Status report 70 - Pebble Bed Modular Reactor (PBMR)

Overview

Full name	Pebble Bed Modular Reactor
Acronym	PBMR
Reactor type	Pebble Bed Type Reactor
Coolant	Helium
Moderator	Graphite
Neutron spectrum	Thermal Neutrons
Thermal capacity	400.00 MWth
Gross Electrical capacity	165.00 MWe
Design status	Conceptual Design
Designers	Pebble Bed Modular Reactor (Pty) Limited (PBMR)
Last update	10-08-2011

Description

Introduction

The Pebble Bed Modular Reactor is a High Temperature Gas Cooled Reactor based on the evolutionary design of the German AVR, THTR and HTR-Modul designs. It is being designed and marketed by PBMR (Pty) Ltd.

Various reactor concepts have been under development since 1996. The original design was based on a direct Brayton cycle as this held and holds promise of higher efficiencies. The maximum achievable power levels for the reactor was increased in several design steps in order to reach a set target for installed cost/kW that would be roughly comparable to coal fired power when lifetime costs were evaluated. As a result the design of the reactor core evolved from the original base of 200 MWt adopted from the HTR-Modul[1] design to reach 400 MWt with an annular core. Because the direct cycle efficiency is very sensitive to gas outlet temperature, a reactor outlet temperature of 900 ˚C was selected with an inlet temperature of 500 ˚C. The development of the turbine generator set went from a 3 shaft system with two high speed and separate high and low pressure turbo units (turbine and compressor mounted on one shaft) and a 3000 RPM turbine/generator, to a single horizontal one shaft arrangement in 2004. This development was the result of the adoption of dry gas seals to separate the turbine from the generator as well as the introduction of a 2:1 gearbox which allowed a significant reduction in turbine blade size due to the higher 6000 rpm speed. Techno-economic studies were carried out on a regular basis. Although no insurmountable technical problems were encountered, it was found that operating at 900 ˚C ROT (Reactor Outlet Temperature) was testing the limits of available approved materials, including the turbine blades. In addition the steep price increases for all materials experienced by all power equipment suppliers increased the cost significantly. Because of the low core power density, a result of the choice to embrace the inherent safety characteristics of modular HTRs, these price increases affected the economic competitiveness of the design more than those of larger power units.

Although some components for the plant could be adapted from the THTR design [2], there was still a need to re-engineer these for the selected design, and manufacture and test these components under real plant conditions. For this reason a test facility, the Helium Test Facility (HTF) [3] was completed in 2007 and is in operation at a test site at Pelindaba near Pretoria. When not used for component testing, this facility can be used for more basic experiments and development work where the tests need to be carried out under reactor conditions using helium at high temperature and pressure. Before this there was another accomplishment, where a micro module Brayton cycle test loop using nitrogen as the working fluid was built and successfully operated in 2003 at the North-West University [4]. This proved that a three-shaft Brayton cycle can be started up and controlled in the manner foreseen in the analyses. At the same university a large test facility was constructed and operated to validate the natural circulation analysis and heat transport parameters for a pebble bed. Results for all these experiments were reported in the open literature or are being prepared for publication [5]. Beside these facilities which were directly "project managed" by PBMR, the company has also sponsored numerous university research projects in areas related to materials, fuel characterisation, graphite development etc. PBMR also paid for technology development done by manufacturers of the graphite, compact heat exchangers and turbines.

Whilst the basic design for the direct cycle was in development, it became apparent since about 2004, that the increases in oil prices and the rising concern about climate change have sparked interest in HTRs as a source of process heat. PBMR received enquiries from several quarters with the result that a separate project team started to investigate the possibilities and the most appropriate reactor size and configuration for that market. As a result of the world financial crisis short term funding for PBMR (Pty) Ltd was curtailed, forcing a rethink of the product priorities. The outcome was a Board resolution to concentrate on the electricity and process heat market using an indirect steam cycle. The direct cycle design is archived with a view to further progress this design when conditions improve and material development catches up with the demanding conditions of the Brayton cycle. Based on the market research and need to avoid time consuming and costly development work, it was decided that the plant should consist of smaller modules to better match the market requirements of limited steam and electricity demand as well as high availability but with assurance of supply. The chosen solution for a co-generation plant is summarised in Table 1 below.

Table 1: Projected operating parameters for a twin unit PBMR-Cogeneration plant

Reactor thermal output [MWt]	250 (x2)
Cycle type	Indirect
Thermodynamic efficiency for power generation part [%]	40
Design lifetime [years]	60
Maximum steam mass flow from both steam generators [kg/s]	195
Maximum generator electric output for 50:50 split of steam between power generation and process steam [MWe]	100
Maximum generator electric output for all electric plant [MWe]	200
Availability target (%)	95
Load following (%)	50-100

Per Reactor	
Number of fuel spheres	360 000
Diameter of a fuel element [mm]	60
Moderator and reflector	graphite
Control rods in reflector	6
Shutdown channels in reflector with boronated spheres	18
Fuel cycle concept	Multi-pass
Average fuel sphere residence time [days]	300
Average burn-up [MWd/t]	80 000 –92 000
Primary coolant	Helium
Helium flow at per reactor [kg/s]	96
Helium inlet temperature [°C]	250
Helium outlet temperature [°C]	750
Helium average pressure [MPa]	6
Fuel average power density [MW/m3]	< 4.0
For Steam Generators	
Steam pressure from steam generator [MPa]	12
Steam temperature from steam generator [°C]	540
Feedwater inlet temperature [°C]	200

Steam flow to reboiler for process steam generation [kg/s]	97*
Steam flow for electricity generation [kg/s]	97*
Process Steam conditions after reboiler (if fitted) [MPa], [°C]	3.55/385*
Process Steam mass flow [kg/s]	83*

* Values can be varied according to customer requirements

As is nowadays the norm for modular HTRs that are seen as Gen IV designs, the PBMR design has the following attributes that contribute to enhanced safety:

- Use of Triple Coated Isotropic (TRISO) fuel particles shown to remain intact to at least 1600 ˚C and with some time delayed failure fractions at even higher temperatures [6,7];
- A geometry that allows the passive dissipation of decay heat to an external heat sink;
- Relatively low power density to aid in the limitation of fuel temperatures following a loss of coolant due to an un-isolatable leak;
- Load following limited to 50-100% to reduce the excess operating reactivity to a value that prevents fuel failure for any scenario of group control rod withdrawal without scram;
- Use of helium as coolant which avoids the effects of phase changes and has a negligible effect on plant reactivity when pressures fluctuate;
- Control rods move only in the reflector and thus avoid any danger of mechanical damage to the fuel spheres;
- Limiting the heavy metal fuel loading to ensure minimum under-moderation so that any water ingress from the coupled steam generator cannot cause undue reactivity addition;
- Use of nuclear grade graphite to ensure minimal corrosion by impurities and low activation at end of life.

