The Reactor Core Neutronic model for the Pebble Bed Modular Reactor

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\textbf{ABSTRACT}
This paper describes the technical aspects of the Reactor Core Neutronic model for the Pebble Bed Modular Reactor (PBMR). Included is a model design review with preliminary simulation results and model constraints. The PBMR Demonstration Power Plant is a First of a Kind Engineering plant which will be used for the production and generation of electricity in South Africa.

The theory and solution techniques used for modelling and simulating the neutronic core are also described. The neutronic model is discussed, as well as the model capabilities and model requirements. The model formulation for the PBMR plant is also derived from GSE’s nuclear (neutronic) simulation model known as REMARK\textsuperscript{©} (Real Time Multigroup Advanced Reactor Kinetics). The derived neutronic model for PBMR is aptly called the PBMR-REMARK\textsuperscript{©} Reactor Core Neutronic model. Preliminary results of the Reactor Core Neutronic model simulations are included and discussed in the paper.

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1. Introduction

1.1. The PBMR plant-technological basis
The PBMR reactor design is primarily based on German High Temperature Reactor (HTR) Technology developed over the past three decades. (A nuclear reactor (heat source) and a direct helium gas turbine set coupled to a generator are used as a power source to generate electricity (Ott and Neuhold, 1985).) A complete description of the PBMR plant can be viewed in Matzner (2004). Fig. 1 is a Uni-Graphix representation of the PBMR Demonstration Power Plant power plant, displaying the main components of the plant. On the left in Fig. 1 is the reactor vessel, and in the horizontal position is the power turbine, which is connected to the electric generator by a differential gearbox. A large ‘shut-off’ valve is used to shut off and separate the two systems (nuclear reactor and Power Conversion Unit) during maintenance. At rated power, heated helium gas flows from the bottom of the reactor vessel at 900 °C to the power turbine at a rate of 190 kg/s. The helium gas exits the power turbine and then passes through a series of coolers and compressors before re-entering the reactor vessel at 500 °C. The recuperator is a complex gas heat exchanger, in which the exiting hot energy-sent helium gas from the power turbine is used to heat up the

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The PBMR-REMARK© model incorporates the following approaches in a subroutine:

(a) Improved quasi-static solution approach to obtain neutron flux distributions.
(b) Six-group delayed neutrons calculated in each mesh cell.
(c) Four-neutron energy groups are used.
(d) Concentrations of four isotopes calculated in each mesh cell: I-135; Pm-149; Xe-135; Sm-149.
(e) Reactivity feedback based on core thermal–hydraulic conditions of fuel temperature and moderator temperature.
(f) Reactivity feedback effects of the Reactivity Control System and the Reserve Shutdown System.
(g) Twenty-three decay heat groups are used to calculate decay heat contributions from U-235, U-238, and Pu-239, based on fission power history. The fraction of total decay heat from each isotope depends on core exposure.

Advanced features incorporated into the model include modelling and calculations for:

(a) Burn-up of fission isotopes for U-235, U-238, Pu-239, Pu-240 and Pu-241.
(b) Determination of fission product isotopic concentrations for Ag-109, Ag-110m, Cs-133, Cs-134, Cs-135, Cs-136, Cs-137 and I-133.
(c) The prediction of critical rod positions. The prediction of critical rod position is to determine the Reactivity Control System rod positions of criticality based on the current core conditions. The following approaches are implemented in a subroutine:
   (i) The prediction is done in parallel to the real-time core physics simulation.
   (ii) At each time point, the core conditions are the base for the prediction calculations.
Fig. 3 is a schematic for the sequence of execution of the PBMR-REMARK© model modules. Shown in Fig. 3 are the inputs for the model calculations and the interaction with the Reactor Core Thermal–hydraulic model. Not indicated in the figure is the output for the neutron detector readings. This output is not shown, since the output values are closely related to the Control and Instrumentation functions of the plant, and are not necessary for the preliminary discussions.

2.2. Model capabilities

The core physics history is available at any given point in time during the simulation operation with respect to model capabilities ((a) to (g)), as follows:

(iii) Pseudo Reactivity Control System positions are moved, without affecting the actual positions, to search for the critical positions for banks 1 and 2 of the Reactivity Control System. Banks 1 and 2 are moved at the same time.

