of 1989, (the ECA) and as such, requires a written authorisation prior to commencement therewith. The process by which information on the project is assimilated, analysed and presented so as to inform an authorisation, is referred to as the Environmental Impact Assessment (EIA) process. The EIA for the proposed 400 MW(t) PBMR DPP commenced in August 2005 with the submission of an application for authorisation of the proposed development to the Department of Environmental Affairs and Tourism (DEAT) (ARCUS GIBB, 2007).

The National Nuclear Regulator Act (Act No. 47 of 1999) regulates nuclear activities. It established the National Nuclear Regulator (NNR) to exercise legislated nuclear regulatory control and assurance. The Nuclear Energy Act (Act No. 46 of 1999) is the leading legislation with regard to the governance of radioactive waste. DEAT, the lead authority on environmental matters, and NNR have agreed to work in close collaboration on the assessment of nuclear related matters associated with the PBMR DPP. A written and signed cooperative agreement was therefore established between DEAT and the NNR, which provides a framework within which DEAT will consult with the NNR on issues related to radiological aspects of the proposed PBMR DPP (ARCUS GIBB, 2007).

The International Atomic Energy Agency (IAEA) safety requirements on Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste, and Transport Safety (IAEA, 2000), state that: "Prior to the granting of an authorization\(^1\), the applicant shall be required to submit a detailed demonstration of safety, which shall be reviewed

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\(^1\) As used here, authorization refers to the granting by a nuclear regulatory body or other governmental body of written permission for an operator to perform specific nuclear activities (IAEA, 2007a), and not an environmental authorization process.
and assessed by the regulatory body in accordance with clearly defined procedures.”

The IAEA fundamental safety principles (IAEA, 2006a) states that the prime responsibility for safety rests with the person or organization responsible for facilities and activities that give rise to radiation risks (Principle 1). This responsibility includes (IAEA, 2006a):

- Ensuring the safe control of all radioactive material that is used, produced, stored or transported; and
- Ensuring the safe control of all radioactive waste that is generated.

The South African legislation and regulations are consistent with these requirements. According to Section 21(1) of the National Nuclear Regulator Act (Act No. 47 of 1999), Eskom is required to request nuclear authorisation in the form of a nuclear installation license from the NNR to site, construct, operate, decontaminate, or decommission the PBMR DPP. Eskom should submit or make available to the NNR, in accordance with agreed timescales, all information that is specified or requested. According to IAEA (2004), safety related information should be presented in the form of a safety analysis report (SAR). The SAR should contain accurate and sufficiently precise information on the plant and its operating conditions in such a way that the NNR will be able to evaluate independently the safety of the plant. This information should include, amongst others (IAEA, 2004):

- A justification of the adequacy of the measures proposed for the safe management of radioactive waste of all types that is generated throughout the lifetime of the plant;
- Measures to control or contain the waste produced at all stages of the lifetime of the plant;
- Measures to safely handle waste of all types produced during all stages of the lifetime of the plant, including provisions for the safe handling of the generated waste while transporting it from the point of origin to the specified storage point;
- Measures to minimize the accumulation of waste produced at all stages of the lifetime of the plant, including measures taken to reduce the waste arising to a level that is as low as practicable;
- Measures to condition\(^2\) the waste produced at all stages of the lifetime of the plant;
- Measures to store the waste produced at all stages of the lifetime of the plant,

\(^2\)Those operations that produce a waste package suitable for handling, transport, storage, and/or disposal. Conditioning may include the conversion of the waste to a solid waste form, enclosure of the waste in containers and, if necessary, provision of an overpack (IAEA, 2007a).
including the quantities, types and volumes of radioactive waste and the need to categorize and separate waste within the provisions for storage; and

— Measures to safely dispose of the waste produced at all stages of the lifetime of the plant, including the measures for ensuring the safe transport of waste to another specified location for longer term storage, if necessary.

1.2 Location of the Proposed Site

The Koeberg Nuclear Power Station (KNPS) site is located within the Eskom Controlled Area on the farm Duynefontein, within the Koeberg Private Nature Reserve, the latter being approximately 3,000 ha in extent. The KNPS site is approximately 2 km from the Duynefontein residential area, 30 km north of Cape Town, and 10 km south of Atlantis, within the City of Cape Town Metropolitan Municipality jurisdiction (see Figure 1). The PBMR DPP is proposed to be located some 400 m south of the existing Koeberg Power Station, inside the access control 1 security fence of the KNPS site (see Figure 2). The PBMR DPP would require approximately 9 ha of the KNPS site, which is approximately 125 ha in extent. The site and surrounding nature reserve are managed according to a formal integrated environmental management system (ARCUS GIBB, 2007).

1.3 Purpose of the Study

The EIA process can be separated into two phases namely the Scoping Phase and the Impact Assessment Phase. During the Scoping Phase for the 400 MW(t) PBMR DPP, several issues were identified for consideration in the Impact Assessment Phase. Some of those issues are related to the management of radioactive waste that will be generated during the operation and decommissioning of the PBMR DPP.

The purpose of this study is to address these radioactive waste management issues identified for the PBMR DPP, in a consistent and objective manner.

1.4 Terms of Reference of the Study

While some of the information presented in this report may meet the requirements for nuclear authorisation in terms of the National Nuclear Regulator Act (Act No. 47 of 1999), it is not the intention of the report to meet these requirements. In addition, some of the information presented in this report may be similar to those required as part of the SAR. However, this specialist study is being conducted as part of the EIA for the PBMR DPP and not to fulfil the requirements of the SAR. In accordance with the precautionary principle, and in the absence of actual data, estimates of the quantities of effluent generated has been based on the emission limits as published in Regulation R.388 promulgated in terms of the National Nuclear Regulator Act (Act No. 47 of 1999).
Figure 1  Regional location of the KNPS.

Figure 2  Proposed location of the PBMR DPP on the KNPS site.
Radioactive waste management is a broad term related to all administrative and operational activities involved in the handling, pre-treatment, treatment, conditioning (processing), storage and disposal of radioactive waste (IAEA, 2007a). This document will not address all these activities for the PBMR DPP in detail. Instead, the scope of the document is limited to specific radioactive waste management issues identified during the Scoping Phase for the PBMR DPP and which are of relevance to the EIA.

The nature of the radioactive waste management practices to be employed is directly related to the characteristics of the waste to be managed. The document consequently provides a description of the sources, quantities, and level of radioactivity of all radiological waste (liquid, gaseous, and solid) to be generated by the PBMR DPP. While the focus of the document is on radioactive waste, the description includes what is termed conventional and mixed waste, where applicable.

The document includes a description of the manner in which all the radiological waste, including tritium, is likely to be managed for the PBMR DPP based on the cradle to grave principle (i.e. from generation to disposal). From this perspective, the document includes a description of:

- how radiological waste may be processed and the potential for processing of radiological waste generated by the PBMR DPP; and

- the current research being undertaken to reduce the volume of radiological waste generated by the PBMR DPP. The document includes a comparison of the estimated volume of waste against the volume of waste generated by a generation 3 nuclear reactor technology. The comparison is made on a per unit energy produced basis.

According to the National Radioactive Waste Management Policy and Strategy (DME, 2005), the Vaalputs site, located in the Northern Cape Province of South Africa is, and will continue to be used as the National Disposal Site for low and intermediate level waste (LILW). The bulk of the LILW currently disposed of at Vaalputs, originates from the KNPS. The document includes:

- An estimate of the amount and the source (clothing etc.) of low and intermediate level radioactive waste likely to be generated by the PBMR DPP;

- A description of the manner in which low and intermediate level radiological waste is currently transported to Vaalputs from the KNPS site;

- A description of the manner in which low and intermediate level radiological waste from the PBMR DPP is intended to be transported to Vaalputs;

- A description of the available capacity for low and intermediate level radiological waste disposal at Vaalputs; and
A description of the manner in which low and intermediate radiological waste is disposed of at Vaalputs.

The feasibility of safely storing all radioactive waste (including high-level waste) over periods of decades has been clearly demonstrated during the operation of existing facilities (IAEA, 2003a) and are applied equally successfully in South Africa. According to DME (2005), the disposal of radioactive waste is regarded as the ultimate step in the national radioactive waste management strategic framework, although a step-wise waste management approach is acceptable. The disposal of high-level waste, however, presents a major challenge nationally and internationally. The document consequently includes:

- A description of international trends and policies with respect to the disposal of high-level radioactive waste;
- A description of the South African policy and strategy on high-level radioactive waste and how this policy compares with international policies;
- A description of the manner in which high-level radiological waste is managed at the existing KNPS site; and
- A description of the proposed manner in which high-level radioactive waste from the PBMR DPP will be managed on-site.

In terms of national legislation, the transport of nuclear fuel is subject to the IAEA Regulations for The Safe Transport of Radioactive Material (IAEA, 2005). The document includes a description of:

- the manner in which nuclear fuel is currently transported to the KNPS site; and
- the manner in which nuclear fuel is likely to be transported from the fuel manufacturing plant at Pelindaba to the proposed PBMR site.

### 1.5 Structure of the Report

To realise the objectives of the study and to fulfil the terms of reference as outlined in Section 1.4, the report is structured as follows:

- Section 2 presents an overview of the nuclear regulatory framework governing the management of radioactive waste in South Africa, as defined by the National Radioactive Waste Management Policy and Strategy (DME, 2005), and the applicable regulations regarding safety standards and regulatory practices.
- Section 3 presents a theoretical overview of the content of a typical Radioactive Waste Management Programme for a nuclear power plant.
- Section 4 presents an overview of the characteristics of the gaseous, liquid, solid,
and other radioactive waste that will be generated by the PBMR DPP.

— Section 5 provides an overview of the radioactive waste management practices envisaged to being part of the radioactive waste management programme for the PBMR DPP, from generation to disposal.

— Section 6 provides the international basis for the management of high-level waste. This summary then serves as the basis of comparison between South Africa’s Radioactive Waste Management Policy and Strategy and international trends and policies.

— Section 7 provides an overview of the manner in which nuclear fuel is currently transported to the KNPS and the manner in which nuclear fuel is likely to be transported to the proposed PBMR DPP.
2 Nuclear Regulatory Framework for the Management of Radioactive Waste

2.1 General

The NNR regulates all nuclear activities in terms of its establishment in the National Nuclear Regulator Act (Act No. 47 of 1999), including the management of radioactive waste. In 2006, new regulations regarding safety standards and regulatory practices as required by the National Nuclear Regulator Act (Act No. 47 of 1999), were Gazetted (Regulation No. R.388 dated 28 April 2006). The main purpose of these regulations is to protect persons, property, and the environment against nuclear damage.

In terms of the Nuclear Energy Act (Act No. 46 of 1999) the authority over radioactive waste and irradiated fuel waste is the Minister of Minerals and Energy. In 2005, the Department of Minerals and Energy (DME) published the National Radioactive Waste Management Policy and Strategy (DME, 2005). The purpose of the policy and strategy document is:

“To ensure the establishment of a comprehensive radioactive waste governance framework by formulating, additional to nuclear and other applicable legislation, a policy and implementation strategy in consultation with all stakeholders.”

The principles for the management of radioactive waste in South Africa are also contained in the National Radioactive Waste Management Policy and Strategy (DME, 2005).

The purpose of this section is to provide an overview of the nuclear regulatory framework governing the management of radioactive waste in South Africa, as defined by the National Radioactive Waste Management Policy and Strategy (Section 2.2), and the regulations regarding safety standards and regulatory practices (Section 2.3).

2.2 National Radioactive Waste Management Policy and Strategy

2.2.1 General

The emphasis of the policy and strategy document is on the nuclear industry (DME, 2005), within which the management of radioactive waste is a national responsibility assigned to the Minister of Minerals and Energy as per the Nuclear Energy Act (Act No. 46 of 1999).
Other drivers for the management of radioactive waste are the Constitution, the National Environmental Management Act (Act No. 107 of 1998) and the National Nuclear Regulator Act (Act No. 47 of 1999). The scope of the policy and strategy document relates to all radioactive wastes, excluding operational radioactive liquid and gaseous effluent (waste discharges) that is permitted to be released to the environment routinely under the authority of the relevant regulators. The policy and strategy thus serves as a national commitment to address solid radioactive waste management in a coordinated and cooperative manner (DME, 2005).

2.2.2 International Radioactive Waste Management Policy Principles

The international community, through the IAEA, has developed a comprehensive set of principles for the safe management of radioactive waste (IAEA, 1995). These basic principles, summarised in Table 1, are applicable to all countries and can be applied to all types of radioactive waste, regardless of its physical and chemical characteristics or origin3.

Table 1 Summary of the IAEA radioactive waste management principles (IAEA, 1995).

<table>
<thead>
<tr>
<th>Principle</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Protection of Human Health</td>
<td>Radioactive waste shall be managed in such a way as to secure an acceptable level of protection for human health</td>
</tr>
<tr>
<td>Protection of the Environment</td>
<td>Radioactive waste shall be managed in such a way as to provide an acceptable level of protection of the environment</td>
</tr>
<tr>
<td>Protection Beyond National Borders</td>
<td>Radioactive waste shall be managed in such a way as to assure that possible effects on human health and the environment beyond national borders will be taken into account</td>
</tr>
<tr>
<td>Protection of Future Generations</td>
<td>Radioactive waste shall be managed in such a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today</td>
</tr>
<tr>
<td>Burden on Future Generations</td>
<td>Radioactive waste shall be managed in such a way that will not impose undue burdens on future generations</td>
</tr>
<tr>
<td>National Legal Framework</td>
<td>Radioactive waste shall be managed within an appropriate national legal framework, including clear allocation of responsibilities and provision for independent regulatory functions</td>
</tr>
<tr>
<td>Control of Radioactive Waste Generation</td>
<td>Generation of radioactive waste shall be kept to the minimum practicable</td>
</tr>
<tr>
<td>Radioactive Waste Generation and Management Interdependencies</td>
<td>Interdependencies among all steps in radioactive waste generation and management shall be appropriately taken into account</td>
</tr>
<tr>
<td>Safety of Facilities</td>
<td>The safety of facilities for radioactive waste management shall be appropriately assured during their lifetime</td>
</tr>
</tbody>
</table>

3 In 2006, the IAEA Fundamentals Safety Principles were published (IAEA, 2006a), combining three safety fundamental publications, including the publication on the safety of radioactive waste management from which these principles were drawn. A summary of the IAEA Fundamentals Safety Principles is presented in Section 6.3.
2.2.3 National Radioactive Waste Management Policy Principles

All radioactive waste management activities in South Africa should be managed in accordance with the set of national principles summarised in Table 2.

2.2.4 Responsibilities

The prime responsibility for safety, including the safety of radioactive waste management, rests with the person or organisation responsible for facilities and activities that give rise to the radiation risks.

Table 2 Summary of the national principles for the management of radioactive waste in South Africa (DME, 2005).

<table>
<thead>
<tr>
<th>Principle</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Polluter pays principle</td>
<td>The financial burden for the management of radioactive waste shall be borne by the generator of that waste</td>
</tr>
<tr>
<td>Transparency regarding all aspects of radioactive waste management</td>
<td>All radioactive waste management activities shall be conducted in an open and transparent manner and the public shall have access to information regarding waste management where this does not infringe on the security of radioactive material</td>
</tr>
<tr>
<td>Sound decision-making based on scientific information, risk analysis and optimisation of resources</td>
<td>Decision-making shall be based on proven scientific information and recommendation of competent national and international institutions dealing with radioactive waste management</td>
</tr>
<tr>
<td>Precautionary principle</td>
<td>Where there are threats of serious irreversible damage, lack of full scientific certainty shall not be used as a reason for postponing cost-effective measures to prevent environmental degradation</td>
</tr>
<tr>
<td>No import nor export of Radioactive waste</td>
<td>In principle South Africa will neither import nor export radioactive waste</td>
</tr>
<tr>
<td>Co-operative governance and efficient national co-ordination</td>
<td>Due to their crosscutting nature all activities involving radioactive waste management shall be managed in a manner that prevents duplication of effort and maximises coordination</td>
</tr>
<tr>
<td>International cooperation</td>
<td>The government recognises that it shares a responsibility with other countries for global and regional radioactive waste management issues. Its actions shall follow the principles in this policy and in relevant regional and international agreements</td>
</tr>
<tr>
<td>Public Participation</td>
<td>Radioactive waste management shall take into account the interests and concerns of all interested and affected parties, when decisions are being made</td>
</tr>
<tr>
<td>Capacity building and education</td>
<td>The government shall create opportunities to develop people’s understanding, skills and general capacity concerning radioactive waste management</td>
</tr>
</tbody>
</table>

The role of government is to establish and sustain an effective legal and governmental framework for safety, including an independent regulatory body (IAEA, 2006a). Consistent with these principles, the policy and strategy defines the responsibilities of government and regulatory bodies in terms of national legislation. The responsibilities of the generators of radioactive waste, or operators of radioactive waste disposal facilities, are of particular interest and include:
— the technical, financial and administrative management of such wastes within the national regulatory framework and within any applicable co-operative governance arrangements;

— development and ongoing review of site / industry specific waste management plans which are to be based on the national radioactive waste management policy & strategy;

— execution of waste management plans by the establishment of appropriate waste management facilities and processes and the development of site / industry specific waste management systems; and

— site / industry waste management in accordance with waste management systems to reflect sustainable development and principles such as continued improvement and Best Available Technology Not Entailing Excessive Cost (BATNEEC) and other elements of the national strategy.

The responsibility of the generators of radioactive waste, or operators of radioactive waste disposal facilities, as the case may be, will be terminated upon closure of the disposal facility at which time institutional control (where required) will commence.

2.2.5 Definition and Classification of Radioactive Waste

For the purposes of implementing the National Radioactive Waste Management Policy and Strategy, South Africa follows the IAEA guidelines regarding the definition and classification of radioactive waste (unless deviations therefrom can be justified). A summary of the waste classification scheme adopted for this purpose is presented in Appendix A4.

2.2.6 Radioactive Waste Management Strategy Principles

This section of the National Radioactive Waste Management Policy and Strategy sets forward specific principles as strategic points of reference for the implementation of a radioactive waste management strategy. Some principles are general, such as the principle of reasonable consensus (for a course of action), or the principles for the development of a new course of action.

Others are more specific to waste management strategies and related to issues such as passive safety, regulatory requirements, hierarchy of waste management options, institutional control, retrievability, transfer of waste, and the dilution of waste.

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4 Note that the IAEA is in the process of revising the radioactive waste classification scheme (Draft IAEA Safety Guide 390), which includes six classes of waste. As a Member State, South Africa is in the process of reviewing the scheme.
2.2.7 Management Structures for Radioactive Waste

In terms of the National Radioactive Waste Management Policy and Strategy, the South African Government has the responsibility to establish appropriate structures for the management of radioactive waste at a national level. For this purpose, two entities will be established:

— the National Committee on Radioactive Waste Management (NCRWM), which will oversee the implementation of the policy and strategy; and

— the National Radioactive Waste Management Agency (NRWMA), which will be responsible for the national management of all radioactive waste disposal.

The national process for implementing the radioactive waste management strategy will be coordinated by the NCRWM. For this purpose, appropriate waste management plans will have to be developed and submitted to the NCRWM for approval.

2.2.8 Financial Provision for Radioactive Waste Management

According to the policy and strategy, government will establish a Radioactive Waste Management Fund (RWMF) by statute. The purpose of the fund shall be to ensure that there are sufficient provisions for the long-term management options of the various waste forms. These shall include:

— fees for disposal activities;

— research and development activities including investigations into waste management/disposal options;

— capacity building initiatives for radioactive waste management/disposal; and

— fees for other activities related to radioactive waste management/disposal.

In keeping with the polluter pays principle, the contributions to the fund will be from the generators of radioactive waste. Generators should enter into an agreement with the RWMF for managing long-term provisions for institutional control measures.

2.2.9 National Radioactive Waste Management Model

The national radioactive waste management model recognises all steps in the radioactive waste management process, from waste generation to the main waste management endpoints and institutional control. The following steps, in particular, are addressed:

— radioactive waste generation;
— pre-disposal management of radioactive waste;

— radioactive waste management options; and

— radioactive waste management end-points.

