Physics and engineering aspects of South Africa's proposed dry storage facility for spent nuclear fuel

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Abstract

The continual increase in electricity dependence for the advancement of society has led to increased demand in electricity globally. This increased demand, among other things such as global warming interventions and energy security have encouraged the need to diversify electricity generation sources. Civilian use of nuclear power dates back to the 1950s. The United States of America and France are currently leading with the highest nuclear power generation in the world, generating 101 GWe and 63 GWe, respectively. Several countries such as China and the United Arab Emirates have committed to new nuclear build in order to increase their nuclear power generation capacities.

Standing against the prospects of growth of the nuclear power industry are technical and non-technical challenges. These include proliferation risk, safety, high capital costs and high-level waste management. Most spent nuclear fuel from power reactors is currently stored in the spent fuel pools on reactor sites, and some have been reprocessed. It is estimated that about 32% (370 000 tons of Heavy Metal) of the total spent fuel generated from power reactors have been reprocessed up to date. With most of the spent fuel pools filling up, alternative interim and long term disposal of spent nuclear fuel solutions have been under investigation from as early as the 1970s. South Africa has planned an interim dry storage facility for the spent nuclear fuel to be established at the existing Koeberg power station. The interim dry storage facility will make use of HI-STAR 100 multi-purpose casks to store spent nuclear fuel until the country decides on final disposal solution. There are many aspects that are critical to safe, efficient and cost-effective long term storage of spent nuclear fuel.

Some of the physics and engineering aspects concerning dry storage facilities are briefly discussed. The aspects presented here are: radiation containment, spent fuel, sub-criticality, decay heat removal, site location aspects, response to seismic events, cask corrosion, transportation infrastructure, operability and monitoring. The study of the three existing dry cask storages from the USA, Hungary and Belgium gives an overview of the dry cask technology in use today. These presentations are based on publicly available reliable information.

The proposed dry storage facility at Koeberg will be in the existing power station footprint using the HI-STAR 100 casks. The decision to have the proposed dry storage facility at Koeberg will minimise related licence applications and part of security installations as the site already has some security. The location of the facility in the power station's footprint also allows for cost-effective and safe transportation of casks from the reactor building to the proposed facility. The modularity aspect of the dry cask storage facility at MV Paks in Hungary should also be employed at Koeberg to allow for more storage. This will cater for additional casks that may need to be stored if more nuclear power plants are procured in the future. South Africa's air traffic around the Western Cape is not as congested as Belgium's. There is, therefore, no need for the casks to be housed in concrete buildings like Doel's. Most of Koeberg's high-level waste would have had a longer cooling time in the pools compared to the minimum cooling time required for the chosen cask technology. This will provide a conservative, safe approach for Koeberg's facility. Dry cask storage technology has provided a reliable interim dry storage solution for several countries. Despite uncertainties for long term disposal options, the proposed dry cask storage facility at Koeberg is a suitable interim storage alternative for South Africa to allow continuous operation of the plant. This conclusion is based on the physics and engineering aspects that have been presented in this minor dissertation.

Keywords: spent fuel, dry cask storage, spent fuel pool

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Nomenclature

AFR	Away-From-Reactor
AR	At-Reactor
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
DGR	Deep Geological Repository
DUPIC	Direct Use of Spent PWR Fuel in CANDU
EDF	Électricité de France (French word which transliterate to *Electricity of France*)
GWd	Gigawatt-days
GWe	Gigawatt – Electric
IAEA	International Atomic Energy Agency
ISFSI	Independent Spent Fuel Storage Installation
kW	Kilo-watt
MOX	Mixed Oxide
mSv	milli Sieverts
MTU	Metric ton of Uranium
MWd	Megawatt-days
MWe	Megawatt – Electric
NRC	Nuclear Regulatory Commission
NUHOMOS	Nutek Horizontal Modular Storage
PGA	Peak Ground Acceleration
PWR	Pressurized Water Reactor
SFP	Spent Fuel Pool
tHM	tons of heavy metal
VVER	Water-water energetic reactor (from Russian *Vodo-vodynoi nergetichesky reactor*)

1. Introduction

1.1. Overview of nuclear energy and current contribution to electricity generation

Since its discovery in the 18th century, electricity has become one of the vital components for societal development all over the world. This continuous dependence on electrical energy for development has led to an increase in demand for power generation capacity throughout the world. In an attempt to meet the growing demand, diversify generation technologies and increase security of supply, several generation technologies are in operation today, including nuclear power. Civil use of nuclear power dates back to the 1950s, where it was initially used for electricity generation in 1950 [1] [2].

Nuclear power currently accounts for about 11% of the world's electricity base-load supply. It is the second largest source of low-carbon electricity generation after hydropower [3] [4] [5]. The United States of America (USA) and France are currently leading in the nuclear power industry, with the largest nuclear power generation capacities, generating 101 GWe and 63 GWe, respectively. Although the USA generates 101 GWe of nuclear power, it only accounts for about 19% of the total domestic electricity generated [6]. France's nuclear generation accounts for about 76% of the total domestic electricity generated [7] [8]. There are currently 31 countries worldwide that have over 440 nuclear reactors generating electricity, with a number of countries planning to expand their nuclear power generation capacities in the coming years. The nuclear power industry is dominated by pressurised water and boiling water reactor technologies (PWR and BWR) [9]. PWR and BWR technologies generate about 87% of nuclear power worldwide [10] [11].

China's current nuclear electricity contributes about 3% of the total installed generation in the country. This is generated using 37 reactors of different technologies with an installed capacity of 34 GWe as of 2017. There are 20 reactors under construction, a plan to increase the nuclear generation to 58 GWe by 2020 and construction of nuclear plants from 2020 which will add 30 GWe. The United Arab Emirates (UAE) is also embarking on a nuclear power program. The construction of four APR-1400 reactors commenced in 2012. These will give a gross output of 5.6 GWe in addition to the current 19 GWe total electricity output. France's energy policy makes provision for reduction of nuclear power generation to 50% from its current 76% [12] [13] [14] [15].

1.2. South Africa's electrical infrastructure

South Africa's only nuclear power plant, Koeberg, contributes about 5% of the 45 GWe of the total installed electricity generation in the country [16] [17]. The plant has been in operation since 1984 and has two pressurised water reactors (PWR) each with an installed capacity of 960 MW of electrical power. Over 90% of the electricity in South Africa is currently generated using coal with most of the power plants located in the northern part of the country [17] [18]. The abundance of coal in the northern part of the country has led to the concentration of these coal-fired power stations in Mpumalanga, Gauteng and Limpopo provinces. South Africa experienced power shortages around 2008 due to lack of generating capacity. The country's initial Integrated Resource Plan (IRP), published in 2010, had projected a nuclear capacity of 9.6 GW to be added by 2030. This was accepted by the cabinet [19]. However, the draft IRP 2016 which was at public consultation stage at the time of this dissertation has projected a delay in construction, with the first contribution of 1359 MW only expected in 2037 [20]. The total projected contribution of nuclear power by 2050 is 20385 MW [20]. This is very likely to change as it depends on the IRP, which gets updated every two years and in turn, depends on economic growth among other factors.

1.3. Some of the challenges of nuclear power

Public acceptance of nuclear power has been an issue in many countries due to the introduction of nuclear fission in the early 1940s during the Second World War and nuclear accidents that have followed after. The most historic nuclear accidents are the 1979 Three Mile Island accident, the 1986 Chernobyl and the 2011 Fukushima accident in Japan. Countries such as Germany decided to phase out nuclear power generation after the 2011 Fukushima accident while other countries like China decided to increase their nuclear power generation capacity despite the unfortunate accidents. Fear of the possibility of nuclear proliferation from nuclear power plants still creeps in many people's hearts today regardless of the treaty on non-proliferation of nuclear weapons signed in 1968 by major countries.

Possibilities of terrorist attacks that can access nuclear waste also contribute to public opposition to nuclear power. Samia Rashad [21]and the interdisciplinary MIT study [22] highlight some key issues that the nuclear industry needs to address in detail for future expansion of nuclear energy. Waste management is one of the critical aspects highlighted in these two studies. South Africa is not immune to public opposition of nuclear power. This was experienced during the construction of Koeberg power station and demonstrations that have been shown in response to Thyspunt site in the Eastern Cape being chosen for a nuclear power station [23]. In addition to nuclear accidents and high capital costs associated with the nuclear build, communities are often worried about nuclear waste.

1.3.1. Classification of radioactive waste from nuclear power

The World Nuclear Association classifies nuclear waste into four categories, namely exempt and very low-level waste, low-level waste, intermediate-level waste and high-level waste. The classification of these wastes is based on the heat generated and level of radioactivity and determines how the waste is handled. Other classification systems use the radioactivity and the half-life of the radionuclide to classify waste [24]. The International Atomic Energy Agency (IAEA) classifies waste into six categories, namely exempt waste, very short-lived waste, very low-level waste, low-level waste, intermediate waste and high-level waste [25]. Classification of waste also differs from country to country [26].

The difference in waste classification systems from country to country arise because of different purposes which classification systems aim to address. Some of the classification systems have been placed to simplify the language and assist in planning while on the other hand, regulatory bodies have classification systems to ensure safety. Even though there is a variation in classification systems, the origin of the waste or contents of the waste is generally used as the basis for these classifications. Other classification of waste uses physical properties such as the phase of the waste (gas, liquid and solid), compatibility, volatility, weight and size [27].

1.3.2. Nuclear fuel cycle

There are two nuclear fuel cycles for nuclear power in use and under research today. These fuel cycles describe steps from the mining of the uranium through its utilisation as fuel in the reactor up to waste disposal [28]. Nuclear fuel cycles are generally classified into three distinct phases, the front-end, reactor and back-end [29]. The front-end activities are associated with the preparation and delivering of the fuel for the reactor. This activity ranges from uranium mining to completion of fuel assemblies. The fuel assemblies must be specific for the reactor type. The reactor extracts energy from the fuel. The back-end activity involves the management of spent fuel from the reactor and safe disposal of all different wastes generated [29] [30]. Below is a diagram adapted from [28], showing steps in a fuel cycle for a typical 1000 MWe reactor.

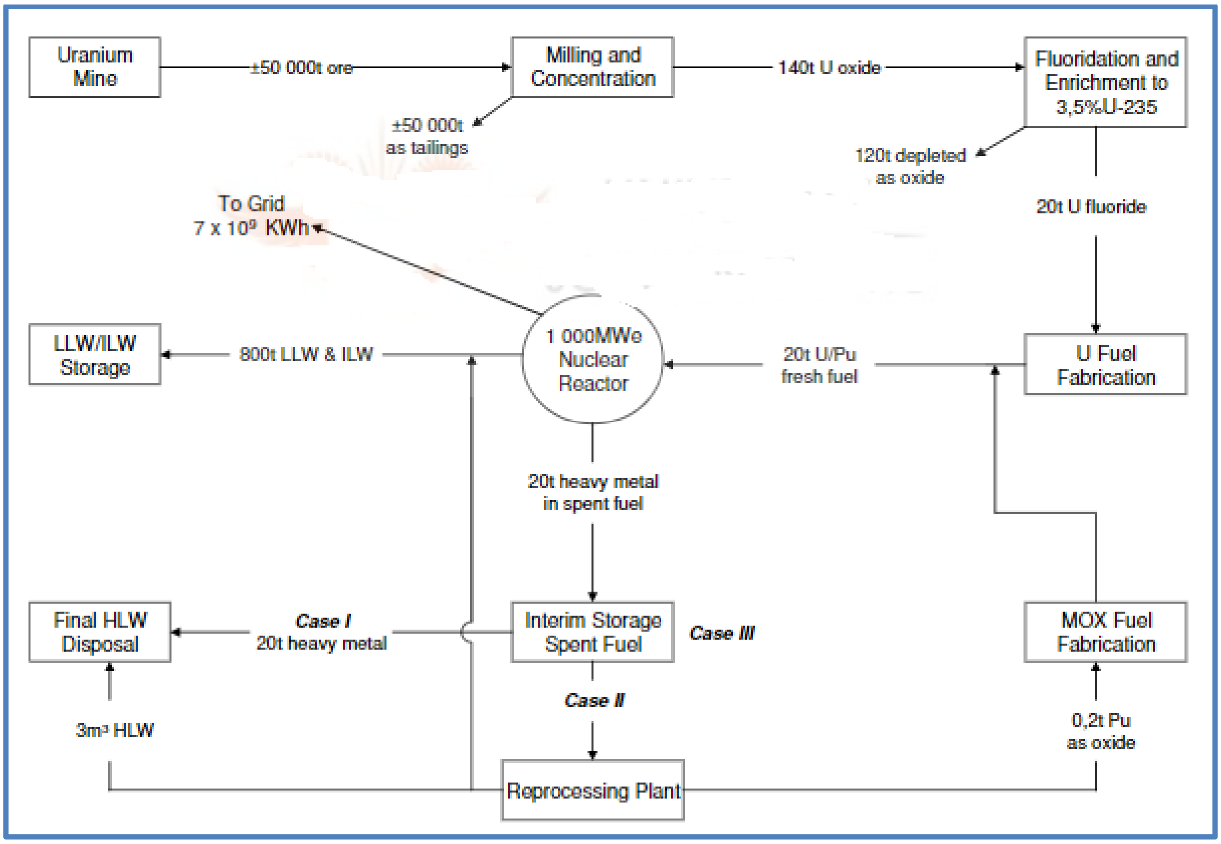


Figure 1-1: Typical nuclear fuel cycle [28]

The enriched uranium-235 is the most common in use today. This uses an enriched (typically ranging from 3-5%) uranium as the initial fuel [31]. Other fuels in use include natural uranium currently in the CANDU technology which has no enrichment. The MOX cycle is another cycle where the plutonium which has been recovered during the reprocessing is mixed with new or used uranium to form a mixed uranium and plutonium oxide fuel [31]. Other fuels that are not discussed in detail here include, thorium and DUPIC [32] [33]. The fuel cycle is either an open or closed cycle. This division of the cycles is necessary due to the possibility of reprocessing at the back end of the cycle. An open fuel cycle is one where the fuel is used in the reactor only once and sent for disposal or interim storage after irradiation in the reactor. A closed cycle is one where the spent fuel is reprocessed to extract uranium after sufficient cooling in the spent fuel pool. In Figure 1-1 above, the open, closed and deferral cycles are shown as cases I, II and III.

2. Literature review: Current spent fuel management

2.1. Introduction

This chapter gives an overview of current spent fuel management in practice today. At the centre of deciding the spent fuel management option is the understanding of the spent fuel itself. A brief overview of a typical spent fuel design is discussed. The spent fuel management options that are discussed here are the spent fuel pool, dry cask storage, reprocessing and deep geological repository.

2.2. Spent fuel overview

2.2.1. Overview of a typical fuel design

The majority of nuclear power plants use fuel that has uranium dioxide pellets which are placed in a sealed tube of zirconium alloy. A typical pellet is cylindrical and about 9 mm in diameter [32]. There is a gap of about 1 mm between the cladding and the fuel pellet to allow the fuel to swell as it heats up inside the reactor without causing damages to the structural integrity of the cladding. The gap also allows for ease of pellet insertion during the manufacturing process and provides space for helium gas to keep a predetermined pressure. A typical PWR cladding thickness is about a millimetre, thereby making the fuel rod (fuel pellet, gap and cladding) to range between 9 and 14 mm in diameter, depending on the reactor design.

The fuel rod acts as the primary confinement of the fuel in the reactor. Zirconium alloy is used for cladding in all reactor technologies in use today. There are slight variations in the mixture of the elements for cladding; an example is a Magnox reactor which uses Magnox alloy for cladding. Magnox alloy is magnesium with approximately 1% of zirconium or aluminium [33]. Some of the essential factors in determining the material for cladding include heat transfer capability, low neutron absorption, especially for thermal neutrons for PWR, mechanical strength and high corrosion resistance [34].

The rod is not filled to capacity to allow space for the release of gases during fission. A number of fuel rods arranged in a specific geometry make up a fuel assembly together with control rods used to control the fission [32] [34]. A fuel assembly typically consists of fuel rods ranging from 14x14 to 17x17 [34]. The geometry of the fuel assemblies also differs from reactor to reactor. There are 3 common geometries in use today. These are hexagonal, circular and square arrangements. Table 2-1 shows some of the parameters of fuel assemblies for different reactors and a diagram of a 17x17 fuel assembly with rod control, adapted from [35], is shown in Figure 2-1. The common reactors presented in Table 2-1 are the Russian VVER-1000, PWR from Westinghouse and EDF and a typical BWR. The VVER is a Russian acronym for “vodo-vodyanoi energetichesky reacktor” which translates to water-water energetic reactor in English. The concept behind this name is that the reactor uses water for both cooling and moderation. EDF is an acronym for French’s Électricité de France, a nuclear electric power company. The BWR is an acronym for boiling water reactor which does not have a pressurizer in its system and therefore allows water to boil inside the reactor.

Table 2-1: Some of the typical parameters of the fuel design of common reactors

Reactor	Electrical Power output [MW]	Geometric arrangement of rods	Fuel pellet Diameter [mm]	Cladding	Fuel assembly array	Number of fuel assemblies	No. of fuel rods
VVER – 1000	1000	Hexagonal	7.6	E110, E635		66	300
PWR (Westinghouse)	1100	Square	7.8	Zircaloy-4	17 x 17	193	264
PWR (EDF)	900	Square	8.2	Zircaloy	17 x 17	157	264
BWR	1000	Square	9.55	Zircaloy-2	9 x 9	764	91-96

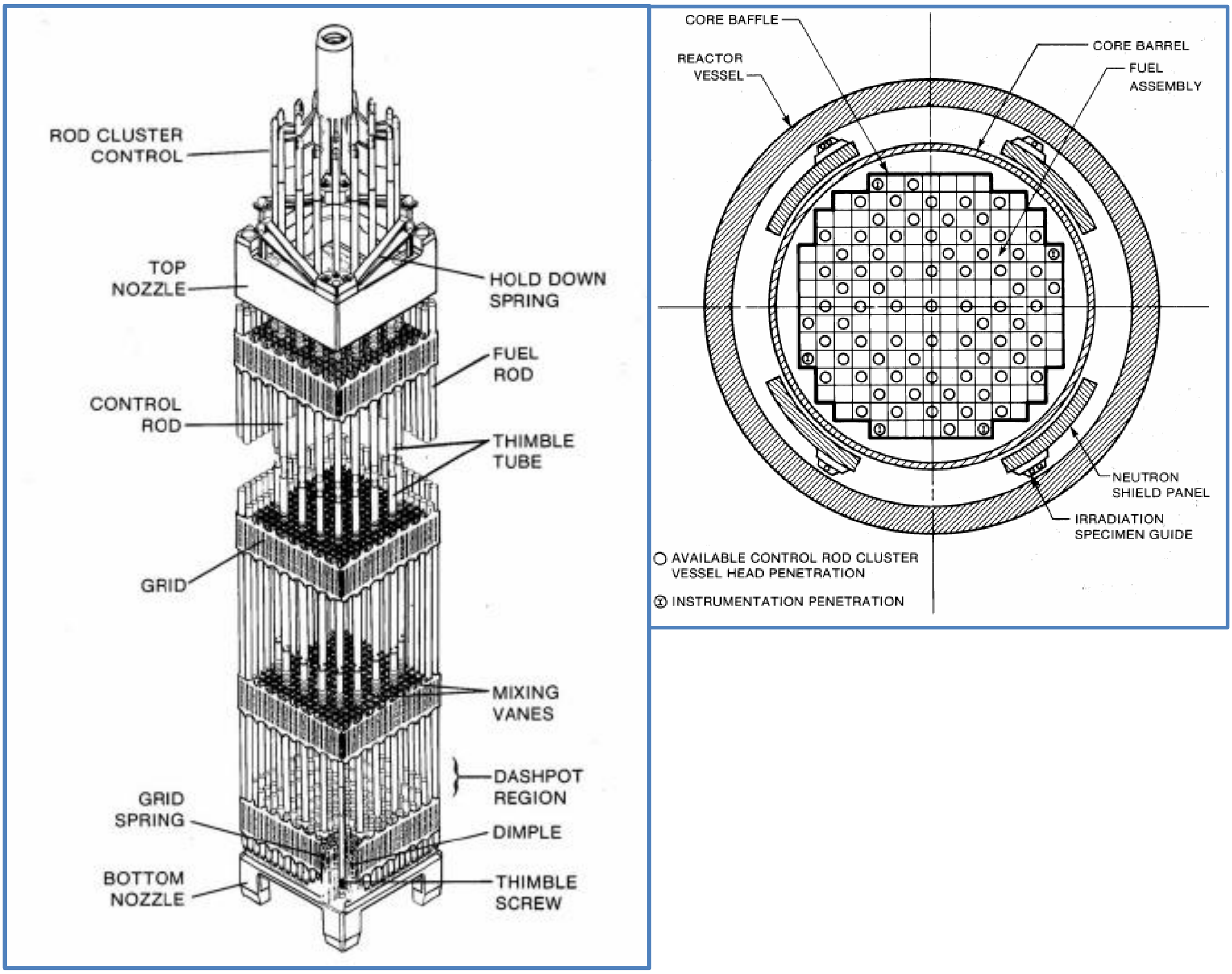


Figure 2-1: Typical PWR fuel assembly with control rods and a cross-section of a reactor core [35]

2.2.2. Spent fuel characteristics overview

A typical PWR like Koeberg's removes and refuels a third of its fuel every 18 months. This means that in a case of an EDF reactor as above in Table 2-1, a third of 157 fuel assemblies will be replaced, i.e. 52 fuel assemblies. The spent fuel will then be transferred to a spent fuel pool for temporary storage for a prescribed minimum number of years depending on the country/reactor technology (at least ten years in almost all reactors [36]). The nuclear composition of spent fuel depends on the specific fuel used (which is determined by reactor type), enrichment, neutron energy spectrum and the fuel burnup [37] [38]. Fuel burnup refers to the amount of thermal energy extracted from fission per unit mass of fuel [39]. Fuel burnup is measured in megawatt-days per tonne (MWd/t) or gigawatt-days per tonne of heavy metal (GWd/tHM).

Most reactors manufactured in the 1970s have a burnup of between 30 and 33 GWd/tHM. Reactors that followed have a burnup of up to 50 GWd/tHM and possible future burnup currently under research will go up to 150 GWd/tHM [40]. Feiveson *et al.* [38] claim that the higher the burnup, the smaller the spent fuel discharged from the reactor. Xu *et al.* [40] also agree that higher burnup reduces the volume and mass of spent fuel per unit mass of electricity generated. However, the storage and disposal volume savings remain uncertain due to higher decay heat. Some of the focus areas of improvement for future nuclear reactors include minimising waste. The relationship between burnup and waste reduction has become a pivotal one for future nuclear reactors.

Below is a table adapted from [38] which shows a typical annual discharge of spent fuel for different reactor technologies. The assumptions made in the below table are:

(i). Reactor with an output of 1GW electrical power

(ii). Reactor with a 90% capacity factor

Table 2-2: Typical annual discharge of spent fuel per reactor technology

Reactor	Typical burnup (GWd/tHM)	Annual discharge of spent fuel (tons)
PWR	50	20
CANDU	7	140
RBMK	15	65

This minor dissertation only focuses on uranium thermal fuelled-reactors. Such reactors have fuel composed of a mixture of isotopes of uranium. The uranium isotopes present in the reactor fuel is Uranium-234, Uranium-235 and Uranium-238. Natural uranium has 0.72% of the fissile Uranium-235 and typical enrichment will increase it to between 3.5 to 4.5%. Higher burnups require higher enrichment.

The spent fuel from a reactor has three categories of radionuclides, namely the actinides, fission products and activation products. This is high-level waste that needs to be treated with the utmost care. Ewing [37] claims that the spent fuel radioactivity at its initial removal from the reactor is a hundred thousand more compared to unused fuel. The approximated radioactivity is about $10^6\ GBq\ MT^{-1}$ (gigaBecquerel per ton of uranium).The spent fuel has the smallest volume of total nuclear waste in a power station but cannot be assembled in small sizes due to heat generation and the level of radioactivity [41]. The fission products constitute about 3-4% of the spent nuclear fuel with a typical mass distribution shown below in Figure 2-2, adapted from [42]. Some of the typical fission products of a PWR include Strontium-90, Iodine-131 and Cesium-137. The half-lives of the fission products vary from a couple of months to a million years. These (fission products) are formed by the disintegration of the fissile material (uranium and plutonium) in the reactor core.

Fission products are radioactive because the ratio of the neutrons to protons is large; this is inherited from the fissile isotopes which undergo fission. In general, the fission products have short half-lives relative to actinides. This short half-life is of the order of tens of years. Most of the fission products also have short decay chains and generally determine the short-term behaviour of the spent fuel

From many fission products generated in the reactor, there are few that are of main concern due to their longer half-lives. The fission products can generally be classified into 3 groups according to their half-lives. These are short-lived, medium-lived and long-lived fission products. The half-lives for these categories are less than 5 years, between 5 and 100 years and 100 years and more, respectively. Hundreds of fission products are short-lived, i.e. less than a year and ultimately would have disappeared entirely in 100 years. Example of short-lived fission products would include Xenon-135 (half-life of 9.2 hours) and Iodine-131 (half-life of 8 days). The short-lived fission products would not be much of a concern due to their short half-life which would make them to have decreased by a factor of million after 100 years

In the medium-lived category, examples include Strontium-90 and Cesium-137, which have half-lives of about 28 and 30 years, respectively. Some of the fission products with half-lives longer than 100 years are Iodine-129 (15.7 million years), Cesium-135 (2.3 million years) and Zirconium-93 (1.61 million years) [43]. Therefore, the long-term activity of nuclear waste due to fission products comes from fission products such as Iodine-129, Strontium-90 and Cesium-137. Iodine-129 decays to Xenon-129 through gamma and beta emissions. Strontium-90 decays to Yttrium-90 with a half-life of 64 hours which ultimately decays to a stable Zirconium-90 [39] [43].

