A comprehensive Thermo-hydraulic neutronic and safety analysis of a 100MWth pebble bed reactor core

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Abstract

The HTMR100 is a 100MW$_{th}$ High Temperature Gas-cooled Reactor (HTGR), which is helium-cooled and graphite moderated. This reactor is of the Pebble Bed type featuring a Once-Through-Then-Out (OTTO) fuelling scheme, this cycle was chosen for the simplicity of the fuel handling system.

The HTMR100 Pebble Bed Reactor (PBR) has a core volume of 27.73 m^3 (core diameter 2.6 m, core height ~ 5.2 m) which results in an average core power density of 3.6 MW/m^3. This reactor utilizes uranium dioxide fuel (UO_2) at a U-235 enrichment of 10 wt% and 10 g of heavy metal (total uranium) per Fuel Sphere (FS). Although there are various types of fuel options that can be utilized in the HTMR100 reactor, ranging from LEU to mixtures of Th/LEU, Th/HEU and Th/Pu, this study will only focus on LEU in the form of UO_2 fuel.

This study investigates the neutronic and thermal–hydraulic modelling of the 100MW$_{th}$ HTMR100 at equilibrium (100% power with the control rods situated at the nominal position for a core having an equilibrium fuel burnup profile) and at transient/accident conditions. The purpose of the investigation is to determine if the maximum fuel temperature remains below the set limit of 1600 °C during a design-base accident event that is defined as a Loss of Coolant Accident (LOCA).

The equilibrium core was simulated using the Very Superior Old Programs (VSOP99) suite of codes to obtain burnup results. The transient behaviour of the core was modelled by a code called Multi-group Time Dependent Neutronics and Temperatures (MGT). MGT assesses the dynamic transient results of the accident scenarios such as the Depressurized Loss of Forced Cooling (DLOFC), Pressurized Loss of Forced Cooling (PLOFC), Control Rod Withdrawal (CRW) and a Control Rod Ejection (CRE) as well as normal operating transients of which a Load Following (LF) operation is a typical example.

The HTMR100 reactor utilizing the enrichment and heavy metal loading as specified does indeed produce the targeted 80 000 MW$_D$/T$_{HM}$ burnup for the OTTO fuel cycle. The study also proves that the VSOP99 and the MGT codes do in fact yield similar results for the HTMR100 reactor with regards to fuel centreline temperatures, outer sphere surface temperatures as well as moderator temperatures for the postulated accident scenarios that were analysed. The results also indicate that the design-base transient safety analysis simulations prove that the fuel temperatures remain below the set value of 1600 °C for the oxide-based fuel. A temperature-volume analysis of a DLOFC design-base event shows that

only ~ 2 % of the fuel reaches the higher temperatures of 1500 °C to 1600 °C in a DLOFC event.

The beyond-design base events with a very low probability of occurrence do however exceed the set value of 1600 °C but the fuel only exceeds this set limit for short periods of time while the transient occurs. The probability that this will lead to fuel damage should be very small since the time at these high temperatures is short and the volume fraction of the core exposed to these high temperatures is small.

Keywords: Fuel Spheres (FS); High Temperature Gas Reactor (HTGR); Low Enriched Uranium (LEU); Multi-Group Time Dependent Neutronics and Temperatures (MGT); Once-Through-Then-Out (OTTO); Pebble Bed Reactor (PBR); Reactor Pressure Vessel (RPV); Very Superior Old Program (VSOP)

1. Introduction

STL Nuclear (Pty) ltd. is developing a 100 MW_{th} Pebble Bed reactor called the HTMR100. This reactor is a HTGR. The HTMR100 uses helium as the coolant and is graphite moderated. This reactor has a cylindrical core with a diameter of 2.6 m, height of ~ 5.2 m, and a core volume of 27.73 m^3 that results in a core having power density of 3.6 MW/m^3. The outer Reactor Pressure Vessel (RPV) diameter is 5.25 m, which resulted from an optimization between the core structures design and the RPV being road transportable. The reactor is fuelled by uranium dioxide (UO_2) ceramic fuel at 10 wt% enrichment (U-235 content) and 10 g of heavy metal per fuel sphere. In this reactor, the fuel spheres pass through the reactor core only once, this is known as the OTTO reactor fuel cycle.

An OTTO cycle was chosen for this reactor due to the simplicity of its fuel handling system and significant cost reductions associated with the operational simplicity of this system (Mulder and Wang, 2016). The OTTO cycle has a higher power peaking factor and would have a higher maximum fuel temperature during a DLOFC event. Therefore, switching from a MEDUL (multiple fuel pass through the core fuelling cycle) to an OTTO cycle results in a reduction of the maximum allowable core power of about 30 %. Even so, the benefits of simplicity and lower plant cost might outweigh the disadvantages of lower power output and thus the OTTO fuel cycle was chosen for the reactor design and neutronic modelling.

In this paper the HTMR100 reactor shall not be coupled to a specific thermodynamic cycle but this reactor still ensures:

- A low excess reactivity due to on-line fuelling cycle (Δk_{eff} ~ 1700 pcm).
- The reactor ensures passive safety. There is no requirement for active safety systems or operator action to prevent fuel damage. This is achieved with a relatively large negative temperature coefficient of reactivity over the entire operational range, a low core power density, a core geometry that will ensure passive decay heat removal and the radionuclide retention capacity of the TRi-Structural ISOtropic (TRISO) particle fuel. Reutler and Lohnert determined that increasing the height/diameter ratio beyond 0.97 in a trade-off between neutron losses versus the advantage of gaining passive decay heat removal via the walls of a steel pressure vessel in the event of a DLOFC (Lohnert and Reutler, 1984).

- The design ensures that the materials of construction remain below the structural design limits and the maximum fuel temperatures in an accident condition remain below the set fuel damage limits
- The reactor and its components/systems are modular in nature with the RPV (5.25 m diameter) being smaller for road transportability and making it easier to fabricate.
- The simplistic fuel handling system ensures reliability through simplicity, which reduces the complexities associated with the recirculation system while achieving the desired burnup of 80,000 MW_d/t_{HM}.

All these features reduce the cost of the reactor facility, while maintaining passive safety.

A key passive safety characteristic of typical HTGR design is keeping the fuel temperatures during a DLOFC event low enough that the probability that radioactive fission products that would be released from the coated particles will be acceptably small. Based on the fuel tests performed in the early days of coated particle development the fuel temperature during a DLOFC event should remain below the set limitation of 1600 °C.

Another crucial safety issue is the risk of accidental steam ingress into the fuel core: The water is a more effective moderator than the graphite matrix of the fuel pebbles and in the graphite reflector. Therefore, steam (water) ingress will substantially increase neutron moderation. As HTR cores are normally under moderated, this increase in neutron moderation will potentially add reactivity to the core and thus to an increase in power output (Lohnert, 1992). If appropriate safety features are not put in place, such an event could potentially lead to a nuclear excursion. The risk of steam ingress is especially high when the core is coupled to a Rankine steam cycle for power conversion. However, this water ingress analysis falls outside the scope of this study. Therefore, the Heavy Metal (HM) (uranium in this case) loading per fuel sphere is kept low in order to achieve an over-moderated core as generic protection against possible water ingress incidents.

This study investigates if the design of the HTMR100 reactor indeed remains below the set limit of 1600 °C for the oxide-based UO_2 fuel for design-base accident events. The neutronic and thermal–hydraulic design of the HTMR100 is assessed and the transient safety analysis is performed to prove that the reactor design can claim passive safety, and that the fuel temperatures remain below the prescribed limits to avoid fuel damage in a LOCA.

The study assesses the core operating at normal conditions at a Maximum Continuous Rating (MCR) of 100 % power (100 MW_{th}) with the control rods inserted to the nominal position of 65 cm from the bottom of the top graphite reflector (in the side reflector). This is called the equilibrium condition at 100 % MCR.

The equilibrium core will be evaluated by the Very Superior Old Programs (VSOP99) suite of codes to obtain equilibrium results. These results are then utilized in a second code called Multi-group Time Dependent Neutronics and Temperatures (MGT). MGT assesses the dynamic transient results of the postulated accident scenarios such as typically the DLOFC event.

The HTMR100 is shown Fig. 1.

Fig. 1. HTMR100 Reactor.

2. The Htmr100 design Description

2.1. Design characteristics

The core design parameters of the HTMR 100 are shown in Table 1 supported by the reactor configuration in as shown in Fig. 1.

Table 1. Reactor core design parameters.

	Nominal value or description	Unit
Thermal power	100	MW_{th}
Primary coolant	Helium	
Moderator	Graphite	
Core geometry	Cylindrical	
Core volume	27.73	m^3
Average power density	3.6	MW_{th}/m^3
Pebble bed diameter	260	cm
Pebble bed height	~520	cm
Angle of defueling cone	30°	
Fuel circulation	1	Pass through core
Core inlet temperature	250	°C
Core outlet temperature	750	°C
Coolant flow rate	38.5	kg/s
Primary coolant pressure	4	MPa
Core loading	~150 000	pebbles/ core
Fuel circulation rate	~125	Pebbles /day
Nominal burn up	80 000	MWd/t_{HM}

The coupled neutron physics and thermo-fluid dynamics design of the HTMR100 reactor features three specific characteristics:

1. In the event of core heat-up the relatively strong negative temperature coefficient of reactivity will cause the nuclear chain reaction die down;
2. Two diverse independent shutdown systems are available. Each of these systems is capable of shutting the reactor down with fresh fuel when the reactor is cold and with one member (of highest reactivity worth) of the system stuck in the fully withdrawn position. Both these systems collectively is called the Reactivity Control and Shutdown System (RCSS). The one system used for operational reactivity control is the Reactivity Control System (RCS) while the system used for safety shutdown is the Reserve Shutdown System (RSS). These systems position the neutron absorbing material in channels in the side reflector. Both of these systems (9 RCS and 9 RSS) move in graphite sleeves located inside the graphite side reflector blocks. The rods in the RCS can be operated independently as a group or as sub-groups, as required by the reactor operating control system. A control/shutdown rod consists of several segments that contain the absorber material, linked together to form articulating joints. The segments each consist of sintered B_4C absorber material, sandwiched between an inner and an outer tube segment. The inner tube segment allows cooling helium gas to flow from the top down on the inside.
3. The design geometry of the core and core structures is chosen to remove residual heat solely by thermal conduction, thermal radiation, and natural convection. The Reactor Cavity Cooling System (RCCS) located outside the RPV is an investment protection measure available during operation and upset conditions to mainly protect the concrete in the reactor building from overheating. The maximum fuel temperature anywhere within the core remains below 1130 °C during normal operation and thus below any value that may lead to fission product release during operation.

The OTTO fuelling scheme has an axial peak in neutron flux at the top section of the reactor core where fresh fuel enters. The temperature profile for such a fuelling scheme is however flat since helium coolant enters at the top of the reactor and heats up as it moves downward through the reactor core. In addition, the top of the core contains fresher fuel than the bottom where the fuel is more depleted. The relatively cooler core at the top combined with fresher fuel results in a higher power density which in turn leads to a relatively high coolant heat-up rate in the top of the core as is evident from Fig. 12.

The excess reactivity in the HTMR100 is approximately 1700 pcm to provide mainly for a Xe-poisoning override during a 100–40 − 100 % thermal power transient during load following.

In summary, a cylindrical core of average height of ∼ 5.2 m, diameter 2.6 m, an OTTO fuelling scheme, inlet/outlet temperatures of 250 °C /750 °C and a unit power output of 100 MW_{th} must be observed per reactor if the maximum fuel temperature is to restrict the potential for any significant radiological release in a postulated DLOFC event.

The bypass flow can be estimated from previous reactor core simulations and experience as well as published data. The bypass flow is assumed to be 9.6 % of the total coolant flow through the core and core structures.

This 9.6 % is further broken down into a control rod cooling percentage 2.3 % (for the control rods in the HTMR100) and for a direct by-pass of 7.3 %. The bypass flow has not been explicitly modelled for the purposes of this paper but shall be considered as the design progresses.