(iv) When no critical positions can be found, the predicted rod positions will be displayed.

(d) Control rod insertion limit. The calculation of control rod insertion limit is based on the following definition and approaches:

The control rod insertion limit calculates the insertion depth of Reactivity Control System Bank 2 of maximum reactivity worth. This calculation is only to be performed in core model stand-alone mode to calculate the insertion limit based on fixed Thermal–hydraulic conditions. When the insertion limit is performed, the rods are inserted until the slope is determined and when \( k_{\text{eff}} \) starts to become positive while the Bank 2 is still inserted, hence the maximum reactivity worth is reached. When this phenomenon occurs it is known as the “horse-shoe” effect (Section 4.2).
Fig. 3. Sequence of execution of the PBMR-REMARK© model modules.

(a) $k_{eff}$ is calculated and available to track for each cycle.
(b) Neutronic fission power is calculated in every mesh and can show power profile on the mesh geometry.
(c) Isotopic inventory concentrations are calculated for I-135, Xe-135, Pm-149, and Sm-149 for shutdown and start-up scenarios.
(d) A Xenon transient may be calculated for the current Xenon-worth based on power history. The Xenon (number) density level during a transient following a planned shutdown (or a trip), as well as the Xenon transient during start-up based on power history and for various power increase ramp rates is available. A similar result is available for Samarium concentration changes during a transient.
(e) Control rod position. PBMR-REMARK© takes into account the Reactivity Control System and Reserve Shutdown System control rod positions. Each control rod is modelled individually, so the model can reflect single or multiple rod movement.
(f) Transients originating from a Power Conversion Unit transient response to various Power Conversion Unit faults (compressor faults, bypass valve or control valve malfunction). This also includes effects due to loss of control supplies, e.g. helium gas, loss of cooling water supplies, increase in cooling water temperature, and water ingress. The neutronic model will respond to transients originating from the Power Conversion Unit through the changes of moderator/coolant temperature in the Reactor Core Thermal–hydraulic.
(g) Perform anticipated transient simulations such as:
(i) Single or multiple control rod ejection and insertion.
(ii) Fuel loading malfunctions; this includes different fuel enrichment compositions, and fuel at different compositions, as required during the start-up process. Fuel enrichment at steady state is expected to be 9.6%.
(iii) Helium flow disturbances and adjustments; such information will be received from the interaction with the thermal–hydraulic model.
(iv) Scram mode operation, Reactivity Control System insertion and Reserve Shutdown System insertion.
(v) Decay heat level following shutdown may be calculated.
(vi) Power increases (various ramp rates) at various times after shutdown, with particular emphasis on control rod behaviour (to determine if any combination of ramp-rate/power history will cause power control problems). The user may introduce power increases (various ramp rates) at various times after shutdown, with particular emphasis on control rod behaviour to check responses.
(vii) Shutdown rod position during restart may be initialized via an Initial Condition set of parameters. This action may be performed through an Instructor Station mode operation (Dudley et al., 2006b).

2.3. The neutronic model

2.3.1. The PBMR-REMARK model equation

The four-group time-dependent, 3D, diffusion equation (Neutronic Model Fundamental equation (1)) is used as the initial neutronic equation (Oett and Neuhold, 1985):

$$\frac{1}{v_g} \frac{\partial \Phi_g(r,t)}{\partial t} = \nabla D_g(r,t) \nabla \Phi_g(r,t) - \Sigma_{a,g}(r,t) \Phi_g(r,t)$$

$$- \sum_{g' = 1, g' \neq g}^{4} \Sigma_{s,gg'}(r,t) \Phi_{g'}(r,t)$$

$$+ \sum_{g' = 1, g' \neq g}^{4} \Sigma_{s,g'g}(r,t) \Phi_g(r,t)$$

$$+ (1 - \beta) \sum_{g' = 1}^{4} v_g \Sigma_{l,g'}(r,t) \Phi_{g'}(r,t)$$

$$+ v_g \sum_{i=1}^{6} \lambda_i c_i(r,t) + v_{ext,g} S_{ext}(r)$$

(1)