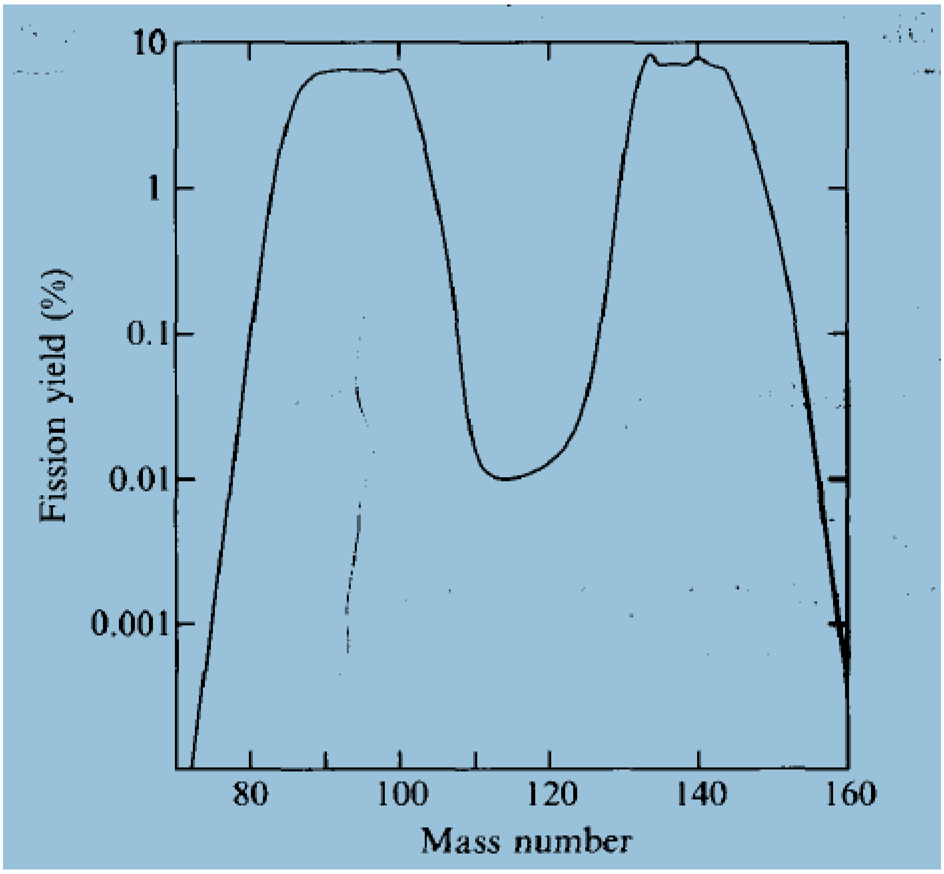


Figure 2-2: Fission products mass distribution [42]

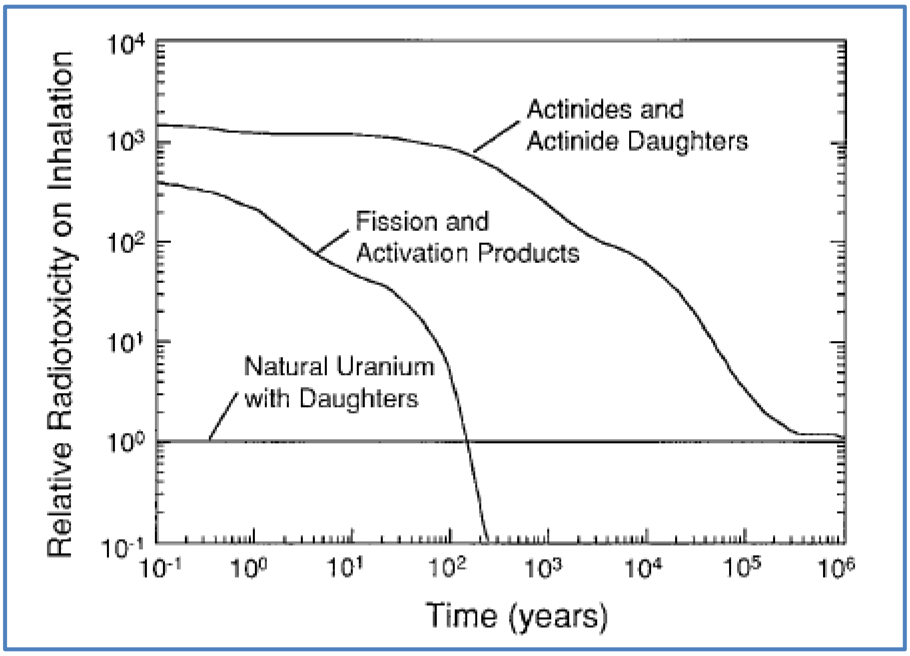
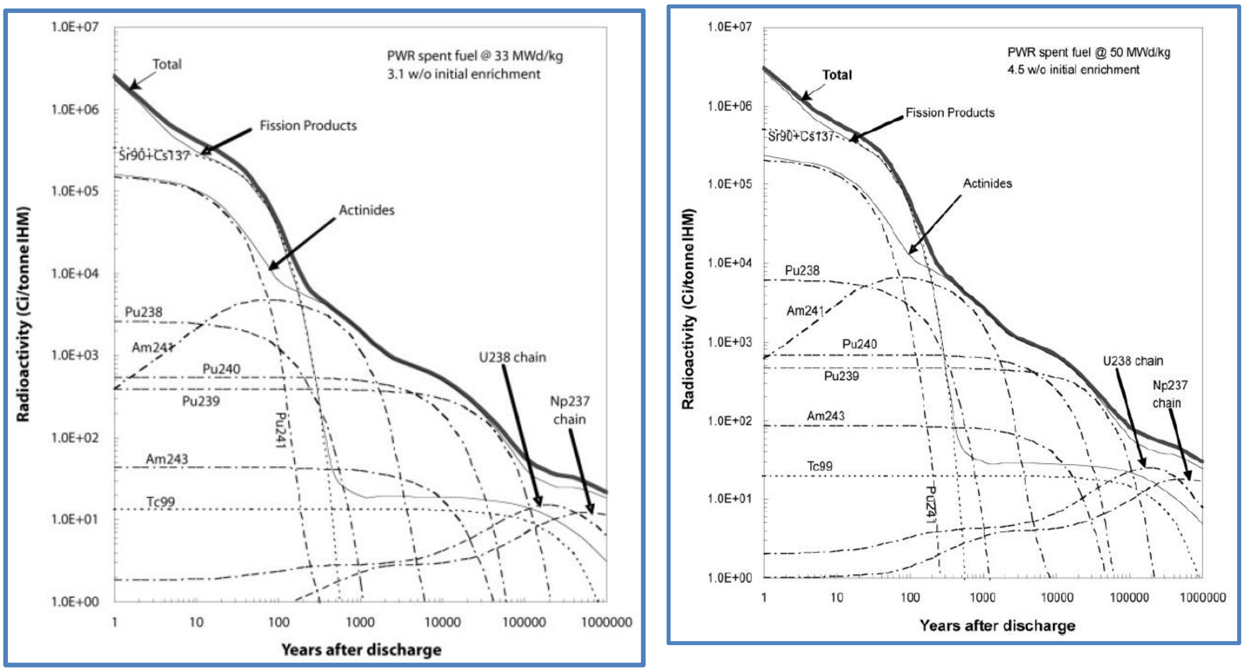


Figure 2-3: Relative radiotoxicity on inhalation of spent nuclear fuel [82]

The actinides are formed when neutrons are captured by the uranium and plutonium isotopes without fission [41] [44]. Actinides are also called transuranic elements. They contribute to heat generation of the spent fuel considerably due to their high activity. Actinides and their daughter products contribute to radiotoxicity for a longer period in contrast to fission products as can be seen from Figure 2-3. Many actinides have longer half-lives in contrast to fission products, of the order of thousands of years. Examples include Plutonium-239 (half-life of about 24000 years, Americium-241 (half-life of about 432 years), Plutonium-240 (half-life of about 6561 years), Plutonium-242 (half-life of 375 000 years) and Neptunium-237 (half-life of about 2.1 million years). They govern the long term behaviour of the spent fuel characteristics.

Activation products are produced from irradiation of materials located in the high radiation level parts of the reactor core [44]. They are generally ignored due to their small contribution to decay heat and radioactivity when compared with fission products and actinides. Examples of activation products include Argon-41, Cobalt-60 and Technetium-99.

Graves *et al.* [45] highlight that the main radiation concern from a spent fuel pool is Cesium-137 due to its relatively long half-life of about 30 years, presence in high volume in the spent fuel, its far-reaching absorption into the human body and transportability in gaseous plumes. Spent fuel tends to be more "rich" in Cesium-137 than in the reactor core. A fission product such as Iodine-131 has a half-life of about 8 days, but its effect of causing thyroid cancer, especially in children, can be mitigated by taking potassium iodide pills [45]. The radioactivity of the fission products and actinides as a function of time after discharge from the reactor is shown in Figure 2-4 below, adapted from [34]. The radioactivity is given in curies per tonne of heavy metal (Ci/tonne HM). A curie is equal to 3.7×10^{10} Becquerel (Bq). These have been computed using computer programs.



(a)

(b)

Figure 2-4: Radioactivity of some of the waste radionuclides as a function of time: (a) a reactor with 33 GWd/tHM and (b) a reactor with 50 GWd/tHM [34]

The radioactivity of the spent fuel is dominated by the fission products up to 1000 years after discharge from the reactor. Typical contributions from Strontium-90 and Cesium-137 are in the range of $2.85 \times 10^{15} Bq$ and $3.7 \times 10^{15} Bq$, respectively. These values differ with the initial fuel enrichment and burnup. The understanding of the long-term physical, chemical and thermal behaviour of the spent fuel is critical for safe spent fuel management. This calls for continuous monitoring and analysis of the spent fuel.

2.3. Spent fuel pool

The amount of decay heat produced by high-level waste and the radioactivity makes it require cooling and shielding. Between 6 and 7% of the heat generated during the normal reactor operation is initially generated by the decay of spent fuel [42]. High-level waste is regarded as one of the major issues arising from the use of nuclear power reactors by various organisations such as the World Nuclear Association and the United States Nuclear Regulatory Commission [46] [47].

The spent fuel pool (SFP) has three main general objectives, namely to cool the spent fuel and thereby prevent heat-up from radioactive decay, to prevent criticality accidents from spent fuel and shield the workers and the environment from radiation [48]. The SFP is also used to store fresh fuel which must be separated from the spent fuel by a sufficient distance to prevent criticality accidents.

It is therefore essential for the SFP system to continually purify, filter and maintain the required water inventory in all different components of the SFP. In most designs, spent fuel pool is a collective term that refers to different systems used to cool the spent fuel. These systems include the spent fuel pool, spent fuel cask compartment and the transfer compartment in a typical Westinghouse design [49] and generation II of the French PWR design. The (European Pressurized Reactor) EPR has six subsystems (functional areas) making up a spent fuel pool; incoming cask transport, cask reception and preparation, fuel removal, rack movements and long term storage [50].

Spent fuel pools are either placed inside the reactor building or the building adjacent to it. Although a third of the fuel is replaced during a fuel cycle, all the fuel assemblies are normally removed to enable reactor vessel maintenance and inspection [48]. This makes the SFP experience the highest heat generation just after all the fuel assemblies have been offloaded. The removal of the fuel assemblies from the reactor vessel is done with the whole space above the reactor vessel immersed in borated water [51]. This removal and movement of fuel assemblies under water require that the water in the SFP is always as optically clear as possible.

The design of spent fuel pool differs from reactor to reactor and sometimes from technology to technology and from site to site [52]. Most current SFPs can be classified into two types: at-reactor (AR) and AFR (away-from-reactor) SFPs. The AR SFPs pools are used for temporary fuel storage while the AFR SFs are used for interim storage of spent fuel prior to reprocessing or waiting for final disposal [52]. The major general differences between the two pools are:

- The spent fuel in the at-reactor is characterised by higher decay power than in the away-reactor pools
- ARs are generally closer to the reactor vessels and are smaller in size

The pools are monolithic structures made of reinforced concrete and stainless steel liners to prevent leakage as well as maintenance of cooling water quality. Some SFPs such as Russia's VVER use borated water for the cooling while others such as BWRs use demineralised water. The depth of SFPs varies but typically is about 12 metres. A wide variety exists in terms of width and length with most typically between 6 and 12 metres in length and 9 to 18 metres in width [53]. Most of the SFPs have minimum water coverage of 3 meters [52] [53]. In Figure 2-5 and Figure 2-6 generic SFP designs for a PWR and a BWR adapted from [52] and [54] are shown.

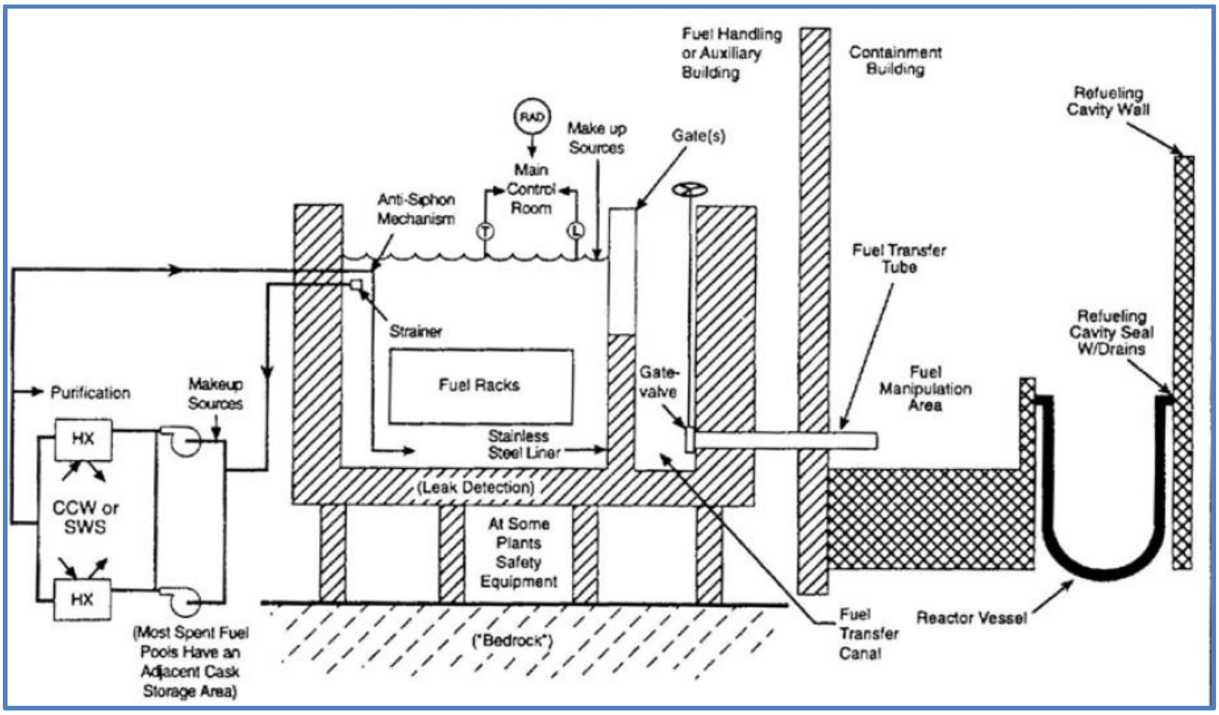


Figure 2-5: Typical SFP design for a PWR [52]

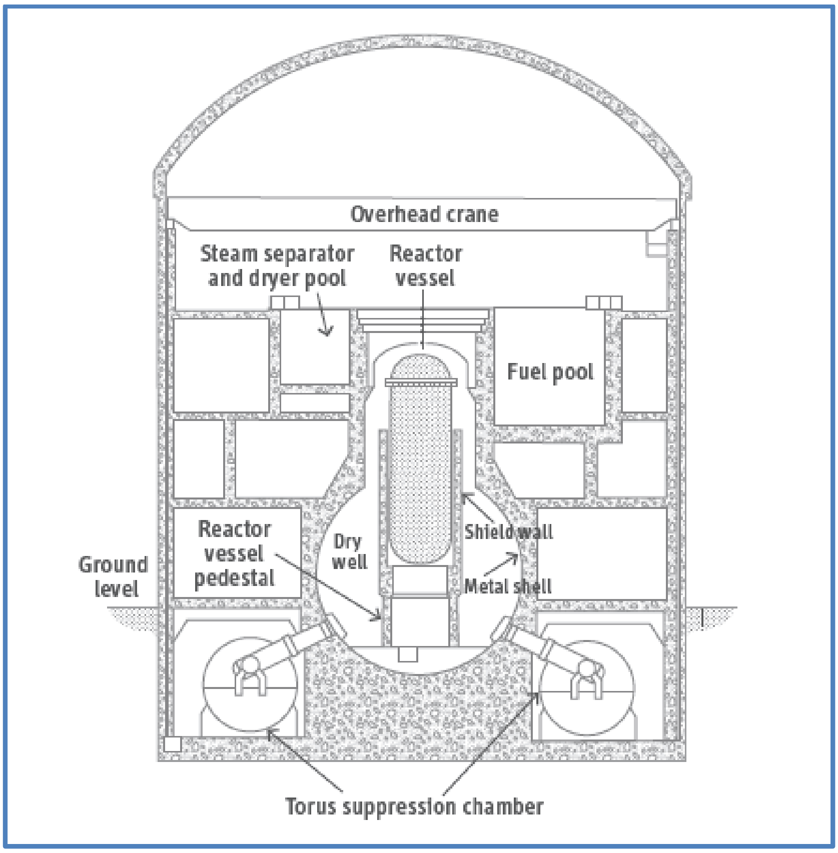


Figure 2-6: Typical SFP design for a BWR [38]

The SFPs are also used to store other components and in-reactor equipment such as control rods, primary neutron source, etc. The irradiated spent fuel is removed from the reactor core and placed in spent fuel racks and gets transferred to the pool. The design of these racks is crucial in preventing criticality accidents in the SFPs. A variety of designs are in use today. Figure 2-7 below shows some of the designs used in PWRs [52].

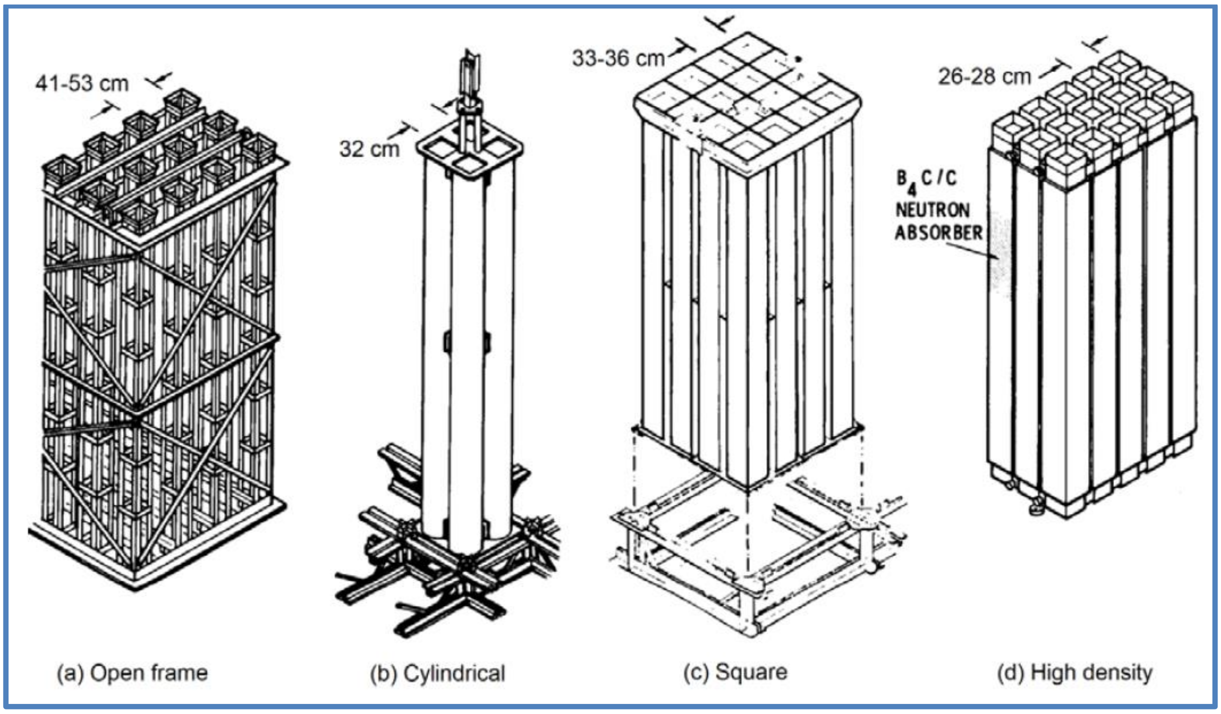


Figure 2-7: Spent fuel rack designs for PWRs [52]

Borated water is used for criticality control in a PWR. This is in addition to sufficient space allocation between the fuel assemblies. The design of the spent fuel racks must ensure the sub-criticality of the spent fuel in the pools. Most nuclear companies in the United States have been expanding their storage capacities by re-racking their spent nuclear fuel pools. This is achieved by decreasing the distances between the fuel assemblies [55]. The SFPs generally have dedicated cooling systems, as shown in Figure 2-5 above. Coolant is pumped through heat exchangers, and heat is rejected into a heat sink of the nuclear power plant. It is clear that the SFPs must be designed to have a lifespan many years more than the reactor. This is to cater for the interim storage of the last fuel to be withdrawn from the reactor before decommissioning of the nuclear power plant. The EPR reactor has a lifespan of 60 years, and its pool is designed with a lifespan of 100 years [50].

Although there has never been a serious spent fuel pool accident (where there was a total loss of coolant, and the accident originated purely from the SFP) in the past, possibilities of losing the SFP cooling circulation exist, and the consequences can be hazardous. If the SFP cooling were to be lost, the decay heat would boil and evaporate the water thereby compromising its shielding and cooling properties. This would then lead to the release of radioactive nuclides inside the SFP room and the environment. When the fuel reaches certain temperatures, hydrogen production will start because of interaction with the Zircaloy cladding and the water. The governing equation of this interaction is given by:

$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + heat$ [56].

This can lead to a hydrogen explosion. Some reports suggest that this is what happened at Fukushima in the reactor core due to the absence of cooling while some suggest that hydrogen was produced in the SFP and travelled back to the reactor since the systems are interconnected [57] [58].

Causes of cooling loss can range from the malfunction of the cooling system to the loss of water inventory in the pools [57]. The design of the SFPs must, therefore, take into account scenarios where one or both of the above occurs. In contrast to reactor LOCA (loss of coolant accident), the SFP LOCA would be slow due to a large amount of water present in the SFP and low heat generation. However, the SFP only has one physical fuel barrier which is the cladding [52]. The worst case scenario should look at a loss of cooling that occurs when the decay heat is at its maximum, i.e. immediately after fuel off-loading.

Nimander [57] considers passive systems that prolong boiling of the cooling water and one that ensures that boiling never occurs. He concluded that a passive safety system is not possible to mitigate the consequences of worst case scenarios using gases at atmospheric pressure. Although other designs may exist of passive cooling, he concludes that large enough spent fuel pools are the simple and efficient way of prolonging the boiling of cooling waters. It should be noted that this conclusion was based on a PWR reactor and may not necessarily apply to all PWRs. With the increase in spent fuel in nuclear power plant sites, the possibilities and economics of having large pools may not be feasible.

Ye *et al.* [59] have also studied the use of a passive system based on heat pipes to remove the decay heat in an SFP. They concluded that such a system would operate safely and prevent accidents even during the highest heat decay. The simulations conducted are only applicable to a CAP1400 reactor. Carasik *et al.* [56] have studied the design of an SFP that has a passive cooling feature. This design is the fluidised bed SFP. “A fluidised bed works by passing fluid or gas through a bed of granular solids at a high enough velocity to suspend the bed material, causing it to behave like a fluid” [56]. The conclusions drawn from simulations that were done from this design was that in addition to providing passive cooling, the fluidised bed SFP would also prevent reactivity accidents. The economics and other related components such as fuel movement design of such SFPs still need to be investigated in detail. Other reports have indicated that the main concern with SFPs is vulnerability to terrorist attacks [48] [54]. This can only be true for SFPs that are not located at reactor sites because reactor sites should be designed to withstand such attacks.

The design of an SFP needs to provide an assurance that the systems of the SFP can withstand the effects of natural phenomena like earthquakes, design basis accidents and support of normal operations as well as the provision of fuel monitoring capabilities. A Westinghouse technology SFP, for instance, is designed to meet General Design Requirements (GDC) 2, 4, 61 and 63 prescribed by the Nuclear Regulatory Authority (NRC) in the United States. The first GDC give assurance for the ability to withstand the effects of natural phenomena [60].

In summary, the following are critical considerations when designing a spent fuel pool:

- Provision of cooling water monitoring capabilities. This includes key parameters in the SFP such as water level, temperature and radiation. The desired design is one that should enable monitoring of these parameters even with a total station blackout.
- Ensuring sub-criticality at all times
- Passive cooling. This has been a major subject of discussion and interest, especially after the Fukushima nuclear accident in March 2011.
- Ability to withstand natural phenomena

- It is evident that SFPs will remain a vital interim storage technology for a number of years. It is therefore essential that designs are done to adapt to store expected future fuel designs.

