

The influence of the number of fuel passes through a pebble bed core on the coupled neutronics / thermal- hydraulics characteristics

by

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ABSTRACT

The increasing demand for energy and the effect on climate change are some of the big drivers in support of the nuclear renaissance. A great amount of energy is spent on studies to determine the contribution of nuclear power to the future energy supply. Many countries are investing in generation III and IV reactors such as the Westinghouse AP1000 because of its passive cooling system, which makes it attractive for its safety. The pebble bed high temperature gas cooled reactors are designed to be intrinsically safe, which is one of the main drivers for developing these reactors.

A pebble bed reactor is a high temperature reactor which is helium-cooled and graphite-moderated using spherical fuel elements that contain triple-coated isotropic fuel particles (TRISO). The success of its intrinsic safety lies in the design of the fuel elements that remain intact at very high temperatures. When temperatures significantly higher than 1600 °C are reached during accidents, the fuel elements with their inherent safety features may be challenged.

A pebble bed reactor has an online fuelling concept, where fuel is circulated through the core. The fuel is loaded at the top of the core and through gravity, moves down to the bottom where it is unloaded to either be discarded or to be re-circulated. This is determined by the burnup measuring system. By circulating the fuel spheres more than once through the reactor a flattened axial power profile with lower power peaking and therefore lower maximum fuel temperatures can be achieved. This is an attractive approach to increase the core performance by lowering the important fuel operating parameters. However, the circulation has an economic impact, as it increases the design requirements on the burnup measuring system (faster measuring times and increased circulation). By adopting a multi-pass recycling scheme of the pebble fuel elements it is shown that the axial power peaking can be reduced

The primary objective for this study is the investigation of the influences on the core design with regards to the number of fuel passes. The general behaviour of the two concepts, multi-pass refuelling and a once-through circulation, are to be evaluated with regards to flux and power and the maximum fuel temperature profiles. The relative effects of the HTR-Modul with its cylindrical core design and the PBMR 400 MW with its annular core design are also compared to verify the differences and trends as well as the influences of the control rods on core behaviour.

This is important as it has a direct impact on the safety of the plant (that the fuel temperatures need to remain under 1600 °C in normal and accident conditions). The work is required at an early stage of reactor design since it influences design decisions needed on the fuel handling

system design and defuel chute decay time, and has a direct impact on the fuel burnup-level qualification.

The analysis showed that in most cases the increase in number of fuel passes not only flattens the power profile, but improves the overall results. The improvement in results decreases exponentially and from ten passes the advantage of having more passes becomes insignificant. The effect of the flattened power profile is more visible on the PBMR 400 MW than on the HTR-Modul. The 15-pass HTR-Modul design is at its limit with regards to the measuring time of a single burnup measuring system. However, by having less passes through the core, e.g. ten-passes, more time will be available for burnup measurement. The PBMR 400 MW has three defuel chutes allowing longer decay time which improves measurement accuracy, and, as a result could benefit from more than six passes without increasing the fuel handling system costs.

The secondary objective of performing a sensitivity analysis on the control rod insertion positions and the effect of higher fuel enrichment has also been achieved. Control rod efficiency is improved when increasing the excess reactivity by means of control rod insertion. However, this is done at lower discharge burnup and shut down margins. Higher enrichment causes an increase in power peaking and more fuel-passes will be required to maintain the peaking and temperature margins than before.

Keywords:

High Temperature Reactors, pebble bed cores, in-core fuel management, multiple fuel passes, flux profiles, power profiles, temperature profiles, burnup measurement.

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ABBREVIATIONS

This table list contains the abbreviations used in this text.

Abbreviation or Acronym	Definition
ABWR	Advanced Boiling Water Reactor
BUMS	Burnup Measuring System
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
DLOCA	Depressurised Loss of Coolant Accident
DLOFC	Depressurised Loss of Coolant Accident
FHS	Fuel Handling System
HTR	High Temperature Reactor
THTR	Thorium High Temperature Reactor
LEU	Low Enriched Uranium
MEDUL	Mehrfachdurchlauf (multi-pass)
MIT	Massachusetts Institute of Technology
OTTO	Once Through Then Out
OTTO-PAP	Once Trough Then Out, Power Adjusted by Poison
PBMR	Pebble Bed Modular Reactor
PWR	Pressurised Water Reactor
RCS	Reactivity Control System
RIA	Reactivity Induced Accident
SAS	Small Absorber Spheres
TINTE	Time Dependant Neutronics and Temperatures
TRISO	Triple Coated Isotropic
VSOP	Very Superior Old Programme

1. INTRODUCTION

1.1 OVERVIEW

In the past few decades there has been a dormancy or decline in the nuclear industry due to major accidents at Chernobyl (Ukraine) and Three Mile Island (USA) and also the power plant construction cost overruns in the 1970's and 1980's. Globally the share in electricity has remained constant at around 16% since the mid 1980s. The output of nuclear reactors increased to match the growth in global electricity consumption.

Today, however, there is a revival in support of nuclear power and drivers such as increasing energy demand, climate change, increasing fossil fuel prices and security of supplies enhance nuclear popularity [1].

According to a study done by the Massachusetts Institute of Technology (MIT), nuclear power is a carbon-free source that can potentially make a significant contribution to future electricity supply [2].

Of the most popular types of thermal reactors today are the Generation III Pressurised Water Reactors (PWR's), which are light water reactors operating under conditions of 290 °C to 325 °C at pressures of approximately 15MPa with an overall efficiency of between 32-33%. The other light water competitor is the Boiling Water Reactor (BWR), which produces steam at 290 °C and 7 MPa. The Advanced Boiling Water Reactor (ABWR) and Advanced Passive 600 MWe (AP600) PWR are considered to be Generation IV as they include simplifications and the addition of passive cooling systems during reactor accidents.

High Temperature Gas Reactor (HTGR) from General Atomic produces steam at approximately 540 °C and 16 MPa while it circulates Helium through its core at temperatures between 815 °C and 870 °C. Overall plant efficiency of high-temperature gas reactors is around 40%. The PBMR 400 MW design is a Generation IV high-temperature gas reactor as it is passively cooled during and after accidents. The operating conditions for this reactor are at temperatures between 500 °C and 900 °C and at a pressure of 9 MPa [3].

Other types of reactors available are the Canadian CANDU (Canada Deuterium Uranium) reactor and breeder reactors [4].

The most attractive concepts of modular High Temperature Reactors (HTR) are their inherent safety characteristics. The first modular HTR pebble bed design that presented these inherent safety characteristics was the HTR-Modul, from the Interatom Company in Germany in the late 1980's [5].

A pebble bed reactor is an HTR which is helium-cooled and graphite-moderated using spherical fuel elements that contain triple coated isotropic (TRISO) fuel particles.

The key to the success of the intrinsic safety lies in the design of the fuel elements that remain intact at very high temperatures, retaining fission products effectively, as well as the low power density of the core and its self-acting decay-heat removal ability. However, these features are challenged when the fuel reaches temperatures significantly higher than 1600 °C.

In order to maintain the safety principle for an HTR the following limitations need to be adhered to:

1. The fuel elements must remain below 1600 °C during operation and accident conditions, and
2. The core design must tolerate loss of coolant and loss of active decay heat removal.

These two conditions can be demonstrated during a depressurised loss of coolant (DLOFC) accident which proves to be the most challenging condition during the life of the plant [6].

On the other hand, for an optimal solution a balance should be obtained between the reactor core design and the safety constraints. One of these design considerations includes the in-core fuel management scheme.

The HTR-Modul designers suggested a 15-pass fuelling scheme [5],[7],[8] while the PBMR 400 MW design is based on the selection of a six-pass scheme [3],[9]. The number of passes refers to the average number of times a fuel sphere is circulated through the reactor (loaded at the top and unloaded at the bottom) before it reaches its target burnup and is thus finally unloaded from the core. Of course, this is only the average behaviour since some fuel will be circulated more and others less due to different core residence times in the various flux positions within the core, and the stochastic loading at the top.

In selecting the circulation rate (and thus the average number of passes) a balance should be obtained between operational requirements (fuel handling system design, burnup measurement times and accuracies, dust generation and so on) and an optimised core design with a flatter power profile and lower peaking factors. The axial burnup profiles need to be considered and, in

this work, different multi-pass cases are evaluated to achieve this condition. This is the primary focus of this study.

The general behaviour of the flux, power and temperature profiles for the HTR Module and other pebble-bed designs have been investigated in the past [6], [10] and [11]. In this investigation the comparisons between the Once-Through-Then-Out (OTTO) and different multi-pass cases are evaluated for the HTR-Modul and the PBMR 400 MW. This investigation goes beyond the previous studies since it addresses factors on power peaking, maximum fuel temperatures, fuel enrichment, discharge burnup and excess reactivity of the control rods for normal operating conditions. The maximum fuel temperatures during a Depressurised Loss of Coolant Accident (DLOFC) event are also included with the exclusion on the evaluation of fission product releases. The main focus is on the HTR-Modul with selected results shown for the PBMR 400 MW.

The number of control rods, their location and insertion depth is of critical importance not only to maintain the excess reactivity, thereby keeping the reactor critical, but it also as part of their function to execute a shutdown operation [12]. When the neutron-absorbing control rods are inserted deeper into the core, it results in a reduction of the neutron flux. On the other hand, a way to increase the neutron flux is by adding more fuel. When the core geometrical boundaries are fixed so that no more mass can be added, the fissile content can be increased by higher enrichments or the heavy metal loading. In [13] it was shown that by adjusting the heavy metal loading, the reactivity and by implication the peaking in the core (due to burn-up) can be reduced. The adjustment of the heavy metal loading to optimize the peaking is not always possible due to aspects such as the need for optimum moderation to limit reactivity effects in the case of a water ingress event. The focus is therefore on the design parameters as given and the adjustment of the heavy metal loading falls outside the scope of this work.

As this study's primary objective is concerned about the evaluation of the influence of different multi-pass cycles on the axial power, temperature and burnup profiles over the core, the secondary objective included in the evaluation, is the effects of the control-rod insertion depth and the different enrichments.

Ultimately the design decisions must be weighed up against economical considerations, eg. a more expensive fuel handling system can improve neutron economy.

1.2 NUCLEAR CORE ANALYSIS

The principle design activities in nuclear analysis when designing a reactor core can be grouped into three general areas [14]:

- The determination of core criticality and power distributions,
- Reactivity and control analysis, and
- Depletion analysis.

The core power distribution is of central importance to both the corresponding thermal analysis and fuel depletion studies of the core. The calculation of the core power depends on parameters such as core enrichment, moderator-to-fuel ratio, core geometry, the location and types of reactivity control and the fuel element design. With regards to thermal design considerations, the ratios of the peak-to-average power densities along with the axial core power profile are of concern as they allow for the determination of whether the thermal limitation on the core performance is exceeded for a given design.

To compensate for the excess reactivity contained in the initial fuel loading as well as the allowance for safe reactor operation, analysis must be performed to determine the amount of negative reactivity or control. The interaction of reactivity with the control rods must be studied for static and dynamic situations as it is necessary for designing the control rod withdrawal and insertion sequences during reactor operation. However, for this study only a partial contribution is made in this regard as only the steady state situations are evaluated.

During reactor operation the fuel composition will change as fissile isotopes are consumed and fission products are produced. The depletion or burnup analysis is the most important analysis (aside from reactor safety) since it indicates the economic aspect of the nuclear reactor. It is also closely related to fuel management where the optimisation of fuel loading, arrangement and reloading are considered for economical power generation within the design constraints.

Except for the exclusion mentioned above, all these aspects are included in this study and are discussed to a greater or lesser extent according to the study objectives.

1.3 GENERAL DEFINITIONS

This section provides definitions for some of the commonly used terms in this dissertation.

1.3.1 Criticality

In order to obtain a stable chain reaction the number of neutrons in the system must remain constant from one generation to another.

Criticality refers to the effective multiplication constant k being equal to one. The effective multiplication constant illustrates a system's ability to sustain a nuclear reaction. It can be defined as

$$k = \frac{v \sum_f \phi}{DB^2 \phi + \sum_a \phi} \quad (1-1)$$

where, $v \sum_f \phi$ is the birth rate of neutrons and $DB^2 \phi + \sum_a \phi$ the leakage rate (neutron loss) summed with the absorption rate for new fissions to occur [15].

In the case where $k < 1$, the system is sub-critical and the chain reaction will not be sustainable, while when $k > 1$, the system is super critical and the chain reaction will go beyond controllability if not prevented. When $k = 1$, the reactor is critical which means that for the neutron generation the neutrons being used are equal to those lost.

1.3.2 Reactivity

The reactivity, ρ , of a system is a measure of the system's deviation from the effective multiplication constant of one.

For a finite reactor the reactivity is defined as $\rho = \frac{k-1}{k}$ and is often quoted in the unit of “pcm”. 1 pcm is equivalent to a reactivity of 10^{-5} .

Positive reactivity is when the reactor is supercritical, $k > 1$ and ρ is positive. Negative reactivity is when the reactor is sub critical, $k < 1$ and ρ is negative.

1.3.3 Burnup

“Burnup” is defined as the energy generated in the fuel during the period of core residence for the total mass of the fuel [16].

$$B = \frac{1}{\rho} \int_0^{\tau} \bar{E}_f \cdot \sum_{f(t)} \phi(t) \cdot dt \quad (1-2)$$

Burnup is the total energy released in fission by a given amount of nuclear fuel (MWd), while specific burnup is the burnup per metric ton of heavy metal originally contained in fuel (MWd/t).

1.3.4 Enrichment

The fuel used in both the HTR-Modul and the PBMR 400 MW is uranium oxide (UO_2) [4], [3]. It consists of a fraction of fissile material mixed with a larger amount of fertile material. The mass fraction amount of fissile material within the mix states the enrichment amount. The mass fraction of the fuel is calculated as follows [17]:

$$f = \frac{rM_{ff}}{rM_{ff} + (1-r)M_{Nf} + M_{O_2}} \quad (1-3)$$

Where the parameter f is the mass fraction of the fuel in the fuel material in units of kg, r is the enrichment or mass ratio of fissionable to total fuel and M , is the isotope mass of ^{235}U for M_{ff} , ^{238}U for M_{Nf} or O_2 in case of M_{O_2} all in units of kg.

From this mass fraction of fuel, the weighted percentage [w/o] of fissile isotopes (^{235}U) is then referred to as the enrichment amount.

1.3.5 Excess Reactivity

Excess reactivity (ρ_{ex}) is the core reactivity that is present when all the control elements are withdrawn from the core. Large values of ρ_{ex} will imply longer core lifetimes, but at the expense of greater control requirements and decreased neutron economy [18].

In pebble bed reactors with on-line refuelling the excess reactivity is not required to overcome burnup and therefore much smaller and only used to overcome xenon build-up during load follow.

1.3.6 Shutdown Margin

The shutdown margin (ρ_{sm}) is the negative reactivity of the core present when all control elements have been fully inserted so that minimum neutron generation can occur.

It is also related to the rate at which the reactor power may be reduced during an emergency shutdown or “scram”.

Large negative values are favourable for reactor control [18].

1.3.7 Control Rod Worth

The power plant is designed to accommodate changes in power. Power changes result in core reactivity changes. The rate of change in power must be accommodated by a corresponding rate of change in reactivity $\left(\frac{d\rho}{dt}\right)$. The rate at which the control system can change reactivity with time is a function of both the control rod speed $\left(\frac{dh}{dt}\right)$ and the differential worth $\left(\frac{d\rho}{dh}\right)$.

$$\frac{d\rho}{dt} = \frac{d\rho}{dh} \cdot \frac{dh}{dt} \quad (1-4)$$

Control rod worth ($\Delta\rho$) is therefore the difference between the excess reactivity (when all the control rods are withdrawn) and the minimum reactivity (when all the control rods are fully inserted) [19].

$$\Delta\rho = \rho_{ex} + \rho_{sm} \quad (1-5)$$

With regards to reactor control, it is favourable when the $\Delta\rho$ values increase positively. Since control rod worth represents reactivity, it is often given in units of pcm or percentage.

1.3.8 Power Density

The maximum value of flux in a uniform bare reactor is always found at the center of the reactor. The power density is also higher at the center of the reactor as it is a function of the neutron flux. It is defined as:

$$P = E_R \Sigma_f \int_{-a/2}^{a/2} \phi(x) dx. \quad (1-6)$$

With the parameter P as the power in units W/cm^2 , E_R the recoverable energy of 200 MeV or 3.2×10^{-11} Joules and $\Sigma_f \phi(x)$ as the reaction rate, where Σ_f is the macroscopic fission cross-section $[\text{neutrons/cm}]$ and $\phi(x)$ is the flux $[\text{neutrons/cm}^2 \cdot \text{sec}]$.

The ratio of the maximum flux over the average flux throughout the reactor is an indication of the extent to which the maximum power density exceeds the average power density. The peak power density will therefore occur at the location of the maximum neutron flux in the fuel element. The power peaking factor is the ratio of the peak to average power densities in the assembly [20].

$$F_{pp} \equiv \frac{\sum_{fF} \phi_F(a)}{\sum_{fF}^{eff} \phi_{av}} \quad (1-7)$$

Where F_{pp} is the power peaking factor in units W/cm^2 , $\sum_{fF} \phi_F(a)$ is the reaction rate for the maximum flux [fissions/sec] and $\sum_{fF}^{eff} \phi_{av}$ the reaction rate of the average flux [fissions/sec].

1.3.9 Equilibrium Cycle

The equilibrium cycle concept is valuable in providing a point of reference for evaluating reactor performance. It does not exist in practise as it implies two conditions namely, that the operating requirements and constraints are invariant with time and secondly, that no unexpected disturbances take place. However, this concept is used in comparing alternate courses of action and can be defined according [19]:

When considering an autonomous system of the form

$$\frac{dx}{dt} = F(x, y) \quad (1-8)$$

$$\frac{dy}{dt} = G(x, y) \quad (1-9)$$

Then the critical point will be where $(x_\tau, y_\tau) \lim_{\delta x \rightarrow 0}$ are such that

$$F(x_\tau, y_\tau) = G(x_\tau, y_\tau) = 0 \lim_{\delta x \rightarrow 0} \quad (1-10)$$

So that the constant valued functions $x(t) \equiv x_\tau$, $y(t) \equiv y_\tau$ satisfy the equations of the autonomous system. The constant-valued solution is called the equilibrium solution of the system [21].

For the pebble bed reactor with the on-line refuelling, the equilibrium core can then be defined as that state the core is in for most of its operational life assuming stable full power operation for a long time. The equilibrium cycle is calculated by performing quasi-static cycle depletion calculations while simulating the fuel management, control rod positions and temperatures. The pebble flow is modelled by the flow channels defining flow paths (channel boundaries) and flow speeds (number of axial meshes). This step-wise calculation process is continued until equilibrium is reached, i.e. the results for each burn-up period is reproduced. This is normally found after 300 cycles, i.e. fuel shuffling steps (depending on starting conditions, model and core residence time).

The cycle achieved at this stage is accordingly termed the ‘equilibrium cycle’, and is characteristic of the largest part of the reactor operational conditions [3].

1.3.10 Residence time

The fuel residence time is the average amount of time the fuel element spends in the reactor [22].

$$\text{Fuel residence time [days]} = \frac{\text{Fuel burnup [MWd/T]}}{(\text{Specific power [kW/kg]} \times (\text{Capacity Factor}))} \quad (1-11)$$

$$\text{Specific power [kW/kg]} = \frac{(\text{Reactor thermal power [kW]})}{(\text{Total mass of fissionable material [kg]})} \quad (1-12)$$

1.4 MOTIVATION FOR THIS WORK AND STUDY OBJECTIVES

This study impacts the safety of the plant directly as it investigates the fuel temperatures which need to remain under 1600 °C in normal and accident conditions. It provides an accurate analysis required even at the early stage of reactor design since it influences design decisions needed on the fuel handling system design, defuel chute decay time and has a direct impact on the fuel burnup level qualification.

It is relevant in that there are still pebble bed technology developments such as the South African and Chinese designs that are similar to the HTR-Modul and the number of passes is central to the safety of the HTR-Modul core [4].

The primary objective of this study is to investigate the influences on the core design with regards to the number of fuel passes.

It also compares the HTR-Modul and its cylindrical core with the PBMR 400 MW design, with its annular core and fixed central column. Emphasis is placed on the influence of the selected target discharge burnup since localised peaking increases with increased burnup. The new contribution made to this field of study is the evaluation of the practical disadvantages of the fuel handling system dealing with pebble flow with an increasing number of pebble passes [6].

As a secondary objective, the core behaviour effects at different enrichment levels as well as the effect of control rod insertion depths during normal operation are also investigated as the latter affects the flux profile with a deeper insertion eliminating the advantage of more passes.

Fission product release evaluation is also a safety aspect that can be considered. However, to do it correctly requires detailed fuel performance calculations which are outside the scope of this study. The maximum fuel temperatures are however given for all cases and can be used as an indication of expected fuel performance.

1.5 DISSERTATION OUTLAY

Chapter 2 describes the theory and background for this study, while Chapter 3 distinguishes between the two reference cases, the HTR-Modul and the PBMR 400 MW which are used as the two reference designs for result comparison. Chapter 4 describes the modelling technique, the calculations as well as the analytical model used and points out the limitations of the HTR-Modul.

Chapter 5 evaluates and discusses the results on the analysis of the HTR-Modul, while Chapter 6 compares and discusses the findings of the PBMR 400 MW.

The study is concluded in Chapter 7 with recommendations for further work. It follows with a reference list and appendices which provide the VSOP code as well as more detail with regards to the results.

2. THEORY

2.1 INTRODUCTION

This chapter provides the theoretical background for the study performed. It discusses general information about high temperature pebble bed reactors and points out the importance of the enveloping safety case. It shows the basics of how the neutron flux (hence the power) and burnup are calculated and explains the principle of in-core fuel management and how it is affected by variations in the number of passes, control rod positions and different fuel enrichment. This chapter concludes with an introductory background to the software tool selected, which is followed with a detailed discussion in Chapter 4.

2.2 ABOUT HIGH TEMPERATURE REACTORS

The High Temperature Reactors (HTR) are defined by the design criteria regarding the core and its surroundings. The in-core consists only of fissile and fertile material together with a very weak absorbing moderator. An inert and gaseous coolant such as Helium is used and coolant outlet temperatures are achieved to accommodate a closed-cycle gas turbine power plant. The core reflectors are ceramic material which can operate at very high temperatures. The fuel should be distributed as uniformly as possible in the reactor to give more heat transfer surface and achieve higher power density in the fuel. It uses a good neutron economy [16].

A prominent distinction that is made between HTR types is the fuel management scheme followed, which is based on the fuel type being utilized. Certain reactors use prismatic block fuel elements which require a shutdown event to refuel, while others use spherical fuel elements (pebbles) that are circulated and refuelled online. The latter reactor type is referred to as a pebble bed reactor.