2.1 A typical Pebble Bed Co-Generation Power Plant (PBMR-CG)

The PBMR-CG is intended as a versatile energy source for the generation of electrical power, process heat or a combination of electricity and process heat (cogeneration).

The PBMR-CG can be described as a standardized nuclear power reactor module with a thermal output of ~250 MW. These modules can be combined in multi-module units for larger plants. Figure 1 shows an example of a two-module unit (steam side excluded), which can be expanded to more units or more modules. This modular concept makes it possible to use the unique safety characteristics of small High Temperature Reactors (HTRs) and to build larger plants by combining more modules. The steam from the modules can be combined to drive larger steam turbines or common steam headers that feed into a process plant.

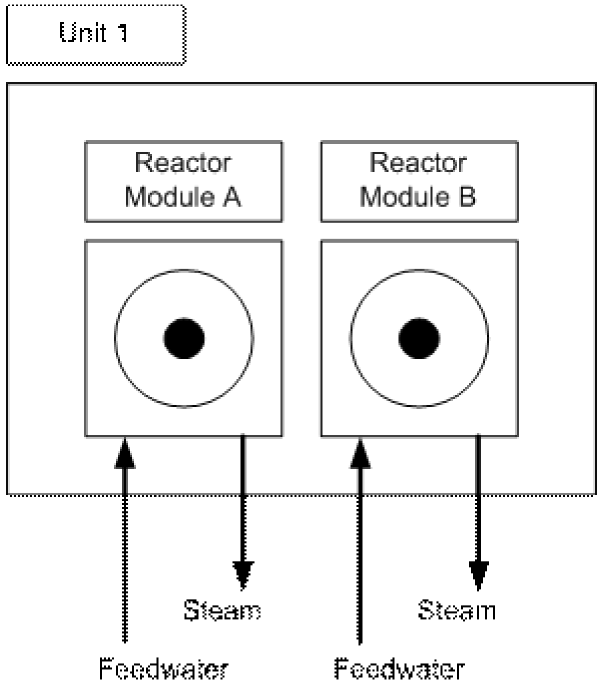


Figure 1: PBMR-CG Unit with Two Reactor Modules

Figure 2 shows a module arrangement consisting of a Nuclear Heat Supply System (NHSS) and a connected water/steam system. Each NHSS module consists of an HTR, a steam generator and a helium circulator.

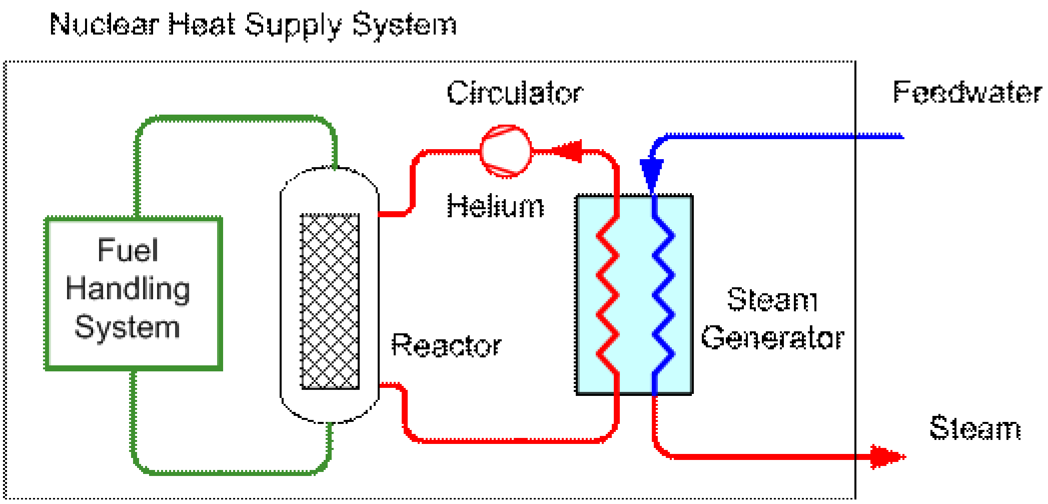


Figure 2: NHSS Module Diagram

The HTR core contains the fuel spheres where heat is generated by nuclear fission of uranium fuel. The helium primary coolant flows through the reactor core where it is heated by flowing around the fuel spheres, and from the reactor to the steam generator. In the steam generator, the hot helium transfers the heat through the heat exchanger to the water, and the cooled helium is circulated back into the reactor core to be heated again. The heat from the hot helium in the steam generator converts the water into superheated steam which is delivered for use in the application side. This steam can then be used to drive applications such as steam turbines for electricity generation, chemical processes or desalination.

2.2 General layout

The flow diagram for a single module with steam generator is shown in Figure 2. The high-temperature reactor core inside the Reactor Unit contains the fuel spheres where heat is generated by nuclear fission of uranium fuel.

2.3 The Reactor Unit

The reactor unit consists of the 3 meter diameter core filled with fuel spheres and surrounded by the graphite reflector. The functions of the reflector are to:

- Reflect escaping neutrons back into the core, to keep the neutron losses as low as possible;
- Provide paths for the control rods and the reserve shutdown system absorbers to enter the core region to shut down the reactor, and
- To provide a heat transport path for the decay heat from the fuel to the Reactor Pressure Vessel so that passive cooling of the core is possible by radiating this heat to a suitable heat sink called the Reactor Cavity Cooling System (RCCS).
- Provide the volume in which the fuel spheres can move through the core but remain in a well defined geometry.
- Provide a flow path for the cold gas to be returned to the top of the core.

The side reflector consists of 24 columns of graphite blocks stacked on the bottom reflector. All the columns are free to move individually but are connected with keys to prevent or minimise gas leak flows. Each of the blocks has a boring in which the control elements can move. The borings are connected with sleeves to guide the rods and again prevent leak flows. The bottom reflector is made from larger graphite blocks and is supported by the core barrel bottom. It also contains slots through which the hot gas can pass through to the collections chamber from where the gas enters the hot gas duct.

In Figure 3, a and b show a cut through the reactor unit with the room for the fuel spheres in the centre, the core unloading device and the top reflector on which the control units are mounted.

