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U based fuel and maximum power size

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ABSTRACT

The decay heat removal capabilities are an important safety feature of the Modular Pebble Bed HTR. The reactor can be designed and optimised that even under loss of cooling accidents the decay heat can be removed by passive means without exceeding defined temperature limits for fuel and structure components. Such a design yields limitation of power output, however. The optimisation of power under constraints of limitation of fuel and structure temperature is an important task, therefore.

Several designs with cylindrical and annular cores were investigated assuming both Pressurised and Depressurised Loss of Coolant Accidents. The ZIRKUS program system was Used for the calculation of the neutronics and thermal-hydraulics calculations.

ZIRKUS contains the THERMIX-KONVEK thermal hydraulics module for the PLOCA and DLOCA analysis.

Annular Cores can produce higher power output without exceeding fuel temperature limits during DLOCA accidents, since heat storage effect of the inner fuel free column have an influence on the maximum fuel temperature by increasing the time at which this temperature is reached.
A thermal isolation of the core can decrease the maximum temperature of the pressure vessel during LOCA. Optimum is reached if both maximum temperature of fuel and RPV remains beyond corresponding limits.

1.Introduction

Increasing the power of the reactor unit is the most effective way to reduce the electricity costs of modular pebble bed High Temperature Reactor power plants [1]. Doing this while keeping the feature of inherent safety is a complex optimisation process of operating data, the core geometry and configuration of structures. The range of design parameters is limited by physical requirements (e.g. ability to shutdown with reflector rods, removal of the decay heat from the core by passive means, stability against Xenon oscillations ...) and technical-economical constraints e.g., (the size and manufacture technology of the reactor pressure vessel, technology used for compensating the pressure drop over the system etc.).

This resulted in diverse design concepts. The majors of the concretised concepts for the modular pebble bed HTR are summarised in Table 1. The HTR-MODULE [2] was the first reactor in which the concept of inherent safety was rigorously implemented. The PBMR-400 [3] represents the most current design.

Table 1: Overview of major design parameters for selected modular high temperature reactor concepts.

Reactor	HTR-MODUL	PBMR-302	PBMR-400
Thermal power [MW]	200	302	400
Core layout	Cylindrical	Annular core with dynamic middle column	Annular core with compact middle column
Outer diameter of core [m]	3	3.7	3.7
Inner diameter of core [m]	-	2	2
Height of core [m]	9.4	9.3	11
Diameter of RPV [m]	6	6.2	6.2
Inlet-/Outlet temperature [°C]	250 / 700	500 / 900	500 / 900
Coolant	Helium	Helium	Helium

The relevant design proposals in the literature can be classified to mainly three options with regard to the core layout:

- core with cylinder geometry,
- annular core with dynamic inner column of graphite pebbles,
- annular core with compact inner graphite column.

Here these diverse design options for the modular pebble bed HTR are discussed with respect to their performance during accidents with passive decay heat removal.

This is done by means of calculations using the ZIRKUS-System and THERMIX/KONVEK code [4, 5]. It models the steady state or time dependent thermal hydraulic behaviour of high temperature reactors in two-dimensional cylindrical geometry. The global solid temperature distribution and the coolant gas flow field are calculated, taking into account heat transport via heat conduction and radiation, as well heat transport and transfer by forced or natural convection. The modelling in THERMIX/KONVEK is based on a porous media approach which involves the solution of the time dependent energy conservation equation for the solid, the quasi-steady mass- and energy conservation equations as well as a simplified (Ergun-type with buoyancy) momentum conservation equation for the gas together with an appropriate set of constitutive equations, e.g. for effective heat conductivity, pressure drop and heat transfer.

In a first step, the characteristic behaviour during **P**ressurised and **D**epressurised **L**oss **O**f **C**oolant **A**ccidents (PLOCA and DLOCA) was analysed in detail. This is done for a reference reactor design with annular core, which was chosen here for comparison and benchmarking purposes. It does not directly correspond to an existing or actually proposed design, although major geometrical parameters are similar to other design proposals (derived from technical and physical constraints). The PLOCA and DLOCA cases are used to analyse and compare the design options with compact and dynamic middle column.

In order to investigate the potential of achieving higher power while bounding maximum fuel temperatures in order to ensure the safe inclusion of fission products, different core layouts, including cylindrical and annular core geometries, are then compared quantitatively. Analyses are done here for the DLOCA case only, since it imposes the strongest requirements on the decay heat removal capacities.

In a next step, the analysis was extended, in addition to the maximum fuel temperature, to include further criteria which determine the safety of the pebble bed HTR, particularly the maximum temperature to ensure the thermal integrity of the **R**eactor **P**ressure **V**essel (RPV). For the chosen design, it is shown that a thermal insulation is beneficial, in spite of its effect of increasing maximum fuel temperatures. As an example, it is demonstrated how the code analyses can be used to optimise design parameters (in this case insulation thickness) which maximise the achievable thermal power while fulfilling the constraints of maximum fuel and RPV temperatures.

2. Computer Codes and methodology

The ZIRKUS [5] program system was used developed by **FRAMATOME ANP** GmbH for HTR analysis. This program system has a modular structure and allows the coupled calculation of power distribution (neutronics), burn-up, refuelling and thermal-hydraulics. For thermal-hydraulics the **FRAMATOME ANP** version of THERMIX and IKE module FRECON [6] are used. For heat-up during a depressurisation accident also the SCALE/HEATING-7 [7] module and for calculation of control rod values the MORSE [8], KENO-VI [9] and MCNP [10] Monte Carlo codes will be used.

2.1 ZIRKUS modular system

The ZIRKUS modular system contains many modules for different tasks and a general data base for exchange of data between the modules. By means of a control monitor modular sequences can be formulated and executed in a flexible way. The main modules of ZIRKUS are following:

- Microscopic cross section processing module: MICROX
- Dancoff factors for double heterogeneous system: NEWA
- Initialisation of fuel data for burnup steps: KUGEL
- Initialisation of mesh nets for diffusion, thermal-hydraulics and burn-up calculations
- Calculation of macroscopic Cross sections: MAGRU
- Refuelling module for different strategies: NIVERM
- Neutron diffusion calculation: HBLOCK
- Calculation of bucklings for spectral zones: BUCK
- Burnup steps: SBURN
- Fast temperature calculations for spectral zones: NECKAR
- Detailed thermal-hydraulic analysis: THERMIX/KONVEK
- Interface for more accurate thermal-hydraulic calculations: VORNEK
- Calculation of afterheat for transients: NZW

A typical sequence of ZIRKUS calculations is shown in Tab.2. The input of every module in the sequence is pre defined as well as the data base transfers. A control sequence consists in control words subsequently calling corresponding modules. The monitor allows to execute different control sequences defined by names.

Table 2: ZIRKUS modular sequence for HTR calculations

KUGEL
NIVERM
NEWA
MICROX
MAGRU
HBLOCK
VORNEK
BUCK
NECKAR (THERMIX)
SBURN
NWZ

- KUGEL

  This module calculates the cell concentrations and the volume shares of each material, for every type of spherical fuel element with coated particles introduced in the reactor core.

- NIVERM

  Calculates the number densities for each cycle of each type of fuel element in core, the reload of the fresh fuel elements, the mean number densities for all burn up zones, the charge/discharge masses and the inventory of the nuclides. For every burn up zone and for every fuel cycle there are tree isotope vectors: one for the upper boundary of the burn up zone, one for the lower boundary of the burn up zone and one corresponding to the average values of the two previous vectors. This is repeated for each zone with the exception of the first axial upper zones where this vector is calculated over a whole burn up computation. These concentration records are then used for the calculation of cross sections for NEWA, the module which calculates Dancoff-factors and to MICROX and MAGRU, while the first two isotopes vectors are passed throw the burn up module for every cycle and are subsequently transferred back to NIVERM. The vector of the upper isotopes concentrations are burnt down in the flux of each zone while the vector of the bottom limit are burnt with the same flux of the adjacent zone below.

- MICROX

  The GAA code MICROX [11] calculates in a two zones model the fast slowing down density spectrum and the thermal flux density spectrum. In every zone the code solves the neutron slowing down and the thermalisation equation on a detailed energy grid of a lattice cell, where it considers two regions: one representing the fuel grains with coating and binder, and the other representing the moderator. The fluxes in the two regions are then coupled by collision probabilities based on the flat flux approximation A second level of heterogeneity can be treated, i. e., the fuel region may have grain structure up to two different types of fuel particles. The coupling between the single fuel elements is realised with the Dancoff-Ginsburg factor (calculated by module NEWA).
  The nuclear MICROX data have in fast energy range GAM energy structure (99 fine groups in the energy zone from 14.9 MeV to 0.414 MeV) and in the thermal energy range GATHER structure (101 energy groups up to 2.38 eV). To take into account the thermal scattering of neutrons in graphite, the fast/thermal energy boundary is here set at 2.38 eV. For the absorbers in resonance region (like Th232, U238, Pu240) there is ultra fine representation of cross sections between 0.4 eV and 3.5 keV for a set of temperatures. After spectral calculation MICROX finally condenses macroscopic and microscopic groups constants to few groups cross sections as specified for neutronics calculations.

- MAGRU

  The scope of the module MAGRU is the calculation of the macroscopic cross section for each burn up zone. For this task it uses the set of average number densities delivered from NIVERM and the set of microscopic cross sections from MICROX. Moreover it calculates the diffusion constant according to BEHRENS to take into account streaming effects in pebble bed core [12].

- HBLOCK

  The diffusion code HBLOCK [13] calculates the neutron flux distribution in the core and the eigenvalue k_{eff}. It solves the diffusion equation by a coarse mesh method, characterised as following:

  - A linear equation system for distinct flux values with nodal points of a R-Z grid is formulated. Each node can contain at most for four materials (original equations system and original mesh).
  - On transferring to the coarse mesh equations, the most part of the original equations are in general omitted the differential operator div D grad is formulated as usual in a five points difference notation. and auxiliary assumptions are made to enable the solution of the remaining equations. The deletion of any equations occurs so that the crossing lines on the original grid are defined as coarse mesh lines and the equations are selected in order to delete the source terms that not belong to the points of the coarse grid lines.
  - The transversal leakage is here determined by a spatial linear interpolation from the respective leakage of each nodal point of the “coarse lines”.
  - The cavity between pebble bed core and top reflector is here simulated with an effective diffusion constant, dependent from the direction.

- VORNEK

  This module’s objective is the determination of the power density distribution, of the fuel element power (averaged over all the fuel element at each place), the fast neutron dose and the mean normalised neutron fluxes in each burn up zone for the subsequent burn up calculations of the module SBURN. The field of the power density can also be axially smoothed (i.e. , discontinuity at the limit of the burn up zones of the fission cross sections can be approximated with the method of the last squared errors and as fission cross sections are assigned at each case the cross sections values in the middle of the zone).

