PBMR Nuclear Design and Safety Analysis: An Overview

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Abstract

PBMR is a high-temperature helium-cooled graphite-moderated continuous-fuelled pebble bed reactor. The power conversion unit is directly coupled to the reactor and the power turbines are driven through a direct closed-circuit helium cycle. An overview is presented on the nuclear engineering analyses used for the design and safety assessment for the PBMR. Topics addressed are the PBMR design, safety and licensing requirements, nuclear engineering analysis results, software verification and validation, and advances in software development.

KEYWORDS: PBMR, Nuclear Design and Safety, Analysis

1. Introduction

The Pebble Bed Modular Reactor (PBMR) being developed [1] in South Africa is a high-temperature helium-cooled graphite-moderated continuous-fuelled pebble bed reactor. The power conversion unit is directly coupled to the reactor and the power turbines are driven through a direct closed-circuit helium cycle. The 400 MWt reactor design, which displays characteristics of Generation-IV reactors, has reached a high level of maturity. Excess reactivity is limited by continuous fuelling, while adequate passive heat removal ensures an inherent safe design with no accident event resulting in significant fission product release.

The nuclear engineering analyses to be performed in support of the PBMR design, plant operation and licensing spans across various disciplines ranging from reactor design, fuel performance, radionuclide and dust transport to radiation shielding and ex-core criticality.

This paper provides an overview of the reactor design and safety analyses performed for the PBMR design. Section 2 gives a description of the PBMR design and features. Section 3 provides an overview of the PBMR safety functions and licensing requirements, while Section 4 focuses on the nuclear engineering analyses with illustrative results. Section 5 provides a short summary of the analysis software, the validation and the future software development advancements that are needed to improve accuracies and to support future designs.