For $g = 1, 4$: $r$ is the space in $\theta$–$r$–$z$ geometry (refer to Fig. 4) and $t$ is the time.
<table>
<thead>
<tr>
<th>Symbol</th>
<th>Definition</th>
<th>SI/metric units</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\lambda_i$</td>
<td>Decay time constant of group I decay heat</td>
<td>s$^{-1}$</td>
</tr>
<tr>
<td>$\nu$</td>
<td>Neutron generation per fission</td>
<td>—</td>
</tr>
<tr>
<td>$\Phi$</td>
<td>Neutron flux</td>
<td>cm$^{-2}$ s$^{-1}$</td>
</tr>
<tr>
<td>$D$</td>
<td>Diffusion constant</td>
<td>cm</td>
</tr>
<tr>
<td>$r$</td>
<td>Space</td>
<td>cm</td>
</tr>
<tr>
<td>$S_{ext}$</td>
<td>External source</td>
<td>cm$^{-1}$ s$^{-1}$</td>
</tr>
<tr>
<td>$t$</td>
<td>Time</td>
<td>s</td>
</tr>
<tr>
<td>$g$</td>
<td>Refers to neutron group $g$; $g = 1, 2, 3, 4$</td>
<td></td>
</tr>
<tr>
<td>$4$</td>
<td>Refers to Group 4; thermal neutron group</td>
<td></td>
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<tr>
<td>$l$</td>
<td>Dimensional index</td>
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<td>$l'$</td>
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<tr>
<td>$k$</td>
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<td></td>
</tr>
<tr>
<td>ext</td>
<td>External</td>
<td></td>
</tr>
<tr>
<td>$\Sigma_1$</td>
<td>Macroscopic absorption cross-section</td>
<td>cm$^{-1}$</td>
</tr>
<tr>
<td>$\Sigma_f$</td>
<td>Macroscopic fission cross-section</td>
<td>cm$^{-1}$</td>
</tr>
<tr>
<td>$\Sigma_s$</td>
<td>Macroscopic scattering cross-section</td>
<td>cm$^{-1}$</td>
</tr>
</tbody>
</table>

A complete in-depth discussion on the neutronic model can be found in Dudley et al. (2006c).

2.3.2. Model assumptions and simplifications
2.3.2.1. Four neutron energy groups. The neutron energy spectrum is divided into four groups. The neutron energy of the four groups is defined as per energy levels:

(a) Group 1: $>0.1$ MeV.
(b) Group 2: $0.1$ MeV–$29$ eV.
(c) Group 3: $29$–$1.86$ eV.
(d) Group 4: $<1.86$ eV.

The four neutron group approach is considered to be adequate for the PBMR plant.

2.3.2.2. All neutrons from fissions are fast neutrons. The neutron emission spectrum ($\chi$) is 0 for (c) and (d). The neutrons from spontaneous fission of external sources are also fast neutrons and use the same emission spectra.

2.3.2.3. All delayed neutrons are fast neutrons. The delayed neutrons are considered to be fast neutrons and use the same emission spectra as the fission neutrons.

2.3.2.4. Six delayed neutron precursor groups. There are approximately 40 delayed neutron precursors out of the 500+ fission product nuclides. Since not all of the delayed neutron precursors have known properties, it is both impractical and inaccurate to calculate the delayed neutrons directly from all the precursors.

Therefore, the overall delayed neutron source rate was experimentally determined as a function of time after exposing a sample of fissionable material to a very short neutron pulse. It was then shown (Ott and Neuhold, 1985) that this delayed neutron source rate can be accurately represented by just six exponential functions. The six group delayed neutron precursors are thus commonly used in neutron kinetics and sufficient for the modelling of the Reactor neutronic’s for the PBMR plant.

2.3.2.5. Decay heat–23 groups. Based on ANSI/ANS-5.1-1979 (ANSI/ANS, 1979), the decay heat in reactors is represented by 23 groups for decay heat from U-235, U-238 and Pu-239. The contribution from each isotope is a function of fuel exposure.

2.3.2.6. Each mesh cell is homogeneous. In each mesh cell, the neutron cross-sections represent the homogeneous properties of all the materials within the cell (as illustrated in Fig. 4). Fig. 4 illustrates the 3D geometry for each mesh cell. The Laplacian operator is shown in Fig. 4 for sector mesh calculations.