- BUCK

  BUCK calculates bucklings for spectral zones from Flux distribution calculated by diffusion module HBLOCK. The bucklings are used for spectral calculation in MICROX. By means of an iterative application the cross sections calculated for spectral zones are consistently weighted so that four or seven energy groups are sufficient for the solution of whole core diffusion equation.

- NZW

  NZW calculates the afterheat distribution and U-239 and NP-239 number density as a function of time for the simulated power history. The calculation scheme is according to DIN 25485E [14]. It uses data calculated from NIVERM and SBURN as well as data pre-calculated from ORIGEN2 with a special cross section library condensed over core spectra.

- NECKAR

  NECKAR is a fast running thermal-hydraulic code for the determination of average fuel and moderator temperatures of spectral zones of MODUL type reactors.

- THERMIX/FRECON

  These modules calculate 2D stationary and transient solid and gas temperature distribution for a pebble bed reactor . THERMIX is the **FRAMATOME ANP** version of the THERMIX code developed by **FZJ**. FRECON was developed at **IKE** and used for both thermal-hydraulic calculations for pebble bed reactors and in an extended version also for calculation of core cooling of LWR under accidental conditions.

- SBURN

  This module calculates the propagation of number densities during a burnup step for a reduced 69 isotope chain model taking into account the not explicitly treated isotopes by means of four pseudo fission products using neutron flux densities calculated from HBLOCK. The burnup steps were performed for all burnup zones (for top middle bottom and of zone) for every pass of pebbles through the core.

- KENO-VI

  This code is a multigroup Monte Carlo code which is used for the calculation of three-dimensional problems such as control rod values or reactivity coefficients. The multigroup data were processed by MICROX with maximum possible group number 193 or by IKE system RESMOD/RESAB/RSYST with 292 energy groups. The number densities for the preparation of macroscopic cross sections will be taken from SBURN/NIVERM for the burnup zones. The cross sections for reflector and control rods are prepared mainly by RESMOD/RSYST.

- MCNP

  For comparisons with deterministic solutions MCNP calculations with the continuous energy cross section representation for the double heterogeneous system were performed. The number densities were taken from SBURN/NIVERM again.

2.2. Method for simulation of Xe behaviour

The simulation of Xe behavior in a pebble bed HTR can be performed by means of at least two-dimensional coupled neutronics, thermal-hydraulics and burn-up calculations. Due to the expected slow transients a quasi-stationary approach can be used for these calculations. For the neutronics calculations first the core conditions for which the detailed Xe transients will be investigated must be defined. These conditions can be determined by performing of a set of coupled stationary neutronics, thermal hydraulics and burn-up calculations for a distinct reload strategy for the pebbles together with the simulation of the flow of pebbles through the core.

For these calculations the program system ZIRKUS will be used which contains program modules for all steps of such coupled calculations. The type of fuel, the pebble flow pattern and the design data of core and the thermal hydraulic parameters as well must be defined for these calculations. The results of these calculations are the zone wise core composition (number densities), temperature distribution, critical control rod position etc. The starting conditions of the Xe transients will be the full power at Xe equilibrium. The Xe transients will be excited by a perturbation of power distribution caused by a movement of the control rods. For the Xe analysis the program system ZIRKUS can be used (as for the calculation of the core conditions) with small time steps to simulate the detailed buildup and decay of iodine and xenon and its influence to power distribution.

3. Results

3.1. Cylindrical cores

The main activity of the first work for this work package was the implementation of all modules described and performing of test calculations. The reference reactor was the MODUL with 200MWth [15]. The burnup 80.000 MWD/t U. The uranium content of one fuel element was 7.0 g, the enrichment 7.87 % U-235. The mesh grid for the burnup and neutronics calculation is shown by Fig. 1, the corresponding relative power distribution and thermal flux distribution can be seen from Figs. 2 and 3 respectively. For this reactor design the maximum fuel temperature after a depressurisation accident was analysed for different MEDUL passes. For every number of passes from 5 to 15 the equilibrium state was calculated by ZIRKUS. Then the afterheat distribution calculated by module NWZ of ZIRKUS was passed to the transient heat conducting program HEATING-7 of the SCALE system. It was conservatively assumed, that there was a prompt depressurisation. The propagation of maximum fuel temperature with time is shown in Fig. 4. From this figure one can see the advantage of MODUL since the maximum fuel temperature even after depressurisation is limited. This maximum temperature, however is dependent from fuel cycle passes. The development of temperature after the depressurisation is shown in Figs. 5 and 6 for two time points after beginning of accident. In Fig. 7 the final result: maximum fuel temperature as a function of recycle passes is shown. The larger the number of passes the lower the maximum temperature. This fact can be used for optimisation since the reactor nominal power can be increased if the number of passes is increased without increase of maximum fuel temperature. In Fig. 8 the peak factor of power distribution is shown as a function of recycle passes.

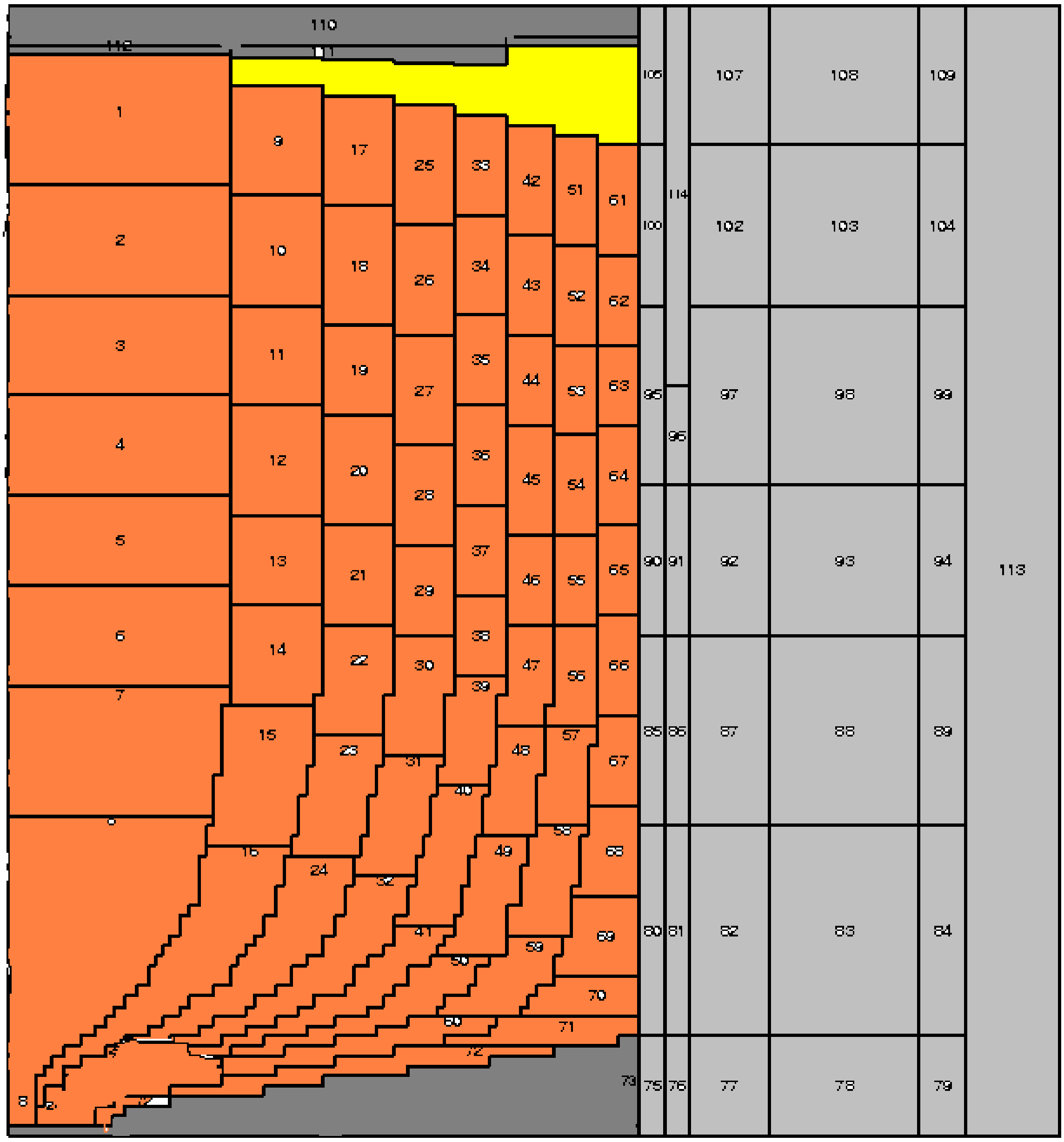

Fig. 1: Zone subdivision of MODUL reactor with 200 MWth (zones 1-72 burnup zones)

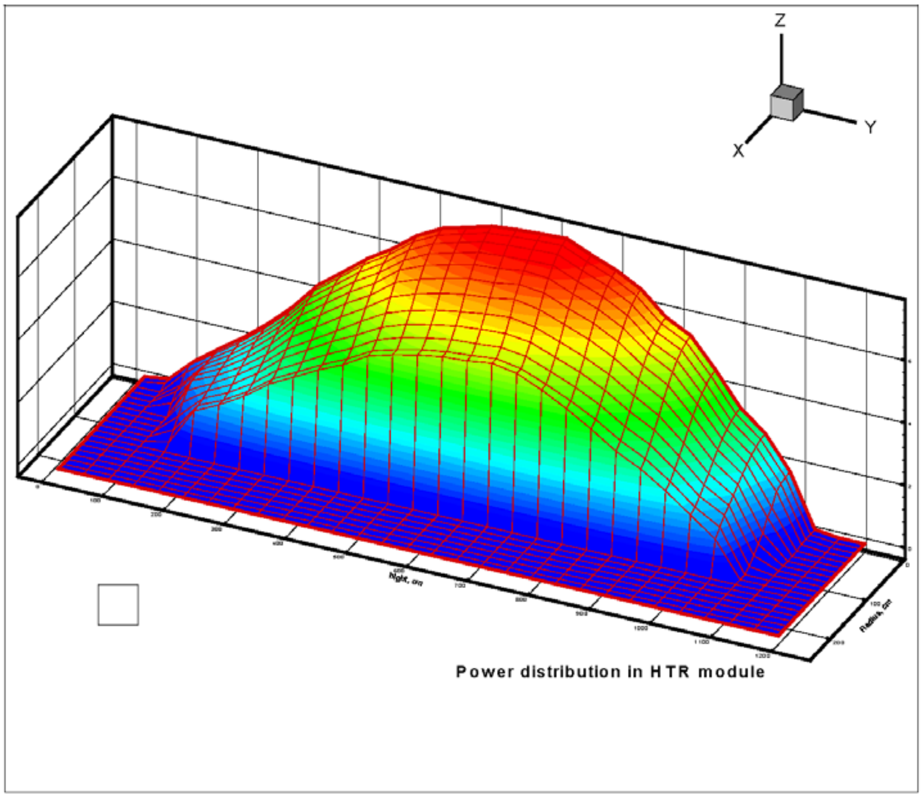

Fig. 2: Relative power distribution in HTR Module in equilibrium cycle (200 MWth)