2.2. Pebble flow characteristics

Information on the flow of pebbles within a pebble bed core is available from numerous German experimental programs documented in (Becker and Ragoß, 1993, Internal memoranda reporting on the investigation of pebble flow curves for the AVR., 2009). Pebble flow experiments were also reported by Kadak and Bazant in (Pohl, 1987) confirming the assumed pebble flow velocities and diffusive characteristics during pebble migration. It was found that the pebble flow is essentially vertical with pebbles moving only a few ball diameters in the horizontal or azimuthal direction. The bottom cone introduces non-vertical flow when directing the fuel spheres to the unloading tube. This effect is only noticed in the lowermost part of the pebble-bed closest to the cone.

The major influence on flow speed and therefore core residence time is fuel spheres in contact with the side reflector. Typically, fuel spheres within 1½ sphere diameter from the reflector will touch it and can therefore be delayed significantly relative to the other fuel spheres. Within the restrictions of the 2-D VSOP representation, the different flow speeds and flow lines are simulated in the VSOP model and is shown in Fig. 2. The inner core flow line is represented by the blue line, this is not showing the central fuel flow line which would be near vertical in shape, followed by the outer inner core flow line in red and the most outer core flow line in green. As can be seen in Fig. 2 the profiles of the lines change due to the cone located in the bottom of the reactor core with the outer fuel travelling slower towards the defueling chute compared to that of the middle and inner core fuel pebbles.

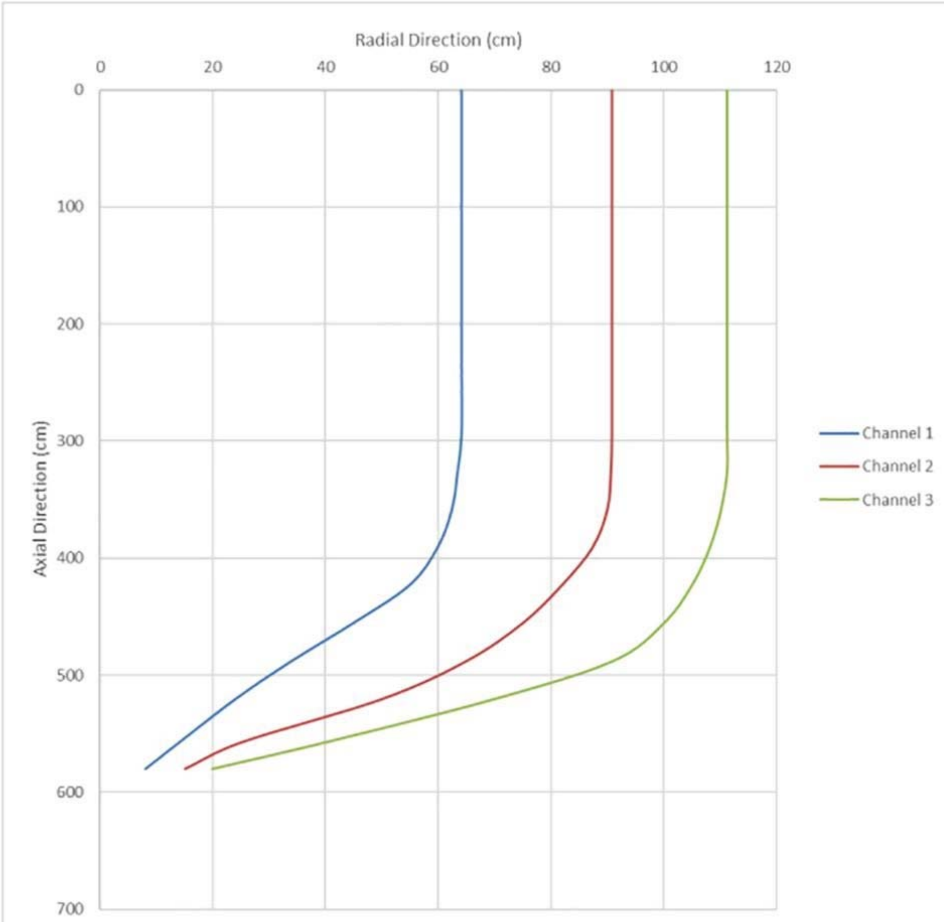

Fig. 2. Pebble flow lines for the HTMR100.

2.3. Fuel sphere characteristics

The defining characteristic of the pebble-bed reactor and the key to the safety and operational simplicity of the HTMR100 is the use of TRISO fuel particles embedded in fuel spheres, or pebbles, as depicted in Fig. 3.

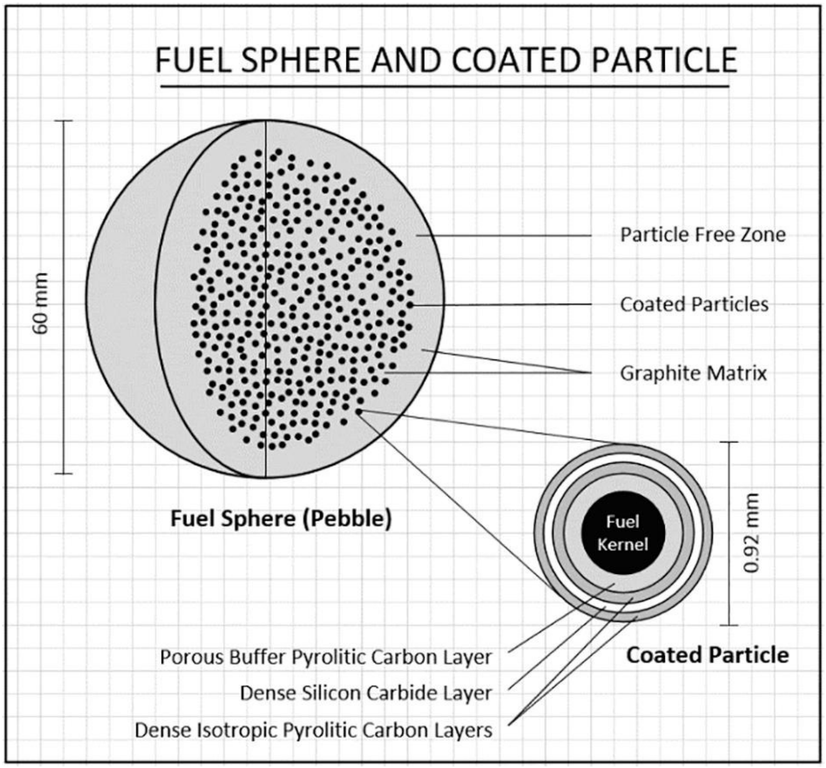

Fig. 3. Fuel sphere depicting multi-barrier TRISO coated particles.

Each pebble contains thousands of coated fuel particles, each consisting of a spherical fuel kernel surrounded by four coating layers. These layers all assist in the retention of fission products of which the SiC layer is the main barrier.

In other fuel cycles, the pebbles may contain fissile and/or fertile material.in

the coated particles such as, thorium, or plutonium in an oxide or carbide form. The pebbles are 60 mm in diameter and contain typically 12,000 ~ 17,000 coated particles. The pebble consists of an inner fuel zone of approximately 50 mm diameter in which the TRISO coated particles are embedded. A fuel-free zone of about 5 mm surrounds the fuel zone. The coated particle consists of an inner fuel kernel containing the fissile/fertile material and is surrounded by four coating layers (Kadak and Bazant, 2004). Each coating has its own function.

The HTR pebble bed has the flexibility to accommodate many different fuel cycles. The fuel cycles that are used in this simulation of the HTMR-100 reactor core is LEU in the form of UO_2.

Although the HTMR100 is designed for UO_2, it is not limited to the use of only UO_2 fuel shown in Table 2.

Table 2. Fuel parameters.

Description	Size	Unit
Fuel pebble outer diameter	6	cm
Thickness fuel free zone	0.5	cm
Fuel kernel diameter	500	μm
Kernel coating material	C/IPyC/SiC/OPyC	
Layer thickness	95/40/35/40	μm
Layer densities	1.05/1.90/3.18/1.90	g/cm^3
Total HM content per FS	10	g
Fuel enrichment	10	wt%
Fuel type	UO_2	

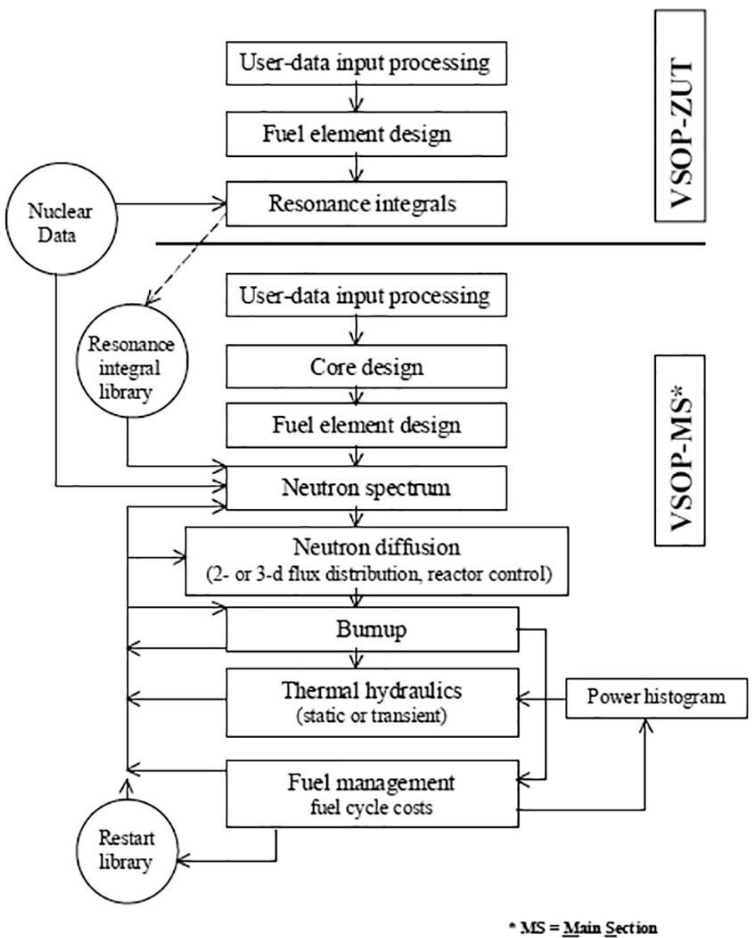

Fig. 4. VSOP code flow sheet for HTGR core physic calculations

3. Simulation methods and models

3.1. VSOP 99 code system

VSOP is a computer code system for the comprehensive numerical simulation of the physics of thermal reactors. The application of the code implies processing of cross sections, the set-up of the reactor and of the fuel element, neutron spectrum evaluation, neutron diffusion calculation, fuel burn-up, fuel shuffling, reactor control, and thermal hydraulics of steady states and transients. The neutronic calculations can be performed in three dimensions. The thermodynamic analysis is restricted to gas-cooled reactors in (r,z)-coordinates. Evaluation of fuel cycle costs over the reactor lifetime is performed using the present worth method (International Atomic Energy Agency (2010)). The VSOP suite of codes can simulate start-up core conditions as well as equilibrium core conditions. The code uses quasi steady state approximations for transient calculations. Fig. 4 shows the VSOP code flow sheet for HTGR core physic calculations.Fig. 5.shows the HTMR100 geometrical layout.