The main radioactive waste management end-points correspond with the waste management options and may be regarded as the outcome of a specific waste management option. Regulated disposal requires continued regulation of the disposal site for a predetermined period where after the site should be placed under institutional control.

2.2.10 Long-Term Radioactive Waste Management Issues

There are two long-term radioactive waste management options employed in South Africa at present:

— above ground disposal in engineered facilities for the bulk of the mining waste; and

— near surface disposal for low and intermediate level radioactive waste at Vaalputs in the Northern Cape province of South Africa.

According to the National Radioactive Waste Management Policy and Strategy, Vaalputs will continue to be used as a National Disposal Site for the disposal of low and intermediate level radioactive waste.

Spent fuel is currently managed through two mechanisms in South Africa: dry and wet storage. Spent fuel from the KNPS is stored in authorised fuel pools at the KNPS site, as well as in casks designed and constructed for the storage of spent fuel. There is enough storage capacity at the KNPS site for the current operational lifetime of Koeberg (DME, 2005). Spent fuel from the SAFARI research reactor at Pelindaba is stored at an authorised dry storage facility as well as in the reactor pool on the Pelindaba site, which is the headquarters of the South African Nuclear Energy Corporation (Necsa).

According to the National Radioactive Waste Management Policy and Strategy, the Government should initiate investigations into the best long-term option for the management of spent fuel. In the interim, spent fuel is, and should continue to be, stored in authorized facilities within the generator’s sites.

5 Control of a waste site (for example, disposal site) by an authority or institution designated under the laws of a country or state. This control may be active (monitoring, surveillance, remedial work) or passive (land use control) and may be a factor in the design of a nuclear facility (for example, near surface disposal facility).
The National Radioactive Waste Management Policy and Strategy recognises that the storage of spent fuel on these sites is finite and not sustainable indefinitely. Government should thus ensure that investigations are conducted within set timeframes to consider the various options for safe management of spent fuel and high-level radioactive waste in South Africa. Included in the options for investigation should be the following:

— Long-term above ground storage at an off-site facility. This is a consideration although it may not be in line with some of the principles for radioactive waste management. The strength of this option is that if technologies that are more appropriate are developed in future, then the waste can be dealt with using those technologies. Storing above ground indefinitely may, however, result in an undue burden on future generations.

— Reprocessing, conditioning, and recycling. An investigation commissioned by the DME has concluded that it would not be advisable to exclude the reprocessing, conditioning, and recycling of spent fuel (DME, 2005). This option is sometimes associated with proliferation concerns. However, as South Africa has concluded the IAEA Safeguards Agreements and the Additional Protocol, this should not be an issue for South Africa.

— Geological disposal. Internationally, geological disposal is currently the most pursued option and as such will require very careful consideration. This option has been under investigation outside of South Africa for the best part of a decade and as such investigations in South Africa should commence as soon as possible. If chosen as a preferred option in South Africa, geological disposal of radioactive waste should take place with an option for retrieving the waste. (The reason for this is to not rule out the possibility of the use of future technology for better management options).

— Transmutation. A fourth option (Transmutation) has been - and continues to be - investigated in a number of countries. However, it has not been proven to be a workable solution and also requires major investment in technology. The Government will continue to monitor developments internationally, but this option will probably not be investigated in South Africa in the near future.

The National Radioactive Waste Management Policy and Strategy (DME, 2005) indicates that the choice of the most suitable option should take due cognisance of the policy principles and should clearly demonstrate how the option satisfies the national policy objectives. All conclusions on investigations should be subject to public scrutiny.

2.3 Regulations on Safety Standards and Regulatory Practices

2.3.1 General

A nuclear regulatory guidance or requirements document aimed specifically at
management of radioactive waste is currently not available in the South African context. The best information available and which the South African industry currently prescribe to, is IAEA requirements and safety guides in the safety standard series related to the pre-disposal management of radioactive waste, storage of radioactive waste, transport of radioactive material, and regulatory control of discharge to the environment.

In South Africa, the Regulations on Safety Standards and Regulatory Practices (Regulation No, R.388 dated 28 April 2006) (Hendricks, 2006), however, provide the necessary standards and principles that should be met to ensure safety in any nuclear installation, including the safety of radioactive waste management. The Basic Licensing Requirements for the Pebble Bed Modular Reactor (NNR, 2007a) are based on and are established to fulfil these principles.

The requirements of the Regulations on Safety Standards and Regulatory Practices (Hendricks, 2006) with respect to radioactive waste management are summarised in Section 2.3.2 to Section 2.3.5 below.

2.3.2 Principal Radiation Protection and Nuclear Safety Requirements

The principal radiation protection and nuclear safety requirements of the regulations apply to actions authorised by, or seeking authorisation in terms of, a nuclear installation license, a nuclear vessel license, or certificate of registration. The application of these requirements should be commensurate with the characteristics of the action and with the magnitude and likelihood of exposure, as determined in a safety assessment. Not all the requirements are relevant to all actions. Requirements contained in the standards are related to:

- dose limits to an individual and risk limits of fatality from actions;
- optimisation of radiation protection and nuclear safety in terms of the ALARA principle;
- measures to control the risk of nuclear damage to individuals must be determined based on a prior safety assessment;
- good engineering practices;
- maintaining and fostering a safety culture;
- retrospective application of the regulations;

6 Authorised action means an action authorised in terms of the National Nuclear Regulator Act (Act No. 47 of 1999).
— regulatory approval of radiation protection and nuclear safety measures;

— accident management and emergency planning, emergency preparedness, and emergency response;

— a multilayer (defence in depth) system of provisions for radiation protection and nuclear safety; and

— establishing, implementing and maintaining a quality management programme.

2.3.3 Requirements Applicable to Regulated Actions

These requirements apply to actions authorised by a nuclear installation license, nuclear vessel license, or certificate of registration, and include the following:

— conductance, submission, and maintaining of operational safety assessments to the regulator;

— controls and limitations on operations, as established in the safety assessments (e.g. radioactive waste acceptance criteria in respect of waste disposal or storage facilities);

— establishing, implementing, and maintaining a maintenance and inspection programme;

— competency and qualification of staff responsible for radiation protection and nuclear safety and for maintaining an appropriate safety culture;

— optimisation of radiation protection in terms of the ALARA principle; application of a dose constraint, annual authorised discharge quantities, radiation dose limitation, medical surveillance and health register, and a dose register;

— establishing, implementing and maintaining a radioactive waste management programme, including the safe storage of radioactive waste and the removal of radioactive material, radioactively contaminated material or radioactive waste;

— establishing, implementing, and maintaining an appropriate environmental monitoring and surveillance programme to verify that the storage, disposal or effluent discharge of radioactive waste complies with the conditions of the nuclear authorisation;

— transport of radioactive material off the site or on any other road accessible to the public in terms of the provisions of the IAEA Regulations for The Safe Transport of Radioactive Material;

— establishing, implementing and maintaining physical security arrangements;
establishing, implementing and maintaining a system of records specified in the nuclear authorisation;

— monitoring of workers in the workplace; and

— occupational exposure to radon.

2.3.4 Decommissioning

These requirements apply to actions authorised by a nuclear installation license, nuclear vessel license, or certificate of registration, which involves the decommissioning of any nuclear installation, plant, or equipment having an impact on radiation protection and nuclear safety, or the release of radioactively contaminated land for other uses. These include the following:

— development and submission of a decommissioning strategy and plan to the Regulator;

— availability of sufficient resources from the time of cessation of the operation to the termination of the period of responsibility;

— conduct decommissioning operations in compliance with requirements listed in Section 2.3.3;

— release of radioactively contaminated land; and

— obligations under other statutes.

2.3.5 Accidents, Incidents and Emergencies

These requirements are applicable to emergency exposure situations requiring protective action to reduce or avoid temporary exposure, and include the following:

— criteria for the definition of a nuclear accident;

— criteria for the definition of a nuclear incident;

— information to be supplied to the Regulator in case of a nuclear accident or incident; and

— emergency or remedial measures to be considered near a nuclear accident.
3 Radioactive Waste Management Programme

3.1 General

IAEA (2007a) defines radioactive waste as material, whatever its physical form, remaining from practices or interventions for which no further use is foreseen:

- that contains or is contaminated with radioactive substances and has an activity or activity concentration higher than the level of clearance from regulatory requirements; and

- exposure to which is not excluded from the IAEA Basic Safety Standards published in IAEA (1996).

An application for a nuclear installation license requires, amongst others, the development of a Radioactive Waste Management Programme (RWMP). This is consistent with the NNR Basic Licensing Requirements for the Pebble Bed Modular Reactor (NNR, 2007a) and the NNR Guideline for Applying for a Nuclear Authorisation (NNR, 2007b).

Some measures pertaining to a RWMP were listed in Section 1.1, including the need to ensure that the resultant radioactive waste meets the requirements for safe handling, transport, processing, storage, and disposal, as applicable to national regulations, and international requirements and recommendations. More specifically, the RWMP should make provision for (IAEA, 2002a):

- keeping the generation of radioactive waste to the minimum practicable, in terms of both activity and volume, by using suitable technology;

- reusing and recycling materials to the extent possible;

- classifying and segregating waste appropriately, and maintaining an accurate inventory for each radioactive waste stream, with account taken of the available options for clearance and disposal;

- collecting, characterizing and storing radioactive waste so that it is acceptably safe;

- providing adequate storage capacity for anticipated radioactive waste arisings;

- ensuring that radioactive waste can be retrieved at the end of the storage period;

- treating and conditioning radioactive waste in a way that is consistent with safe storage and disposal;
— handling and transporting radioactive waste safely;
— controlling effluent discharges to the environment;
— carrying out monitoring for compliance at source and in the environment;
— maintaining facilities and equipment for waste collection, processing and storage in order to ensure safe and reliable operation;
— monitoring the status of the containment for the radioactive waste in the storage location;
— monitoring changes in the characteristics of the radioactive waste, in particular if storage is continued for extended periods, by means of inspection and regular analysis;
— initiating, as necessary, research and development to improve existing methods for processing radioactive waste or to develop new methods, and to ensure that suitable methods are available for the retrieval of stored radioactive waste.

While the national nuclear regulatory framework does not provide specific requirements for the management of radioactive waste, from generation to disposal, it does require the establishment, implementation, and maintenance of a RWMP. The purpose of this section is to elaborate further on elements of a RWMP, as a framework for the management of radioactive waste generated at a nuclear power plant.

The discussion is generic and largely based on guidance presented in IAEA (2002a). The RWMP for the PBMR DPP is still being compiled and will form part of the SAR submitted to the NNR in support of the application for nuclear authorisation (NNR, 2007b) in terms of the National Nuclear Regulator Act (Act No. 47 of 1999). Note however, that the NNR Licensing requirements for the PBMR do not contain specific requirements of what should be included in the RWMP.

3.2 Generation of Radioactive Waste

3.2.1 General

Nuclear power plants generate gaseous, liquid, and solid radioactive waste as by-products of their operations. The nature and the amounts of such waste will depend on the type of reactor, specific design features, operating procedures and practices, including maintenance, refuelling and operational occurrences, the operational history of the plant, and the integrity of the fuel. IAEA (2002a) requires that measures to keep the generation of gaseous, liquid, and solid waste to the minimum practical should be defined within the RWMP and implemented.
3.2.2 Gaseous Radioactive Waste

Although the sources of gaseous radioactive waste differ according to the type of reactor, possible gaseous sources from nuclear reactors include leakage from the coolant, the moderator systems, or the reactor itself; degasification systems for the coolant; condenser vacuum air ejectors or pumps; the exhaust from turbine gland seal systems; and activated or contaminated ventilated air. In all cases, spent fuel in storage or in handling operations is a potential source of gaseous radioactive waste.

3.2.3 Liquid Radioactive Waste

Although the composition of the liquid radioactive waste may vary appreciably according to reactor type, contributions to the waste stream may derive from reactor coolant let-down, evaporator concentrates, equipment drains, floor drains, laundry waste, contaminated oil and waste arising from the decontamination and maintenance of facilities and equipment.

3.2.4 Solid Radioactive Waste

Solid radioactive waste results from the operation and maintenance of the nuclear power plant and its associated processing systems for gaseous and liquid radioactive waste. The nature of such waste and the associated levels of activity vary from plant to plant. Solid radioactive waste consists of: spent ion exchange resins (both bead and powder); cartridge filters and pre-coat filter cake; particulate filters from ventilation systems; charcoal beds; tools; contaminated metal scrap; core components; debris from fuel assemblies or in-reactor components; contaminated rags, clothing, paper and plastic.

3.3 Classification and Segregation of Radioactive Waste

The successful management of radioactive waste depends in part on adequate classification and segregation of the waste. Gaseous radioactive waste should be classified for treatment purposes into waste arising directly from the primary coolant systems of the reactor and waste arising from the ventilation of plant areas.

Liquid radioactive waste, which is mainly water based, should be classified for processing purposes according to its specific activity and its content of chemical substances. Radioactive waste containing boric acid or organic matter, for example, may need special treatment. Non-aqueous radioactive waste such as oil should be segregated for treatment.

Solid radioactive waste should be classified according to its nature and activity; for instance, sludges, cartridge filters, contaminated equipment and components, ventilation filters and miscellaneous items (such as paper, plastic, towels) may be segregated in...
accordance with the type of treatment and conditioning process, such as compaction, incineration or immobilization.

### 3.4 Storage and Characterisation of Radioactive Waste

The storage and characterization of radioactive waste may take place between and as part of steps in radioactive waste management.

Sufficient storage capacity should be made available for all the radioactive waste generated at the nuclear power plant in Normal Operation\(^7\) and in Anticipated Operational Occurrences\(^8\) if the waste cannot be disposed of, discharged, or cleared\(^9\) from nuclear regulatory control.

In the design of storage facilities, account should be taken of the various characteristics of the waste, the possible need for its future retrieval and the potential consequences of any improper handling. Irradiated fuel assemblies contain by far the greatest quantity of radionuclides and represent potentially the greatest hazard. They are required to be stored in a manner that ensures sub-criticality\(^10\) and the removal of residual heat in compliance with established requirements and recommendations.

A margin of storage capacity should always be available in anticipation of any unforeseen events, such as delays in dispatching the radioactive waste from the site or the need for repairs to the storage facility. In order to ensure suitable margins of storage capacity, the available storage capacity for radioactive waste should be carefully controlled by maintaining an inventory of waste and where necessary, its location. For solid radioactive waste, in particular bulky items, full use should be made of the capacity of the store by means of appropriate arrangement or emplacement of its contents in the store.

Excessive accumulation of untreated and/or unconditioned radioactive waste may give rise to hazards and should be avoided if reasonably practicable by means of properly scheduled treatment and/or conditioning.

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\(^7\) Operations within specified operational limits and conditions. This includes startup, power operation, shutting down, shutdown, maintenance, testing, and refuelling (IAEA, 2007a).

\(^8\) An operational process deviating from Normal Operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to Accident Conditions (IAEA, 2007a).

\(^9\) Clearance refers to the removal of radioactive material or radioactive objects within authorized practices from any regulatory control applied for radiation protection purposes from the regulatory body (IAEA, 2007a).

\(^10\) Criticality is attained when at least one of the several neutrons that are emitted in a fission process causes a second nucleus to fission. If more neutrons are lost by escape from the system or by non-fission adsorption in impurities than are produced in fission, then the chain reaction is not self sustaining, and dies out. In this case, the assembly of fissionable material is called sub-critical (Cember, 1983).
Containers for the storage of radioactive waste should be suitable for their contents and for the conditions likely to be encountered in storage in order that the integrity of the container can be maintained over the necessary storage period. Monitoring devices with alarms set at appropriate levels should be provided as necessary to ensure the detection, location, and assessment of any leakage from the containment. Dose rates and surface contamination for waste containers should be measured in accordance with established procedures. The levels of dose rate and surface contamination as measured should comply with the requirements established by the regulatory body.

Waste should be characterized for all steps in radioactive waste management. The characterization process should include the measurement of physical and chemical parameters, the identification of radionuclides and the measurement of activity content. Such measurements are necessary for monitoring the history of the radioactive waste or waste packages through the stages of conditioning, storage, and disposal and for maintaining records for the future. The input into radioactive waste processing should be monitored in order to provide information on the performance of the plant and to help in reducing the amounts of radioactive waste generated.

### 3.5 Processing of Radioactive Waste

#### 3.5.1 General

Processing (pre-treatment, treatment, and conditioning) systems for radioactive waste should be operated and controlled in accordance with written procedures for Normal Operation as well as for Anticipated Operational Occurrences. The design intent and the operational limits and conditions, including authorized discharge limits, clearance levels and the criteria for maintaining doses as low as reasonably achievable, should be taken into account in these procedures.

Waste processing systems should be designed, operated, and maintained in accordance with a programme in which the operational modes of the plant such as startup, full power operation, and outages are taken into consideration.

Radioactive waste should be processed as early as practicable in order to convert it into a passively safe state and to prevent its dispersal during storage and disposal.

Waste packages resulting from the conditioning of radioactive waste are subject to the applicable requirements for handling, transport, storage, and disposal. In order to obtain the required product, all operations should be carried out in accordance with established procedures and subject to quality assurance requirements.
3.5.2 Gaseous Radioactive Waste

In the operation of treatment systems for gaseous radioactive waste, consideration should be given to: the amount of gas to be treated; the activity; the radionuclides contained in the gas; the concentrations of particulates; the chemical composition; the humidity; the toxicity; and the possible presence of corrosive or explosive substances.

If necessary, personnel should wear appropriate protective clothing and breathing apparatus when testing, maintaining or replacing filters so as to minimize the inhalation of particulates accumulated on the filters or the structures.

3.5.3 Liquid Radioactive Waste

In the operation of processing systems for liquid radioactive waste, the amounts of liquids to be treated, the radionuclide present, the activity, the concentrations of particulates, the chemical compositions, the toxicity and the possible presence of corrosive substances should be taken into consideration.

Input streams should be characterized, in particular for new facilities, either before liquid waste streams reach the processing plant or early in the processing activities. By this means, different types of waste can be segregated appropriately and, if various options are available, the most effective methods of processing can be adopted.

For waste conditioning, a suitable matrix material, if any, and a suitable container should be used. The container should be properly filled, closed and labelled in order to produce a waste package suitable for handling, transport, storage and disposal.

3.5.4 Solid Radioactive Waste

Solid radioactive waste may be inhomogeneous, with different physical, chemical and radiological properties within a batch of waste or on a smaller scale, within a single container. Special consideration should be given to representative sampling before processing to confirm compatibility with the intended process.

Input streams should be characterized either before liquid waste streams reach the processing plant or early in the processing activities. By this means, different types of waste can be segregated appropriately and, if various options are available, the most effective methods of processing can be adopted.

A number of processes based on proven technology are available for producing acceptable waste packages. Such processes should be selected based on the characteristics of the waste concerned, with due account taken of radioactive decay.
3.6 Transport of Radioactive Waste

The transport of radioactive waste, both domestically and internationally, is subject to the national and international model regulations for the safe transport of radioactive materials. National and international model transport regulations are generally based on the IAEA Regulations for the Safe Transport of Radioactive Material (IAEA, 2005).

The means (road, rail, or air) for the transport of radioactive waste should be considered at an early stage and its transport should comply with the appropriate regulations. The preparation of waste packages for the transport of radioactive waste should be carried out in accordance with written, approved operating procedures.

3.7 Discharge Control and Compliance Monitoring

3.7.1 General

Prior to the commencement of operations, the operating organization should propose to the regulatory body levels for gaseous and liquid discharges. In proposing such levels, it should be demonstrated that they would result in compliance with national regulations. The purpose of setting levels for discharges is to ensure that radiation doses to members of the public due to the discharges do not exceed a fraction of the dose limit for the public (the dose constraint) when applied to the critical group and that such doses are As Low As Reasonably Achievable. The expected discharges for all operational states of the plant and if possible also for potential future changes in operations should be taken into account in setting the levels to be proposed for discharges.