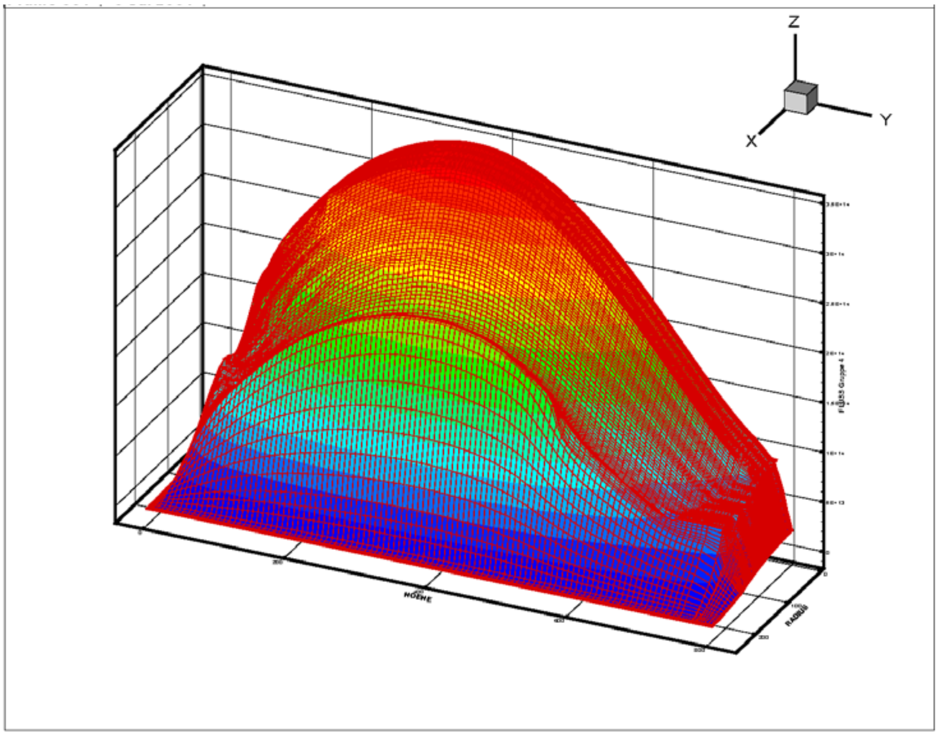

Fig. 3: Thermal flux distribution in HTR Module in equilibrium cycle (200 MWth)

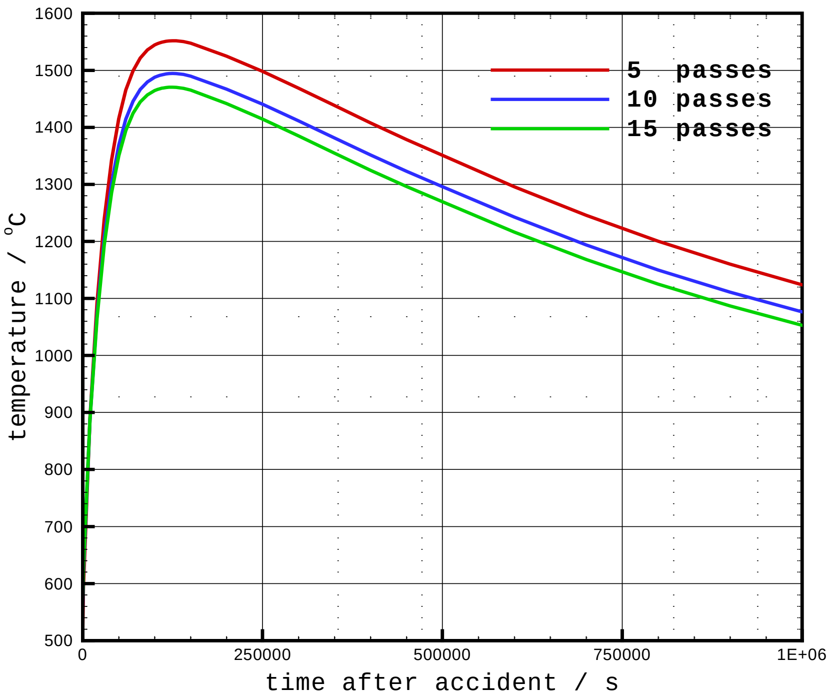

Fig. 4: Maximum fuel temperature as a function of time after depressurisation for different recycle passes for MODUL (200 MWth)

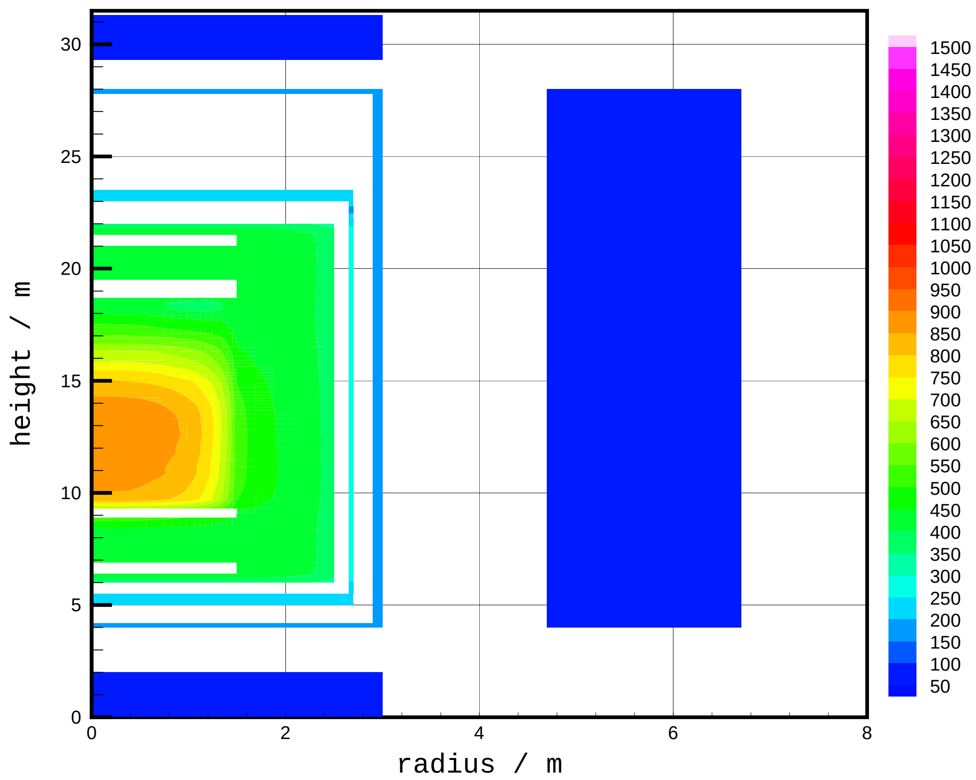

Fig. 5: Temperature distribution 10,000 s after depressurisation for MODUL (200 MWth) with 15 recycle passes and 80.000 MWd/t U

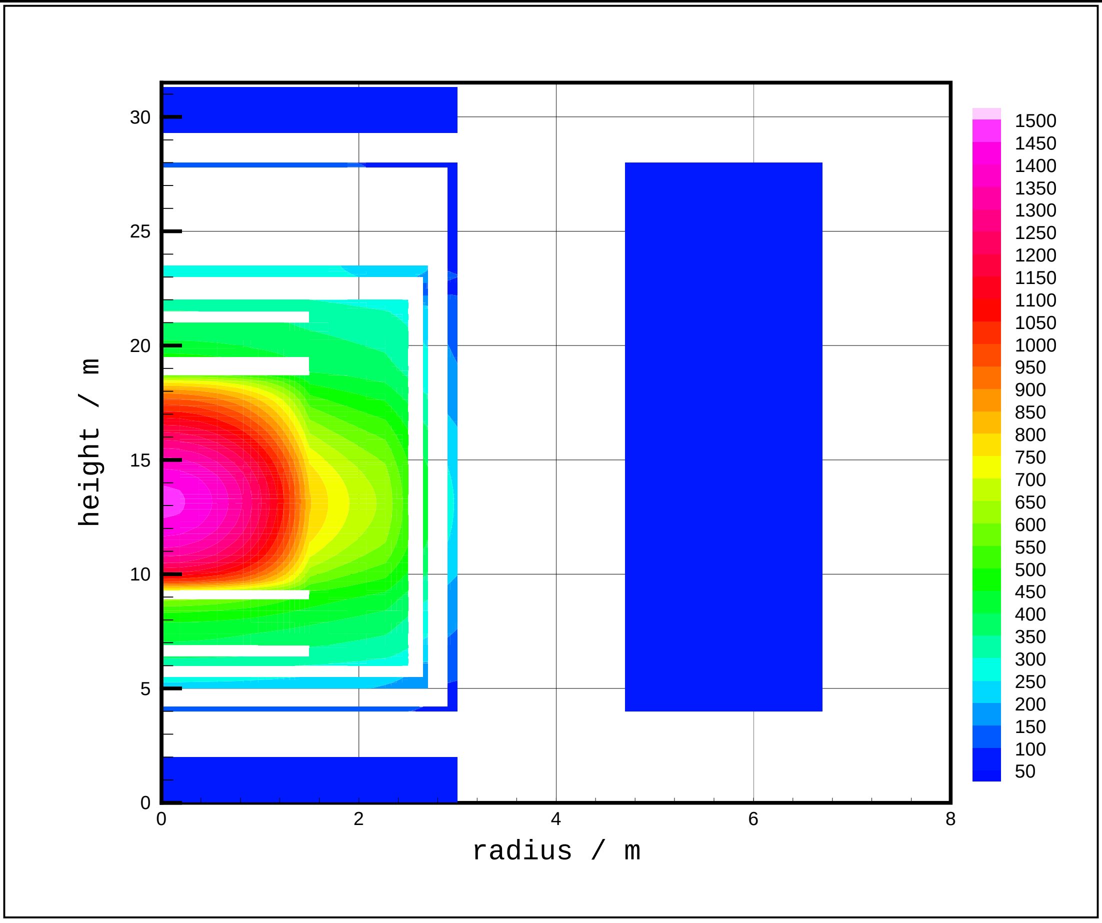

Fig. 6: Temperature distribution 125,000 s after depressurisation for MODUL (200 MWth) with 15 recycle passes and 80,000 MWd/t U