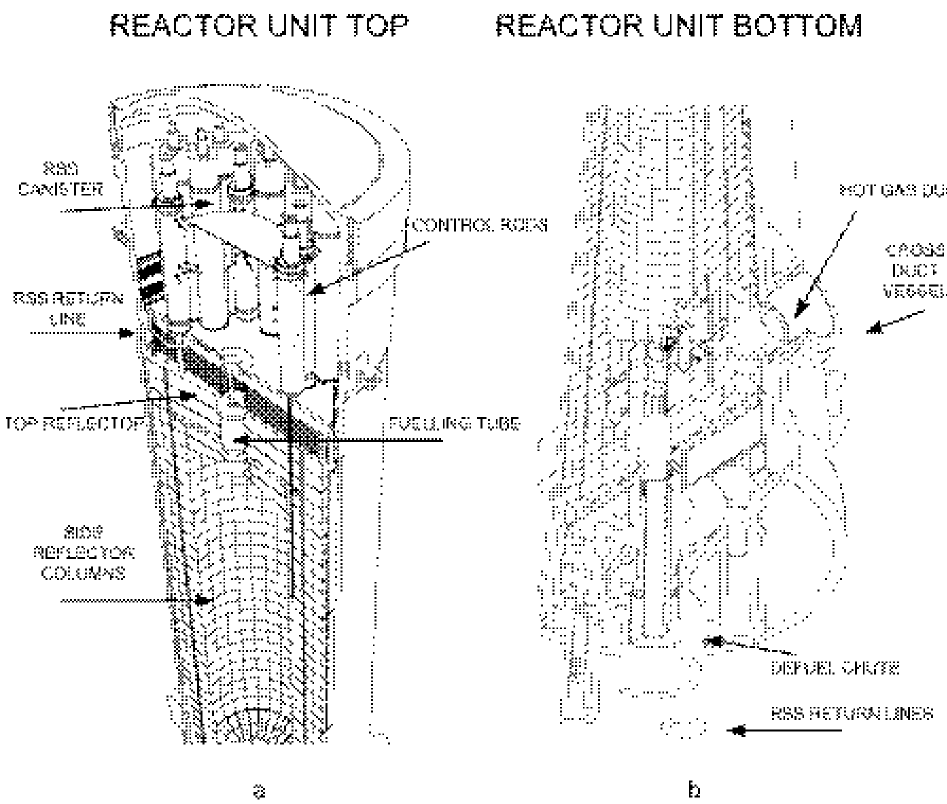


Figure 3: Reactor unit top (left) and bottom (right) showing core internals

In operation, fuel is introduced to the core through an opening in the middle of the top plate and top reflector. The metal top plate seals the core from the top cavity which also houses the Reserve Shutdown System containers. Fuel spheres pass through the core and are removed at the bottom through the Core Unloading Device which is connected to the Defuel Chute. At anyone time there are about 360 000 spheres in the core and with a residence time of 100 days and a recirculation rate of 3000 spheres per day that means each sphere moves on average about 8 cm per day although the flow speed varies across the core diameter. New spheres are introduced at an average of 350 per full operating day and on average the same number, having reached the target burnup, are removed from the return line to the spent fuel storage. The resultant neutron flux and power distribution peaks at a point above the axial middle of the core as the upper part contains more fresh fuel than the lower section.

2.4 The fuel handling system

This is a very important part of the plant as it enables the continuous fuelling and defuelling. Each fuel sphere goes through a number of cycles, the average of which is ~10 passes through the core. After a fuel sphere is extracted from the defuelling chute it is checked for mechanical soundness by the scrap separator and then tested for burnup by investigating the gamma spectrum for the intensity of the Cs-137 peak. Should a sphere exceed the target burnup, it is shunted away to the spent fuel containers located low in the reactor building. Fresh fuel is introduced at the same location to replenish the core content and the replenishment rate is determined by the spent fuel removed and the required excess reactivity. Each fuel sphere is counted at all stations outside the core and a daily balance is calculated to ensure that the core is neither too full or too empty to prevent blockages or a long drop onto the pebble bed. By regulating the addition of fresh fuel the core reactivity can be kept at the design value for normal operation.

Because of the continuous fuelling property of the reactor, the plant could have an availability of 100% if it were not for the need to regularly maintain and/or inspect moving parts in the rest of the plant. Much care is therefore taken to design for on-line maintenance wherever possible and keep unavoidable shut-downs as low as possible by planning for all maintenance tasks early in the design.

2.5 The reactor control system

In accordance with international practice the design provides for two independent and diverse reactor shutdown systems. One system consists of 6 control rods that move in the borings of the side reflector and are used during normal operation to keep the reactor power output constant at the selected levels or to aid in changing power level when this is desired. The design provides for load following capability between 50% and 100% of full power. At 100% power the control rods, which are 6 meter long, are inserted to a depth of 3 meters into the side reflector. The remaining 3 meters of insertion depth available is sufficient to shut down the reactor in the hot operating mode and keep it in that state until the Xenon starts to decay after about 20 hours. If after 20 hours the reactor cannot be restarted, additional negative reactivity needs to be inserted to keep the reactor sub-critical. This is done with the Reserve Shutdown System (RSS) which consists of 18 containers filled with borated graphite pellets that will drop into the remaining borings when the valve opens on loss of power to the holding magnets. The RSS has sufficient negative reactivity to enable the core to be cooled down to maintenance temperature at all times. The positions of the control rods and RSS containers is indicated in Figure 3 b as being under the lid of the Reactor Pressure Vessel (RPV). This arrangement still needs to be verified based on a detail maintenance and availability study. On start-up the small absorber spheres are mobilised by the return line blower which lifts the pellets through the 18 pipes that run inside the RPV.

2.6 The heat transfer system

The heat transfer system takes heat produced in the core and transforms it into usable energy in the form of steam or electricity. The main instrument for this is the Steam Generator which has the helium gas on the inside of the SG pressure vessel and water circulates through pipes within the SG where it is heated up under pressure to produce steam at the required temperature and pressure for further use. The helium circulator which circulates the gas through the core and the SG is mounted on top of the SG. The circulator draws cool gas from the bottom of the SG which is where the feedwater enters the tubes, and forces it through the core from where it re-enters the SG through the co-axial cross structure at the location where the steam exits the SG. The hot gas passes through and past the helical coil tubes of the SG, transferring heat to the water. The hot gas is separated from the SG vessel by an insulated shroud that covers the whole of the tube assembly to the point where the cold water enters the tubes.

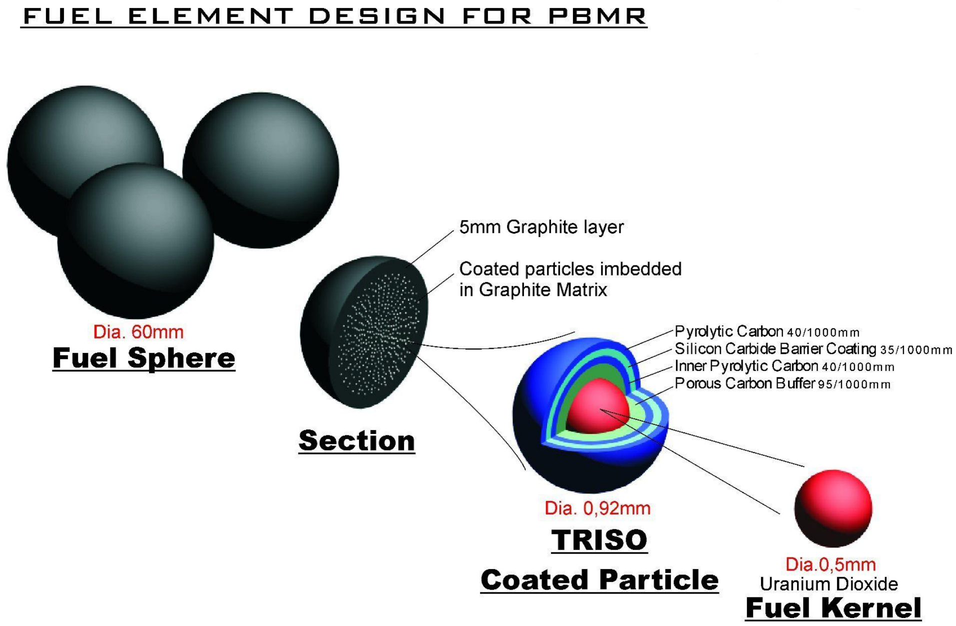
2.7 Operating Features

The PBMR-CG is designed for ease of operation. The main operating features of the PBMR-CG reference design are:

- High reactor availability due to online refuelling. The pebble bed reactor technology allows for online refuelling of the reactor by inserting fresh fuel at the top of the core and extracting spent fuel at the bottom. This capability avoids the need to periodically shut down the reactor for refuelling.
- Increased plant availability due to multiple reactors. The modular concept allows many reactor modules to be combined to supply steam to a turbine or a header in a process plant. The autonomy of each module allows steam production to continue even in the event of a single reactor being removed from service, albeit at a reduced capacity.
- Low equipment count in the primary system, with the major equipment in the primary system being only one reactor, one circulator and one steam generator.
- High thermodynamic efficiency due to a high reactor outlet temperature.
- Highly automated control and protection. The plant is automated for normal operational control and requires minimal operator intervention. The plant also has a fully automated protection system to prevent significant equipment damage or nuclear accidents.
- Slow thermal transients in the primary circuit. The large thermal capacitance in the reactor combined with the negative temperature coefficient of reactivity in the core result in very slow thermal transients after a disturbance. This removes the requirement for quick operator or control system intervention.

2.8 Fuel cycle and fuel types

As described under the fuel handling system, the fuel cycle is continuous fuelling with recycling of fuel that has not reached the burnup limit. The fuel is the form of spheres as used in the AVR and THTR and is described in [8] and recent developments in manufacturing in South Africa in [9]. Of interest is the fact that basically fuel with enrichments between 5 and 20 % can be used because the reactor will eventually reach an equilibrium fuel loading that allows high burnups. Similarly fuel kernels can be made of UO_2, UCO or ThO_2 mixed with enriched uranium.



All these mentioned and other types (i.e. high enrichments) were used and tested in the AVR with great success. The present design is based on the UO_2 kernel with kernel diameter of 500 micron coated with layers of PyC and SiC and approximately 12000 coated particles per sphere. For higher enrichments and burnups the kernel diameter must be reduced to prevent overheating the fuel particle. By judicious use of a mixture of enriched material or Pu and additional Thorium it is relatively simple to design a near breeder with low fuel cycling needs. A full breeder is in principle possible but not presently actively studied until the present concept is fully validated in production reactors.

2.9 Spent fuel strategies

The pebble fuel is extremely well suited for direct disposal in a waste disposal repository due to the very low leaching potential of the graphite surrounding the coated particles and the additional resistance to fission product release from the particles themselves. Fuel from the THTR is presently stored at Ahausen in Germany in air cooled containers while the AVR fuel is still stored at the research centre in Jülich. Arguments against direct disposal are the cost of large volumes of spheres and the potential energy still locked in the spent fuel. Thus no decision on any method of disposal is likely to be made in the near future and may in part depend on the need to dispose of fuel from Koeberg and other planned nuclear installations in South Africa.

3.1 Safety philosophy

The safety philosophy for modular HTRs has been described a number of times in the past 30 years and has been adopted with a few modification by PBMR. It's basis is that an accident equivalent to severe core damage must be inherently impossible by limiting reactivity increases and ensuring that decay heat can be removed passively after a loss of coolant event. The way this is achieved is described in section 2.5. An additional requirement adopted by PBMR is that there should be no need for operator action to ensure public safety within the first 72 hours after an event.

The PBMR-CG reference plant is designed with extensive passive safety features. Even in the case of hypothetical accidents, the decay heat can be dissipated from the primary system to the environment by passive measures alone for a sufficiently long enough time by heat conduction, convection and thermal radiation. This is to a large extent due to the low power density in the core and the neutronically inefficient, but for safety reasons important, long section of the RPV that is uninsulated.

The distinguishing safety feature of the PBMR-CG reference design is the way in which the reactor is designed. The design ensures that the coated particles in the fuel spheres retain practically all radioactive fission products, even during accident conditions, with a total failure of active cooling systems combined with a complete loss of primary coolant. This feature removes the need for additional systems, electrical power and coolant to guarantee coolant circulation through the core after an accident. In this way, it is ensured that the PBMR plant does not pose any danger to the environment or surrounding communities during normal operation or after an accident.

The basic characteristics and subsystem characteristics that provide these high safety levels for a PBMR-CG plant are summarized as follows:

- Ceramic fuel spheres are used, each containing many thousands of coated UO_2 particles that are capable of retaining practically all radioactive fission products up to very high temperatures.
- The reactor core is designed in such a way that during a power failure and other events, the maximum design temperature of the fuel spheres cannot be exceeded.
- Unlike conventional water reactors, no active cooling of the reactor core is needed, even in the event of failures. The decay heat can be removed from the reactor by passive heat transport mechanisms such as heat conductivity, heat radiation and natural heat convection in combination with a heat exchanger that is placed in the citadel surrounding the RPV.
- The shutdown systems are designed and placed in such a way that, when required, the absorber material can be inserted by means of gravity into borings in the graphite reflector. However, even if that fails the strong negative temperature coefficient will still make the core subcritical as soon as the fuel temperature increases by a few degrees.
- The fuel spheres and core structure components are constructed from nuclear-grade graphite. The graphite is capable of withstanding very high temperatures in the core without losing structural integrity.
- Helium gas is used as the reactor coolant. Helium is chemically inert and neutronically insensitive to pressure and temperature. In addition, the helium is not subject to phase changes that can complicate operation. Loss of helium pressure does not lead to fuel failure.
- The reactor and steam generator are in separate steel pressure vessels and designed in such a way that in the event of a failure in the primary cooling system, there is no danger of overheating of components in the

primary system. The layout allows access to the equipment for inspections and maintenance.
- Because of the low expected coolant gas contamination and the advantages of limited venting of escaping gas from design basis leaks, there is no high-pressure building containment. Instead, the gas is vented and a low-pressure controlled release containment is selected as the safest design solution.

3.2 Source Term

The source term can be defined as either the total content of fission products contained within the fuel at a given time or as the maximum amount of this that can be released in the event of an accident. The latter definition is used from here onwards. This source term represents the radioactivity that must be contained within the confines of the reactor site and this drives many of the design constraints. The PBMR source term available outside the fuel which should be used in accident analysis has two components:

The first comprises the circulating and deposited radioactive nuclides that are either in the form of aerosols, or attached to dust. Part of this source term may be released should a leak occur in the pressure boundary. This source term is calculated to be small and will not lead to any exceedance of the regulatory limit, even if the release is unfiltered. Based on detailed analysis of the previous direct cycle design it is expected that analysis of the reference Rankine plant design, using realistic coolant bypasses in the core and conservative values for dust production, will show that the basic assumptions are valid.