2.3.3. The moderator temperature effect

The moderator temperature dependency of cross-sections is represented by a similar equation to the Doppler correction equation shown in Eq. (2) (moderator temperature calculation)

$$
\Delta \Sigma_{a,1} (\text{Doppler}) = \frac{d \Sigma_{a,1}}{d \sqrt{T_f}} \left( \sqrt{T_f} - \sqrt{T_{f,\text{ref}}} \right)
$$

where $\Delta \Sigma_{a,1}$ is the change in macroscopic absorption cross-section, $\Sigma_{a,1}$ is the macroscopic absorption cross-section, $T_f$ is the fuel temperature, $T_{f,\text{ref}}$ is the reference fuel temperature (provided with cross-section data).

2.4. Modelling data requirements

Modelling data requirements were sourced from PBMR as requested by GSE’s technical development team. Data exported was in the form of Data Requests, which included verification of data that was generated. Data that was not available was taken from citation material (citation material refers to internationally published documents), which was found to be acceptable for the modelling requirements. Data generated was from different nuclear codes such as VSOP (Teuchert et al., 1994) and

$$
\mathbf{\nabla} = \left( \begin{array}{c}
\frac{\partial}{\partial r} \\
1 \frac{\partial}{r \partial \theta} \\
\frac{\partial}{\partial z}
\end{array} \right)
$$

Fig. 4. Three-dimensional geometry for each mesh cell.
The Reactivity Control System and Reserve Shutdown System effect on the cross-sections will be considered with the following delta cross-section as in Eq. (3).

$$
\Delta \sum_{\text{Reactivity Control System or Reserve Shutdown System}} = \sum_{\text{No Reactivity Control System or Reserve Shutdown System}} - \left( \sum_{\text{w/ Reactivity Control System or Reserve Shutdown System}} \right)
$$

Within each mesh-cell, $\Delta \Sigma$ (Reactivity Control System or Reserve Shutdown System) is the total cross-section impact of the Reactivity Control System or Reserve Shutdown System in that mesh-cell.

The volume fraction affected by the Reactivity Control System or Reserve Shutdown System in each mesh-cell is calculated based on the Reactivity Control System or Reserve Shutdown System positions. The $\Delta \Sigma$ (Reactivity Control System or Reserve Shutdown System) is weighted with the controlled volume fraction and added into the base cross-section.

### 3. Neutronic model interfacing with other models

The PBMR-REMARK© model interfaces with other systems, namely the Reactor Core Thermal–hydraulic, Power Conversion Unit, Reactivity Control System and the Reserve Shutdown System. A description of each interface is given in the following sections. The Reactor Core Neutronic model has a direct interface with the Reactor Core Thermal–hydraulic model. Information sent to the Reactor Core Thermal–hydraulic model consists of the heat flux profile, and in return the Reactor Core Thermal–hydraulic model sends the heat slab temperature distribution to the Reactor Core Neutronic model for re-calculation. Data communication between the Reactor Core Neutronic model and Reactor Core Thermal–hydraulic model is as shown in Fig. 5.

#### 3.1. The Interface between the Reactor Core Neutronic model and the Reactor Core Thermal–hydraulic model

##### 3.1.1. Connection links from the Reactor Core Thermal–hydraulic model to the Reactor Core Neutronic model

In computing reactivity feedback calculations, the PBMR-REMARK© Reactor Core Neutronic model requires the following information from the Reactor Core Thermal–hydraulic model:

1. **Moderator temperature values.**
2. **Average Uranium sphere fuel temperature values.**

These temperature values are computed by the Reactor Core Thermal–hydraulic model, which models the 3D core geometry.
using lumped heat slabs for calculating heat transfer between the different regions in the Main Power System (Dudley et al., 2006d). The interfacing diagram is shown in Fig. 6, which represents the axial boundaries of the thermal–hydraulic heat slabs and the connection mapping to the Reactor Core Neutronic model axial meshes, shown as a schematic for illustrative purposes only. The thermal–hydraulic heat slabs are discussed in Dudley et al. (2006d).