The first pebble bed reactor to operate was the AVR in Germany (1967) to test the high temperature gas reactor concepts [23]. The second reactor constructed and in operation was the Thorium High-temperature Reactor (THTR). The HTR-Modul was a reactor designed by Interatom in the late 1980's in Germany with a tall, slender core in a steel pressure vessel. This was the first reactor design that presented inherent safe characteristics [7]. The PBMR 400 MW is very similar to the German design. Table 1 presents a summary of the principal core neutronic and thermodynamic design parameters of pebble bed reactor designs [16], [24].

Table 1: Design Parameters Comparison of Different High Temperature Reactors

Parameter	PBMR	HTR-Modul	THTR	AVR
Reactor thermal power	400 MW	200 MW	750 MW	46 MW
Coolant	Helium	Helium	Helium	Helium
Reactor inlet temperature	500 °C	250 °C	250 °C	275 °C
Reactor outlet temperature	900 °C	700 °C	750 °C	950 °C
Mass flow rate	192 kg/s	85 kg/s	296 kg/s	13 kg/s
Operating pressure	9 MPa	6 MPa	3.9 MPa	1.08 MPa
Pressure vessel	Steel	Steel	Pre-stressed concrete	Steel
Reactivity Control System	24 control rods in the side reflector	6 control rods in the side reflector	36 control rods in the side reflector	4 control rods in side reflector noses
Reserve Shutdown System	8 channels in the centre reflector filled with absorber spheres	18 channels in the side reflector filled with absorber spheres	42 rods inserted into the pebble bed	-
Fuel type	Spherical LEU	Spherical LEU	Spherical HEU+THO	Spherical HEU/LEU
Fuel sphere diameter	60 mm	60 mm	60 mm	60 mm
Fuelling method	Multiple recycle	Multiple recycle	Multiple recycle	Multiple recycle
Number of passes	6	15	-	~7
Coolant flow direction	Downwards	Downwards	Downwards	Upwards
Pebble bed inner diameter	2.0 m	Not applicable	Not applicable	Not applicable
Pebble bed outer diameter	3.7 m	3.0 m	5.60 m	3.0 m
Pebble bed height	11.0 m	9.4 m	6.00 m	2.8 m
Volume of pebble bed	~84 m ³	~66 m ³	~125 m ³	~20 m ³
Number of fuel spheres	~452 000	~360 000	~675 000	~100 000
Mean power density	4.8 MW/m ³	3.0 MW/m ³	6 MW/m ³	2.6 MW/m ³
Mean fuel sphere output	~0.9 kW/FS	~0.6 kW/FS	~1.1 kW/FS	~0.5 kW/FS
Status	Being designed	Planned	Operated	Operated
Power conversion cycle type	Direct Brayton (gas cycle)	Indirect Rankine (steam cycle)	Indirect Rankine (steam cycle)	Indirect Rankine (steam cycle)

A typical layout of a pebble bed reactor core consisting of a reactor pressure vessel, a core barrel, reflectors, gas flow channels, control mechanisms and fuel pebbles are shown in Figure 1.

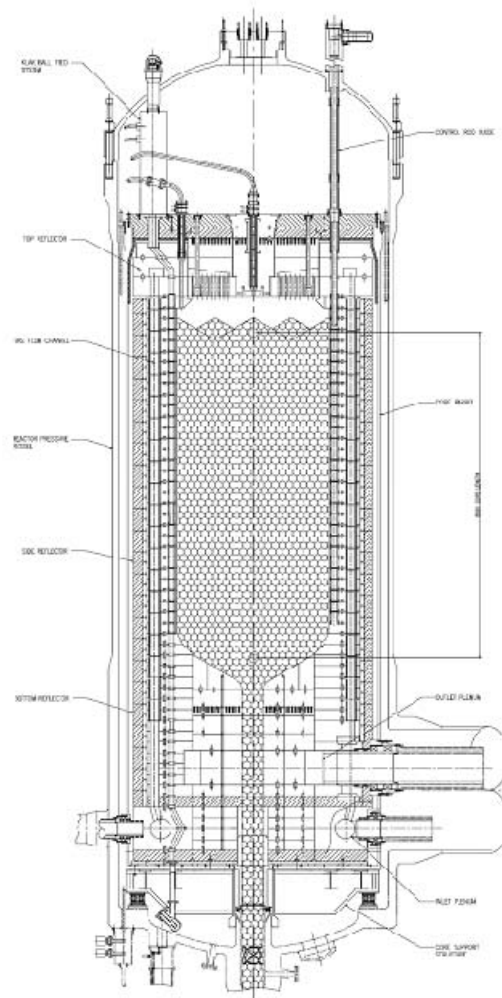


Figure 1: Typical Layout of a Pebble Bed Reactor Core

2.3 THE SAFETY CHARACTERISTICS OF HIGH TEMPERATURE REACTORS

The most attractive concepts of HTRs are their inherent safety characteristics.

The first modular HTR pebble bed design that presented these inherent safety characteristics was the HTR-Modul design, from the Interatom Company in Germany in the late 1980's [7].

A pebble bed reactor is an HTR which is helium-cooled and graphite-moderated using spherical fuel elements which contain triple coated isotropic (TRISO) fuel particles [25], [26] .

The key success factor for the intrinsic safety is the design of the fuel elements that remain intact at very high temperatures, retaining fission products effectively as well as the low power density of the core and its self-acting decay heat removal ability. However, these features are challenged when the fuel reaches temperatures higher than 1600 °C.

In order to maintain the safety principle for an HTR the following limitations need to be adhered to:

1. The fuel elements must remain below 1600 °C during operation and accident conditions and
2. The core design must tolerate loss of coolant and loss of active decay heat removal.

Thermo-hydraulic analysis performed on the HTR-Modul showed that the highest fuel temperatures are obtained during a depressurised loss of coolant (DLOFC) accident [7]. For safe reactor design, DLOFC conditions are to be proven within design limits.

2.4 DEPRESSURISED LOSS OF COOLING ACCIDENT (DLOFC)

A DLOFC accident caused by a pipe break in the primary loop can be considered as the worst case scenario for an HTR. A fast depressurisation of the core causes an absence of forced cooling, which results in a rise in fuel temperature. This has the effect of a large negative temperature reactivity feedback that diminishes the fission power quickly leaving decay heat on which the total power is then determined. Temperatures in DLOFC conditions are considerably higher than in normal operating conditions and maximum fuel temperatures are found in the region of the power peak. This peak can be reduced by the increase in the number of passes which will result in the decrease in fuel temperature.

In the work of Boer [6] it was shown that a ten-pass case will be advantageous with regards to maximum fuel temperature during a DLOFC scenario for the PBMR 400MW design (compared to the reference design using six passes). Radially the power profile showed peaks near the inner and outer reflector (in the case of a PBMR with an annular core). The radial distribution of fissile material can be influenced by creating several radial zones in the core in combination with a multipass recycle scheme. An increase in radial zones provides improved effects [6].

Although a DLOFC accident is much less severe for HTRs compared to the Pressurised Water Reactors (PWRs), it is still considered to be the most stringent design conditions for an HTR [27].

2.5 NEUTRON FLUX CALCULATION

Prediction of neutron distribution throughout the reactor system is required to design a nuclear reactor properly.

The neutron flux can be calculated by either applying Diffusion Theory or Neutron Transport Theory. Both these methods are applicable for the analysis of this study and are therefore discussed, but are not necessarily applied. The next paragraphs provide background and explanation of the respective theories, their application as well as reasons for one method being selected.

2.5.1 The Diffusion Theory

The Diffusion Theory is based on Fick's law first applied in the diffusion of fluids. Fick's law states that if a solution with a higher concentration in one region diffuses from highest to lower concentration, the rate of the solute flow is proportional to the negative of the gradient of the solute concentration.

Fick's law can be written as:

$$J(x) = -D \frac{d\phi}{dx} \quad (2.1)$$

where $J(x)$ is the net number of neutrons that pass per unit time through a unit Area to the x-direction. It has the same units as flux namely $[\text{neutrons}/\text{cm}^2.\text{sec}]$. The parameter D is the diffusion coefficient and has units in cm. D is given approximately by

$$D = \frac{1}{(\Sigma_t - \bar{\mu}_0 \Sigma_s)} \equiv \frac{1}{3\Sigma_{tr}} \quad (2.2)$$

where Σ_t , Σ_s and Σ_{tr} are the total, scattering and transport cross-sections respectively

and $\bar{\mu}_0 \approx \frac{2}{3A}$ is the average cosine of the scattering angle (A is the atom mass number of the scattering nuclei) [28].

Neutrons present at a certain time in a reactor core are described by the equation of continuity given as follows:

$$\int_V \frac{\partial n}{\partial t} dV = \int_V S dV - \int_V \Sigma_a \phi dV - \int_V \text{div}.J dV \quad (2.3)$$

The equation of continuity states that the rate of change in neutrons in volume V are equal to the rate of production of neutrons in V , less the neutrons absorbed in V and neutrons that leaked out of V .

The diffusion equation is the substitution of Fick's law in the equation of continuity. Through this, the following is obtained:

$$D\nabla^2\phi - \Sigma_a\phi + S = \frac{\partial n}{\partial t} \quad \text{where the symbol } \nabla^2 = \text{div grad is the Laplacian.}$$

Because flux is defined as $\phi = nv$ where n is the neutron density and v is the neutron speed, the equation can also be written as

$$D\nabla^2\phi - \Sigma_a\phi + S = \frac{1}{v} \frac{\partial \phi}{\partial t} \quad (2.4)$$

This principle is applied to energy-dependent neutron flux as neutrons in a reactor, to a good approximation, behave like a solute in a solution [29].

Diffusion theory provides a mathematical description of neutron flux derivation when the following assumptions are satisfied:

1. The flux is assumed to be sufficiently slowly varying in space that it can be approximated by expansion in a Taylor series about the origin.
2. The absorption is small relative to the scattering.
3. The neutrons are scattered isotropically.

The first condition is satisfied for most of the moderating and structural materials found in a nuclear reactor, except for the fuel and the control elements.

The second condition is satisfied a few mean free paths away from the boundary of large homogenous media with relatively uniform source distributions.

The third condition is satisfied for scattering from heavy atomic mass nuclei.

As nuclear reactors consist of thousands of small elements, many of them highly absorbing, with dimensions in the order of a few mean free paths or less, diffusion theory is used in nuclear reactor analysis with accurate predictions. It is complimented and supported by transport theory for accurate predictions in areas where it is expected to fail [28].

2.5.2 The Neutron Transport Theory

The transport equation is more sophisticated and more advanced than the diffusion equation and is based on neutron conservation and vector calculus.

It yields angular density $N(\vec{r}, \vec{\Omega}, E, t)$ and angular flux $\psi(\vec{r}, \vec{\Omega}, E, t)$ which provide an exact description of the neutron distribution.

The neutron transport equation is derived as follows:

$$\begin{aligned}
 \underbrace{\frac{1}{v(E)} \frac{\partial}{\partial t} \psi(\vec{r}, \vec{\Omega}, E, t)}_{\text{Time Variation}} = & \underbrace{-\vec{\Omega} \cdot \nabla \psi(\vec{r}, \vec{\Omega}, E, t)}_{\text{Leakage Term}} - \underbrace{\Sigma_t(\vec{r}, E) \psi(\vec{r}, \vec{\Omega}, E, t)}_{\text{Interaction Term}} \\
 & + \underbrace{\int_{4\pi} d\vec{\Omega}' \Sigma_s(\vec{r}, E' \rightarrow E, \vec{\Omega}' \cdot \vec{\Omega}) \psi(\vec{r}, \vec{\Omega}', E', t)}_{\text{Scattering Term}} \\
 & + \underbrace{q_{\text{extern}}(\vec{r}, \vec{\Omega}, E, t)}_{\text{external Source}} + \underbrace{\chi_p(E)(1-\beta) \int dE' v \Sigma_f(\vec{r}, E') \phi(\vec{r}, E', t)}_{\text{Prompt Fission}} \\
 & + \underbrace{\sum_{l=1}^6 \chi_d^l(E) \lambda_l C_l(\vec{r}, t)}_{\text{Delayed Fission}}.
 \end{aligned} \tag{2.5}$$

The rate in neutron production is equal to the loss of neutrons through leakage and neutron interaction and the addition of neutron gain by scattering, external sources, prompt fission and delayed fission reactions. A full description of this derivation can be found in [30].

Due to the heterogeneous nature of a reactor core numerical solutions are required as a one-speed equation cannot be used to solve the complexity of variation in fuel and other materials present analytically as can be done with an homogeneous reactor. The transport equation can be solved stochastically with methods such as Monte Carlo [31].

2.5.3 Selected methods and approach applied for the study

The one-group neutron diffusion theory treats neutrons as if they all have one effective speed, and it suppresses the effects associated with changes in neutron energy. In practice, this simplification would be justified if the cross sections were averaged over the applicable neutron energy distribution. For further simplification, the medium is initially assumed to be uniform [32]. It is also limited under the conditions as described in paragraph 2.5.1 in that it is not applicable

for fuel and control element calculations and it should exclude the boundary of a large homogeneous media.

The diffusion equation fails and does not provide exact results whenever direction dependence of angular flux is strong. Another assumption made by the diffusion equation is that all the neutrons can be characterised by a single speed or energy. This however is not true, as neutrons in a reactor have energies spanning the range from 10 MeV down to 0.01 eV and neutron-nuclear cross-sections depend rather sensitively on the incident neutron energy. This one-speed diffusion equation deficiency can be overcome by the multi-group diffusion approach.

Transportation theory is more sophisticated and advanced than the diffusion equation but is dependent on numerical and stochastic methods for calculations, which make it rather expensive with regards to computer processing.

Though the diffusion equation has its limitations, as a first approximation it can be accepted that neutrons diffuse through the reactor medium. The use of multi-group diffusion codes will also be applicable for the analysis of this study as it forms part of the initial calculations of the concepts design. The model to be used is two dimensional therefore assuming a weak angular flux and neutron density.

In the multi-group diffusion equation approach, one discretizes the neutron energy into energy intervals or groups as schematically shown below:

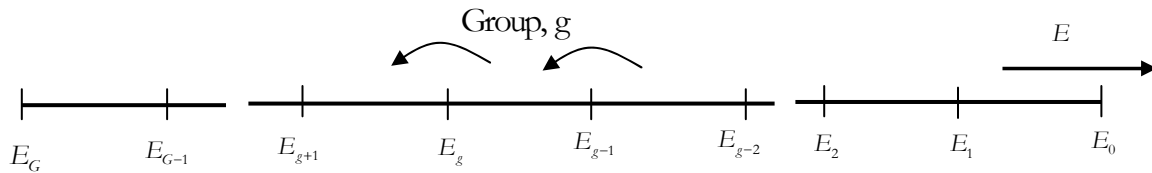


Figure 2: Energy group intervals

In practice two to 20 groups are used for reactor calculations. In CITATION (a sub-program of VSOP-99) a four energy group subdivision is made. For HTR studies a choice of four energy groups have proven to be sufficient [33]. The energy ranges are as follows:

- A thermal energy group with energies between 0 to 1.86 eV,
- An epithermal groups with energies ranging from 1.86 to 29.0 eV and another
- Epithermal group with energies between 29.0 and 1.11 MeV, and
- A fast energy group with energies between 1.11 and 10 MeV.

Few group calculations can only be effective with reasonable accurate estimates of the group constants. It is first calculated by performing a very finely structured multi-group calculation to determine the intragroup fluxes (relying on various models of neutron slowing down and thermalisation). The group constants for the fine spectrum calculation are frequently taken to be the tabulated cross-section data averaged over each of the fine groups. These intragroup fluxes are then used to calculate the group constants for a coarse group calculation [34].

This method of calculating a neutron flux spectrum and then collapsing the cross section data over the spectrum to generate few group constants is the most commonly used and is illustrated in Figure 3.

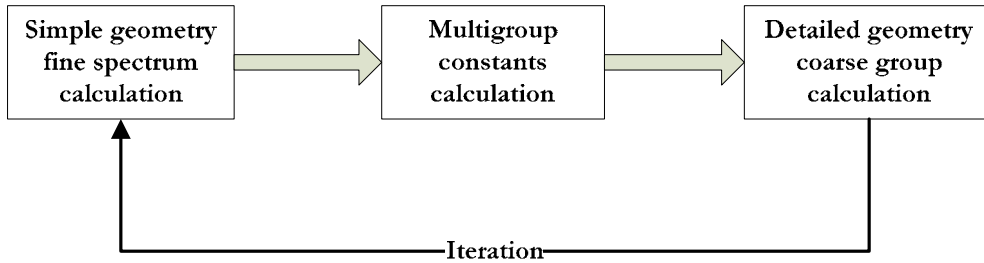


Figure 3: A coarse group calculation

In [33], it is explained that in VSOP-99 the transport equation is solved for the cell, yielding the cell flux $\phi(E,r)$. The neutron flux $\phi(E,r)$ is a function of energy, E and space, r . The integral over broader energy groups g yields the broad group cell flux $\phi_g(r)$ and the broad energy group macroscopic cross section $\Sigma_g(r)$:

$$\phi_g(r) = \int_g \phi(E,r) dE \quad (2.6)$$

$$\Sigma_g(r) = \int_g \Sigma(E,r) * \phi(E,r) dE / \int_g \phi(E,r) dE \quad (2.7)$$

Averaging in space yields the average cell flux $\phi_{av,cell}$ and the average cross section $\Sigma_{av,cell}$:

$$\phi_{av,cell} = \int_{cell} \phi_g(r) dr \quad (2.8)$$

$$\Sigma_{av,cell} = \int_{cell} \Sigma_g(r) * \phi_g(r) dr / \int_{cell} \phi_g(r) dr \quad (2.9)$$

with dr the 3-dimensional volume element. The reaction rate in the cell is achieved by the multiplication of $\Sigma_{av.cell}$ with $\phi_{av.cell}$.

The angular flux $\Psi(r)$, as the solution of the diffusion equation, is the “average cell flux at the position r ”, because by multiplying $\Sigma_{av.cell}(r) * \Psi(r)$ represents the reaction rate in the mesh. Therefore, the function $\Psi(r)$ represents the solution of the diffusion equation extended over the whole reactor.

To summarise, the average neutron flux are calculated by first obtaining the cell averages then determining the composition changes to determine the continuity of the spectrum zones.

In VSOP-99 a number of sub-programs are grouped together for approximating the transport equations in the epithermal and fast neutron energy range (GAM), while thermal energies are treated by the THERMOS code.

GAM-I performs neutron flux evaluation in 68 energy groups (with equal lethargical width) ranging from 10 MeV to 0.414 eV.

It uses the materials homogeneously distributed and applies the P1-approximation. Heterogeneity effects can be included by defining selfshielding factors as derived from other codes. Cross sections of the resolved and unresolved resonances are taken from ZUT.

For the spectrum cell calculation the THERMOS code requires information on how to form a cell configuration (1-D) and how to distribute the different nuclides in the cell.

The THERMOS code performs 1-dimensional cell calculation in 30 energy groups ranging from 0 eV to 2.05 eV.

2.6 BURNUP

The long-term changes in the properties of a nuclear reactor over its lifetime are determined by the changes in composition due to fuel burnup and the manner in which these are compensated.

The fuel composition changes due to various nuclei that transmute because of neutron capture and subsequent decay.

Fissile isotopes such as ^{235}U , ^{239}Pu and ^{241}Pu , are unstable as they are excessively neutron rich (they contain more neutrons than are necessary for their stability). As a result they decay by

and forms more fission isotopes (^{239}Pu). Generally in early core life, and depending on the initial fuel enrichment, the transmutation decay process produces more fissile nuclei than are destroyed, which causes a positive reactivity effect. This process continues until the concentration of transmuted fissile nuclei comes into equilibrium [37].

Over the lifetime of the fuel in the core, the fuel is affected by fuel depletion and compensating control actions, such as control rod movement and burnable poison.

Depletion of fuel will be greatest where the power is the greatest. The initial positive reactivity effect of depletion will enhance the power peaking. Later during the cycle, the negative reactivity effects will cause the power to shift away to regions with higher k_{∞} . Any strong tendency of the power distribution to peak as a result of fuel depletion must be compensated by control rod movement. However, the control rod movement to offset fuel depletion reactivity effects itself produces power peaking; the presence of the rods shields the nearby fuel from depletion and when the rods are withdrawn, the higher local k_{∞} causes power peaking. The effect is similar when using burnable poisons.

2.7 IN-CORE FUEL MANAGEMENT AND OPTIMISATION

Fuel management analysis is about the determination of fuel batches within the core to meet the safety, power distribution and burnup or cycle length constraints for a fuel cycle.

Fuel in the reactor can be modelled as several batches that have been in the core for different lengths of time. The number of batches presents the number of fuel loads that are circulated at a specific time in the core. On refuelling, the batch with the highest burnup is discharged while the remaining batches are re-circulated. The discharged batch is replaced online with a fresh fuel batch. This regime is referred to as a multi-pass recycling scheme.

More energy is extracted from the fuel when the power distribution in the core is as flat as possible, as peaking areas undergo faster burn out and causes higher temperatures. In general, for all types of reactors, different loading patterns have been considered for different reactor designs [38].

To increase fuel performance in high temperature pebble-bed reactors it is necessary to reduce the maximum fuel temperature. An attractive approach to achieve the decrease in the maximum fuel temperature is to reduce the power peaking.

By adopting a multi-pass recycling scheme (see Figure 5) of the pebble fuel elements it is shown that the axial power peaking can be reduced.

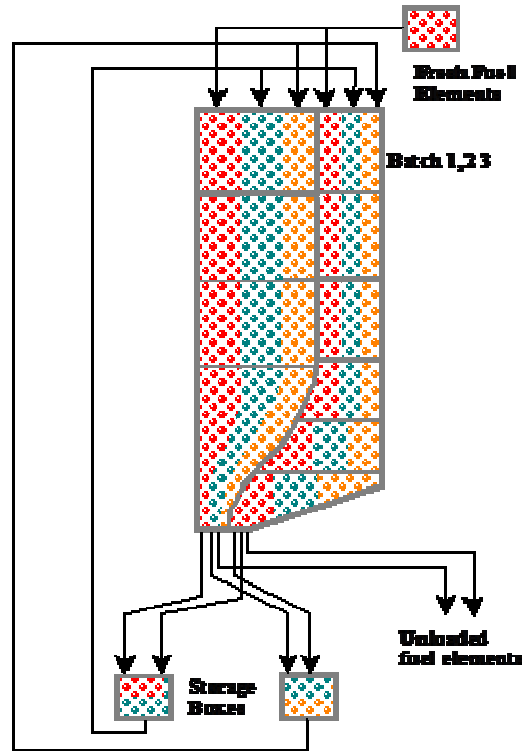


Figure 5: A Depiction of the Multi-pass Recycling Scheme

An optimal fuel loading pattern should also take radial shuffling (the so-called two-zone cores) into account since this can reduce the maximum power density even further. However, this reloading fuel pattern is beyond the boundary for the study at hand. Fuel temperature evaluations pointed out that with this optimisation technique decreases of up to 80 and 300 are obtained for normal operation and Depressurised Loss of Coolant (DLOFC) accident conditions, respectively.