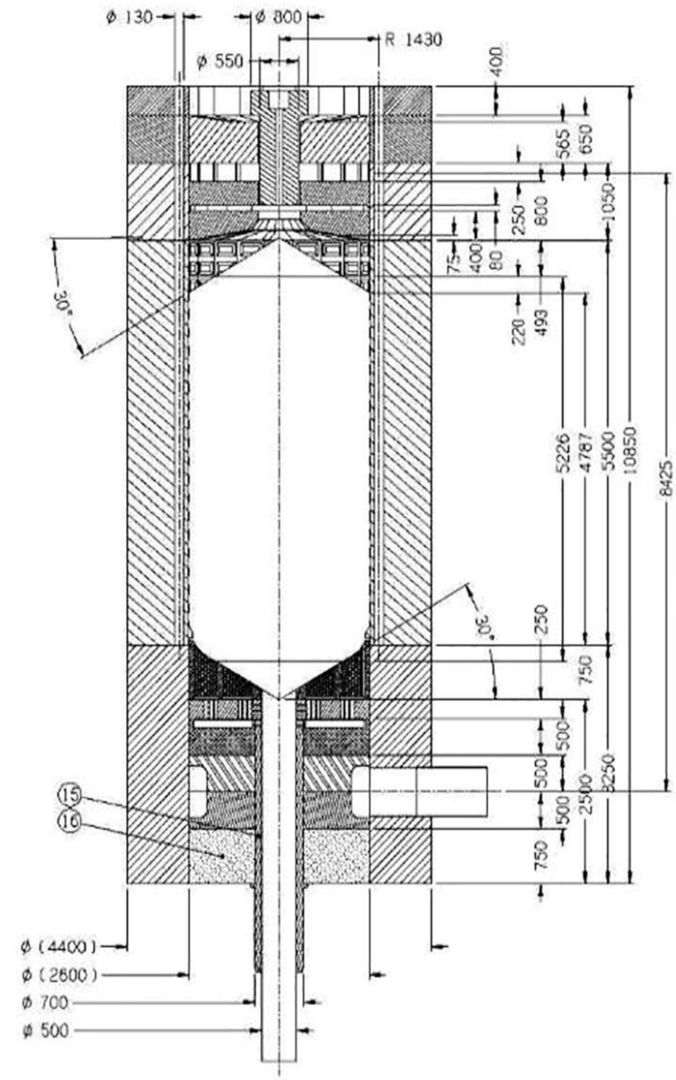

Fig. 5. HTMR100 geometrical layout.

The VSOP 99 system of codes is used to produce input models for the reactor geometry, fuel design, reactor physics characteristics analysis, fuel cycle convergence parameters are deployed to enable the achievement of steady state conditions. As a final step, the parameters are saved to provide restart conditions for in-depth transient studies using the MGT code. MGT performs group-dependent flux evaluations and transient simulations specifically with regards to accident simulations.

The VSOP suite of codes is used to simulate a finite reactor core geometry for the HTMR100 cylindrical reactor core 2D model using the diffusion-based calculation method to simulate various fuel cycles and compare the neutronic results (Hansen, 1975). The reactor core model is based on a conventional pebble bed reactor with the cold helium gas entering the side reflector at the bottom at 250 °C, flowing up in helium risers in the reflector and entering the top plenum at the top of the core. This gas then flows downwards through the core where it heats up and exits at the bottom of the reactor at 750 °C. The conventional material model for THERMIX utilizes a graphite reflector surrounded by carbon bricks, a core barrel and a reactor pressure vessel [12]. Table 3 lists the material selections for the reactor pressure vessel and its internal structures. The RCCS is not explicitly modelled in these simulations; however fixed fluid region boundary conditions are stipulated for both the VSOP and MGT models. A fluid region boundary (or heat sink) of 1 cm thick was assumed to create a natural heat transfer gradient and represent the RCCS.

Table 3. Reactor material selection.

Description	**Material**
Pebble Graphite	A3-3 Graphite
Reflector graphite	Reactor Graphite, SGL, Grade A, NBG10
Static Helium Gap	Helium (1 bar)
Core Barrel (CB)	Stainless Steel SA-240 grade 316
Reactor Pressure Vessel (RPV)	Steel SA-508
Static Air Gap	Air (1 bar)
Top/Bottom Plate	Steel SA-508
Carbon Bricks	Carbon
Fluid Region Model Boundary	Fluid at 50 °C

Furthermore, several important design features are important:

1. The side reflectors consist of isotropic nuclear-grade graphite.
2. A single-zone, cylindrical core with four ‘pebble-flow channels’ has been dimensioned to provide related calculation volumes – simplifying analyses – over the flow regions.
3. The effective core height is fixed at ~ 5.2 m and a core diameter of 2.6 m.
4. A cylindrical discharge funnel was modelled to simulate the discharge tube with a diameter of 0.5 m.

Fig. 6 shows the VSOP 99 geometrical model for the HTMR100 reactor layout. This model shows the discretization of the core, graphite internals, gaps, core barrel and RPV as well as the pebble flow lines associated with this geometry.

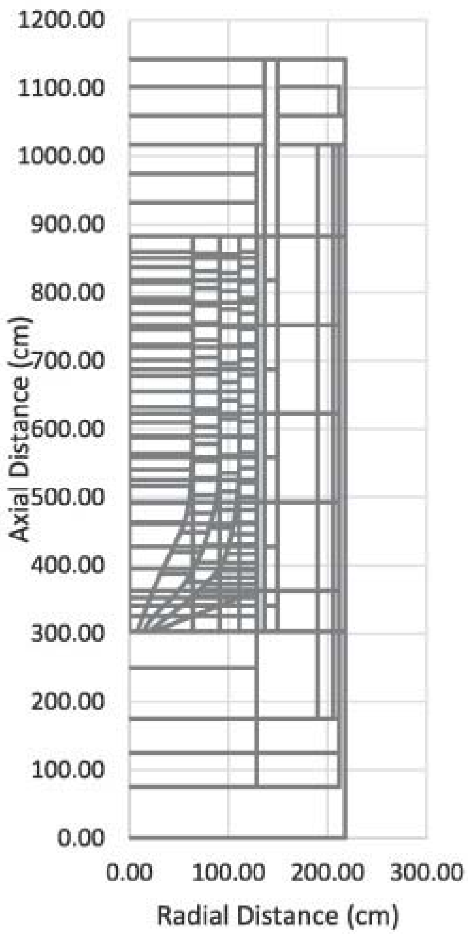

Fig. 6. VSOP 99 geometric model and pebble flow lines.

3.2. MGT code system

The MGT code deals with the nuclear and thermal transient behaviour of the primary circuit of a HTGR, considering mutual feedback effects in two-dimensional R-Z geometry.

MGT assesses the dynamic transient results such as a Load Following (LF) as well as the accident scenarios such as a Depressurized Loss of Forced Cooling (DLOFC), Pressurized Loss of Forced Cooling (PLOFC), Control Rod Withdrawal (CRW) and a Control Rod Ejection (CRE) event. The modular structure of the MGT code can be seen from the schematic in Fig. 7.

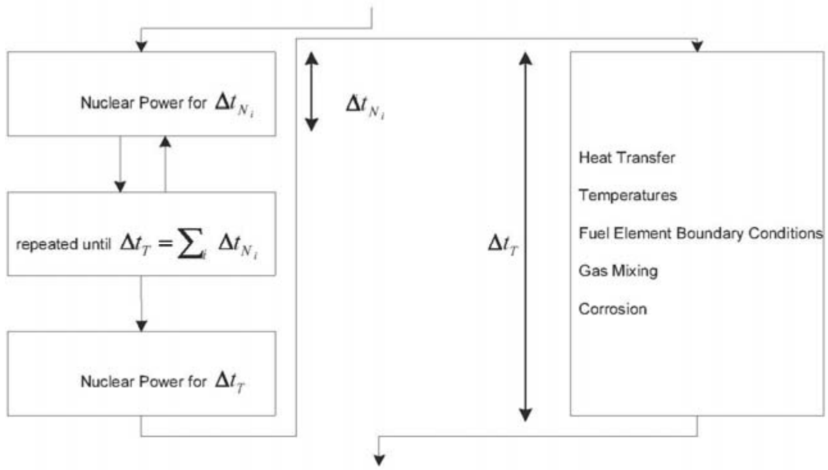

Fig. 7. Modular structure of MGT.

4. Results

4.1. VSOP 99 equilibrium core

Results presented in this section are for the equilibrium cycle and with the control rods positioned at the 'equilibrium' position with its bottom end at a depth of 65 cm below the bottom of the top graphite reflector. The thermal flux averaged over the core volume is calculated at $\phi_{avg} = 3.9x10^{13}\ n/(cm^2.s)$ at 100 MW_{th}.

Table 4 provides an overview of the HTMR100 calculated performance data at equilibrium conditions.

Table 4. HTMR100 performance parameters at equilibrium conditions.

	Value	**Unit**
Change in the overall performance		
Fuel type	UO_2	
Fresh fuel enrichment (in U-235)	10	wt%
Average core enrichment during equilibrium operation	6.22	wt%
Core volume	27.73	m^3
	Value	Unit
Heavy metal content per fuel sphere	10	g
Number of fuel passes through core	1	
Heavy metal loading at start of cycle (fresh core load)	1379	kg
Heavy metal inventory	14348.95	Kg/GW_{th}
Fuel inventory		
U −234	11.57	Kg/GW_{th}
U −235	752.31	
U −236	132.52	
U −237	0.14	
U −238	13250.72	
U −239	0.01	
Np-237	6.90	

Np-238 Np-239 Pu-238 Pu-239 Pu-240 Pu-241 Pu-242 Am-241 Am-242 *m* Am-243 *Cm*-242 *Cm*-244	0.01 1.83 1.97 102.84 45.39 28.98 11.13 1.00 0.02 1.18 0.17 0.25	
Fuel supply - discharge		
U −234 U −235 U −236 U −238 Np-237 Pu-238 Pu-239 Pu-240 Pu-241 Pu-242 Am-241 Am-243 *Cm*-244	0.0122–0.0082 1.2398–0.3876 0.0000–0.1450 11.146–10.608 0.0000–0.0098 0.0000–0.0036 0.0000–0.0962 0.0000–0.0523 0.0000–0.0354 0.0000–0.0179 0.0000–0.0042 0.0000–0.0022 0.0000–0.0005	Kg/GWD_{th}
k_{eff}	1.0000	
Average burn-up	80552.4	MWd/tHM
Maximum burn-up	86740.0	MWd/tHM
Average residence time in core	1227.0	EFPD
Number of spheres per day	124.1 ~ 125	
Particle packing fraction in the pebble fuel zone	9.5	%
	Value	Unit
Thermal performance		
Power peaking factor	2.20	Qmax/Qavg
Maximum power rating of fuel sphere	1.45	kW/FS
Maximum fuel temperature during		
Normal operations	822	°C
DLOFC event (VSOP99)	1548.6	°C
Helium temperature variation at the core outlet:	750	°C
ΔT He = maximum – minimum	750–250 = (5 0 0)	°C
Neutronic performance		
Neutron balance Fractional neutron productions by		
U −235 U −236 U −238 Pu-239 Pu-240 Pu-241 Am-242 *m*	64.40 0.02 0.44 29.32 0.01 5.77 0.02	%
Neutron balance Fractional neutron losses		
Neutron losses in heavy metals	76.77	%
Neutron losses especially (esp.) in fissile isotopes	51.42	%
esp. in U −234 esp. in U −235 esp. in U −236	0.15 32.38 0.51	%

esp. in U −238 esp. in Np-237 esp. in Np-239 esp. in Pu-238 esp. in Pu-239 esp. in Pu-240 esp. in Pu-241 esp. in Pu-242 esp. in Am-241 esp. in Am-243	20.07 0.13 0.03 0.02 16.34 4.23 2.71 0.10 0.05 0.02	
Fission products	6.13	%
esp. in Xe-135	1.86	%
Neutronic performance		
Neutron leakage	15.51	%
Conversion ratio	0.471	
Source neutron/fissile absorbers.	1.945	
	Value	Unit
Capture/fission in fission materials.	0.331	
Fast dose spent fuel elements. e + 21/cm2	2.91	
Fast neutron exposure (energy group 1)		
Maximum upper edge	0.41	E + 21(cm^2 × 365D)
Maximum lower edge	0.03	E + 21(cm^2 × 365D)
Maximum outer edge	1.17	E + 21(cm^2 × 365D)
Thermal neutron flux (energy group 4)		
Maximum upper edge	0.57	E + 14/(cm^2 × *sec*)
Maximum lower edge	0.10	E + 14/(cm^2 × *sec*)
Maximum outer edge	0.98	E + 14/(cm^2 × *sec*)
Average thermal flux	0.39	E + 14/(cm^2 × *sec*)
Average total flux	1.23	E + 14/(cm^2 × *sec*)

Notes: * EFPD = Equivalent Full Power Days.

4.2. Power, flux and temperature distributions for steady state operations

4.2.1. Power distributions

The power profile is a perfect representation of a OTTO fuel cycle, since the fresh fuel enters at the top of the reactor where the fuel is relatively cool. As the fuel slowly moves downward the fissile quantity drops and the resultant power production drops. This is the reason why the OTTO cycle gives you a very high maximum to average power distribution. This can be seen in Fig. 8.