The proposed discharge levels should be based on an assessment of their expected radiological impacts by means of predictive modelling. Expected doses to the most highly exposed individuals should be estimated. It may be necessary to establish by means of habit surveys, which members of the public are potentially the most highly exposed because of the discharges (the critical group or groups in the population). Account should be taken of their location with respect to the plant, food consumption, sources of food and drinking water and any habits or practices that might give rise to higher than average exposure to radiation.

The regulatory body, after considering the submissions of the operating organization, should establish authorized discharge levels. All discharges should be within the discharge levels authorized by the regulatory body.

Compliance with authorized discharge levels should be demonstrated by means of monitoring at the source of the discharge and confirmed by measurement in the recipient environmental media (such as water or air). The monitoring may be by continuous measurement and/or by representative sampling and intermittent measurement, as
appropriate. For intermittent discharges into water, the assessment should be made by means of representative sampling and measurement before, and during and after each discharge, if appropriate.

Provision should be made to enable the prompt detection of any abnormal discharge of radionuclides, and the identification and assay of radiologically significant radionuclides should be performed for both gaseous and liquid discharges.

If an authorized discharge level has been or may have been exceeded, the operating organization should take appropriate steps, such as terminating the discharge and taking corrective actions; estimating the amounts of radioactive substances released; recording all relevant details; report promptly to the regulatory body in accordance with prescribed procedures; and investigating and identifying the causes of any non-compliance.

### 3.7.2 Source Monitoring

Source monitoring refers to the measurement both of discharges and of the radiation field around the source itself. The design of the source monitoring programme should be such that it enables the verification of compliance with external exposure limits and discharge limits and criteria specified by the regulatory body. The monitoring of radioactive discharges may entail making measurements for specific radionuclides or gross activity measurements as appropriate. Measurements should normally be made before or at the point of release (for example, the stack for atmospheric discharges or the discharge pipeline for a liquid discharge).

### 3.7.3 Environmental Monitoring

An environmental monitoring programme should be implemented in accordance with the requirements of the regulatory body. A pre-operational programme should be implemented two to three years before the planned commissioning of the plant. This pre-operational programme should provide for the measurement of background radiation levels near the plant and their variation over and between the seasons. It should also provide the basis for the operational programme of environmental monitoring and should include the routine collection and radionuclide analyses of various samples, such as samples of vegetation, air, milk, water, sediment, fish and environmental media collected from several fixed and identified locations off the site.

The operational programme should be implemented as an extension of the preoperational programme. The samples taken during the operational programme should be similar to those taken in the preoperational programme, but they may be collected at different intervals (for example, milk may be sampled more frequently and sediment less frequently). The operational programme should be reviewed in the light of experience and it should be modified if necessary. The programme should be designed to provide information for the purposes of:
— confirming the adequacy of control over effluent discharges;

— correlating the results of environmental monitoring with data obtained from monitoring at the source of the discharges;

— checking the validity of environmental models used in establishing authorized limits;

— fostering public assurance;

— assessing trends in the concentrations of radionuclides in the environment.
4 PBMR DPP Radioactive Waste Characteristics

4.1 General

From the definition of radioactive waste (see Section 3.1), it is clear that not all radioactive waste is the same, but may differ significantly in its origin, physical, radiological, chemical, and even biological properties. Nuclear power plants generate gaseous, liquid, and solid radioactive waste as by-products of their operations and decommissioning activities. It should therefore come as no surprise that the management of radioactive waste generated from these activities is directly related to its inherent characteristics.

A key feature of the PBMR DPP design process is the continuous strive to minimize the production of all radioactive waste, both in terms of volume and activity content, as required for new reactor designs. The PBMR DPP Safety Case Philosophy Document states that the PBMR DPP design strives to minimize the generation of radioactive waste throughout its lifetime (including decommissioning) through appropriate processing, conditioning, handling, and storage systems (PBMR, 2007c).

The reduction of waste at the design stage of the PBMR DPP is facilitated by the application of good practices for radiological zoning, provision of active drainage and ventilation, appropriate finishes, recycling of slightly contaminated water, and the use of current best practices for the handling of solid radioactive waste (PBMR, 2007a).

The purpose of this section of the report is to provide an overview of the characteristics of the radioactive waste that will be generated by the PBMR DPP. The discussion is divided into gaseous radioactive waste (Section 4.2), liquid radioactive waste (Section 4.3), solid radioactive waste (Section 4.4), and other PBMR DPP waste (Section 4.5). The discussion will cover the source (origin) of radioactive waste, quantity (volume) of waste, and level of radioactivity associated with the waste type.

4.2 Gaseous Radioactive Waste

The main kinds of gaseous radioactive waste generated in the operation of nuclear power stations are (IAEA, 2003c):

- effluents from ventilation systems in buildings;
- off-gas from systems for primary coolant degasification in nuclear reactors; and
- off-gas from the venting of storage tanks.
The activity of gaseous waste is dependent on its origin. Building ventilation air usually has lower contamination levels than process or coolant off-gas or off-gas from the venting of liquid waste storage tanks. Consideration should also be given to whether the radioactive material is present in particulate, aerosol, or gaseous form. The main sources of gaseous radioactive waste identified for the PBMR DPP include the following (PBMR, 2007e):

- **Activation of reactor cavity air**: The reactor cavity air is subject to neutron flux fission during power operation. Neutrons of thermal energy activate the constituents of the air, which become radioactive. The Reactor Building Heating, Ventilation and Air-Conditioning (HVAC) System then routes the reactor cavity air to the Power Conversion Unit (PCU) Cavity, from where it is discharged to atmosphere.

- **Leakage from the main power system at power**: During power operation, leakage of pressurized helium from the Main Power System and the associated radioactive contaminants, are collected and discharged to atmosphere by the Reactor Building Heating, Ventilation and Cleaning System.

- **Discharge of residual helium in the Main Power System (MPS) prior to full depressurization for maintenance outage**: In those years where maintenance outage is programmed, it is necessary to depressurize the MPS prior to beginning maintenance. During depressurization of the MPS down to 10% of MPS inventory, the helium is routed to the Helium Inventory Control System (HICS) and stored in the HICS storage tanks. The remainder of the helium inventory is discharged to atmosphere via the HICS and the Reactor Building HVAC System.

- **Gaseous waste from the helium purification system**: During power operation, the Helium Purification System (HPS) of the HICS receives a continuous flow of MPS helium, which is purified and returned to the MPS. Amongst other contaminants, the HPS removes and stores the CO\(_2\) containing C-14. Periodically, this gaseous waste containing CO\(_2\) is discharged to the environment via the Reactor Building HVAC System.

- **Re-suspension of activity during maintenance outage in the reactor building**: During maintenance outage in the Reactor Building, residual contamination is re-suspended by the performance of maintenance tasks. This activity is removed from the Reactor Building rooms and discharged to the atmosphere by the Reactor Building HVAC system.

- **Re-suspension of activity during normal power operation and maintenance outage in the service building**: During maintenance outage and normal power operation in the Service Building, residual contamination is re-suspended by the performance of maintenance tasks. The activity is removed from the Service Building rooms and discharged to the atmosphere by the Service Building HVAC System.
— Gaseous waste generated by Anticipated Operational Occurrences: In terms of releases of activity to environment, the limiting Anticipated Operational Occurrences event chosen is the Small Break Filtered (0 to 10 mm)\(^\text{11}\). Activity released from this event is discharged to the atmosphere by the Reactor Building HVAC System.

The gaseous radioactive emissions via the exhaust chimney consist of the following (AFROSEARCH, 2002):

— Noble gas, iodine, C-14, H-3 and aerosol emissions caused by leaks in the primary cycle and the systems that contain primary coolant; and

— Iodine, C-14 and H-3 emissions from the storage containers for radioactive contaminated helium.

Expelled air from the reactor cavity is responsible for the emission of Ar-41 and most of the aerosol activity. Table 3 presents a conservative estimate of the annual gaseous radioactive waste releases under Normal Operation and Anticipated Operational Occurrences conditions from the PBMR DPP (PBMR, 2006). Other nuclides are also released, but their quantity and consequential effects on health are insignificant.

<table>
<thead>
<tr>
<th>Significant Radionuclide</th>
<th>Normal Operation (Bq)</th>
<th>Anticipated Operational Occurrences (Bq)</th>
<th>Total (Bq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Noble Gas</td>
<td>≤ 9.60E+11</td>
<td>≤ 6.89E+12</td>
<td>≤ 7.85E+12</td>
</tr>
<tr>
<td>Ar-41</td>
<td>≤ 3.54E+12</td>
<td>None</td>
<td>≤ 3.54E+12</td>
</tr>
<tr>
<td>I-131</td>
<td>≤ 4.23E+06</td>
<td>≤ 3.75E+06</td>
<td>≤ 7.98E+06</td>
</tr>
<tr>
<td>Total Iodine</td>
<td>≤ 9.97E+08</td>
<td>≤ 7.75E+07</td>
<td>≤ 10.7E+09</td>
</tr>
<tr>
<td>Metallic Fission Products</td>
<td>≤ 1.54E+06</td>
<td>≤ 1.76E+08</td>
<td>≤ 1.78E+08</td>
</tr>
<tr>
<td>H-3</td>
<td>≤ 1.74E+13</td>
<td>≤ 4.40E+13</td>
<td>≤ 6.14E+13</td>
</tr>
<tr>
<td>C-14</td>
<td>≤ 1.21E+12</td>
<td>≤ 5.72E+10</td>
<td>≤ 1.27E+12</td>
</tr>
</tbody>
</table>

4.3 Liquid Radioactive Waste

The main sources of liquid radioactive waste generated in nuclear power stations are (IAEA, 2003c):

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\(^{11}\) A 10 mm break size refers to a crack in a pipe which is equivalent to a hole in that pipe with a diameter of 10 mm. The Small Break refers to an un-isolated break < 10 mm in diameter in the Main Power System (MPS), Inventory Control System (ICS) or Fuel Handling and Storage System (FHSS) (PBMR, 2007).
— drainage water from laundries, showers, floors, and equipment;
— organic liquids;
— liquids from decontamination, which may contain complexing agents; and
— liquids resulting from chemical processes.

The generation of liquid waste within the PBMR is restricted to those Structures, Systems, and Components (SSC) and operations, which use liquids. Because the most active parts of the PBMR within the Pressure Boundary (PB) are not water cooled, liquid process drains are restricted to potentially contaminable systems. The followings sources of liquid radioactive waste are identified within the PBMR DPP (PBMR, 2007d):

— liquid radioactive waste contributed by the Reactor Building Sump;
— liquid radioactive waste contributed by the Decontamination System (DS);
— liquid radioactive waste from the Helium Purification System (HPS) of the Helium Inventory Control System (HICS);
— liquid radioactive waste from the hot laboratory and laundry; and
— liquid radioactive waste from the washroom.

The main contributors to liquid waste are the decontamination facility, laundry, sump system, as well as the showers and wash room. The volume of liquid radioactive waste generated from these sources is less than 2,000 m³ per annum in total (PBMR, 2006), consisting of the following (AFROSEARCH, 2002):

— decontamination facility and laboratories (∼480 m³ per annum);
— laundry (∼500 m³ per annum);
— showers (emergency and health physics) and washrooms (∼100 m³ per annum); and
— sump system (HICS, PLICS, and HVAC systems) (∼365 m³ per annum).

The main activity in the liquid release from the HPS is tritium, which arises from activation of He-3 in the coolant. This quantity is very dependent on the leak tightness of the system and the resulting need to top up the helium inventory with fresh gas. Besides the tritium, small amounts of iodine and C-14 in the form of CH₄ are also present in the HPS.
4.4 Solid Radioactive Waste

4.4.1 General

Appendix A summarises the radioactive waste classification scheme defined in the National Radioactive Waste Policy and Strategy. The bulk of the solid waste that will be generated at the PBMR DPP falls under low and intermediate level waste (LILW), while high-level waste in the form of spent fuel will be the highest activity waste.

LILW can be divided further into short-lived LILW (LILW-SL) and long-lived LILW (LILW-LL). Nuclear facilities generate many different types of LILW in a wide range of radionuclide concentrations and in various physical forms and of various chemical compositions. In addition to operational waste streams, waste also arises during the decommissioning of nuclear facilities and restoration activities. Incidents or accidents may also generate waste in variable amounts and of variable composition.

LILW with low levels of activity, generated in the controlled radiological areas of a nuclear power station, generally comprises of refuse that may or may not be contaminated with minute quantities of radioactive material. This waste is usually in the form of trash, clothing, masks, gloves, plastics, insulation material, paper, concrete, wood, debris, soil, or other protective clothing. LILW with intermediate level of activity, consists of sludges, spent ion exchange resins, filter cartridges, precipitation flocculants, evaporator concentrates, or irradiated scrap metal. This waste is more radioactive than the refuse but less radioactive than spent fuel.

4.4.2 Sources of Solid Radioactive Waste

During normal plant operation, solid radioactive waste is generated from the following PBMR DPP activities (PBMR, 2007d):

- Plant maintenance, including decontamination, cleaning, replacement and disposal of systems, components and parts;
- Conditioning and replacement of dust filters in the Fuel Handling and Storage System and the Helium Purification System;
- Operation of the liquid waste plant that generates solid waste in the form of spent resin, used filters, and evaporator residue; and
- Laundry, when highly contaminated protective clothing may be discarded as unsuitable for washing.

According to the processing requirements, this waste is divided into compactable waste, non-compactable waste, and abnormal waste. Figure 3 presents a summary of the
sources of low and intermediated level solid radioactive waste that will be generated at the PBMR DPP. Table 4 summarises the PBMR DPP low and intermediated level solid radioactive waste characteristics.

![Diagram of Solid Waste](image-url)

Figure 3 Sources of solid radioactive waste that will be generated at the PBMR DPP (PBMR, 2007d).

### 4.4.3 Compactable Waste

Compactable waste typically consists of items such as discarded clothing, solid cleaning materials, wrappings, and HVAC filters (PBMR, 2007d). This waste will typically be classified as LILW, compacted in a metal container.
Table 4  Summary of some of the solid LILW characteristics (PBMR, 2006).

<table>
<thead>
<tr>
<th>Solid Waste Produced</th>
<th>Waste Package Volume (m$^3$)</th>
<th>Number of Containers per Non-Maintenance Year</th>
<th>Number of Containers During 6-Year Maintenance</th>
<th>Number of Containers During 18-Year Maintenance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Metal Containers(^1)</td>
<td>0.210</td>
<td>&lt; 100</td>
<td>&lt; 150</td>
<td>&lt; 150</td>
</tr>
<tr>
<td>Concrete Containers (Max. 6,300 kg)</td>
<td>2</td>
<td>&lt; 5</td>
<td>&lt; 20</td>
<td>&lt; 20</td>
</tr>
<tr>
<td>Special Containers for LLW(^2)</td>
<td>~12</td>
<td>-</td>
<td>-</td>
<td>~100</td>
</tr>
</tbody>
</table>

**Notes:**

1  This includes about 20 containers of embedded waste having a mass of 600 kg each. The remainder of the containers contain 200 kg of waste each.

2  Special storage containers for LLW will be produced once in the plant’s life.

4.4.4  Non-Compactable Waste

Non-compactable waste includes waste such as metal parts or process components that are generally solid but can also contain voids. Typical sources of non-compactable waste include Fuel Handling and Storage System (FHSS) graphite dust, filters in the FHSS, Dry Gas Seal Supply and Recovery System (DSRS), turbine and Liquid Waste System (LWS) residue from the evaporator, spent ion exchange resin vacuum cleaner waste, Helium Purification System (HPS) solid waste and effluent, and discarded solid objects (PBMR, 2007d).

This waste will typically be classified as LILW, solidified in a specially designed concrete container.

4.4.5  Abnormal Waste

Abnormal waste is generally considered as beyond the routine handling capabilities of the installed waste management system, due to characteristics such as excessive volume, high activity concentration, radionuclide composition, physical or chemical form, and its non-compactability. For the PBMR DPP, abnormal waste may be generated after equipment failure, but also include waste resulting from replacement of the graphite reflector blocks and control rods, as well as possible replacement of turbine, control rod drivers and other large Structures, Systems, and Components (SSC) (PBMR, 2007d).

The large components may include valve motors, pipes, vessels and similar items, while high activity components include all activated or contaminated waste (excluding fuel that is normal waste) that exceeds the requirements of LILW. Control rods, when removed from the reactor at the time of core replacement will be packed in a separate container with suitable shielding to cool down and to be reclassified (PBMR, 2007d).
4.4.6 Spent Fuel as High-level waste

In general terms, high-level waste (HLW) includes spent fuel (if it is declared as waste); radioactive liquid containing most of the fission products and actinides present in spent fuel (which forms the residue from the first solvent extraction cycle in reprocessing) and some of the associated waste streams; or any other waste with similar radiological characteristics (IAEA, 2007a).

Spent fuel refers to nuclear fuel that has been irradiated in a nuclear reactor (usually at a nuclear power plant) to the point where it is no longer useful in sustaining a nuclear reaction (IAEA, 2007a).

The KNPS generates approximately 32 tons of spent fuel each year, which would add up to 1,280 tons over a 40-year lifetime. Each spent fuel assembly contains radioactive materials that fall into three categories12:

— The first category contains the fission products (such as caesium, iodine, strontium, and xenon) which are created when uranium or plutonium nuclei are split. They are the most radioactive components of spent fuel when it leaves the reactor vessel for the fuel pool but they decay to low levels relatively quickly and after 1,000 years only about 400 GBq of the longest-lived fission products such as I-129, remain.

— In the second category are the actinides, which are isotopes of uranium and heavier metals including plutonium. These are long-lived materials, which take 10,000 years to decay to about 800 GBq.

— The third category contains the structural materials of the fuel assemblies, which become radioactive through irradiation by neutrons. They only add a small amount of radiation to the spent fuel assembly total and decay in about 500 years to less than 200 GBq.

The bulk of the HLW that will be generated by the PBMR DPP consists of spent fuel removed from the reactor core, although other larger items may also be classified as HLW. A 165 MWe PBMR module will generate a nominal 35 tons of spent fuel pebbles per annum.

4.5 Other PBMR DPP Waste

4.5.1 General

Since the PBMR DPP is not yet commissioned, records of waste other than radiological waste that will be generated are not available at present. As an indication of the type of waste that could be expected, this section provides an overview of conventional and hazardous waste produced during the first two quarters of 2007/2008 at the KNPS.

4.5.2 Conventional Waste

Conventional waste generated by a nuclear power station consists of, amongst others, domestic waste, garden waste, medical waste, building rubble, paper, and metals. Table 5 presents a summary of the conventional waste generated at the KNPS.

Table 5 Summary of the conventional waste generated at the KNPS (2007).

<table>
<thead>
<tr>
<th>Description</th>
<th>1st Quarter</th>
<th>2nd Quarter</th>
</tr>
</thead>
<tbody>
<tr>
<td>Domestic Waste (20 m³ Compact. Unit)</td>
<td>100</td>
<td>240</td>
</tr>
<tr>
<td>Seaweed/Sludge (15 m³ Container)</td>
<td>165</td>
<td>75</td>
</tr>
<tr>
<td>Garden and General (10 m³ Container)</td>
<td>170</td>
<td>140</td>
</tr>
<tr>
<td>Lagging (Man min fib) (30 m³ Container)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Medical Waste (kg)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Medical Waste (m³)</td>
<td>0</td>
<td>180</td>
</tr>
<tr>
<td>Dirty Thinners (210 L drums)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Resin (m³)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Sand Blast Grid (m³)</td>
<td>20</td>
<td>20</td>
</tr>
<tr>
<td>Fluorescent Tubes (crushed) (# Tubes)</td>
<td>1,788</td>
<td>0</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Description</th>
<th>Quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Amount Produced (m³)</td>
<td>30</td>
</tr>
<tr>
<td>Building Rubble &amp; Concrete Recycled (m³)</td>
<td>0</td>
</tr>
<tr>
<td>Wood (m³)</td>
<td>10</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Description</th>
<th>Quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Paper &amp; Cardboard: Amount Recycled (kg)</td>
<td>8,140</td>
</tr>
<tr>
<td>Printer Cartridges (#Cartridges)</td>
<td>371</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Description</th>
<th>Quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Copper (kg)</td>
<td>0</td>
</tr>
<tr>
<td>Stainless Steel (kg)</td>
<td>0</td>
</tr>
<tr>
<td>Aluminum (kg)</td>
<td>0</td>
</tr>
<tr>
<td>Brass (kg)</td>
<td>234</td>
</tr>
<tr>
<td>Mild Steel (Scrap Metal)(30 m³ container)</td>
<td>48,366</td>
</tr>
<tr>
<td>Precious Metal (PC Boards etc.) (kg)</td>
<td>0</td>
</tr>
</tbody>
</table>

4.5.3 Hazardous Waste

Hazardous waste, other than radioactive waste, generated by a nuclear power station consists of, amongst others, used gas, batteries, chemicals, asbestos, solvents, oils, and diesel. Table 6 presents a summary of the hazardous waste generated at the KNPS.
Table 6  Summary of the hazardous waste generated at the KNPS (2007).