Spent fuel pools are designed to house the spent fuel for the duration of the reactor operation. The rate of filling up the SFPs depends on the fuel design, frequency of refuelling and the size of the SFP. Most of the SFPs in operation today were designed to be interim storage where the spent fuel would stay for a maximum of about 10 years before transportation for final disposal. The SFP technology has proven to be mature with one issue remaining, the introduction of efficient and cost-effective passive cooling systems.

2.4. Dry cask storage

It has been highlighted above that the increase in the amount of spent fuel in nuclear power plants is leading to decreased space in the SFPs. Dry cask storage facilities were designed to solve this challenge. Dry cask storage systems were initially designed for one purpose, to store the spent fuel after it has been cooled in the SFP with the intention to “buy” more time to find viable long term solutions. Others have argued that dry casks were initially designed to transport spent fuel to reprocessing facilities. In the United States, most dry casks were initially licenced to operate for 20 years but were later extended to 40 years seeing that no long term solutions were agreed upon yet and that they posed no foreseeable technical danger [61]. The US NRC study on the casks has concluded that the casks can be used for up to 60 years.

The nuclear industry has now developed and licenced dry cask storage facilities that are used for both transportation and storage of spent nuclear fuel. Most of the transportation aspect has been from the reactor site to the reprocessing plant. Such casks are generally called dual-purpose dry casks. There have been proposals of multi-purpose casks which are expected to perform the storage, transportation and disposal of spent nuclear fuel. These are not yet licenced and most are in the design phase [62]. A variety of designs exists for dry cask storage facilities but most can be traced to have similar characteristics. The variation in design ranges from the purpose, orientation, material and siting [63]. General requirements of dry casks are similar to those of SFPs with few additions. These requirements include radiation containment, managing of decay heat, prevention of criticality, resistance to earthquakes, floods, etc. [64].

Hambley *et al.* [62] give three general classifications of dry storage, namely the vault, casks and silos. Each category has a variety of designs within each category. A vault is a large ferroconcrete structure with multiple large cavities used to store spent fuel. The external structure of a vault is used for radiation shielding. Heat removal is achieved through forced or natural air convection [65]. Below is a picture adapted from [66], showing the modular vault dry storage system at Paks nuclear plant in Hungary.

A vault storage system can be constructed above or below the ground. This dry storage system requires installations to be done in very stable geographic areas with high-level security and robust monitoring systems. It is desirable to have a vault storage system within the perimeter of the nuclear power plant. This would eliminate additional site establishment costs, security and transportation of the spent fuel from the SFP to the vault dry storage system. Other countries that have employed such storage systems include Canada, France, the United Kingdom and the United States of America [67].

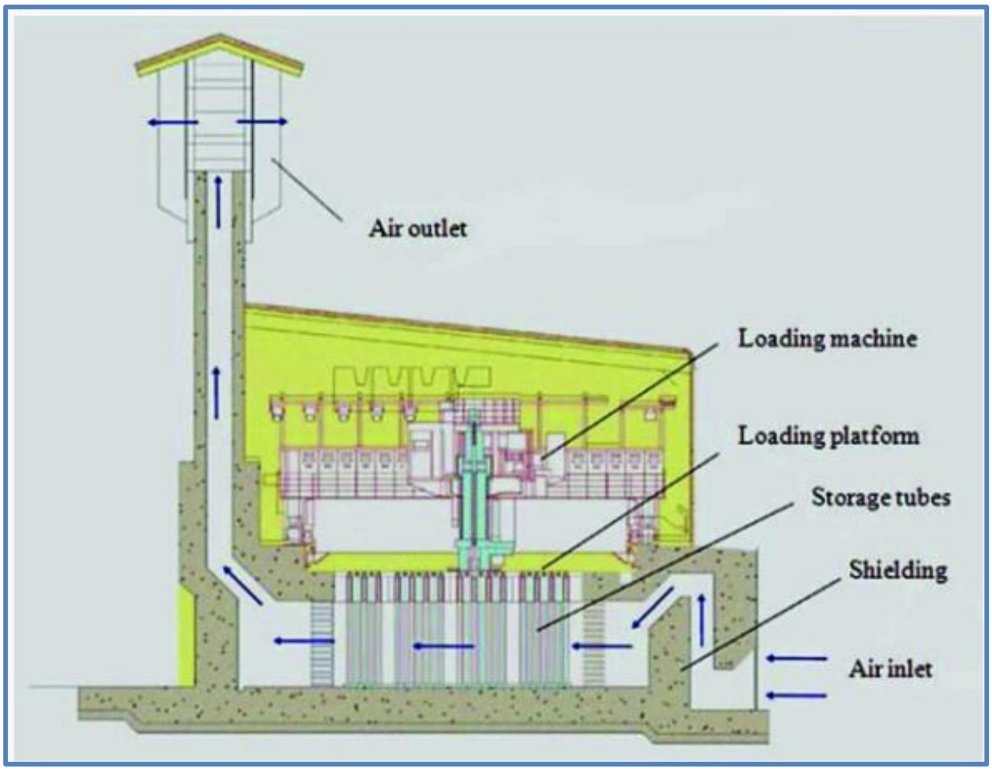


Figure 2-8: Vault storage system at Paks nuclear power plant [66]

Romanato [68] defines silos as “horizontal or vertical concrete cylinders with metallic canisters inside them”. The spent fuel is stored in the canisters. Natural air convection is used to cool the spent fuel using special ducts. In contrast to a building or structure in a vault system, the concrete provides radiation shielding and the metallic canisters perform the containment function. These have been deployed in countries such as Armenia, Korea and the United States of America [65]. Below is a picture of a NUHOMS (Nutek Horizontal Modular Storage) adapted from [69] showing a dry storage facility making use of silos.

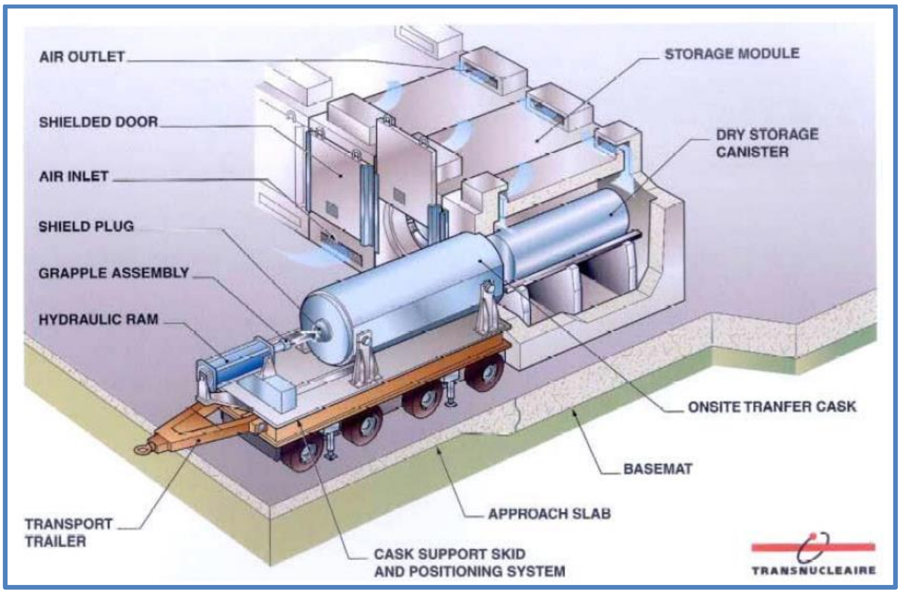


Figure 2-9: NUHOMS dry storage modular system [69]

The dry cask technology has two major groups which include metal and concrete casks. It is clear that the design of such casks should ensure leak tightness and radiation shielding. The cooling of the fuel inside a dry cask is passive. A combination of heat conduction through solid materials and natural convection plays a role in heat removal. The use of concrete, lead, steel and other cask materials are used to provide radiation shielding. The geometry of the fuel assemblies inside the dry cask is designed to control criticality [48]. Dry casks also come in horizontal and vertical orientations. Dry cask storage, like spent fuel pools, can be Away-From-Reactor (AFR) or At-Reactor (AR). Most metal casks are used for both transportation and storage [70]. Countries such as Germany, the USA and Switzerland have already employed dry cask storage. Below is a picture showing typical metal and concrete casks, adapted from [68].

Figure 2-10: Metal casks on the left and concrete casks on the right [68]

A table detailing a CASTOR 440/84 metal cask parameters is given in appendix A and B along with a diagram showing major components of the cask; both these are adapted from [70]. The design of casks is specific for the reactor and fuel technology in use. The amount of time that the spent fuel spends in the SFP before it can be removed varies from country to country according to regulatory bodies. Excess water must be removed from the fuel prior to transporting it to a dry cask facility. This is done to decrease corrosion and pressure-related challenges [62].

Dry storage systems provide the following:

- Mobility,
- modularity, and
- passive cooling and shielding.

Heat dissipation of a dry cask storage system is lower than that of an SFP and therefore cannot store fuel that has just been off-loaded from the reactor. It is also worth noting that the dry storage facilities at Fukushima nuclear power plant during the 2011 earthquake and tsunami remained intact and were reported not to have had any safety concerns [45].

The IAEA [71] gives a comprehensive cost analysis for spent fuel storage where the above dry storage options are briefly analysed. The report identifies major categories of costs associated with spent fuel storage, i.e. capital costs, operational and maintenance costs (O&M), decontamination and decommissioning costs (D&D). Figure 2-11 from [71] further explains what each category comprises of:

Category of costs	Project phase	Remark
CAPITAL	Project Definitions	Alternatives are evaluated to select the best option. A plan for project implementation is established.
	Design Engineering	The facility is designed. The investment plan established.
	Regulatory Approval	Safety analysis documents are prepared. Licences are issued by authority for the facility.
	Construction	The storage facility is built.
O&M	Spent Fuel Loading	Spent fuel is placed in storage in the facility. Dry storage casks/modules are procured.
	Storage Only	Monitoring is carried out and protection of the stored spent fuel is provided.
	Unloading	Spent fuel is removed from storage. Spent fuel is transferred to a transportation cask (if applicable). Spent fuel is shipped to off-site destination.
D&D	Decontamination and Decommissioning	The fuel storage facility is decontaminated and dismantled. Site is restored to its original condition.

Figure 2-11: General categories of costs associated with spent fuel storage [71]

The capital costs of AR SFPs are included in the initial capital costs of the entire nuclear power station and work out cheaper due to other costs that would have been covered. These costs include land, security systems, transportation, etc. Spent fuel pools that are further from the reactor incur more capital costs, especially if it is at another site. The cost of spent fuel management varies from 1.5% to 10% of the total power station construction cost [71].

2.5. Reprocessing

Reprocessing and recycling is one of the spent fuel management options in use today by several countries, including France, Japan, the United Kingdom, India and Russia. Other countries such as Switzerland, Belgium and Netherlands do not have reprocessing facilities but send their spent fuel for reprocessing. Switzerland sends their spent fuel for reprocessing in France and UK while Belgium and Netherlands send them to France [72] [73] [74]. This spent fuel management option is also referred to as reprocessing and recycling or closed fuel cycle. The justification and interest in the further development of reprocessing date back to the beginning of civil application of nuclear power in the early 1950s. The main idea was to recover the fissile material to be re-used in the fast breeder reactors, thereby making the economics of nuclear power more competitive [75] [28]. However, the first deployment was to recover plutonium for the production of weapons during World War II in the 1940s [76].

Reprocessing is primarily aimed at recovering the unburnt uranium (U-235), U-233 and plutonium (Pu-239). Plutonium is generated from the fertile Uranium (U-238) and Uranium-233 is bred from fertile Thorium. In [77], five key factors for reprocessing are highlighted. These are optimising waste and disposal conditions, minimum impact on the environment, increasing economic competitiveness of nuclear power, increasing proliferation resistance and uranium resource conservation. There are two processes used in reprocessing of spent nuclear fuel. These are pyrochemical and hydrometallurgical (aqueous) processes.

The end products of reprocessing are therefore Uranium, Plutonium and conditioned radioactive waste. The Uranium and Plutonium recovered are then fabricated to form MOX (Mixed-oxide) fuel which is then recycled in the pressurised water reactors. The use of MOX fuel assemblies requires minor changes in the reactor for its effective operation. This is due to the difference in the nuclear physics of uranium and plutonium. The isotopes of plutonium in reprocessed PWR fuel are Pu-238, Pu-239, PU-240, Pu-241 and Pu-242. The probability of fission is almost the same in fast reactors and only the odd-numbered isotopes in PWR. The neutron flux resonance region for MOX fuel is higher which effectively reduce the reactivity worth of control rods. This leads to a compromise in shutdown margins and hence a need for minor changes in the control rods in the reactor [78].

The nuclear physics of the MOX fuel also vary in terms of the effect of delayed neutrons in comparison to that of uranium dioxide. The thermal absorption probability of the control rod is reduced in MOX fuel because the prompt neutron life is short [78].

In comparison to the normal fabrication of uranium fuel, the MOX fuel fabrication is substantially costly and more hazardous. Twala [28] shows that reprocessing of spent nuclear fuel was accepted on the basis of promising economic competitiveness, which assumed certain conditions. However, these were later challenged with increased proliferation risks and health risks, among others.

2.6. Deep geological repository

One of the current solutions to high-level nuclear waste is a deep geological repository (DGR). The US NRC defines a deep geological repository as an excavated underground facility designed, constructed and operated for safe and secure permanent disposal of high-level nuclear waste. A combination of engineered barrier systems, site's natural geology and geochemical systems are used to isolate nuclear waste from the environment [47].

Canada is currently in the process of finalising a site for a deep geological repository. Finland's deep geological repository in Olkiluoto is expected to be operational in 2023. Other countries such as Korea, Japan and France are in the engagement phase with local communities, members of civil society and all concerned parties with regard to siting of DGRs [79].

Proposals of DGRs also include a reversibility feature wherein at a later stage the spent nuclear fuel can be retrieved to be reprocessed or completely destroyed. This is in the hope of finding technical solutions that would make reprocessing more viable or discoveries that could completely "destroy" the waste. It is not yet known how this reversibility feature impacts the cost and safety of the DGRs. The DGRs deploy the multi-barrier concept, which is summarised below:

(i). prevention of the release of radioactive waste,

(ii). delay of the release of radioactive waste, and

(iii). delay of the release and dilution of radioactive waste if they still escape to the environment despite the above.

Below is a table adapted from [34] which shows functions of different barriers for DGRs. The contributions to safety from different barriers are slightly different and come at different times.

Barrier	Function
Matrix (glass, concrete, bitumen)	Confinement Limitation of the RN release
Container and overpack	Transport Retrieval Protection of the packages
Clay engineered barrier (optional)	Protection of the packages Delay migration
Geological barrier	Protection of the first barriers Delay migration Dilution

Figure 2-12: Functional analysis of different DGR barriers [34]

In many designs, the prevention of the release of radioactive waste is achieved by using glass or concrete to enclose the spent fuel in casks. Engineering barriers such as overpack and clay layer are used to protect the spent fuel packaging and delay any release of radionuclides. The dilution of radionuclides is achieved using the geological barrier which is the host rock. The designs will differ from site to site depending on the host rock, nuclear waste to be disposed and other factors. With the current technology, the DGRs need to confine the waste for at least 10 000 years for vitrified waste from reprocessing and more than 100000 years for waste directly from the open fuel cycle. The geological repository also requires not only the understanding of the long-term spent fuel behaviour but also the chemistry of the geology for that site. The geological chemistry of the site should also study the changes that will be brought by these DGRs.

3. Aspects of physics and engineering of interim dry storage

In this chapter, some of the most important physics and engineering aspects of dry cask storage are briefly discussed. These include radiation containment, spent fuel, sub-criticality, decay heat removal, site location aspects, corrosion of the casks, operability and monitoring.

3.1. Radiation containment

Radiation containment is about the retention of all radiation within the cask and prevention of any leaks into the environment. The current practice involves a multiple-barrier approach where the first shield is the cladding. The materials used to manufacture the casks then become second and third barriers of radiation. In some facilities, the casks are placed in buildings in order to add another barrier for radiation containment from the environment. The principle of radiation containment is based on the attenuation of the radiation as it interacts with matter. Radioactive nuclides emit different types of radiation which interact with matter in different ways. These include alpha, beta and gamma radiations.

Alpha radiation is the emission of two protons and two neutrons at high speed/energy which are bound together into a particle like the helium nucleus [80]. This is called alpha decay and the two neutrons and two protons emitted would be called alpha particles, i.e. two protons and two neutrons make one alpha particle. In this phenomenon, both mass and charge is lost by the parent nucleus undergoing alpha decay. The mass decreases by four units while charge decreases by two. An example of alpha decay is that of a uranium-238 given by the following equation:

$$^{238}_{92}U \rightarrow \, ^{234}_{90}Th + \, ^{4}_{2}He + \gamma$$

Beta decay is the emission of an electron or positron from the nucleus. This is a phenomenon that occurs in nuclides that are proton or neutron-rich. In a proton-rich nuclide, the proton is converted into a neutron with the emission of a positron and a neutrino. This is called beta plus decay and the positron is called the beta plus particle. In a neutron-rich nuclide, the neutron is converted into a proton with the emission of an electron and an antineutrino. This is called beta minus (β^-) decay and the electron emitted is called a beta minus particle. In both beta minus and plus decays, the mass number remains unchanged while the atomic number of the nuclide increase or decrease by one, respectively. Beta decay can be summarized by the following equations, which represent the conversion of the neutron into a proton or vice versa with the release of leptons and anti-leptons.

$$n \rightarrow p + e^- + \bar{v} \text{ … Beta minus decay}$$

$$p \rightarrow n + e^+ + v \text{ … Beta plus decay}$$

Examples of beta minus decay and beta plus decays are that of carbon-14 decaying to Nitrogen-14 and carbon-10 decaying to boron-10 respectively. The equations of these processes are given as follows:

$$^{14}_{6}C \rightarrow \, ^{14}_{7}N + \bar{v} + e^-$$

$$ {}^{10}_{6}C \rightarrow {}^{10}_{5}B + v + e^{+} $$

Gamma radiation comprises of emission of high energy photons. The number of nucleons is not altered during gamma decay. Gamma radiation usually follows beta and alpha decays. This process generally starts with a neutron captured by a target nucleus to form a compound nucleus in an excited state which decays by emitting gamma rays. An example is the formation of U-239 by capturing a neutron. This phenomenon is expressed below:

$n + {}^{238}_{92}U \rightarrow {}^{239}_{92}U^{*}$ … Formation of the excited compound

${}^{239}_{92}U^{*} \rightarrow {}^{239}_{92}U + \gamma$… De-excitation of the excited compound

All these processes are radioactive decay phenomena for unstable nuclides. The interaction of radiation with matter can be broadly divided into three groups. These are the interaction of radiation with charged particles, photons and neutral particles. All these see matter in terms of its basic constituents, which are electrons, neutrons and protons [81].

3.1.1. Charged particles

Interaction of charged particles with matter is usually presented as two groups, heavy charged particles which include alphas, protons, muons and pions and light charged particles which include electrons and positrons. All charged particles interact with matter primarily by electromagnetic force. Charged particles continuously lose energy and are deflected from their incident direction as they pass through matter. Both inelastic collisions and elastic scattering are the driving processes behind the energy loss and deflection from the incident direction of the charged particles. The inelastic collisions occur with the atomic electrons of the material which the particle interacts with, and elastic scattering occurs from the nuclei of the material. The inelastic collision is primarily responsible for the energy loss of these charged particles. The distance travelled by charged particles through the material before they lose all their energy is called the range of the particle.

Heavy charged particles transfer their energy into the matter during the inelastic collision, thereby causing ionisation or excitation of the atoms of the matter they interact with. The rate at which a particle loses energy per unit length as it interacts with matter is called the stopping power of the medium. This can be calculated using the Bethe-Bloch equation given below:

$$ -\frac{dE}{dx} = \left(\frac{ze^2}{4\pi\varepsilon_0}\right)^2 \frac{4\pi Z\rho N_A}{Amv^2}\left[\ln\left(\frac{2mv^2}{I}\right) - \ln(1-\beta^2) - \beta^2\right] $$

This equation includes relativistic effects and the symbols are described in Table 3-1 below.

Table 3-1: Bethe-Bloch formula parameters

$v = \beta c$	Ion velocity
ze	Ion charge
m	Mass of the electron
N_A	Avogadro's constant = 6.02214086 x 10^{23}
A	Atomic mass number of the stopping material
Z	Atomic number of the stopping material
ρ	Density of the stopping material
I	Mean ionization energy

The negative sign indicates the fact that the energy of the particle decreases with distance. It is common to give the stopping power value in units of energy loss per mass per unit area by dividing by the density, ρ. This is called mass stopping power. Tables of range and stopping power have been computed but further analysis can be used to understand the effects of changing variables in the Bethe-Bloch equation. These variables include the particle's energy, particle type as well as the stopping medium.

The variation of the mass stopping power of protons in an aluminium medium is given in Figure 3-1, adapted from [42]. The variation shape is typical for any charged particle in any medium. Both scales in the figure are logarithmic.

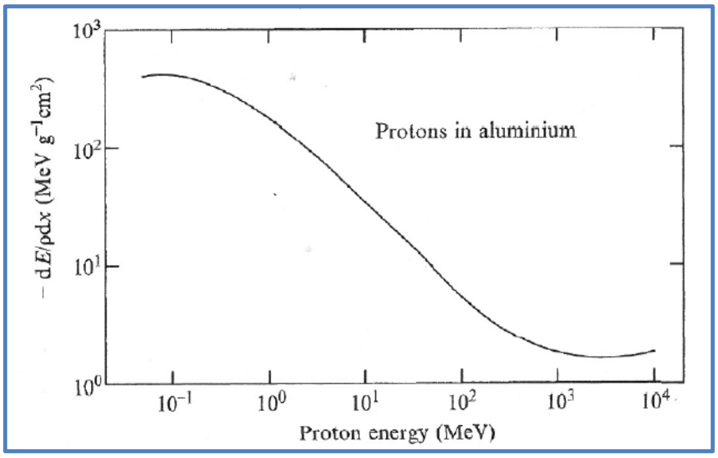


Figure 3-1: The variation of mass stopping power of protons in aluminium medium [42]

As seen in Figure 3-1 above, the energy dependence is dominated by the $\frac{1}{v^2}$ term between energies of approximately 1 MeV and 1GeV. In this region, the mass stopping power can be approximated with the following formula:

$$\frac{dE}{dx} = \frac{constant}{E^k}$$

The value of k is approximately 0.8. A heavy charged particle will travel through a medium starting with some energy until it loses all its energy i.e. the energy changes of the particle will move from right to the left of Figure 3-1. The rate of energy loss increases as the particle energy decreases, thereby increasing the number of ions in the medium per unit distance. This number of ions produced per unit distance is called ionization density, which increases along the path of the particle. The ionization density will be at its highest just before the particle goes to rest after losing all its energy. The particle's energy will go to zero, thereby defining the range. The plot of the ionization density against the distance travelled is called the Bragg curve. It is not discussed here in detail but worth mentioning that the treatment of some localized tumours is based on the enhancement of ionization in the Bragg peak using heavy charged particles. A Bragg curve is shown in Figure 3-2 below, adapted from [82].

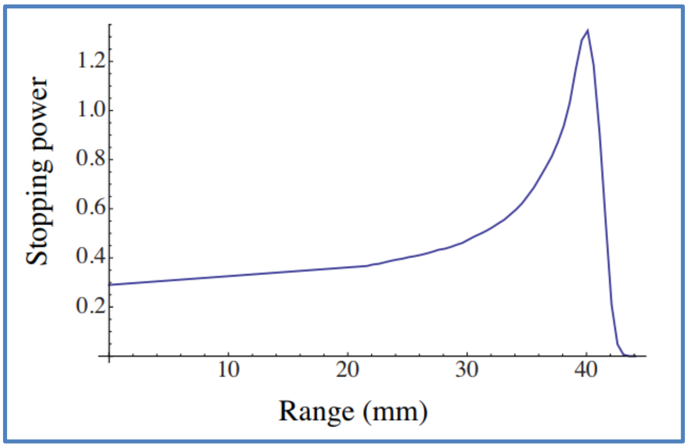


Figure 3-2: Bragg curve for protons adapted from [82]

The light charged particles also interact in a similar manner as the heavy charged particles with the exception of their high speed due to their very light mass. This makes them have a much smaller stopping power in comparison to heavy charged particles and much more penetrating. Electrons also lose more energy in a single collision when compared to heavy charged particles due to their light mass. Their paths are also erratic due to large deflections they experience when they collide with other electrons. These collisions also cause sudden changes in the direction and speed, which result in the emission of electromagnetic radiation. This is called bremsstrahlung radiation. The total energy loss for electrons is, therefore, due to collisions with electrons of the medium and radiated energy [82] [42] [81].