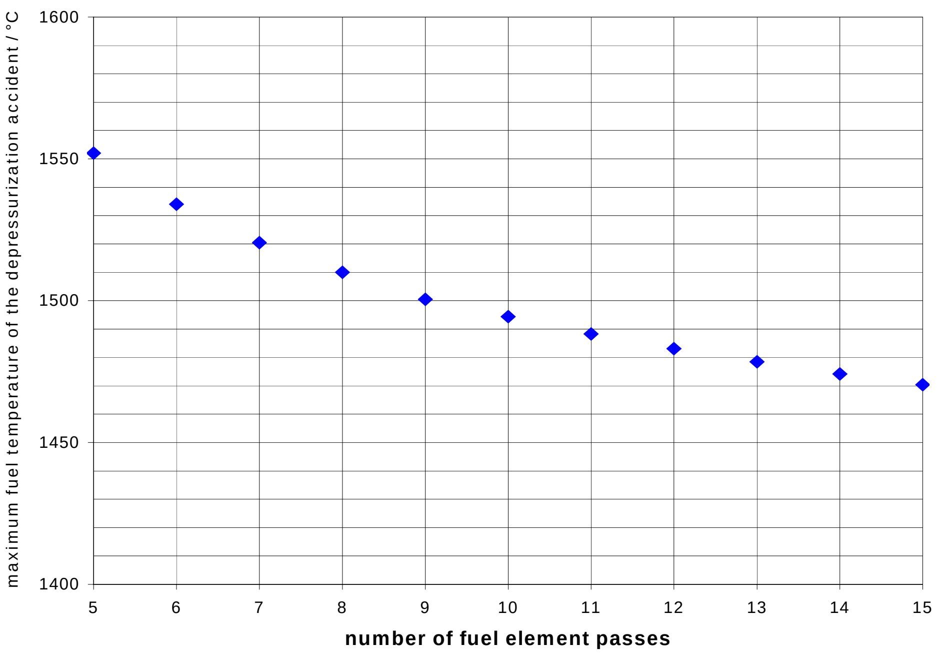

Fig. 7: Maximum fuel temperature after depressurisation for MODUL 200 MWth, 80.000 MWd/t U, as a function of recycle passes

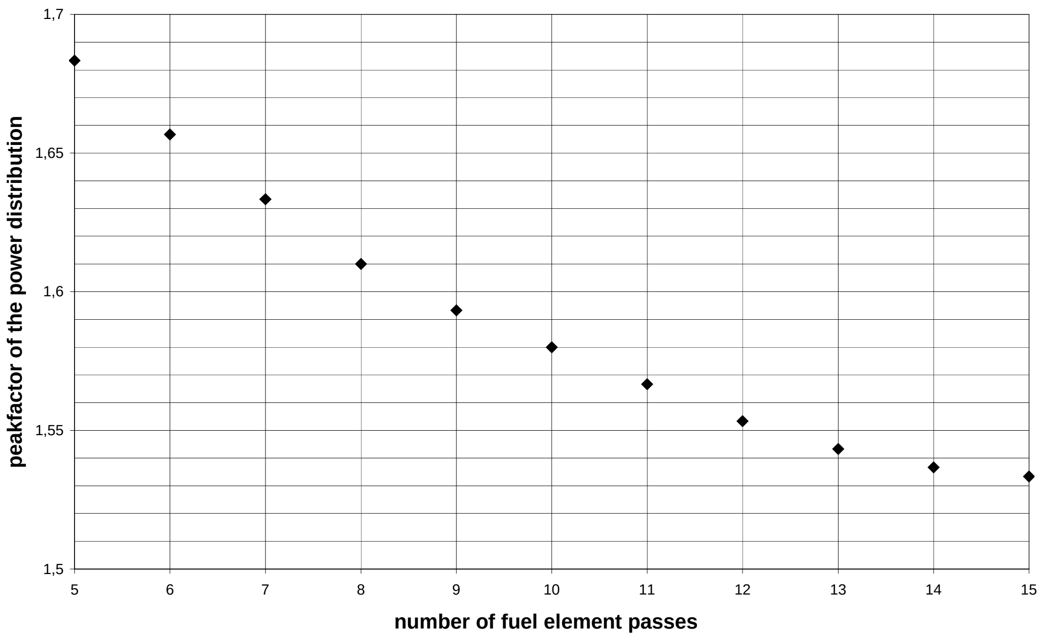

Fig. 8: Peak factors for MODUL 200 MWth, 80.000 MWd/t U, as a function of recycle passes

The further characteristics of HTR cores are safety related parameters such as reactivity coefficients of moderator and fuel temperature and reactivity changes due to water ingress (if it has to be regarded under accidental conditions). The temperature coefficients are strongly dependent from U-load as it is seen in Figs. 9 and 10 for fuel and moderator coefficients respectively for 6-10 g U/pebble. Furthermore, in Fig. 11, the dependency of the moderator coefficient from Xe-inventory is shown. Fig. 12 shows the reactivity increase due to water ingress for different U loads/pebble. It is clear to identify that for high U-loads (>78 U/pebble) the reactivity increase is very large due to the strongly under moderated core. If water ingress (large amounts of water) cannot be excluded, the core must be designed correspondingly.

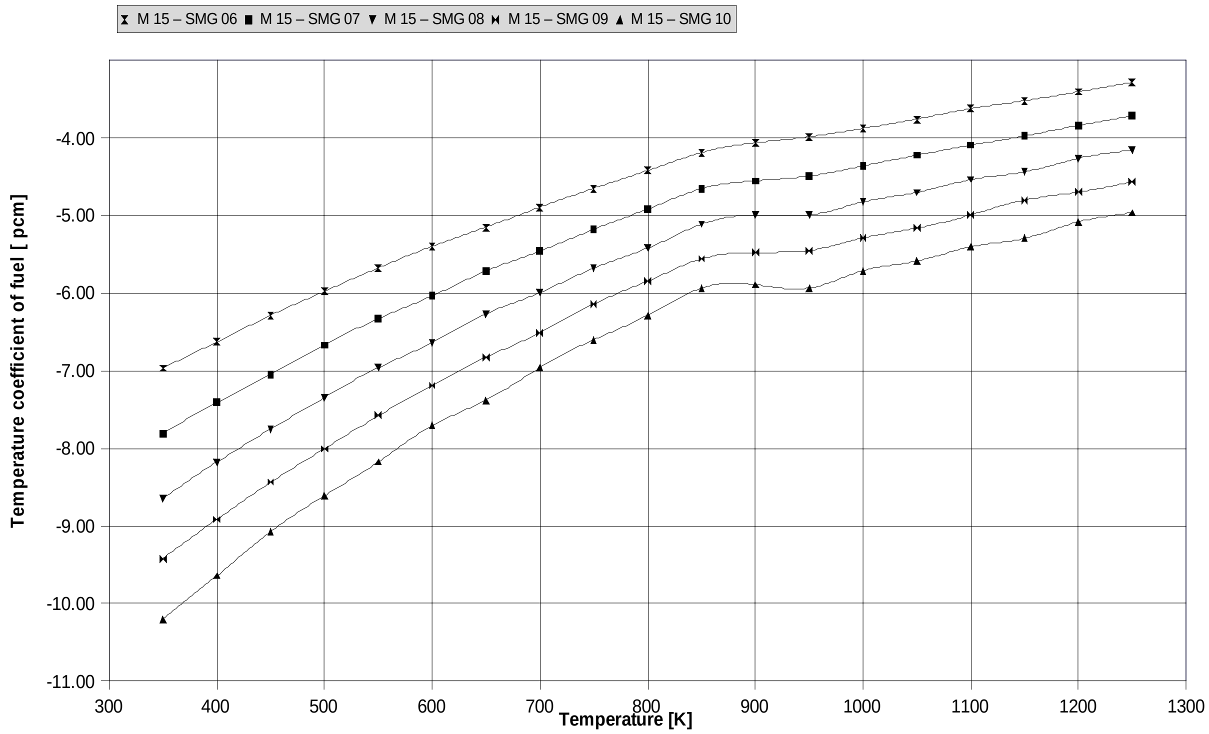

Fig. 9: Temperature coefficients for fuel for different U-loading : 6, 7, 8, 9, 10 g U/Pebble

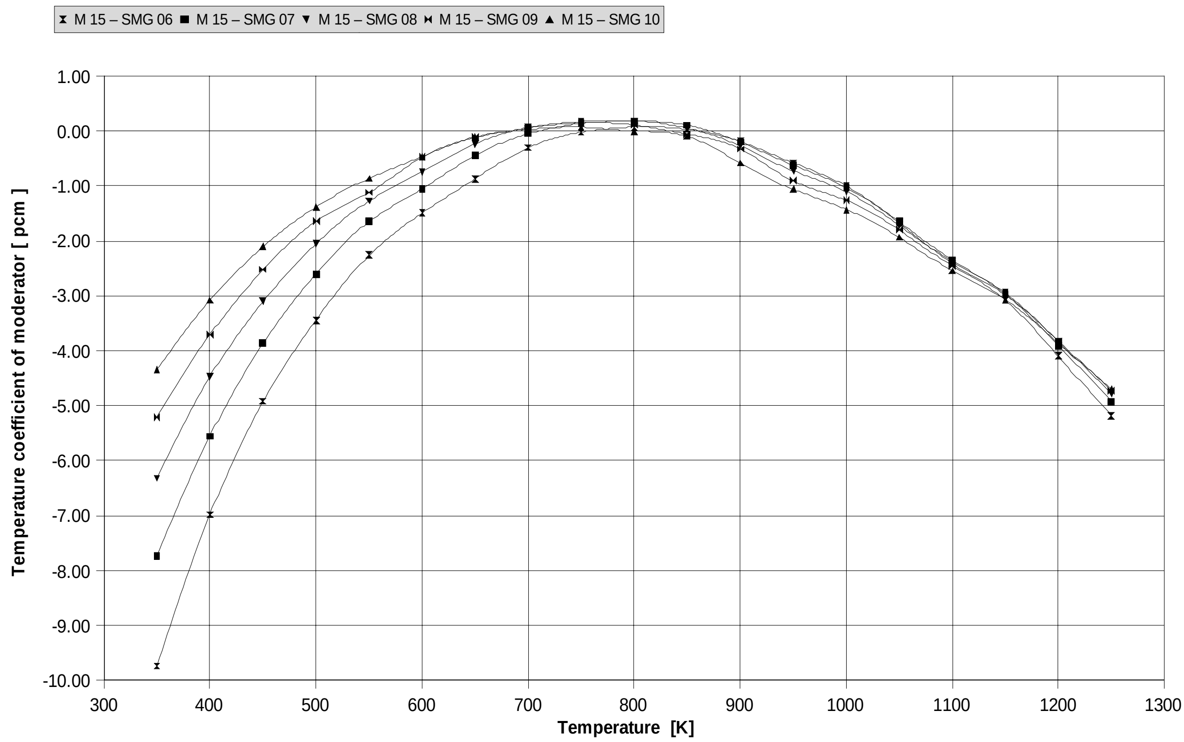

Fig. 10: Temperature coefficients for moderator for different U-loading : 6, 7, 8, 9, 10 g U/Pebble