The second source term may arise due to delayed releases from the fuel resulting from heat-up of the fuel particles, should all active cooling fail. A part of increased fission product release from the fuel is due to already failed particles as well as a small percentage of additional failures caused by increased temperatures. Another part is due to enhanced diffusion of some metallic fission products from the particles. Most of these releases will be absorbed in the structural graphite, but for the initial bounding analysis it is assumed that they find their way into the reactor confinement building. Because the confinement, after a pressure pulse, can be regarded as a low-pressure containment, controlled releases through High Efficiency Particulate Air (HEPA) filters and carbon filters, as well as significant plate-out in the structures, such delayed releases will also lead to a very low off-site dose.

The quality and diversity of defence in depth in the systems needed is greatly determined by the magnitude of the second source term.

3.3 Barriers and Defence-in-Depth

The most important barriers to fission product release are the coatings of the fuel particles. High manufacturing requirements and strict quality control will ensure that the amount of free fissile or fertile material in the graphite matrix will be very low, and the percentage of fuel particles with missing or failed coatings will meet the requirements. At the temperatures at which the reference design will operate, the diffusion of metallic fission products out of the fuel will be minimal, ensuring a low activity level in the primary circuit.

A second barrier is provided by the Helium Pressure Boundary. This enclosure assures that only very small amounts of helium and nuclides circulating within it are released into the building during normal operation. Those that are, are released through the stack, and only in the case of a leak will the carbon filter be introduced to remove specific aerosols such as iodine, caesium, and strontium compounds, which may be present.

A third barrier is the confinement building. This is that part of the building which houses the primary coolant pressure boundary. The confinement is designed for very low leakage at low pressure, and to prevent damage to components important to safety, as well as to contain the build-up of higher activity gas in the delayed phase of a depressurisation event. The confinement is designed to vent releases for leaks > 10 mm equivalent, that cannot be isolated, until the primary gas is at atmospheric pressure. If the overpressure in the building needs to be relieved, the escaping gas will be passed through dust filters to remove particulates. This is considered to be a system introduced to show As Low As Reasonably Achievable (ALARA) application. After venting, the confinement is closed and, as a functioning low-pressure containment, it will either retain the activity or, through a filtered release, ensure that any delayed releases cannot lead to the regulatory limit being exceeded, and can be shown to be ALARA.

Defence-in-Depth is provided by active systems that can circulate the gas on failure of the main circulator and with a loss of grid power. This auxiliary system can be supplied from diesels as are the additional heat exchangers and water supply. For small unisolatable leaks the ventilation system has additional carbon filters that can remove aerosols before release through the stack. Due to the design philosophy, all D-i-D engineered safeguards fulfil mainly an investment protection function and help keep internal releases to a minimum for ALARA purposes.

3.4 Reactivity Control

Because of the continuous fuelling, there is no need for a large amount of excess reactivity to allow for burn-up. This also implies a lesser need for a large capability for reactivity control. For the reference design, the control during normal operation is effected by six control rods that together have enough negative reactivity to allow load following between 50% and 100%, and be able to shut down the reactor from any operating point, and keep it shut down in the hot condition until the Xe peak has passed. To enable shutdown to maintenance conditions, or for subcriticality beyond 20 h, the present design includes 18 Small Absorber Sphere channels that together provide enough negative reactivity to keep the reactor subcritical in maintenance conditions (i.e. Xe-free at a temperature below 200 ˚C). The high negative temperature coefficient of reactivity also helps to quickly take the reactor subcritical in the event of a sudden temperature increase, even before the rods are inserted. Stopping the main circulator will also lead to a quick increase in core temperatures, enough to immediately shut down the reactor. Changes in reactivity due to erroneous fuel loading (either too much or too little fresh fuel) will be detected by the control rod positions and, because it is a slow process, the operator can take remedial action without having to shut down the reactor. All reactivity transients will be analysed as part of the safety case.

3.5 Heat Removal

The Reactor Cavity Cooling System (RCCS) is a means to remove residual heat passively for a defined time, and indefinitely with the use of an active system after refilling the cooling system. For this to work, the RPV and the core need to be long and narrow. There must also not be any insulation in the belt region of the RPV, to allow heat radiation and convection to the water filled cavity cooler. Analysis will show, as has been done for the direct cycle, that with the loss of active core cooling by the main circulation system, the cavity cooler and/or the building structural materials will be able to limit the increase in fuel temperature in the most affected region of the core to below the allowable fuel temperature limit. The design of the cavity cooler or RCCS will be subject to safety, reliability and economic trade-offs.

3.6 Chemical Attack

A specific concern applicable to graphite moderated reactor designs that regulators have raised is the fact that graphite at high temperatures may oxidise in the presence of air or water. This is called chemical attack and the possibility of a graphite fire needs to be excluded by design. It therefore warrants a special section in the safety provisions description.

During normal operation, the Helium Purification System ensures that the coolant contains a minimum of impurities capable of reacting with the graphite of the fuel and structures. However, there are two potential events in which reactive substances can enter the core region:

Because the water pressure in a Rankine cycle is higher then the primary circuit gas pressure, the most likely possibility for moisture ingress is a steam generator leak, which will cause water vapour to enter the core, where it can attack graphite that is at a temperature > 800 °C. The oxidation products are CO and H, both of which can form flammable mixtures with air when released from the pressure boundary. As a steam generator leak does not imply a pressure boundary leak, there is little likelihood of releases to the building atmosphere. Analyses of such releases are very dependent on the building size and layout, but for the HTR-Modul design, it could be shown that no flammable mixture can result. Similar analysis will be performed when the design matures. The oxidation process will result in a small increase in radioactivity released from the graphite, and this will be added to any source term when relevant. The accident will be terminated by removing the water vapour or by cooling the core, depending on the result of analysis on the most effective method for various scenarios.

The second major possibility for chemical attack (assuming limited potential for oil ingress) is air ingress through a sizeable pressure boundary break. This potential cannot be evaluated until more details on pipe and vessel connections to the RPV are available. However, a hypothetical accident assuming a Double-ended Guillotine Break (DEGB) of the cross-duct vessel was analysed for the HTR-Modul in Germany. This showed limited potential for airflow, due mainly to the pebble bed airflow resistance. The effect of oxidation of fuel spheres and graphite structures due to potential pipe breaks will be evaluated to judge the severity of the problem, and to enable a decision to be made regarding what types of countermeasures, if any, need to be provided in the design.