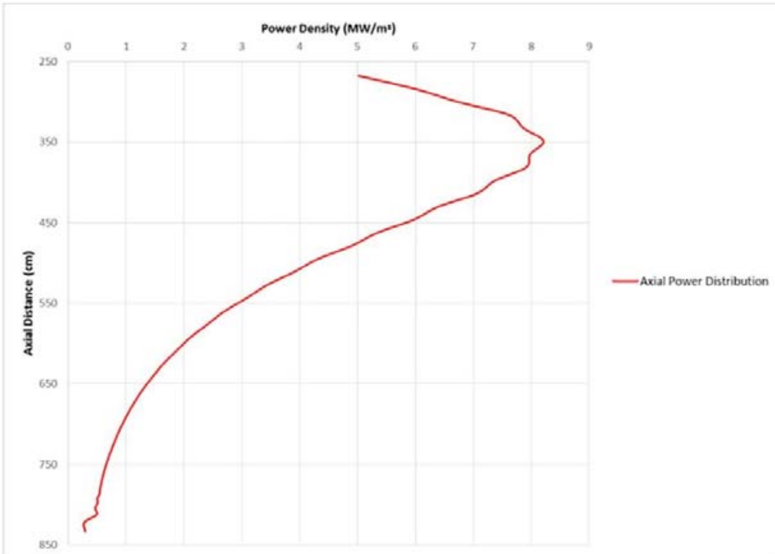

Fig. 8. Axial power distribution.

The radial distribution also shows clearly that the reflector peak producing power on the outer edge of the core due to neutrons being thermalized and reflected, this can be seen in Fig. 9.

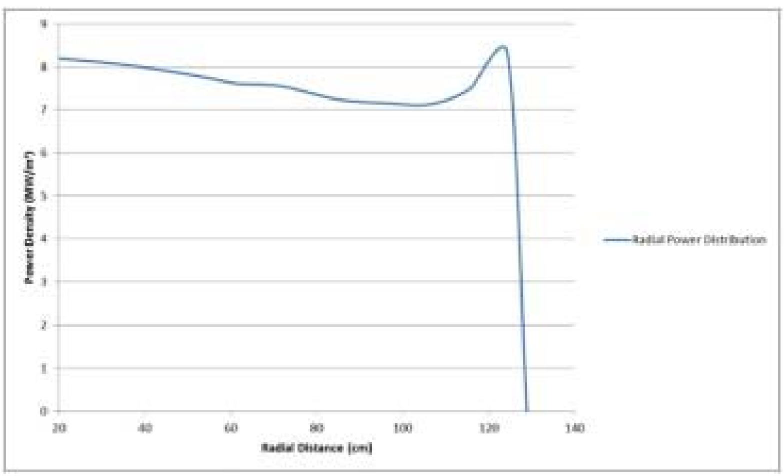

Fig. 9. Radial power distribution.

4.2.2. Flux distributions

In Fig. 10 the fast flux is represented by the blue line has no peaks. The thermal flux represented by the red line shows the peaks due to thermal neutron reflection on the top and bottom of the core.

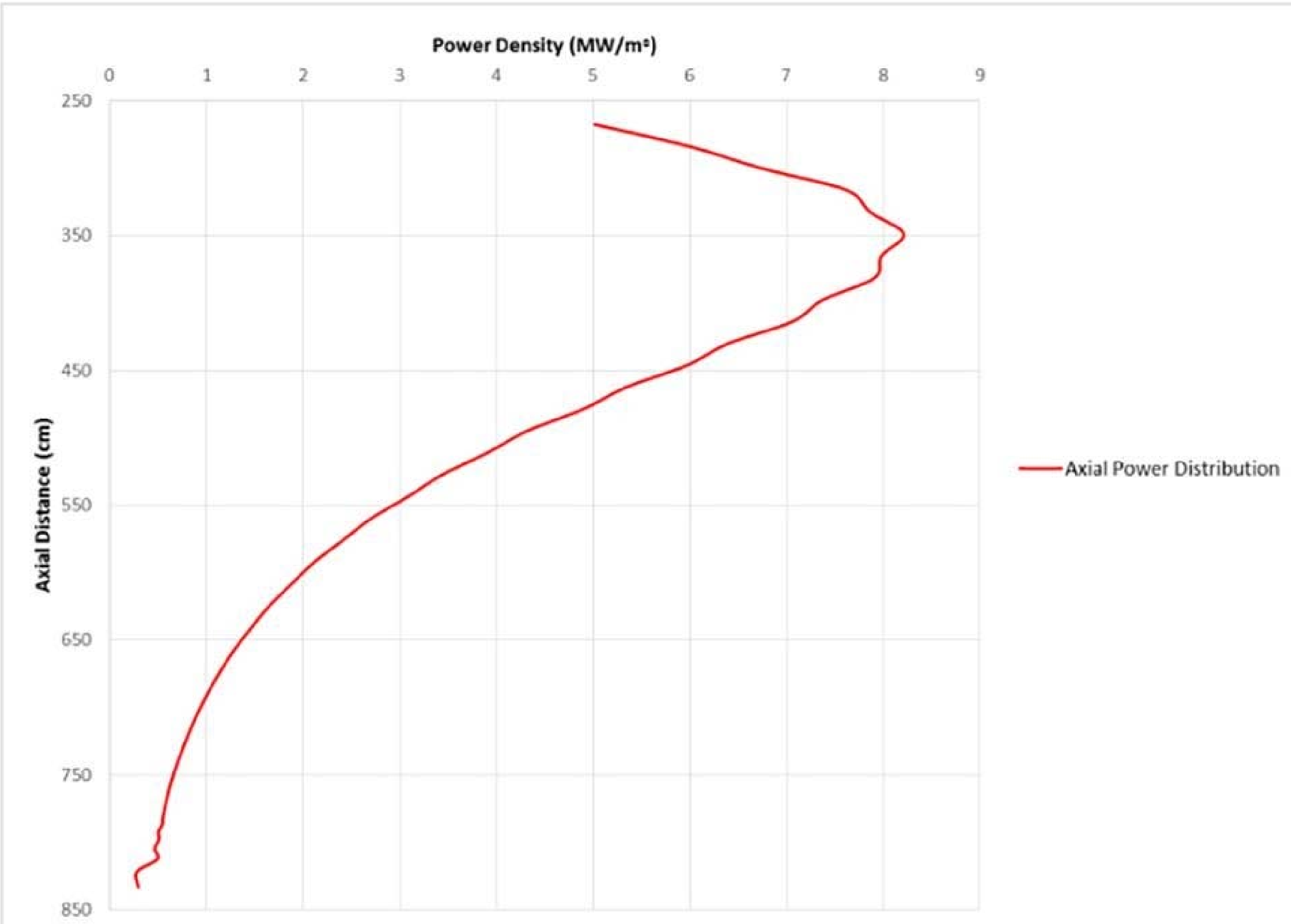

Fig. 8. Axial power distribution.

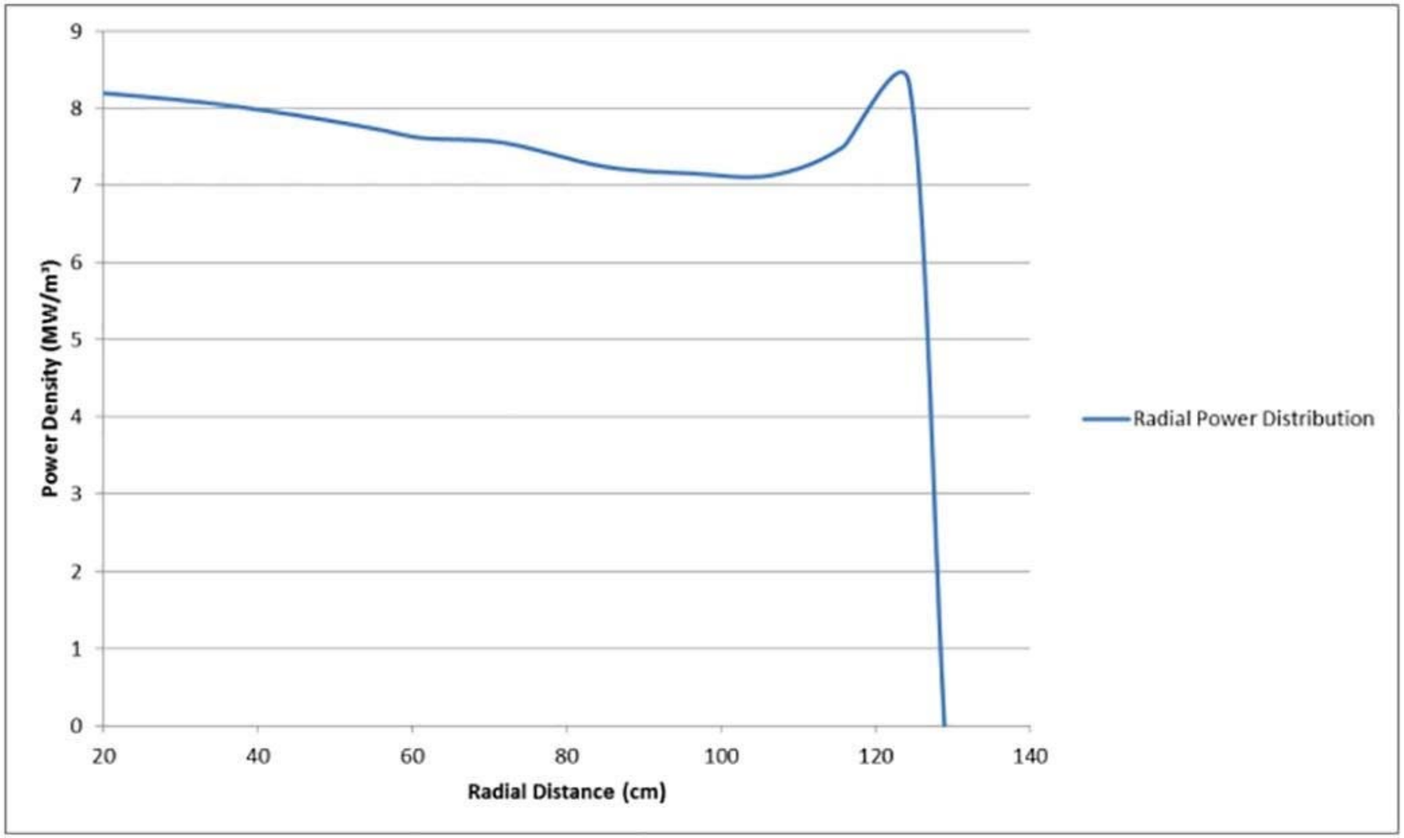

Fig. 9. Radial power distribution.

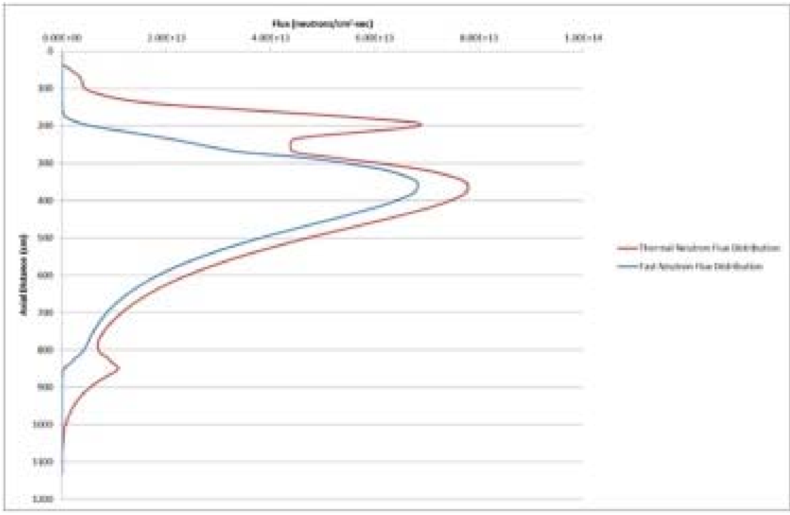

Fig. 10. Axial flux distribution.