3.1.2. Photons

There are 3 possible interactions from photons, namely, photoelectric effect, Compton scattering and pair production. The lack of electric charge in the photons makes them different from the rest. Photoelectric effect is when the energy of the photon is converted into releasing an electron from the atom, typically in one of the inner electron shells of the atom. The atom may then de-excite by releasing less tightly bound electrons. These electrons are called Auger electron. In some instances, an electron from a higher shell fills that vacancy in the inner shell with a release of an X-ray photon which is known as X-ray fluorescence [81]. The X-ray photon may interact with the medium. The electron that gets released from the inner shell called the photoelectron would have kinetic energy given by the following equation:

$K_e = E_\gamma - B_e,$

where E_γ is the incident photon's energy and B_e is the binding energy of the electron. This phenomenon is for low-energy photons of less than 0.1 MeV. In Compton scattering, an incident photon with energy E_γ scatters from a nearly free electron. The photon emerges with less energy E'_γ and the electron scatters with kinetic energy T. This kinetic energy is expressed as follows:

$T = E_\gamma - E'_\gamma.$

The scattered photon's energy is given by the following equation:

$$E'_\gamma = \frac{E_\gamma}{1 + \left(\frac{E_\gamma}{mc^2}\right)(1 - cos\theta)}$$

Theta, θ is the angle between the incident photon path and the scattered photon path. The scattered photon's energy will be at its minimum when the angle between the scattered photon path and incident photon's path, θ is 180° and maximum when θ is 0°. Compton scattering is dominant in photons of energies between 0.1 MeV and 10 MeV. The derivation of the scattered photon's energy is given in appendix A. Compton scattering is not as dependent on the atomic number Z and incident photon's energy E_γ as the photoelectric effect.

Pair production is when the entire incident photon's energy is transformed to form an electron-positron pair. The total kinetic energy is given by the following equation:

$$K_{-e} + K_{+e} = E_\gamma - 2mc^2$$

The two terms on the left, K_{-e} and K_{+e} are the electron and positron kinetic energies respectively, E_γ is the incident photon's energy, m mass of the electron and c is the speed of light. This pair production energy equation shows that there is a minimum incident photon energy threshold for this interaction to take place. This energy threshold is $2mc^2$ = 1.022MeV. This interaction is therefore only dominant for photons of energies approximately above 10 MeV. The presence of a heavy nucleus is also required for the conservation of both momentum and energy in order for this process to take place. This makes pair production to have some independence on the atomic number Z.

The remaining incident photon energy which does not get transferred to the absorbing medium during the interaction will be in the form of rest masses of electron and positron. The positron will slow down and get attracted to an electron just before it comes to rest. The interaction of these two will result in annihilation in which the rest masses of the positron and the electron are converted into 2 gamma rays. These gamma rays will each have energy of 0.511 MeV and will be emitted in opposite directions. This will ensure that momentum is conserved and may, in turn, interact with absorbing medium by either photoelectric absorption or Compton scattering as explained above. This phenomenon is called positron annihilation [44].

3.1.3. Neutrons

The interaction of neutrons with matter is dependent on the neutron's energy and the type of nuclei they interact with. Neutrons do not experience Coulomb repulsion force as they approach a nucleus due to the fact that they are neutrally charged. Thus neutrons will travel in straight paths and will either scatter or be absorbed in the medium. In the former, the neutron's energy is lost in successive collisions. The scattering process can either be elastic or inelastic. In elastic scattering, the total kinetic energy between the neutron and the nucleus it interacts with is conserved. The inelastic scattering process leaves the nucleus in an excited state which may later decay by gamma radiation. This is possible when the neutron has sufficient energy to excite the nucleus.

The absorption process can occur via different reactions, namely, radiative capture, fission and other reactions (n, x). In a radiative capture process, a neutron is absorbed into the nucleus, thereby forming the higher next isotope in an excited state of energy. The newly formed higher isotope will de-excite by emitting gamma rays. Fission reaction is when a nucleus splits into two fission fragments as a result of absorbing a neutron with the release of a couple of neutrons. Other reactions, (n,x) is when the interaction of the nucleus with the neutron results in the emission of x charged particle. The possible particle x includes alpha, neutron, proton, deuteron, etc. If the emitted particle is an alpha, the reaction would be written as (n, α). The dependency of the interaction of neutron with matter on the neutron's energy has led to the classification of the neutrons in terms of their energies. These are given in Table 3-2 below.

Table 3-2: Neutrons according to their energy

Class	Energy
Thermal neutrons	~ 0.025 eV
Epithermal neutrons	~ 0.1 – 100 keV
Fast neutron	~100keV – 100 MeV
High energy neutrons	E > 100 MeV

The main summary that can be deduced from the discussion above is that radiation interacts with matter on the basis of seeing matter at its basic constituents, which are nuclei and electrons. The structure of the matter at its atomic level can be altered and in many cases, it is not a desired alteration. The effect of radiation on matter is dependent on the type of radiation, energy, intensity and the material (medium) which the radiation interacts with. The biological effects on living things exposed to these radiations can be either direct or indirect action. Direct action refers to the irreversible biological damage of living tissue as a result of interaction with highly ionizing particles [42].

Indirect action begins with ionization of simpler molecules which creates chemically active free radicals. A free radical is an atom, ion or a molecule with no paired valence electron in the outer shell [83]. These free radicals are chemically reactive and diffuse longer distances to reach atoms of living tissue in the body. They then induce chemical changes in biological tissue. This indirect action of radiation on living tissue dominates the total disruption arising from radiation. The results are the changes in biologically complex systems such as chromosomes or genetic mutation [42].

There are a couple of concepts that have been developed to characterize the damage of radiation on matter. These include absorbed dose, equivalent dose and effective dose. Absorbed dose (D) is the absorbed radiation energy per unit mass and is measured in Gray (Gy), where 1 Gray equals 1 Joule per kilogram. In radiation protection, the average absorbed dose (D_T) for a tissue or organ is preferred as it gives a better characterization. This would be given by the following formula:

$$D_T = \frac{\varepsilon_T}{m_T}$$

where ε_T is the total energy deposited in a tissue or an organ and m_T is the mass of the tissue or the organ. Certain molecules such as DNA in organisms have repair mechanisms that occur when they have not been too damaged. This makes the biological effects of radiation to be dependent on the rate of dose absorbed in the tissue. Thus, a dose that could damage 100% of a given population of cells can be applied in fragments in a slow rate to reduce the number of cells damaged. Another concept used to characterize radiation damage in tissue is the relative biological effectiveness (RBE) which refers to the biological response to a given radiation dose relative to that induced by 250-keV X-rays or gamma rays [42]. The RBE is radiation energy dependent and therefore complicated to work with in practice which has led to the establishment of radiation weighting factor, w_R. This is given in Table 3-3 which is adapted from [84] and [42]. The weighting factor is obtained by averaging the RBE over a range of energies for particular radiation [42].

Equivalent dose (H) is a quantity used to indicate the biological effects of radiation exposure [42] [84]. The equation used to calculate the equivalent dose is given below:

$$H_T = w_R \times D_{T,R}$$

The subscript T represent tissue, $D_{T,R}$ is the average absorbed dose in tissue T for a given specific radiation R. The equivalent dose is measured in Sievert (Sv). If the issue is exposed to multiple types of radiation, the equivalent dose is given by a weighted sum of all radiation types contributions. Different parts of the body have different sensitivity to radiation and have therefore been given tissue weighting factor, w_T. This weighting factor is used to calculate an effective dose, E, which specifies the sum of the equivalent doses to different tissues in the body. The effective dose is given by the following formula:

$$E = \sum_T w_T H_T \text{ (Sv)}$$

Table 3-3: Radiation weighting factors adapted from [84] and [42]

Radiation type	Energy (MeV)	Radiation weighting factor
Photons	All energies	1
Electrons and muons	All energies	1
Protons and charged pions		2
Alpha particles, fission fragments, heavy ions		20
Neutrons	< 0.01	5
	0.01 – 0.1	10
	0.1 – 2	20
	2 - 20	10
	>20	5

Some of the weighting factors for specific body organs adapted from [84] are given in Table 3-4 below.

Table 3-4: Weighting factor for some of the body organs adapted from [84].

Organ	Weighting factor, w_T
Bladder	0.04
Skin	0.01
Gonads	0.08
Liver	0.04

The goal of radiation containment in dry casks is therefore to ensure that all radiation in the SFAs is retained within the casks. The governing formula is given as follows:

$$I(x) = I_0 \, e^{-\mu x}$$

Where:

$$I(x) - \ quantity\ of\ radiation\ after\ thickness\ x$$

$$I_0 - \ initial\ \ radiation\ quantity$$

$$\mu - linear\ attenuation\ coefficient\ of\ the\ mateial$$

$$x - is\ the\ thickness\ of\ the\ material$$

Table 3-5 below shows some of the commonly used materials to achieve gamma radiation shielding in the casks. The desired material must have the highest linear attenuation coefficient and provide a practical way to be incorporated in the cask design. The material must also be able to withstand other phenomena such as temperature fluctuation and pressure.

Table 3-5: Linear attenuation coefficient of different materials at different gamma energies [80]

Absorber	**At 100keV** (cm^{-1})	**At 200keV** (cm^{-1})	**At 500keV** (cm^{-1})
Lead	59.7	10.15	1.64
Air	0.000195	0.000159	0.000112
Water	0.167	0.136	0.097
Aluminum	0.435	0.324	0.227
Iron	2.72	1.09	0.655
Carbon	0.335	0.274	0.196

The desirable dose rates for the casks are those whose effects on human and the environment are minimal. The principle of ALARA (As low as reasonably achievable) is used in nuclear facilities to minimize radiation doses and limit the release of radiation into the environment. The maximum dose rates have been calculated and recommended by international organizations such as the International Commission on Radiological Protection (ICRP). These are given in the table below and the current achievable dose rate limits for some of the casks are tabulated. The units used for the dose limits as seem in Table 3-6 are milliSieverts (mSv) and millirem (mrem). The former (Sievert) is the SI unit for effective dose and is related to roentgen equivalent man (rem) by the following equation: 1 Sievert = 100 rem. These effective dose is the measure of the amount of radiation absorbed and its biological effects [87] [88].

Table 3-6: The ICRP recommended dose rates vs achievable rates in some of the casks in use [84]

ICRP 60 recommended dose limit	Occupational effective dose (mSv)	Public (mSv)	TN-24P (mrem)	VSC-24 (mrem)	HI-STAR 100 (mrem)
Whole body	20	1	35 mrem/hr	20 mrem per hour 50 on the top	191.26

3.2. Spent fuel

During the feasibility study of interim dry storage, the spent fuel is not only a vital aspect to be analyzed in detail but one of the first because it determines a lot of other decisions. There are two main aspects of spent fuel, namely, the quantity and characteristics of the spent fuel. The spent fuel quantity determines the size of the required casks and the facility. This quantity can either be the number of the spent nuclear fuel assembly expected at the end of the power station's lifespan or a fraction of those spent nuclear fuel assemblies in a case of other plans such as reprocessing at a later stage. The physical properties such as the dimensions of the SFAs and shapes are also part information analyzed in the design of the casks.

In [71], some of the most important characteristics of spent nuclear fuel to be considered include the type of fuel, burnup, required minimum cooling time in the SFP, condition of the fuel cladding, radionuclides inventory and activity. An analysis of the spent fuel characteristics also includes damaged fuel which may require special designs for the casks. This information allows for the prediction of long term spent fuel behaviour which is one of the critical inputs in the design of casks for optimum performance. The radionuclide inventory depends on the fuel type, fuel cycle and neutron flux. The inventory will, in turn, determine the activity of the spent fuel.

3.3. Sub-criticality

One of the key aspects of safe spent fuel management is ensuring sub-criticality of the spent fuel. Criticality in nuclear reactors refers to a condition where the number of neutrons produced by fission in one generation equals the number produced in the preceding generation. This is the condition required for a sustained nuclear fission chain reaction. This condition corresponds to a value of k_{eff}, the effective multiplication factor equals to 1 and reactivity, ρ equals to 0. Mathematically, the effective multiplication factor can be defined as follows:

$$k_{eff} = \frac{neutrons\ produced\ from\ fission\ in\ one\ generation}{Neutron\ absorption\ in\ the\ previous\ generation + neutron\ leakage\ in\ previous\ generation}$$

If the neutrons produced are greater than the neutrons absorbed and leaked, the reactor is said to be supercritical which corresponds to an effective multiplication factor of greater than one. This will result in a neutron flux increase in every generation. If the neutrons produced are less than the absorption and leakage, the value of k_{eff} will be less than one. The neutron flux will decrease in each generation and, the reactor will be said to be sub-critical. The effective multiplication factor is therefore expressed as follows:

$k_{eff} = \varepsilon.p.f.\eta.l_f.l_t$, where

ε = fast fission factor = $\frac{number\ of\ fast\ neutrons\ produced\ by\ all\ fissions}{number\ of\ fast\ neutrons\ produced\ by\ thermal\ fissions}$

$$p = resonance\ escape\ probability = \frac{number\ of\ neutrons\ that\ reach\ thermal\ energy}{number\ of\ fast\ neutrons\ that\ start\ to\ slow\ down}$$

f = thermal utilization factor = $\frac{number\ of\ thermal\ neutrons\ absorbed\ in\ the\ fuel}{number\ of\ thermal\ neutrons\ absorbed\ in\ all\ reactor\ materials}$

$$\eta = \text{reproduction factor} = \frac{number\ of\ fast\ neutrons\ produced\ in\ thermal\ fission}{number\ of\ thermal\ neutrons\ absorbed\ in\ the\ fuel}$$

$$l_f = \text{fast non-leakage probability} = \frac{number\ of\ fast\ neutrons\ that\ do\ not\ leak\ from\ reactor}{number\ of\ fast\ neutrons\ produced\ by\ all\ fissions}$$

$$l_t = thermal\ non-leakage\ probability$$

$$= \frac{number\ of\ thermal\ neutrons\ that\ do\ not\ leak\ from\ reactor}{number\ of\ neutrons\ that\ reach\ thermal\ energies}$$

The desired state for the spent fuel is sub-criticality, which will ensure that no further nuclear fission takes place. The spent fuel contains approximately between 1 and 1.5% of fissile material in the form of U-235 and U-239 [82]. This makes it a necessity to incorporate sub-criticality conditions in the cask design. This is achieved by either removing the neutrons in the spent fuel or creating a condition for them not to cause fission. The "removal" of neutrons is done by using neutron-absorbing materials when designing the casks. Examples of neutron-absorbing materials include boron, gadolinium, hafnium and cadmium. These are materials with high neutron absorption cross-section for thermal neutrons. While these have some of the highest thermal neutron absorption cross-sections, they can also increase moderation of fast neutrons to thermal neutrons. This can result in absorption of these thermal neutrons in the remaining fissile material of the waste, thereby initiating another nuclear chain reaction. The design is therefore critical to ensure that the materials perform the function of absorption effectively and dominate any possible moderation that can initiate a chain reaction.

Table 3-7: Typical materials used for thermal neutron absorption [85].

Material	**Thermal neutron absorption cross-section (barns)**	**Possible forms in which they can be used in cask designs**	**Issues**
Gadolinium-157	259 000	Aqueous	Poor corrosion resistance
Boron-10	3800	Boral steel, borated concrete, borated polythene	Helium buildup thereby increasing pressure in cask
Chlorine-37	100	Polyvinyl Chlorine (PVC)	Degrades quicker
Hafnium-177	7200	Hafnium boride, alloys of hafnium and zirconium	
Cadmium-113	62 000		High cost

Most parts of the internals of the casks are therefore made from neutron-absorbing materials in order to ensure sub-criticality. These neutron-absorbing materials are also called neutron poisons. The other solution in achieving sub-criticality is spacing the spent fuel assemblies sufficiently far apart from each other. This is to increase the distance a neutron would have to travel before it interacts with a fissile uranium atom. The further apart the fuel assemblies are, the better the chances of sub-criticality. However, the further the spent fuel assemblies are apart, the bigger the casks as well as the cost. Others [86] have suggested decreasing the value of the effective multiplication factor by coating the spent fuel rods with neutron absorbing material before loading them into the casks. This would create an additional neutron absorbing capabilities in the casks.

The need to maintain the effective multiplication factor at unity during operation of the reactor and the ability to monitor any change gives a need for expression of reactivity in terms of the effective multiplication factor. The definition of reactivity is therefore given as follows:

$$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

The generally accepted value of the multiplication factor to ensure sub criticality is a value of no greater than 0.95. In [87], the multiplication factor of 0.95 is said to include all uncertainties benchmarking and calculated with 95% probability at 95% confidence level. If the maximum allowed multiplication factor is 0.95, the corresponding value of ρ will be -0.05. The traditional analysis of criticality of spent nuclear fuel assumes that the fissile nuclide is at its highest reactivity as the fresh fuel with the highest enrichment and no irradiation. This has allowed for the over-conservative design of spent fuel pools and casks. Designs of spent fuel pools and casks are now taking into account the irradiation of the spent fuel by applying the concept of burnup credit. This allows for dense spent fuel placement into the casks but requires detail calculations and analysis of the spent fuel to be done. If the burn-up credit approach is employed, detailed criticality analysis must be demonstrated.

3.4. Decay heat removal

The engineering approach on the safety of spent nuclear fuel is based on the concept of defence-in-depth philosophy. This is an application of different barriers to prevent accidents and incidents from occurring or minimize the effects if an accident occurs. As discussed above, the primary containment barrier in spent fuel is the cladding. However, like any material, the optimum performance of the cladding can only be achieved within certain conditions. One of those conditions is an acceptable temperature range. Once the temperature exceeds a certain threshold, the integrity of the fuel cladding is compromised and this can result in the escape of the radionuclides. This, therefore, makes efficient decay heat removal from the SFAs one of the most important factors in dry storage facilities.

It has been highlighted that the spent nuclear fuel comprises of 3 types of nuclides, namely, actinides, fission products and activation products. The actinides and fission products continue to release energy that is converted to heat as they decay in various phenomena explained in 3.1 above. Jang *et al.* [88] show a typical variation of cladding temperature for various fuel cycles.

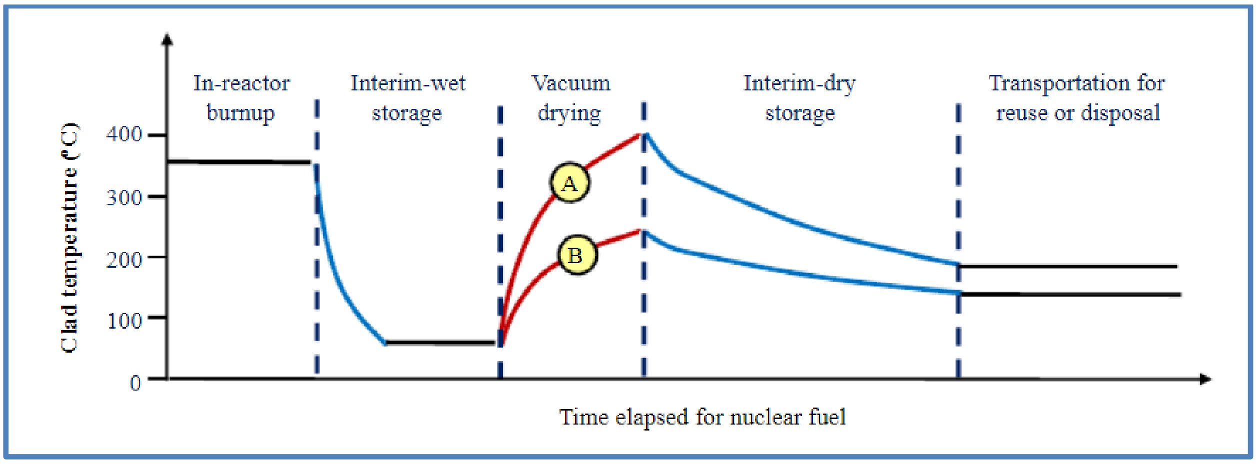


Figure 3-3: Typical variation of cladding temperature at different fuel life cycle [88]

Although no indication of specific time frame has been made in Figure 3-3 above, typical time frames can be allocated to these fuel cycles. A typical nuclear power station time profile for this figure will, therefore, be as follows:

Table 3-8: Typical time frame for fuel life cycles at a nuclear power plant

Fuel life cycle	In-reactor burnup	Interim-wet storage	Vacuum drying	Interim-dry storage	Transportation for reprocessing or disposal
Time (months)	54	120	0.2	120 - 600	1

The in-reactor burnup is certain with exceptions of damaged fuel that gets removed before the end of the full 18-month cycle. Yang [89] points out that continued fuel design improvements have brought the number of damaged fuel rods to less than 0.1%, i.e. on average, 99.99% of the fuel rods remain in the reactor for the planned duration with no damage. This is a significant improvement from a big portion of fuel rods and sometimes the whole core in the 1970s.

The interim dry storage can range from 10 years to 50 years (lifespan of the casks) depending on whether there is a final disposal strategy yet. The time required for the transportation of the spent fuel for reprocessing or to repository facility is also dependent on the destination and transportation mode used. From Figure 3-3 above, the vacuum drying method is critical as it can raise the cladding temperature to values even higher than the temperature while in the reactor. However, the vacuum drying is not discussed in detail in this dissertation mainly because the most important heat to remove is the decay heat, resulting from radioactive decay of fission products.

This decay heat must be evacuated from the spent fuel assembly in dry casks to prevent damaging the fuel cladding and the casks. The dry casks use passive heat removal methods. This is an important aspect of dry cask storage as it eliminates the need for active power supply and mechanical devices. The passive heat removal methods used are conduction, convection and radiation. These are briefly explained below with the aid of Figure 3-4 adapted from [90].

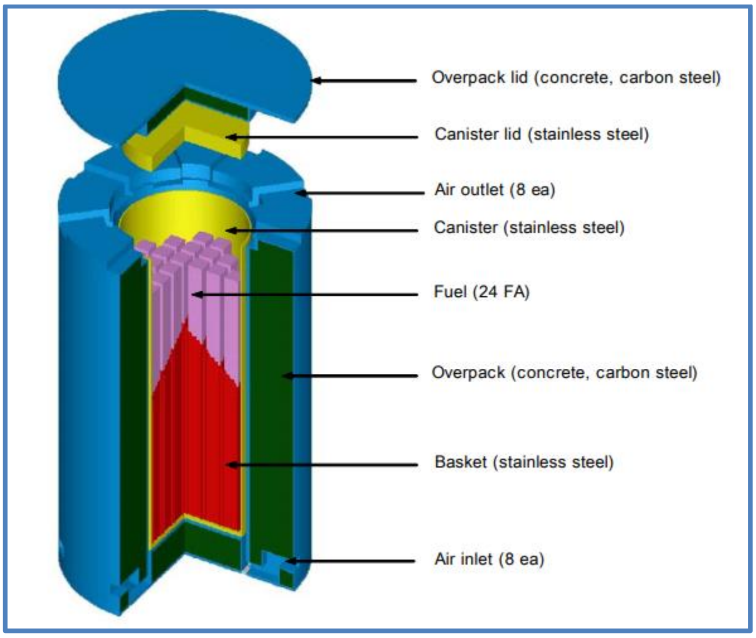


Figure 3-4: Concrete cask overview adapted from [90]

The spent fuel assemblies are in contact with the canister usually made of stainless steel. Heat is therefore conducted out of the cask. Most cask designs have air inlets and outlets as seen in Figure 3-4 that allow for air to flow naturally into the cask, thereby cooling the canister. The air does not flow into the canister but flows into the bottom of the cask and out at the top outlet of the cask. The heat from the cask would then be ejected into the atmosphere. Most of the decay heat is ejected through this natural convection of air flowing through the passage from the bottom inlet to the top outlet. However, some of the heat is ejected via conduction transfer through the outer concrete surface. Casks are designed to evacuate a certain amount of heat which limits the number of spent fuel assemblies that can be loaded in the casks. Computer programs such as computational fluid dynamics and COBRA (Cooling & Boiling in Rod Arrays) are used to design and verify heat transfer and cask performance. The design must cater for normal, abnormal and accident conditions. Causes of abnormal and accident conditions would generally range from air inlets and outlets blockage to fires and earthquakes.

The detailed thermal design and characterization are not discussed here but the basics are briefly presented. Kim et al. [91] highlight some of the key parameters in heat removal and transfer which include temperature rise between the air inlet and outlet, air velocity, air mass flow rate and canister heating power. The following table gives some of the key parameters.