<table>
<thead>
<tr>
<th>Description</th>
<th>1st Quarter</th>
<th>2nd Quarter</th>
</tr>
</thead>
<tbody>
<tr>
<td>Used SF6 Gas (kg) (removed form Koeberg 6,6kv circuit breakers)</td>
<td>0</td>
<td>50</td>
</tr>
<tr>
<td>40L Fixer Developer (L)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Batteries (Ni-Cad) (25 L drums) (L)</td>
<td>0</td>
<td>275</td>
</tr>
<tr>
<td>Chemical Containers (m³)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Chemicals (L)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Chemicals gas (m³)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Redundant Iodine tablets (kg)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Asbestos (m³)</td>
<td>10</td>
<td>3</td>
</tr>
<tr>
<td>Solvents (210 L drums) (L)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Transformer oil (topping up only, no replacement) (L)</td>
<td>8,100</td>
<td>110,000</td>
</tr>
<tr>
<td>Dirty diesel (L)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Fyrquel (210 L drums) (L)</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Oil filters (m³)</td>
<td>0</td>
<td>2</td>
</tr>
<tr>
<td>Unknown substance (210 L drums) (L)</td>
<td>0</td>
<td>0</td>
</tr>
</tbody>
</table>
5 Management of PBMR DPP Radioactive Waste

5.1 General

Section 2 presented an overview of the nuclear regulatory framework within which radioactive waste must be managed in South Africa, while Section 3 presented an overview of the elements of a typical radioactive waste management programme for a nuclear power plant. A similar programme will be developed for the PBMR DPP as part of the conditions for a nuclear authorisation application. Section 4 provided a summary of the anticipated gaseous, liquid and solid radioactive waste which the PBMR DPP will produce.

The purpose of this section is to provide an overview of the radioactive waste management practices envisaged as being part of the radioactive waste management programme for the PBMR DPP, from generation to disposal. The reactor Waste Handling System (WHS) is designed to handle, store and discharge low and intermediate level liquid and solid waste generated during Normal Operation, maintenance activities and Anticipated Operational Occurrences (PBMR, 2007c).

As shown in Figure 1, radioactive waste management comprises all the administrative and operational activities involved in the handling, pre-treatment, treatment, conditioning, transport storage, and disposal of radioactive waste. Conditioning of waste typically includes immobilisation and packaging of waste, treatment includes volume reduction and activity removal, while pre-treatment refers to activities such as collection, segregation, chemical adjustment and decontamination (IAEA, 2007a).

![Diagram of Radioactive Waste Management](image)

**Figure 4** Breakdown of the various activities associated with radioactive waste management (IAEA, 2007a)

The discussion will include the management of gaseous waste in Section 5.2 and liquid waste in Section 5.3, followed by an overview of the management practices envisaged for low and intermediate level waste (LILW) in Section 5.4, and high-level waste (HLW) in
Section 5.5. Where appropriate, the discussion will include the processing (pre-treatment, treatment, or conditioning) of radioactive waste.

General guidelines for the pre-disposal management of LILW and HLW is presented in IAEA (2003c) and IAEA (2003b), respectively. Guidelines for storing, as part of the predisposal management of radioactive waste, is presented in IAEA (2006d).

5.2 Management of Gaseous Radioactive Waste

5.2.1 General

The characteristics for the gaseous radioactive waste that will be generated by the PBMR DPP were presented in Section 4.2.

The best way to reduce discharges of gaseous radioactive waste from a nuclear power plant is to keep the source to the minimum activity practicable. IAEA (2002a) further propose the following to reduce the generation of gaseous waste at a nuclear power plant:

— Use filters for separating aerosols or iodine from the gaseous discharges;

— Use delay systems (charcoal beds, tanks) to allow the radioactive materials in the gases to decay; and

— Use treatment for volume reduction (such as those using recombiners, absorbers, and pressurised storage, which may also function as a delaying system.

5.2.2 Processing of Gaseous Waste

Although not a conventional radioactive waste processing system, the most important gaseous waste processing Structure, System and Component is the PBMR Heat, Ventilation and Air Conditioning System (HVAC), which consists of a number of individual systems (PBMR, 2007e). Radioactive gases, aerosols, and dust particles are removed from within the Reactor Building by purging, filtering and recirculation as part of the HVAC system (PBMR, 2007c). The other HVAC system significant in terms of the safety function performed, is the Service Building HVAC System.

The Reactor Building HVAC System is designed with an exhaust High Efficiency Particulate Air (HEPA) filter train that is switched on. An activated charcoal absorber filter bank is switched in line in series with the HEPA filter bank on detection of high radioactivity in the exhaust ducting. All releases during Normal Operation, including the discharge of residual Main Power System (MPS) helium inventory prior to maintenance outage, are discharged to the atmosphere with filtration. Before the discharge of residual MPS helium prior to maintenance outage, the charcoal filter system is switched on and
discharge is filtered by HEPA and charcoal filters. In the event of a small MPS break Anticipated Operational Occurrence, the charcoal filters of the Reactor Building HVAC System located in the exhaust air outlet are switched on automatically, and the air is filtered by a HEPA filter and a charcoal absorber bed (PBMR, 2007e). Radioactive gas samplers will be analysed to quantify the amount of radioactive gas released to ensure limits are not exceeded. All releases to the atmosphere are from a height of approximately 60 m above ground level. The dilution factors are specific to the design of the ventilation system and the release height.

5.2.3 Annual Authorised Discharge Quantities

Section 3.7 introduced the concept of discharge control and compliance monitoring as part of the Radioactive Waste Management Programme (RWMP) for a nuclear power plant. As stated, the purpose of setting levels of discharges is to ensure that radiation doses to members of the public due to discharges do not exceed a fraction of the dose limit for the public (dose constraint) when applied to the critical group and that such doses are as low as reasonable achievable (ALARA).

The PBMR DPP Safety Analysis Report (SAR) presents discharge levels for gaseous emissions based on an assessment of their expected radiological impact to the most highly exposed individual (PBMR, 2007e). It includes best estimates and design annual activity discharges for Normal Operations, as well as best estimates and design activity discharges from Anticipated Operational Occurrences. The NNR will review these proposed discharge levels, with the view to approve Annual Authorised Discharge Quantities (AADQ) for the PBMR DPP. Once operational, compliance with the PBMR DPP AADQ for gaseous emissions will be demonstrated by means of monitoring at the source of the discharge and confirmed by measurement in the recipient environmental media.

5.3 Management of Liquid Radioactive Waste

5.3.1 General

The characteristics for the liquid radioactive waste that will be generated by the PBMR DPP were presented in Section 4.3.

The best way to reduce discharges of liquid radioactive waste from a nuclear power plant is to keep the source to the minimum activity practicable. IAEA (2002a) further propose the following to reduce the generation of liquid waste at a nuclear power plant:

- use filters of different kinds to separate undissolved radioactive substances from liquids;
- use ion exchange resins, more or less specialized for the purpose, and use
standard methods to separate both dissolved radioactive substances from liquids; and

— use evaporators to separate both dissolved and undissolved substances from liquids.

The management of liquid waste is controlled by the Liquid Waste System (LWS), which function is to “collect, segregate, treat, analyse, and discharge potentially radioactive and chemically contaminated liquid radioactive waste generated during Normal Operation including maintenance activities.” The LWS is designed to perform this function during Normal Operation and Anticipated Operational Occurrences. These functions are performed with a view to ensuring compliance with regulatory dose limits and to ensure that doses to the public and workers are below limits and as low as reasonable achievable (ALARA) (PBMR, 2007d).

5.3.2 Processing (Treatment) of Liquid Waste

In order to accommodate the various source of liquid effluents, the Liquid Waste System (LWS) is split into the collection-, ion exchange-, precipitation-, and release subsystems (PBMR, 2007d).

The collection subsystem captures effluent from the Reactor Building sumps and drains, as well as effluent from the laboratories and washrooms (basins and emergency showers) located in the Service Building. In principle, fire suppression effluent from the controlled areas in the surface building will be collected and transferred to the LWS (PBMR, 2007d).

The ion exchange subsystem is used to process effluent arising from the Reactor Building sump (including effluent from Decontamination System operations) and laboratory drains effluent, by means of an ion exchanger. Effluent is received in the analysis tank in batches and then analysed by performing gross gamma analysis. The following conditions apply (PBMR, 2007d):

— If the activity levels are low enough, no processing is required to reduce the activity. Other actions such as neutralisation or chemical correction to comply with the requirements of the various non-nuclear authorisations may still be necessary. The low activity effluent is first pumped to the collection tank and then to the pre-release tank of the Release System.

— If the activity exceeds a preset limit, a radionuclide specific analysis is required to assess the release against the accumulated release data to determine if processing is required. If processing is indicated, then the effluent is pumped from the analysis tank to the anion and cation bed demineralizers via an oil separator and filters. The processes effluent is then collected in the neutralization tank, where chemical adjustment may be made before pumping the effluent to the
collection tank and then to the pre-release tank of the Release System.

Effluent is processed in the ion exchange system in batches to ensure segregation. Pre- and post-treatment of the effluent is required to ensure that the quality of the effluent is adequate to give reasonable decontamination, and to ensure that the discharge complies with the appropriate water quality requirements (PBMR, 2007d).

The precipitation subsystem processes effluent from the washrooms that may be contaminated with soap. Effluent from the washroom is collected in a recessed sump, from where it is pumped to one of the laundry analysis tanks for processing between laundry effluent processing operations. Effluent from the laundry washing cycle is delivered directly to the laundry analysis tank.

The treatment steps for the precipitation subsystem are determined by the measured activity and composition of the effluent. If precipitation is necessary, the effluent should be preconditioned for precipitation. Additives are added manually. A permanently installed agitator is used to mix the tank content. After mixing, the tank is isolated and the effluent is allowed to segregate (PBMR, 2007d). Precipitate will be removed in a way that will limit the water content.

The cleared fraction is pumped to the laundry analysis tank where the activity is determined. If the activity concentration is within the allowable limit, the batch is released to the laundry pre-release tank. If the activity concentration exceeds set levels, the effluent can be transferred to the underflow collection tank, or it can be returned to the laundry settling tanks for reprocessing (PBMR, 2007d).

Effluent from the ion exchange subsystem and laundry is transferred to different pre-release tanks for processing before release into the Main Heat Sink System’s (MHSS) seawater return pipe, where it will be analysed to obtain authorisation for release (PBMR, 2007d).

The release subsystem consists of the pre-release tank and the laundry pre-release tank. Effluent from the ion exchange system intended for discharge is transferred to the pre-release tank. The tank is isolated to prevent further effluent ingress. The effluent is analysed by radionuclide-specific analysis and the activity of each radionuclide to be discharged is quantified. A check is made against the discharges authorized in the applicable period to date and then submitted to the appropriate individual for authorization to discharge. Following authorization, the contents of the pre-release tank are discharged. The discharge is monitored by the Radiation Monitoring System. If a high signal on the discharge monitor is registered, the discharge is automatically terminated by closing the discharge valve and stopping the pump. The remaining effluent is then transferred back to the collection tank, and from there, to the analysis tank for further processing (PBMR, 2007d).

The laundry pre-release tank serves to collect effluent and to provide storage for analysis
prior to release. When sufficient effluent has been accumulated, the tank is isolated to prevent further effluent ingress. The effluent is analysed by radionuclide-specific analysis and the activity of each radionuclide to be discharged is quantified. A check is made against the discharges authorized in the applicable period to date, and then submitted to the appropriate individual for authorization to discharge. Following authorization, the contents of the laundry pre-release tank are discharged. The discharge is monitored by the Radiation Monitoring System. If a high signal on the discharge monitor is registered, the discharge is automatically terminated (PBMR, 2007d).

As a limiting case, tritiated water generated from the Helium Purification System (HPS), is stored in the tritiated water storage tank for sampling and analysis prior to discharge via the Main Heat Sink System (MHSS), following authorization. This effluent will not be monitored online during discharge, but will be metered to provide a check on the volume discharged (PBMR, 2007d).

Monitoring includes registering the activity, flow rate, and duration of the release, storage of samples and retention of data and authorizations. Effluent from the Release Subsystem is released into the MHSS's seawater return pipe, which is then routed to the Koeberg Outfall. No liquid waste is released into the storm water drain. All potentially radioactive liquid waste is routed to the KNPS main cooling seawater outflow at 40 to 80 m$^3$ sec$^{-1}$. This ensures significant dilution of radioactivity prior to reaching the sea.

Release of other industrial effluent is also monitored, but does not include monitoring of the radiological properties, since the waste streams has no pathway that could lead to contamination. All possibly contaminated effluent is treated as radioactive liquid waste. Industrial effluent is drained into the sea (PBMR, 2007d).

### 5.3.3 Reuse of Processed Liquid Waste

Where possible, processed water, rather than fresh potable water, is used to minimize the plant's water consumption. This should apply to the rinsing of tanks and to flush resin and sludge transfer lines. Processed water is also used for rinsing equipment and supply lines in the Immobilization Plant Room where possible. If the activity of an effluent batch destined for discharge is found to be high enough to warrant processing, the supply lines from that source will be rinsed with processed water to remove, as far as is possible, residual activity and reduce the potential for cross-contamination (PBMR, 2007d).

### 5.3.4 Annual Authorised Discharge Quantities

Section 3.7 introduced the concept of discharge control and compliance monitoring as part of the Radioactive Waste Management Programme (RWMP) for a nuclear power plant. As stated, the purpose of setting discharge levels is to ensure that radiation doses to members of the public due to discharges do not exceed a fraction of the dose limit for the public (dose constraint) when applied to the critical group and that such doses are as
low as reasonable achievable (ALARA).

The PBMR DPP Safety Analysis Report presents discharge levels for liquid emissions based on an assessment of their expected radiological impact to the most highly exposed individual (PBMR, 2007d). It includes best estimates and design annual activity discharges for Normal Operations. The NNR will review these proposed discharge levels, with the view to approve Annual Authorised Discharge Quantities (AADQ) for the PBMR DPP. Once operational, compliance with the PBMR DPP AADQ for liquid emissions will be demonstrated by means of monitoring at the source of the discharge and confirmed by measurement in the recipient environmental media.

### 5.4 Management of Solid Radioactive Waste

#### 5.4.1 General

The characteristics for the solid radioactive waste that will be generated by the PBMR DPP were presented in Section 4.4.2.

The function of the Solid Waste System (SWS) as part of the Waste Handling System (WHS) is to segregate, handle, analyse, process, store, and transport potentially radioactive and chemically contaminated solid radioactive waste generated during Normal Operation, maintenance activities, and upset conditions of the PBMR DPP (PBMR, 2007d). These functions are performed in order to maintain releases of radioactive materials within regulatory limits and as low as reasonable achievable (ALARA). Note that handling of spent fuel and other fuel-related waste is not part of the SWS, but part of the Fuel Handling and Storage System (PBMR, 2007d). Section 5.5 discusses the management of high-level waste.

Operational solid radioactive waste generated by the PBMR DPP will generally be in the category LILW-SL. The Radioactive Waste Management Programme that will be developed for the PBMR DPP will include procedures for the predisposal management (processing, storage, and transport) of the solid waste. Generally, it will be handled similar to the operational waste generated at the KNPS, after which it will be disposed of at the national radioactive waste disposal facility at Vaalputs. The basic activities for the processing of PBMR DPP solid waste are depicted in Figure 5.

Guidelines for the predisposal management of LILW radioactive waste is presented in IAEA (2003c).
5.4.2 Processing of Compactable Waste

Compactable waste is generally treated by compaction to reduce the waste volume. A Radiation Protection Officer (RPO) will monitor the containers for surface contamination and it will be cleaned if necessary. Compacted waste is stored until it can be transferred for disposal (PBMR, 2007d).

An example of this waste type is shown on the left in Figure 6, while the standard metal container for disposal at Vaalputs is shown on the left in Figure 8.

5.4.3 Processing of Non-Compactable Waste

Non-compactable waste includes waste such as metal parts or process components that are generally solid but can also contain voids. Waste containing dispersible activity is immobilised in a solid matrix, in order to prevent spreading of contamination if the storage container is damaged. The radiological properties of the waste will determine if standard metal containers or concrete containers are used for conditioning (PBMR, 2007d).

An example of the immobilised waste is show on the right in Figure 6, while the standard concrete container used for disposal of non-compactable waste at Vaalputs is shown on the right in Figure 8.

5.4.4 Processing of Abnormal Waste

If the large component abnormal wastes (e.g. valve motors, pipes, vessels, and similar items) are uncontaminated, they will be accumulated in a designated location and clearance will be requested when deemed necessary. Bulky parts may be processed in an on-site uncontrolled workshop.

Treatment of the contaminated waste will consist of disassembly for decontamination, disassembly to remove and segregate parts, or cutting of parts into smaller pieces. If it is
decided to decontaminate the Structures, Systems, and Components, they will be moved to the decontamination facility. These large items will be handled individually. The components will be marked and kept in a controlled interim storage area until release is confirmed or until further processing is undertaken (PBMR, 2007d).

*High activity components* include all activated or contaminated waste (excluding fuel that is normal waste) that exceeds the requirements of LILW. The following handling options can be considered (PBMR, 2007d):

- decontamination in order to remove the active layers (not applicable to activated components);
- storage on site until activity has decayed to a level when processing and disposal is feasible;
- disposal in a non-standard but approved waste container; and
- processing in a manner specific to a particular component.

When choosing the most suitable process, the dose from the whole life cycle of the component must be considered. An ALARA study will be required to optimize the choice.
Remote handling will be required or the material must be packed into containers that provide adequate shielding (PBMR, 2007d).

5.4.5 Storage of Solid Radioactive Waste

Provision is made to store compacted waste on site for up to three years. Normally, waste will be removed every year with the KNPS waste. The inventory of waste to be shipped will then be compiled. The containers will be removed from the store and transferred to the KNPS waste store. An example of such a store in use at the KNPS is shown in Figure 7. This process will include compilation of shipments, preparation of documentation, updating the inventory and final inspection of the containers for surface contamination and completeness of the data set (PBMR, 2007d).

Some of the abnormal waste will require storage to allow for decay of activity prior to disposal at Vaalputs as LILW-SL. Some of the waste in this group should be kept in storage on site until a facility is available where handling of abnormal waste and its decontamination can be accomplished. Abnormal waste that does not meet acceptance criteria for LILW-SL will be retained in on-site storage until a suitable repository is licensed (PBMR, 2007d).

Figure 7 Example of a LILW store at the KNPS used for the storage of LILW.
5.4.6 Disposal of Solid Radioactive Waste

The South African Nuclear Energy Corporation (Necsa) owns and operates the Vaalputs radioactive waste disposal facility. In terms of the National Radioactive Waste Management Policy and Strategy, this is the designated facility for the disposal of LILW in South Africa. Disposal at the site is carried out in terms of a nuclear authorisation granted by the NNR under the National Nuclear Regulator Act (Act 47 of 1999). The bulk of the LILW disposed of at Vaalputs at present and for which authorisation was granted, is generated by the KNPS.