First, the thermal–hydraulic heat slab temperatures are mapped into the core radial and axial nodes assuming symmetric angular distribution. Then, the radial ring temperature is weighted by the angular power to ring power ratio to obtain the temperature in each Reactor Core Neutronic model mesh.

With reference to Fig. 2, from the top view, there are 23 radial zones and 24 azimuthal sections (refer also to Fig. 4). In addition to this, there are 35 axial nodes. Multiplying these numbers gives an overall number of nodes of 19,320. These neutronic nodes are linked in a weighted ratio to the thermal–hydraulic slab nodes as shown in Fig. 6. Fig. 6 is a section of the thermal–hydraulic heat slab mapping extracted from the Reactor Core Thermal–hydraulic model (Dudley et al., 2006d).

3.2. Other interfaces with the Neutronic model

As illustrated in Fig. 3, the Reactor Core Neutronic model interacts with the Reactivity Control System and the Reserve Shutdown System. This interface is currently only implemented as internal Reactivity Control System and Reserve Shutdown System drivers in the PBMR-REMARK© Reactor Core Neutronic model. Information on the position of the Reactivity Control System or filled length of the Reserve Shutdown System is sent from the Reactivity Control System/Reserve Shutdown System Drive System to PBMR-REMARK© Reactor Core Neutronic to calculate the reactivity impact. In addition, neutronic core power levels and flux readings are sent to Reactivity Control System/Reserve Shutdown System controllers. This is mainly for Control and Instrumentation implementation.

4. Reactor Core Neutronic model simulations

It would be of interest to the Modelling Engineer to examine the robustness of the model under extreme cases of simulation such as rapid rod withdrawal or rapid rod insertion. Of course, such extremities are not expected in real life for the reactor plant control operations. However, there is the case of a reactor operating at steady state, when for some unknown reason there is a rapid one rod ejection or multiple rod rapid withdrawal. Such case studies are needed and results should be interpreted with an open mind (i.e. a model test of inserting the control rods to a maximum insertion limit shall not be done in reality, merely as a modelling verification process). Alternatively, a simple modelling test is to run the plant at steady state, 100% power output, and then to insert the groups of control rods as would normally be operated. The observation of $k_{\text{eff}}$, neutronic power changes, and poisoning effects would be noted. This type of simulation for any nuclear plant is standardized and easily comparable with simple verification tools, such as using other nuclear codes to generate the simulation. Part of the PBMR plant’s uniqueness is in the control of the reactivity by the operation of the control rods and for complete shutdown, by inserting thousands of small neutron absorbing spheres (Reserve Shutdown System) in conduits in the core. Besides the inherent safety of the neutronic core, i.e. it can be shown that if there is a loss of forced
cooling, the neutronics, due to the negative temperature effect, will shut itself down. This has been shown in the Thermal–hydraulics simulation result as a Depressurized Loss of Forced Cooling (DLOFC) (Dudley et al., 2006d).

Construction of the PBMR plant will start in 2008. The projected Fuel-on-site date is 2013. This implies that operational plant data for the period (2007–2013) is not available for validation purposes. The Verification and Validation procedure plan (PBMR, 2007) is to verify the models in a structured step-wise process involving all the parties concerned. The final Verification and Validation on the plant model will be completed once real plant data is available. During the interim period, comparable results of selected simulation scenarios are available. There are simulation results for the reactor core neutronics, which are obtainable from codes such as the VSOP software tool and TINTE codes.

4.1. Simulation—insertion of 24 control rods and small absorber spheres from a position of steady state 100% power output

This simulation for the Reactor Core Neutronic model demonstrates the PBMR-REMARK© capabilities. The results are displayed in Fig. 7. Fig. 7 is a snapshot of the simulation result taken from the Jade suite of simulation software from GSE Systems (Dudley et al., 2006b). The model is initially at steady state with a 100% power output. There are 24 control rods, grouped alternately into banks of 12 rods. When the control rods in Group 1 are inserted into the core, the neutronic power and $k_{\text{eff}}$ decrease. This action is then followed by the total insertion of the remaining control rods (Group 2). A rapid decrease in neutronic power is observed. After normalization has been achieved (i.e. steady state), Small Absorber Spheres from the Reserve Shutdown System are released into the absorber sphere conduits. (The insertion of the Small Absorber Spheres has the effect of a reactor shutdown.) Although the Small Absorber Spheres are released from the top of the reactor vessel (and filling of the conduits under the effect of gravity), the effect is to fill the conduits from the bottom up. A rapid decrease in neutronic power and $k_{\text{eff}}$ is observed. During this transient, there is a gentle increase in Xenon (poisoning effect). Throughout the simulation, the PBMR-REMARK© model remains stable. This simulation demonstrates the robustness of the model.