When circulating the fuel elements only once through the core, an exponential burnup shape over the axial length of the core is achieved (Figure 6). With the increase of passes, the profile changes to a more cosine shape. This is due to a more even spread of fissile material over the core and the decrease in burnup difference. This causes the power peaking to shift closer to the centre of the core, also moving the associated maximum fuel temperature towards the bottom as shown in Figure 7. It has also been observed that the increase in passes beyond six passes is not advantageous with regards to fuel temperature [6].

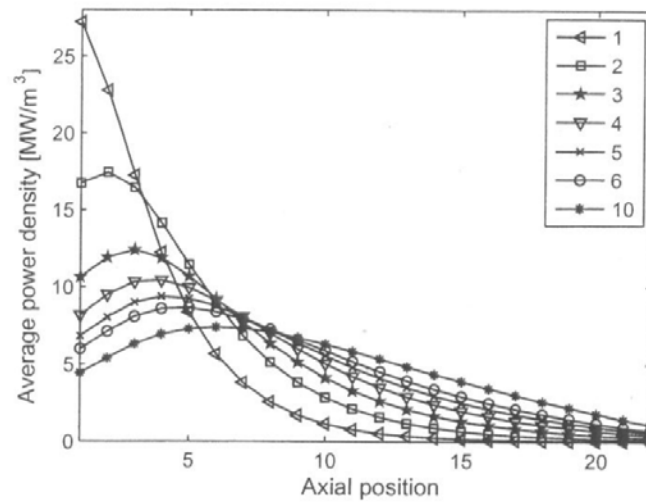


Figure 6: The axial power profile as a function of the axial position with the total number of fuel element passes as a parameter. [6]

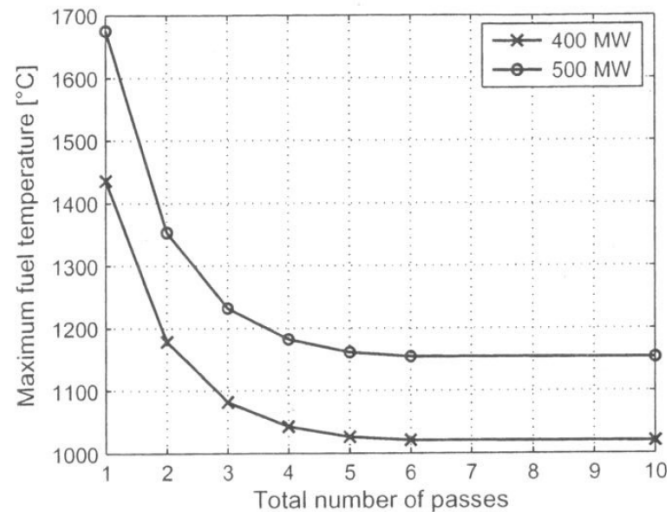


Figure 7: The maximum fuel temperature during normal operation as a function of the total number of fuel element passes for both 400 MW and 500 MW reactor power [6]

2.8 THE OTTO, OTTO-PAP AND MEDUL FUELLING PHILOSOPHIES

An alternative option to the Modul – also referred to as the MEDUL (Mehrfachdurchlauf, translated from German as “multi-pass”) – fuelling regime, is the Once-Through-Then-Out (OTTO) and the Once-Through-Then-Out, Power- Adjusted-by-Poison (OTTO-PAP) fuelling method [39].

With the OTTO method the fuel spheres passes through the core only once before being disposed. This method simplifies the Fuel Handling System (FHS) design.

During OTTO-PAP fuelling, fuel spheres are initially being added daily, on-line, in accordance with the fuel burnup. As the elements are discarded fresh fuel is added.

The Modul fuelling scheme allows for a continuous circulation of a fuel sphere until the fuel is depleted and then discarded. Fuel depletion is measured by a Burnup Measuring System that measures the ^{137}Cs gamma peak used to derive the burnup, above the background gamma noise [40].

In order to adhere to the passive safety characteristics of an HTR like the PBMR (which follows a Modul fuelling scheme), a sufficiently flattened neutron flux profile over the core is required as the power density profile is directly derived from the flux profile.

In the OTTO-PAP fuelling method a flattened flux profile resembling that of the Modul fuelling regime, is achieved by adding pre-calculated B_4C coated particles within a predetermined number of graphite spheres intermixed with the fuel spheres. The burnable poison content is optimised in the fuel design as a function of a specific core height and desired burnup. A higher power output together with an optimised overall fuel utilisation is achieved by this manipulation.

The advantages of the OTTO and OTTO-PAP fuelling scheme are simplification of the overall fuel management regime, operational ease offered by off-line defuelling and capital cost saving due to the simplified component requirements.

Previous studies have shown that the OTTO (without the burnable poisons) method is limited to a maximum thermal power output of $120 \text{ MW}_{\text{th}}$ due to its characteristic flux shape [39].

This states that this option is not viable for a $200 \text{ MW}_{\text{th}}$ reactor. However, the OTTO method is included as part of this study for evaluation and comparison.

2.9 EFFECTS OF CONTROL ROD POSITIONS IN THE PEBBLE BED CORE

The Reactivity Control System (RCS) or control rod function is to balance the excess reactivity by insertion, in order to keep the reactor critical. Excess reactivity is the additional reactivity available in the core during operating conditions. The purpose of excess reactivity is to allow flexibility in reactor operations such as the overcoming of xenon build up and is the result of a fresher fuel loading mixture than what is required to keep the reactor critical at operational conditions.

The control rods strongly affect the overall flux and hence the power distribution. The control rods can be used to shape the power distribution in a more favourable profile for core thermal performance and fuel burnup [41].

The insertion depth of the control rods (and therefore the excess reactivity) at normal operating conditions has to be managed to limit reactivity excursions and to ensure safe shutdown.

A postulated reactivity induced accident (RIA) which is considered by a nuclear regulator when evaluating plant safety is the ejection of the control rods. A nuclear power plant must be able to demonstrate that a reactivity transient and consequent energy release associated with the ejection of the highest worth of a control rod are within acceptable limits [19].

The bounding RIA is determined for each design based on a combination of deterministic and probabilistic analysis. Typically this would be the control rod group withdrawal or the maximum operational speed of a single control rod ejection (typically 1cm/s), as for the HTR-Modul.

The positioning of the control rods thus play an important role in defining the bounding RIA but also in normal operation and when determining accident maximum fuel temperatures.

The most significant parameter to determine the core reactivity, core temperatures and control rod worth are the position of the control rods. For the PBMR 400 MW a maximum insertion depth of 3.4 m was proposed, which gives an excess reactivity of 3.4% and still fulfils the 1% shutdown margin [12].

The maximum insertion depth of the control rods must hold for normal operation and allow flexibility for load follow, but then it must also be strict enough to assure cold shutdown.

2.10 EFFECTS OF FUEL ENRICHMENT ON THE CORE

Two primary determinants for the multiplication or criticality (see paragraph 1.3.1) on a thermal reactor core are the fuel enrichment and the volume ratio of moderator to fuel.

The fuel enrichment indicates the initial density of fissile isotopes available that are required for the fission chain reaction, hence the importance for the criticality calculations.

The fission reaction rate is one of the key parameters used to determine the neutron flux within the core. High fuel enrichment will result in higher neutron flux in the core, but to maintain the same thermal power, more neutron absorber ions are then required. [42]

2.11 SIMULATION TOOLS/ COMPUTER CODES

For this research project the main simulation tool identified for use is the VSOP-99 code system. It is a comprehensive numerical simulation of the physics of thermal reactors. It implies the setup of the reactor and of the fuel element, processing of cross sections, neutron spectrum evaluation, neutron diffusion calculation (in two or three dimensions), fuel burnup, fuel shuffling, reactor control, thermal hydraulics and fuel cycle costs (although the latter will be excluded for this research study).

As shown in Figure 8, the VSOP-99 code consists of two basic code sections namely the VSOP-ZUT and the VSOP-Main Section (or VSOP-MS). VSOP-ZUT calculates the fuel element design based on the user input data and sets up the resonance integrals, while VSOP-MS performs the core design calculations and fuel element design, and calculates the neutron spectrum, neutron diffusion, burnup and thermal-hydraulics [43].

All neutronic calculations from VSOP-99 can be applied, but for transient analysis, such as a Depressurised Loss of Cooling (DLOFC) event, the Time Dependant Neutronics and Temperatures (TINTE) code [44] is more applicable.

TINTE was developed to investigate the nuclear and thermal transient behaviour of high temperature reactors, with full neutron, temperature and xenon feedback effects taken into account in two-dimensional r-z geometry.

The code uses a one-dimensional leakage iteration method to solve the two-dimensional partial differential equations resulting from neutron and heat diffusion theory. The code can be used for a large variety of transient analysis with largely varying time constants. In this work the code was used to calculate the time evolution of the DLOFC accident with the focus on the fuel temperatures.

The core composition data generated from VSOP-99 will be used as the input data required for the TINTE model.

After simulation all the model data needs to be extracted from the VSOP report files and needs to be reworked. The reworking and additional calculations will be performed in Microsoft Excel.

3. HTR MODUL AND PBMR DESIGN DESCRIPTIONS AND COMPARISON

3.1 INTRODUCTION

The HTR-Modul core design, from the Interatom Company in Germany in the late 1980's, represented the first modular HTR pebble-bed reactor design with inherent safety characteristics [5]. The principle safety feature of this design concept is that even in the case of failure of all active cooling systems and the coolant, the fuel element temperatures remain within the limits at which there is virtually no fission product release from the fuel. In addition, the excess reactivity of the core is limited by the on-line refuelling, which minimizes the possible effects of reactivity-induced transients.

Two different modular pebble-bed core design concepts are included in this study. The first, a cylindrical core configuration, is based on the HTR-Modul design [5],[8] while the second is the annular core design with the fixed central column, the PBMR 400 MW [3] design.

The two different geometrical configurations were selected to investigate if the same conclusions could be drawn about the sensitivities of the selected parameters for the number of passes. Furthermore, the fuel enrichment, number and position of the control rods, and the gas inlet and outlet temperatures are different in the two designs. The two designs thus probably represent the bounding cases of small modular pebble bed designs.

Details of the two designs relevant to this study are given below.

3.2 DESCRIPTION OF THE HTR-MODUL DESIGN [5]

The HTR-Modul power plant consists of standardized reactor units with a power rating of 200 MW_{th} each, in order to maintain the inherent safety features of the small high-temperature reactors. The power conversion is via a helical tube steam generator with the helium coolant being circulated by the primary circuit blower.

The HTR-Modul design was used as the basis of the model of a cylindrical core of 3 m in diameter and a height of about 9.4 m. The use of a Rankine cycle (steam cycle) means that the coolant inlet/outlet temperatures are 250 °C/750 °C. The VSOP-99 core model is shown in Figure 9. Note that the core (in blue) is divided into five separate pebble flow channels. The number of axial meshes represents the number of burnup steps and fuel movements before the fuel is unloaded at the bottom. In this model fuel needs 15 steps to travel the core axial height in the inner three channels while 16 and 20 steps are required for the outer two channels,

respectively. This represents the slower movement of fuel close to the outer core boundary where fuel spheres interact with the stationary reflector graphite blocks.

In the model, six control rods were modelled in the outer reflector – the position is shown in Figure 9. The control rods are used to bind about 1.2% of excess reactivity needed to overcome the xenon build-up during the defined load follow capability of 100%-50%-100% power. This means that the control rods were inserted in the side reflector to a depth of about 3 m below the bottom of the top reflector or 2.5 m below the level of the top of the pebble bed.

The control rods are not full length and have an active absorber length of only 6.5 m.

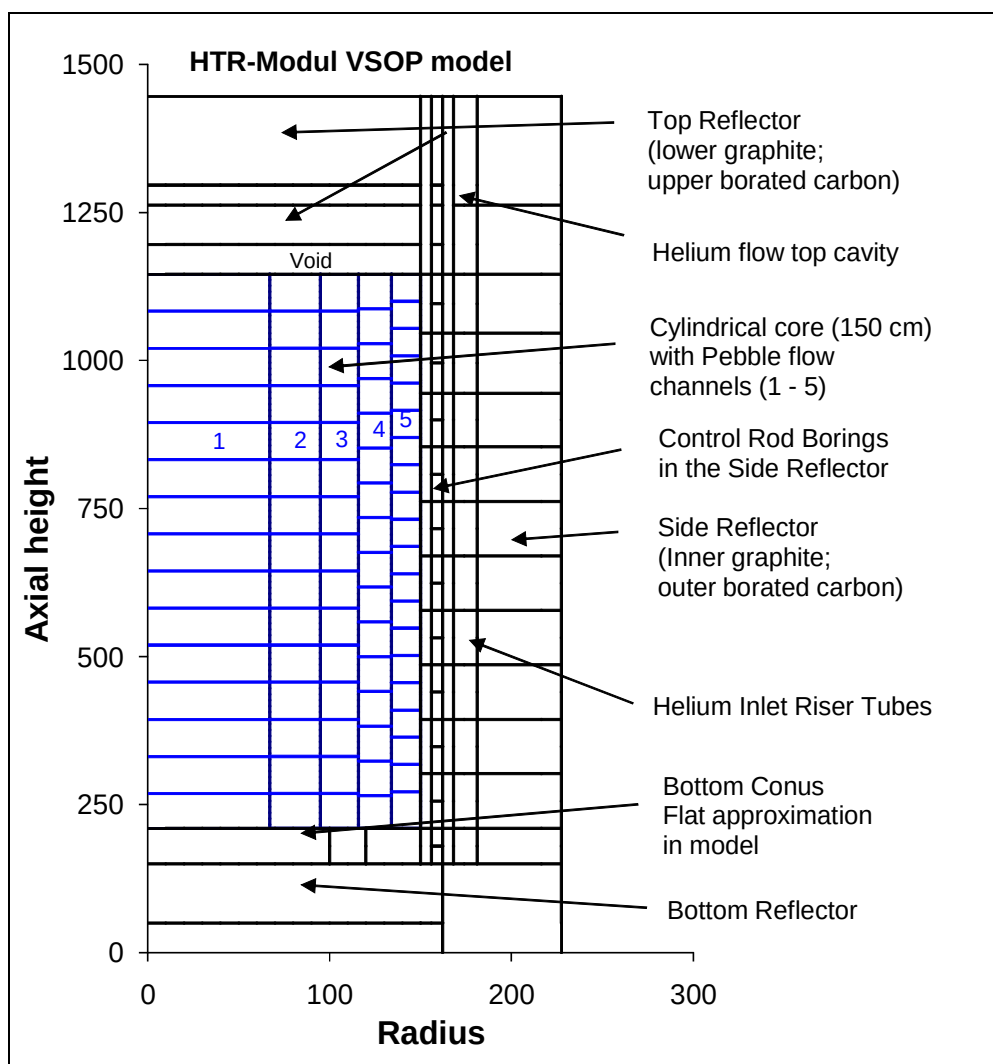


Figure 9: VSOP-99 core geometrical model (R-Z cut) for HTR-Modul [units in cm]

The core volume contains about 360,000 fuel spheres. One fuel element contains a total of 7 g uranium enriched to about 8%. The fuel design is the standard German design pebble fuel with a diameter of 6 cm with an inner fuel core that contains ~ 11600 TRISO-coated UO_2 kernels with

a diameter of about 0.5 mm. Each kernel is coated with layers of pyrocarbon (PyC) and a layer of silicon carbide (SiC), all within the fuel sphere graphite matrix.

The fuel is unloaded through a single defuel chute positioned below and in the centre of the core with a cone-shaped bottom reflector structure to channel the fuel spheres towards the chute. This is not included in the HTR-Modul VSOP-99 model where a flat bottom reflector was assumed. This approximation will not influence the relative effects of the different cases studied.

The defuel chute diameter and length determines the overall time in which fuel spheres can decay from where they leave the active core until they reach the core unloading device (at the bottom) and the burnup measuring station. The decay time is essential to ensure adequate decay of the short-lived fission product so that the background radiation levels are small enough for the specific ^{137}Cs gamma peak to be measured accurately. This aspect is addressed later. The HTR-Modul defuel chute therefore has a diameter of 50 cm and a total length of 11.5 m.

3.3 DESCRIPTION OF THE PBMR 400 DESIGN [3]

The Pebble Bed Modular Reactor (PBMR) concept can be described as a high-temperature helium-cooled, graphite-moderated pebble bed reactor with a multi-pass fuelling scheme. Coupled to the reactor is a power conversion unit, which comprises of a turbo-generator unit based upon a recuperated, direct, closed-circuit helium cycle (the Brayton cycle). The design is for 400 MW_{th} with a coolant helium inlet/outlet temperature of 500°C/900°C.

The design is for an annular core with an outer diameter of 3.7 m and an inner diameter of 2 m shaped by the fixed central reflector. The effective cylindrical core height is 11 m. This model is presented in Figure 10.

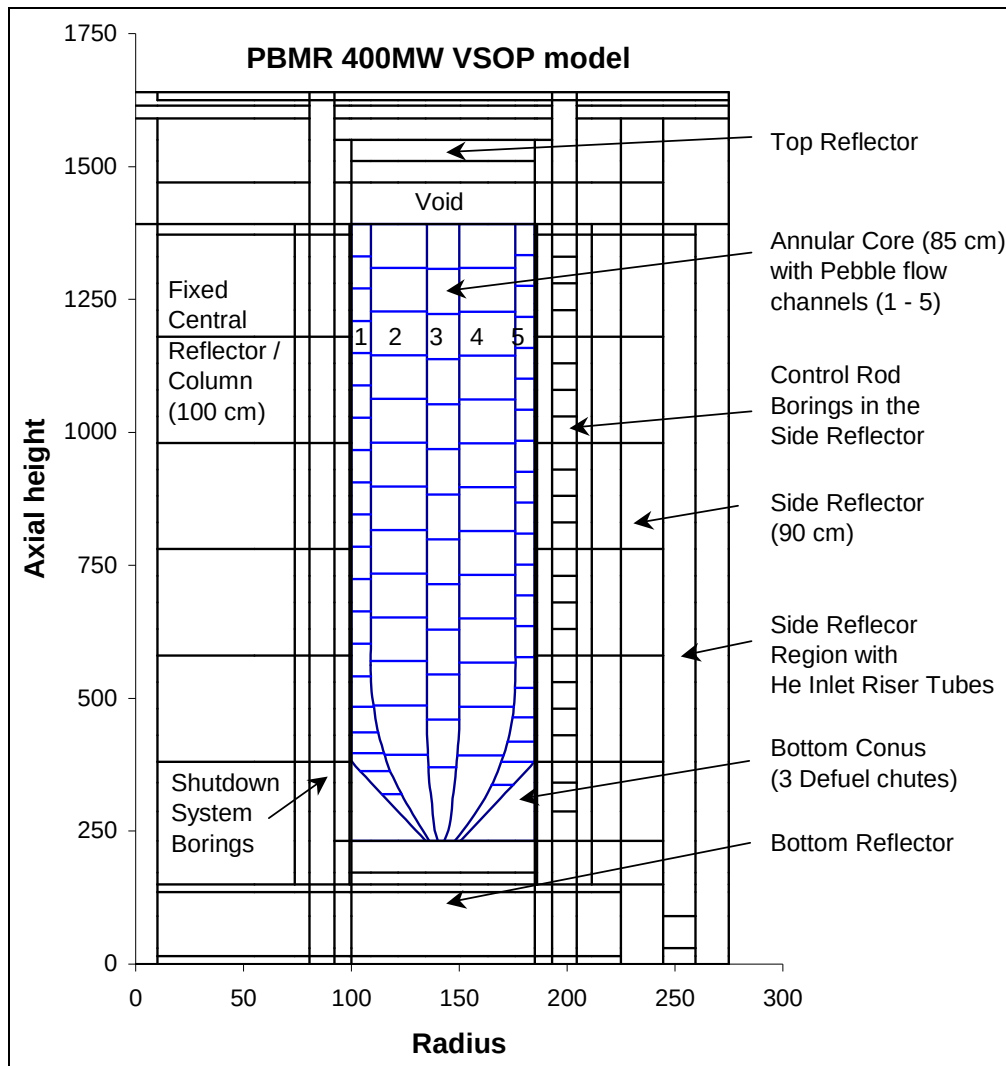


Figure 10: VSOP-99 core geometrical model (R-Z cut) for PBMR 400 MW VSOP Model [units in cm]

The Reactivity Control System (RCS) consists of 24 partial length control rod positions in the side reflector, with 12 upper and 12 lower rods, when fully inserted. The rods have an effective length (neutron absorbing material) of 6.5 m.

During normal operation all 24 control rods are inserted to the same insertion depth of 2.0 m below the upper reflector, thus an effective 1.2 m below the top of the pebble bed. This represents about 1.4% in excess reactivity needed to overcome the xenon build-up during the defined load follow capability of 100% - 40% - 100% power. The fuel design is the standard German pebble fuel (as for the Modul) but with ~15,000 TRISO-coated UO_2 kernels, 9 gram uranium loading and enrichment to 9.6 wt% U-235. Three defuel chutes are equally spaced on the annulus and are used to unload the fuel. This has the advantage of a longer decay time before fuel burnup is measured and also more time for burnup measurement at the three burnup measurement stations (with a cost disadvantage of needing three systems compared to one).

3.4 OVERALL DESIGN PERFORMANCE AND CHARACTERISTICS COMPARISON

A summary of the equilibrium results obtained from literature for the reference designs, are shown in Table 2. The HTR-Modul [40], has a burnup of 80,000 MWd/t with a core residence time of 1007 days. A fuel sphere has a Uranium-loading of 7g and a circulation time of 70 days. The average power density is 3 MW/m^3 , with a maximum power rating per fuel sphere of 1.4 kW and temperature of 865 °C. For the load follow requirement of 100-50-100% an excess reactivity of 1.2% is calculated to overcome the negative reactivity of xenon build-up. The overall leakage is about 13% and it has a conversion ratio of 0.47.

The PBMR 400 MW [3], [24], design has a 90 000 MWd/t burnup with a fuel sphere spending about 970 days within the reactor and doing a circulation in 159 days through the core. A fuel sphere is loaded with 9 g of Uranium and has a maximum power peak at 2.7 kW with a maximum temperature of 1130 °C. The excess reactivity for the 100-40-100% load follow is 1.4%. The neutron loss is 14.8% and the conversion ratio is 0.45.