Early in 2007, Necsa submitted an Authorisation Change Request (ACR) to the NNR for the disposal of a national inventory of LILW. The inventory derived for this purpose includes potential future PBMR LILW. The post-closure radiological safety assessment (PCRSA) prepared for this purpose is extensively documented in Van Blerk (2007a; 2007b) and Kozak (2007). The assessment context for the 2007 Vaalputs PCRSA assumed a total of 10 PBMR type reactors (Van Blerk, 2006). The NNR still have to respond to the ACR.

Beyleveld (2006) presents a detailed description of the waste containers used for the disposal of LLW and ILW at Vaalputs. Standardised containers (in terms of dimensions and mass) are being used as far as practicable to ensure uniformity, compatibility, and safe handling during all waste management processes. A 210 L metal container and the KNPS types C1, C2, C3, and C4 concrete containers are currently being regarded as standard containers. Examples of these are shown in Figure 8.

Near surface trenches are currently being used as disposal concept at Vaalputs. As shown in Figure 9, two sets of trenches are presently being used for the disposal of LLW and ILW. The area set aside for LILW disposal is 500 m by 700 m. Figure 10 presents the provisional trench layout until the assumed closure date of 2036, which takes into account a ratio between LLW to ILW of 3:1. The provisional trench layout may be changed in future in accordance with disposal needs. According to Figure 10, the following trenches are being reserved for PBMR LILW (Van Blerk, 2007a):

- Section B: 7 trenches for PBMR ILW that can accommodate 2,800 type C1 concrete containers; and
- Sector D: 60 trenches for PMBR LLW that can accommodate 99,000 210 L metal containers.

LLW were assumed to be compacted without any concrete or cement stabilization. ILW were assumed to be solidified in a cement matrix inside the concrete container, similar to current operational practices at the KNPS.
Figure 8  Waste packages currently being used for the disposal of LILW at Vaalputs are metal containers (left) and concrete containers (right).

The national inventory of LILW assumed for the purpose of the 2007 Vaalputs PCRSA is presented in Table 7, while the inventory of waste assumed to be generated by the PBMR DPP is presented in Table 8. Meyer et al. (2006) describes in detail the assumptions made to derive the national inventory, including the screening of radionuclides argued to be of no significance in terms of long-term safety.

The PBMR contribution to the national inventory in Table 7 (the activity distribution was taken as 2.59 to 1 for LLW to ILW) represents an initial estimate. The radionuclides and activities presented in Table 8 represent an improved estimate of the nuclides and associated activities that can be expected at the end of operational life of the PBMR DPP (PBMR, 2007f).

A comparison between Table 7 and Table 8 shows that, with the exception of a few nuclides (e.g., Sr-90, U-232, U-233, U-234, Ni-63, and Cd-113m), the activity of most nuclides is less than what was assumed for the national inventory. However, a number of long-lived nuclides (e.g., Be-10, Ca-41, Ni-59, Mo-93, Nb-94, Th-232) were not considered in the 2007 Vaalputs PCRSA. These radionuclides will be included in subsequent iteration of the Vaalputs PCRSA. It is expected, however, that the activity concentration of these radionuclides are such that it will have an insignificant effect on the regulatory compliance of Vaalputs as a disposal facility for LILW.
Figure 9  Examples of the near surface trenches used for the disposal of LLW (top) and ILW (bottom) at Vaalputs.
Figure 10 Provisional trench layout for the disposal of LILW at the Vaalputs site.
Table 7  Nuclide inventory of the potential future PBMR waste as at 2036, and the national inventory of LILW used in the 2007 Vaalputs Post-Closure Radiological Safety Assessment (Meyer et al., 2006). A revised inventory is presented in Table 8.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>PBMR LLW (Bq)</th>
<th>PBMR ILW (Bq)</th>
<th>Total LILW</th>
<th>Radionuclide</th>
<th>PBMR LLW (Bq)</th>
<th>PBMR ILW (Bq)</th>
<th>Total LILW</th>
</tr>
</thead>
<tbody>
<tr>
<td>Am-241</td>
<td>5.55E+09</td>
<td></td>
<td></td>
<td>Pu-239</td>
<td>4.94E+08</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Am-242m</td>
<td>4.18E+06</td>
<td></td>
<td></td>
<td>Pu-240</td>
<td>3.71E+08</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Am-243</td>
<td>2.37E+07</td>
<td></td>
<td></td>
<td>Pu-241</td>
<td>1.10E+10</td>
<td></td>
<td></td>
</tr>
<tr>
<td>C-14</td>
<td>3.75E+12</td>
<td>1.45E+12</td>
<td>5.21E+12</td>
<td>Pu-242</td>
<td>1.79E+06</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cd-113m</td>
<td>5.55E+06</td>
<td></td>
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<td>Co-60</td>
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<td>Tc-99</td>
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<td>Tru</td>
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Table 8  The estimated LILW inventory for the PBMR DPP at the end of operational life (PBMR, 2007f).

<table>
<thead>
<tr>
<th>Fission Products from Fuel and Impurities</th>
<th>Fission Products from Graphite Impurities only:</th>
<th>Activation Products from Graphite Impurities</th>
<th>Actinides from Graphite Impurities</th>
<th>Metallic Dust (other)</th>
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<tbody>
<tr>
<td>Nuclide</td>
<td>Half Life (Bq)</td>
<td>Nuclide</td>
<td>Half Life (Bq)</td>
<td>Nuclide</td>
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<td>Sr-89</td>
<td>50.5d 1.23E+10</td>
<td>Be-10</td>
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<td>Y-91</td>
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<tr>
<td>Ag-110m</td>
<td>249.9d 5.08E+13</td>
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<td>35.15d 1.08E+10</td>
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<td>Tc-99</td>
<td>2.13E5y 5.14E+06</td>
<td>Ca-41</td>
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<td>Sr-90</td>
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<td>1.57E7y 2.44E+04</td>
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<td>I-131</td>
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<td>284.3d 2.39E+10</td>
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<tr>
<td>I-135</td>
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<td>Br-84</td>
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<td>W-188</td>
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<td></td>
<td></td>
<td></td>
<td></td>
<td>Tritium</td>
</tr>
</tbody>
</table>
The number of PBMR containers assumed for the national inventory of waste consist of (Meyer *et al.*, 2006):

- 119,967 210 L metal containers, with a total volume of 25,193 m³ and a total weight of 21,594 tons.
- 2,336 C4 concrete containers, with a total volume of 893 m³ and a total weight of 14,716 tons.

The number of containers estimated for the operational life of the PBMR DPP (see Table 4) is much less than the number of containers assumed for the PBMR, with disposal capacity to spare for PBMR waste (see Figure 10).

### 5.4.7 Transport of Solid Radioactive Waste to Vaalputs

At present, all radioactive waste disposed of at Vaalputs is being transported to the site by road. In terms of the safety standards and regulations, transport should be carried out according to the provisions of the IAEA Regulations for the Safe Transport of Radioactive Material (IAEA, 2005). According to these Regulations, transport of radioactive waste to Vaalputs is subject to the following general provisions to protect persons, property, and the environment:

- an appropriate radiation protection programme to ensure adequate protection for workers and the public along the transport route. Compliance criteria for this purpose are published in the safety standards;
- an emergency response programme and procedures in the unlikely event of an accident or incident during the transport of radioactive waste; and
- a quality assurance programme for the design, manufacturing, testing, documentation, use maintenance, and inspection of waste packages to ensure compliance with the relevant provisions of the Regulations.

The bulk of the waste currently disposed of at Vaalputs originates from the KNPS. Waste is being transported to Vaalputs in consignments in specially designed trucks. The transport route from the KNPS to Vaalputs (508 km in distance) is (and *vice versa* from Vaalputs to the KNPS): Malmesbury, Moorreesburg, Piketburg, Citrusdal, Clanwilliam Dam, Klawer, Vanryhnsdorp, Kliprand, Bitterfontein, Kliprand, and Vaalputs (Eskom, 2000).

The number of metal and concrete containers shipped to Vaalputs annually, varies depending on the availability of open trenches, and meeting the Vaalputs waste acceptance requirements. On average, 160 concrete containers and 720 metal containers are being shipped to Vaalputs annually. This equates to about 32 concrete containers consignments and 6 metal container consignments, given that 5 concrete containers and
120 metal containers are transported per shipment. The shipment schedule is agreed between the KNPS and Vaalputs management (e-mail communication 25/10/2007, Rodelo Bougard).

The preparation of a shipment of solid waste is being done according to the procedure described in Eskom (2005). The purpose of this procedure is:

- to implement the requirements of the Waste Acceptance Criteria for Vaalputs, the International IAEA transport regulations, and any other applicable standards and/or procedures for all shipments of solid radioactive waste generated at the KNPS;
- to describe radiation protection responsibilities for processing of solid low and intermediate level radioactive waste and its accompanying documentation;
- to process and administrate solid radioactive waste containers;
- to provide guidance to radiation protection personnel for surveillance aspects and requirements of radioactive waste handling, prior to and during its shipment; and
- to set out steps to be followed in order to determine the radioactivity in radioactive waste containers.

The emergency plan for the transport of radioactive waste to Vaalputs is described in Eskom (2000) and includes:

- responsibilities;
- general and specific instructions for drivers;
- Normal Operation instructions for drivers;
- instructions for drivers in case of radio failure, in an event of (or involvement in) a traffic accident, or mechanical breakdown;
- instructions for the Central Alarm Station principle instructor;
- emergency instructions in the event of any abnormal occurrence during shipment of radwaste; and
- transport emergency response plan.
5.5 Management of High-level Waste

5.5.1 General

The characteristics of high-level waste (HLW) that will be generated by the PBMR DPP were presented in Section 4.4.6.

It follows from Section 2.2 that South Africa still has to formulate a strategy for the long-term management of HLW, including spent fuel. Until such time, all spent fuel is stored temporarily either in spent fuel pools (wet storage), or in dry cask storage facilities (dry storage). This allows the shorter-lived isotopes to decay before further handling, a management strategy that is acceptable from a safety perspective. An international panel of experts on the long-term storage of radioactive waste (IAEA, 2003a) developed a position paper (IAEA, 2003a), in which they clearly state that the storage of radioactive waste has been demonstrated to be safe over some decades and can be relied upon to provide safety as long as active surveillance and maintenance is ensured.

The Radioactive Waste Management Programme to be developed for the PBMR DPP will include procedures for the predisposal management of HLW, including storage of the spent fuel from the PBMR DPP.

Further guidelines for the predisposal management of HLW is presented in IAEA (2003b).

5.5.2 Storage of KNPS Spent Fuel

Since the KNPS began operating, Eskom has looked at various options for storing spent fuel from the facility. The KNPS was originally designed to store spent fuel on site for only 5 years, after which the fuel would be sent off for reprocessing. This option did not materialize, since the process was and remains a very expensive option. During the late 1980s, the fuel pools were re-racked in order to store fuel for a 10 to 15 year period (Eskom, 2004).

Eskom evaluated options such as reprocessing of spent fuel, buying special storage casks and building a facility, in which to house these casks or putting high density racks into the spent fuel pools. In 1996, a decision was taken to implement high density racks (wet storage). For this purpose, a two region (Region I and Region II) storage pit is currently being used (Eskom, 2004):

- Region I consists of three high-density rack modules. Within this region, 210 cells are provided to store unirradiated fuel with a maximum initial enrichment of 5.0 wt. % U-235. It can thus accommodate one complete core loading (157 fuel assemblies), plus one fresh batch (53 fuel assemblies) with a maximum enrichment of 5.0 wt. % U-235, or assemblies that have not reached the
necessary discharge burn-up for storage in Region II.

Region II contains high density spent fuel storage racks designed to receive spent assemblies which have undergone a minimum burn up to meet the acceptance criterion for storage in this storage region. It consists of 14 rack modules with 1,326 spent fuel storage locations (25 of which are administratively blocked, because fuel assemblies stored there could be damaged by a Safety Shutdown Earthquake (SSE) event. The acceptance criterion for this region will be met when the spent fuel assemblies to be stored in this region have reached the minimum average discharge burn up as a function of their initial enrichment. Spent fuel assemblies that do not meet the necessary discharge burn-up requirement may be stored in Region II, and still meet the criterion for sub-criticality, by being stored in a chequerboard arrangement.

The total spent fuel storage ability of the super high-density storage racks is therefore a maximum of 1,511 assemblies. This would be reduced due to the storage positions that have to be kept free of fuel in a chequerboard arrangement (Eskom, 2004). The safety analysis for the fuel rack design took into consideration criticality prevention, decay heat removal, and prevention of fuel damage (Eskom, 2004).

5.5.3 Storage of PBMR DPP Spent Fuel

The storage of PBMR spent fuel is much easier than for fuel elements or rods from conventional nuclear reactors, as no safety graded cooling system is needed to prevent fuel failure.

The Fuel Handling and Storage System (FHSS) circulates, replenishes, and removes fuel spheres to support the reactor core fuel management requirements, while the reactor is operating at power. The Sphere Storage Subsystem (SSS) of the FHSS provides a facility for the storage of spent fuel, used fuel, and graphite spheres. This facility has sufficient capacity to store all the spent fuel produced throughout the life of the plant. These spheres can be kept in situ for a further 40 years after the decommissioning of the PBMR DPP (PBMR, 2007b).

Should a HLW repository or any other long term management solution for spent fuel not be available after 80 years, the storage facility on site (or elsewhere) will have to be upgraded and refurbished to store and manage such spent fuel and other HLW for a further extended period (e.g. 40 years).

Longer storage allows residual heat generated by the fuel and the radioactivity of the spent fuel to decrease. The rationale for on site storage of spent fuel is to allow thermal and radioactive cool down of the spent fuel spheres. Most of the radioactive decay for high activity shorter lived isotopes will have taken place in the 40 years of on-site storage. Thermal heat generated during the decay process of different radioactive isotopes within the fuel spheres, is dissipated through a purpose designed water cooling
system. This system is linked to the main cooling water system of the PBMR DPP, which consists of a primary closed circuit of fresh water, and a secondary once through system, which employs sea water. In the event of coolant loss, natural convection cooling (cooling by air) will be employed. The fuel decays to less than 2 kW.m\(^{-3}\) after about a year of storage. After 40 years, the spent fuel can be more readily handled and managed. Note that the activity concentrations of long-lived isotopes will not be significantly affected by radioactive decay during this period. Depending on the half-life, it could take hundreds to thousands of years to decay to insignificant levels.

The reactor is continuously replenished with fresh or reusable fuel from the top, while spent fuel is removed from the bottom. After each pass through the reactor core, the fuel pebbles are measured to determine the amount of fissionable material left. If the pebble still contains a useable amount of fissile material, it is returned to the reactor at the top for a further cycle. Each pebble passes through the reactor about six times and lasts about three years before it is spent, which means that a reactor will use 15 total fuel loads in its design lifetime of 40 years.

The spent fuel storage pools are in a secure area inside the reactor building below ground level. These pools are located on the aseismic nuclear island (seismicity-protected area) and further protected by the citadel that is constructed around the reactor. The storage pools are approximately 18.5 m deep with the water level at 17.5 m. Pending confirmation from the structural building design, the pools will have the following characteristics (PBMR, 2007b):

- Double pool walls will be utilized for leak identification purposes. A small gap between the walls with a suitable drainage system and inspection point, will allow the distinction between groundwater seeping inwards or pool water leaking outwards.

- The inner walls are clad with a suitable liner to protect and seal the concrete surface. Depending on the type of lining, a leak monitoring system will be provided at joints and other possible initiating points in order to identify and locate leaks.

- Provision is made for insertion and deployment of ISI and underwater repair equipment for the vessel and pool liner surface.

Since the spent fuel is contained inside the storage vessels, the pool water is not expected to carry any fission products escaping from the fuel. The main contaminants will be activated impurities inside the water and elements liberated in the water due to radiolysis\(^{13}\). Pool water purification is done on-stream by utilizing the plant services Demineralization Water System (DWS). Inhibitors could be added to the pool water to

\[^{13}\text{Dissociation or disintegration of molecules by radiation.}\]
minimize the corrosive effects on the storage vessels, support structures, and pool liner (if needed) (PBMR, 2007b).

Larger items (classified as HLW) will be stored in purpose designed storage casks and sufficient space has been provided for the storage of such casks within the HLW storage area that is also located on the nuclear island. Such items will be generated during the refurbishment of the reactor after 20 years of operation and typically be reclassified after a decay period.

5.5.4 Reprocessing of PBMR DPP Spent Fuel

Reprocessing is a chemical process to separate any usable elements (e.g. uranium and plutonium) from fission products and other materials in spent fuels. Usually the goal is to recycle the reprocessed uranium or place these elements in new mixed oxide fuel (MOX).

While reprocessing of spent fuel is not excluded as an option for spent fuel management in the National Radioactive Waste Management Policy and Strategy (See Section 2.2.10), there is no intention to reprocess the PBMR DPP spent fuel at present. The main reason for this is the high burn-up of uranium in the fuel spheres (i.e. very little residual fuel) and the very high cost associated with spent fuel reprocessing.

5.5.5 Processing of PBMR DPP Spent Fuel

The reactor design caters for removal of part of the reflectors during the operational lifetime due to the radiation-induced degradation of the graphite close to the fuel core. A programme to make the graphite more resistant to degradation due to radiation is being developed, thereby reducing the specific waste production rate (PBMR, 2005). To deal with the PBMR DPP spent fuel, the following options are being considered at present (PBMR, 2005):

— The direct disposal route implies storage of spent fuel for up to another 40 years after final shutdown of the plant, after which it will be transferred to shipping/storing casks for shipment and disposal at a designated high-level waste repository.

— The second option involves reduction of the high-level waste volume by removing the matrix graphite and outer pyrolytic graphite of the coated particles. It is then also the intention to remove the radioactive contaminants and the C-14 isotope from the graphite and to reuse the cleaned graphite for the production of fresh fuel. This option can reduce the volume of high-level waste to only 4 % of that of the first option. The viability of the volume reduction process has been demonstrated at laboratory scale. Further work aimed at demonstrating the technology on a large scale is currently being carried out at a co-working university, and as part of the European nuclear waste minimization programme.
The waste management option must also be justified in terms of dose to the workers and other environmental impacts stemming from the intended process.

— Consideration is also given to reduction of the minor actinides in the fuel. Such a reduction programme will require closing the fuel cycle, with fuel reprocessing as a necessary intermediate step. Reprocessing of fuel (and therefore minor actinide reduction) will have to be seen as part of future international collaboration on fuel cycle matters.

5.5.6 Comparison with Generation 3 Nuclear Reactor Radioactive Waste

The ratio between the PBMR DPP and a Generation 3 nuclear reactor (PWR) of spent fuel storage space required (m$^3$) per unit of electrical energy produced, currently is in the order of 20 (i.e. for every 1 m$^3$ of storage space which a PWR like the KNPS requires, the PBMR DPP will require 20 m$^3$). If the waste minimization programme is successful, then the ratio PBMR:PWR would be reduced to 1 cubic metre per unit of electrical energy produced (PBMR, 2007).

5.5.7 Non-proliferation of Radioactive Waste

The extent to which enriched uranium is used in the PBMR DPP - to the point where it is no longer of use in the core - is much greater than in the present nuclear power reactors. The remaining fissile material that could be extracted from the spent PBMR fuel is of no use for proliferation purposes. This, coupled with the level of technology of coated particle fuel reprocessing, protects the PBMR fuel against the possibility of nuclear proliferation or other covert misuse. In particular, the PBMR poses the following non-proliferation attributes:

— low enriched fuel is used (less than 20%);
— a closed system for fuelling and de-fuelling with on-line tracking of fuel or graphite sphere location. This reduces the possibility of clandestine introduction of target material or the protracted diversion of core nuclear material;
— only sufficient excess reactivity is allowed to cater for temperature effects and to provide for equilibrium and transient fission product poisoning, the latter occurring during load following operation. The clandestine introduction of neutron absorbing target material into the core will upset this reactivity balance and will have a noticeable effect on the core physics;
— the PBMR is designed to store all spent fuel generated during the operational lifetime of the reactor in the facility; and
— should reprocessing become a viable option, the high burn-up achieved by the PBMR fuel produces a mix of plutonium isotopes that does not favour the production of a reliable nuclear explosive device.
5.5.8 Disposal of Spent Fuel

The National Radioactive Waste Management Policy and Strategy (see Section 2.2) clearly suggests that a long-term management strategy for spent fuel in South Africa has not been agreed upon. Internationally, several counties are in the process of formulating and developing long-term management solutions for their spent fuel. The preferred solution is geological disposal\textsuperscript{14}, mainly for its passive safety features, multiple safety functions in terms of natural and engineered barriers, containment of the waste, and excellent ability to isolate the waste from the biosphere and humans over the long term.