Table 3-9: Some of the key parameters in the thermal performance of dry casks

$\dot{m}$	Overall air mass flow rate (kg/s)
ρ	Density of air ($\frac{kg}{m^3}$)
A	Cross-sectional area of the air outlet (m^2)
C_P	Specific heat of the air ($\frac{J}{kgK}$)
ΔT	Differential air temperature between inlet and outlet (K)
u	Velocity of the air ($\frac{m}{s}$)
h	Natural convective heat transfer coefficient
T_S	Temperature at the surface of the cask
T_A	Ambient temperature
σ	Stefan-Boltzmann constant = 5.67 $\times\ 10^{-8} kg\ s^{-3} K^{-4}$
ε	Emissivity of the exterior surface of the overpack
A_s	Surface area of the overpack

Using the parameters given in Table 3-9, the following equations are used to understand the basics of decay heat removals in casks. These are adapted from [92]. The overall mass flow rate of air in the cask is given as:

$$\dot{m} = \rho u A$$

It follows that the heat, Q_A that is transferred into the atmosphere through the flow of the air via the bottom inlet and out the top outlet is given by:

$$Q_A = \dot{m} C_P \Delta T$$

The heat transferred into the atmosphere from the surface of the cask is given as follows:

$$Q_s = hA(T_S - T_A) + \sigma \varepsilon A_S ({T_S}^4 - {T_A}^4)$$

Decay heat generated by the spent fuel assemblies can be measured and is generally axially non-uniform. This is due to non-uniform burnup in the fuel assemblies [92]. The Wooten-Epstein model, presented in [93], predicts net radial heat transfer from the SFA. The model takes into account the radiative and conduction heat exchange. It is given by the following heat flow equation:

$$Q_{we} = \sigma\, C_0\, F_\varepsilon A_{SFA} \left[{T_C}^4 - {T_B}^4\right] + 13.5740\, L K_{CS} [T_C - T_B]$$

The parameters are defined in the below:

$$C_0 = Assembly\ geometry\ factor$$

$$= \frac{4N}{(N+1)^2}, when\ N\ is\ odd.$$

$$= \frac{4}{N+2}, when\ N\ is\ even.$$

$$N\ is\ the\ number\ of\ rows\ or\ columns\ of\ rods\ in\ a\ square\ array$$

$$A_{SFA} = fuel\ assembly\ heat\ transfer\ area = 4\ \times SFA\ width\ \times\ length$$

$$L = length\ of\ the\ fuel\ assembly$$

$$K_{CS} = spent\ fuel\ assembly\ constituent\ materials\ volume\ fraction\ weighted\ mixture\ conductivity$$

$$T_C = hottest\ fuel\ cladding\ temperature, T_B = fuel\ assembly\ temperature$$

$$Q_{WE} = net\ radial\ heat\ tranfer\ of\ an\ SFA$$

$$\varepsilon_C\ and\ \varepsilon_B\ emissivity\ of\ fuel\ cladding\ and\ fuel\ assembly\ repectively$$

σ *is as defined in Table 3-9.*

3.5. Site location aspects

Although not necessarily a physics and engineering aspect, the site for the cask it is essential. The site must have an adequate footprint for the facility and provide ease of access for whatever means of transport chosen to transport the casks. The decision to place the dry storage facilities within the power plant site reduces legal and regulation approval timelines. This can then give more time for the design phase. The soil type at the location of the facility may cause an increase in construction of the dry cask storage if more civil works are carried out. The knowledge of the site's peak ground acceleration (PGA) is also a major input in designing of the cask. PGA refers to the maximum ground acceleration at a location during an earthquake event. Earthquake-induced ground movements occur in vertical and horizontal directions. In many cases, the horizontal component (peak horizontal ground acceleration) is larger than the vertical component and hence its use in many engineering applications. The use of PGA is discussed further in 3.6. PGA is expressed in g, the gravitational acceleration, where $1g = 9.81\ m/s^2$. The effective peak acceleration (EPA) which refers to the maximum ground acceleration to which buildings corresponds is used in seismic engineering. This value is typically between $\frac{2}{3}$ to $\frac{3}{4}$ of the PGA [94]. In a case where the dry cask storage is installed in the nuclear power station site, these would be known already.

3.6. Response to seismic events

The dry casks are placed on concrete pads designed to support the casks when loaded with the spent fuel. This is shown in Figure 3-5 below. In contrast to other civil structures, dry casks are free-standing and not anchored. They are susceptible to motions of different kinds due to man-made hazards or natural phenomena such as earthquakes which cause ground movements. These motions have been classified into 3 main categories, namely, rocking, sliding and a combination of both rocking and sliding [94]. The factors that influence the response of the casks to ground shaking include the following:

- Radius to centre height ratio of the cask
- The friction between cask and pad
- Characteristics of ground shaking motion such as amplitude, duration, etc

These motions can result in casks collisions against each other if the horizontal displacement is large enough. The damage from these collisions can result in a compromise of the structural integrity of the casks thereby releasing the radioactive material. Dangol [94] shows that sliding motion will occur when the friction force between the pad and the cask is smaller than the horizontal seismic force.

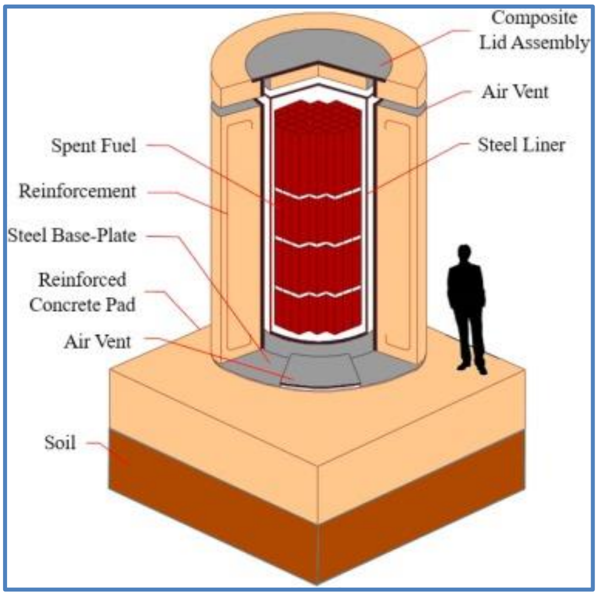


Figure 3-5: Dry cask on a pad adapted from [95]

This can be expressed in the following equation:

$$\mu_{fc}(mg - ma_v) < ma_h$$

Where μ_{fc} is the friction coefficient, a_h and a_v are the horizontal and vertical accelerations respectively, g is the gravitational acceleration and m is the mass of the cask. The equation can further be expressed as follows:

$$\mu_{fc} < \frac{a_h}{g - a_v}$$

Dangol [94] further shows that the governing equation for a cask to fall is given by the following equation: $(mg - ma_v)r < ma_h h_{dh}$, where r is the radius of the cask and h_{dh} is the diameter to height ratio of the cask. This is further simplified as follows:

$$\frac{r}{h_{dh}} < \frac{a_h}{g - a_v}$$

The above equations show that the friction coefficient and height to diameter ratio of the cask are important factors in the casks responding to ground movements. The design must, therefore, ensure that the casks can withstand these vertical and horizontal ground movements. Hartnick [96] also shows that casks with higher diameter to height ratios can withstand higher horizontal acceleration. The casks can also be subjected to ground movements due to explosion-induced shock. In such cases, the vertical ground acceleration component is usually higher than the horizontal component.

3.7. Corrosion of the casks

Spent fuel casks are subjected to the phenomenon of corrosion, which is the interaction between materials and the environment resulting in the degradation of the properties of the material. This interaction results in a formation of oxide or/and hydrated-oxide compounds on the surface of the metal. The casks are generally prone to two types of corrosion mechanisms which are stress corrosion cracking (SCC) and galvanic corrosion where there are different kinds of metals in contact with each other. Their environments make them susceptible to chloride stress corrosion cracking (CSCC) because of the presence of chloride ions [97] in the vicinity of the sea. Other factors required for this phenomenon to take place are the presence of tensile stress and elevated temperatures [97]. In [98], Enos and Bryan show that residual tensile stresses exist near welds on the body of the cask for welded casks. This corrosion can, therefore, result in component failure.

It is therefore essential to have robust monitoring and inspections of the cask continuously to pick up these developments in the casks and initiate remedial actions. The phenomenon of corrosion can be addressed by using materials that are less reactive with oxygen and employing cask manufacturing processes that reduced tensile stress but still deliver quality casks. Aluminium, on the other hand, is very reactive with oxygen thereby encouraging corrosion. However, it reacts faster with the oxygen in the air to form an aluminium oxide which is unreactive. This aluminium oxide forms a thin layer at the surface of the material and thereby preventing cracks and further corrosion. This has been applied at Paks storage facility where the fuel storage tubes (FST) have been sprayed with aluminium coating [99]. Another method used to counteract corrosion is galvanizing. This is a method where a corrosion-susceptible material such as iron is covered with a more reactive metal like zinc. The zinc and oxygen will form a zinc oxide layer thereby preventing corrosion on the covered metal

3.8. Transportation infrastructure

The transportation of the cask from the SFPs to the dry storage facility is also a vital component of safe spent fuel management. It needs a reliable and suitable transportation infrastructure. These infrastructures include a well-engineered route (for ground transport) and suitable trucks or ships if sea transportation is utilized. The planning of the transportation infrastructures should also consider possible transportation from the interim storage to a centralized storage or reprocessing facility in the future. This is important for countries such as South Africa that have not decided on the final disposal strategy yet.

3.9. Operability

The facility needs to provide ease of operation of the facility throughout its life span. This includes continuous monitoring and maintenance, which could also involve personnel to do visual inspection on the casks. This will then require the knowledge of the dosage that personnel can pick up when working in this storage facility. The facility, therefore, should not only provide sufficient physical space for personnel to work around but also ensure the dose limits are not exceeded. A dry storage facility has less maintenance compared to wet storage facilities. This is due to their nature of providing passive cooling as discussed in chapter 2. The tools and equipment used for maintenance activities can, therefore, remain unused for extended period of time. It is important to regularly check their functionality and maintain them as well in order to ensure their reliability when needed to perform. In [100], other operability factors to be considered are highlighted. These include fail-safe systems such as the use of distinctive shapes such that parts cannot be misused, swapped or lost.

The cask selection for countries which have not decided on the final disposal strategy must also incorporate retrievability of spent fuel. This is to keep the option for future reprocessing if economics become favourable. However, this option can compromise the safety of the casks.

3.10. Monitoring

Any action required to avert abnormal conditions or incidents and accidents is preceded by knowledge of the system conditions. Monitoring also forms a backbone of safe spent fuel management. The monitoring includes temperature, pressure and radiation. The parameters monitored must be recorded and analyzed regularly to track any changes. The monitoring information and reports analyzed by the operators should not only be sent to the relevant chief technical staff of the plant, but also to both national and international regulatory bodies. These would include the National Nuclear Regulator (NNR) and international organizations such as IAEA, WANO, etc. The purpose of these submissions would be to prove that the performance of the casks is within the expected and agreed standards. A technical sound plan should be submitted should a deviation from the expected cask performance be seen.

4. Evaluation of existing interim dry storage facilities

4.1. Introduction

This chapter gives a brief evaluation of the existing interim dry storage facilities in operation from 3 countries. The evaluation of these facilities focuses on some of the unique physics and engineering aspects found in these facilities. The facilities presented here are the Connecticut Yankee in the USA, Paks in Hungary and Doel in Belgium. The chapter presents a general overview of the facility, location, technology in use and the characteristics of the spent fuel to be stored in these facilities. All these facilities have been in operation for at least five years, and they use different casks, which diversify the analysis in this chapter.

4.2. Connecticut Yankee spent fuel storage facility

4.2.1. General overview of the facility

The Connecticut Yankee dry cask storage facility was built to store spent nuclear fuel from Connecticut Yankee nuclear power station, formally known as Haddam Neck Nuclear power plant. The Connecticut Yankee nuclear power station is located in Connecticut, about 200 km northeast of New York City in the United States. The nuclear power station operated for 28 years with an installed capacity of 619 MWe. It was shut down in 1996 and decommissioned between 1998 and 2007. The power station is said to have been shut down for economic reasons [101]. The dry storage facility has stored 1019 spent fuel assemblies [102] [103]. The construction was completed in 2002, and the transfer of the fuel from the spent fuel pools was done between May 2004 and March 2005. It is reported to have taken 3 years from procurement of materials to completion (transfer of the spent fuel) of the project [102].

4.2.2. Location and physical dimensions

The facility is located in the power station property, which is about 500 m from the decommissioned reactor site. It occupies an area of 21x70 m^2. The facility has 43 casks of which 3 have been used to store reactor vessels internals from the decommissioned PWR reactor. Figure 4-1 and Figure 4-2 below, both adapted from [104], show the top view and Connecticut Yankee dry storage facility.

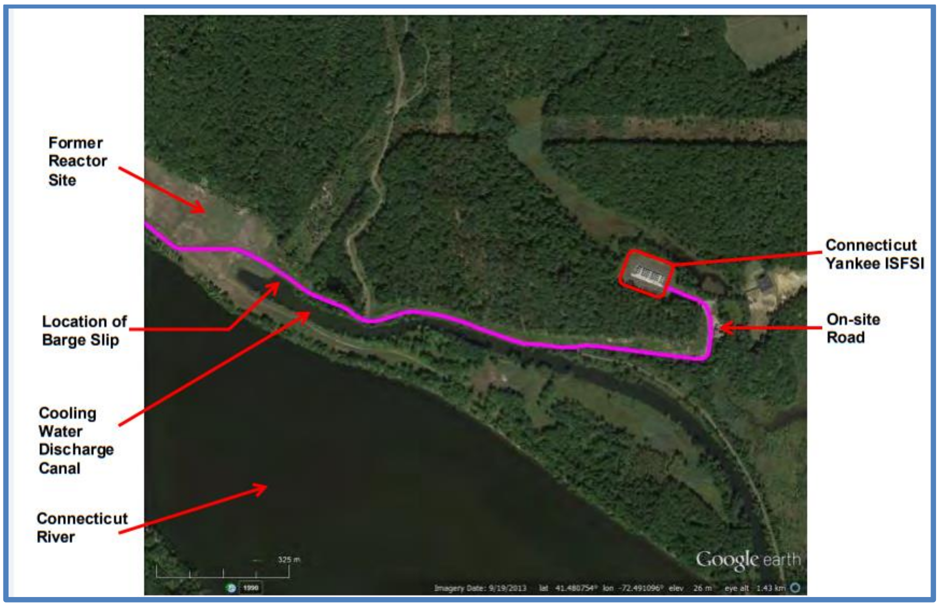


Figure 4-1: The top view of ISFSI at Connecticut Yankee adapted from [104].

Figure 4-2: Aerial view of the Connecticut Yankee ISFSI at Connecticut Yankee adapted from [104]

4.2.3. Technology

The focus on this section is only given to the casks used for spent fuel from Connecticut Yankee unless specified otherwise. The Connecticut Yankee ISFSI has used multipurpose casks from NAC international, called the NAC MPC which were designed specifically for spent fuel from Connecticut

Yankee and hence preceded by “CY’. The NAC multipurpose canister (MPC) system has 3 major components, namely [105]:

- CY-MPC Transportable Storage Canister (TSC), which houses either 24 or 26 spent fuel assemblies [106].
- CY-MPC Vertical Concrete Cask (VCC) serves as storage for the TSC. It also provides structural support, radiation shielding and convection cooling of the spent fuel [106].
- CY-MPC Transfer Cask (TRF) is used to move the loaded TSC into and out of the VCC. It also provides radiation shielding while the transfer of the TSC from the SFP to the VCC takes place [106].

Below is a diagram adapted from [106], which shows the 3 main components of the NAC-MPC system.

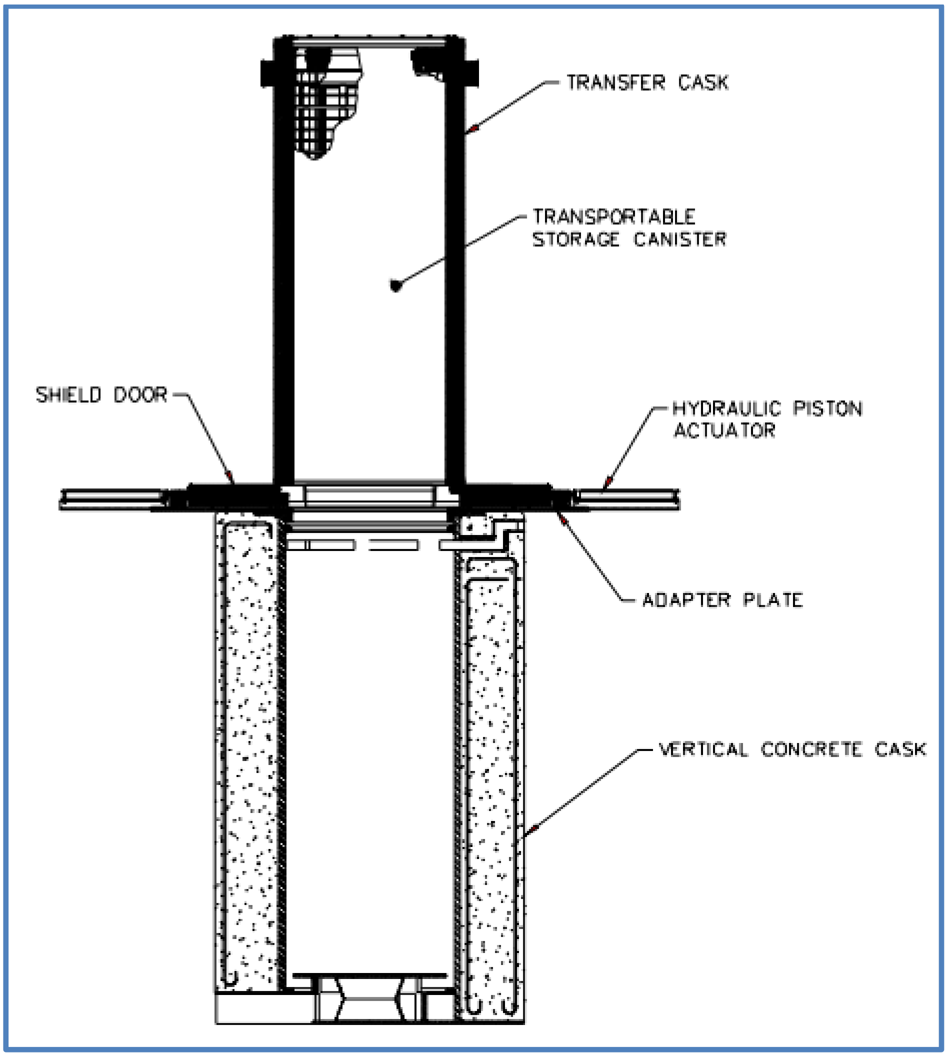


Figure 4-3: Main components of the NAC multipurpose canister system adapted from [106].

The VCC is made from reinforced concrete with a steel liner inside. The concrete and steel liner performs the functions of neutron and gamma radiation shielding. In addition to the provision of radiation shielding, the concrete wall also provides structural strength for the protection of the TSC against natural phenomena. The transfer cask is made up of a variety of materials including steel, lead and NS-4-FR neutron shield making it a multiwall component. The combination of these materials is aimed at limiting radiation dose to less than 300 mrem/hr [106]. These casks have a maximum allowable burnup of 45 GWd/tHM and 60 GWd/tHM for PWR spent fuel assemblies with a required minimum cooling time of 5 years in the spent fuel pool [104].

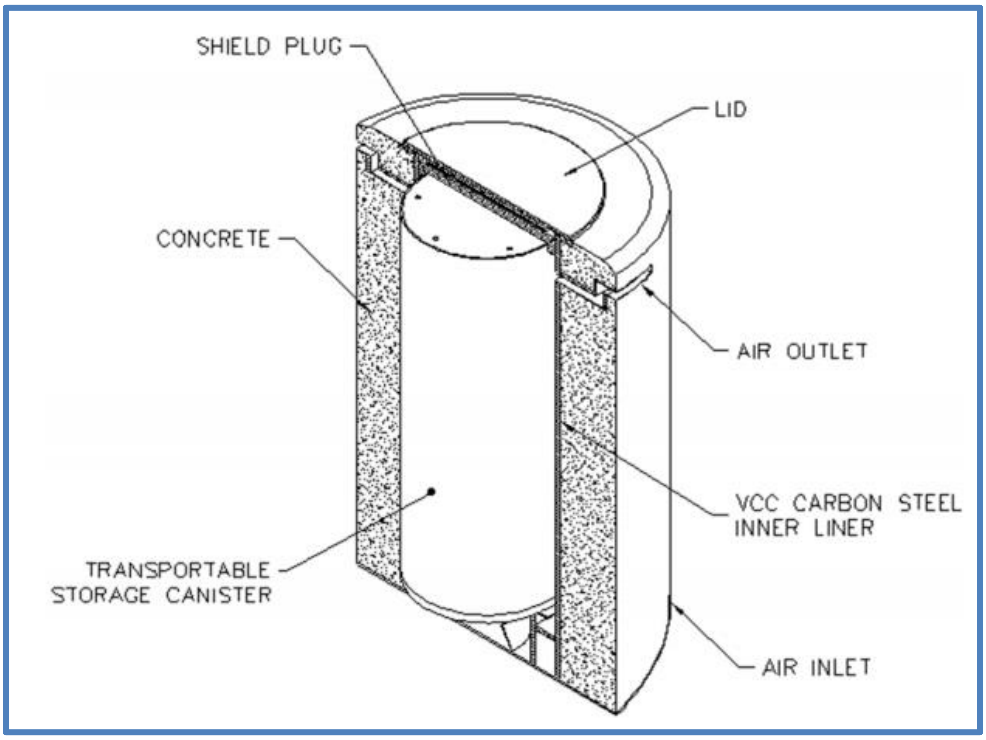


Figure 4-4: CY-MPC storage system adapted from [105]

The casks have been placed on a concrete pad with a thickness of more than half a meter [102]. Some of the physical properties are shown in the tables presented below.

Table 4-1: Physical parameters of the NAC multi-purpose cask system

NAC MPC major component	**Outside diameter (m)**	**Length (m)**	**Capacity (no. of fuel assemblies)**	**Weight (tons)**
Transportable Storage canister	1.79	3.85	26	32

Table 4-2: Physical parameters of the transport cask used at Connecticut Yankee

Transport cask	Gross weight (tons)	Height (m)	Outside Diameter (m)	Concrete Thickness (m)	Steel thickness (m)	Lifespan (years)	Sidewall dose rate (mrem/hr)
Vertical Concrete Cask (VCC)	93	4.84	3.25	0.5	0.09	50	170

Table 4-3: Physical parameters of the transfer cask used at Connecticut Yankee

NAC MPC major component	Outside diameter (m)	Inside diameter (m)	Height (m)	Nominal empty weight (tons)	Sidewall dose rate
Transfer cask	2.26	1.82	4.1	53.5	200 mrem/hr

4.2.4. Spent nuclear fuel at Connecticut Yankee ISFSI

The first batch of fuel to be discharged from the PWR was 23 fuel assemblies in 1971 and the last to be discharged was 157 fuel assemblies in 1996. The 157 fuel assemblies in the last discharge have been highlighted as the typical total number of fuel assemblies in a PWR. These reactor vessel internal wastes is what the US NRC classifies as Greater-Than-Class C Low-Level radioactive waste (GTCC). GTCC waste from nuclear power stations include activated metal hardware such as control rods, spent fuel disassembly hardware and filters [109]. The burnup of the fuel assemblies ranges from 38 GWd/tHM to 40GWd/tHM (Gigawatt-days per metric ton of uranium). The spent fuel initial enrichment was 4.03 % [105].

4.3. Dry Storage facility at Paks in Hungary

4.3.1. General overview of the facility

The dry storage facility was established to provide interim storage for the spent nuclear fuel from the only nuclear power plant in Hungary located at the MVM Paks site. The power plant has four VVER-440 type reactors [67]. Reactor units 1, 2, 3 and 4 were commissioned in 1982, 1984, 1986 and 1987, respectively, with an initial operating license of 30 years [110]. Each nuclear reactor initially generated 440 MW of electricity, making the total electrical output of the plant to be 1760MW. The plant has had advancements and adjustments on some of the major equipment, improving the generation output to 500 MW of electricity per reactor (per generator on each reactor). The electricity output of the plant contributes about 50% of the total current electricity generation in Hungary [110] [111] . The operating license has been extended by 20 years for reactor units 1 and 2, thereby giving a new decommissioning date of 2032 and 2034 for each reactor respectively. Reactor units 3 and 4 are also in the process of lifespan extension by 20 years [110].

The initial spent fuel management involved transporting the spent fuel to the then Soviet Union. The decision to establish the dry storage facility was taken as a result of issues related to transportation to the then Soviet Union. The dry storage facility was built in stages with the first phase completed in 1997 and the last one out of the five completed in 2012. The first four phases which were completed between 1997 and 2007 consist of 16 vault modules, each with a capacity to store 450 fuel assemblies. The last phase of the facility, completed in 2012 has 4 modules, each with a capacity to store 527 fuel assemblies [67] [110]. The current storage capacity is 9308 fuel assemblies. Provision was made to make it possible to add up to 33 vault modules which will give a total capacity of 16159 spent fuel assemblies (assuming vaults 21 to 33 also store 527 SFAs) [99]. Figure 4-5 below, adapted from [110] shows the layout, vault modules and years in which they were completed. The planned lifespan of the facility is 50 years [67] [112].

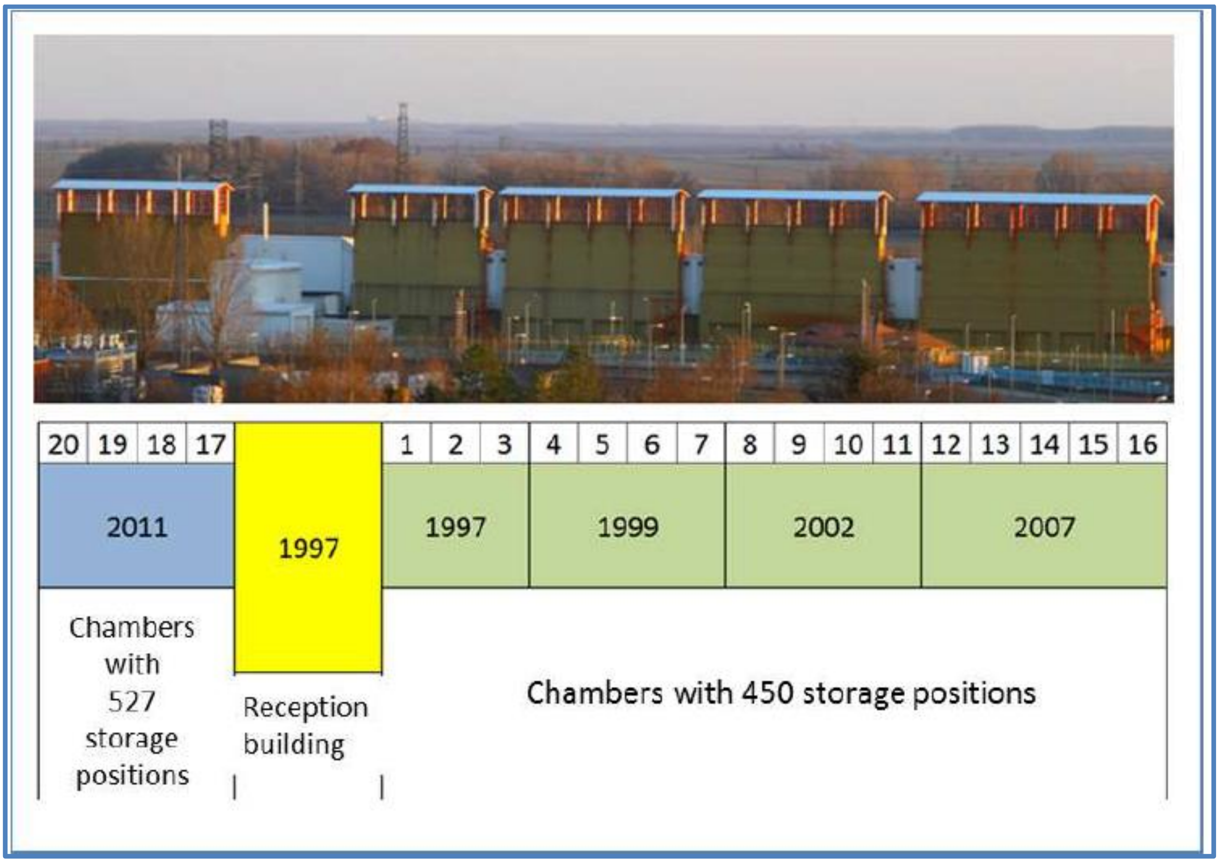


Figure 4-5: Current layout of Paks dry storage facility [110]

4.3.2. Location and physical dimensions

The facility is located adjacent to the nuclear power plant at Paks, which is approximately 127 km south of Budapest, the capital city of Hungary. Below is a map in Figure 4-6 showing the location of Paks where the nuclear power plant and the dry storage facility are located.