The radial thermal flux distribution has a pronounced peak on the outer radial edge due to the thermal neutron reflection as seen in Fig. 11.

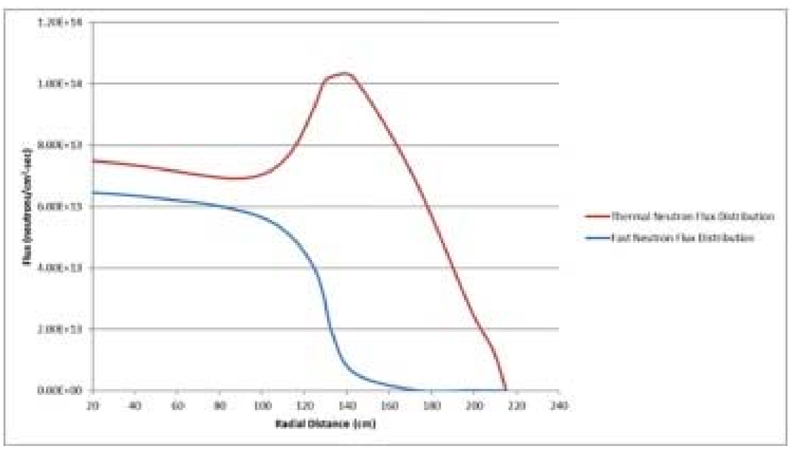

Fig. 11. Radial flux distribution.

4.2.3. Temperature distributions

The temperature of the fuel in the centre of the core is the highest and increases towards the bottom part of the core. The OTTO cycle provides a relatively flat axial temperature profile within the core after the high initial coolant heat-up rate. This can be seen in Fig. 12.

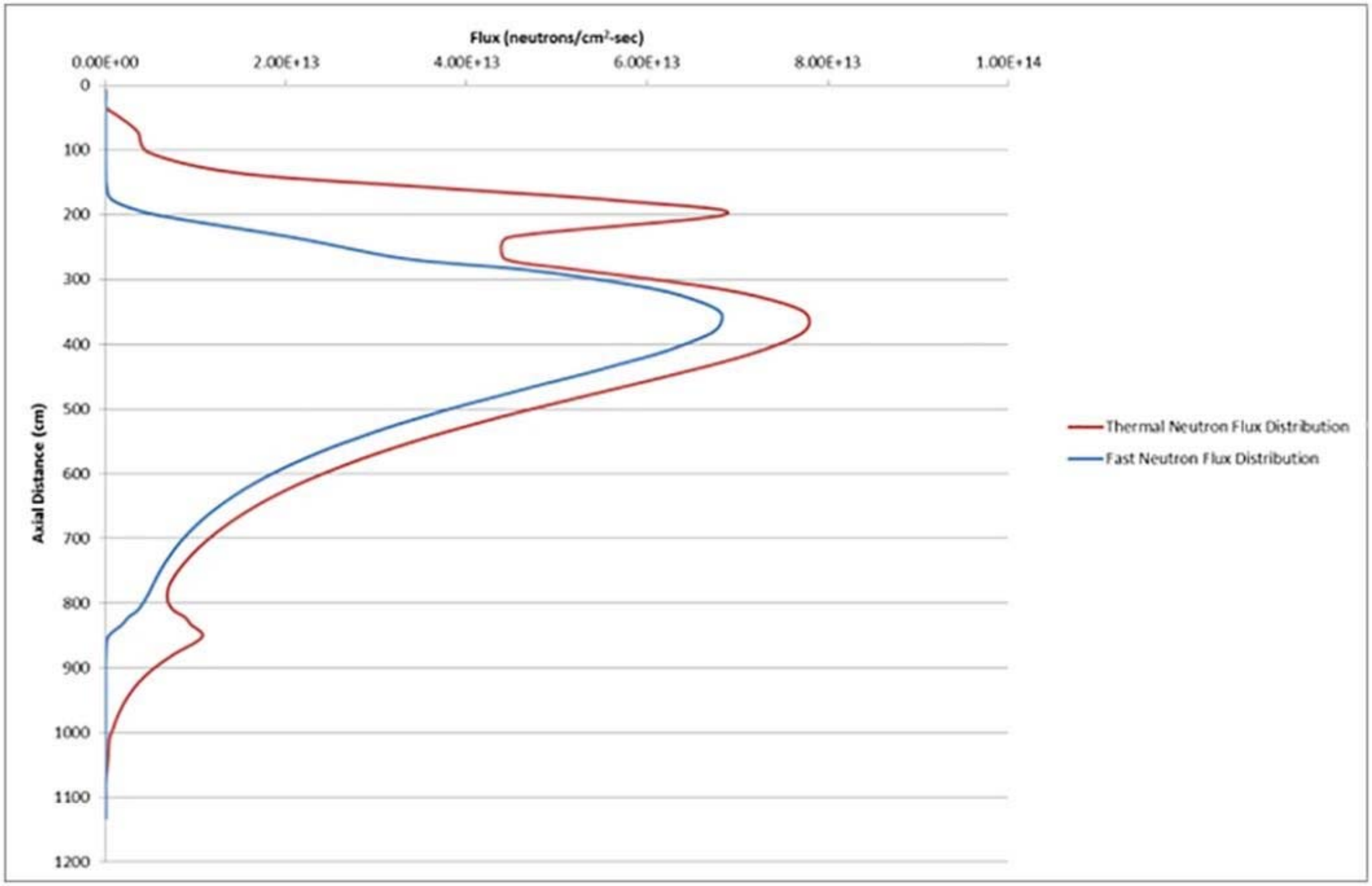

Fig. 10. Axial flux distribution.

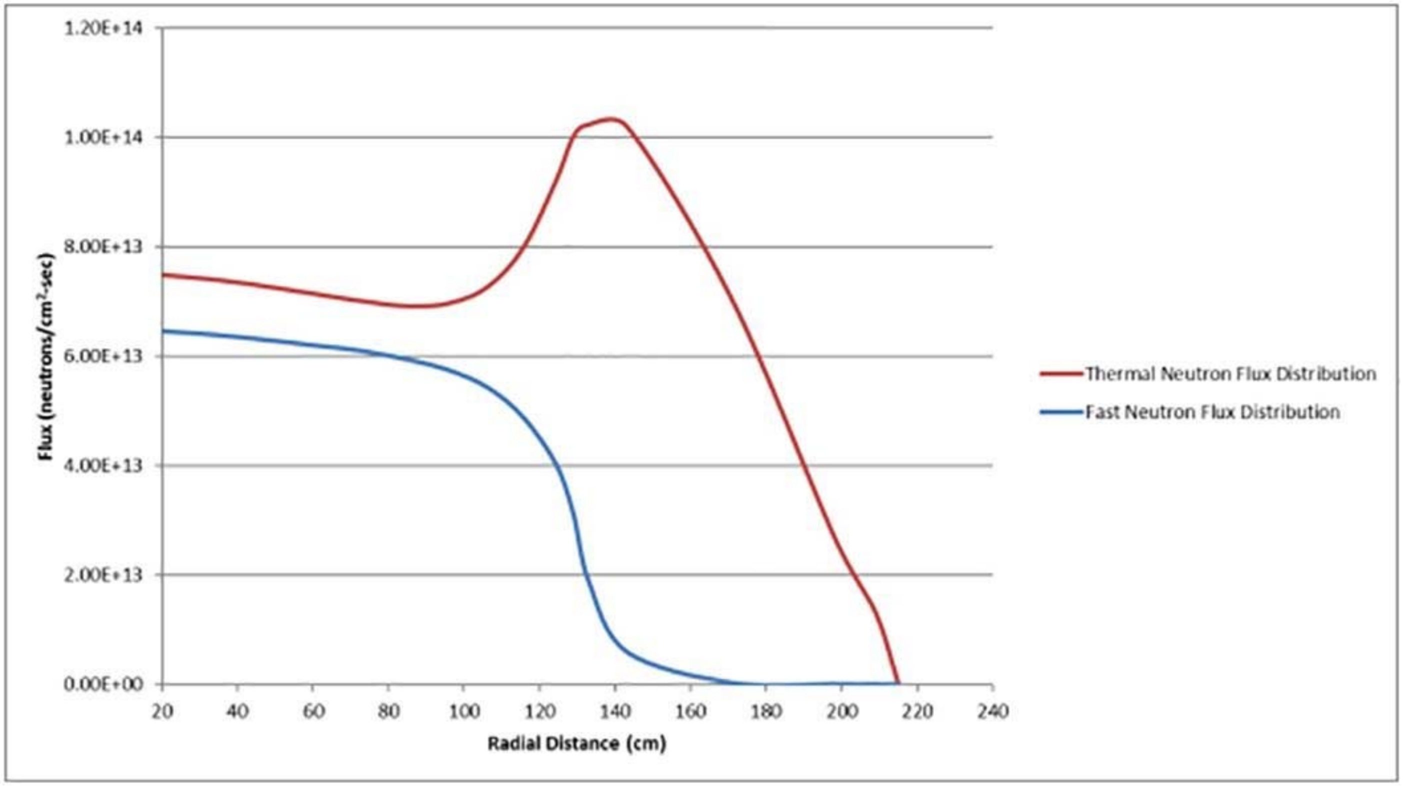

Fig. 11. Radial flux distribution.

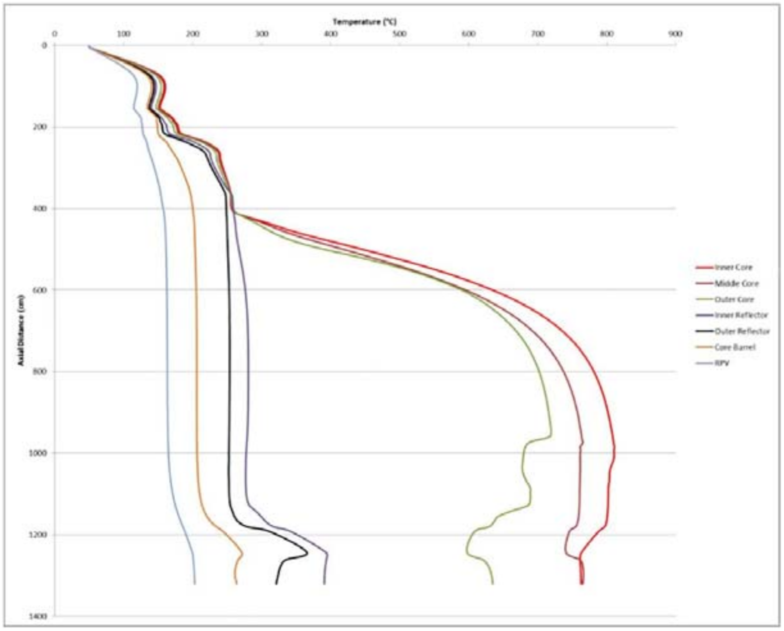

Fig. 12. Axial temperature distribution.

The radial temperature distributions also show the inner core being the hottest with the temperatures decreasing outwards. The core barrel and RPV operates well within their specified material temperature ranges. This is shown in Fig. 13.

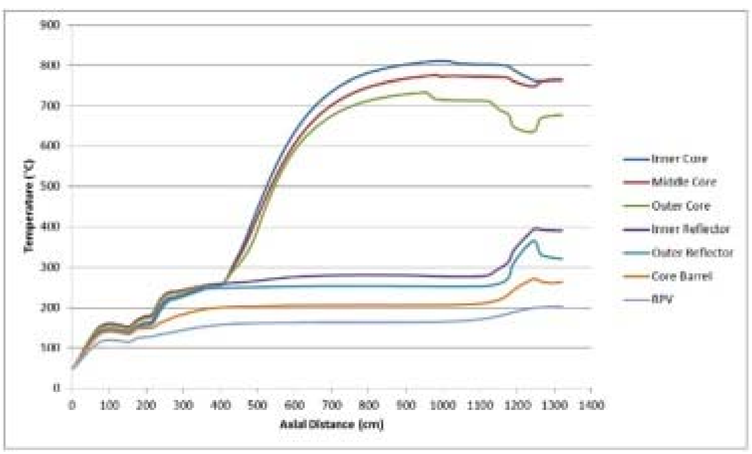

Fig. 13. Radial temperature distribution.