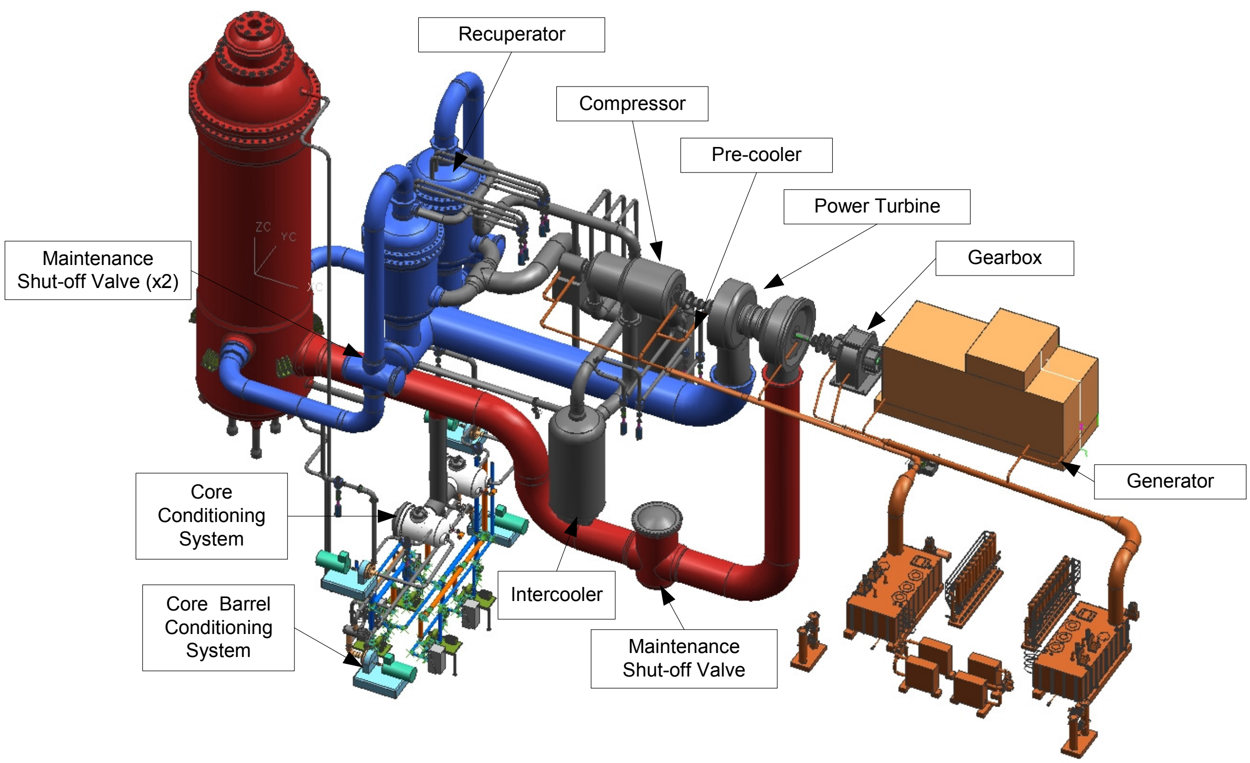
2. PBMR Description

The PBMR design has its origin in the German high-temperature nuclear reactor technology [2,3]. The most innovative part of the PBMR design is the integration of the reactor and the power conversion unit in a direct gas thermodynamic cycle configuration.

The PBMR core is designed for the low enriched uranium Triple-coated Isotropic (TRISO) fuel spheres that were developed in Germany from 1969 to 1988. More information on the PBMR fuel design and qualification is provided in Reference [4]. The PBMR core contains about 451 000 fuel spheres in an annular fuel region defined by a fixed central graphite reflector and an outer graphite reflector. The reactor is continuously fuelled through three loading tubes and three discharge chutes. The fuel spheres are circulated several times through the core until they reach the target burn-up. By adjusting the recirculation rate of the fuel spheres, the average number of passes through the reactor, before the fuel needs to be discharged to the spent fuel tanks, can be selected. Currently, an average of six passes is specified for the PBMR design.

The on-line fuelling allows a small excess reactivity that is just enough to overcome the xenon build-up for the power load-follow requirement. The reactor reactivity is controlled by two independent systems, namely the Reactivity Control System (RCS) and the Reserve Shutdown System (RSS). The RCS consists of 12 control rods and 12 shutdown rods in the side reflector and is used to control the reactivity in the core, for example during load follow, and to quickly shut the reactor under all operating conditions. The RSS consists of eight units that can insert Small Absorber Spheres (SAS) into the eight borings of the central reflector. The SAS are inserted to shut the reactor down to 'cold' conditions for maintenance operations. More information on the PBMR reactor design requirements and analyses are presented in [5]. The basic layout of the Main Power System (MPS), consisting of the Reactor Unit (RU) and the Power Conversion Unit (PCU) is illustrated in Fig. 1.

Figure 1: Basic layout of the PBMR Main Power System



The PBMR uses a direct integrated RU and PCU gas turbine thermodynamic cycle (Brayton cycle) to drive the turbine-generator. The PCU contains the power turbine, recuperator, pre-cooler, low-pressure compressor, intercooler and high-pressure compressor, along with piping and valves. The PBMR is designed for a rated thermal power of 400 MWt. The Power

Turbine (PT) runs at 6 000 rpm and drives the generator through a reduction gearbox, which reduces the generator speed to 3 000 rpm. The maximum rating of the turbine-generator unit is 175 MWe.

The reactor, the helium gas containing equipment of the PCU, and the spent fuel storage tanks are located within the reactor building on a seismically qualified nuclear island. The Reactor Pressure Vessel (RPV) is protected within a Reactor Cavity with a wall thickness of 2.5 m, manufactured from reinforced concrete. All of these structures together with the turbines are housed within the nuclear island, which is constructed of 1.5 m thick reinforced concrete to form the outer shell. The gearbox forms the separation between the generator and the rest of the PCU, with the generator being located in the building attached to the nuclear island.

By design, provision has been made to accommodate the storage of spent fuel in the Nuclear Island for the 40-year design life of the plant, and thereafter for a further period, if so required.

3. Safety Functions and Licensing Requirements

The PBMR design includes a combination of inherent and passive safety features to ensure the performance of safety functions and to prevent, limit, control, or mitigate the consequences of anticipated operational occurrences and postulated accidents. Compared to traditional Light Water Reactors, PBMR and other advanced High-temperature Gas-cooled Reactor (HTGR) designs rely more on inherent characteristics and passive design features that prevent significant fission product transport from the fuel, core, and reactor pressure boundary, and rely less on active systems that mitigate or delay radioactive material transport beyond the reactor pressure boundary to the environment.