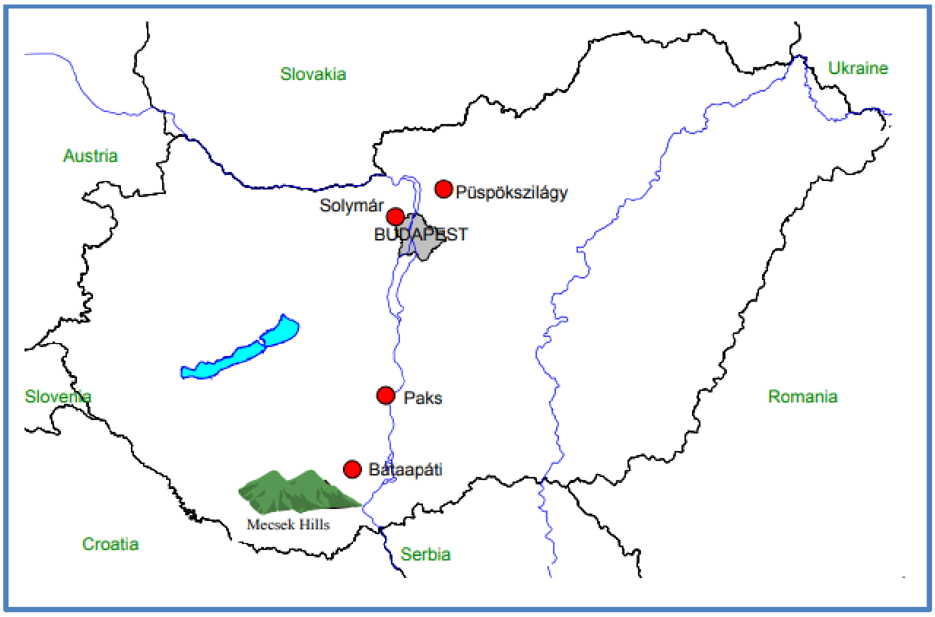


Figure 4-6: Location of Paks nuclear power plant and the dry storage facility, adapted from [113]

Figure 4-7: Paks dry storage facility adapted from [111].

4.3.3. Technology

The Paks dry storage facility uses the modular vault dry storage (MVDS) technology [99]. Three main areas make up the complete dry storage facility, namely transfer cask reception building (TCRB), the charge hall and the storage modules (vaults). The TCRB serves as a fuel reception and dispatch area. The charge hall is an area where transportation from the TCRB to the vaults takes places using a fuel handling machine. The spent fuel is then stored in the vaults [99]. Below is a diagram in Figure 4-8, adapted from [99] showing different areas and components of the MVDS at Paks dry storage facility.

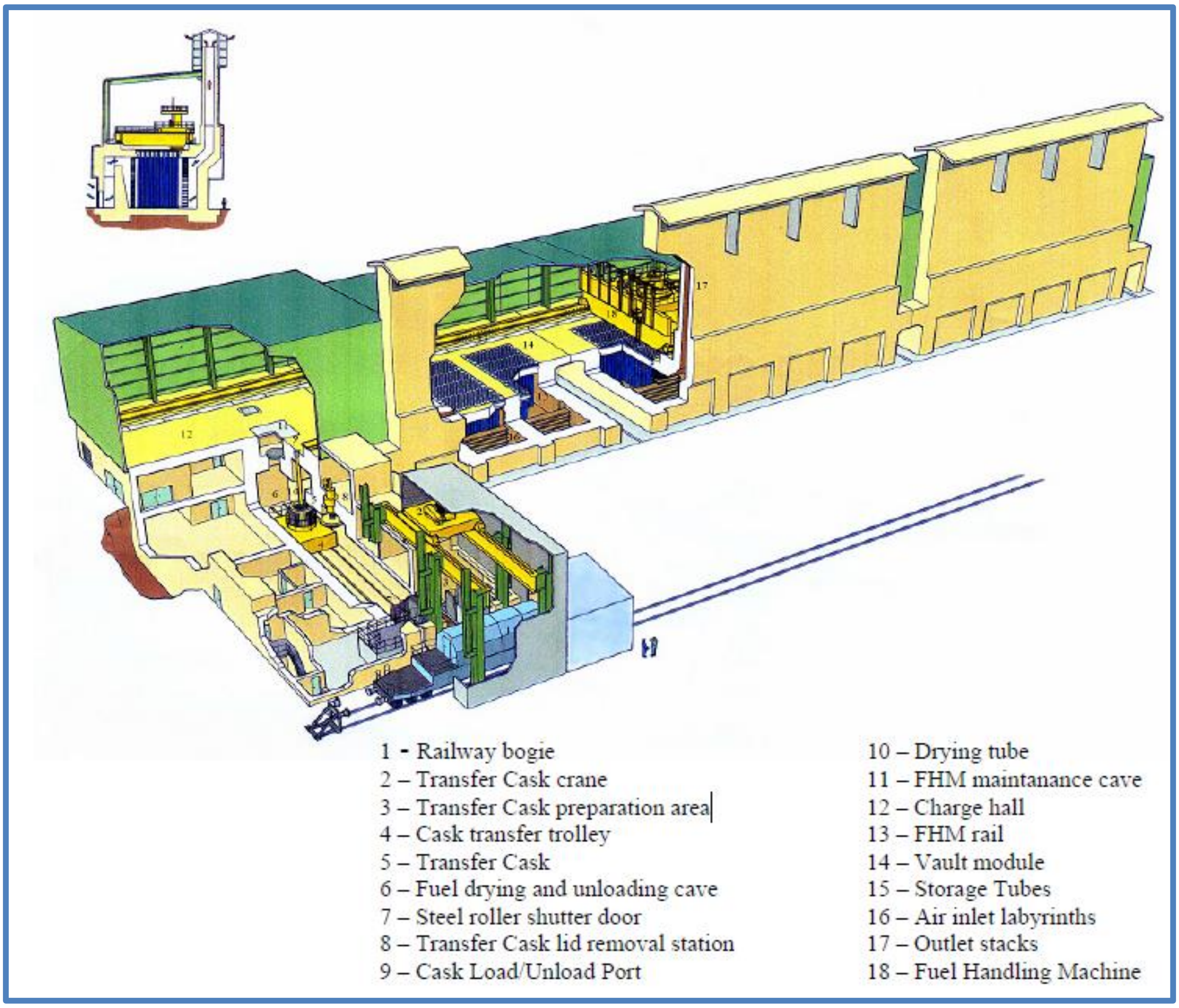


Figure 4-8: Different components of the MVDS at Paks adapted from [99]

In the TCRB, the unloading of the fuel and fuel drying activities takes place in a protected area from the personnel. The area is closed off with a steel roller shutter door [99]. Fuel drying refers to the removal of moisture from the spent fuel assemblies prior to storage in the vaults (in this case) or casks [114]. The fuel handling machine is housed in the charge hall which protects it from weather-driven deterioration. The roof of the charge hall is made from steel. The spent fuel is placed in a vertical sealed fuel tube which is retained in a concrete vault. Each fuel tube holds one spent fuel assembly. It is made from carbon steel [99]. The first 16 vault module stores a total of 450 fuel assemblies each, i.e. 450 fuel tubes and the last 4 stores 527. The removal of decay heat from the vault type of system is achieved by circulating air through the tubes as illustrated in the figure below, adapted from [115].

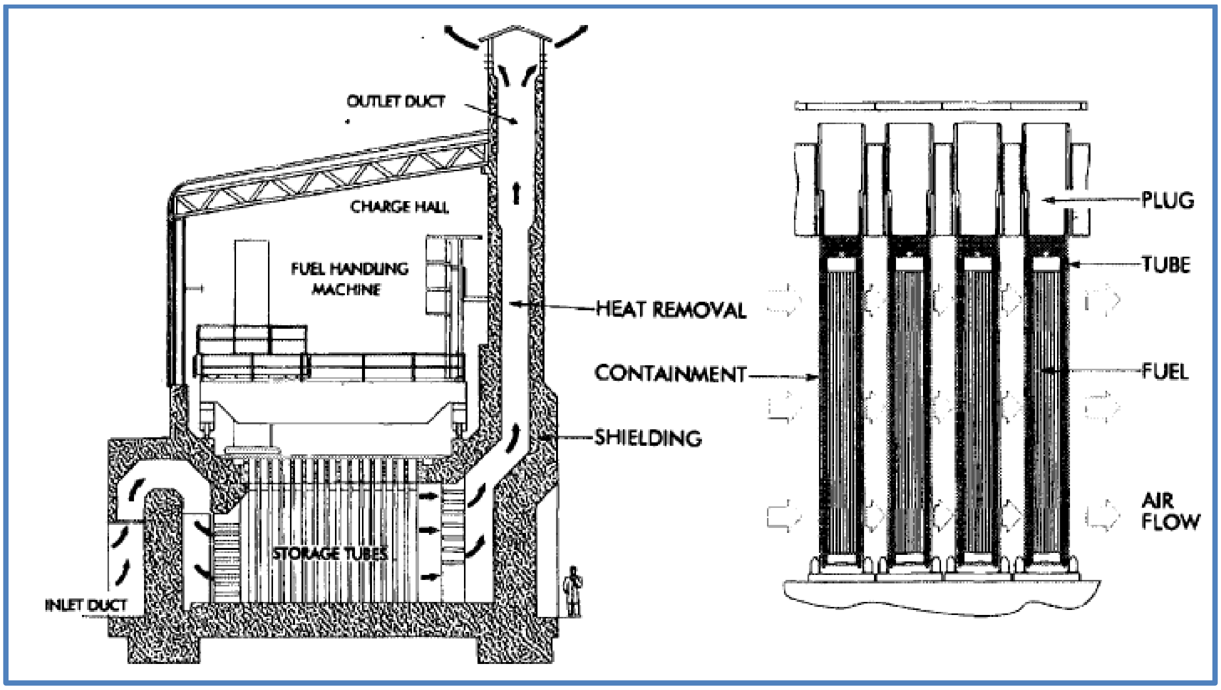


Figure 4-9: Schematic overview of the vault dry storage cooling at Paks [115]

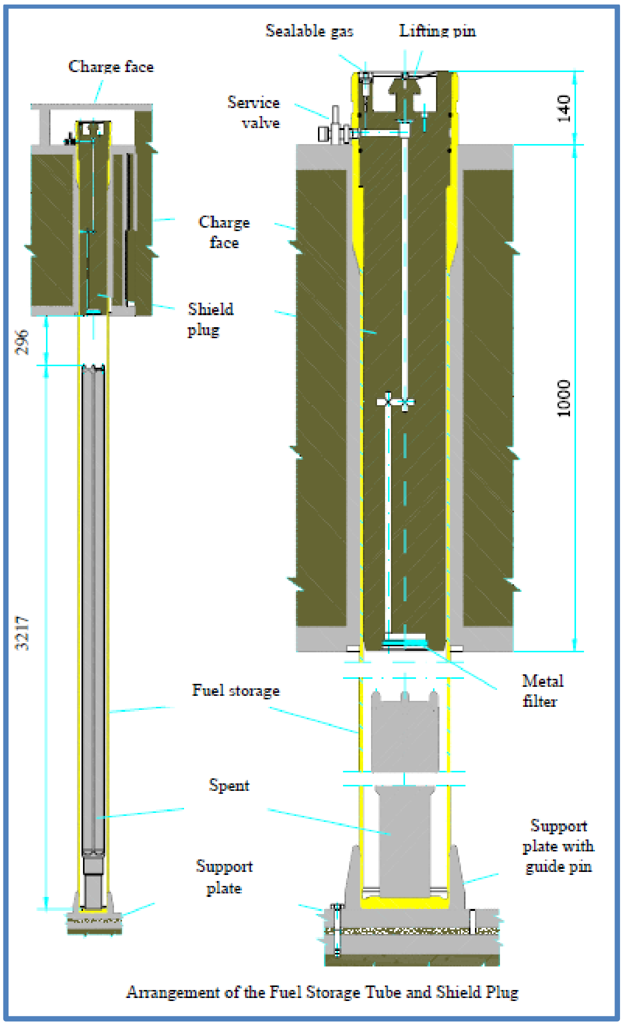


Figure 4-10: The fuel storage and tube and shield plug arrangement adapted from [115]

The SFAs are transported from the SFPs using the C30 casks. The cask was licensed with the characteristics shown in Table 4-4 below. The tubes in the vaults are filled with nitrogen after placing the SFAs. The nitrogen provides a corrosion-free environment inside the vaults.

Table 4-4: Characteristic of the C30 cask at Paks [105]

Maximum number of fuel assemblies	30
Total heat production	< 15kW
Average burnup of the cask load	< 30 GWd/tHM
Initial enrichment	< 3.7 %
Cooling time	Minimum of 3 years

4.3.4. Spent nuclear fuel at Paks nuclear plant

Some of the major physical characteristics of the spent fuel at Paks nuclear power plant are presented in Table 4-5 below. This information is adapted from the IAEA's report on operating experience with nuclear power plants in member states [116].

Table 4-5: Some of the major physical characteristics of spent fuel at Paks nuclear power plant adapted from [116].

Reactor unit	No. of fuel assemblies (control rod assemblies)	Refueling frequency (months)	Average burnup (GWd/tHM)	Power (MWe)	Fuel type	Enrichment [%]	% of FAs replaced
Paks 1	349 (37)	12	37	500	UO2	3.6	30
Paks 2	349 (37)	12	37	500	UO2	3.82	
Paks 3	349 (37)	12	37	500	UO2	3.82	
Paks 4	349 (37)	12	37	500	U02	3.82	

4.4. Doel dry storage facility in Belgium

4.4.1. General overview of the facility

The interim dry storage facility presented here is at Doel nuclear power plant in Belgium. Doel nuclear power station is one of the two nuclear power plants in Belgium and has an output of 2919 MW of electricity from the four nuclear reactors [117]. Reactor units 1 and 2 each have an output of 433 MWe. Unit 3 has an output of 1006 MWe and unit 4 generates 1047 MWe. The first two units (1 and 2) were commissioned in 1974 and 1975 respectively. The last two (3 and 4) were commissioned in 1982 and 1985, respectively [117]. Unit 1 and 2 were initially licensed for 40 years of operation. Their lifespan was extended by 10 years each thereby giving them a new decommissioning date of 2025. Reactor units 3 and 4 were also licensed for 40 years each. It is also worth noting that units 1 and 2 were based on the USA's Westinghouse design, while units 3 and 4 were based on the French's Framatome [117].

The interim dry storage facility was established to provide interim storage of spent nuclear fuel from Doel nuclear power station. The initial capacity of the facility upon its commission in 1995 was 53 casks and was extended in 1998 to accommodate 165 casks capacity [118]. The initial spent fuel management involved reprocessing the spent fuel at France's reprocessing facility, La Hague. It was only in 1993 when the reprocessing arrangement ceased was the dry storage option pursued [119]. The facility was designed for a lifespan of 50 years [118].

4.4.2. Location and physical dimensions

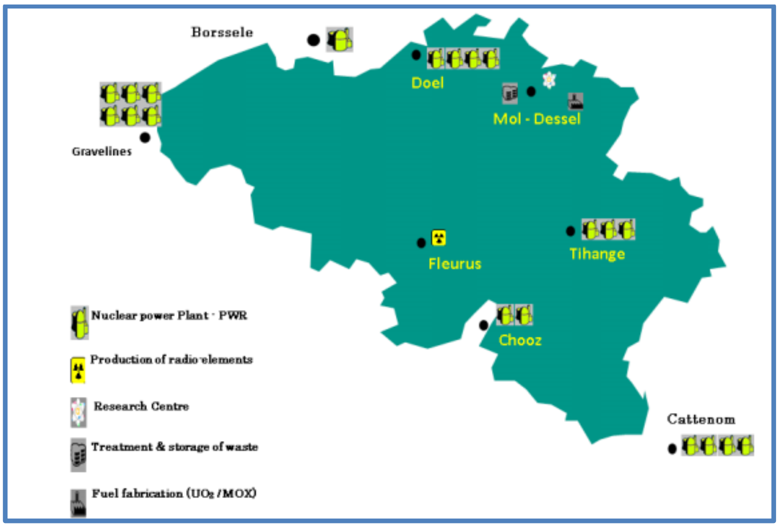


Figure 4-11: Map of Belgium showing the location of Doel nuclear power plant [73]

The storage facility is located on the nuclear power plant site at Doel. Shown above is a map adapted from [73], showing Doel nuclear power plant where the dry storage facility is located. The power plant and the dry storage facility are approximately 80 km north of Belgium's capital, Brussels. It is 3 km from the border of Belgium and Netherlands.

4.4.3. Technology

The dry storage facility utilizes dual-purpose metal casks from Areva Cogema Logistics, formerly known as Transnucleaire. The spent fuel from reactor units 1 and 2 uses TN-24 SH casks [120]. The spent fuel from reactor units 3 and 4 use casks TN-24 D and TN-24 XL, respectively [121]. The use of different casks is a result of different fuel designs for the reactors at Doel power plant. These casks are stored in a concrete building shown below.

Figure 4-12: Cask storage building at Doel dry storage facility [122]

The physical parameters of the cask used at Doel are shown in the table below. These were compiled from [120] and [122].

Table 4-6: Some of the characteristics of the casks used at Doel dry storage facility

Reactor Unit	Cask	Number of maximum fuel assemblies	Cask thermal load (kW)	Maximum allowed enrichment	Minimum required Cooling time (years)	Maximum burnup (GWd/tHM)	Surface dose rate (mrem/hr)
Doel 1 & 2	TN24SH	37	29.9	4.25	5	55	200
Doel 3	TN24DH	28	19.74	4.1	7	55	
Doel 4	TN24XLH	24	33	4.3	7	55	

The casks from the TN family are made from forged carbon steel. The closure system comprises of 2 forged steel lids with two metallic O-ring gaskets. The fuel basket which holds the SFAs is made from borated stainless steel plates which provide sub-criticality in the cask. Gamma shielding is achieved using carbon steel. The neutron shielding function is performed by a solid resin compound between the inner and outer shell. Figure 4-13 below shows the typical layout of the TN 24 cask. Corrosion in the TN casks is prevented by thermally spraying zinc-aluminium coating in the cask cavity surfaces and the outer carbon steel. An additional titanium-aluminium oxide coating is applied in the cask cavity surface to enhance thermal emissivity. The cask is pressurized with helium at the end of the fuel loading which also provides and inert environment that discourages corrosion [123].

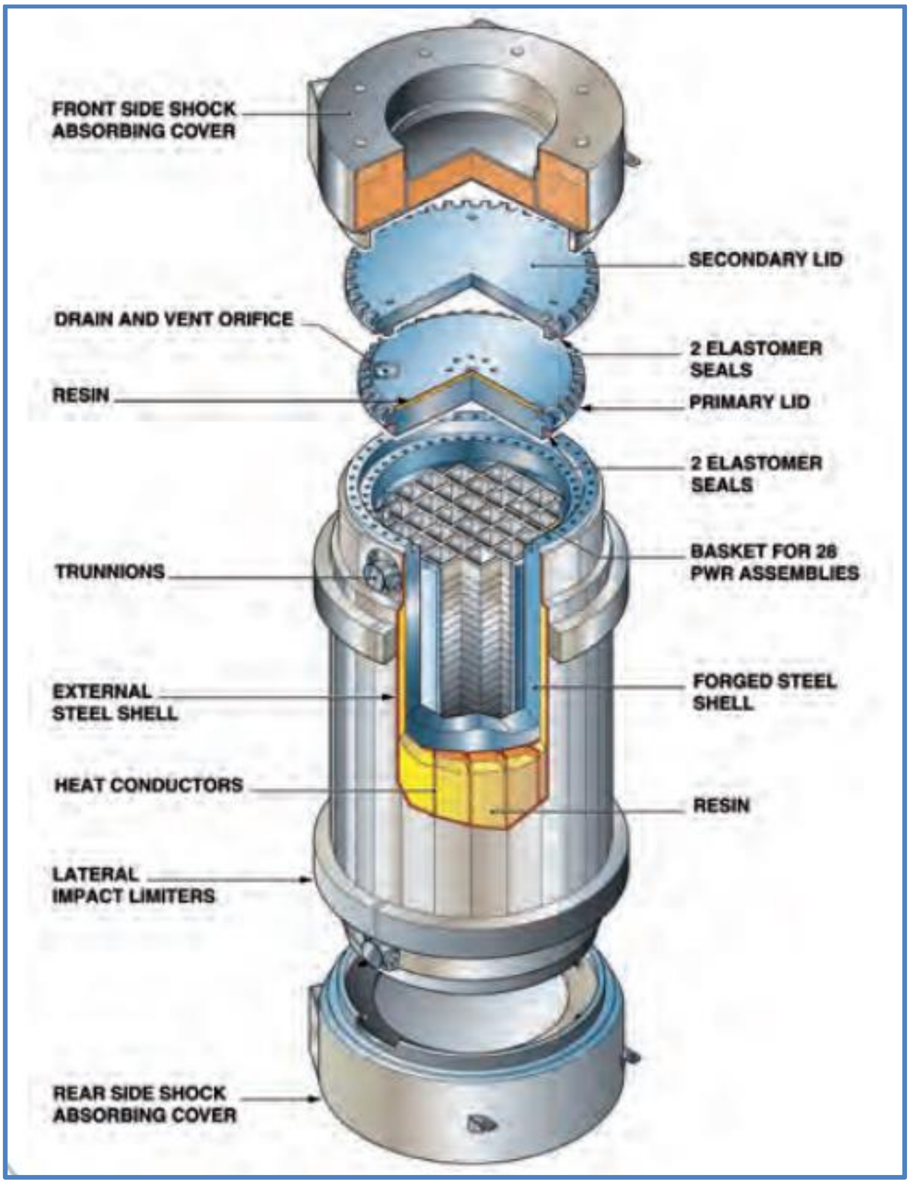


Figure 4-13: TN24 cask [124]

4.4.4. Spent nuclear fuel at Doel dry storage facility

The spent fuel from all the four reactors at Doel nuclear power plant has three types of fuel assembly design. Reactor units 1 and 2 share the same fuel assembly design while units 3 and 4 have different fuel assemblies design. Table 4-7 below gives the physical properties of the fuel assemblies design for the fuel at Doel.

Table 4-7: Fuel assemblies design for the reactors at Doel [121]

Reactor unit	Fuel assembly length (m)	Arrangement
Doel 1 and 2	2.90	4 x 14
Doel 3	4.03	17 x 17
Doel 4	4.79	17 x 17

Table 4-8: Some of the characteristics of the spent fuel from Doel nuclear power plant

Reactor Unit	No. of fuel assemblies	Refueling frequency (months)	Average burnup (GWd/tHM)	Power (MWe)	Fuel
Doel 1	121 (21)	12	45	433	UO_2
Doel 2	121 (21)	12	45	433	UO_2
Doel 3	157 (32)	12	49	1006	UO_2/MOX
Doel 4	157 (32)	12	45	1038	UO_2

4.5. Brief comparison of the dry storage facilities

A summary of some of the key physics and engineering of the three dry storage facilities presented above is given in Table 4-9 below. The first noticeable aspect is that the Paks dry storage has large spent fuel assemblies' capacity requirement even though its total electricity output is almost half that of Doel. This is due to the reactor type, VVER, which has more than double the number of SFAs of that of Doel. The reactor type which determines the spent fuel assemblies' design will, in turn, determine the size of the facility. Paks dry storage has the lowest dose rate of the three storage facilities. The dose rate is what was measured on site

Table 4-9: Summary of the key features of Connecticut Yankee, Paks and Doel dry storage facilities

Physics/engineering aspect	**Connecticut Yankee (619 MWe)**	**Paks (1760 MWe)**	**Doel (2919 MWe)**
Reactor type	PWR	VVER	PWR
Dry storage type	Concrete cask	Vault	Metal cask
Cask/SFAs Orientation	Vertical	Vertical	Vertical
Dry storage capacity	1019 SFAs	9308 SFAs	4400 SFAs
	40 casks	16 vaults	165 casks
Cooling mechanism	Natural convection	Buoyancy-driven ambient air flow	Natural convection
Dose rate limit at the cask surface	170 mrem/hr	26 mrem/hr	200 mrem/hr
Radiation shielding	Concrete	Concrete, carbon steel	Carbon steel and neutron absorbing resin compound
Sub-criticality	Fuel geometry	Fuel geometry	Geometrical arrangement of fuel basket. Hydrogen and borated plates.
Transport from SFP to the dry storage facility	Road	Rail, using C30 casks	Road

5. Analysis of South Africa’s nuclear waste management

Chapter 4 presented an overview of some of the interim dry storage facilities around the world that are currently in operation. This chapter focuses on analysing the proposed interim dry storage facility at South Africa’s only nuclear power plant, Koeberg. The analysis is at a high level because the facility has not been designed yet. There are, therefore, a number of assumptions made to make the analysis complete. The assumptions are stated in the relevant sections of the chapter. The analysis only focuses on some of the physics and engineering aspects of the interim dry storage. The chapter starts by giving a brief overview of the current high-level waste management.