4.3. VSOP 99 DLOFC results

Fig. 14 shows the VSOP99 results of the analysis of the fuel temperature history after a DLOFC event has occurred. The maximum fuel temperature of 1548.6 °C is reached at 18-hours after the onset of the DLOFC event. VSOP99 is not a tool to perform transient analysis and takes a quasi-steady state approach to perform this simulation. The results do however provide a good correlation between VSOP99 and MGT, which is shown in the section below. The MGT maximum DLOFC temperature is 1563.30 °C (between 17 and 18 h) which is a

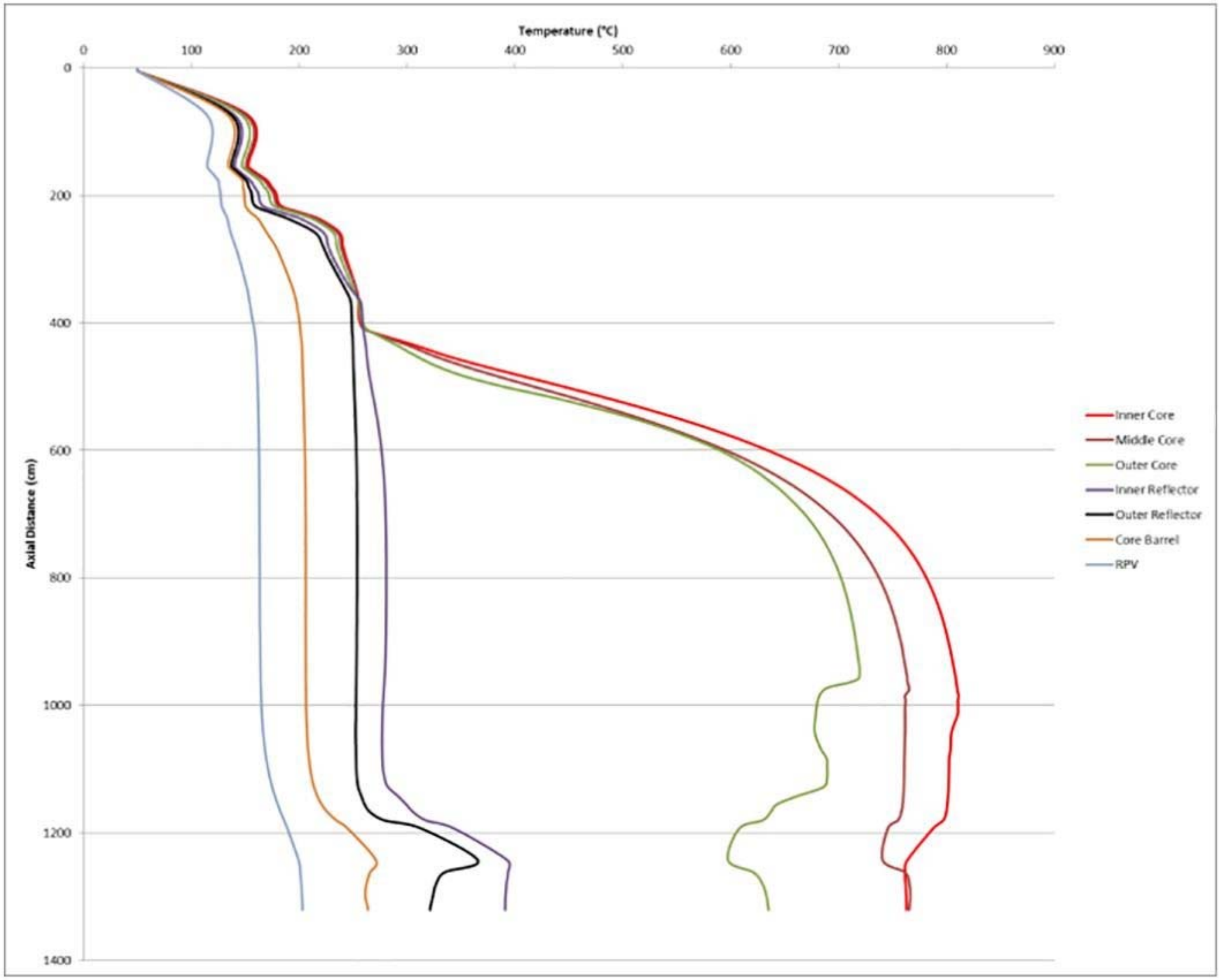

Fig. 12. Axial temperature distribution.

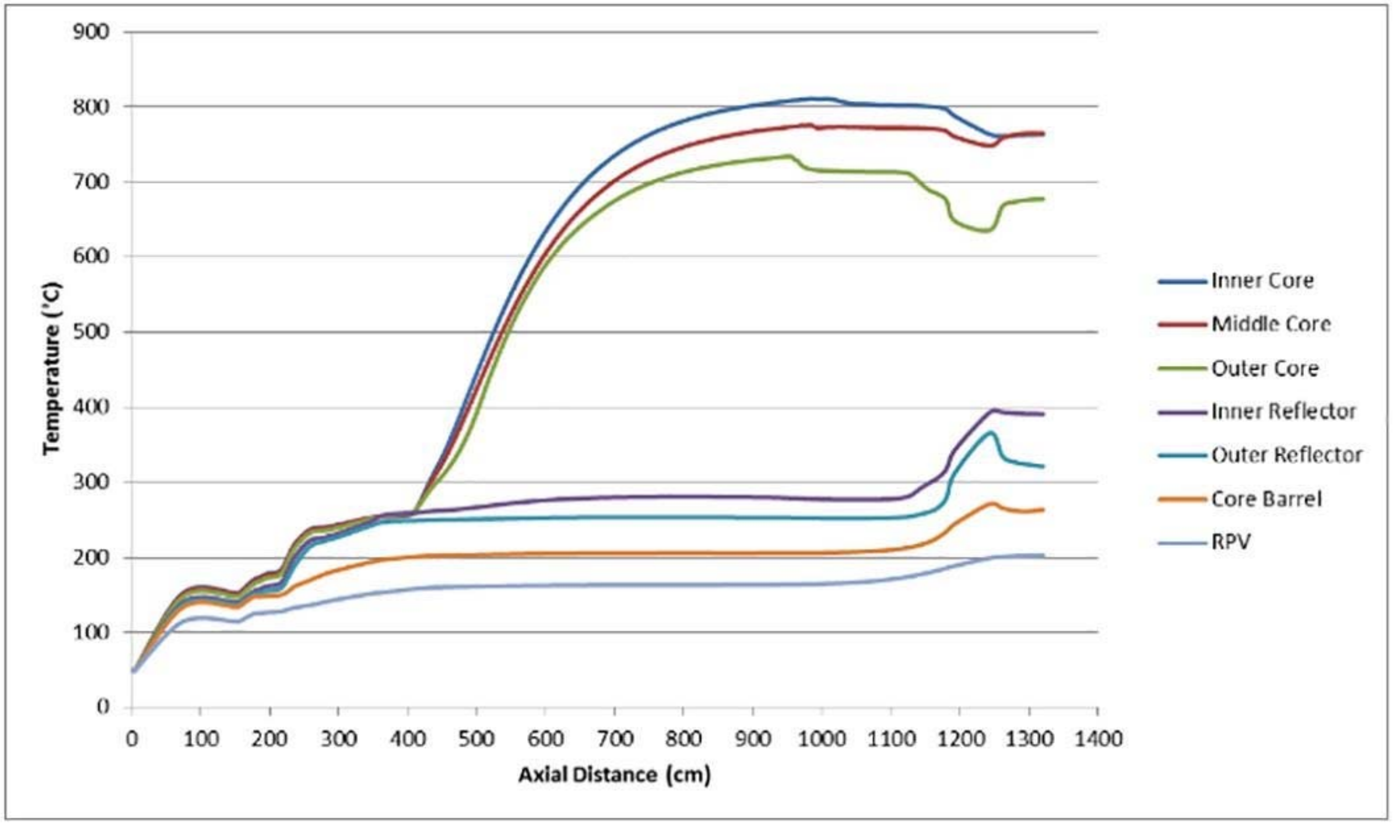

Fig. 13. Radial temperature distribution.

difference of 14.7 °C and should provide some degree of validation of the VSOP calculations or vice versa.

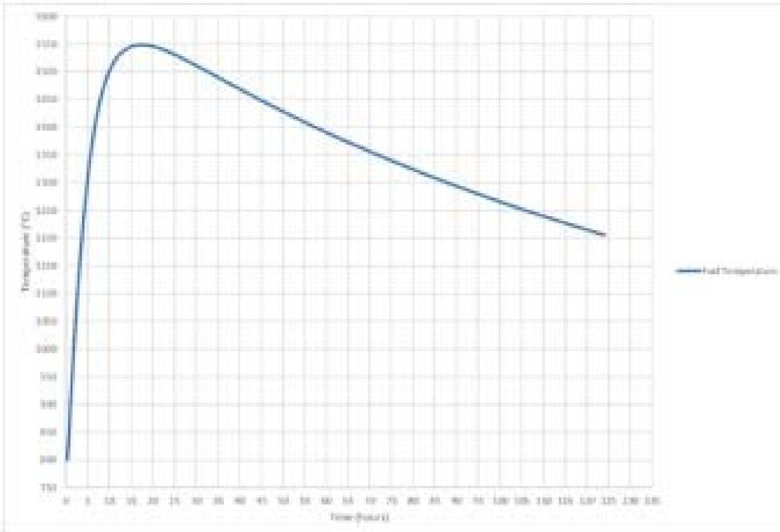

Fig. 14. VSOP 99 Maximum DLOFC temperature results.

Fig. 15 shows the temperature-volume analysis of a DLOFC design-base event which indicates that only about ~ 2 % of the entire fuel volume in the reactor would experience temperatures between 1500 °C − 1600 °C in a DLOFC event.

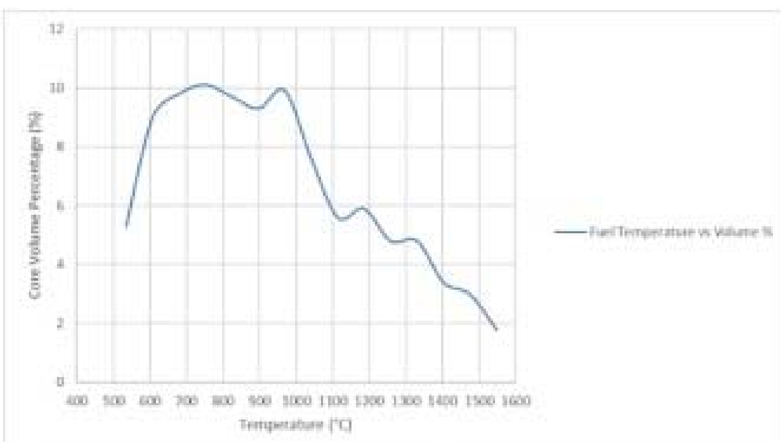

Fig. 15. VSOP 99 temperature-volume analysis.

4.4. MGT transient results

The MGT code package assesses the dynamic transient result, which includes in this case a 100 %-40 %-100 % Load Following operation as well as the various accident scenarios such as a DLOFC, PLOFC, CRW and a CRE event.