The most important inherent characteristics of the PBMR which contribute to the fulfillment of the fundamental safety functions are:

1. A fuel and core design with a low excess reactivity and an overall negative temperature coefficient of ***reactivity*** sufficient to accommodate any foreseeable reactivity insertions during start-up and power operations without damage to the fuel.
2. A core design that ensures that post-shutdown decay heat removal is achievable through conduction, natural convection and radiation heat transfer, due to the core dimensions, low power density of the core and high thermal capacitance of the core structures. Peak temperatures remain below the structural design limits, and the fuel temperature is kept below the limit where serious degradation of the coated particles would lead to a significant activity release.
3. High-quality ceramic coated-particle fuel of proven design, which adequately retains its ability to ***confine*** radioactive fission products over the full range of operating and accident conditions.

In addition to these inherent safety characteristics, the PBMR design includes additional safety features to provide defense in depth for each of these fundamental safety functions.

The PBMR safety analyses necessary to demonstrate attainment of the safety objectives are based on requirements [6] from the South African National Nuclear Regulator (NNR), which is also fairly similar to the IAEA safety guide requirements [7]. The PBMR safety assessment is performed by a combination of both probabilistic and deterministic methods, and includes a selection of events where failures or combinations of failures could potentially have radiological consequences.

The NNR has defined the basic licensing requirements in two deterministic categories, namely Category A and Category B, and a probabilistic Category C. These categories are based upon the frequency of events. A third deterministic category is inferred which refers to the events which are beyond the Category B frequency limit. The design dose limits, based upon the consequences of the events, are also defined in terms of the categories. For the purpose of the safety analyses, all initiating events are classified into three groups, namely Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and events that are categorized as Beyond Design Basis Accidents (BDBA). The Beyond Design Basis events are analysed to show that there is no sudden and significant increase in dose when a change in accident frequency occurs. A summary of the licensing categories, the dose limits and events grouping are given in Tab. 1.

Table 1: PBMR licensing requirements and dose limits

Category	Description and Frequency	Public Dose	Worker Dose
A	<u>Normal Operation</u> Planned operation and maintenance <u>Anticipated Operational Occurrences</u> (AOO) Frequency > 10^{-2} / plant year Events where the plant continues operating after the event, or the fault is repaired without any need for damage analysis.	0.25 $mSv.y^{-1}$	20 $mSv.y^{-1}$
B	<u>Design Basis Accidents</u> (DBA) Frequency: 10^{-2} to 10^{-6} / plant year Events not expected during plant life, but must be designed for. Plant operation is not automatically assumed to recommence after the event.	50 mSv/event	500 mSv/event
Beyond B	<u>Beyond Design Basis Accidents</u> (BDBA) Frequency: <10^{-6} / plant year Events are analysed to demonstrate no cliff-edge increase in the dose		
C	All possible events that could lead to exposure		

Conservative analysis is used to demonstrate compliance with regulatory limits for accidents within the Design Basis, while Best Estimate (BE) analysis is used to demonstrate compliance with As Low as Reasonably Achievable (ALARA) targets and BDBA targets. Best estimate analysis is achieved by using best estimate inputs in best estimate models using best estimate assumptions. Conservative analysis is achieved by using either conservative inputs or best estimate inputs with uncertainty intervals in either best estimate or conservative models. When best estimate and uncertainty is used, a Monte Carlo analysis is performed, varying the inputs over their entire range.

The starting point for the deterministic and probabilistic safety analyses is the identification of an extensive list of initiating events for the AOO, DBA and BDBA groups. From these a bounding and representative set of Licensing Base Events (LBEs) is chosen for detailed deterministic analysis. For the PBMR, the dose requirements are relevant to the MPS leak and break events and releases from non-MPS systems only. Transients, for example, do not lead to a release of radioactive material from the Helium Pressure Boundary (HPB) for the PBMR.

4. Nuclear Engineering Analyses

The PBMR nuclear engineering analyses span across various disciplines, for example, reactor neutronics and thermal fluids, fuel design and fuel performance, radionuclide and dust transport, fuel source terms and material activation, radiation transport and shielding, radiation safety and radiological protection, and ex-core criticality. These analyses, in essence, evaluate the fulfilment of the design for all operational modes and licensing events with respect to the following requirements:

1. Reactivity Control
2. Heat Removal
3. Confinement of Radioactivity

In the following sections, analysis results are provided to illustrate the PBMR response for some of the safety functions.

4.1 Control of Reactivity

For the "Reactivity Control" requirement, the following must be demonstrated:

1. Reactor shutdown capability during operation and maintaining sub-criticality for "cold" conditions.
2. Effective core reactivity control during operation and the ability to do daily load-follow
3. Damped xenon oscillations for normal operations scenarios, such as load-follow, as well as other possible operator induced initiating conditions.
4. Inherently safe features of the reactor under all operating conditions and licensing events, for example, the passive reactor shutdown for total control-rod withdrawal (TCRW) with no operator intervention
5. Maintaining sub-criticality in fuel storage, which is for fuel storage before usage, during operation and for the spent fuel.