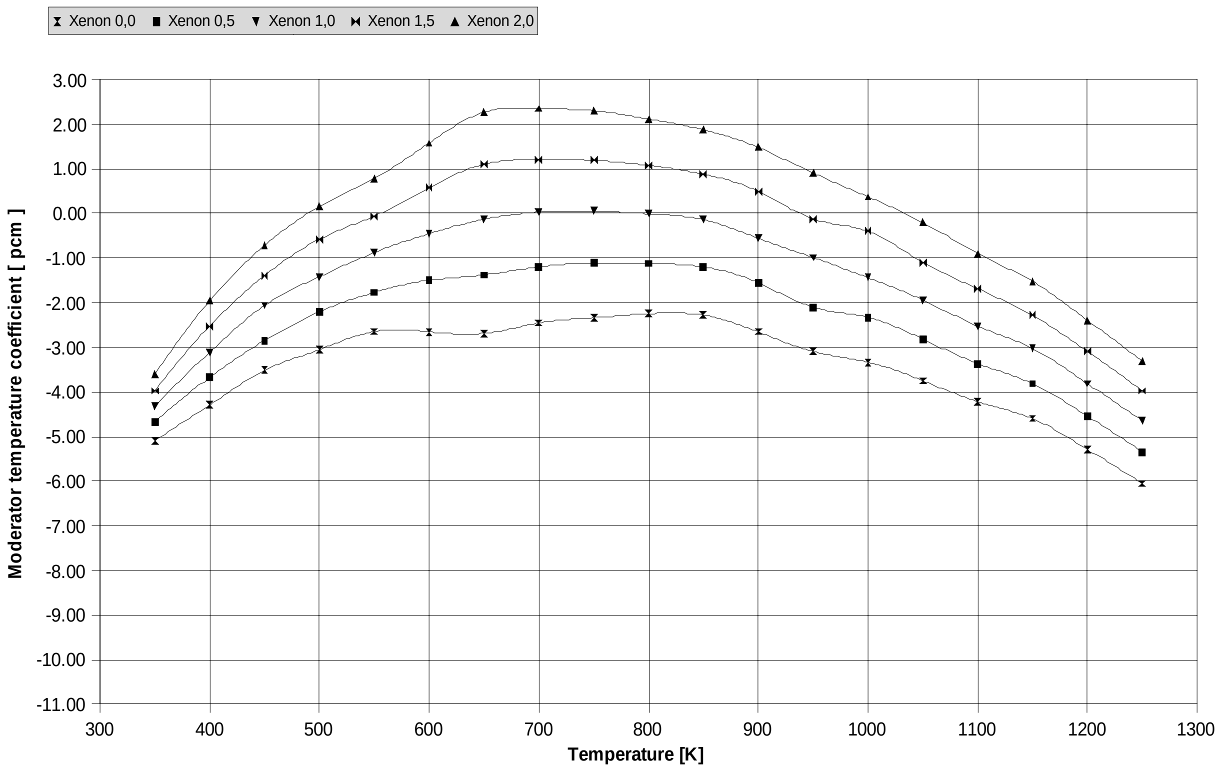

Fig. 11: Moderator coefficient as a function of Xe inventary (Xe 1 means Xe equilibrium at nominal conditions)

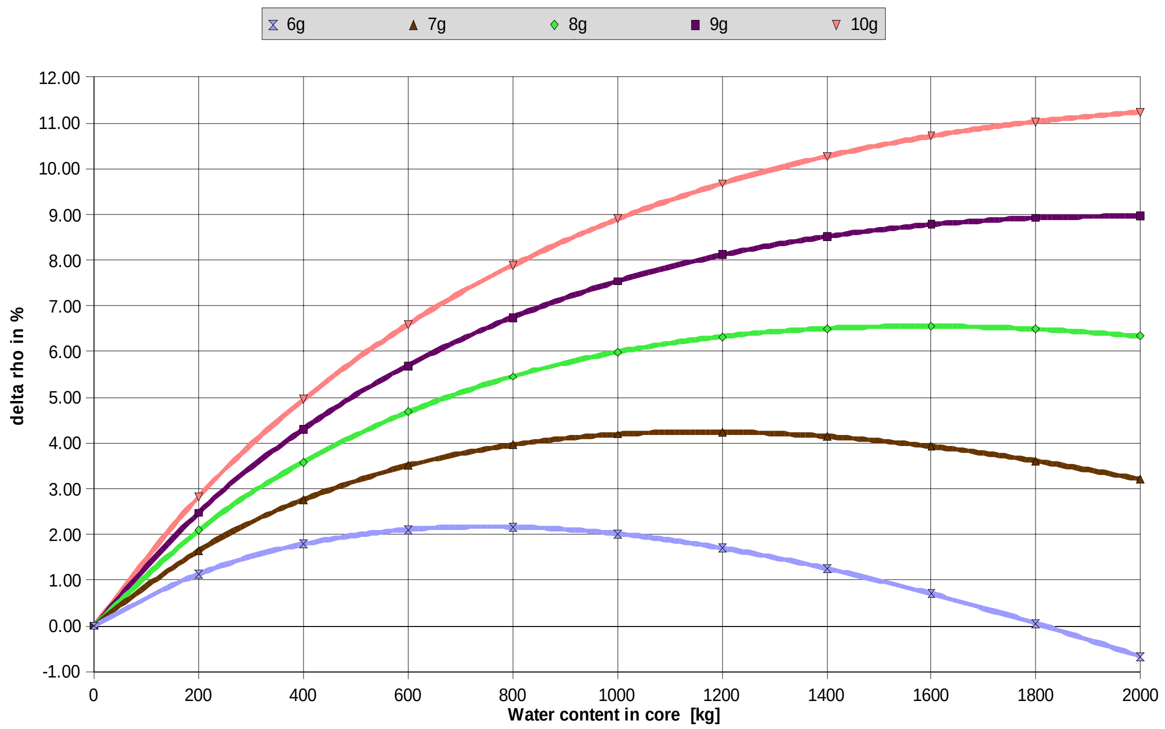

Fig. 12: Reactivity change due to water ingress for different U load: 6 – 10 g U/pebble

3.2. Analysis of Characteristic behaviour during LOC Accidents for annular cores

The ability to remove the decay heat without reaching fuel temperatures which could lead to a massive release of fission products under all possible scenarios is essential for inherent safety of HTRs. Scenarios with loss of the forced circulation are most challenging with this respect. For their analysis, it is important to take into account the thermal inertia and heat transfer resistances of relevant design elements, as well as proper initial and boundary conditions.

The calculations here are carried out for a reference reactor design. A sketch showing the basic structures and the major geometrical dimensions is given in Fig. 13. The emphasis here is not to investigate a specific given design, but rather to derive trends and to quantify the effect of design variations in the right order of magnitude. Therefore, the investigated layout has been partly simplified, concentrating on the major design elements. Key data have partly been borrowed from current concepts or follow from technical feasibility considerations. E.g., the core height has been limited to 11 m, motivated by considerations on stability against Xenon oscillations. The diameter of the RPV of 6.4 m has been chosen to be within economical and technical feasibility constraints. Two design variants with annular core are investigated in this section, for both a dynamic middle column and a compact middle graphite column. The core dimensions and the nominal operating conditions are given in Table 3.

The radial and axial shape of the power profile will clearly have an influence on the heat-up and thus the maximum fuel temperatures. However, the exact power profile at the occurrence of the accident depends on various factors such as the fuelling scheme, control rod position etc. and therefore can be optimised separately. In order to facilitate comparison and analyses, an approximate, simplified power distribution has been chosen for the calculations, assuming a flat radial distribution and a cosine-shaped axial distribution with a peak factor of 1.5. The temporal development is calculated with respect to the Way-Wigner approximation. For optimised designs ZIRKUS calculations were performed and the correct power and afterheat distributions were used.

As initial conditions for the analysed transients the steady (equilibrium or operating) states, calculated according to the values given in Table 3, are used. Boundary conditions are provided by the RCCS assumed to be the unique heat sink in the system at a constant temperature of 40°C during the transients.

For the material properties such as thermal conductivities, emissivities and specific heat capacities of helium, graphite, carbon and steel standard correlations or values have been used which were also applied in the HTR-MODULE case [16]. The effective heat conductivity of the pebble bed was evaluated according to the model of Zehner/Schlünder [17].

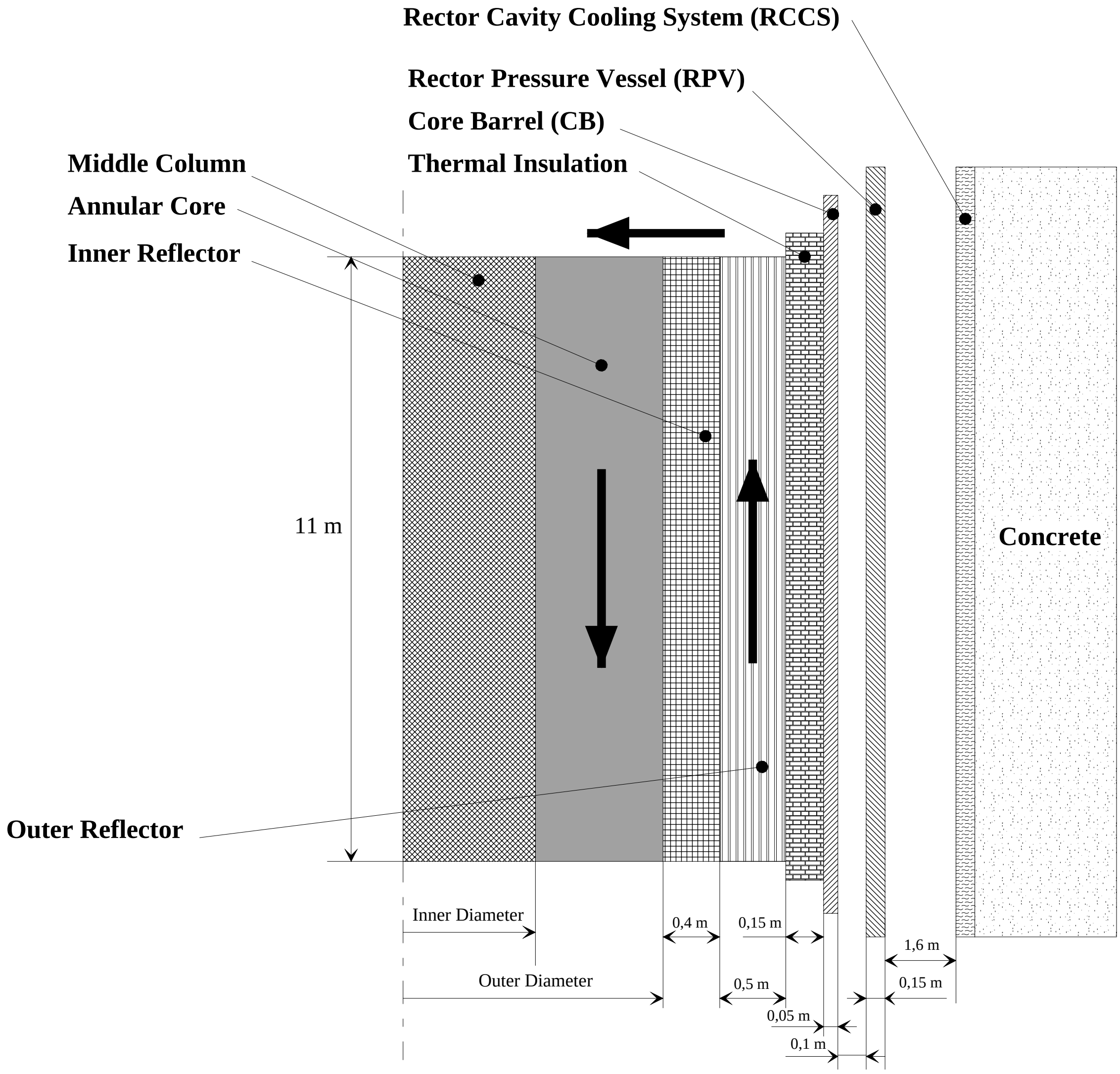

Fig. 13: Geometrical sketch of the reference design.