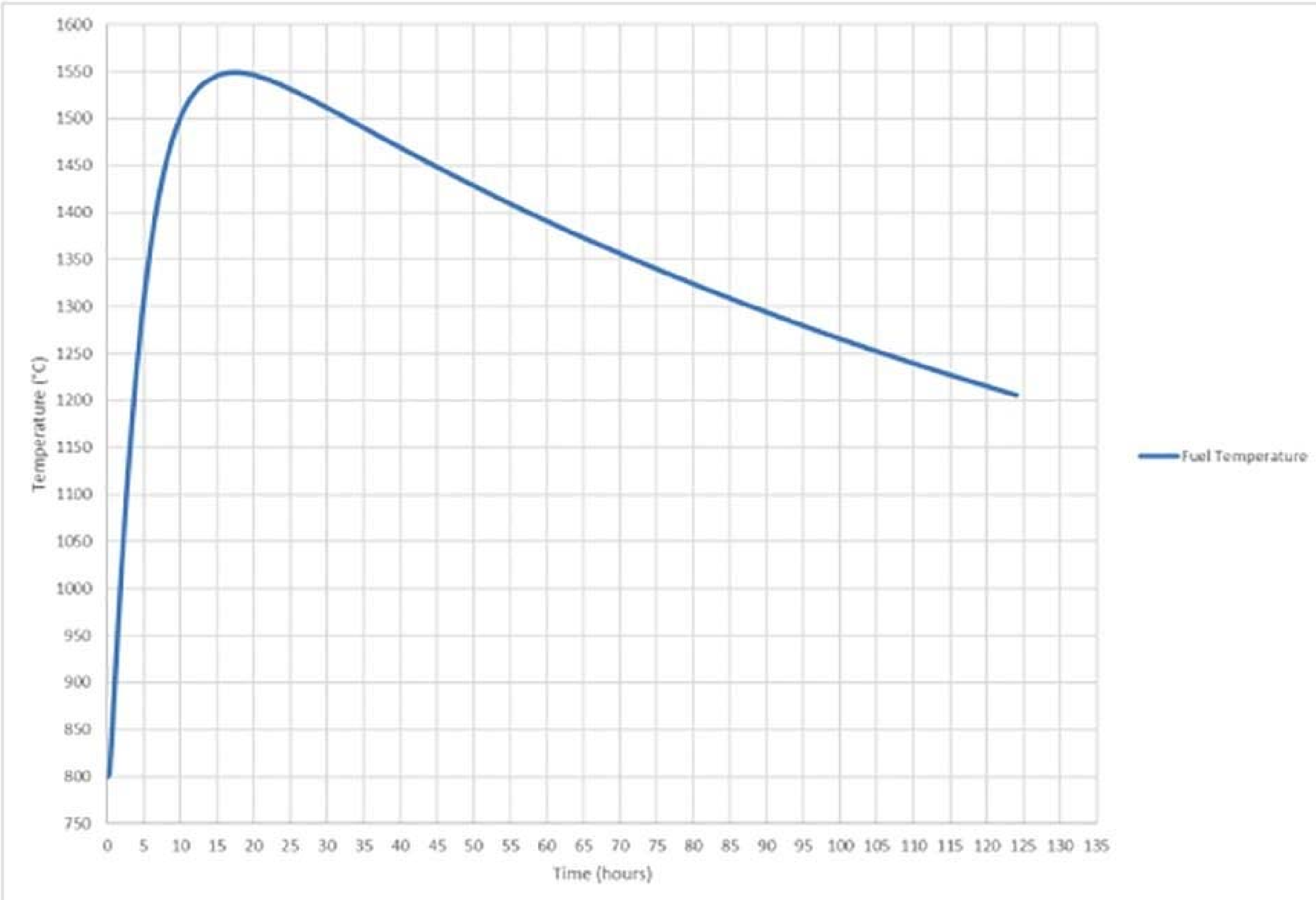

Fig. 14. VSOP 99 Maximum DLOFC temperature results.

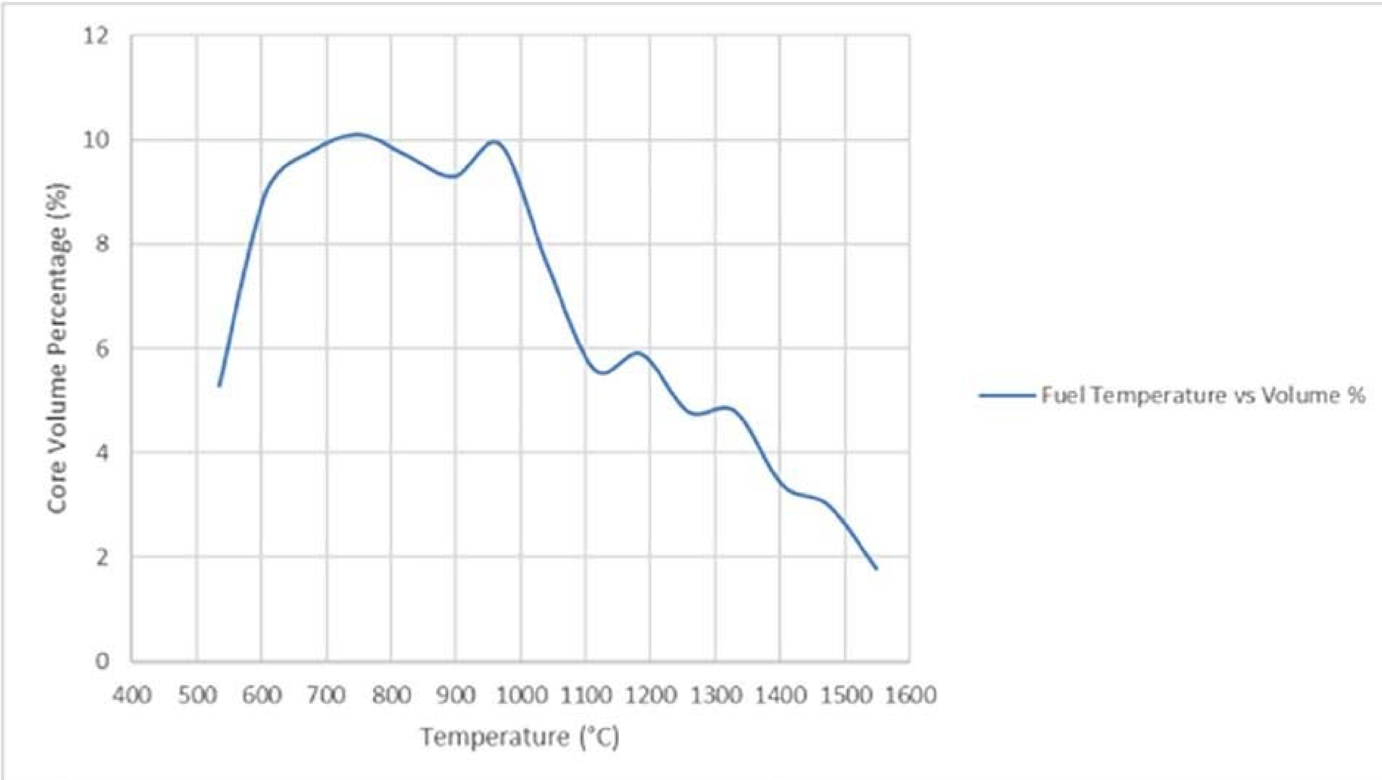

Fig. 15. VSOP 99 temperature-volume analysis.

4.4.1. DLOFC with SCRAM

Fig. 16 shows the maximum fuel temperature for a DLOFC event with a rapid insertion of control rods (SCRAM action) The average moderator and fuel temperatures are also shown for a DLOFC event and lie on the same line. The maximum fuel temperature is 1563.30 °C.

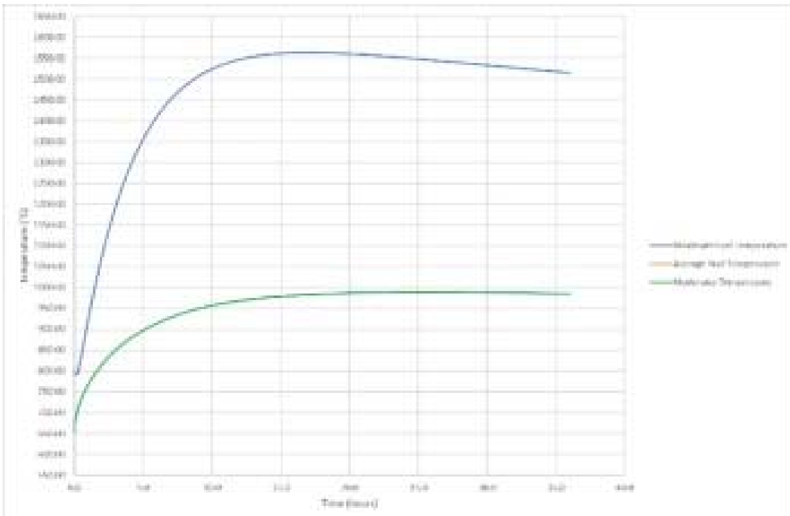

Fig. 16. Temperatures following a DLOFC with SCRAM (local heat).

Fig. 17 shows the maximum fuel temperatures for both a DLOFC and PLOFC events. It must be noted that the PLOFC fuel temperature is significantly lower (1168.4 °C) than the DLOFC fuel temperature (1563.3 °C) due to the removal of heat by convective and conductive heat transfer by the helium that remains in the core at pressure. During a DLOFC event, a vacuum is assumed in the core and no gas is available for heat transfer thus increasing the maximum fuel temperatures.

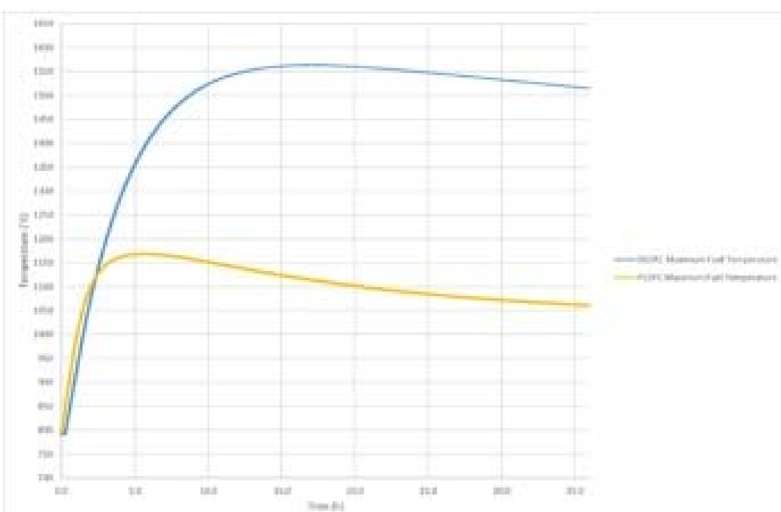

Fig. 17. Maximum fuel temperatures following a DLOFC & PLOFC with SCRAM (local heat).

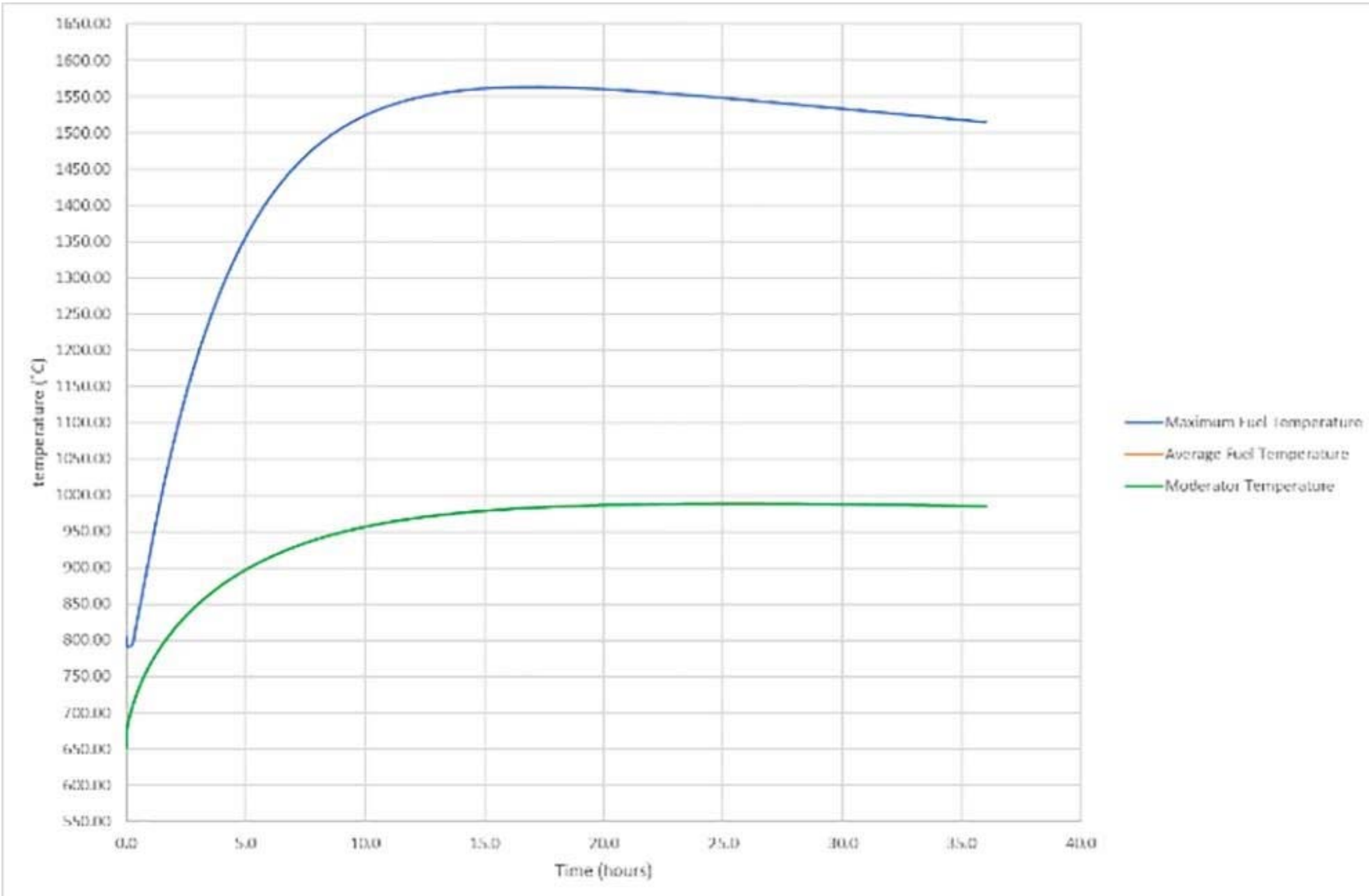

Fig. 16. Temperatures following a DLOFC with SCRAM (local heat).