The PBMR design provides for two diverse methods to control the core reactivity. Firstly, the reactor has a negative temperature coefficient of reactivity over the entire operating range. Therefore, by opening the bypass valve, the helium inlet temperature increases, with a slight rise in the mean core temperature, and causes the reactor to become sub-critical. Secondly, the core reactivity is controlled by the insertion or extraction of the RCS rods.

During normal operation the RCS rods are partially inserted to a depth that provides for a small excess reactivity that is just enough to overcome the xenon build-up for the power load-follow requirement. The reactivity requirement for the core to perform 100–40–100% power load follow was calculated to be 1.4%Δk (or 1400pcm), thereby bounding the reactivity addition even in the case of a full control-rod withdrawal. If the load-follow requirement does not need to be fulfilled the RCS can be further withdrawn. This will lead to a lower excess reactivity and thus also a larger shutdown margin for the RCS. The excess reactivity also allows continual operation of the reactor at full power for more than a week without refuelling. During this time the control rods will slowly be stepped out to compensate for the loss of reactivity due to burnup, however, daily load follow will then not be possible.

The RSS must be capable of shutting the core down and keeping it sub-critical at a cold shutdown average core temperature of 100 °C. The RSS is designed to be able to render the reactor sub-critical under all normal operating conditions and for AOOs and DBAs that require no rapid changes in reactivity, and to keep it sub-critical in the long term. Tab. 2 presents the reactivity balance for the RCS at hot conditions and the RSS at cold conditions for 100°C and four days after shutdown. More information on the reactor reactivity coefficients and control is provided in Reference [5].

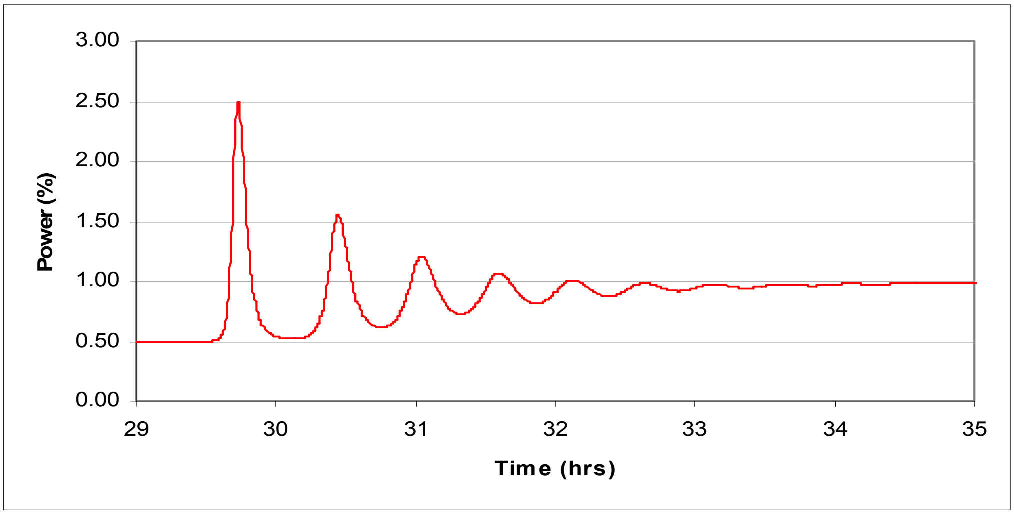
Table 2: Reactivity balance (%Δk) for RCS at Hot and RSS at Cold Conditions

Description	**Hot – RCS**	**Cold – RSS**
Normal Operation (RCS 2 m inserted, RSS out)	**-**	**-**
RCS extracted	+ 1.4%	+ 1.4%
Operating temperature down to 100 °C	n/a	+ 1.6%
Xenon and fission product decay (after four days)	n/a	+ 4.4%
Total requirement	**+ 1.4%**	**+ 7.4%**
24 RCS or 8 RSS inserted	- 8.9%	- 10.1%
Reactivity shutdown margin	**- 7.5%**	**- 2.7%**

Previously published work [8] gives a strong indication that the PBMR will exhibit xenon stability. Detail analyses [9] are also performed to confirm the xenon stability for the annular PBMR core. Also for the PBMR it can be concluded that the axial and radial xenon and power oscillations are strongly damped for several load-follow scenarios as well as for possible operator induced initiating conditions. The damped oscillations only cause relatively small temporary changes in the fuel temperatures and power densities without exceeding any design limits.