Section 6 will review some of the trends and strategies followed internationally for the long-term management of HLW, including spent fuel.

5.6 Assessment of Impact of Radiological Waste Management

The results of the impact assessment of radioactive waste management associated with the PBMR DPP are presented in Table 9 to Table 11.

\textsuperscript{14} The term geological disposal refers to the disposal of solid radioactive waste in a facility located underground in a stable geological formation (usually several hundreds of meters or more below the surface) so as to provide long term isolation of the radionuclides in the waste from the biosphere (IAEA, 2006b).
Table 9  Potential impact of radioactive waste management associated with the PBMR DPP.

<table>
<thead>
<tr>
<th>Impact</th>
<th>Nature</th>
<th>Intensity</th>
<th>Extent</th>
<th>Duration</th>
<th>Probability</th>
<th>Confidence</th>
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<tr>
<td>Radioactive waste that will be generated by the PBMR DPP consists of gaseous, liquid and solid waste, the latter of which can be divided into LILW and HLW.</td>
<td>Negative</td>
<td>High</td>
<td>Local</td>
<td>Long-term</td>
<td>Improbable</td>
<td>High</td>
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<tr>
<td><strong>With Mitigation</strong></td>
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</table>

Table 10  Consequence of potential impacts of radioactive waste management associated with the PBMR DPP.

<table>
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<tr>
<th>Impact</th>
<th>Consequence</th>
<th>Probability</th>
<th>Confidence</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Impact 1: Impact of radioactive waste</strong></td>
<td>Medium</td>
<td>Improbable</td>
<td>High</td>
</tr>
</tbody>
</table>

Table 11  Significance of potential impacts of radioactive waste management associated with the PBMR DPP.

<table>
<thead>
<tr>
<th>Impact</th>
<th>Consequence</th>
<th>Probability</th>
<th>Significance</th>
<th>Confidence</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Impact 1: Impact of radioactive waste</strong></td>
<td>Medium</td>
<td>Improbable</td>
<td>Low</td>
<td>High</td>
</tr>
</tbody>
</table>


6 International Basis for Management of High-level Waste Disposal

6.1 General

The IAEA fundamental safety principles are clear that the prime responsibility for safety - including the safe control of radioactive waste management - rests with the person or organisation responsible for facilities and activities that give rise to the radiation risk (IAEA, 2006a). However, the ultimate responsibility for ensuring the safety of spent fuel and radioactive waste rests with the State, through the establishment of a legal and governmental infrastructure for nuclear, radiation, radioactive waste and transport safety (IAEA, 2000). This is confirmed in the Preamble to the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste (IAEA, 2006c).

Witherspoon and Bodvarsson (2001) provide an extensive review of the geological challenges in radioactive waste isolation, which includes status reports on waste isolation projects from 32 countries. According to the review, there are two basic challenges in perfecting a system of radioactive waste isolation: choosing an appropriate geological barrier (host medium), and designing an effective engineered barrier. The review highlighted the positive contribution of underground research laboratories (URL) to waste isolation research, and the challenges with public acceptance of the management of radioactive waste isolation projects. Other highlights include (Witherspoon and Bodvarsson, 2001):

- approval of Decision in Principle for a final disposal facility for high-level waste (HLW) to be built at Olkiluoto in Finland in 2001;
- developments in site selection and specialised waste emplacement equipment in Sweden;
- the development of the Yucca Mountain project in the USA, which has been approved since the publication of the review; and
- the developments at the Waste Isolation Pilot Plant (WIPP) located in New Mexico, which is the world’s first deep geological disposal facility.

Spent fuel is regarded differently by countries — as a resource by some and as a waste by others. The strategies for its management also vary, ranging from reprocessing to direct disposal. However, in both cases a final solution is needed and it is generally agreed that disposal deep in geological formations is the most appropriate solution (IAEA, 2007b).

In all countries, the spent fuel or the high-level waste from reprocessing are currently...
being stored, usually above ground, awaiting the development of geological repositories. While the arrangements for storage have proved to be satisfactory and have been operated without major problems, it is generally agreed that these arrangements are interim, that is they do not represent a final solution (IAEA, 2007b). It is becoming increasingly important to have final disposal arrangements available so as to be able to demonstrate that nuclear power is sustainable and that it does not lead to an unsolved waste problem.

A summary of the National Radioactive Waste Management Policy and Strategy (DME, 2005) is presented in Section 2.2, according to which Government should initiate investigations into the best long-term option for the management of spent fuel.

The purpose of this section is to discuss the international basis for the management of HLW disposal, and to compare the National Radioactive Waste Management Policy and Strategy to that basis. The discussion begins with an overview of the applicable articles contained in the Joint Convention on Spent Fuel and Radioactive Waste Management, in Section 6.2.

In Section 6.3 to Section 6.8, basic concepts for radioactive management from the international literature are reviewed. These concepts are compared with current South African policy in Section 6.9.

### 6.2 Joint Convention on Spent Fuel and Radioactive Waste Management

The objectives of the Joint Convention on Spent Fuel and Radioactive Waste Management as stipulated in Article 1 of Chapter 1 are (IAEA, 2006c):

- to achieve and maintain a high level of safety worldwide in spent fuel and radioactive waste management, through the enhancement of national measures and international co-operation, including where appropriate, safety-related technical co-operation;

- to ensure that during all stages of spent fuel and radioactive waste management there are effective defences against potential hazards so that individuals, society and the environment are protected from harmful effects of ionizing radiation, now and in the future, in such a way that the needs and aspirations of the present generation are met without compromising the ability of future generations to meet their needs and aspirations; and

- to prevent accidents with radiological consequences and to mitigate their consequences should they occur during any stage of spent fuel or radioactive waste management.
South Africa is a signatory to the Joint Convention, with its entry into force as of February, 2007\textsuperscript{15}. The Joint Convention is legally binding on its contracting parties and requires that spent fuel and radioactive waste management are conducted with regard to accepted norms of safety. The safety norms are derived from the recommendations of the international safety standards, which establish best safety practices based on worldwide experience in the field (IAEA, 2007b).

The Convention applies to the safety of spent fuel management when the spent fuel results from the operation of civilian nuclear reactors. Chapter 2 (Article 4 to Article 10) deals directly with the management of spent fuel and contains the following provisions (IAEA, 2006c):

**Article 4 General safety requirements**

Each Contracting Party should take the appropriate steps to ensure that at all stages of spent fuel management, individuals, society and the environment are adequately protected against radiological hazards. In so doing, each Contracting Party should take the appropriate steps to:

- ensure that criticality and removal of residual heat generated during spent fuel management are adequately addressed;
- ensure that the generation of radioactive waste associated with spent fuel management is kept to the minimum practicable, consistent with the type of fuel cycle policy adopted;
- take into account interdependencies among the different steps in spent fuel management;
- provide for effective protection of individuals, society and the environment, by applying at national level suitable protective methods as approved by the regulatory body, in the framework of its national legislation which has due regard to internationally endorsed criteria and standards;
- take into account biological, chemical and other hazards that may be associated with spent fuel management;
- strive to avoid actions that impose reasonably predictable impacts on future generations greater than those permitted for the current generation; and
- aim to avoid imposing undue burdens on future generations.

\textsuperscript{15} See the website http://www.iaea.org/Publications/Documents/Conventions/jointconv_status.pdf for a status report.
Article 5 Existing facilities

Each Contracting Party shall take the appropriate steps to review the safety of any spent fuel management facility existing at the time the Convention enters into force for that Contracting Party and to ensure that, if necessary, all reasonably practicable improvements are made to upgrade the safety of such a facility.

Article 6 Siting of proposed facilities

Each Contracting Party shall take the appropriate steps to ensure that procedures are established and implemented for a proposed spent fuel management facility:

— to evaluate all relevant site-related factors likely to affect the safety of such a facility during its operating lifetime;

— to evaluate the likely safety impact of such a facility on individuals, society and the environment;

— to make information on the safety of such a facility available to members of the public; and

— to consult Contracting Parties in the vicinity of such a facility, insofar as they are likely to be affected by that facility, and provide them, upon their request, with general data relating to the facility to enable them to evaluate the likely safety impact of the facility upon their territory.

In so doing, each Contracting Party shall take the appropriate steps to ensure that such facilities shall not have unacceptable effects on other Contracting Parties by being sited in accordance with the general safety requirements of Article 4.

Article 7 Design and construction of facilities

Each Contracting Party shall take the appropriate steps to ensure that:

— the design and construction of a spent fuel management facility provide for suitable measures to limit possible radiological impacts on individuals, society and the environment, including those from discharges or uncontrolled releases;

— at the design stage, conceptual plans and, as necessary, technical provisions for the decommissioning of a spent fuel management facility are taken into account; and

— the technologies incorporated in the design and construction of a spent fuel management facility are supported by experience, testing or analysis.

Article 8 Assessment of safety of facilities

Each Contracting Party shall take the appropriate steps to ensure that:
⎯ before construction of a spent fuel management facility, a systematic safety assessment and an environmental assessment appropriate to the hazard presented by the facility and covering its operating lifetime shall be carried out; and

⎯ before the operation of a spent fuel management facility, updated and detailed versions of the safety assessment and of the environmental assessment shall be prepared when deemed necessary to complement the assessments referred to in previous paragraph.

**Article 9 Operation of facilities**

Each Contracting Party shall take the appropriate steps to ensure that:

⎯ the licence to operate a spent fuel management facility is based upon appropriate assessments as specified in Article 8 and is conditional on the completion of a commissioning programme demonstrating that the facility, as constructed, is consistent with design and safety requirements;

⎯ operational limits and conditions derived from tests, operational experience and the assessments, as specified in Article 8, are defined and revised as necessary;

⎯ operation, maintenance, monitoring, inspection and testing of a spent fuel management facility are conducted in accordance with established procedures;

⎯ engineering and technical support in all safety-related fields are available throughout the operating lifetime of a spent fuel management facility;

⎯ incidents significant to safety are reported in a timely manner by the holder of the licence to the regulatory body;

⎯ programmes to collect and analyse relevant operating experience are established and that the results are acted upon, where appropriate; and

⎯ decommissioning plans for a spent fuel management facility are prepared and updated, as necessary, using information obtained during the operating lifetime of that facility, and are reviewed by the regulatory body.

**Article 10 Disposal of spent fuel**

If, pursuant to its own legislative and regulatory framework, a Contracting Party has designated spent fuel for disposal, the disposal of such spent fuel shall be in accordance with the obligations of Chapter 3 relating to the disposal of radioactive waste.

**6.3 General Principles**

Disposal of radioactive waste has to be planned and implemented in a way that considers
long-term safety and radiation protection of the public and environment without imposing an undue burden on the future generations (IAEA, 2006b). Disposal involves the emplacement of waste in approved, licensed, and specified facilities. Internationally, the strategy adopted at present to protect the public and environment without imposing an undue burden on the future generations is to concentrate and contain the waste and to isolate it from the biosphere. The degree of containment and isolation of radioactive waste depends on the performance of the disposal system as a whole and it is necessary to consider the integrated performance of any waste disposal option adopted.

The fundamental principles for radioactive waste management also reflect the basic international consensus on the overall structure of ensuring waste safety. The ICRP principles for radiation protection were extended to focus on disposal issues (ICRP, 1997, 2000b, a), and these also reflect the basic principles elaborated by IAEA (1989; 1995; 2006a). In addition, these principles are embodied in the legal framework for radioactive waste management, which is described in detail by IAEA (2000).

According to the IAEA fundamental safety principles, radioactive waste disposal facilities must meet a series of ten basic principles (IAEA, 1989, 1995, 2006a):

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**Principle 1: Responsibility of safety**
The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks.

**Principle 2: Role of government**
An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.

**Principle 3: Leadership and management for safety**
Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.

**Principle 4: Justification of facilities and activities**
Facilities and activities that give rise to radiation risks must yield an overall benefit.

**Principle 5: Optimization of protection**
Protection must be optimized to provide the highest level of safety that can reasonably be achieved.

**Principle 6: Limitation of risks to individuals**
Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.

**Principle 7: Protection of present and future generations**
People and the environment, present and future, must be protected against radiation risks.
— **Principle 8: Prevention of accidents**  
All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.

— **Principle 9: Emergency preparedness and response**  
Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents.

— **Principle 10: Protective actions to reduce existing or unregulated radiation risks**  
Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.

The application of these principles imposes unique constraints on radioactive disposal facilities. Their safety must be assured over unprecedented timescales, and must be capable of functioning in these far distant times without human intervention (Principles 6 and 7). These unique constraints must be kept in mind when considering options for waste disposal.

These principles apply to all periods of the lifetime of the disposal facility. Consequently, operational safety assessments are conducted to adhere to these principles, as well as conducting a post-closure safety assessment for a very long period after facility closure. Safety assessments for the operational period follow well-established patterns for establishing the safety of nuclear facilities. Post-closure safety assessments, however, require consideration of additional features of the repository, conducted for very long periods.

The timescales of interest for post-closure safety assessment depend on the nature of the waste disposal system and the external influences on it, and the longevity of the radionuclides in the wastes. Short-lived radionuclides might only require assessments for a timescale in the order of $10^3$ years, whereas longer lived radionuclides might require assessment over timescales in excess of $10^4$ to $10^5$ years. Timescales can also be determined by national legislation governing the disposal of radioactive waste. Regardless, safety considerations are applied to radioactive waste requiring consideration of possible impacts on the system over many generations. These impacts are evaluated using a post-closure safety assessment of the disposal system.

A key challenge for post-closure safety assessment is the need to try to account for future changes in the disposal system, even over relatively short timescales. Indeed, IAEA (2003d) notes that challenges associated with projecting the behaviour of the system over long timescales is one of the key aspects that distinguishes post-closure assessments from operational safety assessments. Significant releases from a radioactive waste disposal facility might not occur for many hundreds or thousands of years after disposal. Over such timescales, it is clearly unrealistic to forecast human habits and behaviour.
This problem is compounded by the fact that changes to the disposal system, due to factors such as climate change, are also likely to occur over the timescales of potential interest. Changes may also arise from the natural evolution of the disposal system (for example the degradation of engineered barriers). Thus, any safety assessment of a disposal system must inevitably remain an estimate of what will actually occur in the future at a given location. It must not be seen as a prediction of future impacts (IAEA, 2003d).

### 6.4 Interdependencies in Waste Management

Basic steps in radioactive waste management are pre-treatment, treatment, conditioning, storage, and disposal. Pre-treatment refers to all activities prior to modification of the waste in its chemical or physical form for subsequent conditioning. Conditioning refers to modifications of the waste to prepare for disposition of the waste. Storage refers to both onsite and offsite storage of the waste, in which there is no intention to permanently leave the waste at the storage facility. Disposal refers to final disposition of the waste with no intention to retrieve the waste.

There are interdependencies among and between steps in waste management. Decisions on radioactive waste management made at one step may foreclose alternatives for, or otherwise affect, a subsequent step. Furthermore, there are relationships between waste management steps and operations that generate either radioactive waste or materials that can be recycled or reused. It is desirable that those responsible for a particular waste management step or operation generating waste adequately recognize interactions and relationships so that safety and effectiveness of radioactive waste management are balanced. This includes taking into account identification of waste streams, characterization of waste, and the implications of transporting radioactive waste. Conflicting requirements that could compromise operational and long-term safety should be avoided.

Some of the key interdependencies occur between conditioning and disposal. Conditioning technologies should be undertaken with final waste form and compatibility with the disposal technology in mind. For instance, conditioning waste in a cementitious waste form should be planned with the dimensions and chemical conditions of the final repository in mind. Similarly, the repository design should account for existing conditioned waste, to ensure that the repository will be adequate to dispose of waste intended for disposal.

Since the steps of radioactive waste management occur at different times, there are, in practice, many situations where decisions must be made before all radioactive waste management activities are established. To the extent possible, the effects of future radioactive waste management activities, particularly disposal, should be taken into account when any one radioactive waste management activity is being considered (IAEA, 1995).
The interdependencies in the stages of waste management provide a linkage between predisposal and disposal phases of waste management.

6.5 Strategy for the Disposal of Radioactive Waste

6.5.1 General

Chapter 4 of the Joint Convention requires the existence of a national legal framework for radioactive waste management (Article 19). This national legal framework must take cognisance of the overall strategy for managing waste within the country. The strategy in turn must account for waste arisings, and provide a route for disposition of all waste streams. Such a strategy requires planning the overall system of disposal within a country.

6.5.2 National Planning for Waste Disposal

Nuclear power at national level is controlled at two levels. The government provides legislation, while the regulatory body supervises and control nuclear installations and the operating organization.

The government establishes a legislative and statutory framework for the regulation of nuclear installations. Clear separation of responsibilities and organization is necessary between the regulatory body and the operating organization.

A legal framework needs to be established to provide for the regulation of nuclear activities and for the clear assignment of safety responsibilities. The government of a country that uses nuclear installations is responsible for the adoption of legislation. This legislation should separate exploitation and surveillance of the nuclear installation between operating organizations and the regulatory body. The primary objectives of such legislation should be:

- to provide the statutory basis for establishing a regulatory body;
- to provide the legal basis for ensuring that nuclear installations are sited, designed, constructed, commissioned, operated and decommissioned without undue radiological risk to the site personnel and to the public, and with proper regard to protecting the environment;
- to provide adequate financial indemnification to third parties in the event of nuclear accident, in view of the potential magnitude of damage and injury which may arise from the accident; and
- to provide the regulatory body with the power to establish and enforce the necessary regulations with respect to nuclear safety.
The government of a member state embarking on or implementing a nuclear power program, establishes a regulatory body for the surveillance of such a program. The planning for a regulatory body and the development of legislation should start in advance of the construction of the first nuclear installation.

The regulatory body acts independently of designers, constructors, and operators to the extent necessary to ensure that safety is the only mission of the regulatory personnel. An additional important function of the regulatory body is to communicate independently its regulatory decisions and opinions and their bases to the public. The regulatory body has licensing, inspection and enforcement responsibilities and must have adequate authority, competence, and resources to fulfil its assigned responsibilities.

Expertise must be available to the regulatory body in a sufficiently wide range of nuclear technologies. Depending on the activities conducted in the country, expertise should cover the following functional areas:

- specification and development of standards and regulations for safety;
- issuing of licenses to operating organizations, following appropriate safety assessments;
- inspection, monitoring and review of the safety performance of nuclear installations and operating organizations;
- requiring corrective actions of an operating organization where necessary and taking any necessary enforcement actions, including withdrawal of a license, if acceptable safety levels are not achieved;
- advocacy of safety research; and
- dissemination of safety information.

By contrast, operating organizations are responsible for:

- specifying safety criteria;
- assuring itself that the design, construction and operation of the installation meet the relevant safety standards;
- establishing policy for adherence to safety requirements;
- establishing procedures for safe control of the installation under all conditions, including maintenance, and surveillance;
- controlling fissile and radioactive materials;
- training its staff; and
ensuring that responsibilities are well defined and documented.

The fulfilment of these responsibilities is done in accordance with applicable safety objectives and requirements established or approved by the regulatory body.

The operating organization will usually delegate operating authority to the onsite management of the installation, which has the direct day-to-day control. Accordingly, the operating organization has the responsibility to monitor the effectiveness of safety management at the installation and to take necessary measures to ensure that safety is maintained at the desired level.