The analysis for water ingress shows that there is a natural maximum of water vapour that can enter the core due to the partial pressure at the core temperature. However, if uncontrolled circulation of gas continues and there is also no stop to the feedwater supply, moisture can enter the core, oxidise the graphite and the resulting gas and moisture will continue circulating till the primary pressure relieve valve operates. In order to prevent this scenario, the reactor protection system will act to stop the circulator as soon as moisture is detected. Additional measures enacted by the control system are to dump water till the pressure is equalised in addition to stopping the feedwater pumps. Several lower level protection measures are also foreseen to make an unchecked ingress and circulation of moisture a well beyond design basis accident.

3.7 Design against external events

Generally the design requirements for external events, particularly seismic events, is dictated by the location of the plant and the probability and expected magnitude of earthquakes. Because the modular HTR design can only be competitive if the design is standardised and compatible with mass production, a design basis ground acceleration has to be selected for the building and component designs. The present basis for the a-seismic design of the reference plant is a ground acceleration of 0.3g. All components and in particular the long slender RPV are supported in such a manner that the resulting component acceleration does not exceed the elastic/plastic deformation limit of the metal or creates other serious damage of the components of the RPV and the core internals.

Another design requirement which is probably affecting all new reactor designs, is the protection against aircraft or other missile impact. As a design basis a Boeing 767 aircraft is taken as a maximum impact commercial plane and a standard military fighter plane for point impacts against which the building must be designed. The requirements are that the result of the impact may not disable any safety related structures or lead to high releases of radioactivity. In addition there shall be no danger of a fuel fire inside the confinement building. A possible solution is to place the main systems underground as shown in Figure 5.

3.8 Probabilistic risk assessment

The role of PRA in the PBMR design is twofold. In the design phase it is used to ascertain whether safety and or investment protection functions are robust enough to meet the design goals for reliability and availability. It is also used to aid in decision making when several solutions to a design issue are possible. The main role of PRA is expected to be in the licensing stage where most regulators nowadays expect to see a level 3 PRA as part of the licensing submissions. As many regulators are also looking at risk informed licensing and safety classification, the risk assessment is performed at many stages in the design process to ensure design decisions are supported by the risk assessment in the licensing application. Another definition for level 3 PRA needs to be agreed upon with the regulator as a core melt or severe fuel damage frequency as end result is inappropriate for this design.

4.1 Fuel composition

Proliferation is interpreted as the ability to remove irradiated fuel and extract fissile material of sufficient purity to make a powerful atomic explosive. Studies have shown that any spent fuel containing plutonium can be made to explode with low yield but potentially large release of radioactivity. Proliferation resistance is thus the property of the reactor that makes it difficult to extract high grade plutonium or breed U-233. Assuming that PBMR reactors are only built and operated by countries that have signed the NPT, it can be taken as a fact that regular inspections on the fuel usage and fuel inventory balances will be carried out by the IAEA. In pebble bed reactors the average dwell time of a fuel sphere in the core is about 100 days and fuel is removed after 3 years in the core at which point the burnup is in excess of 80 GWd/t. When fuel is removed from the core, the balance of fissionable nuclides and actinides is as given in table 1

Table 2 Approximate content of Pu isotopes per sphere for 6 cycles

Cycle Nr	Pu-239	Pu-240	Pu-241	Pu-242
1 (g/sphere)	.036	.009	.002	-
2 (g/sphere)	.042	.019	.010	.002
3 (g/sphere)	.043	.022	.015	.005
4 (g/sphere)	.043	.023	.020	.010
5 (g/sphere)	.043	.023	.021	.015
6 (g/sphere)	.042	.023	.022	.020

The mixture for spent fuel (cycle 6), is very unsuitable for a nuclear explosive and if a user desires a better mixture it would be necessary to remove fuel spheres after one pass when the burnup is still < 10 GWd/t. Such a means of operation is very quickly detected due to the excessive use of fuel and the fact that the fuel balance will deviate very much from the expected value. At 10 GWd/t the amount of Pu-239 per sphere is about .036 g/sphere which means that nearly 100 000 spheres need to be removed after the first pass to obtain a workable quantity of Pu. This is basically impossible to do undetected and takes at least 6 months of reactor operating time. The same product can be achieved in a much shorter time in a standard LWR.

4.2 Technical detection features

In the normal operation cycle about 3000 spheres are circulated each day and of these about 350 are discarded as spent fuel and replaced with fresh spheres. These values are very dependent on the number of cycles considered optimal in terms of fuel burnup and minimisation of dust production and mechanical fuel damage. Each sphere is counted on insertion and again on leaving the core on several places in the burnup measurement process. Every day the balance of spheres sent to spent fuel storage, returned to the core and fresh fuel added must balance and this accounting can be verified by IAEA inspectors. Thus wilful removal by unauthorised persons will be detected very quickly by the operator and if the operator is trying to divert such fuel the inspectors will certainly detect it before a significant amount has been removed.

Besides the physical detection of fuel removal, any abnormalities can be detected by the fact that early removal of fuel will make it impossible to reach an equilibrium core and would need an adjustment of the initial enrichment to prevent the core from becoming uncontrollable, even if additional graphite spheres were to be added.

Over and above the properties listed above it was found to be difficult to extract spent fuel from the fuel spheres [9] To do this the graphite spheres need to be deconsolidated so that the graphite can be separated from the coated particles. Then the coatings have to be removed before the kernel can be recovered. All this is considerably more work than dissolving a LWR fuel pin with high Pu content.

Safety and security (physical protection)

It is assumed that any PBMR type power plant will have the standard access restrictions that apply nowadays to any nuclear power plant. However, even if an insider were to try to sabotage the plant, there is only limited damage that such a person can do. This is because the module design is such that there is no equivalent to core damage even when all the coolant inventory is lost or replaced by air. If the helium coolant is lost due to a deliberate action, and the active cooling interrupted at the same time, the state of the core will revert to depressurised conduction cooldown to the cavity cooling system and the design intent is that no active countermeasures need be taken for 72 hours to restore either active core cooling or active cavity cooling. Any late releases as described in section 3 will be filtered by the carbon and HEPA filters with minimal effect on the public. Thus the pebble bed reactor is a very unattractive target for terrorist attacks or other potential saboteurs except for publicity purposes.

Description of turbine-generator systems

The present PBMR-CG plant design is for a process heat plant with co-generation. A typical possible plant flow diagram is given in Figure 4. It shows the reactor as heat source with a steam generator the output of which is partly reheated for use in process heat applications and another part drives the various stages of a small turbine.