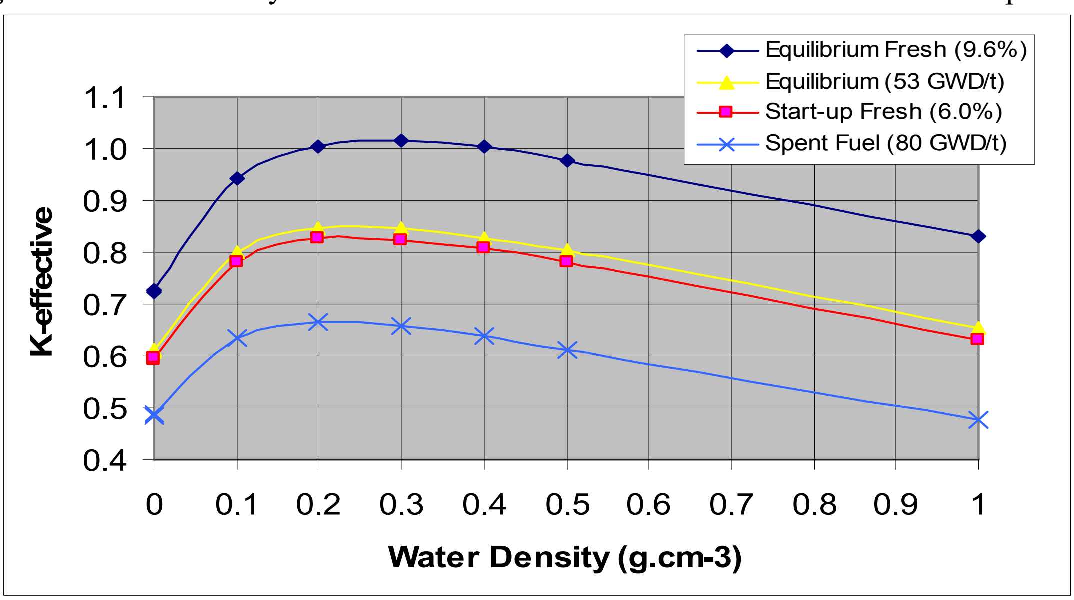
As part of the PBMR reactivity control safety assessment, various AOO, DBA and BDBA reactivity insertion events are analysed. These include, for example, overcooling, total control-rod withdrawal [10], inadvertent RSS removal, core compaction due to earthquakes [11], and top reflector drop. These events are modelled to demonstrate that they do not result in a helium pressure boundary failure or any radioactivity release. One of the TCRW cases is a “Pressurised Loss of Forced Coolant” case, where the PCU was tripped to the equalization pressure (70 bar) and 0 kg/s coolant mass flow rate reached within 5 seconds of the transient, followed by a “Total Control Rod Withdrawal” of all 24 RCS rods over a 200 second period (1 cm/s withdrawal rate). The reactor goes subcritical immediately after the PCU trip, due to the increased moderator (graphite) and fuel temperature and this feedback initially suppresses the effect of the TCRW and reactor power decreases. As the temperature then decreases again, and with the xenon decay, the reactor becomes re-critical after 29 hours. Fig. 2 [13] shows the power oscillation, as a percentage of full power, due to the re-criticality and the temperature increases and decreases.

Figure 2: % Power oscillations illustrating re-criticality after 29 hours



The reactivity control requirement is also applicable to the ex-core fuel storage. The main subcriticality requirement for the design of the used Fuel Tank (UFT) is that the UFT must be critically safe, containing all the fuel spheres from the PBMR core at any time of the PBMR operation. The approach for the UFT subcriticality design is based on maximum possible fuel load reactivity in the PBMR core, also taking credit for burnup, and is illustrated in reference [12] for both the start-up and equilibrium fuel loading scenarios. The UFT design is also evaluated for water ingress, loading of graphite spheres in-between the fuel spheres and for very conservative scenarios for clustering of most reactive fuel. Fig. 3 illustrates the UFT reactivity for the water ingress event when filled with different fuel. The analyses show that the UFT will also be critically safe for the equilibrium and start-up fuel in the event of water ingress.