The regulatory body issues licenses, so it has to make reviews during the lifetime of the installation. Systematic safety reassessments of the installation in accordance with the regulatory requirements should be performed throughout its operational lifetime, with account taken of operating experience and significant new safety information from all relevant sources.

### 6.5.3 Main Factors Used in Selecting Disposal Options

Final disposition of waste is understood to refer to final disposal. Other options, such as long-term storage, do not represent final disposition. At some time in the future, all such alternative options must end in final disposal, for only disposal meets the fundamental principles of radioactive waste management (IAEA, 1995, 2006a) in the long term.

The initial stages of planning for waste disposal must address the types of waste existing and those planned to be produced. Given the types and volumes of waste to be addressed, as well as the costs and societal factors, it may be necessary to develop one or more disposal facilities.

Radioactive waste disposal facilities are faced with rigorous constraints on their performance. Their safety must be assured over unprecedented timescales, and must be capable of functioning in these far distant times without human intervention. These unique constraints must be kept in mind when considering options for waste disposal.

Perhaps the most important fundamental distinction that arises in characterizing disposal systems is depth to the waste. Disposal systems are characterized as either near-surface or geological disposal systems. Existing guidance puts the distinction between the two types of systems at “a few tens of meters below the ground surface” (IAEA, 1994c). More fundamentally, however, the distinction between near-surface and geological disposal facilities relates to the degree of isolation from human activities provided by the overburden soil.

Given the uncertainty about future conditions at the site, a basic concept is that near-surface disposal facilities should be, to a certain extent, intrinsically safe. This concept is
based on the idea that over the long time periods for which safety must be assured, intrinsic safety is achieved by limiting the activity concentrations \((\text{Bq kg}^{-1} \text{ or Bq m}^{-3})\) acceptable in near surface disposal. Limiting the activity concentrations ensures that even if the disposal facility experiences a major disruption, resulting doses will not be excessive. In particular, experience in safety assessment has shown that doses resulting from inadvertent human intrusion are the most significant for near-surface disposal facilities, and it is commonplace for activity concentrations to be established based on human intrusion analyses.

Inadvertent human intrusion is assumed to occur after some time has passed after closure of the facility. At that time, it is assumed that institutional control of the site is lost, and that intrusive activities can proceed at the site without inhibition. It is important to note that these assumptions do not generally reflect an intention to release the site from institutional control, but are instead a recognition that human institutions are fallible, and that the facility should be safe even if memory of its existence is lost. Generally, it is assumed that institutional controls can be relied upon to prevent inadvertent human intrusion for 100 to 300 years.

Once activity concentrations acceptable for near-surface disposal have been established, remaining waste streams must be consigned to a disposal facility in which the likelihood of human intrusion is low, and for which certain types of severe intrusion events are impossible. This is accomplished by consigning these wastes to a deep geological disposal facility. Consequently, a geological disposal facility may be appropriate for wastes in addition to HLW or spent fuel, if they are inappropriate for near-surface disposal.

IAEA (1994a) established a waste classification system to assist in identifying disposal options appropriate for each class of waste. This system is shown in Table 12\textsuperscript{16}. The system was derived from, and is a generalization of, the earlier classification system published by IAEA (1970). This system differs from systems of quantitative waste activity limits, in that only a limited amount of the activity-based information is included in the waste classification system, and additional qualitative features of the waste are included.

Wastes in this system are defined as follows:

- **Exempt Waste** is of such low concentration that it can be exempted from further regulatory control in accordance with clearance levels, as the radiological hazard is negligible.

- **Low- and Intermediate-Level Waste** exceeds exemption status, and also includes more highly active waste, which may include waste that requires remote handling. This category is further subdivided into **Short-lived** and **Long-lived** categories,
which relate to the intended disposal technology.

— **High-Level Waste** is defined simply as that which requires a higher degree of isolation from the environment for long periods of time. These wastes will normally require both shielding and cooling.

### Table 12 Waste Classification System of the IAEA (IAEA, 1994a).

<table>
<thead>
<tr>
<th>Waste Class</th>
<th>Typical Characteristics</th>
<th>Disposal Options</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exempt Waste</td>
<td>Activity at or below clearance levels</td>
<td>No radiological restrictions.</td>
</tr>
<tr>
<td>Low and Intermediate Level Waste</td>
<td>Activity above clearance levels; heat output less than 2 kW m(^{-3}).</td>
<td>Near-surface or geological disposal facility.</td>
</tr>
<tr>
<td>Short Lived Waste</td>
<td>Concentration of long-lived alpha radionuclides less than 4000 Bq g(^{-1}) in any package and 400 Bq g(^{-1}) averaged over all packages.</td>
<td>Near-surface or geological disposal facility.</td>
</tr>
<tr>
<td>High-level waste</td>
<td>Heat output greater than 2 kW m(^{-3}) and long-lived radionuclide concentrations exceeding the limitations for short-lived waste</td>
<td>Geological disposal facility.</td>
</tr>
</tbody>
</table>

Consequently, it can be seen that there is generally a need for two types of disposal systems, near-surface and geological disposal systems. Some countries have opted to use geological disposal for all classes of waste. The decision about the use of near-surface disposal is based on three primary factors: cost, perceived safety, and land use. Near-surface disposal is less expensive to build and operate than geological disposal facilities. Consequently, from cost and operations viewpoints, there is no reason to use geological disposal facilities for LLW and certain classes of ILW. Furthermore, since many countries find near-surface disposal acceptable from a safety perspective, these costs may seem to be excessive. However, since the safety of geological disposal systems is perceived to be greater than that of near-surface disposal, some countries have gained improved public acceptance by using geological disposal for all waste. Also, not all countries have appropriate land available to develop near-surface facilities, leading them to choose the deeper option.

The waste classification system and waste acceptance criteria should be closely linked in the strategy for any specific country. One of the primary bases for developing waste acceptance criteria is the waste concentration, which in turn is a fundamental part of the waste classification system.

A variety of design alternatives have been proposed for both near surface and geological disposal facilities (Witherspoon and Bodvarsson, 2001; IAEA, 2007b). The selection of a design alternative among these options is generally made for practical considerations rather than any fundamental consideration. Numerous analyses comparing the function of a variety of near-surface designs show that all are capable of functioning to provide adequate safety.
6.6 Features of the Repository System

IAEA (2006b) describes several requirements for safety functions of a geological disposal facility: multiple safety functions, containment, and isolation.

- Multiple safety functions: The natural and engineered barriers should be selected and designed to ensure long-term safety by means of multiple safety functions. That is, safety should be provided by multiple barriers whose performance is achieved by diverse physical and chemical processes.

- Containment: The engineered barriers, including the waste form and packaging, should be designed to provide a high level of containment of the waste, especially during the period when the waste produces significant quantities of heat and when radioactive decay can significantly reduce the hazard posed by the waste.

- Isolation: The disposal facility should be sited in a suitable geological formation and at sufficient depth to provide isolation of the waste from the biosphere and humans over the long term, at least for several thousands of years. Isolation, in this definition, is contrasted with containment, used in near-surface disposal descriptions.

Safety cases for geological disposal facilities and their supporting assessments must gather all the necessary information so as to elaborate convincing hypothesis on the functioning of the disposal facility. The safety case involves conduct of a safety assessment, which evaluates the functioning of the repository under all credible alternative external influences. However, additional arguments are included in the safety case, to develop a reasonable assurance of safety in all time periods of the disposal regime.

The disposal of radioactive waste is intended to isolate the waste from the accessible environment during a period sufficiently long to allow substantial decay of the shorter lived radionuclides and, in the longer term, to limit releases of the remaining radionuclides. In order to achieve these objectives, a multiple barrier concept is employed in which the waste form, the engineered barriers, and the site itself all contribute to the isolation of the radionuclides. Multiple barrier concepts have been developed for both near surface and geological disposal options. It has reached a state of maturity due to the experience gained from developing and operating near surface repositories, and from associated research and development. Both have provided valuable information for improvements in repository design and the technologies needed to implement them. Robust designs of engineered barrier systems should be employed in which a combination of physical barriers and chemical controls can provide a high level of isolation (IAEA, 2006b).

The major components of a disposal system generally include the waste form, the waste package, the engineered barrier system, the natural barrier system (geosphere), and the
biological setting (biosphere) of the site (see Figure 11). The waste form is the solid matrix in which the radionuclides are immobilized after treatment and/or conditioning, prior to packaging. The waste package, consisting of the waste form and container, is designed to meet the requirements for handling, transport, storage, and disposal. In order to limit the release of radionuclides and other contaminants, some packages include additional features such as absorbing materials and liners.

Figure 11 Typical barriers in a geological repository design.
The primary components of repositories that mitigate releases are the near-field and the far-field. The near-field encompasses the engineered barriers of the waste package (composed of the waste form and a container) plus a backfill. Included in the near-field is a portion of the immediately surrounding rock that is significantly affected by the presence of the repository. The far-field consists of the undisturbed natural barriers (e.g., host rock and hydrologic setting). Taken together, these two sub-systems define a repository system of multiple, redundant and complementary barriers that act to assure the safe isolation of nuclear waste from the biosphere (see Figure 12 and Figure 13).

The biosphere is not considered a barrier to radionuclide release. IAEA (1999) suggested a stylized approach for selecting critical groups and biospheres in future situations where human behaviour or biosphere conditions cannot be known with any certainty. This approach is consistent with that adopted in areas of radiological protection where it is impracticable to establish the precise characteristics of exposed individuals. For example, a stylized ‘reference man’ is used in calculating annual limits of intakes and generic models of radionuclide behaviour are used to calculate dose coefficients.

6.7 Radioactive Waste Disposal Facilities Life Cycle

6.7.1 General

A disposal facility is developed in a staged manner. At each stage of the repository lifetime, it is necessary to demonstrate that the facility will be safe as proposed (IAEA, 2006b).

At each stage of the life cycle of the disposal facility, it is necessary to maintain the safety case, to ensure that safety will be maintained throughout. This means that the safety case needs periodic revision and updating to incorporate the most up-to-date information. As the understanding of the facility and its environs grows, the safety assessment may need to be updated as well. Such updates may be established in law, requiring a periodic review, or they may be requested by the regulatory authority to ensure that the safety case remains relevant to current practices and understanding at the site.

6.7.2 Site Selection and Characterization

IAEA (1994c) discusses features of siting near-surface disposal facilities, and IAEA (1994b) discusses features of siting geological disposal facilities. The purpose of siting in either case is to identify a location that provides adequate geological stability and functionality to contribute to the long-term safety of the repository. It is particularly useful to note that the object of a siting process is not to identify the single best site possible, but rather to identify an adequate site among a number of possibilities. The goal of a siting process is to identify a suitable site in the areas of (Savage, 1995):

Page 77
Figure 12 Cutaway view of the engineered barrier within an emplacement drift of the Yucca mountain geological repository (Dyer and Voegele, 2001).

Figure 13 Features of a geological repository, showing engineered barriers, geosphere components, and the biosphere (www.goldsim.com).
— long-term safety;
— safety in the operational period (short-term safety);
— technical feasibility;
— social acceptance;
— environmental considerations; and
— cost.

While technical considerations play a necessary role in site selection, political considerations have become far more important in recent years. Consequently, sites are currently often chosen based primarily on acceptance by the local population. With this additional constraint, it is necessary to ensure that site characteristics are adequate for the purpose of waste disposal.

Important technical characteristics for an acceptable site are stability and a lack of excessive complexity. Stability refers to geological stability: the site should not experience dramatic and unpredictable morphological changes over the period of concern for the safety assessment. An example of such changes for a geological disposal facility might be fault displacement or complex fracturing, leading to uncertainty in groundwater flow directions and rates. A lack of excessive complexity is intended to be a relative term, since all natural systems are complex at some level of detail. However, the intention here is to avoid locations with extreme types of behaviour, such as karstic formations, which defy attempts to understand their behaviour to the extent needed to produce a satisfactory safety case.

6.7.3 Design

The design of the repository should minimize the need for active maintenance after site closure, and should complement the natural characteristics of the site to reduce any environmental impact. The design should take into account operational requirements, closure plan and other factors contributing to waste isolation and stability of the repository, such as protection of the waste from external events.

Geological disposal facilities include engineered barriers, which together with the emplacement medium and its surroundings, isolate the waste from humans and the environment. The engineered barriers include the waste package and other human made features such as overpacks, mined excavations, and backfills, which are intended to prevent or delay radionuclide migration from the repository to the surroundings.

Although disposal is defined as the emplacement of waste in an approved location without the intention of retrieval, some jurisdictions may nevertheless require that retrievability be designed into a repository. If the ability to retrieve waste is a design
requirement, it should be considered in the design process in such a way as not to compromise long-term performance capabilities.

The design of any monitoring program should not compromise the long-term performance of the disposal system.

6.7.4 Construction

The construction stage can only start after regulatory authorization has been issued. This usually requires that safety assessment documentation has been reviewed, the detailed repository design has been approved, the respective licensing procedures have been completed and an appropriate quality assurance program has been established. Construction of the repository may be carried out in a phased manner; in particular, it can continue and extend into the operational phase to provide additional disposal space for waste as it becomes available and is received at the facility. Depending on the size of the facility and national circumstances, the period of time from concept development to completion of construction activities may range over several decades (IAEA, 2006b).

6.7.5 Operation

The operational phase usually comprises the following activities: commissioning, waste receipt and emplacement. It is sometimes also considered to include closure (including backfilling and sealing), operational monitoring and surveillance, and any emergency activities (IAEA, 2002b). However, these are usually considered to be separate from the operational phase, and require a separate license.

The license to operate the repository may be subject to conditions imposed by the regulator to ensure that operations are consistent with the applicable regulations. In addition to radiological and industrial safety requirements for these activities, there may be requirements for physical security, fire protection and other safety related matters (IAEA, 2002b).

The operational period may also include variable periods of storage and pre-disposal conditioning and packaging of wastes. The license to operate the repository may be subject to conditions imposed by the regulator to ensure that operations are consistent with applicable regulations.

In addition to radiological and industrial safety requirements for these activities, there may be requirements for physical security, fire protection and other safety related matters.

Emplacement of waste comprises both physical placement in the repository and subsequent management until that part of the repository is covered or sealed. The
repository may have a number of units progressively constructed and used for disposal. As soon as a particular part of the repository is filled with waste to its capacity (and under some conditions even when it is in operation), voids around the waste packages are usually filled with backfill material. It may also be necessary to protect that part of the repository with a temporary cover or seal to limit infiltration of water and to provide radiation shielding.

During operation of the repository, the operator must be able to demonstrate that the repository is performing as designed with respect to its impact on workers, members of the public and the environment, and is in compliance with the license conditions. This may require, for example, inspections of waste emplacement activities, monitoring required under the terms of the license, assessment of worker exposures, and operation of a monitoring system to detect any abnormal releases from the repository. The repository operational period may last between 30 and 40 years for near-surface facilities, and a hundred years or more for geological disposal facilities.

6.7.6 Closure

Closure refers to technical and administrative actions taken at the end of its operational period to put the repository in its final state ensuring long-term safety. Closure of the repository takes place after the receipt of waste ceases and waste emplacement operations have been completed. Engineered barriers, in particular the final cover, are emplaced to ensure integrity of the repository, to minimize the ingress of infiltrating water to the waste, thereby limiting radionuclide releases, and to reduce the likelihood of disturbance by human activities. Closure should be conducted in a manner that ensures proper post-closure performance of the repository, accounting for design changes that have been updated through the operational period (IAEA, 2006b).

6.7.7 Post-Closure

The post-closure period of the repository life-cycle refers to the time in which the repository is developed to its final state, and is performing its function of isolating waste from humans and the environment. The post-closure period is often considered to be further subdivided into periods of active institutional control and passive institutional control. Some form of institutional control may be assumed to remain in place for a period of around 100 to 300 years after the repository is sealed (IAEA, 1999). Institutional control will preclude any inadvertent human intrusion into the repository and disruptive natural events are not expected to occur over this period. Therefore, the International Commission on Radiological Protection (ICRP) system of protection for practices in normal situations applies, including dose limitation, etc. This system of protection is further elaborated upon in the International Basic Safety Standards (IAEA, 1996). Institutions designated for post-closure control of repositories can be instrumental in providing scientific and technical support for safety in the following ways (IAEA, 2002b):
Consequence reduction. Once a situation giving rise to excessive radiation exposure is identified, the institution can evaluate a range of options intended to reduce exposure. This is usually referred to as remediation or intervention. It is necessary to consider whether any action is justified; for example, remedial actions should result in more good than harm.

Reduction of the likelihood of the consequence arising. Institutional control measures, such as the construction and maintenance of fences and other physical security measures, markers, land use controls and archives can all be seen as means to reduce the likelihood of the waste being disturbed. It is important not only to reduce the likelihood of radiation exposures being received, but also to reduce the likelihood of engineered barriers being impaired.

Monitoring of sites. Post-closure monitoring can serve several functions. It can provide an early warning of system malfunctions that might lead to unacceptable impacts on individuals and the environment. It can also help in verifying the intended overall performance of the disposal system.

6.8 Societal and Other Aspects

6.8.1 Public Acceptance

One of the most challenging tasks facing radioactive waste management is to explain safety assessment to stakeholders in a way that enables all concerned to play a meaningful role in the risk management and decision-making process. The US National Research Council has stressed,

“No matter how well analysts perform risk assessments, the impact and information content may be lost if the results are not communicated effectively to the people who need to use the information (NRC, 1996)”

In today’s political situation, the public plays a large role in making decisions about environmentally sensitive projects, and any project associated with radioactivity comes under particular scrutiny. It is a fact of today’s world that nuclear projects are held to higher standards than other activities. Consequently, there is a need for particular attention to be paid to public communication, safety assessment, and quality assurance parts of the project, which tend to be the focus of public attention.

6.8.2 Natural and Archeological Analogues

Natural systems where processes occur that are assumed to be similar to those in a repository environment are generally termed natural analogues. Closely linked to the studies of natural analogues are studies of ancient human made materials, provided the processes and conditions to which they have been subjected are natural. Studies of
archaeological and historical artefacts, ancient buildings, and anthropogenic sources of radionuclides such as nuclear weapons fallout can all be included in the field of analogue studies.

The relationship between natural or archaeological systems and a radioactive waste repository are inevitably imperfect and consequently it is difficult to apply the results of analogue studies directly in a quantitative way, for example to perform quantitative validation of models or to provide values for parameters used in these models (IAEA, 2002b). Consequently, natural and archaeological analogues are generally considered to be of greater use in communication with the public in a qualitative sense, than of direct technical use in safety assessments.

An additional use of natural analogues is to establish the role of natural indicators and fluxes, against which the behaviour of the disposal system can be compared. This is considered to represent a use of multiple lines of reasoning to bolster the safety case (IAEA, 2006b).

### 6.9 Comparison of South African Policy with the International Basis

DME (2005) provides the overall policy on and strategic framework for the management of radioactive waste for South Africa, including high-level. Section 2.2 presents a summary of the policy and strategy.

DME (2005) establishes a set of national radioactive waste management principles, which are explicitly linked to the basic IAEA principles. DME (2005) cites a version of the IAEA principles that are no longer current, but the differences between the older IAEA principles and the current ones are not dramatic, and represent the same underlying ideas. The IAEA basic principles are used to derive the following nine national principles (see Table 2):

- polluter pays principle;
- transparency in all aspects of radioactive waste management;
- sound decision-making;
- precautionary principle;
- no import or export of radioactive waste;
- cooperative governance and efficient national coordination;
- international cooperation;
- public participation; and
capacity building and education.

These national principles are not directly derived from the international principles, but rather include extensions of the IAEA principles for the management of radioactive waste (IAEA, 1995) to the South African national context. In particular, the principles: No import or export of radioactive waste, International cooperation, and Capacity Building and Education are not found in the international principles, whereas the remaining principles can be considered to be implicitly considered in the IAEA principles.