Table 2: Overall design performance comparison at equilibrium conditions

Description	Units	HTR-Modul	PBMR 400 MW
Global equilibrium data			
Average residence time in core	Days	1007	970
Average discharge burnup	MWd/t	80 000	90 000
U235 Enrichment	w/o	7.8	9.6
Heavy metal loading per FS	g	7	9
Number of Fuel Spheres (FS)		360 000	420 000
Number of Fuel Passes (batches)		15	6
Time per pass	Days	70	159
Calculated thermal performance			
Average power density	MW/m^3	3.0	4.8
Maximum power rating of FS	kW/FS	1.4	2.7
Maximum fuel sphere temperature	°C	865	1 130
Calculated neutronic performance			
Requirement for 100 – 50 - 100% load follow for HTR-Modul and 100 – 40 - 100% load follow for PBMR 400 MW	$\% \Delta k$	1.2%	1.4%
Leakage (including absorption of the partially inserted control rods)	%	13.0%	14.8%
Average thermal flux ($<1.86\text{eV}$)	$10^{13} \cdot \text{cm}^{-2} \cdot \text{sec}^{-1}$	7.3 ($E \leq 2.38\text{eV}$)	7.8 ($E \leq 1.86\text{eV}$)
Average fast flux ($>0.1\text{MeV}$)	$10^{13} \cdot \text{cm}^{-2} \cdot \text{sec}^{-1}$	7.7 ($E > 2.38\text{eV}$)	3.3 ($E \leq 0.1\text{MeV}$)
Conversion ratio (ratio of production over loss of fissile nuclides)		0.47	0.45

The relative axial power profiles can be seen in Figure 11, which shows the typical behaviour of the 5th channel through the height of the core. These profiles are generated from the reference models used for this study.

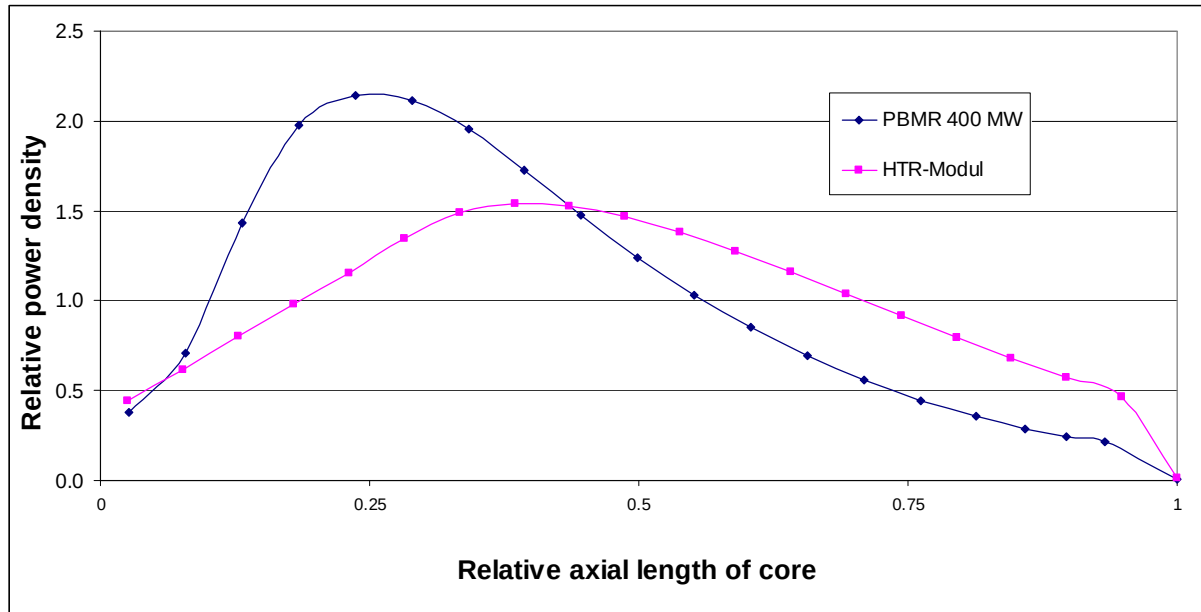


Figure 11: The relative axial power profiles of the reference models of the HTR-Modul and PBMR 400 MW

3.5 FUEL CHARACTERISTICS

As mentioned, both reactors, the HTR-Modul and the PBMR 400 MW, use the standard German pebble fuel design. The fuel spheres are 6 cm in diameter with a 5 mm graphite layer on the outside that serves as a clad and a moderator. In the center are the TRISO-coated particles imbedded in a graphite matrix. The TRISO-coated particles consist of three coatings namely Pyrolytic Carbon, Silicon Carbide and an inner Pyrolytic Carbon that covers the Uranium Dioxide (UO_2) fuel kernel. The configuration and layout of a typical fuel pebble is shown in Figure 12.

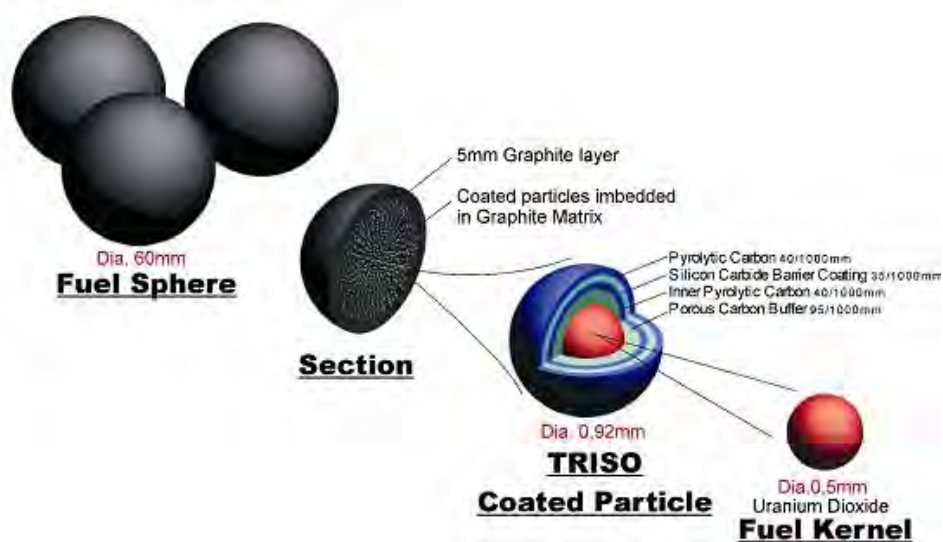


Figure 12: The pebble fuel configuration

Table 3 gives a summary with regards to the fuel design data. Although the average power per fuel sphere are higher for the PBMR 400MW than for the HTR-Modul, it is still well below the fuel design limit of 4 kW per fuel element.

Table 3: Summary of Fuel Sphere Characteristics

Description	HTR-Modul	PBMR
Fuel type	LEU	LEU
Fuel particles	UO ₂	UO ₂
Number of fuel kernel particles per pebble	11,600	15,000
Pebble radius	3 cm	3 cm
Thickness of fuel free zone	0.5 cm	0.5 cm
Density of graphite in matrix/fuel-free zone	1.74 g/cm ³	1.76 g/cm ³
U-235 enrichment of uranium (equilibrium fuel)	7.8 wt%	9.6 wt%
Weight of fuel loading	7 g	9 g
Number of pebbles in the core	360,000	420,000
Power output per pebble	0.6 kW	0.9 kW

4. ANALYSIS

4.1 INTRODUCTION

The same methodology was applied for the HTR-Modul and the PBMR 400 MW. The two software packages (VSOP-99 and TINTE) that were used in the study are described in the first section. This is followed by the detailed methodology applied for the different fuel-pass models and the selected sensitivity studies.

4.2 MODELLING TECHNIQUE

When performing a core design, certain decisions have to be made to fix the design. However, it is not always possible to visualise the impact of the trade-off between all these decision factors when trying to achieve an optimal design. Therefore, it is logical to do preliminary analyses on these decision factors to show the contribution and overall effect on the core design. Typical decision factors to consider are the power rating, core dimensions, fuel design, fuel management, target burnup, control rods, enrichments, reactivity and operating coefficients.

In order to perform such analyses a modelling tool with all the interdependencies coupled is required. These interdependencies are defined by theory based on physics with input data defining the fundamentals such as the resonance integrals and cross sections, the fuel element design and the neutron spectrum.

The modelling tool selected is VSOP-99 (see paragraph 4.3 and 4.4), which is a neutronic calculation tool for HTR pebble bed reactor design. It calculates the neutron diffusion, burnup, thermal hydraulics and fuel management.

Results are influenced by the input parameters defined by the user. The typical results of interest are the general results for the equilibrium cycle, the power distribution, the temperature distribution, the burnup measurement, the flux and fast neutron fluence, temperature coefficients, control and shutdown calculations, xenon behaviour, dust generation and fuel handling costs.

This approach, as indicated in Figure 13, was followed for this study. Sensitivity studies have been done to evaluate the fuel management criteria (incl. number of passes and target burnup), control rod insertion depths and enrichments. Results are compared based on the general equilibrium cycle, the power distributions (for normal and accident conditions), temperature distributions (for normal and accident conditions), burnup measurement (incl. the burnup time and decay time), flux and fast neutron fluence and reactivity control and shutdown calculations.

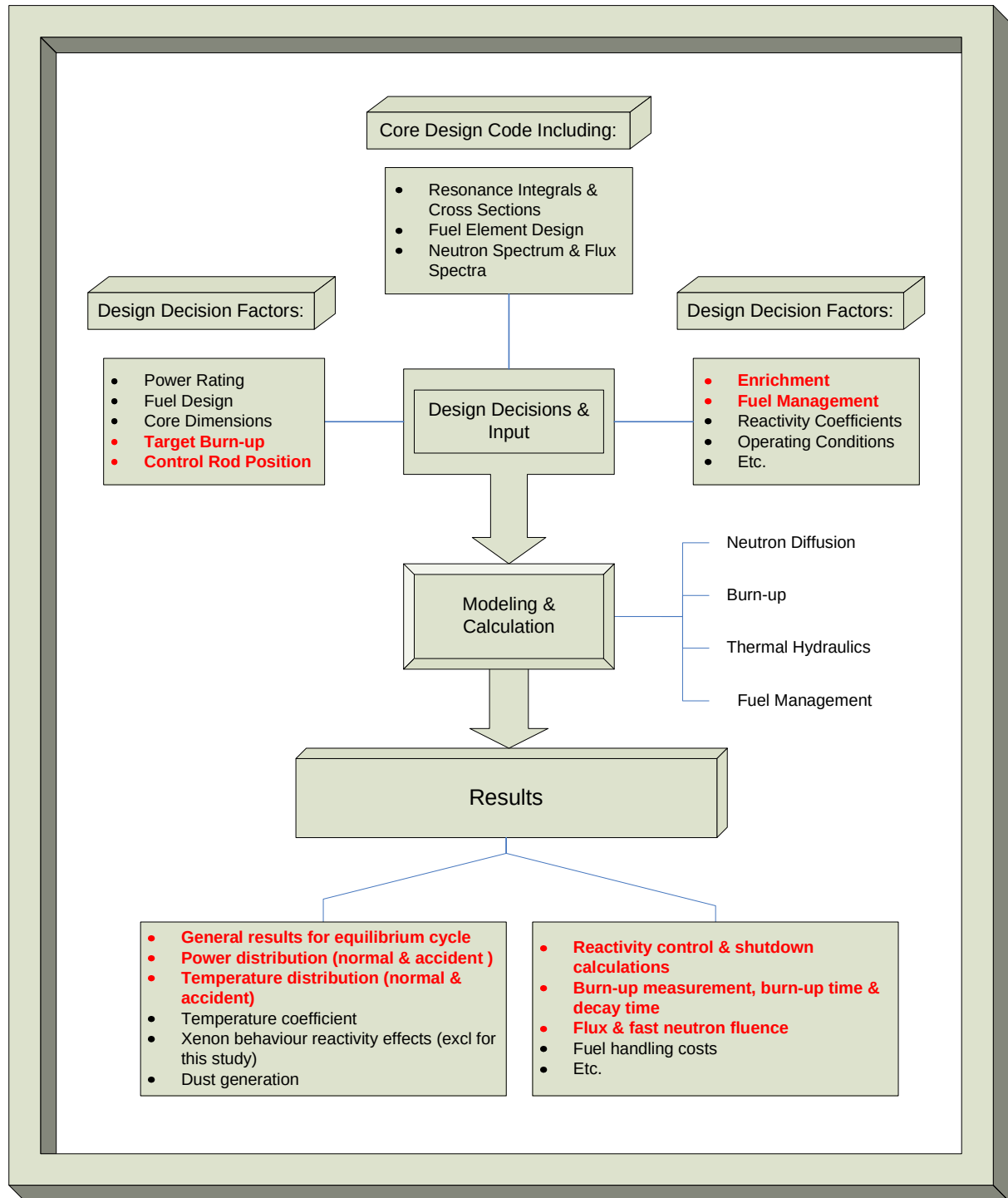


Figure 13: The study modelling concept

For this study three types of models were created in accordance with the objectives for both the HTR-Modul and the PBMR 400 MW:

- The different fuel-pass models;
- The different control rod position models; and
- The different enrichment models.

4.2.1 Different Fuel-Pass Models

Various cases from the OTTO up to 15 passes are assessed. The difference between the OTTO cycle and the multi-pass cycle suggested by the HTR-Module is in the number of times a fuel sphere circulates through the core before being discarded. In the case where fuel spheres pass through the core multiple times there is a better mixture between fresh and used fuel spheres, while in the case of the OTTO, no mixing takes place.

The models achieved equilibrium conditions as they were run to be critical at long-term hot full-power conditions with the fuel refuelling tempo balanced by the fuel depletion. The discharge burnup is therefore the criteria to be used to compare the fuel-pass models.

4.2.2 Control-Rod Position Models

The method of equivalent cross sections and equivalent boron concentration is used to model the control rods. In this study the control rod positions were adjusted to evaluate the difference on the burnup, the flux and power profiles as well as the shutdown margins. The rods were inserted 1 m deeper and then also removed by 1 m and then 2 m, respectively. As before, the equilibrium cycle is used as the basis for comparison

4.2.3 Enrichment Variation Models

In a final variation, the impact of different fuel enrichment scenarios on fuel management and the number of fuel-passes selected are considered. Comparisons are done for the equilibrium cycle with regards to the spread of discharge burnup, fuel temperatures, leakages and power peaking.

4.3 CALCULATION METHOD

The reactor unit is modelled using a two-dimensional cylindrical symmetry comprising the pebble bed, the reflector structures, the core barrel, the reactor pressure vessel and the control rods inclusive of helium or air, wherever present.

To model the fuel sphere circulation, multiple passes through the core as well as the varying speeds of the fuel spheres must be addressed. In VSOP-99 this is modelled in a two-dimensional way by first obtaining results from a detailed three-dimensional pebble flow model from other analysis or experimental data. By applying this method, the radial and axial trajectories are configured to present the results of the three-dimensional model.

For the radial trajectory the reactor core is divided into a number of radial channels, which span the core from the top to the bottom. With the axial trajectory the reactor core is divided into a number of regions. A fuel sphere is modelled to enter the first region of the channel and then in subsequent steps (called burnup steps) it moves to the next region below. By setting the number of regions in each channel to be different, the different axial trajectories can be modelled as having different speeds. The number of regions for each channel of the different reactor models, the HTR-Modul and the PBMR 400 MW are indicated in Table 4.

Table 4: Number of regions for each channel in the HTR-Modul and PBMR 400 MW model

Channel number	Number of regions	
	HTR-Modul	PBMR 400 MW
1	15	20
2	15	13
3	15	13
4	16	13
5	20	20

To account for the fuel isotopic concentrations of the multiple passes through the core, each region is subdivided into a number of batches. The fresh fuel is loaded into the first batch of the model. The fuel that has completed the first pass is then loaded into the second batch and so forth. In the case of the OTTO cycle, where fuel passes through the core only once, only one batch for each region is defined while a ten-pass needs ten different batches in the region.

In VSOP-99 the fuel is modelled to move down through the channels and the last-pass spent fuel batches are modelled to be discarded at the bottom of the core while batches with the same number (each presenting a pass) for each channel are then lumped together. The average burnup and isotopic concentrations of this joint fuel is stored in a storage box. This value is then used when it is reloaded for the next pass for each channel.

In VSOP-99 the neutron flux is solved by CITATION, a code using the multi-group two-dimensional steady-state diffusion equation with the method of inner iterations and power iterations. The diffusion equation is discretized in terms of space and energy.

The output from the CITATION results are the flux profiles for the groups considered, and the effective multiplication, k_{eff} . The power profile is then determined from the flux profile. The cross sections required for the CITATION calculations are calculated in the codes GAM-I and THERMOS.

The thermal hydraulic calculations are done with the THERMIX code, which calculates the mass and momentum conservation as well as the helium, solid material and, importantly, the fuel sphere radial temperatures.

The power produced per fuel sphere per fuel pass is determined from the CITATION calculations. Given the power produced per fission, the total number of fissions per fuel sphere can be determined, which then yields the burnup per fuel sphere.

The equilibrium core can then be defined as that state when the reactivity, burnup, flux profiles and temperature profiles remain constant for subsequent burnup cycles. In each case the model was set to achieve equilibrium status.

From this status the isotopic concentrations, fuel power history and temperatures were used as input as TINTE steady state conditions to perform the transient calculation for the DLOFC events.

4.4 CALCULATIONS

Calculations performed by VSOP-99 and TINTE are done on a Linux as well as a Windows platform. The calculation time for the VSOP analysis depended on the series of the .fil input files applicable for the analysis, and running time varied between 10 minutes and 4 hours. The TINTE analysis was done in less than 8 hours per run.

A total of 27 different VSOP-99 models, including the two reference models, were created. Table 5 gives a summary of the models.

Table 5: VSOP-99 models for the neutronic calculations

Analysis Study	VSOP-99 Model
Models created for HTR-Modul	
Reference:	200MW-ref
Number of passes study analysis:	200MW-1 batch
	200MW-2 batch
	200MW-4 batch
	200MW-6 batch
	200MW-8 batch
	200MW-10 batch
	200MW-12 batch
	200MW-14 batch
Control rod positions analysis:	200MW-10 batch_1m down
	200MW-10 batch_1m up

Analysis Study	VSOP-99 Model
	200MW-10 batch_2m up
Enrichments analysis:	200MW-6 batch_6%
	200MW-6 batch_10%
	200MW-10 batch_6%
	200MW-10 batch_10%
	200MW-12 batch_6%
	200MW-12 batch_12%
Models created for PBMR 400 MW	
Reference:	400MW-ref
Number of passes study analysis:	400MW-10 batch
	400MW-12 batch
Control rod positions analysis:	400MW-10 batch_1m down
	400MW-10 batch_1m up
	400MW-10 batch_2m up
Enrichments analysis:	400MW-6 batch_20%
	400MW-10 batch_20%
	400MW-12 batch_20%

All the models are based on the two reference models namely, the 200MW_ref and the 400MW_ref, which represents the HTR-Modul and the PBMR 400 MW respectively. TINTE transient calculations were done for the different pass study analyses and the enrichment analyses for the HTR-Modul.

A model consists of a variety of different input files programmed to follow a sequence of events presented in the .fil file. All models are based on the same methodology and sequence of events, therefore the following description is applicable to all the models.

The following data files were used as input files for a model:

P400dp3v103_resint_99_5	dimensions.dat
gam_99_4	adage
therm1515	
s.i	* lif.i
t.i	** cr-worth.i
e.i	s.fil
ee.i	t.fil
* tint.e.i	e.fil

ee.fil	* lif.fil
* tinte.fil	** cr-worth.fil

The following script or batch files are used as the executable file (depending on VSOP-99 being executed in Linux or Windows):

run-vsop.sh (Linux)
run.bat (Windows)

The following data files are created as the output files:

s.o	t.keff	ee.phiform
t.o	e.keff	t.powform
e.o	ee.keff	e.powform
ee.o	** cr-worth.keff	ee.powform
** cr-worth.o	t.phiform	* Nakura
s.keff	e.phiform	* Tinte

* Only applicable to models that were prepared for TINTE calculations.

** Only applicable to models that were prepared for the control rod positions.

Table 6 indicates the deviation from the reference model for the specific analysis models. It consists of a listing of all the parameters that are adjusted by means of user input manipulation. It is listed per input file, applicable VSOP-99 card and then the respective parameter that changed for the analysis case at hand. The parameters are named in accordance to the VSOP-99 Manual [43].

Table 6: Model deviations from the respective reference model

Analysis study	Input File	Card	Parameter
Number of passes	s.i	BI2	IBATCH
		V1	MBATCH
		V1	MSTOB
		V7	NCH5
		V7	MULT
		R21	IX1
		R21	NRX

Analysis study	Input File	Card	Parameter
		R21	NSB
		R21	R2
	t.i	R9	DELDAY
	e.i	R9	DELDAY
		R21	IX1
		R21	NRX
		R21	NSB
		R21	R2
	* lif.i	LF1	KMAX
		LF1	LMAX
		LF3	DT
Control rod positions	s.i	V7	NREAD
		V8	L
		V8	DEN
	t.i	R9	DELDAY
	e.i	R9	DELDAY
	** cr-worth.i	R9	XTDOWN
		R19B	IBAE(I)
Enrichments	s.i	D7	ANR
	t.i	R9	DELDAY
	e.i	R9	DELDAY

The models were prepared in the sequence indicated in the run-vsop.sh and run.bat files. The model results were obtained by changing the applicable parameters to presenting the conditions applicable to each case.

4.5 MODEL APPROXIMATIONS AND LIMITATIONS

There are some approximations and limitations made on the reference models.

Both the models are created to present a two-dimensional configuration of the core. This means that some of the three-dimensional spatial and other neutronic effects are simplified for a two-dimensional approach. An example of these effects is the modelling of the control rods.

The control rods are modelled using the method of equivalent cross sections and equivalent boron concentrations [45]. In the two-dimensional models used, the control rods are modelled by defining a grey curtain of boron-10 with a given concentration, in the reflector region. This concentration is calculated by creating a three-dimensional neutronic model in CITATION. The boron cross sections of the control rod regions are evaluated using a one-dimensional transport calculation (making use of XSRNPM, a module of the SCALE system in this case). Two

calculations are then performed in CITATION, one with the control rods inserted and another where the control rods are fully withdrawn. A resulting Δk_{CITA} can then be calculated.

Similarly, a second two-dimensional VSOP-99 calculation can be done. In this calculation the controls are also fully withdrawn, resulting in Δk_{VSOP} . By postulating that the reactivity effect (fully inserted *vs* fully withdrawn) of the control rods in both calculations should be the same, the boron concentration in the two-dimensional VSOP-99 is adjusted so that $\Delta k_{CITA} = \Delta k_{VSOP}$.