Figure 3: UFT criticality for different fuel scenarios and water between the fuel spheres



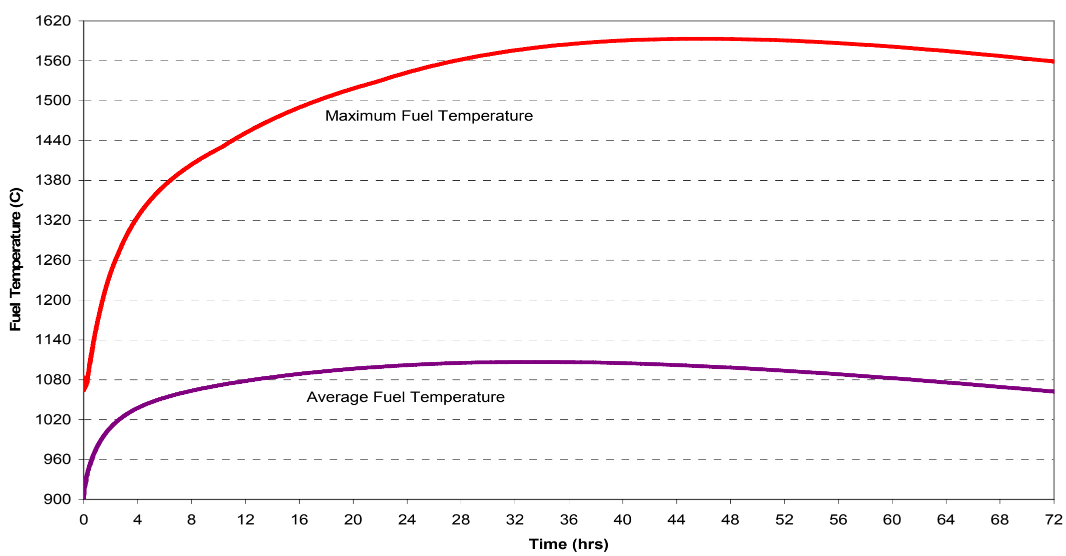
4.2 Heat Removal

The "Removal of Heat" requirement is to ensure that under all reactor operation conditions and events the heat can be removed from the reactor. The PBMR consists of two active heat removal systems, namely the self-sustained PCU-Brayton thermodynamic cycle and the Core Conditioning System (CCS). The passive heat transfer path from the core to the heat sink forms part of the passive heat removal capability that ensures sufficient fission product retention by the fuel even in the event of a total loss of forced cooling. Although the maximum fuel temperatures for excursions without active cooling do not exceed the radiological release limits, systems such as the CCS are designed as defense-in-depth to keep the core at normal operating temperatures during upset conditions.

For normal operation, one of the most important heat removal design requirements is to minimise the bypass flow within the core. The range of possible bypass flow values is bounded by analyses for best and worst cases. In the best case the assumption is made that the side reflector is completely sealed with the keys between the reflector blocks and that the gaps between the blocks are closed. For worst-case leakage the assumption is made that the keys are positioned in the centre of the key-gaps between the blocks (not touching any block) and the gaps between the blocks are for cold conditions (no thermal expansion).

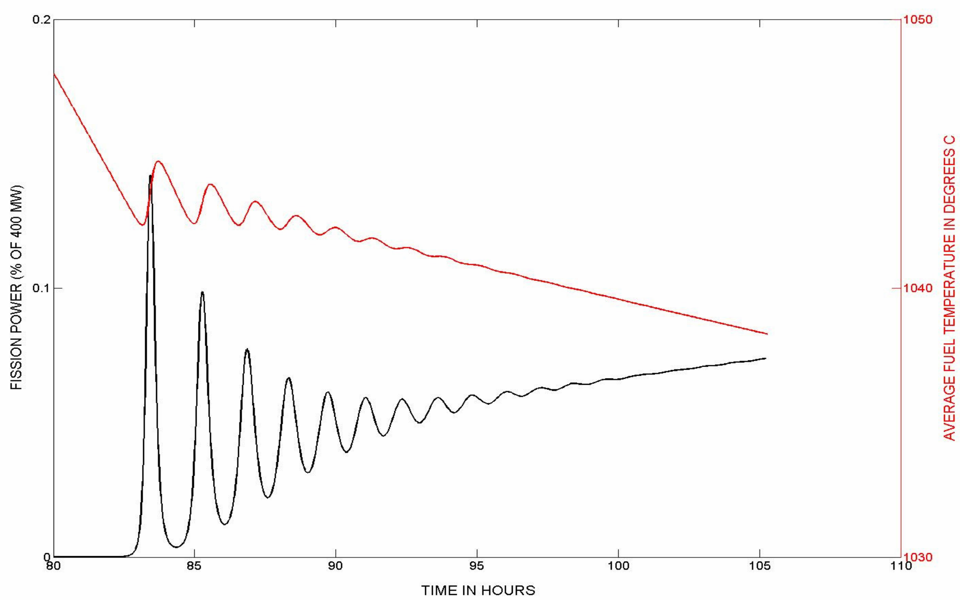
For a large pipe break, the DLOFC (Depressurised Loss of Force Cooling) analysis evaluates the passive heat removal of decay heat from the reactor under depressurised conditions. Fig. 4 shows the maximum fuel temperature and the average fuel temperature for a DLOFC event [13]. The event starts by simultaneously reducing the reactor inlet pressure from 90 bar to 1 bar and the inlet mass flow rate from 192.7 kg/s to 0.0 kg/s. A reactor scram is initiated at t=24 hours by inserting all 24 control rods over a period of 20 seconds.