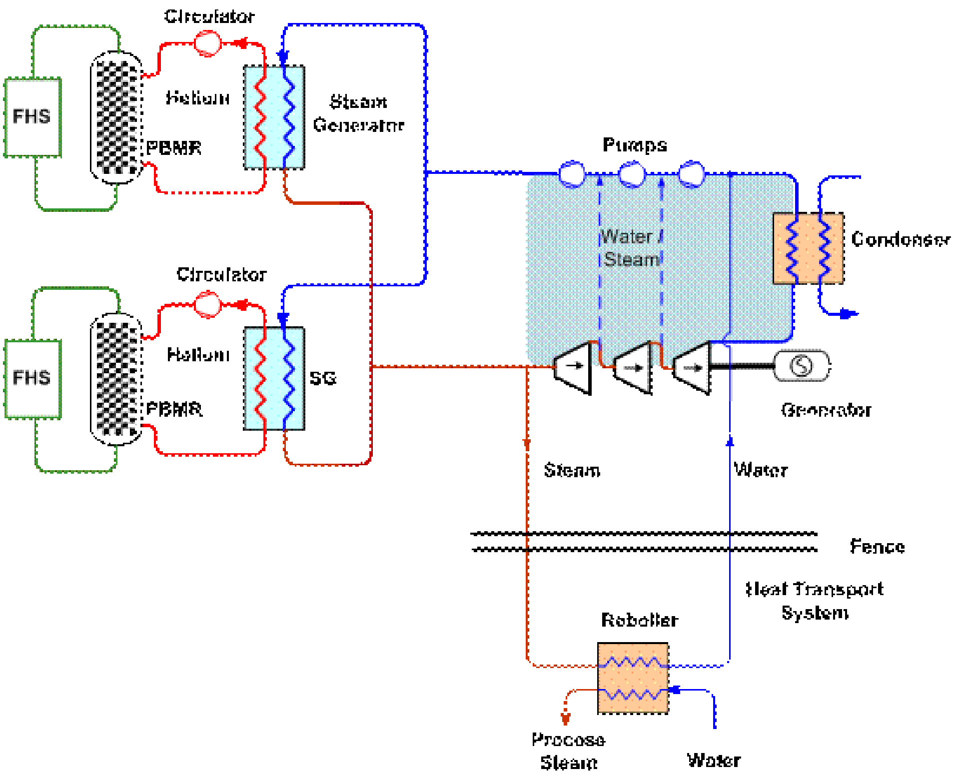


Figure 4: Simplified layout of PBMR- CG plant.

Electrical and I&C systems

The electrical supply system consists of a normal AC system based on the local standard of the client country and a DC system which has a battery backup with 24 hour supply for important safety systems and 2 hours for the main control system. Reserve diesel power systems will be part of the design but is dependent on local requirements and availability of backup grid connections. For the co-generation pants no supply to the grid is envisaged unless the process plant has no need for all the generation potential, in which case export via the transformer yard will be possible.

The I&C system has the following 3 subsystems

1. The plant control system which is a digital distributed system to control all normal plant operational parameters. It displays in the control room. For the standard two module plant there will be one control room with separate control desks for each module manned by an operator per module as well as a supervisor. Instrumentation signals are supplied through redundant and physically separated routes for enhanced safety and availability;
2. The Equipment Protection System (EPS) has a limited function to protect important equipment from damage,

it will override the control systems should limiting conditions be exceeded;

3. The Reactor Protection System (RPS) can override the other 2 systems. It is supplied from dedicated sensors and pre-amplifiers and buffered from the other systems that may make use of the same inputs but cannot influence the RPS. The main function of the RPS is to ensure safe reactor shutdown when certain limits are exceeded. The minimum parameters that are input into the RPS are the power levels (based on neutron fluxes from 12 detectors around the RPV), the outlet temperature of the coolant gas, moisture sensors in the gas inlet channels and reactor period measurements. Other RPS functions will include stopping the main circulator should moisture be detected and preventing start-up unless basic conditions are satisfied. Further functions may be added based on the safety analysis once all accident sequences have been fully explored.

Spent fuel and waste management

With the selected enrichment of 8%, the fuel burnup is about 82000 MWd/t, which is nearly double that of LWRs. This by itself already makes a large contribution to waste minimisation, although the amount of long lived radioactive nuclei in the fuel itself is, of course, higher. There is in principle no reason why even higher enrichments and burnups cannot be attained and some fuel in the USA and Germany have already been tested for very burnups and Pu particles have been burned in the Dragon Reactor to as high as 700 000 MWd/t [10] A modest goal that sticks to the maximum accepted enrichment of uranium to below 20% would be 200 000 MWd/t. This still requires many years of fuel development and qualification.

The way spent fuel will be managed for PBMR plants depends to a large extent on the policy of the host country. If reprocessing to obtain actinides is desired, the spent fuel will be stored on-site in large transport containers and removed to the reprocessing plant when wanted. However, as there is at present no reprocessing plant or acceptable proven industrial way of reprocessing HTR type fuel, this may not initially be a chosen solution. Due to the excellent fission product retention capabilities of the graphite spheres, especially for actinides, it is presently considered to be more optimal to store the fuel until final burial in a waste depository becomes possible. The disadvantages of the pebble fuel is the large volume of graphite that accompanies the small coated particles, making for high transport costs. However the advantage is low unit heat production which will make it possible to store the fuel within 10 years after use without overheating the storage medium, whether rock or clay. In the end, economics and local policy will dictate the chosen solution.

Besides spent fuel the reactor produces other types of radioactive waste. Wastes from HTRs have generally a different nuclide composition than those from LWRs, but the treatment is basically the same. What is different, is that because helium as a gas leaks easily from any container, there is a steady loss of helium for which the design value is that it should not exceed 0.1% of the inventory per day. For ease of calculation and because it is difficult to analyse all loss locations and systems, a conservative assumption is made that non condensable nuclides like noble gases, carbon 14 and a percentage of the tritium produced leak out together with the helium. This leakage is directly into the building and these nuclides cannot be filtered out. This then constitutes a daily release of radioactive waste that goes into the atmosphere and it is augmented by the Ar-41 produced by neutron irradiation of the air in the reactor cavity. The ventilation system of the reactor cavity is designed to have an average air retention time of 24 hours which allows for the major part of the Argon to decay, but it still constitutes a non-reducible source of activity. Together the calculated dose from these non filterable nuclides is well below any regulatory limit or EPA guideline. The rest of the nuclides will eventually reach an equilibrium between production and removal in the helium purification system. Tritium is combined with oxygen to form HTO and is condensed out and released with the coolant water. It may be possible to solidify this tritiated water, but it then creates solid waste which has to be stored in a waste disposal site at high cost and doubtful benefit. Operational data will indicate if the level of Tritium in the water justifies solidification of some sort. All other waste from maintenance and general activities will be treated the same as in present day operating reactors.

To realistically minimise C-14 production it is necessary to reduce the amount of nitrogen leaking into the system from maintenance and fuelling activities. Also it seems, based on the THTR experience, entirely possible to keep the helium leakage to 10% of the design value. This in turn will also reduce the production of H-3 through the He-3 + n = H-3 + p reaction which is the major contributor to the free tritium in the system.

Another source of waste is due to decommissioning. Activation of steel and other metal parts is a fact of life. Despite the low neutron fluence on the RPV, the amount of radioactive steel will, per MWe produced, be higher than that of LWRs because of the low power density and large vessel size. Another major source will be the graphite used as reflector material. This graphite contains radioactive nuclides due to irradiation of impurities, adsorption of materials, like Cs and Sr, that have a high affinity to graphite and, of course, C-14 and Tritium. It is feasible to refine this graphite and remove the impurities and perhaps reuse the resulting material. However the complications of such waste recycling in terms of cost and operator dose have not been quantifiable up till now and this remains an issue for the future.