A limitation to the HTR-Modul model used, is that it doesn't make provision for the cone shape geometry of the reflector above the defuel chute. This approximation will not influence the relative effects between the different cases studied.

For the one-pass study which presents the OTTO cycle an approximation was made with regards to batch definition. THERMIX doesn't allow for a single batch calculation as it goes into an infinite calculation loop. This is due to a limitation in the software code. To overcome this obstacle the files were manipulated to create 2 batch cases with the second batch containing a negligible amount of fuel which would not influence the burnup values.

4.6 MODEL UNCERTAINTY

The respective reference models form the platform for all the other analysis models. Therefore, it is vital to evaluate the accuracy and representation of the developed models. The original HTR-Modul models were predominantly created in ZIRKUS [40], which is an HTR program that can be used to design single-cycle or multiple-recycle cores, from initial to equilibrium core. The PBMR 400 MW model was originally created in VSOP-99.

Both the reference models for the study are created in VSOP-99 and are based on information and the sources discussed in chapter 3. After being set-up, the core equilibrium results from the created models were compared to the data published. Table 7 shows the differences in the results for the HTR-Modul [40].

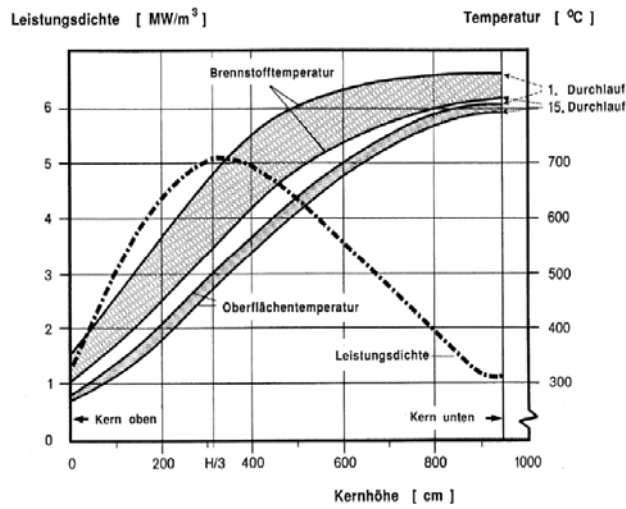
Table 7: HTR-Modul reference model results evaluation

Description	Units	HTR-Modul Published data	HTR-Modul Reference model results
Design parameters			
Thermal Power	MW	200	200
Core diameter (avg.)	m	3	3
Core height (avg.)	m	9.4	9.4
Primary coolant temperatures (avg.)	°C	250/700	250/750
Primary system pressure	Bar	60	60
Number of reflector rods		6	6
Small Absorber Sphere (SAS) units		18	18
Global equilibrium data			
Average residence time in core	Days	1007	1047
Average discharge burnup	MWd/t	80 000	83 767
U235 Enrichment	w/o	7.8	8
Heavy metal loading per FS	g	7	
Number of Fuel Spheres (FS)		360 000	356 845
Number of Fuel Passes		15	15
Time per pass	days	70	70
Calculated thermal performance			
Average power density	MW/m ³	3.0	3.02
Maximum power rating of FS	kW/FS	1.4	1.42
Maximum fuel sphere temperature	°C	865	980
Calculated neutronic performance			
Requirement for 100 – 50 - 100% load follow for HTR-Modul and 100 – 40 - 100% load follow for PBMR 400 MW	% Δk	1.2%	1.2%
Leakage (including absorption in the partially inserted upper control rods)	%	13.0%	13.4%
Average thermal flux (<1.86eV)	10 ¹³ .cm ⁻² .sec ⁻¹	7.3 ($E \leq 2.38eV$)	7.8 ($E \leq 1.86eV$)
Average fast flux (>0.1MeV)	10 ¹³ .cm ⁻² .sec ⁻¹	7.7 ($E > 2.38eV$)	2.6 ($E \leq 0.1MeV$)
Conversion ratio (ratio of production over loss of fissile nuclides)		0.47	0.45

Minor differences are found in the comparison. The burnup is slightly higher and the number of fuel spheres is slightly less. The higher burnup can be result of the 0.2% higher enrichment and the resultant 40 day longer residence time. This is the cause for the higher maximum fuel temperature of 980 °C resulting from the calculations. This difference of 115 °C indicates that the temperature results of the developed model are pessimistically calculated for the HTR-Modul compared to the older models and codes.

Figure 14 (a) shows the typical behaviour of the HTR-Modul [8], while (b) presents the graphs of the developed model. The developed model is accepted as a close representation of the original calculation model.

a)



b)

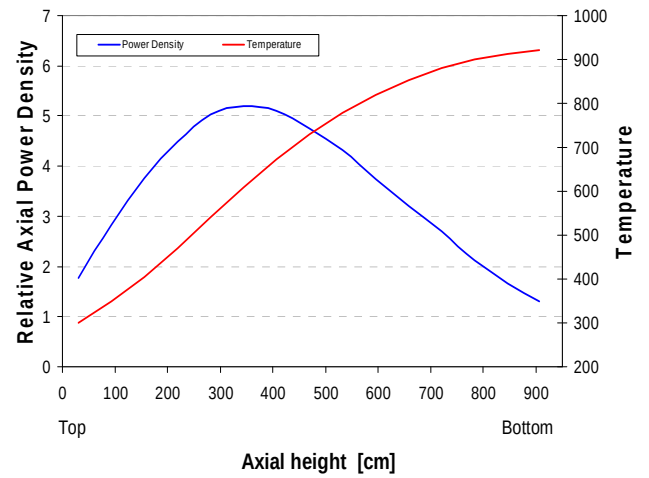


**Figure 14: Axial power density and temperature profile comparison of the HTR-Modul–
(a) from original data and (b) results from the developed model**

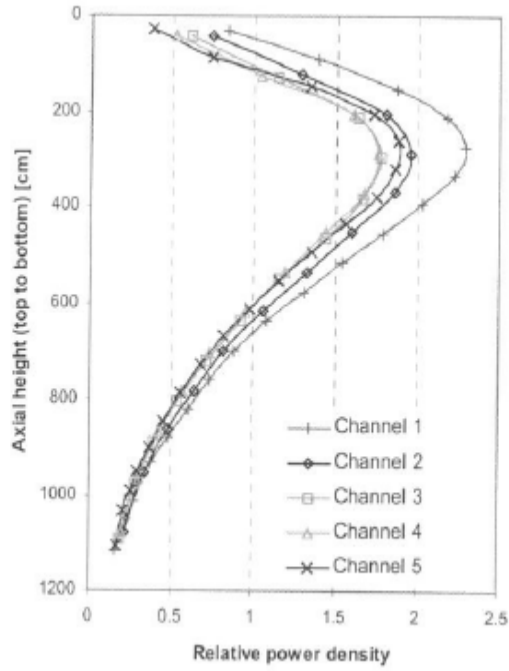
Table 8 demonstrates discrepancies between the developed reference model and the PBMR 400 MW model. As with the HTR-Modul, the burnup is also slightly higher and the same justification applies. The maximum fuel sphere temperature is 26 °C lower compared to published data [3]. This means that the fuel temperature profile is slightly better distributed for the developed model, but that the results are optimistic calculated for the PBMR 400 MW. The differences are also due to different versions of VSOP-99 being used for the two sets of analysis.

Table 8: PBMR 400 MW reference model results evaluation

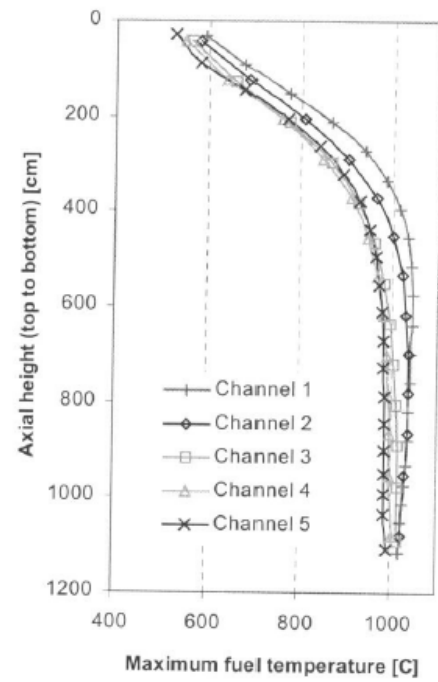
Description	Units	PBMR 400 MW Published data	PBMR 400 MW Reference model results
Design parameters			
Thermal Power	MW	400	400
Core diameter (avg.)	m	2 (inner) /3.7 (outer)	2 (inner) /3.7 (outer)
Core height (avg.)	m	11	11
Primary coolant temperatures (avg.)	°C	500/900	500/900
Primary system pressure	Bar	90	90
Number of reflector rods		24	24
Small Absorber Sphere (SAS) units		8	8
Global equilibrium data			
Average residence time in core	Days	970	956
Average discharge burnup	MWd/t	90 000	94 023
U235 Enrichment	w/o	9.6	9.6
Heavy metal loading per FS	g	9	9
Number of Fuel Spheres (FS)		450 000	451 519
Number of Fuel Passes		6	6
Time per pass	days	159	159
Calculated thermal performance			
Average power density	MW/m ³	4.5	4.7
Maximum power rating of FS	kW/FS	2.7	3.0
Maximum fuel sphere temperature	°C	1 130	1 104
Calculated neutronic performance			
Requirement for 100 – 50 - 100% load follow for HTR-Modul and 100 – 40 - 100% load follow for PBMR 400 MW	% Δk	1.4%	1.4%
Leakage (including absorption in the partially inserted upper control rods)	%	14.8%	14.8%
Average thermal flux (<1.86eV)	10 ¹³ .cm ⁻² .sec ⁻¹	7.8 ($E \leq 1.86eV$)	8.0 ($E \leq 1.86eV$)
Average fast flux (>0.1MeV)	10 ¹³ .cm ⁻² .sec ⁻¹	3.3 ($E \leq 0.1MeV$)	3.4 ($E \leq 0.1MeV$)
Conversion ratio (ratio of production over loss of fissile nuclides)		0.45	0.45

Figure 15 shows the respective power and temperature distributions of the published data of the PBMR 400 MW compared to the developed model. The profiles show close resemblance. Based on the results shown in Table 8 and in Figure 15 the developed model is accepted for the analysis relevant to this study.

a)



b)



c)

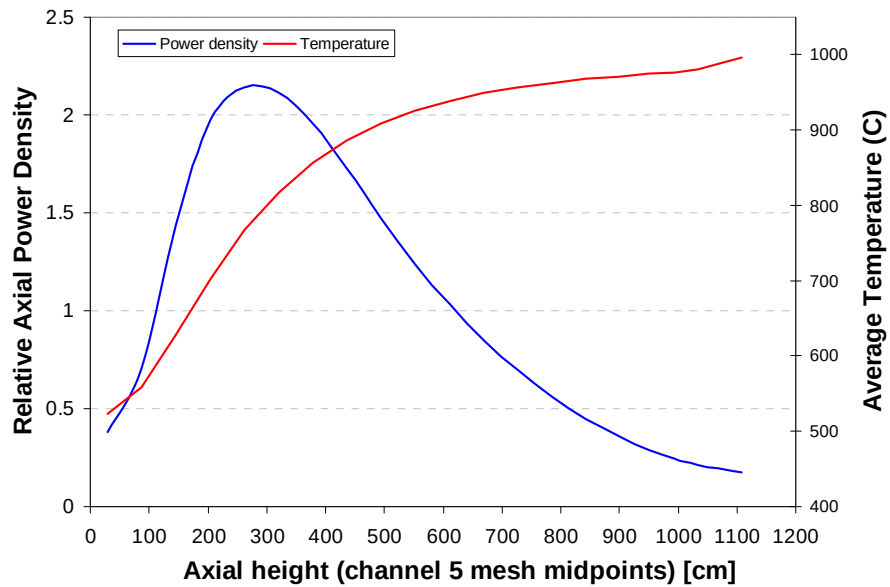


Figure 15: Axial profile comparisons of the PBMR 400 MW – (a) power density profiles from original data [3] (b) temperature profiles from original data [3] and (c) power and temperature profiles from developed model

4.7 EVALUATION CRITERIA

The following characteristics were selected to evaluate the results of the different fuel-pass models:

- The core overall performance, including the achievable discharge burnup and core neutron leakage;
- The power densities and profiles;
- The fuel average and maximum temperatures during normal operation;
- The fuel maximum and average temperatures during a depressurized loss of coolant (DLOFC) accident;
- The practical considerations of the burnup measuring system. This includes the amount of fuel to be circulated per day, the decay time in the defuel chute and the burnup measurement time;
- The expected spread in the fuel sphere burnup distribution and the implications for fuel qualification and testing.

The following characteristics were selected to evaluate the different control rod position models:

- The influence of control-rod positioning on the other evaluation criteria including the power profiles;
- The influence on the excess reactivity and shutdown margins.

The following characteristics were selected to evaluate the different enrichment models:

- Power profiles for different fuel enrichments
- Temperature profiles for the different fuel enrichments
- Fuel maximum and average temperatures during a DLOFC accident

For flux, power and temperature profile evaluation the following criteria was applied:

Data from the most outer channel of the core's results were selected to do profile comparison. The reason is that it is more sensitive for external effects such as that of the control rods. With regards to the temperature profiles, the same selection was made. This simplifies the comparisons significantly. Results have also shown that the axial profiles are all similar (see Figure 15 a and b) and since the change in the number of passes is predominantly an axial fuel burnup distribution or flattening, all radial profiles show the same relative effects.

The two profiles are very similar as the power profile is derived from the flux profile. Figure 16 and Figure 17 show typical examples from the HTR-Modul reference case. Note that the power profile is only over the area of the core and doesn't include the reflectors as with the flux profile. The detail flux, power and temperature profiles can be seen in Appendix 9.2.

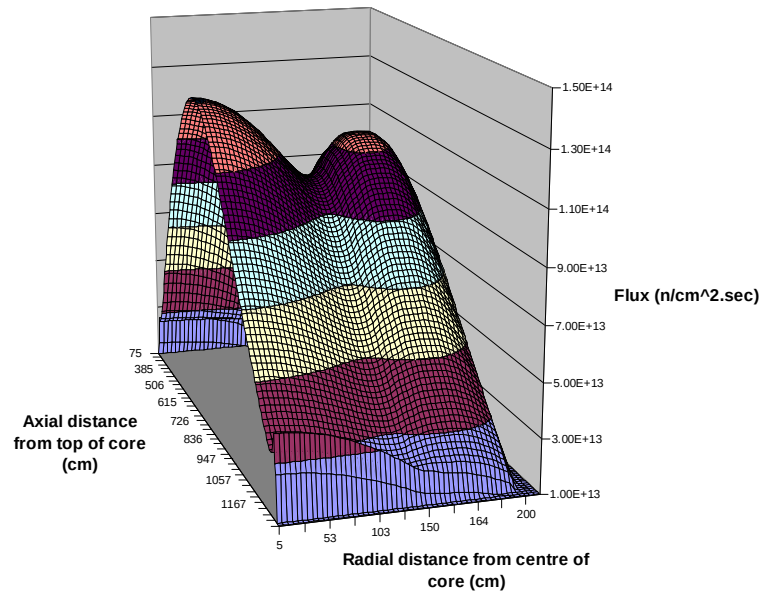


Figure 16: A typical flux profile for the the HTR-Modul

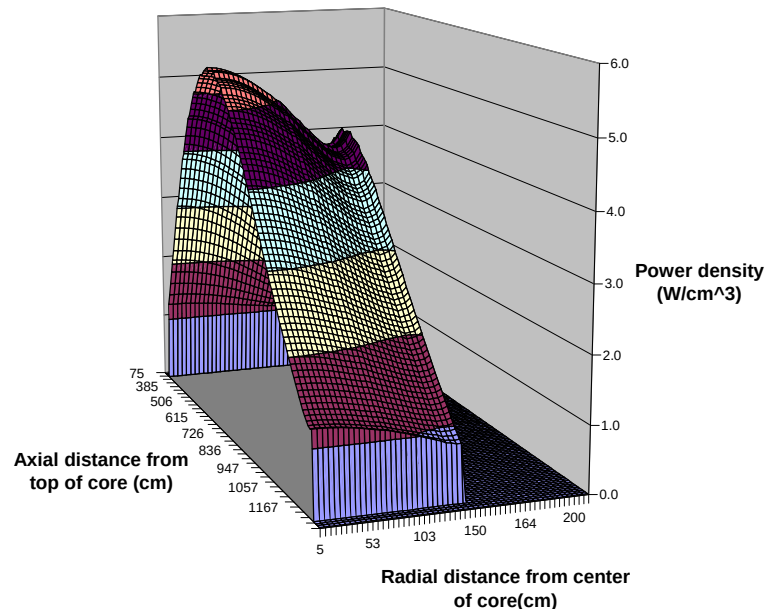


Figure 17: A typical power profile for the the HTR-Modul

5. RESULTS AND DISCUSSION OF FINDINGS ON THE HTR-MODUL

5.1 INTRODUCTION

The HTR-Modul has a cylindrical core of 9.4m in height and 3m in diameter, with only one defuel chute. The primary focus of the analysis performed for this reactor is to evaluate the relative effects on the temperature and the power profiles, the burnup and the DLOFC conditions on the different multipass cases. Cases from one to fifteen passes, with a total of nine models, were selected to do this evaluation. Note that a “Case 1” refers to a “1 Batch” or “1-Pass” analysis and it continues so forth. The discussion which follows is with regards to the second focus. This is concerned with the control rod position and movement on the power and excess reactivity as well as the improved burnup when higher enrichments are considered.

5.2 OVERALL PERFORMANCE

In Table 9 the overall performance of the different fuel-pass models are evaluated by comparing the average discharge burnup of fuel in the equilibrium cycles. The much lower burnup achieved for the OTTO cycle represents a 20% reduction in fuel utilization when compared with the 15-pass model. The penalty is huge for the OTTO cycle and two-pass models, but after six-passes or eight-passes the reduction is small ($\sim 1.6\%$) – also see Figure 18. The fuel remains in the core for nearly three years before discharge but in the one-pass case this is nearly seven months shorter. The residence times therefore reflect the same picture as the discharge burnup.

Table 9: Discharge burnup and leakage of different fuel-pass models

Case	Fuel residence time [days]	Av. discharge burnup [MWd/t]	Neutron leakage
1	827	66465	16.9%
2	950	75864	15.1%
4	1013	80997	14.0%
6	1031	82468	13.7%
8	1038	83058	13.6%
10	1042	83380	13.5%
12	1044	83567	13.5%
14	1046	83700	13.5%
15	1047	83767	13.4%

The discharge burnup variation is due to the differences in the core reactivity that result from different burnup distributions and core characteristics. The power defect (fuel temperature,

xenon) and the neutron leakage are the main contributors. The net leakage shows a 3.5% higher value for the OTTO cycle, decreasing to only 0.3% higher leakage for the six-pass model. The peaked power and flux shapes (see later results) are responsible for the high leakage cores.

5.3 EQUILIBRIUM CYCLE POWER AND TEMPERATURES

The most important effect of multiple circulations of the fuel spheres is the resultant flattened power profile. In the single pass or OTTO cycle, all the fresh fuel or fuel with low burnup is located in the upper part of the reactor while all the depleted fuel is in the lower part. When the multi-pass approach is followed fuel is circulated faster through the reactor and fuel that is not fully depleted is returned to the top of the core. Every region in the core will thus contain a mixture of fuel with different degrees of depletion. In the case of a two-pass model the core will contain fuel that is being passed through the first time and fuel that is busy with its second pass through the core. This is the same as for multi-pass models.

In Figure 18 the effect is clearly illustrated. The average burnup at the top of the core, just as fuel is loaded, and at the bottom of the core, at the defuel chute, are shown and labelled “Top of core” and “Bottom of core”. The burnup difference is also given. So, whereas the difference is very large for the OTTO cycle with one-pass, the difference decreases as the number of passes increases and reduces to less than 10,000 MWd/t for more than ten-passes.

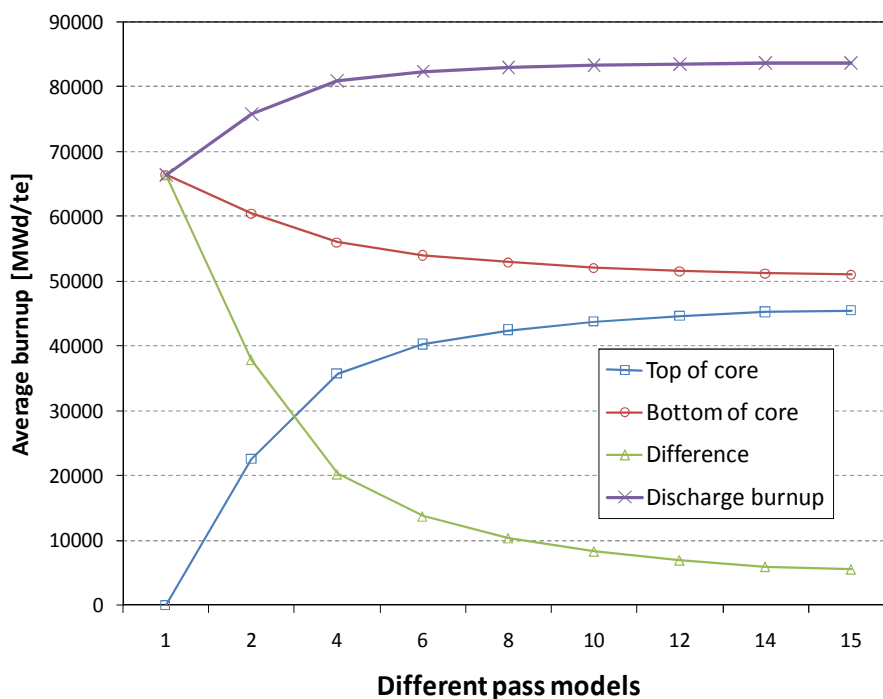


Figure 18: HTR-Modul burnup information for the different number of fuel-pass models

The most important effect of the flatter axial burnup profile and mixture of pebbles of multiple circulations and thus burnup in all regions is the resultant flattened power profile.

With the OTTO cycle the power is very peaked towards the upper part of the core where all the fresh fuel is located, as can be seen in Figure 19. The more passes made by the fuel before discharge the flatter the power profile becomes. Figure 19 shows the relative power profile in the outer pebble flow region (region 5 in Figure 9) of the core, next to the reflector and is normalized to an average of one. The changes in the profile for the eight-pass or more are small with the major benefit seen after increasing the number of passes to six.