4.1.1. Model simulation scenario, poisoning effect—discussion

Suppose the reactor, after running at steady state, is shut down as fast as possible from a high power level operating condition. The observation is that the thermal-neutron level is effectively reduced to zero, and consequently the decay term due to thermal-neutron absorption of the Xe-135 no longer exists. Consequently, the Xe-135 concentration builds up to a maximum from which the I-135 has been previously formed. Ultimately, the radioactive decay of the Xe-135 to Cs-135 takes over, and the total Xe-135 concentration drops off. The time involved in this process can be shown for a peak in Xenon concentration appearing in approximately 11 h after shutdown. The magnitude of this peak again depends upon the initial steady power level of the reactor and its specific design. However, the peak Xenon poisoning may be many times the value of the equilibrium Xe-135 poisoning. It will be recognized that an entire range of poisoning conditions can exist, depending upon the magnitude of the initial power level and the extent to which the power level is shut down.

4.2. Simulation—the inflection of $k_{\text{eff}}$ when rods are theoretically inserted to maximum depth

In another simulation test, the effect is demonstrated. The 24 control rods are inserted into the core to their maximum depth, whilst at the same time observing the change in $k_{\text{eff}}$ throughout the test. A classical model simulation result using the VSOP nuclear design code is depicted in Fig. 8, where the control rods are initially in a fully withdrawn position. The simulation is performed with an iso-deltic temperature assumed at 100 °C and the value of $k_{\text{eff}} = 1.015$ as indicated. The 24 control rods are then inserted to a depth of 150.9 cm, which is seen to be the steady state position.
of the 24 control rods during equilibrium operation of the reactor. Here the value of $k_{\text{eff}}$ is 1.000. The Reactivity Control System is then operated such that Group 1 (12 control rods in alternating positions when viewed from above) is inserted to its maximum position of approximately 570 cm below the flattened top of the pebble bed. Note that the value of $k_{\text{eff}}$ decreases from 1.00 to 0.97 as the Reactivity Control System is further inserted and Group 1 has reached its deepest position. During the simulation, Group 2 of the control rods (12 control rods at alternating positions to Group 1) is further inserted until the maximum insertion point is reached at approximately 1021.5 cm below the flattened surface of the pebble bed.

The value of $k_{\text{eff}}$ continues to decrease to a minimum value of 0.925. In the event of further insertion, the characteristic curve now displays the effect, whereby $k_{\text{eff}}$ starts to increase again to a value of 0.93. The result obtained should be comparable with other nuclear codes such as VSOP. Of course, such a simulation is not to be performed for training or test purposes, but merely as a verification with other codes that the new neutronic model behaves accordingly, and is comparable with other neutronic codes. Note that the top of the reactor vessel is at position 0 cm, and as the control rod travels into the reactor vessel, this motion and distance are in a positive direction.

### 4.3. Simulation—insertion of neutron absorbing small absorber spheres into the reactor core

In this simulation, the reactor neutronics is assumed to be at a specific stable cold condition of 100 °C. All 24 control rods are in a position such that a value for $k_{\text{eff}}$ of 1.075 is obtained. For the model simulation, the reactor core in this state, Small Absorber Spheres are inserted into the eight channels housed within the fixed central column region, under the free fall of gravity.

These Small Absorber Spheres are constructed of a mixture of graphite and a neutron absorbing material (Boron Carbide, B$_4$C). This system provides a diverse, independent means to shut down to cold conditions. The Small Absorber Spheres are poured in from containers within the top region of the reactor vessel, but filling will be done under positive pressure difference from the bottom of the reactor vessel, so as to break the fall of the spheres.