5.1. Koeberg’s current high-level waste management

The two pressurized water reactors at South Africa’s only nuclear power station are of the French design from generation II, model CP-1. The first reactor was commissioned in 1984 and the second one in 1985. The table below shows some of the physical characteristics of reactors.

Table 5-1: Some of the physical parameters of pressurized water reactors at Koeberg

Electrical power output	930MW
Initial licensed lifespan (extended)	40 years (60 years)
Fuel	Uranium dioxide (UO_2)
Enrichment	3.9%
Pellet dimensions	15 mm long, 8mm diameter
Zircaloy tube dimensions	4m long
Active fuel length	3.66 m
Fuel rod (245 pellets inside a zircaloy tube)	4m long, 9.5mm in diameter
Fuel assembly	17 x 17 square lattice =264 fuel elements and 25 for control rods and other
Fuel assembly cross section	21 x 21 cm^2
No. of Fuel assemblies	157
No. of fuel elements in the core	157 x 264 = 41448
Mass approximation of uranium	70 tonnes

At present, South Africa has a nuclear waste disposal facility site for low and intermediate nuclear waste. Vaalputs, situated in the Northern Cape, was built in 1983 and covers about 10 000 hectares of land [125]. The facility is managed by the South African Nuclear Energy Corporation Limited (NECSA). The low and intermediate-level wastes from Koeberg are sealed, marked and stored on-site and transported to Vaalputs using trucks. On average, it is reported that 458 steel drums and 158 concrete drums are shipped to the Vaalputs repository site every year. The intermediate waste is mixed with concrete before it can be sealed into marked concrete drums. This is done to prevent radioactive material from escaping during accidents scenarios.

Koeberg currently stores its high-level waste on-site in the spent fuel pool. Although the volume of high-level waste is small by industry standards, it poses health risks if not handled properly. The initial design and plan was that the spent fuel pools would store the spent fuel for a maximum of 5 years (about 382 fuel assemblies) and thereafter be sent for reprocessing. Reprocessing has not been realised as yet due to economic reasons [36] [126]. Each SFP can hold 1507 fuel assemblies.

There is also a space for 157 fuel assemblies reserved for emergency core off-loading which makes the SFP only to have 1350 spaces [127]. The following table shows some of the physical dimensions of the design of the pools:

Table 5-2: Physical dimensions of the SFP at Koeberg [128]

Length	12.6 m
Width	8.5 m
Depth	12.4 m

An average of 56 fuel assemblies is transferred to the SFP every 18 months. The exact number of fuel assemblies to be replaced in the core is 52 in theory (i.e. a third of 157 fuel assemblies) as per the design and calculated operation. However, during an outage, damaged fuel assemblies that may exist in the core are also replaced. The number of fuel assemblies to be removed also differs with load factor, some fuel assemblies' energy might have been extracted at a higher burnup rate than anticipated and therefore require replacement sooner than planned. Koeberg has accumulated 2173 fuel assemblies (spent fuel) since its operation in 1984 [129] up to 2016. These were generated by the two PWR units in their 32 years of operation, with eight years of operation remaining from the initial 40 year licence period. Of the 2173 fuel assemblies generated, 112 have been transferred to four Castor X/28F casks which are currently housed in the Cask Storage building on site. The spent fuel pool at Koeberg power station for reactor 1 was filled in March 2018, and that for reactor 2 was filled in September 2018 [130] [68] [69]. It is also worth noting that the spent fuel that has been transferred to the casks were 'old' and had low enrichment of less than 3% and burnup of less than 30GWd/tHM.

With the decision to delay the final disposal strategy, a re-racking route was adopted in 1996. Re-racking is making use of high-density racks to store more spent fuel in the SFPs. Re-racking at Koeberg has called for separate regions in the spent fuel pools, region 1 and 2. Region 1 has about 210 rack positions and stores the highly reactive spent fuel which has spent the least amount of time in the reactor. The fuel assemblies are placed further apart to avoid criticality accidents. Region 2 stores less reactive fuel and has more rack positions. The assemblies are closer to each other compared to region 1. Both racks in these regions are constructed using stainless steel with plates of borated steel attached to the outside surface. The borated stainless steel used at Koeberg consists of 1.7% boron [36]. There are currently plans to remove spent fuel from the SFPs to casks that will be initially stored in the cask storage building and later be moved to a planned interim storage facility on site. This process was expected to happen between 2017 and 2025. At the time of this dissertation, the process had been delayed and was expected to begin between the end of 2019 and the beginning of 2020.

5.2. South Africa's proposed dry storage facility

South Africa's proposed spent fuel management for high-level waste will be rolled out in 3 phases [129]. Phase 1 is the procurement of fourteen HI-STAR 100 metal casks from Holtec International and transfer of older spent fuel from the spent fuel pools to these casks. The casks will be stored in the cask storage building located at Koeberg power station. This phase will also require modifications in the cask storage building to accommodate more casks [129] [131]. In addition to the cask storage, there is an application to the National Nuclear Regulator (NNR) to use spent fuel inserts as an interim arrangement [131] [129]. This is expected to make more space in the SFPs. Phase 2 will involve procurement of additional 30-40 metal or concrete casks and the transfer of more spent fuel from the SFPs to these casks. These additional casks will still be stored in the cask storage building. Phase 3 will see an establishment of a transient interim storage facility on site. This will be installed on the existing power station site [129] [132]. The casks at Koeberg will be multi-purpose because they must do both transportation and storage functions. The transportation will be from the cask storage building to the interim dry storage.

Below is an aerial overview of Koeberg power station showing the proposed sites for the interim dry storage facility. These two sites were chosen as viable sites within the power station after applying various criteria from six sites that were initially identified. The proposed facility will be a 12800 m^2 in size and accommodate about 160 casks [133].

Figure 5-1: Proposed sites for the interim dry storage facility at Koeberg power station [129]

The interim dry storage facility (TISF) is expected to provide the storage until a realisation of a centralised Interim Storage Facility (CISF) which will store all radioactive waste in the country [134] [133]. South Africa has not committed to any final spent fuel strategy yet.

5.2.1. Spent fuel quantity at Koeberg

The quantity of the spent fuel generated by Koeberg from 1984 to the end of its 60-year extended lifespan is calculated as follows:

Number of fuel assemblies per reactor=157

A third is replaced every 18 months ~ 53

Number of fuel cycles in a 60-year period=40

Number of SFAs generated in 60 years=40 X 53=2120 SFAs per reactor

Total number of SFAs from the two reactors after 60 years of operation=4240 SFAs

However, provision must be made for cases where more SFAs are replaced due to damage. The proposed dry storage facility at Koeberg will have a capacity of 3840 SFAs (160 casks x 24 SFAs capacity/per cask). If the centralised dry storage facility has not been realized at the end of the 60 years of Koeberg reactors' lifespan or no final disposal strategy decided upon, the SFPs can be expected to continue to store at least 400 SFAs. The pros and cons of designing interim dry storage that will store all the SFAs generated in the 60 years will need to be then studied. The design capacity decision will also be based on possible future decisions that the country can take with regard to final disposal option and other related nuclear power decisions. For example, these two scenarios briefly outlined below can influence the design capacity differently:

(i). South Africa builds no more nuclear power plants

If South Africa decides not to build any more nuclear power plants, the dry storage facility will only be required to store just over 4240 SFAs. The design can therefore, be such that the current capacity will easily be increased to store the remaining 400 SFAs. There may be no need to extend the facility if there will be a final disposal decision taken by 2055. This is because the last 400 SFAs from approximately eight last fuel replacement outages will only be required to be moved out of the spent nuclear fuel pool from 2049 assuming a 10-year cooling period in the spent fuel pool.

(ii). South Africa builds more nuclear power plants according to IRP 2018

According to the IRP 2018, new nuclear power plants would only come in post-2045 [135]. Assuming a cooling time of 10 years in the spent fuel pool, the spent fuel will only move to dry cask from 2055. In this year, the dry storage facility will be 30 years old provided it will have started operations in 2025. The facility would need to be extended then if it is envisioned to accommodate the spent fuel from the new nuclear plants or designed with the capacity to allow for ease of expansion. The other alternative would be to build the dry storage facility in the other proposed sites within Koeberg power station or new sites in the new nuclear power stations.

The continued decline in electricity demand coupled with high pressure to increase renewable energy contribution are diminishing the prospects of nuclear power growth in South Africa, at least by introducing a delay in new build or even a possibility for nuclear power to be not part of the future energy mix. On the other hand, the expected decommissioning of the Eskom's old coal plants in the period between 2023 and 2050 necessitates baseload supply increase. These factors introduce a number of uncertainties at the moment. In this minor dissertation, the opinion is that the dry storage capacity should be guided by scenario 1. This will be taking advantage of the modularity aspect of dry cask storage while also allowing for nuclear waste technology to evolve for possible affordable future reprocessing technologies.

5.3. The HI-STAR 100 Cask

5.3.1. General information

The HI-STAR 100 is a multi-purpose cask designed by Holtec International for storage, transport and disposal of spent nuclear fuel. It is the proposed cask to be used at Koeberg [129]. The HI-STAR 100 name is an acronym from Holtec International Storage, Transport and Repository Cast System [136]. It consists of a multi-purpose canister (MPC) which houses the fuel and an overpack which houses the MPC. The cask also has impact limiters at the top and bottom of the overpack that are attached during transportation. These are used to absorb kinetic energy during normal and accident drop conditions [93]. The diagram of the HI-STAR 100 cask system (MPC with the overpack), adapted from [137] with the MPC partially inserted is shown in Figure 5-2. This diagram is for visual presentation. The MPC has a variety of designs that can accommodate different fuel designs.

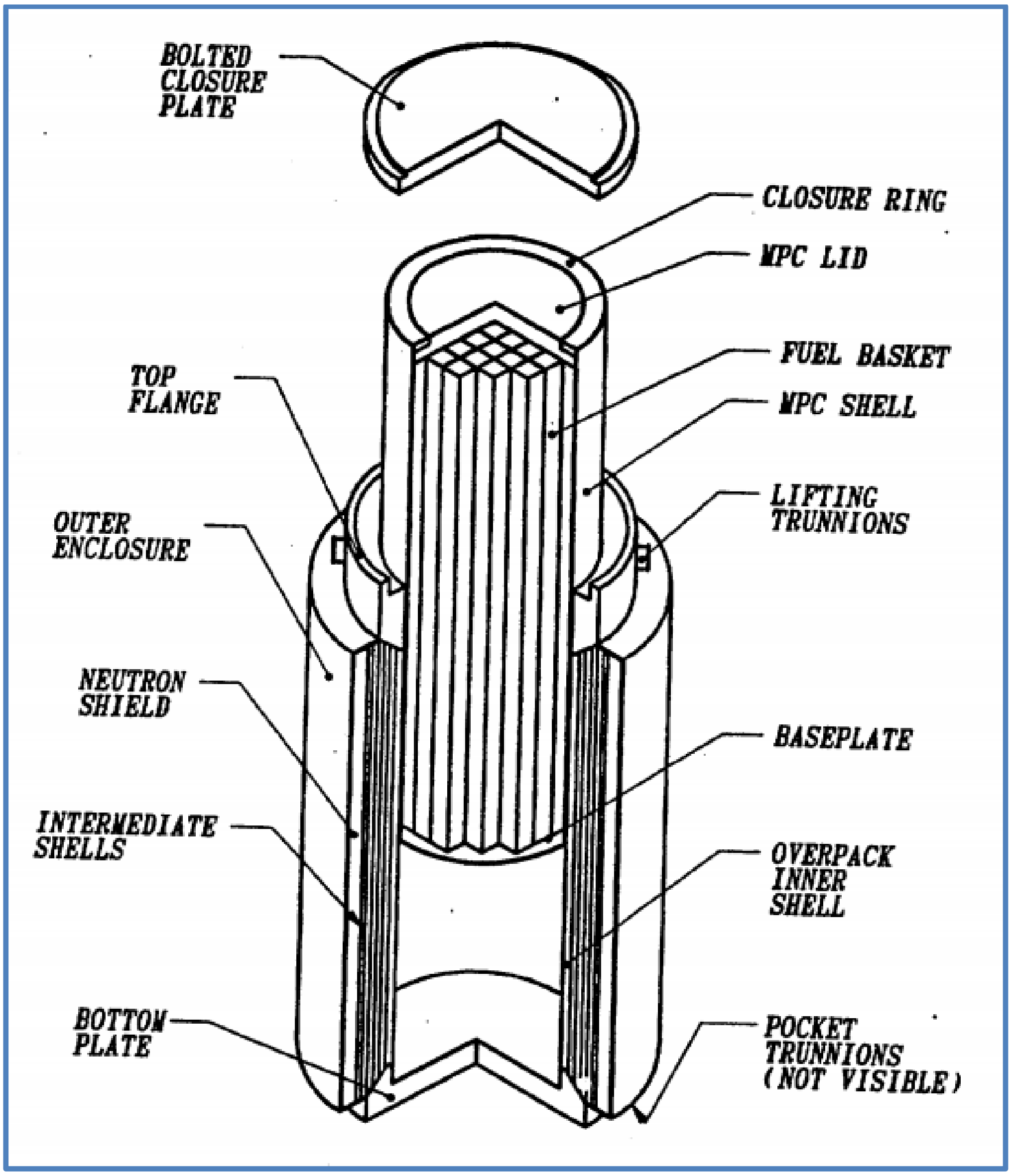


Figure 5-2: The HI-STAR 100 cask system with the MPC partially inserted into the overpack [137]

The cask for 24 PWR SFAs weighs about 37418 kg when fully loaded with spent nuclear fuel. The overpack weighs about 110 545kg, therefore, a fully loaded cask with 24 PWR fuel assembly will weigh a total of 147963 kg.

The impact limiters, attached at the bottom and top of the overpack are also shown in Figure 5-3 below, also adapted from [137]. These are made of corrugated sheets of aluminium alloy. Some of the characteristics of the HI-STAR 100 cask system for PWR SFAs are shown in Table 5-3 below. The cask system was licenced by the US Nuclear Regulatory Commission in 1995 and was one of the first casks to be licensed in the world. The granting of the licence was preceded by a variety of tests ranging from free drop test, corner drop and many more. The system has been designed with a lifespan of 40 years [93].

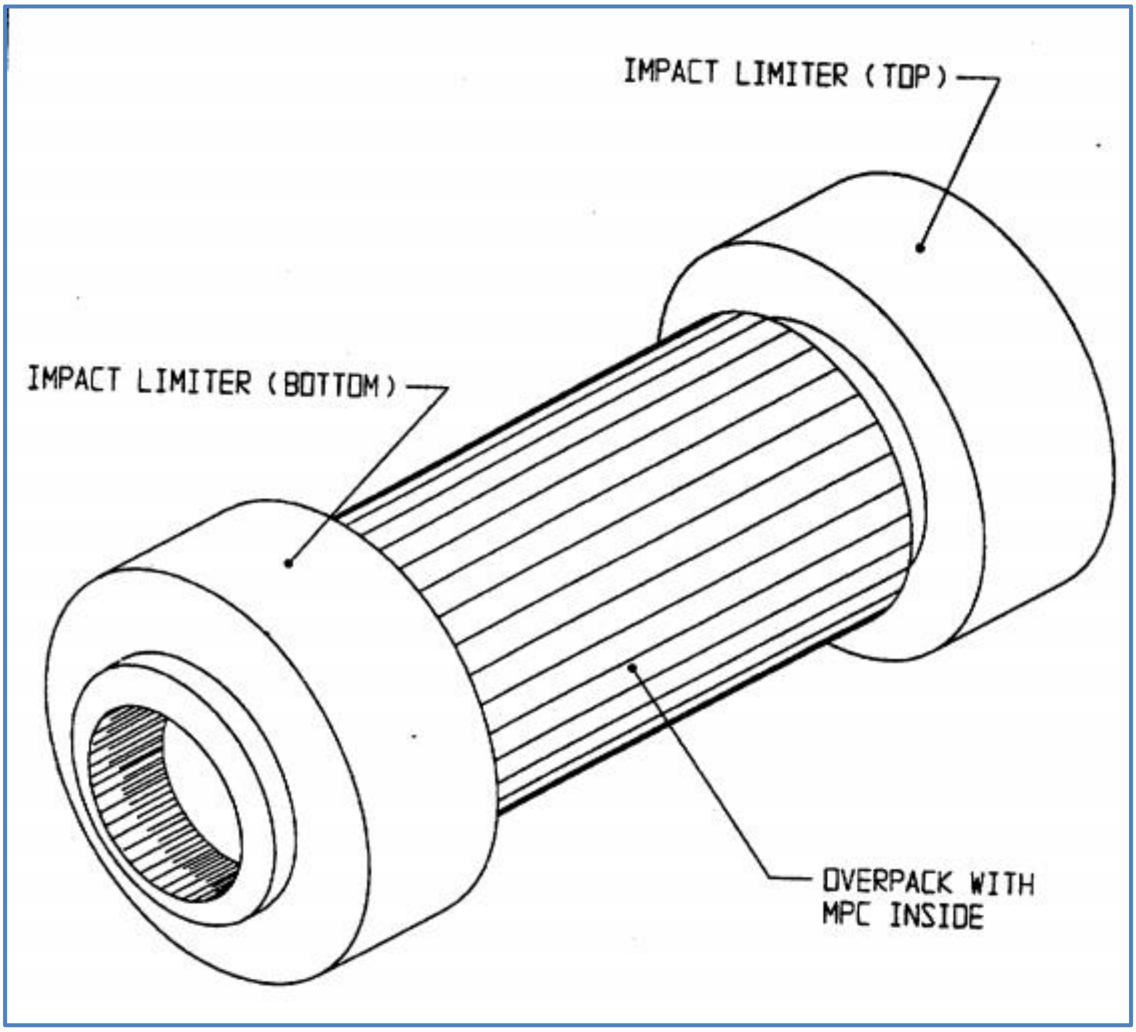


Figure 5-3: The impact limiters on the HI-STAR 100 cask system [137]

The MPC is made of austenitic stainless steel alloy, and the overpack is made of ferritic steel. Austenitic stainless steel is a group of stainless steel containing between 16 and 26% chromium and up to 35% of nickel. It is said to have the highest corrosion resistance [138]. Corrosion resistance is an essential requirement for casks, especially if they will be in an outdoor environment and close to the sea. Typical ferritic steel contains between 10 and 27% of chromium and does not have nickel [138] [97].

Table 5-3: Some of the characteristics of HI-STAR 100 cask proposed to be used at Koeberg

Cask	Orientation	No. of spent fuel assemblies (PWR)	Maximum fuel enrichment	Maximum burnup (GWd/tHM)	Maximum decay heat load	Maximum design weight of canister (kg)
HI-STAR 100	Horizontal during transportation and vertical during storage	24	4.6% (17 X17 fuel assembly)	44	20kW	36287

5.3.2. Spent fuel storage capacity of the HI-STAR 100 cask

The HI-STAR 100 cask that has been proposed to be used at Koeberg's dry storage facility stores 24 PWR spent fuel assemblies with spent fuel characteristics as shown in Table 5-3 above. The maximum number of SFAs to be stored in a 160-cask facility will be 3840.

5.3.3. Sub-criticality of the HI-STAR 100 cask

The sub-criticality of the spent fuel in the HI-STAR 100 cask is achieved using four factors:

- the geometry of the fuel basket designs
- neutron absorbing material in the fuel basket
- limit on the maximum fuel enrichment (4.6% as shown in *Table 5-3*)
- limit on the minimum soluble boron concentration in the water during loading of the fuel.

The first two factors are incorporated into the design of the cask. The maximum calculated effective neutron multiplication factor, k_{eff}, for a HI-STAR 100 cask are given in below. This is a cask with the MPC loading 24 fuel assemblies. Using the neutron multiplication factor, k_{eff}, the corresponding value of reactivity, ρ, is given by the following formula as discussed in 3.3 above:

$$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

Table 5-4: Maximum neutron multiplication factor for HI-STAR 100 cask [93]

Fuel assembly arrangement	Soluble concentration of boron during fuel loading (Parts per million)	Maximum allowable fuel enrichment for U-235 (wt% U-235)	Maximum k_{eff}	Corresponding reactivity value , ρ
17 x 17	No boron	4%	0.9368	-0.0675
17 x 17	400	5%	0.9264	-0.0794

Koeberg's fuel enrichment of 3.9% is below the values in Table 5-4, and if soluble boron of minimum amount of 400 ppm is in the spent fuel pool, the neutron multiplication will be substantially lower than 0.95 thereby ensuring sub-criticality of the spent fuel in the casks. Other MPC designs for the HI-STAR 100 such as MPC-24 allow for storage of damaged fuel as well. The geometrical arrangement shown in Figure 5-4 below is maximized for sub-criticality. The term boral in Figure 5-4 indicates composite material made from boron carbide and aluminium alloy [93].

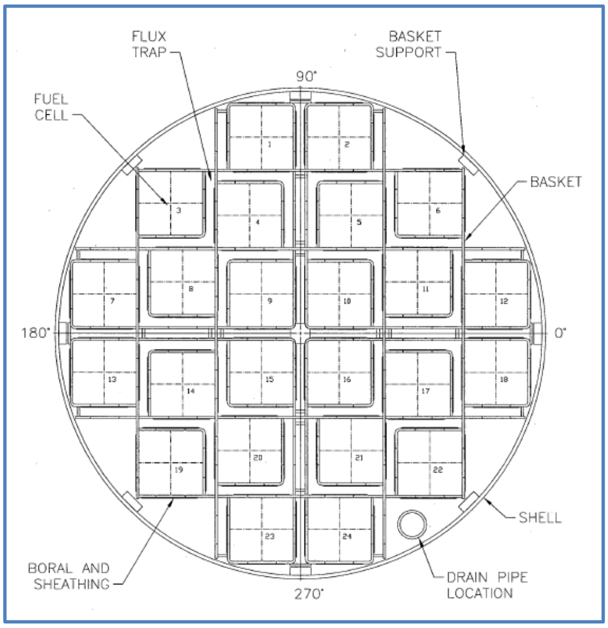


Figure 5-4: Top view of the MPC-24 for the HI-STAR 100 cask [93]

The fuel baskets contain boron-10 which has a neutron capture cross-section of 3840 barns for thermal neutrons. The outer dimension of the MPC shown in Figure 5-4 is 1.7367 m.

5.3.4. Radiation containment in the HI-STAR 100 cask

Radiation shielding in the HI-STAR 100 cask begins at the MPC and goes all the way to the overpack. The steel and borated plates in the MPC attenuate gamma and neutron radiation. The overpack is made of carbon steel which attenuates gamma radiation. The inner and outer diameters of the overpack are 1.74625 m and 2.4384 m respectively. This gives the overpack a thickness of about 0.69215 m. It consists of the inner, first intermediate and intermediate shells as shown in Figure 5-5 below. These are assembled in plate stock forms. One of the carbon steel types used in the HI-STAR 100 overpack is SA516 Grade 70, which consists of the elements in Table 5-5 shown below. The cask design assumes an internal gamma radiation exposure of the order of 10^{10} rads (1 rad equals 0.01 gray) from the spent nuclear fuel. Damage of the material in the cask due to neutron radiation exposure is only expected to be experienced at $10^{19} n/cm^2$ [93].