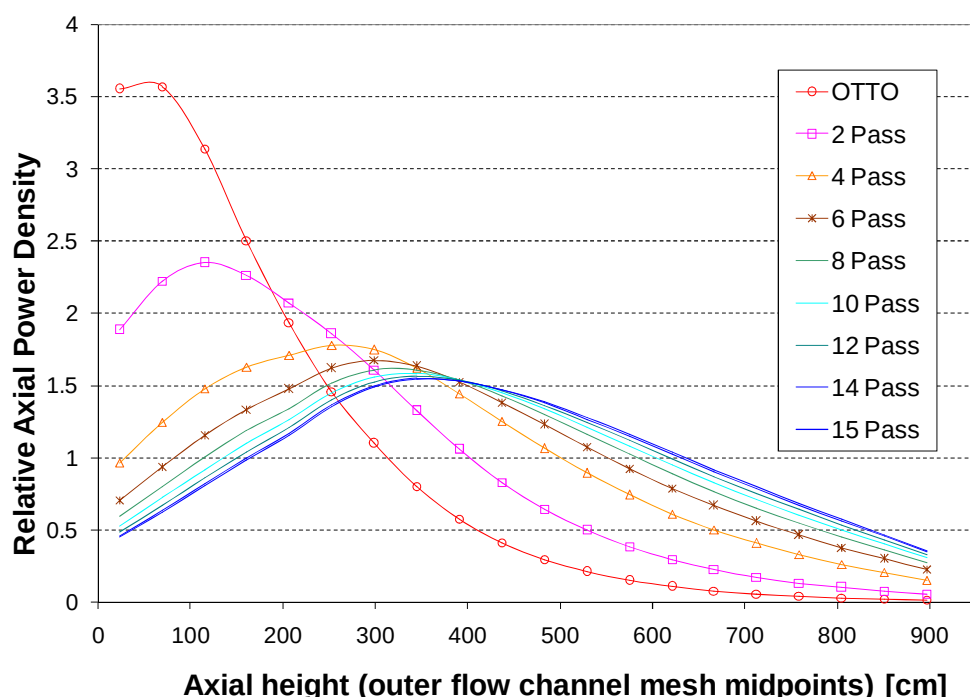


Figure 19: HTR-Modul axial power profile for the different number of fuel-pass models

The average fuel temperatures axial profile in the outer pebble flow channel is given in Figure 20. The temperature profile is a function of the power profile as the helium coolant, flowing from top to bottom, is heated up. This changes the fuel surface temperatures and the associated heat transfer coefficient and thermal material properties.

The fuel temperatures of the OTTO cycle, with its power peaked towards the top, increases rapidly in the upper part of the core but then becomes nearly constant in the lower part of the core, where virtually no power is being generated. In contrast the 15-pass model fuel temperatures initially increase slowly in the upper part, then show a larger increase as the power peaks around 300 cm. The slope then decreases again towards the bottom as the power density decreases.

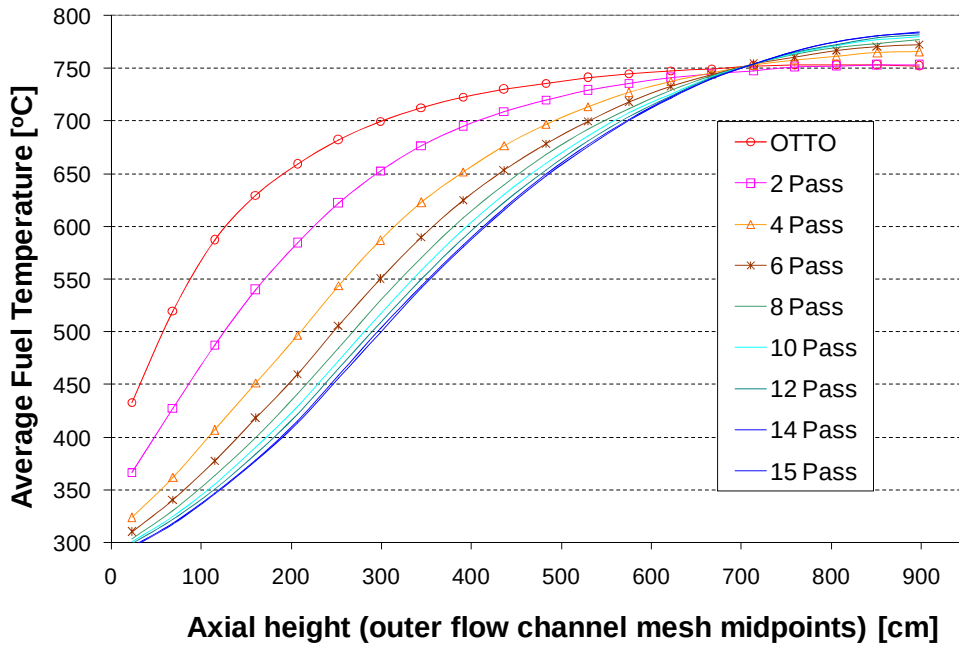


Figure 20: HTR-Modul axial average fuel temperature profile for the different number of fuel-pass models

However, the maximum fuel temperature is larger in the multi-pass cases (compared to the OTTO or two-pass case), but the absolute difference is only 72 °C and the fuel will only spend a relatively short time at these higher temperatures. To place this in perspective, the average fuel temperature in the OTTO cycle is 134 °C higher than in the 15-pass case. These temperatures indicate the relative effects for the purpose of this study and are based on the outer pebble flow region. However, the inner pebble flow region shows higher peaking values and can be observed from the figures in Appendix 9.2.

The most important effect of increased fuel-passes is the flattened axial profile. There is also a smaller improvement in the radial power profile. This is shown in Figure 21. The relative power produced in the five flow channels or zones is shown. The faster the fuel is circulated or the more passes the fuel makes before discharge, the better the mixing of fuel with different burnup. This tends to flatten the radial power profile and thus further reduces the power peaking.

Table 10: Summary of the global thermal performance of the different pass models for the HTR-Modul design

Case	Max. Fuel Power [kW]	Av. Fuel Temp. [°C]	Max. Fuel Temp. [°C]
1	2.25	756	908
2	1.88	719	951
4	1.59	675	965
6	1.50	653	972
8	1.47	641	975
10	1.44	633	977
12	1.43	628	978
14	1.43	624	979
15	1.42	622	980

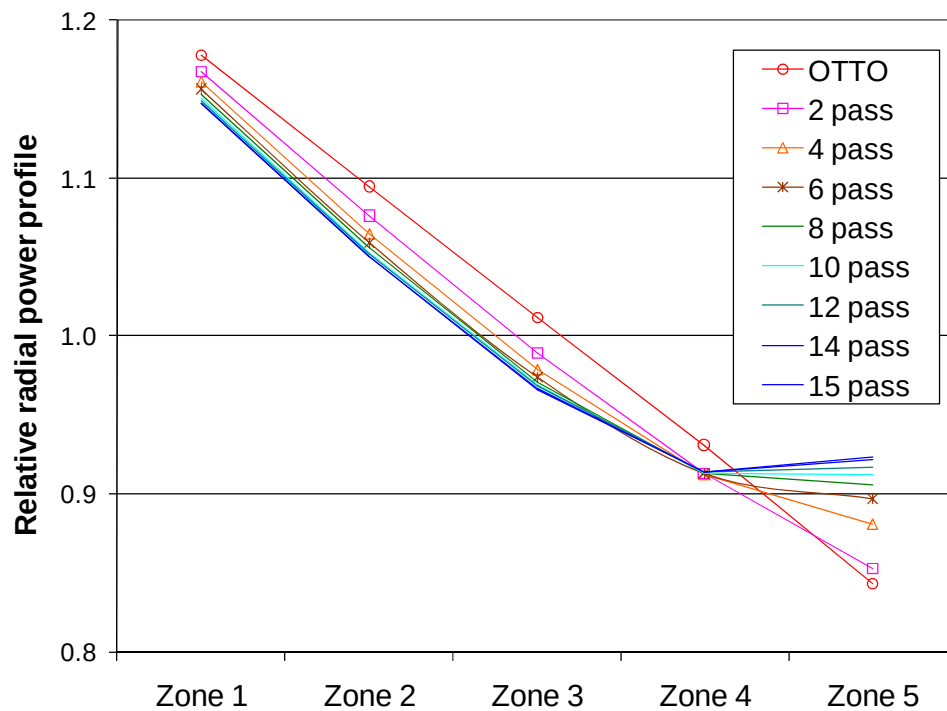


Figure 21: HTR-Modul relative radial power profile for the different number of fuel-pass models

5.4 ACCIDENT CONDITION TEMPERATURES

One of the most severe accident conditions for high temperature reactors such as the PBMR is a depressurised loss of forced cooling (DLOFC) accident. For this event it is assumed that all active forced cooling by helium stops and that the coolant is released from the pressure boundary

to atmospheric conditions. The heat capacity of the remaining helium and the effectiveness of natural convection within the core are significantly reduced. Once forced cooling is stopped the fuel will heat up and the negative temperature coefficient of reactivity will cause the reactor to become sub-critical.

After shutdown the power produced in the core is by decay heat with the main contribution by the short-lived fission products. The heat removal mechanism of the pebble bed is radiation and conduction between pebbles, and the thermal conductivity within the pebbles.

The high thermal capacity of the graphite contributes to a very slow transient and relatively lower fuel temperatures. As time progresses following such an event, heat is transferred to the reflectors (side reflectors and also the central column in the annular core design). The ultimate heat sink is through the reactor metallic structures through radiation and natural convection to the cavity cooling system or the building structures.

The results for the average fuel temperatures are shown in Figure 22. The fuel temperature increases initially but as the decay heat production reduces and the heat removal increases, it reaches a maximum and starts to decrease.

Interesting to note is that for the lower pass cases one to four, an increase in the maximum average fuel temperature can be observed as the number of passes increases, while the maximum average fuel temperatures decrease as the number of fuel-passes increase from four to 15. The turnaround point occurs at the four-pass case. To verify the phenomena an additional case for three passes was performed and included in the comparison.

This average fuel temperature phenomenon is due to the specific power profiles and the helium coolant that is heated flowing from the top to the bottom of the core. In the case of loss of coolant flow, there is no more convective heat transferred by the helium to the lower core volumes. With the OTTO cycle, depleted fuel elements (with very low power and decay heat) are found at the bottom of the core and therefore on average the fuel temperature for a DLOFC are lower compared to a four-pass where the mix between fresh and depleted fuel elements is distributed more between the top and the bottom of the core (and therefore also the power and decay heat).

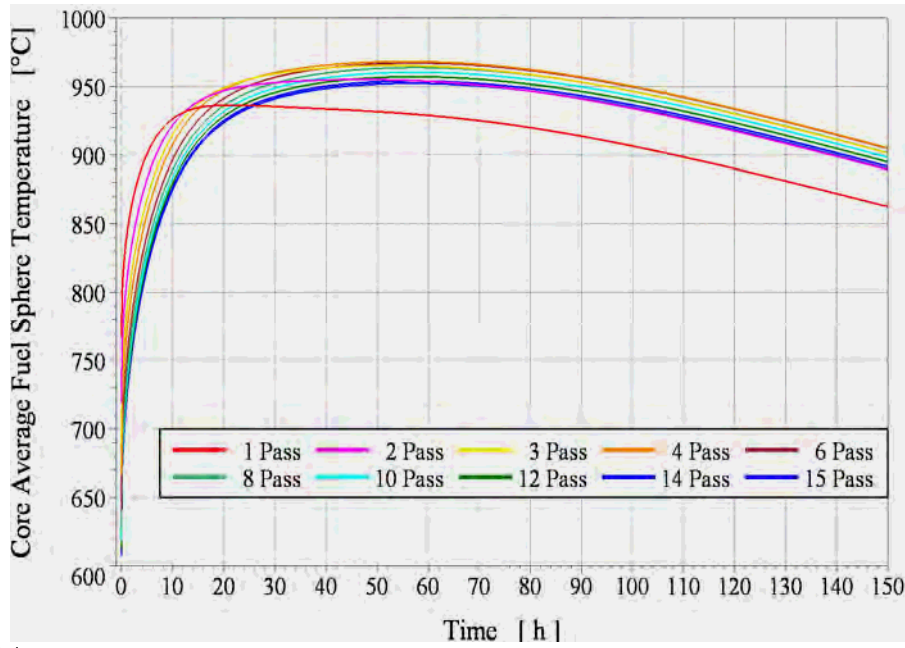


Figure 22: The average fuel temperatures during a DLOFC event for the different number of fuel-pass models

Due to the higher concentration of fresh fuel at the top in the OTTO cycle and the associated high power and decay heat only at the top, the turnaround of the average fuel temperature is achieved earlier (about 15 hours) as the large lower core volume (low decay heat) cools quickly. The increased heat capacity with increased temperature also enhances this effect as more heat can be stored in the upper hot fuel volume. In the four-pass case, with a much more even power and decay heat distribution, the turnaround is only reached at around 55 hours.

The decrease in maximum average fuel temperature for fuel-pass cases four to 15 is due to the flatter axial power and decay heat profiles. More or less the same decay heat is therefore distributed more evenly over the core volume.

During a DLOFC the maximum fuel temperatures for fuel-pass cases one to 15 ranges between 1400 °C and 1940 °C (Figure 23), with the highest temperature for the one pass or OTTO case. The OTTO and two-pass cases challenge the passive safety principle keeping the fuel limit below 1600 °C, while the three pass case is close to this limit but is still within range. This is the reason why the OTTO cycle is limited to only 120 MWt, as discussed in [39].

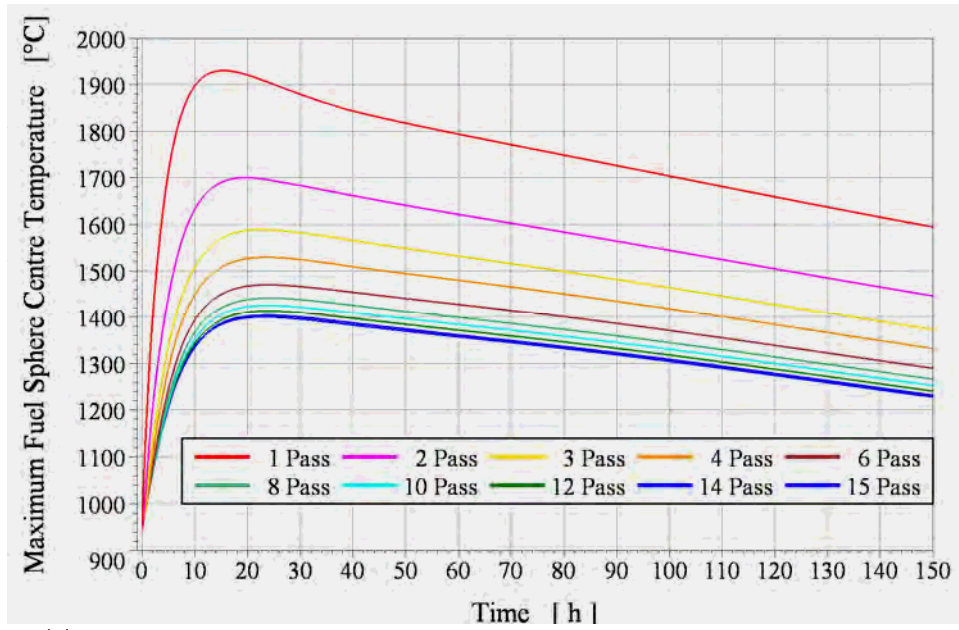


Figure 23: The maximum fuel temperatures during a DLOFC event for the different number of fuel-pass models

The benefit of additional fuel-passes to lower the maximum and average fuel temperature during DLOFC accidents is clear. The incremental benefit unfortunately decreases for each additional fuel-pass added. For example, the decrease in the maximum fuel temperature is ~ 50 °C when the number of fuel-passes increases from six to ten, while it is only ~ 20 °C from ten to fifteen passes.

5.5 BURNUP DISTRIBUTION AND PRACTICAL CONSIDERATIONS

The requirement specification of the fuel handling and burnup-measuring system is directly influenced by the decision to use the multi-pass model. The main parameters that need to be taken into account are the number of fuel spheres to be circulated and the time available for the burnup measurement. Other factors such as the required size of the buffer lines for fresh and spent fuel or the decay time of fuel after it is removed from the core until it reaches the defuel-machine must also be taken into account. In this section the close interconnectivity between the hardware requirements and the choice of the fuel-pass model is highlighted and no opinion is expressed if the requirements on the fuel and burnup measurement system can be met.

In Table 11 the number of fuel spheres to be handled per day is given for each fuel-pass model, followed by the number of fresh fuel spheres that need to be loaded per day (and thus also the number of fuel spheres that will be removed to the spent fuel handling system). The decay time refers to the time spent in the defuel chute below the core before fuel spheres are unloaded and was based on a defuel chute diameter of 50 cm and height of 11.5 m.

The measuring time is then the time available in seconds to make the burnup measurement, assuming a 22/24 hour operation. The minimum 50 hour decay time and 10 seconds measurement time was defined in the HTR-Modul project [46], [40]. The decay and measurement time is critical to accurately measure the Cs-137 gamma peak, used to derive the burnup, above the background gamma noise.

Table 11: Fuel sphere circulation rates and time for decay in the defuel chute and for measuring the fuel sphere burnup

Case	#/day	Fresh/ day	Decay time [hrs]	Measure time [sec]
1	431	431	593	171
2	751	375	341	96
4	1409	352	182	49
6	2077	346	123	32
8	2750	344	93	23
10	3425	342	75	18
12	4101	342	62	14
14	4777	341	54	11
15	5115	341	50	10

The results show that the requirements of the HTR-Modul are met with the 15 fuel-pass model (this was the design choice). For all the other fuel-pass models the decay times increase as well as the time available for burnup measurement. Both of these time increases will improve the accuracy of the burnup measurements and simplify the system designs. A careful balance should therefore be found between the benefits of more fuel-passes and the accuracy of the burnup measurement system.

Burnup measurement accuracy cannot be compromised since many of the advantages of the multi-pass system could then be nullified. These advantages include the overall performance (average discharge burnup may decrease, peaking may increase if depleted fuel is re-circulated when it should not be) and the fuel qualification needs, as discussed below.

5.6 BURNUP DISTRIBUTION AND FUEL QUALIFICATION

The fuel qualification plan is strongly influenced by the fuel-pass model. The operating envelope for normal operation includes the peak fuel-sphere power, fuel temperatures, fast fluence and the discharge burnup, and the maximum fuel temperatures in accident conditions. All of these are

directly influenced by the fuel-pass model as shown before. In selecting the burnup value for fuel qualification, some margin should be added to the expected discharge burnup. This margin must take care of the statistical variation in the burnup of the fuel being measured and the accuracy of the burnup measurement system.

In Table 12 the average burnup increment for the last pass through the core is given as well as the average time it would take to complete the cycle. For the one-pass model no burnup measurement is made but the spread is large and the model shows fuel spheres with a burnup of up to 72,630 MWd/t. It is clear that, although the average discharge-burnup is only 66,465 MWd/t, the fuel qualification will have to be proven to a much higher level, say 80,000 MWd/t.

With the increase in the number of fuel-passes the spread in burnup (due to a single pass) decreases and becomes less important. For the six-pass model the spread is less than 2,000 MWd/t (on an average discharge-burnup of 82,468 MWd/t – see Table 9). The accuracy of the burnup measurement and the burnup that will be gained if the fuel again passes through the core is now important. Fuel returned to the core will take another 172 days on average before reaching the burnup measurement device again. The higher the number of fuel passes, the smaller this effect would be. In the case of the 15-pass model the fuel will circulate in only 70 days and gain less than 3,700 MWd/t in burnup.

In summary, the fuel qualification burnup target will be much closer to the average discharge burnup value for the higher fuel-pass cases. The benefit could be substantial in fuel economics and fuel qualification costs. On the other hand, it places a strict requirement on the burnup measurement system to be accurate with the minimum decay and measurement times, as discussed before.

In the case of all fuel-pass models (except the OTTO) the accuracy of the burnup measurement system and the statistical probability that a spent fuel element that should be discarded, is erroneously re-circulated to the core, needs to be considered as part of the safety case.

Table 12: Burnup added in the last pass and duration of the pass for different fuel-pass models

Case	Days per pass	Last pass burnup	Spread in burnup	Spread as % of pass
1	827	66465	11449	17.2%
2	475	30674	6222	8.2%
4	253	14368	2961	3.7%
6	172	9348	1870	2.3%
8	130	6920	1576	1.9%
10	104	5493	1159	1.4%
12	87	4553	992	1.2%
14	75	3887	870	1.0%
15	70	3623	821	1.0%

5.7 CONTROL ROD POSITIONS

The purpose of the control rods is to control the reactivity and the excess reactivity of the core. If control rods are inserted deeper during full power equilibrium conditions, more excess reactivity is available to overcome xenon poisoning during load-follow power changes.

To witness the effect of the control rod position on the power profile behaviour, the control rods were inserted 1 m deeper than the normal operating position and withdrawn in 1 m increments until they were 2 m above the normal operating position.

For each case the power was kept at 200 MW and equilibrium conditions were obtained. The ten-pass case was selected as the reference case since it is a good average representation of the six to 15-pass cases. According to the 200 MW reactor design the six control rods are positioned in the side reflector and inserted 2.96 m below the top reflector (~2.46m below the flattened pebble bed level).

It can be observed from Figure 24 that the axial peaking of the relative power density decreases with just less than 0.2 MW/m³ as the control rods are inserted and it also shifts to the middle of the core causing a flattened power profile.

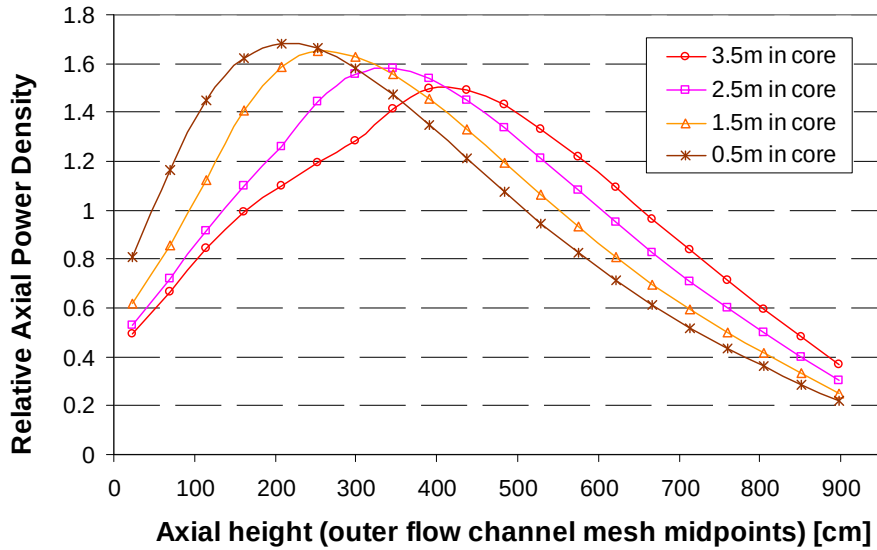


Figure 24: HTR-Modul axial relative power density profile for the different control rod position cases

On average the fuel temperature also declines (see Figure 25) as the control rods are inserted deeper into the reactor although the maximum value of the average temperature of 780 °C remains the same for each case.