Table 3: Core layout and operational data for the reference design

Core inner diameter [m]	2
Core outer diameter [m]	3.7
RPV outer diameter [m]	6.4
Inlet temperature [°C]	500
Outlet temperature [°C]	900
Thermal power [MW]	425
Pressure [bar]	90
Helium mass flow rate [kg/s]	204.5

3.2.1. DLOCA case

In this case failure of the circulation of coolant and a loss of pressure are assumed, e.g. due to the rupture of a main coolant pipe. The reactor is also supposed to be shut down. Due to the relatively low density of the coolant at atmospheric pressure (1 bar compared to 90 under pressure) the heat transfer through natural convection is negligible. For both designs under considerations, the behaviour is quite similar. The reactor starts to heat up due to the decay heat. The initial temperature profile under operating conditions, with maximum temperatures at the core exit, is transformed in an axially essentially symmetric profile imposed by the heat source distribution (see Fig. 14). The maximum temperatures in the core continuously increase, until they reach a maximum around 2,5 days (see Fig. 15), when the decreasing decay heat can be removed from the core region by heat conduction and radiation.

The time, when the maximum fuel temperature is approached, marks also a transition from transient to quasi-steady behaviour. This can especially be seen from the radial temperature profile in the core. During heat-up, the temperatures in the unheated middle column lag behind the temperatures in the annular core. When the maximum temperature is approached, the radial profile over the central column vanishes (see fig. 14). The subsequent cool-down then follows a quasi-steady behaviour, in which the developed temperature profile is practically maintained.

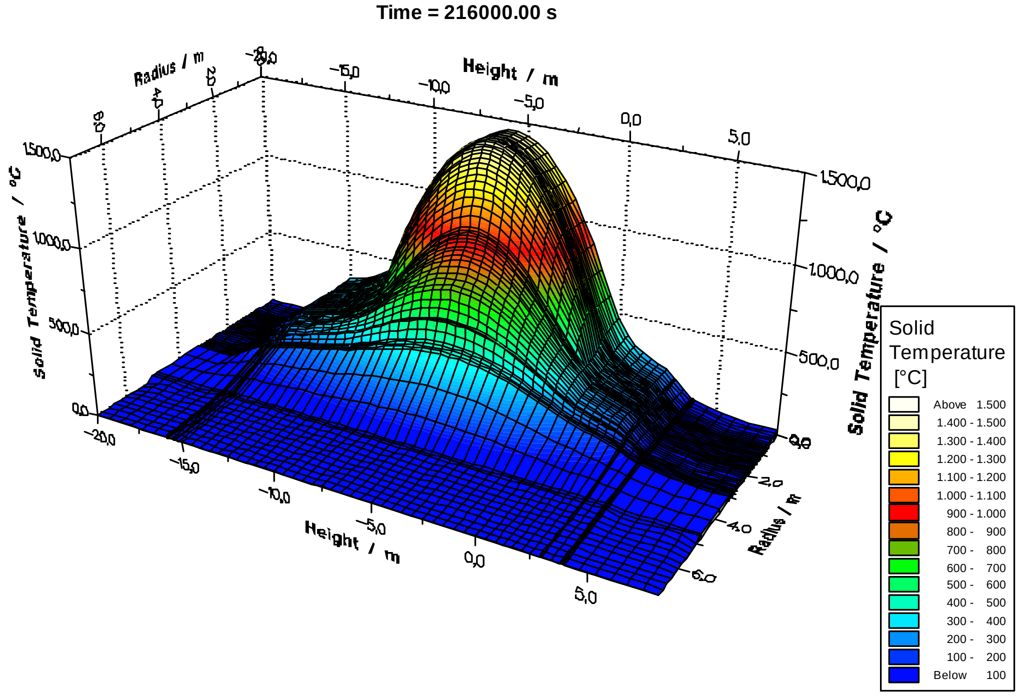

Fig. 14: Solid temperature profile at the point when the maximum temperature is obtained in the DLOCA case for the annular core design with solid graphite middle column.
(Note that the top of the core corresponds to height 0m and the bottom to –11 m.)

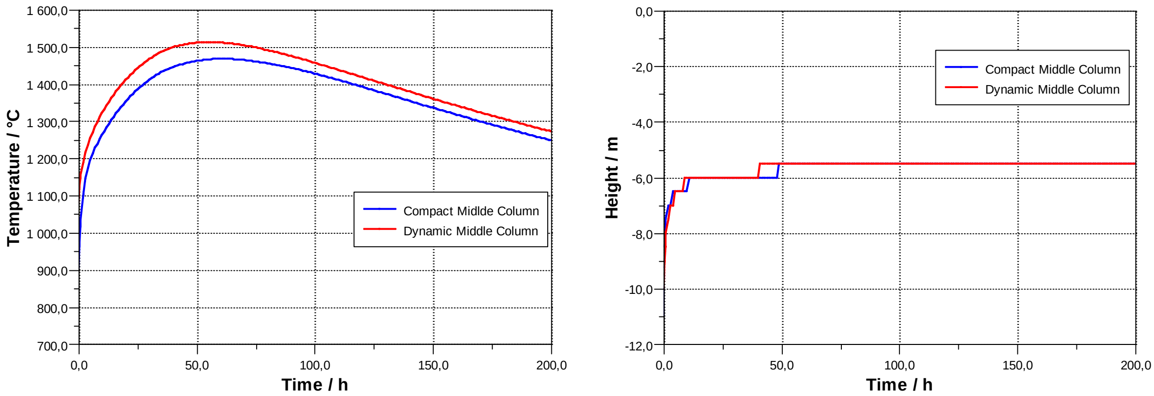

Fig. 15: Comparison of maximum fuel temperature (left) and axial position of the hot spot (right) versus time for designs with annular core in the DLOCA case

For the DLOCA case, the differences between the designs with dynamic and fixed middle column are relatively small. The maximum fuel temperature remain within acceptable limits. The higher temperatures reached in the case with dynamic middle column is mainly caused by the lower thermal inertia of the pebbles compared to a massive graphite column. Thus, the maximum temperature is reached at an earlier time, when the decay heat to be removed is still at a higher level.

3.2.2. PLOCA case

In this case a failure of the circulation of coolant is assumed and the reactor is supposed to be shut down but remains under pressure. A natural convection flow develops under the influence of driving temperature differences, which in contrast to the DLOCA case strongly affects the decay heat redistribution. This can be observed from the development calculated for the design with dynamic middle column given in Fig. 16 which shows the gas flow and temperature distribution at different times. Due to a large initial radial temperature gradient, which results from operational conditions with cooled middle column and hot core annulus, a strong natural circulation loop has developed after 4s. Helium rises in the outer hot annulus, heating up the upper parts, and flows downwards through the central part, releasing heat to the cold middle column. As a result, the hot spot moves from its initial position at the bottom towards the top of the core (see Fig. 18, left).

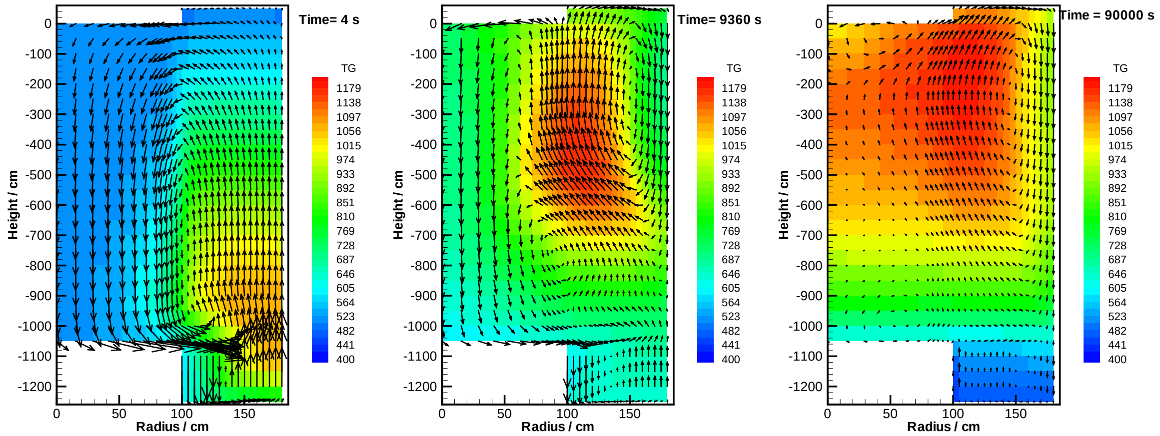

Fig. 16: Gas velocity (arrows) and gas temperature (colour shade) distributions at different times in the PLOCA case for the annular core design with dynamic middle column.

While the radial temperature difference between core and middle column is successively reduced, a second convection loop develops due to the increasing radial temperature gradient between core and reflector (see Fig. 16, middle), which is cooled by conduction and radiation. The heat flux redistributed through the two loops exceeds at around 2,6 h the local decay power in the hot spot region, resulting in the first peak and subsequent decrease of the maximum fuel temperatures given in Fig. 18.

Finally, the first circulation loop practically disappears due to the continuous heat up of the middle column (see Fig. 16, right). With only the reflector as major heat sink, the heat redistribution by convection is less effective. This leads to an at first renewed increase of the maximum temperature, until the core finally cools down with decreasing decay power (see Fig. 18).