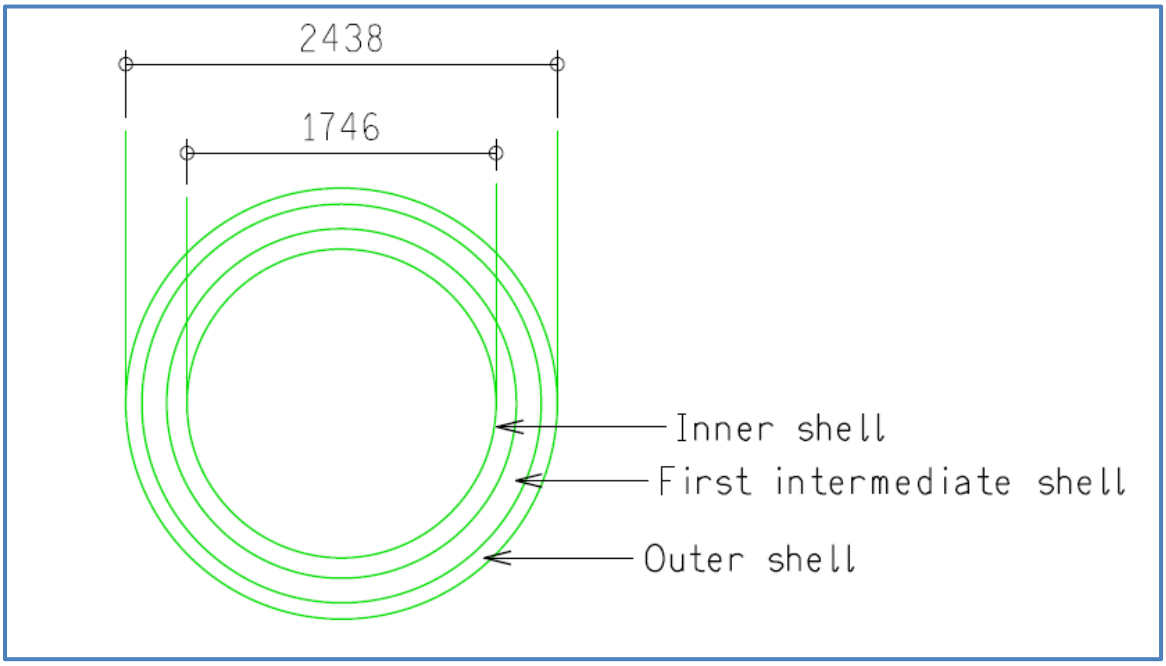


Figure 5-5: Simplified cross-section view of the HI-STAR overpack with dimensions given in millimetres

Table 5-5: Elements in the SA516 Grade 70 carbon steel and their weight

Element	**Elemental concentration (%w)**	**Linear attenuation coefficient (cm^2/g) at 4MeV**	**Density, ρ ($g/cm^{3)}$)**	**Total linear attenuation coefficient, μ_i [ρ $\times$ (cm^{-1})**	**Weighted total linear coefficient, μ [$\%w \times \mu_i$] (cm^{-1})**
Carbon (C)	0.22	3.047×10^{-2}	1.7	0.052	0.0114
Silicon (Si)	0.6	3.24×10^{-2}	2.33	0.076	0.046
Manganese (Mn)	1.17	3.213×10^{-2}	7.47	0.240	0.281
Phosphorus (P)	0.03	3.172×10^{-2}	1.823	0.058	0.002
Sulfur (S)	0.03	2.293×10^{-2}	2.0	0.045	0.001
Nickel (Ni)	0.3	3.444×10^{-2}	8.908	0.307	0.092
Chromium (Cr)	0.3	3.325×10^{-2}	7.19	0.239	0.072
Molybdenum (Mo)	0.08	3.496×10^{-2}	10.28	0.359	0.029
Aluminum (Al)	0.02	3.106×10^{-2}	2.7	0.084	0.002
Copper (Cu)	0.3	3.318×10^{-2}	8.96	0.297	0.089
Vanadium (V)	0.02	3.141×10^{-2}	6.11	0.192	0.004
Total linear coefficient of carbon steel (SA516 Grade 70)					0.6294

The overpack also consists of a neutron-absorbing material called Holtite-A. It is defined as "a poured-in-place solid borated synthetic neutron-absorbing polymer" [93]. The properties of Holtite-A material are given in appendix D.

The calculated dose rates on the exterior surface of the HI-STAR 100 casks are lower than the limit specified by the NNR. The highest dose rate is at the bottom of the cask, which is the gamma radiation of 115.63 mrem/hr. The second-highest dose rate is at the side of the cask which is the neutron radiation at 78.65 mrem/hr. Other gamma and neutron radiation dose rates at different positions of the cask are given in Table 5-7 below, along with the limits specified by the United States Nuclear Regulatory Commission and South Africa's National Nuclear Regulator. The gamma energies approximated for this is between 3 and 4 MeV [136]. The NNR limit of 20 mSv a year can only be exceeded with the following scenario: if a worker were to stand by the surface of the cask for a minimum of 2.54×10^7 hours in a year. This is not possible as a year has 8760 hours. This has been calculated as follows using values in Table 5-6:

$$Cask\ radiation\ at\ the\ surface = 78.65\frac{mrem}{hr}$$
$$1\ rem = 0.01\ Sv, \therefore 1\ mrem = \ 1 \times 10^{-5}\ Sv$$
$$therefore\ 78.65\ mrem = 7.865 \times 10^{-4} Sv = 7.865 \times 10^{-7} mSv$$

$$To\ reach\ 20mSv: \frac{20\ mSv/hr}{7.865 \times 10^{-7}\ mSv} = 2.54 \times 10^7 \text{ hours}$$

This scenario is very conservative because it assumes that the distance between the worker and the overpack surface is zero, i.e. the worker is in contact with the surface of the overpack. It also assumes that the worker is not wearing any protective clothing. Even if the radiation at the cask surface is the maximum 200 mrem as specified by the U.S Nuclear Regulatory Commission, the hours required for a worker to reach 20 mSv would be 10 000.

Table 5-6: Calculated maximum external dose rates for the HI-STAR 100 cask with 24 PWR fuel baskets [93]

Radiation type (mrem/hr)	**Top of the cask**	**Side of the cask**	**Bottom of the cask**
Gamma	0.52	1.35	115.63
Neutron	3.63	78.65	11.38
Total	4.15	80	127.01
US NRC limit (mrem/hr)	200	200	200
NNR limit to workers (mSv/year)	20	20	20

5.3.5. Decay heat removal in the HI-STAR 100 cask

Decay heat in the HI-STAR 100 cask is passively transferred from the fuel in the MPC to the environment, as shown in Figure 5-6 below. The MPC basket design also incorporates the elimination of the structural discontinuities in the MPC in order to reduce thermal heat resistance to heat flow. The helium inserted in the cask has a higher thermal conductivity than air, as can be seen in Table 5-7. The calculated maximum fuel and fuel cladding temperatures are 500 and 422 °C respectively.

Table 5-7: Thermal heat conductivity of different materials used for the MPC and the overpack

Material in the MPC	Thermal conductivity (W/m.K)	
	At 366 K (93.33 ° C)	At 644 K (371.11° C)
Helium	0.098	0.158
Aluminum	84.4	97.4
Air	0.017	0.027
Material in the overpack		
Carbon steel	24.4	22.4
Cryogenic steel	23.8	22.3
Holtite-A	1	1

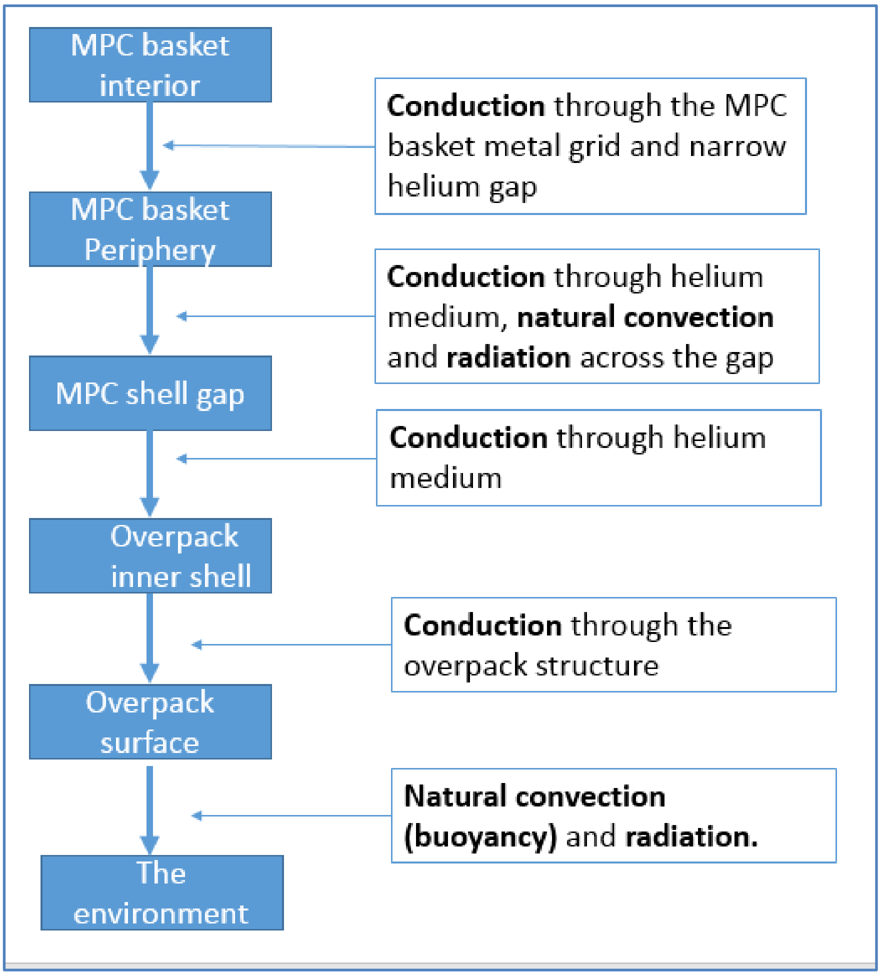


Figure 5-6: Decay heat removal in the HI-STAR 100 cask

The calculation of decay heat removal is complex and done using computer programs as discussed in 3.4. The following is a simplified calculation to demonstrate the decay heat removal phenomenon. The calculation is an approximation of the net radial heat transfer from the SFA interior using the Wooten-Epstein equation that was presented in section 3.4 of this dissertation. The following assumptions are noted: The spent fuel related parameters used are those at Koeberg power station.

Table 5-8: Values used to calculate Wooten-Epstein heat transfer

Parameter	Value	Source
ε_C	0.80	From [93]
ε_B	0.36	From [93]
N	17	From 17 x 17 SFA as indicated in *Table 5-1*
SFA width	0.21 m	From *Table 5-1*
L	3.66 m	From *Table 5-1*
K_{CS}	0.4791	From [93]
T_C	773.15 K	From [93]
T_B	695.15 K	From [93]

$$C_0 = \frac{4N}{(N+1)^2} = \frac{4 \times 17}{(17+1)^2} = 0.2099$$

$$F_\varepsilon = \frac{1}{[\frac{1}{\varepsilon_C} + \frac{1}{\varepsilon_B} - 1]} = \frac{1}{[\frac{1}{0.8} + \frac{1}{0.36} - 1]} = 0.3303$$

$$A_{SFA} = 4 \times SFA\ width \times SFA\ length = 4 \times 3.66 \times 0.21 = 3.0744\ m^2$$

$$\therefore the\ first\ term\ is\ given\ as:$$

$$= \sigma \times C_0 \times F_\varepsilon \times A_{SFA} \times ({T_C}^4 - {T_B}^4)$$

$$= (5.67 \times 10^{-8}) \times (0.2099) \times (0.3303) \times (3.0744) \times (773.15^4 - 695.15^4)$$

$$= 1496.226 \frac{kg}{s^3 K^4} m^2 K^4$$

$$= 1496.226\ \text{W}$$

$$the\ second\ term\ is\ given\ by:$$

$$= 13.5740 \times L \times K_{CS}(T_C - T_B)$$

$$= 13.5740 \times 3.66 \times 0.4791 \times (773.15 - 695.15)$$

$$= 1856.563\ \text{W}$$

$$\therefore the\ net\ radial\ heat\ tranfer = Q_{WE} = 3352.789\ W/\text{hr}$$

In this scenario, heat transfer is dominated by the second term, which represents the conduction heat transfer mechanism. The other assumption made here is that heat generation is uniform in the spent fuel assembly.

5.3.6. Response of the HI-STAR 100 cask to seismic events

The maximum horizontal acceleration of the HI-STAR 100 is 0.473g [96]. The peak ground acceleration (PGA) calculated at Koeberg power station site was 0.3g [139]. The cask can be expected to provide a satisfactory seismic response.

5.3.7. Corrosion of the HI-STAR 100 cask

The austenitic stainless steel in the HI-STAR 100 overpack is mostly prone to stress corrosion cracking (SCC). In environments close to the seas such as the proposed sites for the dry storage facility at Koeberg, the environment supports the phenomenon of SCC, i.e. there is a presence of chloride salts. The helium in the MPC provides a non-aqueous and inert environment that slows down the corrosion phenomenon in the MPC. The steel surface of the overpack is coated with carboline while the inner cavity is coated with thermaline 450 [93]. The physical properties of these are given in Appendix E.

5.3.8. Transportation of the HI-STAR 100 cask

The HI-STAR 100 casks can be transported using heavy haul trailers by road or rail. The casks are transported in a vertical orientation with impact limiters as elucidated above to absorb the kinetic energy in case of an accident, thereby preventing damage to the casks. The proposed dry storage facility and haul route at Koeberg will be within the existing nuclear power plant's footprint, as shown in Figure 5-1 above. This should minimise risks during transportation of the casks as there will be no public interference. Sea transportation can also be used for spent nuclear fuel from the proposed Koeberg dry storage facility to a centralised storage facility in South Africa if it will be along the coast or in other countries outside Africa. An evaluation of a possible route from the proposed dry storage facility to inland parts of the country which has the potential to host a centralised storage facility should also be considered.

6. Conclusion

The role of nuclear power and some of its challenges have been presented briefly in this minor dissertation report. While the contribution of nuclear power continues to grow in some countries, others have decided to reduce its contribution to electricity generation. This is due to a number of issues, including spent nuclear fuel management. The spent fuel pool technology can be qualified as one of the spent fuel management strategies that have matured over the years and is providing interim storage for all countries with nuclear power reactors. However, this interim storage has capacity challenges and needs active cooling. The introduction of dry storage facilities does not replace the need for spent fuel pools but rather makes them return to their intended shorter-term interim storage function while the spent fuel decay heat decreases to an acceptable level. This short term is typically between 3 to 10 years. It is also noted that both spent fuel pool and dry cask storage are interim storage technologies.

Although dry storage facilities are also regarded as an interim storage option, they aid in spent fuel management and in particular addressing the space and active cooling requirement challenges of the SFPs. However, if the capacity and active cooling challenges of the spent fuel pools are resolved, the need for dry storage will be eliminated. South Africa's proposed interim dry storage facility for spent nuclear fuel, expected to be installed by 2025 was presented briefly in this minor dissertation. Some of the major physics and engineering aspects were discussed. The physics aspects discussed here include radiation containment, spent fuel quantity and characteristics and sub-criticality. The engineering aspects discussed are decay heat removal, site location factors, corrosion of the casks, transportation infrastructures, operability and monitoring.

While the proposed interim dry storage facility will accommodate most of the SFAs from Koeberg's reactors, there will be about 9% of the casks that will not be accommodated at the end of a 60-year reactor life. However, should the power plant not operate up to the intended 60-year period, the facility can accommodate all the SFAs up to 54 years of operation. If it is desired to have all the SFAs accommodated at the facility, casks which store more SFAs can be procured later. The dry storage facility can then be extended. One of the significant advantages of dry storage facilities is that they provide modularity just like Paks dry storage facility as presented in this minor dissertation. This feature of dry storage technology should be considered for South Africa in order to address this aspect.

The chosen HI-STAR 100 casks have satisfactory and conservative radiation containment and sub-criticality measures. The spent fuel at Koeberg power plant would have spent more than the required minimum cooling time of 5 years in the spent fuel pool before loading them into the casks. Most of the spent fuel would have spent at least 20 years in the SFP. In addition to many more years of cooling in the spent fuel pool, the burnup at Koeberg is significantly lower than that of the calculated 40 GWd/tHM, i.e. Koeberg's average burnup is 33 GWd/tHM. The fuel enrichment is also below the maximum of 4% of the HI-STAR 100 cask. The decay heat removal capability of the HI-STAR 100 cask is satisfactory for the spent fuel at Koeberg. This is also supported by the longer cooling time the spent fuel would have had in the SFPs as highlighted above.

The proposed facility at Koeberg will be in the existing power station footprint, which is also the case for all other facilities presented here. This will allow for ease of transportation of the cask from the cask building to the dry storage facility. It will also minimise the time for applicable nuclear license and security installations as the site already has some security. South Africa's land area is more than 30 times bigger than of Belgium. The airports in South Africa are also less busy than Belgium's. These factors together with the geographical location of South Africa in contrast to Belgium's, makes air traffic congestion the least of South Africa's worries, thereby reducing the probability of plane crash incidents on the proposed facility. There is, therefore, no need for South Africa's proposed dry cask storage facility to be housed in a concrete building like that at Doel.

While the proposed dry storage facility may not present an apparent technical problem, other non-technical aspects should be considered when deciding on the facility. One of the key aspects is the timeline of the facility based on the proposed delay in new nuclear reactors and decreased energy demand in contrast to the original IRP 2010. A delay in building the facility sounds reasonable on the basis that there would be less electricity generated from the country. However, Koeberg is a baseload power station and is likely to remain like that for a long time due to its lowest operation cost compared to other power stations [140]. This implies that even though the electricity demand has decreased with the possibility of continuing to decrease, Koeberg will continue to generate spent fuel waste continuously.

A delay in the facility construction will imply that phase 1A of the proposed spent fuel management which involves procurement of 7 metal casks to be stored in the modified cask building. Assuming the same casks as the ones in the cask building are procured, the seven casks would store 196 fuel assemblies (28 FAs per cask). This would be equivalent to 6 outages, i.e. three outages per reactor, thereby giving three additional years from 2018 before the spent fuel pool and cask building are full. The decision to delay the interim storage beyond 2021 would therefore not be favourable if the power station is to operate up to its initial 40-year lifespan or the extended 60-year lifespan. The transportation of the casks from the spent fuel pool does not present any apparent challenges as it will be within the power station footprint with no interference from the public. The cask can also withstand the expected maximum ground peak acceleration at Koeberg.

After writing this minor dissertation, it is the opinion of the writer that the proposed dry storage facility at Koeberg is satisfactory to provide an interim storage solution to allow for continuous operation of Koeberg nuclear power plant. This opinion is based on the physics and engineering aspects presented in this minor dissertation. The continuous operation of Koeberg nuclear power plant is vital for South Africa, especially at these current challenging economic times for the country. The power plant has the lowest operating costs, and the dry storage will avoid premature closure of the power plant. Future research for those wishing to add to the analysis of the suitability of this technology can include cost of the casks.

Appendices

Appendix A: Derivation of the scattered photon's energy in Compton scattering

The incident photon has an incident energy E_γ. The scattered photon will have energy E'_γ. This is shown below.

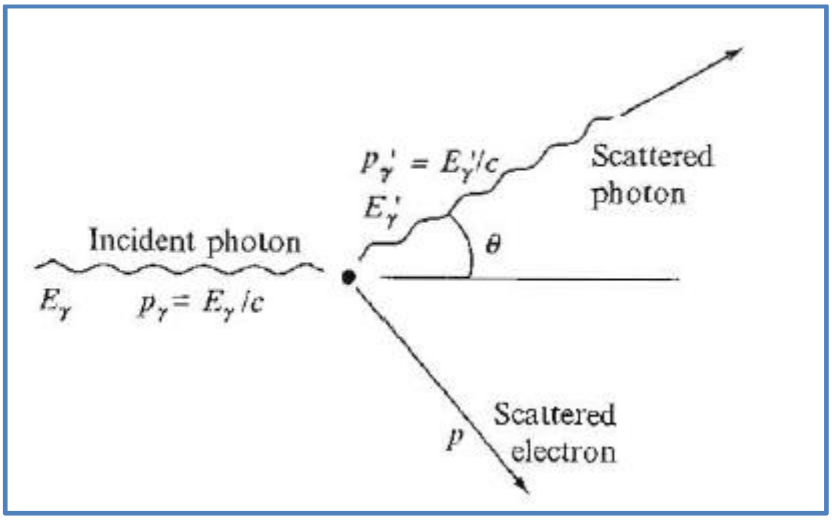


Figure 0-1: Compton scattering adapted from [42]

Conservation of energy gives the following equation for the scattered electron:

$$T = E_\gamma - E_\gamma' = E - mc^2 \qquad \text{(A1)}$$

The conservation of momentum also gives the following equations for the incident photon and scattered photon respectively:

$$p_\gamma = \frac{E_\gamma}{c} \text{ and } p'_\gamma = \frac{E_\gamma'}{C} \qquad \text{(A2)}$$

The momentum of the scattered electron must, therefore, be added vectorially, as shown in Figure 0-2 below.

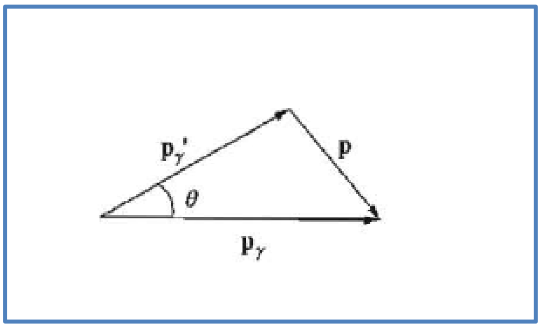


Figure 0-2: The vector relationship between the momentum of the incident photon, scattered photon and the scattered electron

The cosine rule is applied to give the following equation:

$$(pc)^2 = {E_\gamma}^2 + ({E_\gamma}')^2 - 2E_\gamma E'_\gamma \cos\theta \qquad \text{(A3)}$$

Substituting for p_γ and $p_\gamma{}'$ in equation (A3) from equation (A2) and using the relation $E^2 = p^2\, c^2 + m^2\, c^4$ for the electron, (A3) becomes:

$$(pc)^2 = E^2 - m^2c^4 \qquad \text{(A4)}$$

Eliminating E from (A4) and (A1), the scattered photon energy can be expressed as follows:

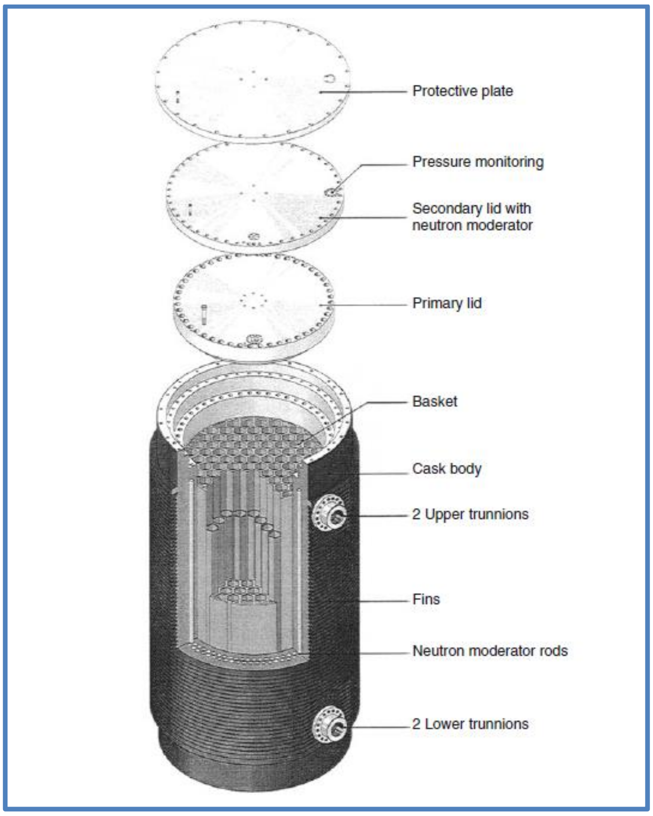
$${E_\gamma}' = \frac{E_\gamma}{1+\left(\frac{E_\gamma}{m^2}\right)(1-\cos\theta)}$$

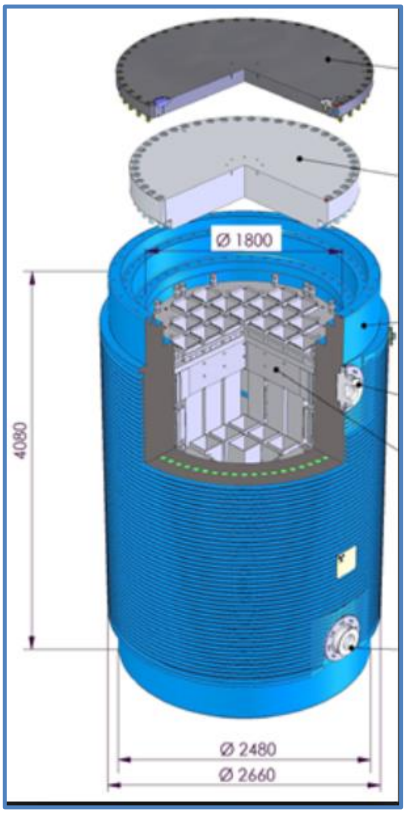
Appendix B: Parameters of a CASTOR 440/84 metal cask adapted from [70]

Table 0-1: Parameters of CASTOR 440/84 metal cask

Maximum Heat output	21kW
Maximum enrichment	3.65%
Maximum burnup	42 GWd/tHM
Typical Cooling time	60 months
Cask length	4.080 m
Cask diameter	2.660 m
Wall thickness	370 mm
Cask body material	Ductile cast iron
Lid material	Stainless steel
Mass of transportation configuration	131 tons

Appendix C: Detailed diagram of a CASTOR 440/84 metal cask as adapted from [70] and [141]





The above metal cask is designed to store 24 PWR fuel assemblies. The dimensions shown on the right are in mm. The following table shows some of the parameters of the cask [70].

Appendix D: Properties of Holtite-A adapted from [93]

PHYSICAL PROPERTIES	
% ATH	62 (nominal)
Specific Gravity	1.68 g/cc (nominal)
Max. Continuous Operating Temperature	300°F
Hydrogen Density	0.096 g/cc minimum
Radiation Resistance	Excellent
CHEMICAL PROPERTIES (Nominal)	
wt% Aluminum	21.5
wt% Hydrogen	6.0 (nominal)
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B_4C	1.0 (nominal)

Figure 0-3: Properties of Holtite-A adapted from [93]

Appendix E: Properties of carboline

Carboline as used in coating of the HI-STAR 100 cask is a dense cross-linked polymer used to minimize surface corrosion in casks.

Continuous Maximum Temperature resistance	218 °C
Maximum excursion	232 °C

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