DME (2005) next establishes roles and responsibilities for various organizations, in keeping with international practice (e.g. IAEA, 2000; 2006a). In addition, DME (2005) establishes policies on the interpretation of South African basic principles for radioactive waste management, and establishes a specific timetable for the implementation of these policies. The policies established in DME (2005) are clearly in agreement with international concepts and approaches for HLW management.

The South African waste classification system appears to be derived directly from the IAEA system of classification (IAEA, 1994a).

DME (2005) discusses several options for disposition of HLW and spent fuel. The only option that results in final disposition of the waste is in a geological disposal facility; other options are only modifications of waste (e.g. reprocessing), or delay of the final decision (e.g. long-term storage). Consistent with international thinking, the policy and strategy recognise that the storage of spent fuel is finite and not sustainable indefinitely and that investigations need to be conducted to consider various options for safe management of spent fuel and HLW in South Africa. Given the long lead times necessary to site, license and construct a geological disposal facility, it is therefore considered necessary for South Africa to begin initiating a repository program.

While DME (2005) provides the overarching principles and policies for regulatory practices, it does not address more specific elements necessary for siting, design, licensing and construction, and operating of a high-level waste repository as required in Article 4 to Article 10 of the Joint Convention. Other elements of the regulatory regime that remain to be developed in South Africa include:

- guidelines for operational and post-closure radiological safety assessments;
- specific safety criteria for high-level waste disposal and time frames over which they need to be applied;
- financial elements of the regulatory regime;
- criteria for public involvement and transparency; and
- considerations of institutional control, retrievability, and recordkeeping.
Therefore, it is concluded that current South African policies on HLW management are consistent with international practice, but that additional detailed regulation is needed on specific issues relevant to long-term management and disposal of HLW. A summary of internationally accepted requirements for geological disposal have recently been established (IAEA, 2006b). However, these requirements should be supplemented from the experiences of several national programs that are within a decade of operating a geological repository for HLW and spent fuel, notably Finland, Sweden, and the USA.
7 Transport of Nuclear Fuel

7.1 General

The safety standards for regulatory practices (Regulation No. R.388 of April 2006), requires that radioactive material or any other equipment or objects contaminated with radioactive material, when being transported off the site or on any other road accessible to the public, must be transported in terms of the provisions of the IAEA Regulations for The Safe Transport of Radioactive Material (IAEA, 2005). The purpose of these regulations is to protect people, property, and the environment during the transport of radioactive material. This protection is achieved by (i) requiring containment of the radioactive material, (ii) control of external radiation levels, (iii) prevention of criticality, and (iv) prevention of damage caused by heat. The requirements should be applied in a graded approach to contents limits for packages and conveyance. This means that the level of application will vary between the transport of solid waste to Vaalputs and the transport of fresh or spent nuclear fuel.

Unirradiated nuclear fuel for the KNPS is imported from France and transported by road from the Cape Town harbour to the KNPS site. The fresh fuel for the PBMR DPP will be manufactured at Pelindaba in North West Province. This fuel will also be transported by road from the fuel manufacturing plant at Pelindaba to the PBMR DPP site.

The purpose of this section is to present an overview of these transport processes, starting with an overview of the manner in which nuclear fuel is currently transported to the KNPS site, in Section 7.2, and followed by a description of the manner in which nuclear fuel is likely to be transported to the proposed PBMR DPP, in Section 7.3.

7.2 Transport of Nuclear Fuel to the KNPS

Transport of nuclear fuel to the KNPS is carried out in terms of the provisions of the IAEA Regulations for The Safe Transport of Radioactive Material (IAEA, 2005), and the US Code of Federal Regulations Part 73.

Imported nuclear fuel elements are delivered at Cape Town harbour. At this point, the KNPS security group is responsible for the overall safety and protection of the fuel, in conjunction with the South African Police Service (SAPS), Crime Intelligence and the National Intelligence Agency (NIA).

Fresh nuclear fuel is delivered every 16 to 18 months. The fuel is loaded onto a 40 ft open container when delivered and loaded onto a normal truck. The loaded vehicles travel by road in a convoy to the KNPS, protected by the SAPS.
Protection of nuclear fuel in transit is of utmost importance. For this purpose, internal and external threats were identified, which forms the basis for a security plan. The security plan includes a road traffic plan.

7.3 Transport of Nuclear Fuel to the PBMR DPP Site

7.3.1 General

Transport of nuclear fuel from Pelindaba to the PBMR DPP site is a licensed procedure, the license being issued by the NNR. Necsa (2001) describes a licensing strategy, according to which transportation is subject to two licensed processes: the packages, in which the material is loaded, and the transport action (e.g. route, method, procedures, etc.).

The licensing strategy is based on national and international guidelines and regulations to ensure the safe transport of radioactive material. The most important of these are the IAEA Regulations for The Safe Transport of Radioactive Material (IAEA, 2005). The provisions of these regulations are not based on quantitative risk assessment, and they do not require such assessment to be undertaken. However, certain parts of the total transport action will be subject to quantitative assessments (Necsa, 2001).

Note that the EIA for the transport of nuclear fuel from Pelindaba to Koeberg was done as part of the EIA for the fuel manufacturing plant to be constructed at Pelindaba (PBMR EIA Consortium, 2002), and as such falls outside the scope of this report. A description of the process is provided as authorised is provided hereunder.

A Framework Transportation Plan (FTP) was developed to deal with the transport of nuclear fuel from Pelindaba to the KNPS site. The FTP is described in detail as Annexure 5 of the Environmental Impact Report (PBMR EIA Consortium, 2002), and covers pre-transport planning, pre-transport readiness/verification, and loading-transport-uploading.

The initial consignment will comprise approximately 60 tons (or about 330,000 fuel spheres), with annual replenishment of about 20 tons (about 110,000 fuel spheres) for 39 years. Current planning is for six fresh fuel packages to be loaded onto a pallet to facilitate handling, storage, and transport. Four pallets will be loaded into a 6 m ISO freight container in a single layer, so that 24 packages, containing 24,000 fuel spheres, will be transported in one such ISO container. The reactor fresh fuel storage area will be able to accept 72 packages, equivalent to 3 ISO containers. Loading into and out of the ISO containers will be performed with a forklift.

7.3.2 Pre-Transport Planning and readiness/Fitness Verification

Pre-transport planning includes the selection of appropriate packages, containers, and
vehicles, emergency planning for the transport process (including preparedness and contingencies, and security), role of the NNR, route planning, loading and off loading facilities, and insurance and securities.

Pre-transport readiness/fitness verification includes verifying operator and driver fitness, vehicle fitness, proto team/escort fitness, and route fitness, as well as maintenance and inspection of vehicles.

7.3.3 Transport Procedures

The transport process will be managed and controlled by transport procedures that include amongst others (Necsa, 2001):

- Travel restrictions (e.g. speed, weather, time of day);
- Selection of alternative routes by providing information on the relative hazards associated with each route;
- Driver certification, training and experience;
- “Fitness for purpose” test on vehicle, which comprises prescribed vehicle inspections and inspection procedures;
- Shipment and unit sizes;
- Convoy (size and composition);
- Vehicle security;
- Composition of human escorts (radiation Protection Officers and Security Officers);
- Operating technical specifications;
- Work procedures, which include package preparation, as well as handling and labelling requirements;
- Administrative requirements;
- Radiation protection;
- Fire protection;
- Quality assurance;
- Packaging and packaging inspection requirements;
- Emergency response plans, which includes risk categorization; and
- Responsibility matrix for all activities.
7.3.4 Transport Alternatives Considered

Road transport is the preferred alternative due to more limited handling, low volumes, and low frequency of movement of material. Other alternatives that were considered include rail and air transport.

7.3.5 Quantity of Nuclear Fuel to be Transported

PBMR fuel consists of uranium oxide particles covered with protective layers, embedded in graphite. A fuel element is a 60 mm sphere containing 9 grams of uranium and approximately 200 grams graphite.

The mass of the fuel relative to the uranium oxide will increase about twenty fold after the addition of graphite, which implies 3.5 tons uranium oxide to approximately 66 tons fresh fuel in total. Quantities and transport requirements for manufactured fuel are indicated in Table 13.

<table>
<thead>
<tr>
<th>Table 13 Transport requirements per reactor (manufactured fuel) (PBMR EIA Consortium, 2002).</th>
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</thead>
<tbody>
<tr>
<td><strong>Annual Transport</strong></td>
</tr>
<tr>
<td>----------------------</td>
</tr>
<tr>
<td>First fuel load (once)</td>
</tr>
<tr>
<td>Reloads (per year)</td>
</tr>
</tbody>
</table>

7.3.6 Transport Route

The fuel will be transported from Pelindaba to Cape Town by road. The preferred route is the R512 from Pelindaba to Randburg, N1 from Randburg via Kroonstad, Winburg, Bloemfontein, Colesberg, Three Sisters, Beaufort West, Laingsburg, Touws River, Worcester, Paarl to Cape Town.

The alternative route is the R512 from Pelindaba to Randburg, the N1 from Randburg, N12 from Johannesburg South via Potchefstroom, Klerksdorp, Wolmaransstad, Bloemhof, Warrenton, Kimberley, Strydenburg, Britstown, Victoria West, Three Sisters, Beaufort West, Laingsburg, Touws River, Ceres, Hermon, Wellington to Cape Town.

It is estimated that there will be about 20 consignments of nuclear fuel between Pelindaba and the PBMR DPP site in total.
7.3.7 Nuclear Safety

Package containment design is less stringent than for uranium oxide powder, as the basic design of the fuel is self-containment of the uranium up to high temperatures. Each package of restricted volume is built with a double walled cavity filled with a neutron absorbent material. This ensures that a criticality accident cannot take place. The principle of mass control and geometric subcritical (see footnote 10) are applied to the containers used for the transportation of enriched uranium and fresh fuel. Loading of the individual packages in a freight container will also be controlled.

7.3.8 Effect on the Environment

The radiological hazard of fresh uranium fuel is considerably less than that of spent nuclear fuel. Due to the former’s encapsulated and contained nature, it is in turn much less hazardous than the transport of uranium concentrate yellow coke (Ammonium Diuranate) from mines, which is a routine activity.

Transport containers will be designed according to IAEA standards for the transport of radioactive material. Radiation exposure through the walls of the container is below the limit of 2 mSv and decreases rapidly with distance, e.g. moving from 1 m to 10 m, exposure is decreased by a factor of 100. During transport, the only potential effects on the environment would arise from accidents and the release of Uranium from the fuel.

In the event of an accident, the fuel packages are designed to remain intact and will not release any fuel. In the unlikely event of fuel being released, the effect on the public or the environment should be negligible. The Environmental Impact Report discusses several accident conditions associated with fires, drop tests, penetration tests and immersion tests (PBMR EIA Consortium, 2002). Other issues that were considered include hijacking and theft, vehicle breakdowns, route hazards, community conditions, national and provincial policies, public acceptance, and selecting a suitable contractor.

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17 A criticality accident sometimes referred to as an excursion, or a power excursion occurs when a nuclear chain reaction accidentally occurs in fissile material, such as enriched uranium or plutonium. This releases neutron radiation, which is highly dangerous to surrounding personnel and causes induced radioactivity in the surroundings (http://en.wikipedia.org).
8 Conclusions

The PBMR DPP generates liquid, gaseous, and solid radioactive waste as by-products of operational conditions and decommissioning activities. The solid radioactive waste is divided further into compactable waste, non-compactable waste, abnormal waste and spent fuel. Waste other than radiological waste that will be generated can be divided into conventional and hazardous waste.

Radioactive waste management practices envisaged for the PBMR DPP are consistent with the IAEA guidelines for a Radioactive Waste Management Programme for nuclear power stations, from generation to disposal.

The PBMR DPP strives to minimize production of all solid, liquid, and gaseous radioactive waste, both in terms of volume and activity content, as required for new reactor designs. This is being done through appropriate processing, conditioning, handling, and storage systems. In addition, production of radioactive waste is minimized by applying good practices for radiological zoning, provision of active drainage and ventilation, appropriate finishes, and the use of current best practices for the handling of solid radioactive waste. Where possible, the PBMR DPP reuses or recycles materials. For example, slightly contaminated processed water rather than fresh water is used to minimize water consumption.

Processing of gaseous and liquid waste is aimed at reducing activity levels in the reactor building and in effluent generated as part of operational conditions. It also ensures that radiation doses to members of the public due to discharges to the environment (i.e., controlled discharges) do not exceed a fraction of the dose limit for the public (dose constraint). For this purpose, Authorised Discharge Quantities (AADQ) is defined for these waste steam. Compliance monitoring will be done at the source and in the environment. Processing of solid waste is aimed at reducing the volume of waste (e.g., compaction), containing dispersible activity (e.g. immobilisation), or reducing the activity of abnormal waste (e.g. decontamination). The processing and conditioning of solid waste are conducive to safe storage and consistent with the Vaalputs waste acceptance criteria.

Systems are designed store processed solid radioactive waste for a period of up to three years within the facility. The storage containers are consistent with the requirements for the disposal of solid waste at the radioactive waste disposal facility at Vaalputs. The waste unsuitable for disposal at Vaalputs will be stored on site until a suitable facility is available.

The transfer and associated transport of the waste to Vaalputs will be done in conjunction with waste shipments from the KNPS. This will be done according to the appropriate provisions of the IAEA Regulations for the Safe Transport of Radioactive Material, subject
to a graded approach. The objective of the Regulations is to protect persons, property, and the environment from the effects of radiation during the transport of radioactive material. In terms of the Regulations, the transport process is subject to radiation protection, emergency response, quality assurance, and compliance assurance programmes.

The concept for the disposal of solid waste at Vaalputs consists of near-surface trenches using metal containers for low-level waste, and concrete containers for intermediate level waste. The long-term safety of the facility, which complies with international best practices for the disposal of low and intermediate level waste, has been demonstrated for a national inventory of radioactive waste. The inventory derived for this purpose, included waste of 10 potential future PBMRs. Vaalputs therefore has more than enough capacity to dispose of the solid waste estimated to be generated by the PBMR DPP.

Processing of PBMR DPP spent fuel involves making the graphite more resistant to degradation due to radiation and thereby reducing the specific waste production rate. In addition, the intention is to reduce the volume of spent fuel by removing the graphite matrix and outer pyrolytic graphite of the coated particles. It is also the intention to remove radioactive contaminants and C-14 isotopes from the graphite to reuse the cleaned graphite for the production of fresh fuel.

The current ratio between the PBMR DPP and a Generation 3 nuclear reactor (PWR) of spent fuel storage space required (m³) per unit of electrical energy produced, is in the order of 20 (i.e. for every 1 m³ of storage space which a PWR like the KNPS requires, the PBMR DPP will require 20 m³). If the volume reduction programme is successful, then the ratio PBMR:PWR would be reduced to 1 cubic metre per unit of electrical energy produced.

The Fuel Handling and Storage System manages the storage of PBMR DPP spent fuel. This facility has sufficient capacity to safely store all the spent fuel produced throughout the life of the plant, and to store the spent fuel for a further 40 years after decommissioning if needed. It is thus only after 80 years that the storage facility on site (or elsewhere) will have to be upgraded to store and manage spent fuel. This should provide sufficient time to define and develop a long-term management strategy for the PBMR DPP spent fuel, e.g. a geological disposal facility or an alternative.

While reprocessing of spent fuel is not excluded as an option for spent fuel management, there is no intention to reprocess the PBMR DPP spent fuel at present. The main reason being the very high burn-up of uranium in the fuel sphere and the very high cost associated with spent fuel reprocessing.

The existing transport of fresh nuclear fuel to the KNPS and from Pelindaba to the PBMR DPP site is subject to the provisions of the IAEA Regulations for the Safe Transport of Radioactive Material, subject to a graded approach.
International trends and policies with respect to spent fuel and high-level waste management is based on the provisions of the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management. Internationally, this waste is currently being stored (usually above ground), awaiting the development of geological repositories. While the arrangements for storage have proved to be satisfactory and have been operated without problems, it is generally agreed that these arrangements are interim, and do not represent a final solution.

The two basic challenges in perfecting a system of radioactive waste isolation is choosing an appropriate geological barrier (host medium), and designing an effective engineered barrier. Underground research laboratories made a very positive contribution to waste isolation research, while public acceptance of radioactive waste isolation projects remains one of the major challenges.

The National Radioactive Waste Management Policy and Strategy is consistent with international practice for the management of high-level waste. However, additional, more detailed regulations are needed on specific issues relevant to long-term management and geological disposal of high-level waste. A summary of internationally accepted requirements for geological disposal have recently been established (IAEA, 2006b). These requirements should be supplemented from the experiences of several national programs that are within a decade of operating a geological repository for high-level waste and spent fuel, notably Finland, Sweden, and the USA.
9 References


PBMR. (2007f). Total Nuclide Load as Input for the EIA. DIT001125, PBMR (Pty) Ltd, Centurion, South Africa.


Appendix A

National Radioactive Waste Classification Scheme
<table>
<thead>
<tr>
<th>Waste Class</th>
<th>Waste Description</th>
<th>Waste type / Origin</th>
<th>Waste Criteria</th>
</tr>
</thead>
<tbody>
<tr>
<td>HLW</td>
<td>Heat generating radioactive waste with high, long and short lived radionuclide concentrations.</td>
<td>Used fuel declared as waste or used fuel recycling products. Sealed sources.</td>
<td>Thermal power &gt; 2 kW m⁻³. or Long-lived alpha, beta and gamma emitting radionuclides at activity concentration levels &gt; levels specified for LILW-LL. or Long-lived alpha, beta and gamma emitting radionuclides at activity concentration levels that could result in inherent intrusion dose (the intrusion dose assuming the radioactive waste is spread on the surface) above 100 mSv per annum.</td>
</tr>
<tr>
<td>LILW-LL</td>
<td>Radioactive waste with low or intermediate short-lived radionuclide and intermediate long-lived radionuclide concentrations.</td>
<td>Irradiated uranium (isotope production). Un-irradiated uranium (nuclear fuel production). Fission and activation products (nuclear power generation and isotope production). Sealed sources.</td>
<td>Thermal power (mainly due to short lived radio nuclides (T½ &lt; 31 y) &lt; 2 kW m⁻³). and Long-lived radio nuclides (T½ &gt; 31 y) concentrations. Alpha: &lt; 4000 Bq g⁻¹ Beta and gamma: &lt; 400000 Bq g⁻¹ (Maximum per waste package up to 10x the concentration levels specified above). or Long-lived alpha, beta and gamma emitting radionuclides at activity concentration levels that could result in inherent intrusion dose (the intrusion dose assuming the radioactive waste is spread on the surface) between 10 and 100 mSv per annum.</td>
</tr>
<tr>
<td>LILW-SL</td>
<td>Radioactive waste with low or intermediate short-lived radionuclide and / or low long-lived radionuclide concentrations.</td>
<td>Un-irradiated uranium (nuclear fuel production). Fission and activation products (nuclear power generation and isotope production). Sealed sources.</td>
<td>Thermal power (mainly due to short lived radio nuclides (T½ &lt; 31 y) &lt; 2 kW m⁻³. and Long-lived radio nuclide (T½ &gt; 31 y) concentrations. Alpha: &lt; 400 Bq g⁻¹ Beta and gamma: &lt; 4000 Bq g⁻¹ (Maximum per waste package up to 10x the concentration levels specified above). or Long-lived alpha, beta and gamma emitting radionuclides at activity concentration levels that could result in inherent intrusion dose (the intrusion dose assuming the radioactive waste is spread on the surface) below 10 mSv per annum.</td>
</tr>
<tr>
<td>VLLW</td>
<td>Radioactive waste containing very low concentration of radioactivity.</td>
<td>Contaminated or slightly radioactive material originating from operation and decommissioning activities.</td>
<td>Clearance or authorized discharge or reuse criteria and levels approved by the relevant regulator.</td>
</tr>
<tr>
<td>NORM-E (enhanced activity)</td>
<td>Radioactive waste containing enhanced concentrations of NORM.</td>
<td>Scales Soils contaminated with scales</td>
<td>Long-lived radionuclide concentration: &gt; 100 Bq g⁻¹.</td>
</tr>
</tbody>
</table>