In the design with compact middle column, the initial temperature distribution resulting from operating conditions shows no significant radial gradients. These just evolve from the heat up of the core. At first, two major loops exist, with hot gas rising in the center of the heated annulus and flowing down along the colder middle column and reflector (see Fig. 17, left). As the middle column is successively heated up, the inner loop diminishes and is displaced to the upper center. The long-term flow pattern is dominated by a loop with gas being heated while rising in the inner part of the core and partly along the middle column and releasing heat while sinking along the cooled reflector (see Fig. 17, right). The relatively large heat conductivity of the middle column (as compared to the core) affects the temperature distribution through axial conduction which flattens the axial temperature profile. As a result, the axial position of the hot spot remains close to the middle of the core, similar to the DLOCA case.

The larger heat capacity and axial heat conduction of the compact middle column, together with the lower starting value, leads to generally lower maximum temperatures for this design (see Fig. 18). For both designs the maximum temperatures reached in the PLOCA case are well below those reached in the DLOCA scenario.

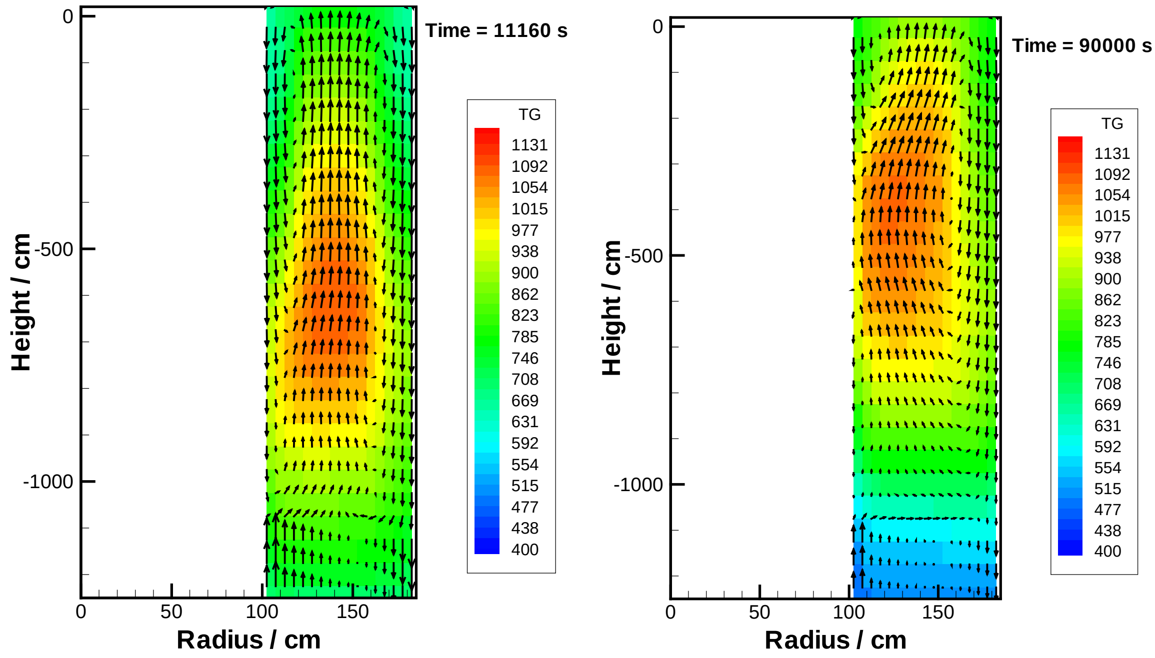

Fig. 17: Gas velocity (arrows) and gas temperature (colour shade) distributions at different times in the PLOCA case for the annular core design with compact middle column.

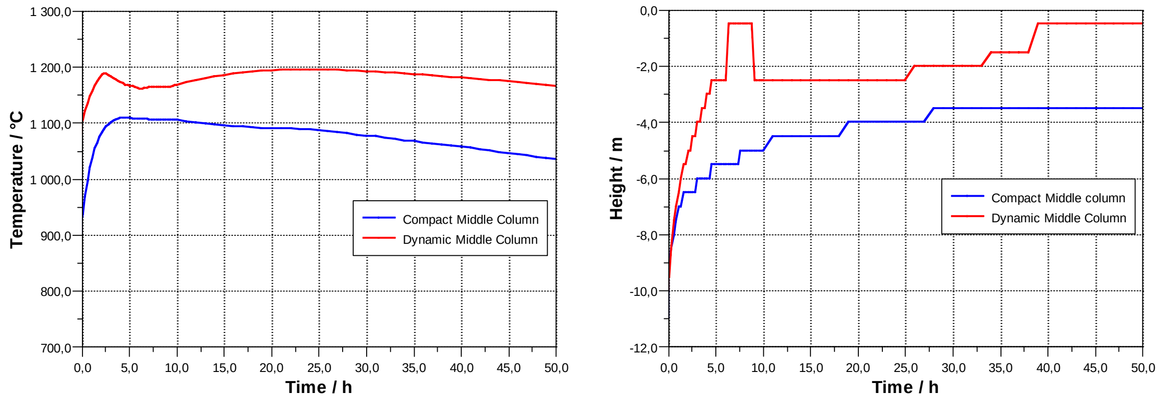

Fig. 18: Comparison of maximum fuel temperature (left) and axial position of the hot spot (right) versus time for designs with annular core in the PLOCA case.

3.3. Limitation of core height due to Xe-transients

For the pebble bed reactors with large core heights the stability of the power distribution in respect to Xe oscillations must be scrutinized and carefully investigated. For HTR designs, the increase of the core height is an obvious means to increase the power of the inherently safe modular HTR without sacrificing the maximal tolerable fuel temperature of 1600°C even in cases of a depressurization accident or rod expulsion scenarios. To reach a core power in the range of other modular HTR concepts, here, the core height must be higher than for cores with other reload strategies or designs. At least at the first sight economic reasons seem to dedicate this measure.

For large core heights, however, Xenon oscillations may be excited – e.g. by load following processes – which could cause undamped periodic changes in axial power distribution with increasing amplitude.

A control of this effects would require an active control of the axial power shape which would be in contradiction to the claimed inherent passive features of the modular HTR concept; apart from that it is difficult to imagine how such an axial power control could be realized in a modular HTR. No convincing technical solutions have been brought forward; probably they do not exist – even in principle.

It is reasonable, to require from the very beginning that the undamped Xenon oscillations must be absolutely excluded for any modular HTR. This requirement poses directly the question of the maximum allowable core height.
Therefore, before choosing a distinct core height, the conditions for stable conditions for both full power and reduced power (part load) must be investigated in detail to ascertain that unmapped Xenon oscillations during full power and part load operation are avoided, since these axial power oscillations would damage or even destroy the reactor fuel elements.

Such calculations were performed with both the ZIRKUS systems and the transient core simulation program RZKIND [18] developed from FANP. Examples of typical Xe oscillations are given in Fig. 19 (relative Xe-concentration at core top, core center and core bottom) and Fig. 20 (control rod positions) after a reduction of power from 100 % to 50 % and increase of power after 6 hours (if Xe-maximum is reached). The Fig. 20 shows that the amplitudes of the (damped) oscillations are dependent from core height; the higher the core the larger the amplitudes. The corresponding axial peak in power distribution must be considered to avoid too high fuel temperatures during operation.

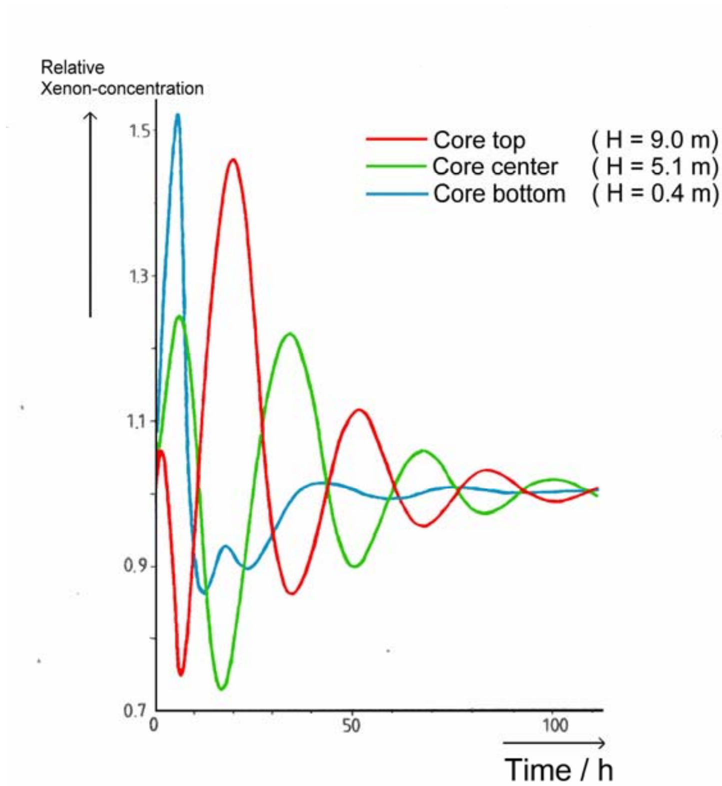

Fig. 19: HTR-Module: relative Xenon concentration for various core heights

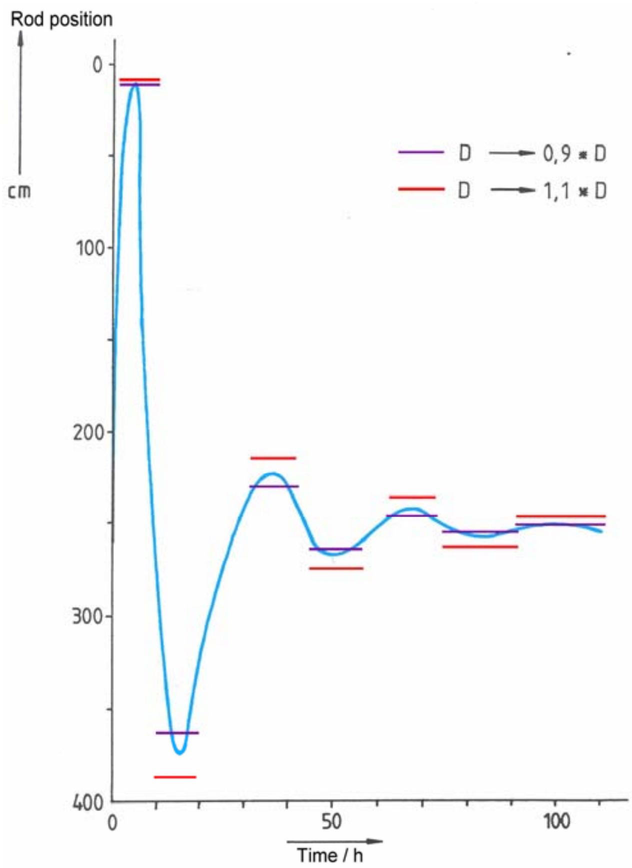

Fig. 20: HTR-Module: Rod positions to balance Xenon oscillations

4. Quantitative comparison of design options

The purpose of this section is to quantitatively compare different designs. Such a comparison can of course not be unique, since the result will depend on the choice of conditions and evaluation criteria. Here, a systematic approach is chosen which complies with the general objective of the present paper, which is to investigate the potential to increase the thermal power of modular Pebble Bed HTRs without loosing inherent safety features. The maximum fuel temperature reached during DLOCA will be limited for all cases to the same value, considered to have a sufficient margin from the technological limit (about 1600 °C), for which the integrity of the fuel coatings has been proven experimentally. This value has been chosen corresponding to the maximum fuel temperature of 1470°C obtained for the reference design with solid central graphite column. Then the maximum achievable thermal power, determined such that this criterion is still met, is used to compare different cases and designs.