Plant layout

The design of PBMR-CG and other variants allow for the placement of the main structures above ground, partly embedded or completely embedded whatever the local regulations require or what the local geology might impose. Figure 5 shows a design that is completely embedded, with the helium primary circuit below grade and all the main safety systems as well. Steam will be transported to above ground level end users, in the figure indicated by the turbine building. A steam line break is not a serious event as overcooling is unlikely and slow and all that is needed is for the control rods to be inserted.

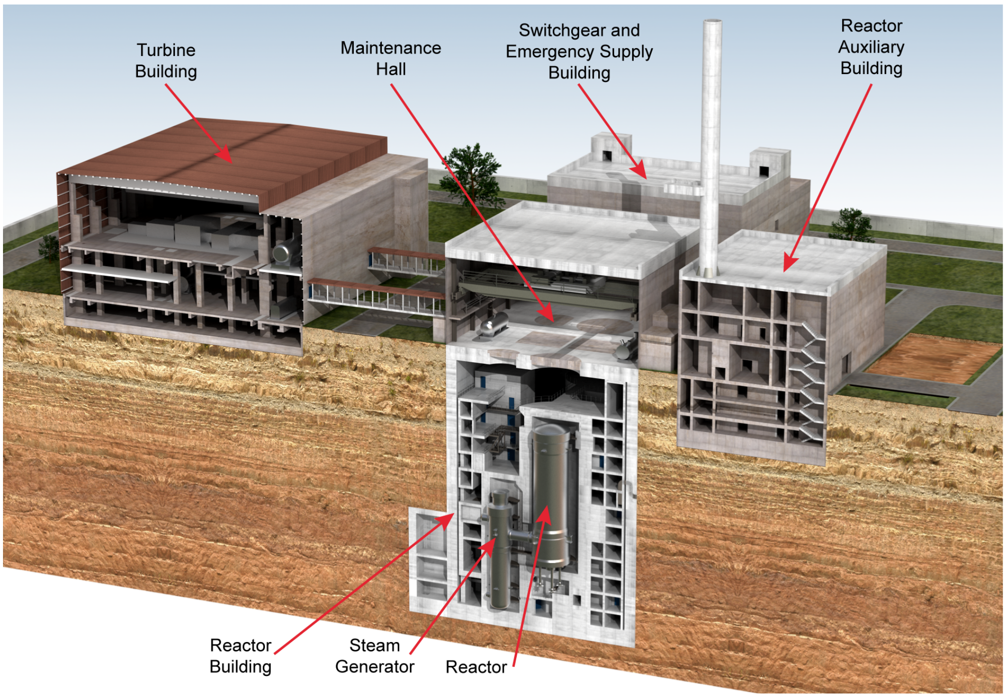


Figure 5: Depiction of the main plant located below ground level to ensure complete protection against missiles

Plant performance

Plant performance targets are based on 95% power operation for a lifetime of 60 years. In these 60 years provision is made for one SG replacement although based on the design and mild operating conditions a 60 year lifetime appears to be well attainable. In addition provision is made to replace the inner part of the side reflector once every 20 years, although 30 years appears to be an attainable figure based on the temperature and neutron fluence conditions at the surface of the reflector. Based on these assumptions the following targets have been adopted:

Reliability	One unplanned outage per year
Availability	95% with 2.4% for unplanned outages
Maintenance intervals	Every 5 years for turbine maintenance

Construction time n^{th} plant	36 months

The low construction time is based on completed site preparation work, modular factory construction of all major systems and concurrent construction wherever possible.

Fuel design and use depends on qualification and fuel development. For the present the same type fuel as used for the German programme is taken as the basis. Fuel costs are very dependent on quantity as coated particle and sphere fabrication are major cost items. Thus the enrichment will play less of a role in the fuel specification than a reduction in manufacturing costs. Higher enrichment also translates into higher burnup and this is the long term goal to reduce fuel costs and also reduce spent fuel quantities.

Development status of technologies relevant to the NPP

Due to the fact that the indirect cycle HTR has had many predecessors, there is not much new development that needs to be conducted. However as there was a hiatus in HTR built, some areas of know-how have to be recovered. These are particularly in the area of fuel manufacture and steam generator design. Since a number of years fuel particle manufacture on laboratory scale was resumed or newly instituted in the USA, South Africa and China in particular [8]. Using proven manufacturing techniques, fuel was produced and is being irradiated to high burnups. The next step up to a fuel factory requires the placement of orders for HTR power plants. Helical coil SGs, if this is the selected technology, were produced for the THTR and also for test beds in Germany. However new manufacturers will need to take the place of the previous suppliers who have discontinued work in these directions or have ceased to exist.

Graphite of the type used in the AVR and THTR cannot be made anymore as the raw material supply has changed. However an equivalent graphite has been developed in Germany and trial reflector blocks produced. Still needed is a graphite irradiation programme which depends on sufficient funding being available.

Future potential applications for HTRs are seen as high temperature heat for production of hydrogen or direct use in steel manufacture etc. Where the gas outlet temperature at present is 750 ˚C, which is well within present day technology, so-called Very High Temperature Reactors (VHTRs) need to produce gas at 950 ˚C and this will need development of high temperature steels and alloys not presently available or accepted in the nuclear technology. These developments need to be driven by governmental programmes before a viable VHTR project can be considered. Allied to this is the development by the chemical industry of a cost effective and sound process for producing hydrogen on an industrial scale using the high temperature process.

Deployment status and planned schedule

The most recent deployment date for the direct cycle DPP-400 design was to be 2014 with a module to be built at the Koeberg nuclear site near Cape Town in South Africa. Whether the indirect cycle design will be considered in any new nuclear construction in South Africa will depend on the South African government's plans for nuclear build to be decided upon later in 2010. In the meantime the PBMR company is developing the conceptual design for a process heat co-generation plant for the NGNP programme in the USA.

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Technical data

General plant data

Reactor thermal output	400 MWth
Power plant output, gross	165 MWe
Power plant efficiency, net	40 %
Mode of operation	Load follow
Plant design life	60 Years
Plant availability target >	95 %
Primary coolant material	Helium
Moderator material	Graphite
Type of cycle	Indirect

Nuclear steam supply system

Steam flow rate at nominal conditions	195 Kg/s
Steam pressure	12 MPa(a)
Steam temperature	540 °C
Feedwater temperature	200 °C

Reactor coolant system

Primary coolant flow rate	96 Kg/s
Reactor operating pressure	6 MPa(a)
Core coolant inlet temperature	250 °C
Core coolant outlet temperature	750 °C
Mean temperature rise across core	500 °C

Reactor core

Average core power density	4.0 MW/m^3
Fuel element type	Spherical
Outer diameter of elements	60 mm
Fuel residence time	10 Months
Average discharge burnup of fuel	92 MWd/Kg