Figure 4: Maximum and core-averaged fuel temperatures (^{o}C) for a DLOFC event



Should the control rods not be inserted, the reactor becomes re-critical after 83 hours. Fig. 5 [14] shows the average fuel temperatures and the power oscillation due to the re-criticality and the temperature increases and decreases.

Figure 5: Power and core-averaged fuel temperature oscillations due to re-criticality



4.3 Confine Radioactivity

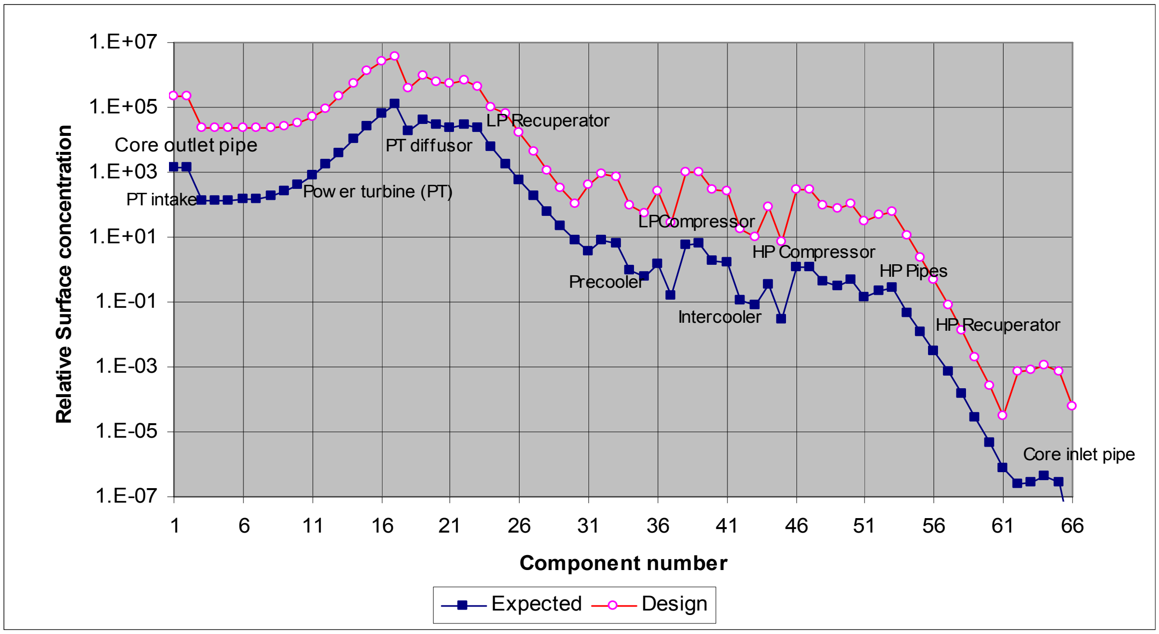
The "Retention of Radioactivity" requirement is evaluated by considering all possible sources of radioactivity and the processes by which radioactivity can be transported through the system. The transport of radioactivity can firstly be viewed as the transport of radiation, for example the neutrons and gammas that are transported through the whole plant originating at the reactor [15]. Secondly, the radioactivity is transported by radionuclides that enter the coolant having escaped from the fuel spheres.