These results indicate that it might be beneficial to insert the control rods a bit deeper to obtain a better axial power profile. Alternatively it could be achieved by inserting poison rods in the side reflector. However, this will affect the control rod behaviour and shutdown margins and will need careful consideration.

On the other hand it also needs to be noted that the maximum temperature increases from 947 °C to 991 °C with the insertion of the rods. This will have an impact on the safety analysis during accident conditions as higher maximum temperatures can emerged.

As the control rods are inserted into the core the excess reactivity is increased and a decrease in the average discharge burnup is observed with a difference of up to 2114 MWd/t (see Table 13). It is also shows that, although the control rod worth increases, enhancing the rod efficiency in this scenario, the shutdown margin (at operating conditions) decreases by 1.66 % to only 1.46 %.

Therefore, it is evident that by inserting the control rods deeper into the core, a preferred power profile and temperature profile can be achieved, but at the expense of the discharge burnup and higher maximum temperatures. In order to achieve the same shutdown performance more control rods would be required.

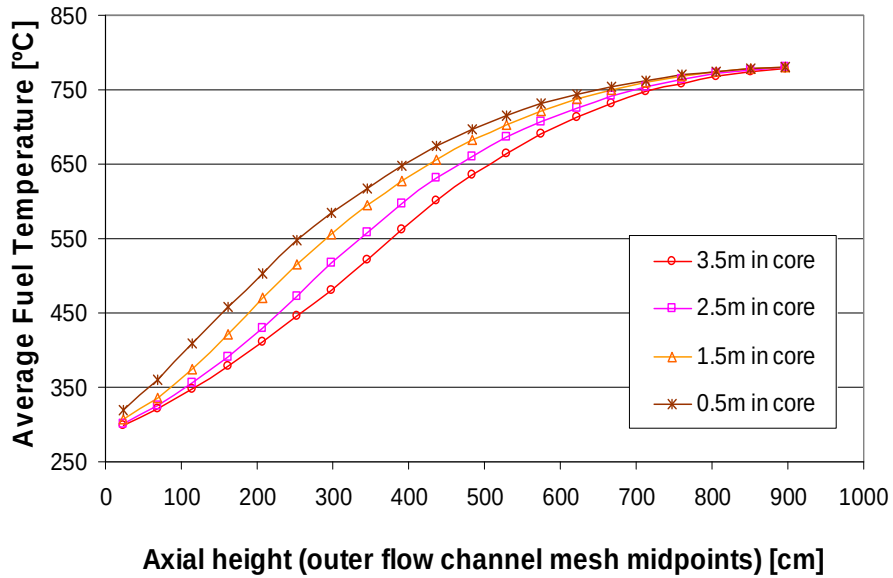


Figure 25: HTR-Modul axial average fuel temperature for the different control rod position cases

The higher excess reactivity will also aggravate postulated control-rod induced reactivity excursions due to the higher control rod efficiency.

Table 13: Discharge burnup, control rod worth and shutdown margins for the different control rod position cases

Case	Average discharge burnup [MWd/t]	Control rod worth [% Δk_{eff}]	Shutdown margin [% Δk_{eff}]	Max. Fuel Temp. [°C]
0.5m	84499	3.39 %	3.12 %	947
1.5m	84116	3.55 %	2.73 %	961
2.5m	83380	3.68 %	2.12 %	977
3.5m	82385	3.74 %	1.46 %	991

5.8 FUEL ENRICHMENT VARIATIONS

Fuel enrichment studies were performed to observe the variations in burnup as well as the impact on power peaking and fuel temperatures. The scope is limited to changing the enrichment values without changing the heavy metal loading. For the HTR-Modul the fuel has an enrichment of 8wt% of ^{235}U . In these sensitivity analyses three enrichment values of 6%, 8% and 10% were considered for the ten -pass fuel model.

Figure 26 shows the trend for the normalised relative power density as enrichments are increased (also to be observed in Table 14). This is to be expected as a larger mass of ^{235}U (due to the higher enrichment and relatively more reactive fuel) is in the upper part of the core than in the

lower part. The difference in the average burnup at the top of the core compared to the bottom of the core will increase in a similar way as seen for the multi-pass study (in Figure 18) and a similar behaviour in power peaking and temperatures is therefore to be expected.

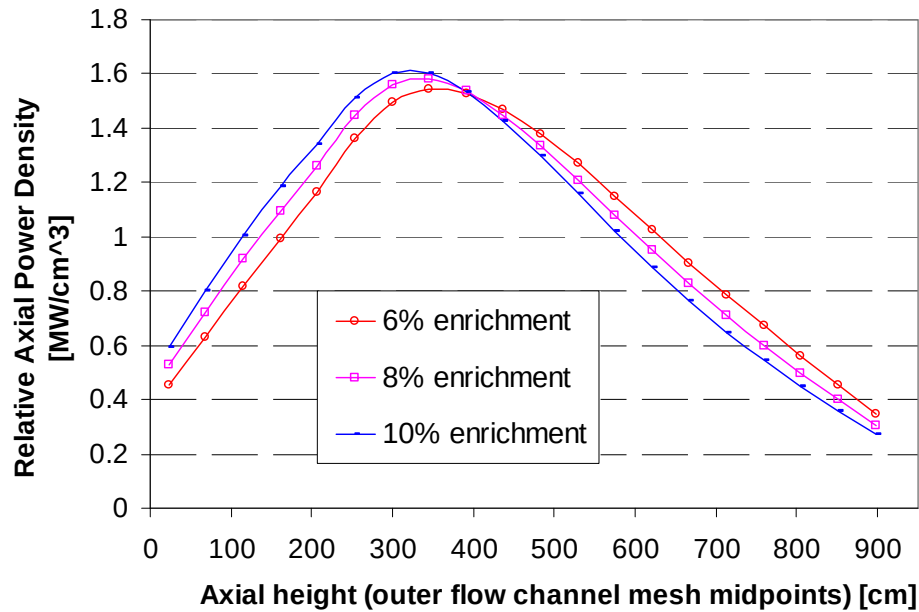


Figure 26: HTR-Modul axial relative power density profile for different fuel enrichments

From what has been seen in Paragraph 5.7, the control rods play a significant role in the shape of the power profile. If it were not for leakage at the top of the control rods as well as the way in which they were inserted, the maximum power would have been at the very top of the core (lowest burnup). For the same reason the power also starts to decline towards the bottom of the core. From Table 14 a decrease in neutron leakage with about 1% occurs as the enrichment goes up. As the enrichment increases from 6% to 10% the average discharge burnup almost doubles. Considering these factors it explains why the power density is shifting upwards on Figure 26 and why the power peaking increases (Table 14).

Table 14: General information for the different enrichments presented for the HTR-Modul

Case	Av discharge burnup [MWd/t]	Neutron leakage	Power peaking	Max fuel temp. [°C]
6%	54499	14.1%	2.19	981
8%	83380	13.5%	2.57	977
10%	110024	13.1%	2.93	975

The average fuel temperature is lower for the low enrichment case (Figure 27) but the maximum fuel temperature is marginally higher with a difference of only 6 °C.

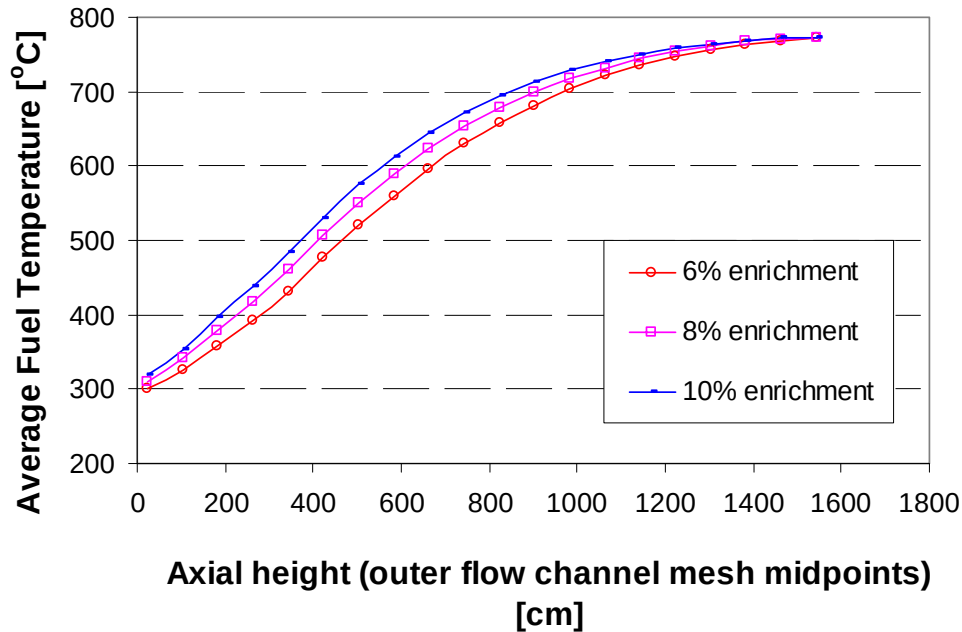


Figure 27: HTR-Modul axial average fuel temperatures for different fuel enrichments

During a DLOFC accident the maximum fuel temperature for all three of the enrichment cases remains below 1600 °C.

Figure 28 shows that the highest maximum fuel temperature of about 1560 °C is achieved by the highest enrichment case of 10 wt%.

The difference in the average burnup between the top and bottom of the core increases with increased enrichment. In order not to increase the power peaking and the fuel temperatures, the number of fuel-passes have to be increased. This is important to bear in mind during the reactor and fuel handling system design since the enrichment may be increased at a later stage for better fuel economy. This will require spare capacity for the fuel handling system to be able to increase the number of fuel spheres that can be circulated in a day and also to increase the measuring time available at the burnup measurement system. The PBMR 400 MW, with three defuel chutes and measurement lines, has some spare capacity that would allow such an upgrade later.

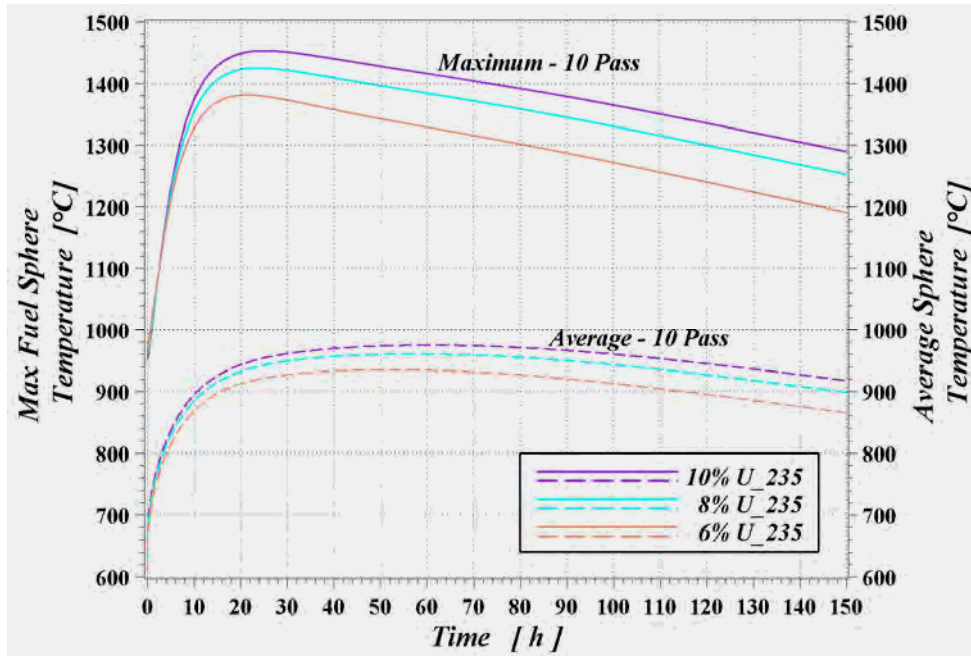


Figure 28: Average and maximum fuel temperatures during a DLOFC event

6. RESULTS AND DISCUSSION OF FINDINGS ON THE PBMR 400 MW

6.1 INTRODUCTION

The PBMR 400 MW design has specific characteristics that may change the impact that the selected fuel-pass model has on the selected evaluation criteria such as the power and temperature profiles. The core is taller (effectively 11 m *vs* 9 m), annular in shape and narrow (85 cm width). The heavy metal loading of the fuel is different (9 g/pebble *vs* 7 g/pebble), the reference enrichment is higher (9.6 % *vs* 8 %), the control rods more (24 compared to only 6) and when fully inserted, they extend to the total axial height in two banks [24]. The multi-pass studies were restricted to only three cases.

The design choice of six fuel-passes was selected as the reference while ten and 12 fuel-pass cases were added. The PBMR 400 MW design fuel-pass study should confirm whether or not the trends and conclusions of the HTR-Modul multi-pass study are directly applicable or whether the annular design introduces other unexpected results.

6.2 OVERALL PERFORMANCE

In Table 15 the overall performance of the different fuel-pass models is evaluated by comparing the average discharge burnup of fuel in the equilibrium cycles.

Table 15: Discharge burnup and leakage of different fuel-pass models for the PBMR 400 MW design

Case	Fuel residence time [days]	Av. discharge burnup [MWd/t]	Neutron leakage
6	956	94023	14.8%
10	972	95582	14.6%
12	975	95952	14.6%

A small increase in the average discharge burnup is shown with the associated small decrease in the total neutron leakage from the core. This agrees with the behaviour seen for the HTR-Modul.

Although the discharge burnup is higher for the PBMR 400 MW design, the core residence time is shorter (956 days for six-passes compared to 1031 days for six-passes of fuel in the HTR-Modul). This is the net effect of the increased initial enrichment (9.6% *vs* 8%) that increases the burnup and thus residence time and the higher thermal power (400 MW *vs* 200 MW) that

decreases the residence time. The behaviour is therefore as expected with the same trends as the HTR-Modul.

6.3 EQUILIBRIUM CYCLE POWER AND TEMPERATURES

We have seen before that the most important effect of the multiple fuel-pass models is the flattened power profile. The mixture of pebbles is the same from multiple circulations and thus the smaller variation of burnup in all regions, irrespective of the design. One would thus expect the same behaviour. Figure 29 shows the relative power profile in the outer pebble flow region (Region five in Figure 10) of the core, next to the reflector. The relative power values are normalized to an average of one. It is shown for the six, ten and 12 fuel-pass models and compared to the same cases of the HTR-Modul.

For the HTR-Modul fuel initially enriched to 10% was used in order to remove the effects of enrichment differences as shown above. In each case the graphs are plotted against the physical axial height of the core.

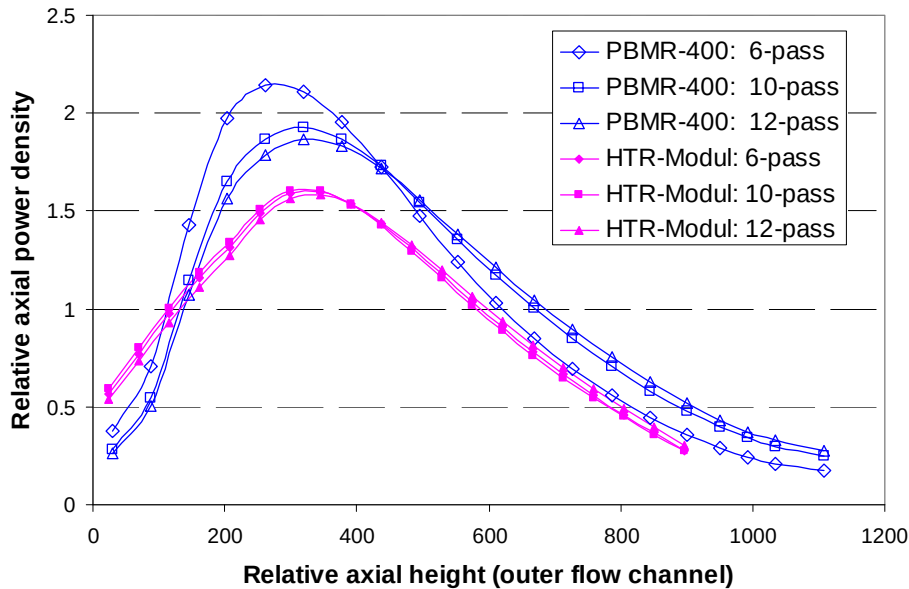


Figure 29: PBMR 400 MW axial power profile for the different number of fuel-pass models compared to the HTR-Modul results

The first observation is that the peaking in the PBMR 400 MW is larger than for the HTR-Modul and that the benefit of increasing the number of fuel passes from six to ten and 12 is larger. Even though the changes in the profile for the eight-pass or more are small in the HTR-Modul, a

significant benefit is still seen in the PBMR 400 MW design after increasing the number of passes from ten to 12.

Another observation is the more pronounced effect of the control-rod insertion on the axial power profile for the PBMR 400 MW design. The 24 control rods inserted depresses the power in the upper part of the core while the six control rods of the HTR-Modul have a much smaller effect. The control rods are also much closer to the fuel in the relatively thin annular core compared to the relatively larger distances of the cylindrical HTR-Modul.

The average fuel temperatures axial profile in the outer pebble flow channel is given in Figure 30. As before, the average temperatures decreased as the passes increased, while the peak temperature actually increased slightly.

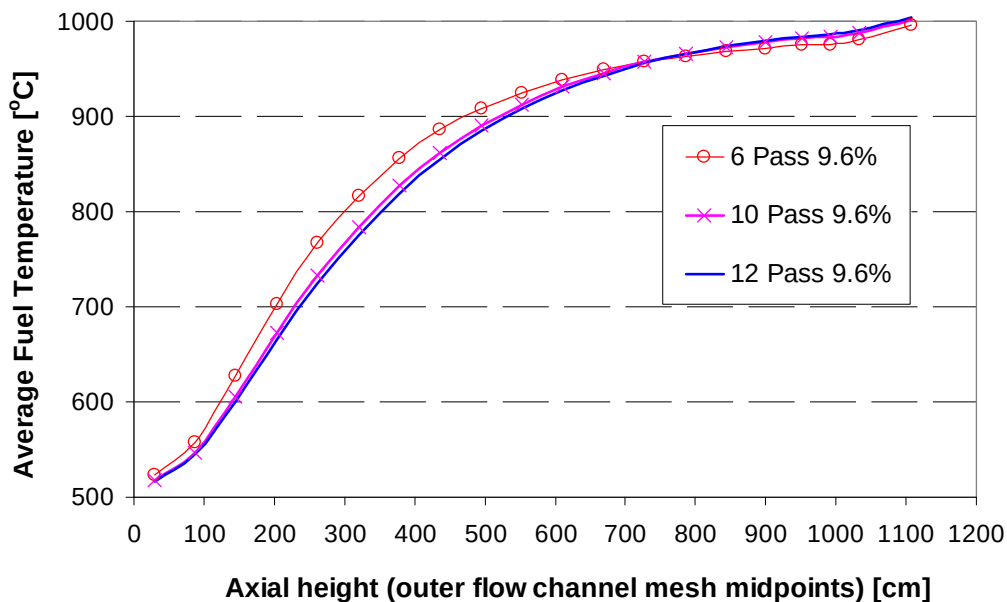


Figure 30: The PBMR 400 MW axial average fuel temperature profile for the different number of fuel-pass models

The results for the PBMR 400 MW are summarized in Table 16. The lower peak power density and the lower average fuel temperatures are shown for more passes as before.

The power and temperature behaviour of the PBMR 400 MW design is thus similar to that of the HTR-Modul, but due to the higher peaking and the effect of the control rod design it benefits even more from additional passes.

Table 16: Summary of the global thermal performance of the different pass models for the PBMR 400 MW design

Case	Max. Fuel Power [kW]	Average Fuel Temperature [°C]	Maximum Fuel Temperature [°C]
6	3.00	884	1104
10	2.86	871	1107
12	2.83	868	1109

6.4 BURNUP DISTRIBUTION AND PRACTICAL CONSIDERATIONS

The practical implications of different fuel-pass models on the circulation and measurement times were already discussed for the HTR-Modul design. For the PBMR 400 MW annular core design the requirements are less stringent since three defuel chutes (equally spaced in the annulus) and therefore three different fuel lines and burnup measurement devices were included in the design. So, despite the higher fresh fuel load (472 fuel spheres compared to 346 fuel spheres for the six-pass cases), the decay time and measurement times are multiplied by three.

The spread in the discharge burnup, based on the last pass in the model, was also shown for the HTR-Modul. The spread is typically small and thus within the measurement accuracy. In the PBMR 400 MW design the picture looks slightly different. The radial power is peaked close to the inner reflector that corresponds to longer core residence times of the fuel spheres. The fuel movement at the edge of the reactor is slower due to the interaction with the reactor wall (reflector blocks). The integrated power (or additional burnup) for fuel slowly passing in the high power radial volume is thus large compared to the fuel in the centre of the annulus where the power is lower and the pebble flow speed is faster. The spread in the burnup for the last pass is therefore in the order of four times larger for the PBMR 400 MW design.

The spread in the discharge burnup as a function of the number of passes and for two enrichments are shown in Figure 31 and Figure 32.