Since the design with dynamic middle column showed inferior capabilities with respect to decay heat removal compared to the design with solid middle column, the further investigations concentrate on the latter. This design, denoted as case II in the following, is compared to a design with cylindrical core (case I). The core diameter for the cylindrical case is set to 3 m. In order to keep the comparisons comprehensible, the cases used for comparison have been derived from the reference design presented, i.e. geometrical layout and calculation conditions have been kept the same. In a second investigation the diameter of the compact middle column is varied, while keeping all other parameters unchanged (outer diameter of the core is 3,7m).

The choice of the power conversion technology is also an important factor which affects the overall economical and technical performance of a HTR. Currently, either the direct Brayton cycle (B) using a helium turbine or a classical Rankine cycle (R) with steam generator and steam turbine are discussed as options. For the present analyses, both options have been considered through different levels for inlet and outlet temperatures.

Table 4 summarises the results for the two cases that have been investigated. Significantly higher thermal powers can be realised with annular cores, since larger amounts of decay heat can be passively removed by heat conduction. This is mainly due to geometrical effects. Heat storage effects of the inner column also have an influence on the maximum fuel temperature by increasing the time at which this temperature is reached. This can also inferred from Table 5, showing that the decay power that can be removed with maximum temperature is practically the same for the different initial conditions (i.e. Rankine and Brayton) and characteristic for a given geometrical layout.

The results on attainable power rather should not be interpreted as absolute figures, also with respect to the simplifying assumptions which have been chosen to rather support comparison of different options relative to each other. E.g. the assumed radial power profile could lead to an underestimation of the maximum fuel temperature and thus to an overestimation of the attainable thermal power. On the other hand, the conclusion on achievable power increase going from cylindrical towards thinner annular cores would be even more pronounced taking into account actual radial power profiles.

Table 4: Major design parameters and maximum achievable thermal power for different reactor designs considered for the comparison

Design	*I*		*II*	
Core height [m]	11		11	
Inner core diameter [m]	-		2	
Outer core diameter [m]	3		3.7	
RPV outer diameter [m]	5.7		6.4	
Variant	*Brayton*	*Rankine*	*Brayton*	*Rankine*
Inlet temperature [°C]	500	250	500	250
Outlet temperature [°C]	900	750	900	750
Thermal power [MW]	**220**	**250**	**425**	**464**
Mean power density [MW/m^3]	2.83	3.21	5.07	5.54
Helium mass flow rate [Kg/s]	105.9	96.24	204.5	178.6
Pressure drop over the core [bar]	1.27	0.84	3.55	2.18

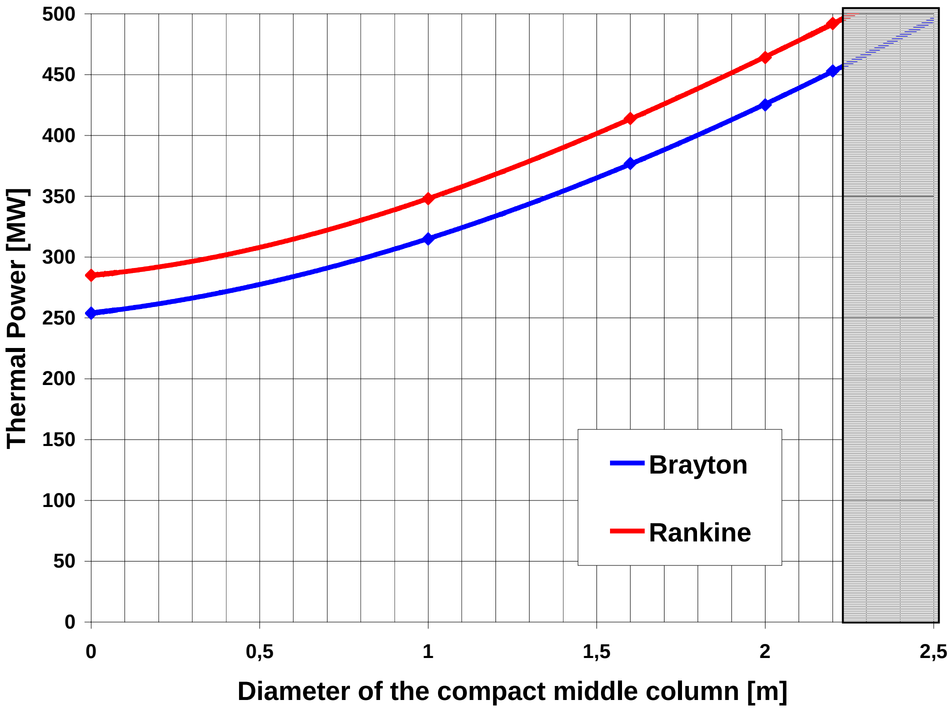

Fig. 21: Thermal power versus diameter of the middle column

Table 5: Results which characterise the design geometry

Case	I/B 220MW	I/R 250MW	II/B 425MW	II/R 464MW
Power removed at max. temp. [MW]	1.01	1.07	1.59	1.62
Time corresponding to max. temp. [h]	29	38	61	77

The potential for optimisation of the diameter of the compact middle column is indicated by the results in Fig. 21. From this diagram it can be seen, that for small diameters up to 1 m on average 6.1 MW additional power can be produced per 10 cm increase of the diameter of the middle column. For higher values this amount is doubled, giving 12.7 MW additional power per 10 cm. However, increasing the diameter of the middle column has of course technical limits and side effects, which finally may cancel the merits.

Such technical limits are also touched with respect to the pressure drop over the core. The higher power that can be achieved with the annular core designs requires larger helium mass flow rates and thus leads to larger pressure losses, which means that major technical difficulty are met in the annular core designs with higher power. E.g. in the cases with Rankine cycle the well proven single-stage blower technology, as used in the major HTR reactors and projects, can not be used (the pressure drop over the hole primary cycle is limited for this technology to 1.3 bar). In the Brayton cycle, turbo compressors are used to compensate the pressure drop and thus higher values are manageable. On the other hand, with too high pressure drop the leak flows increase heavily and can deteriorate the performance of the reactor.

5. The temperature of structures and their impact on the design performance

Although the fuel temperature remains within prefixed limits during the DLOCA accident the reference design can not considered as optimum in terms of inherent safety. The mechanical stability of the Reactor Pressure Vessel e.g. can no be guaranteed if it is expand to too high temperatures for a long time (Fig. 22). At this point a second safety criterion is needed.

This criterion can in a first attempt be defined within the same philosophy (simplicity and grade) as the criterion for the fuel temperature. The optimisation of design can be then made with respect to both criteria.

To further develop this idea, the reference design is taken as an example. As shown in Fig. 22 for the case with a thermal power of 425 MW, the temperature of the RPV reaches during DLOCA values, which could not be considered as harmless anymore (maximum 450 °C). If now a limit for this temperature is fixed at 380 °C, in the meaning that value under this limit are safe, the reactor has to be redesigned under consideration of both limits.

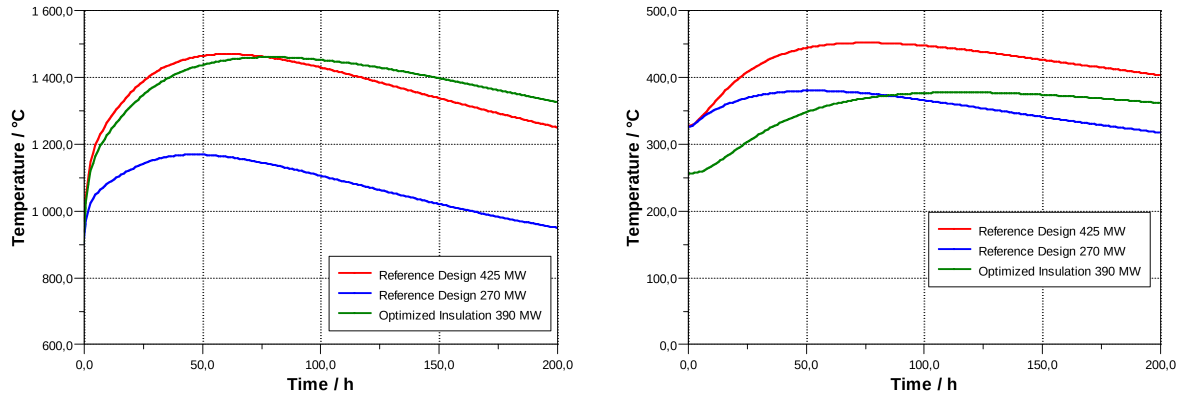

Fig. 22: Comparison of maximum fuel temperature (left) and maximum RPV temperature (right) versus time for the reference geometry cases and for the case with optimised insulation

An evident manner to do it is to reduce the power until both temperatures remain under the limits. The producible power in this case for the geometry is about 270 MW. At this point the maximal temperature of the RPV is just below the limit but the fuel temperature far below the limit.

If further the thickness of thermal insulation between outer reflector and the core barrel is varied, there is a point where both maximal temperatures are exactly within the limits. In this case the producible power is around 390 MW. The thickness of the carbon blocks insulation is then 0.5 m instead of 0.15 m in the original case. In this case an RPV with a diameter of 7.1 instead of 6.4 m is required, which represents a major technological challenge.

6. Conclusions

In this paper the behaviour of different design options for the modular pebble bed HTR during loss of forced cooling accidents is analysed. It is shown, that if properly designed, the modular pebble bed HTR can be considered as inherent safe in the sense that the maximal fuel temperature never exceed limits during both pressurised and depressurised loss of coolant accidents.

The potential to maximise the thermal power with annular core designs while keeping the safety features is investigated. Particularly, the potential of producing more power by increasing the diameter of the middle column is quantified. Additional technical complexities anticipated if this potential is tapped are mentioned and discussed.

Finally, it is investigated how the performance of a design is affected if an other safety criterion in addition to the maximal accident temperature of the fuel, e.g. the maximal accident temperature of the RPV, is included in the optimisation.

The increase of core height will be limited by the pressure drop in core and by Xe oscillations which excite power profiles which can lead to high operational fuel temperatures.

If the technical feasibility of solid inner columns is investigated in detail, such cores can lead to comparable high power of up to 400 MW without loss of the main inherent safety features.

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