As part of the fuel qualification it must be shown that only a very small percentage of fuel particles may have been manufactured with badly formed coatings from which the fission products can easily escape. Depending on the fuel particle temperatures certain fission products, in particular Ag-110m, can also diffuse through intact particle coatings. A design criterion, to restrict the best estimate maximum fuel temperature during normal operation to below 1130 ˚C, was therefore imposed. Other possible sources of radionuclides within the coolant are, for example, the activation of contaminants in the coolant or impurities in the graphite.

Some of the fission and activation products continue to circulate within the coolant while others plate out on surfaces. Abrasion of the fuel spheres and graphite reflector blocks generate dust and some of the radionuclides may adhere to the dust. A portion of the dust continues to circulate while the rest is deposited onto surfaces or within low-flow gaps or slits.

The transport of radioactivity is evaluated for both normal operations and various accident and licensing events. The normal operation analyses are performed for all operational modes and states, including maintenance. During normal operation a small percentage of helium leaks out of the HPB into the citadel (which the HVAC system is designed to filter) and released into the atmosphere. More information on the radionuclide transport and plate-out analyses and validation results can be found in Reference [16]. Fig. 5 shows for example, the Ag-110m plate-out distribution through the PCU after 36 years full power operation.

Figure 5: Ag-110m PCU Surface Concentration Distribution after 36 Years



In the event of a pipe break, the escaping helium transports the circulating activity generated during normal operation out of the pressure boundary, through the citadel and ultimately into the atmosphere. Depending on the break size and location some of the deposited dust may re-suspend and be transported out. If the break results in a loss of forced reactor cooling, the decay heat will increase the fuel temperature above normal operating temperatures resulting in a delayed release many hours after the break. Therefore, the passive "Heat Removal" design requirement also influences the "Retention of Radioactivity" requirement. Analyses are also performed in the pipe break events to evaluate the possible air ingress and degradation of the graphite by chemical attack.

Depending on the break scenario, a significant portion of the delayed release may be retained within the reactor and pressure boundary. Also, depending on the break scenario, a significant portion of the delayed release may be retained within the citadel. This is referred to as the citadel retention. For the worker dose-rate assessments it is assumed that the workers will be evacuated before the delayed release of fission products. The activity released from the citadel is transported to the site boundary for which public dose assessments are performed.

5. Analysis Software

The PBMR nuclear engineering analyses are performed using various code systems, many of them legacy software obtained from the German HTR programme. PBMR has therefore adopted a formal process to perform reverse engineering and verification and validation.

The main software currently being used for the PBMR nuclear engineering design and safety assessments are as follows:

1. VSOP99 (2- or 3-dimensional neutronics with 2-dimensional thermal hydraulics) [17,18]
2. TINTE (2-dimensional core thermal hydraulics, transients and 2-group kinetics) [17,19]
3. GETTER (metallic fission product release) [20]

4. NOBLEG (noble gas fission product release) [20]
5. SCALE (fuel depletion and ex-core criticality) [21]
6. MCNP (reactor neutronics, ex-core criticality, radiation transport and shielding) [22]
7. FISPACT (material activation) [23]
8. RADAX (radionuclide and dust transport and plate-out in the pressure boundary) [16]

Various international HTR benchmark efforts are also on-going, of which some are also reported on the PHYSOR-2006 conference. Validation results are also used from many experimental facilities from the German HTR reactor design programme, but also PBMR specific experimental facilities built or being built within South Africa.

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

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

Some of the method development and improved code system functions that need to be addressed are the following:

1. Option to do either approximate or detailed fluid dynamic analysis, both linked with the neutronics.
2. Updated neutron cross-section models and data, with a fuel sphere assembly code that includes improved spectrum and feedback effects.
3. Improved control rod cross-section model to be integrated within the code system.
4. Adding a theoretically sound method to calculate the non-local heat deposition in the reactor, important to determine the reflector temperature and cooling requirements.
5. Integrated fission product release model that provides release rates as part of the reactor design output.
6. Integrated Monte Carlo uncertainty analysis
7. Capability to specify different fuel-sphere packing fractions within the reactor that are correctly addressed by the fluid dynamics and automatically handled by the fuel shuffling management system.
8. Updated methodology to perform control-rod criticality searches
9. Evaluation of aspects such as more advanced homogenization techniques (to mix the different fuel batches occupying a volume), more advanced core neutronics solvers, such as nodal or simplified transport, or application of advanced reflector model techniques.

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