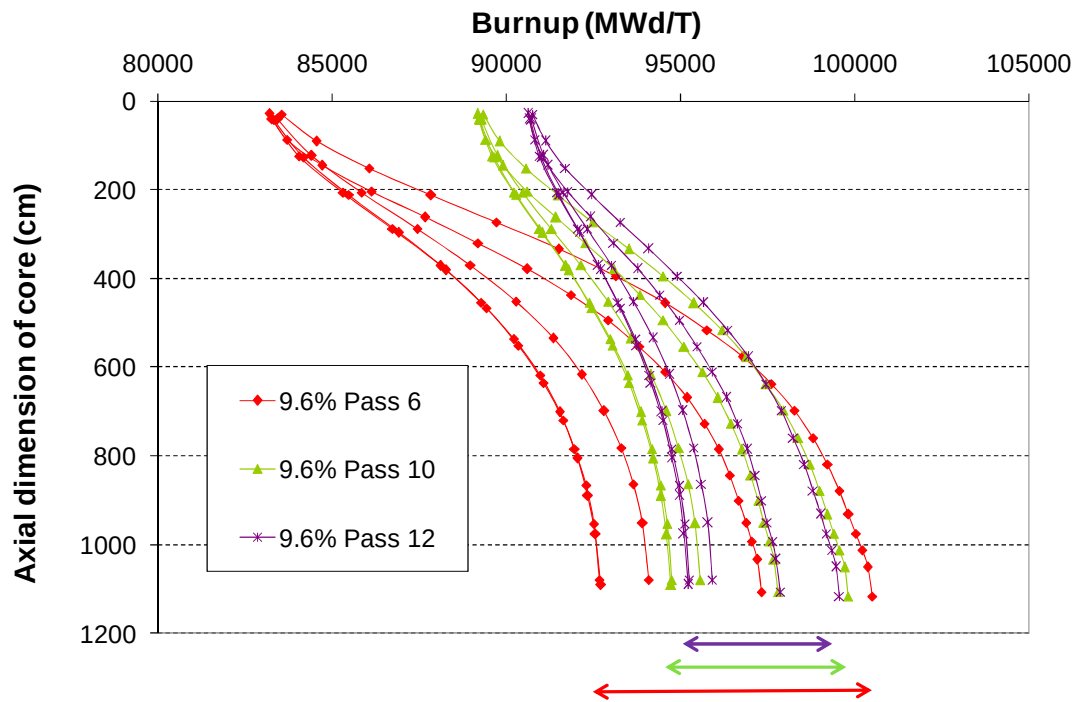


Figure 31: PBMR 400 MW discharge burnup distribution for the different number of fuel-pass models with 9.6% enriched fuel

In the first graph the last pass burnup is shown with 9.6% enriched fuel and for six, ten and 12 passes respectively, while the second graph is for 19.75% enriched fuel. The spreads are shown by the arrows below the graph. The decreasing spread for more passes is immediately apparent in both graphs, while the absolute spread increases for the increased initial enrichment, mainly because the absolute discharge burnup value is higher.

It is important to include the spread or distribution of the fuel burnup in the design and performance evaluation of the burnup measurement system. The VSOP-99 model approximation, where the fuel being reloaded is averaged (or mixed) in the “storage box” before being returned to the core, is over-predicting the effect. A more even distribution of burnup values is expected in real life, but the general behaviour of an increase in spread with a decrease in the number of fuel-passes is still to be seen.

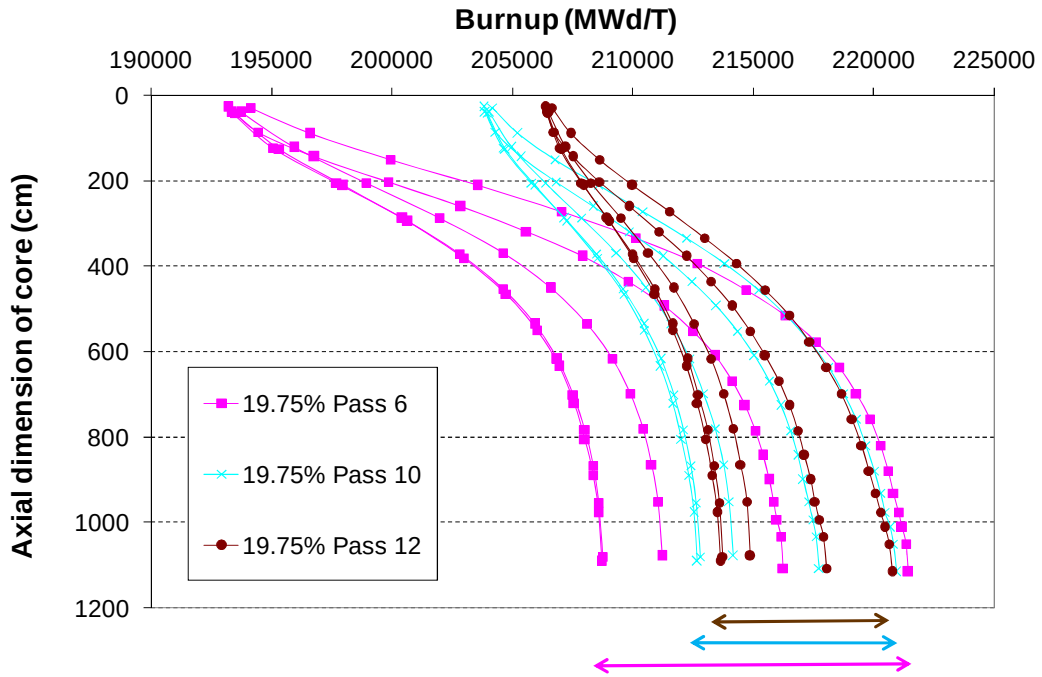


Figure 32: PBMR 400 MW discharge burnup distribution for the different number of fuel-pass models with 19.75% enriched fuel

6.5 CONTROL ROD POSITIONS

Like the HTR-Modul, the study was repeated for the PBMR 400 MW with regards to the varying control rod positions. For the PBMR 400 MW design 24 control rods are positioned 1.987m radially from the centre of the core (13.7 cm into the side reflector) and inserted 2 m axially below the top reflector [24].

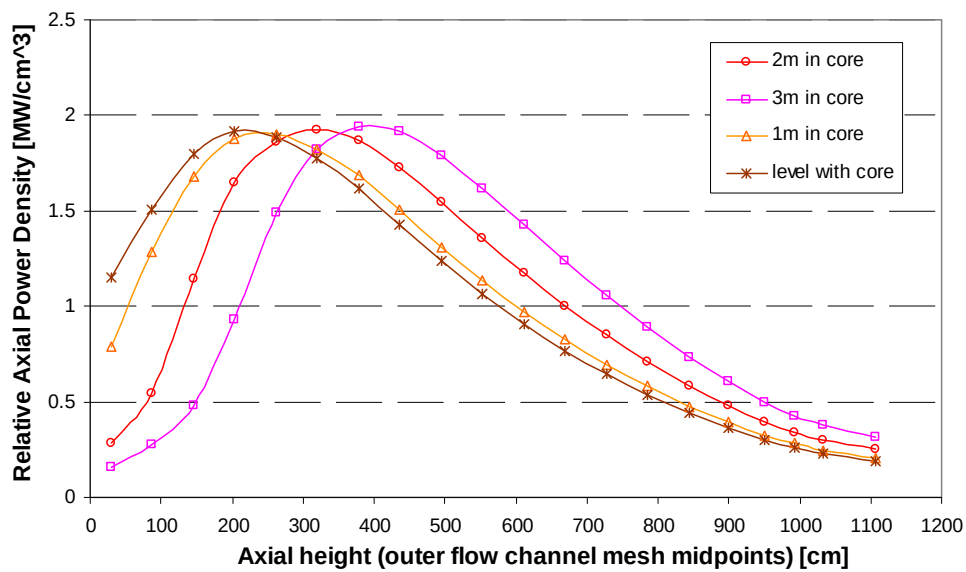


Figure 33: PBMR 400 MW axial relative power density profile for the different control rod position cases

For comparative purposes, the same procedure was followed by inserting the control rods 1 m below the designed position. The control rods were lifted in 1 m increments until they were 2 m above the original designed position, which in this case would be just out of the core. These results were then compared.

Figure 33 shows that the power density peaking shifts towards the bottom of the core as the control rods are inserted deeper into the core. A slight increase in the power density peaking can also be observed. This differs from the HTR-Modul as the control rods that were inserted deeper do not tend to flatten the power profile. This is due to the strong absorbing effect of the control rods as there are 24 control rods in the PBMR 400 MW. These control rods are also positioned closer to each other than the six control rods in the HTR-Modul. The core (85 cm annulus) is also much thinner.

The average fuel temperatures decrease (Figure 34) as the control rods are inserted into the core with a slight rise of less than 10 °C in the maximum average temperature.

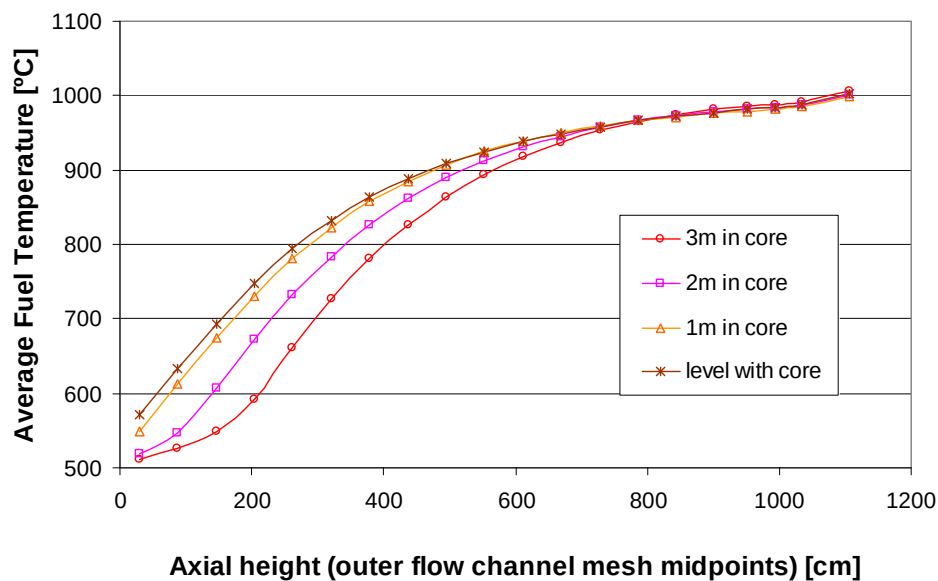


Figure 34: PBMR 400 MW axial average fuel temperatures for the different control-rod position cases

With a difference of 1152MWd/t, Table 17 shows that the decrease in the discharge burnup of the PBMR 400 MW is a less dramatic than that of the HTR-Modul. The use of two half-length banks of control rods (2x12) that overlap in the middle in the PBMR 400 MW design ensures a much larger hot shutdown margin, although the same trend of decreasing margin is shown. As with the HTR-Modul there is a rise in the maximum fuel temperature with a difference of 32 °C from fully withdrawn to full insertion conditions.

Table 17: Discharge burnup, control rod worth and shutdown margins for the different control rod position cases

Case	Average discharge burnup [MWd/t]	Control rod worth	Shutdown margin	Max. Fuel Temp. [°C]
0m	95987	8.19%	8.69%	1091
1m	95944	8.40%	8.65%	1095
2m	95571	9.13%	8.59%	1107
3m	94835	9.91%	8.37%	1123

6.6 FUEL ENRICHMENT VARIATIONS

The study for different fuel enrichments was repeated for the PBMR 400 MW, but only two cases were considered - the one with a 9.6% enrichment and another with a 19.75% enrichment. The latter represents the upper limit of the typical definition of low enriched fuel (LEU) [47]. Both cases were for a ten-pass fuel circulation.

As expected the relative results observed were similar to those of the HTR-Modul. Figure 35 shows the normalised power density, again indicating a higher peaking for the 19.75% enrichment. Also the local power peaking is 1.6 times higher than the 9.6% enrichment case (see Table 18).

Figure 36 shows that the average fuel temperature is lower for the 9.6% enrichment case and, contrary to what was observed for the HTR-Modul (see Table 14) where the maximum fuel temperature decreases slightly, the maximum fuel temperature is almost 150 °C higher for the 19.75% enrichment case in the PBMR 400 MW design (see Table 18). The higher peaking and higher starting fuel temperatures might challenge the safety design principle of the PBMR 400 MW (DLOFC fuel temperature below 1600 °C).

As with the HTR-Modul, the average discharge burnup doubles as the enrichment value doubles. There is a decline of 1.2% of the neutron leakage in the higher enrichment case as shown in Table 18.

Table 18: General information on the varying enrichment cases for the PBMR 400 MW

Case	Average discharge burnup [MWd/t]	Neutron leakage	Power peaking	Maximum fuel temperature [°C]
9.6%	95571	14.6%	3.23	1107
19.75%	214186	13.4%	5.15	1254

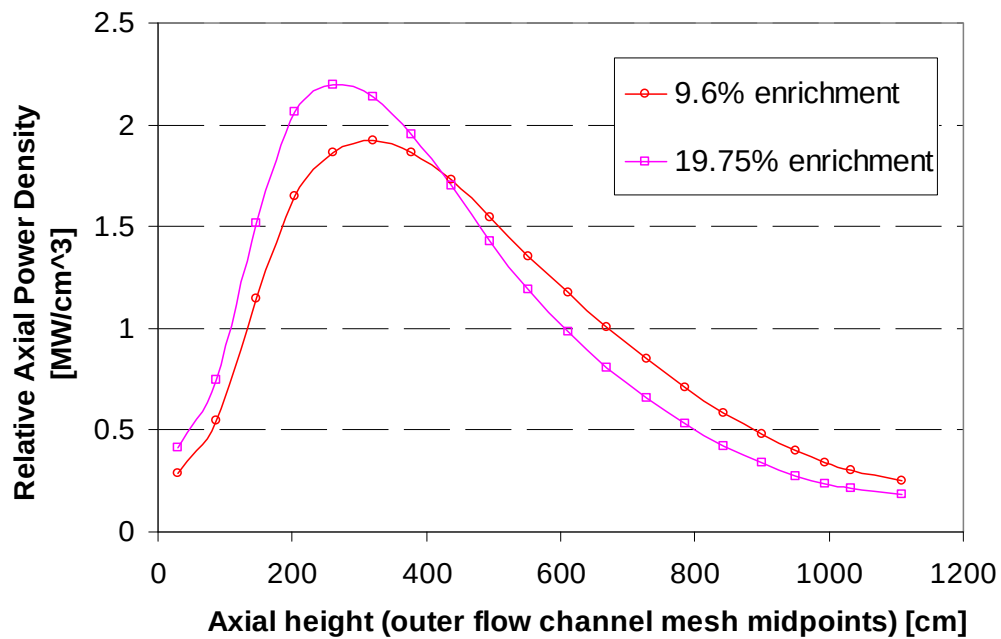


Figure 35: The PBMR 400 MW axial relative power density profile for different fuel enrichments

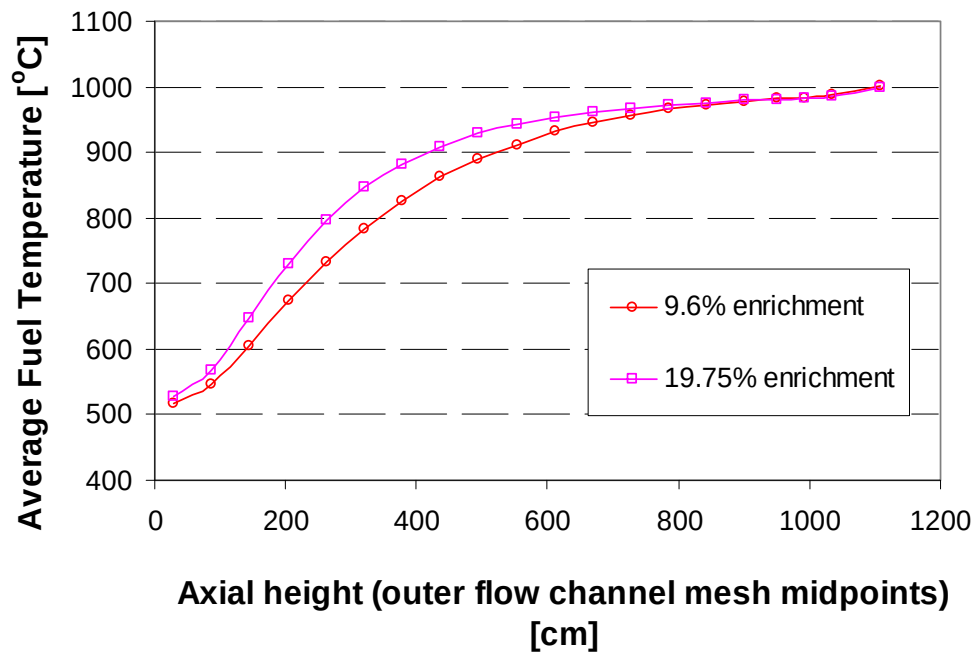


Figure 36: PBMR 400 MW axial average fuel temperature for different fuel enrichments

7. CONCLUSION AND RECOMMENDATIONS FOR FURTHER WORK

7.1 CONCLUSIONS

The primary objective of this study, evaluating the different multi-pass fuel cycles for the different reactor designs, have been fulfilled.

The created reference models for the HTR-Modul and the PBMR 400 MW was compared to previous published work. Comparison showed that the HTR-Modul reference model temperature results are pessimistically calculated in this study and for the PBMR 400 MW reference model the temperature results are optimistically calculated.

By comparison, the multi-pass case study of the HTR-Modul showed that, with the increase in the number of fuel passes from a four-pass case, the results in general improved with regards to the power and temperature profile, the discharge burnup and DLOFC temperatures. Results of the PBMR 400 MW showed the same tendency. In the case of a higher multi-pass number the burnup qualification target is closer to the expected average discharge burnup and therefore it is more beneficial in terms of fuel qualification effort and costs.

The following conclusions with regards to the primary objective can be made with the increase in number of fuel passes and in comparison with the HTR-Modul and PBMR 400 MW designs:

- For the HTR-Modul, the discharge burnup increased with 17 311 MWd/t from one to 15 passes, but only with 1099 MWd/t from six to 12 passes. This can be compared to the PBMR 400 MW where it increased with 1929 MWd/t.
- The neutron leakage decreased for the HTR-Modul with 3.5% from one to 15 passes and 0.2% from six to 12 passes. A slight increase of 0.2% is shown for the PBMR 400 MW.
- The maximum fuel power (per sphere) decreased for the HTR-Modul with 0.83 kW from one to 15 passes and 0.07 kW from six to 12 passes, which is equivalent to the PBMR 400 MW.
- The average and maximum fuel temperatures decreased with the lowest values at 15 passes for the HTR-Modul. A difference of 6 °C is shown for the maximum fuel temperature between six and 12 passes. This is opposite to that of the PBMR 400 MW which shows a slight increase in the maximum fuel temperature of 5 °C but a decrease in the average fuel temperatures.
- In general the DLOFC maximum and average temperatures also decreased except for the first and second pass case where the average temperatures were lower due to the

depleted fuel spheres at the bottom of the core causing less decay heat (and a large volume that cools down quickly decreasing the average).

- For the HTR-Modul and the PBMR 400 MW there is a decrease in the decay time and the available measuring time. Increases in both these times will improve the accuracy of the burnup measurement system. The 15-pass case of the HTR-Modul adheres to the minimum specified requirements (by design) while all other cases with fewer passes surpasses this. The PBMR 400 MW has three defuel chutes to allow more time for doing measurement. Detail results are shown in Appendix 9.1.
- The expected spread in the fuel sphere burnup decrease for the HTR-Modul and PBMR 400 MW. This is beneficial for the fuel qualification as it can reduce discarded fuel costs.

By comparing the profiles and evaluating the two designs from a multi-cycle fuel circulation point of view, it shows that in most cases the increase in the number of fuel passes not only flattens the power profile, but also improves the overall results. The results improvement decreases exponentially and from ten passes the advantage of having more passes is getting less significant. The effect of the flattened power profile is more visible on the PBMR 400 MW than on the HTR-Modul. The 15-pass HTR-Modul design is at its maximum as it is limited by the measuring time of the burnup measuring system. The PBMR 400 MW has three defuel chutes allowing longer decay and burnup measurement times which improves measurement accuracy, and, should still have the capacity to increase the number of passes. The power peaking results did still improve significant after six passes and could thus benefit from an increase number of passes without an increase in the fuel handling costs (if the HTR-Modul requirements is taken as being representative)

The secondary objective of performing a sensitivity analysis on the control rod insertion positions and the effect of higher fuel enrichment has also been achieved. Control rod efficiency is improved when increasing the excess reactivity by means of control rod insertion. This is however at the expense of the discharge burn-up and shut down margins. The HTR-Modul's power profile is more flattened by this effect while for the PBMR 400 MW a simple downward power shift is seen with no significant effect on the maximum relative power. The postulated control rod induced reactivity excursions is however aggravated and would thus disqualify the use of the control rods as a way to decrease the axial peaking.

The following conclusions can be made with regards to the HTR-Modul and PBMR 400 MW control rod position study:

- By withdrawing the control rods, it improved the average discharge burnup with 1119 MWd/t for the HTR-Modul and only with 416 MWd/t for the PBMR 400 MW.

- The withdrawal of the control rods leads to higher average temperatures but lower maximum temperatures
- The control rod worth increased with 0.06% and 0.72% by inserting the control rod 1 m deeper in the core, but the shutdown margin decreased with 0.66% and 0.22% for the HTR-Modul and the PBMR 400 MW respectively.

Higher enrichment causes an increase in power peaking and more fuel-passes will be required to maintain the peaking and temperature margins as before. The next summary is made in concluding the enrichment study:

- The average discharge burnup almost doubled to 110 024 MWd/t as the enrichment increased from 6% to 10% for the HTR-Modul. The PBMR 400 MW showed similar results as the burnup doubled to 214 186 MWd/t for an enrichment that increased from 9.6% to 19.75%.
- On the HTR-Modul, the power peaking increased with 0.74% for higher enrichment and with 1.92% for the PBMR 400 MW.
- The maximum fuel temperature decreased with 6 °C for the HTR-Modul, but increased with 147 °C for the PBMR 400 MW.
- The maximum and average fuel temperature during a DLOFC event is within the accepted limits for the higher enrichment (10%) for the HTR-Modul.

The detailed comparison of the results on the evaluation criteria is given in Appendix 9.1 and all the flux, power and temperature profiles is shown in Appendix 9.2.

Analyses of the number of fuel passes are important when performing reactor concept design. It is not only significant for improving initial reactor performance in terms of fuel management and reactor safety but it also impacts the RCS, BUMS and FHS designs and could require design changes that can become very costly due to stringent requirements from the core design on these systems. Careful consideration must be given when designing the system as a whole.

7.2 RECOMMENDATIONS FOR FUTURE WORK

For future work a more detailed investigation can be performed on the statistical probability of a spent fuel element being erroneously circulated by the burnup measuring system. A study of the impact of the additional burnup acquired if a fuel sphere is circulated again and the effect of the burnup measurement accuracy on such probabilities will make a valuable contribution. A lower burnup measurement accuracy caused by shorter decay and measurement times for more fuel-passes, may discount many of the advantages of models with more passes.

For further investigation on fuel qualification regarding the burnup spread, it must be considered that VSOP-99 has a limitation when reloading the batch fuel as it averages the values in the storage boxes. This is acceptable for this study as it points out the effect, but for a specific detailed analysis on this topic, the limitation should be corrected. This can easily be done by adding trace amounts of fuel as batches and then following actual histories. Alternatively, the methodology used in VSOP-99 will have to be generalised.

Fuel performance calculations would be complimentary to this study in order to show evaluation analysis on fission product releases. It should also be interesting to attempt quantifying the impact of higher dust generation due to the increased number of passes. The trade-off with added burnup should be of interest.

8. REFERENCES

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