This system is known as the Reserve Shutdown System. Hence from Fig. 9 it can be seen that as the tubes for the Reserve Shutdown System are filled from the bottom up, the effect on $k_{\text{eff}}$ is much more pronounced, as the Reserve Shutdown System channels are filled and reaches almost two-thirds of the axial height position. The model shows that $k_{\text{eff}}$ is 1.07 at a filled height of 560 cm from the bottom. As the chutes are filled, $k_{\text{eff}}$ drops off dramatically to 0.97. This is a characteristic curve for the Reactor Core Neutronic

![Fig. 8. Reactivity Control System characteristics with equilibrium temperature distribution using the VSOP nuclear code.](image)
model and is comparable with other nuclear codes under the same conditions.

4.4. Simulation results discussion

4.4.1. Simulation case study comparable to 4.2

In this simulation, a comparable test is performed utilizing the new PBMR-REMARK© neutronic model, and performed under the same conditions as the VSOP model simulation for case study 4.2. When all the control rods have reached the maximum insertion depth, an effect can be observed with $k_{\text{eff}}$. The PBMR-REMARK© simulation result is shown in Fig. 10. This is a characteristic signature print for the pebble bed type of reactor.

4.4.2. Explanation of the effect on $k_{\text{eff}}$

When both banks of control rods are inserted, the first bank will stop at a point where its top has reached the bottom of the top reflector. Control rods in Group 2, however, have the ability to travel further into the core (downwards) to a prescribed depth.

When observing the region of control rod overlap (Fig. 8) towards the centre of the core height, as control rods in Group 2 travel deeper into the core, this overlap region becomes shorter. Since the flux/power distribution has its highest value in the region of the overlap (cosine distribution), the decreasing overlap will have an effect later on.

As the control rods in Group 2 reach a specific depth, the control rod 'bite' indeed becomes less significant than the loss of control rod 'bite' in the overlapping region. The flux then gets pushed into other regions of the core, leading to an increase in the overall core reactivity, which is the phenomenon known as the effect which is observed. This is a theoretical result (the effect), since technically it is not possible to insert the second group below its maximum insertion level. Otherwise the reactor would have a problem.

This is the reason why the insertion depth limit of control rods in Group 2 is 920 cm below the top rim of the reactor (refer to Fig. 2).

The simulation of the maximum control rod insertion is a test performed on the neutronic model, which would indicate the maximum reactivity worth (Fig. 11). This simulation is performed on the neutronic model in stand-alone mode to calculate the insertion limit based on a fixed thermal–hydraulic condition. When the insertion limit test is performed, the two groups of control rods are inserted until the slope of $k_{\text{eff}}$ turns. At this point of inflection of $k_{\text{eff}}$, the maximum reactivity worth is reached.

4.4.3. Simulation case study comparable to 4.3

In this simulation, using the PBMR-REMARK© neutronic model, Small Absorber Spheres are inserted into the Reserve Shutdown System tubes. The identical simulation is performed as in study case 4.3. A similar behaviour is observed as in the case of the VSOP result shown in Fig. 12.

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**Fig. 10.** JTopMeret representation of control depth penetration and the effect.  

**Legend Lines:**  
- **Black** $k_{\text{eff}}$  
- **Cyan** Neutronic Power  
- **Red** Control Rods Bank 1 / Position  
- **Blue** Control Rods Bank 2 / Position  
- **Magenta** Rod Insertion limit  
- **Orange** Small Absorber Sphere fill position

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**Fig. 12.** Reserve Shutdown System characteristics with Reactivity Control System fully withdrawn at 100 °C and Xe decay activated, simulation in the PBMR-REMARK© Reactor Core Neutronic model.
5. Conclusion

The PBMR-REMARK© Reactor Core Neutronic model was developed using the well-established REMARK© modelling tool for Pressurized Water Reactors and Boiling Water Reactors. The simulations presented for the neutronic model have been demonstrated to be robust and stable when the model has been operated under extreme and severe conditions. Further Verification and Validation tests will be performed before the Reactor Core Neutronic model can be accepted as part of the PBMR Plant Training Simulator. Preliminary results obtained from the PBMR-REMARK© Reactor Core Neutronic model simulations compare favourably with VSOP modelling data. This was demonstrated with the "horse-shoe" effect.

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