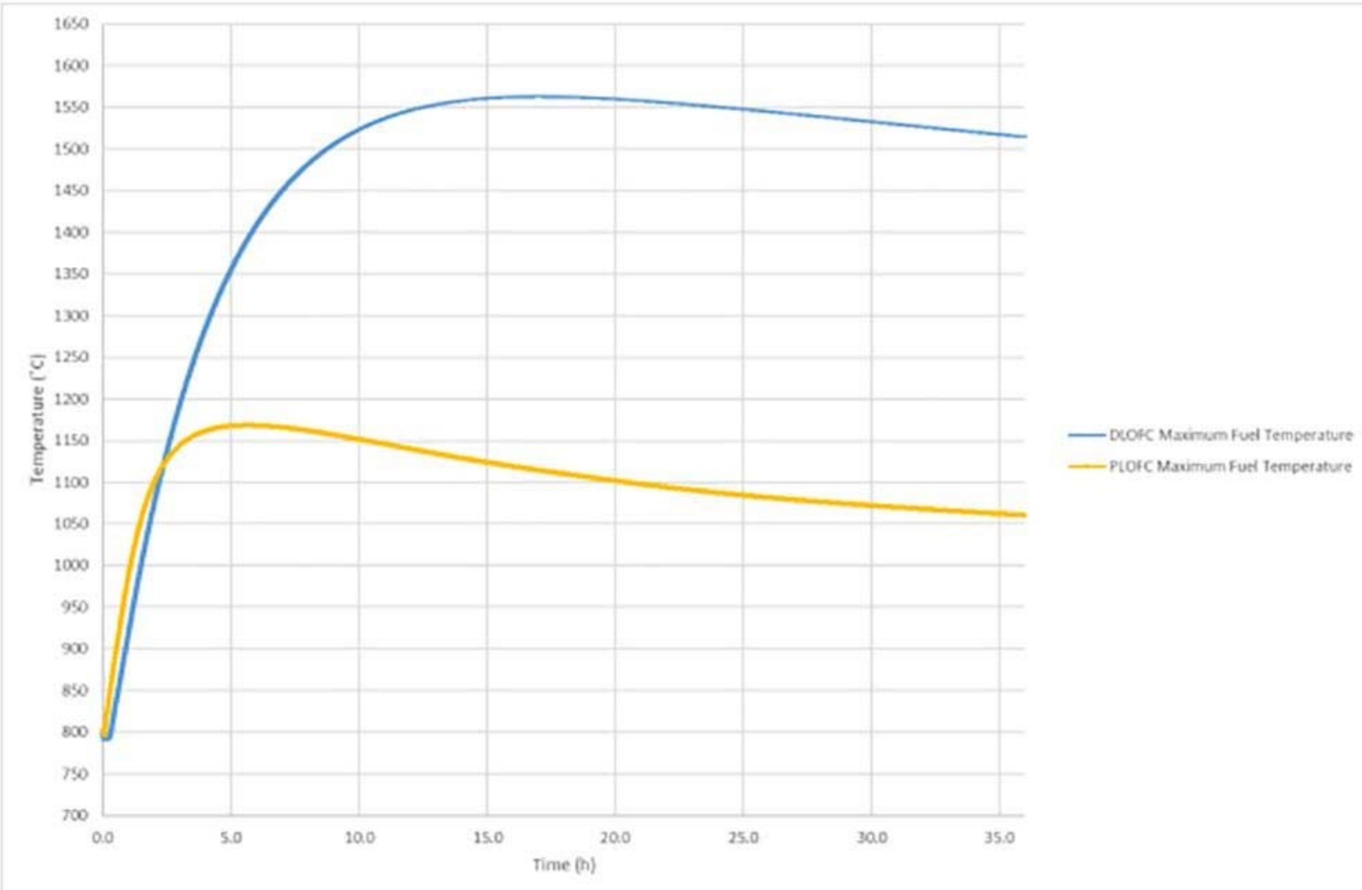

Fig. 17. Maximum fuel temperatures following a DLOFC & PLOFC with SCRAM (local heat).

In Fig. 18 with the maximum DLOFC fuel temperatures for both the local (in core heating) and non-local heating (in core and reflector heating) cases with SCRAM are shown. The non-local heating case (core and reflector heat generation) has a higher maximum fuel temperature of 1583 °C compared to the local heating case (in core only) of 1563 °C, this is due to the heat build-up which occurs within the reflectors due to radiation interaction within the materials which generates heat. When this occurs the heat transfer gradient between the core and the reflectors is reduced and a hot reflector will not absorb as much energy as a cooler reflector. The heat built up in the core in the local heat only case leads to a slightly higher maximum fuel temperature. If the reflectors are to be cooled the non-local maximum heat curve would reduce to a lower value than the local heat curve. The Non-Local case ran for 24 h whereas the Local case ran to 36 h. This made no difference as the only interest was the maximum fuel temperatures for these two scenarios.

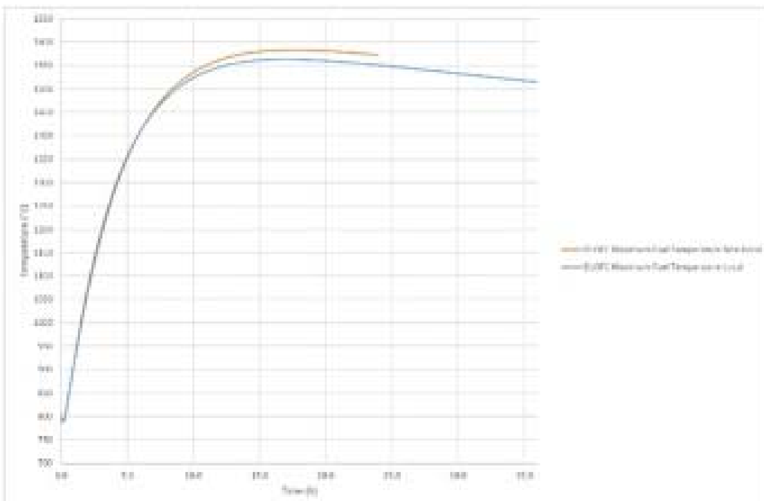

Fig. 18. Maximum fuel temperatures following a DLOFC with SCRAM (local & non-local heat) - no cooling in the reflectors.

4.4.2. Load following

In Fig. 19 a load following is initiated at the 0-hour mark where the power is reduced from 100 % to 40 %, kept at 40 % for 6 h and ramped back up to 100 % power. The reactivity vs Xe-135 % is shown for a period of 48 h. The figures to follow will enlarge the images from 0 to 15 h where the Load Following occurs. The enlarged images shows the load following phenomena.

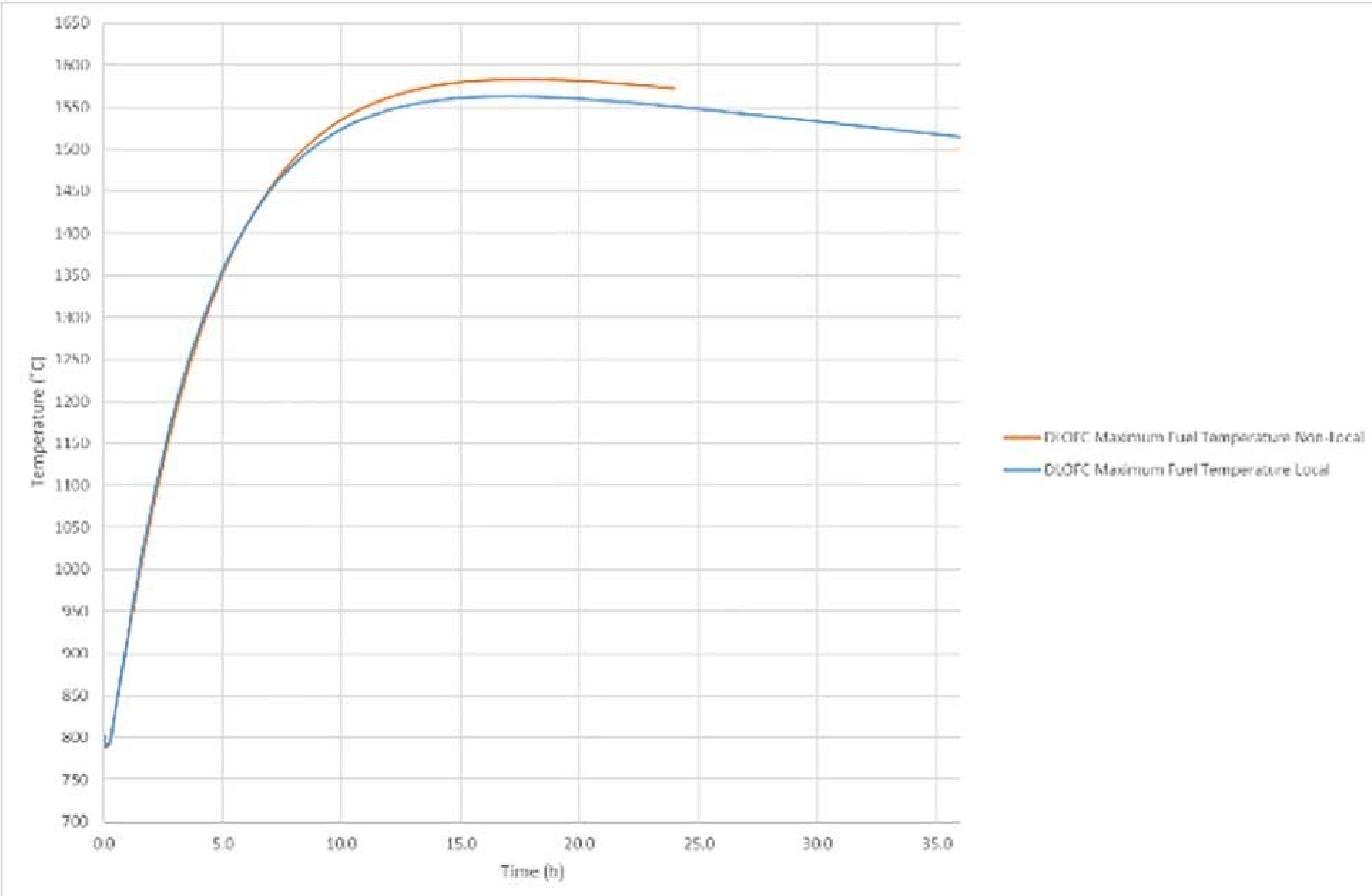

Fig. 18. Maximum fuel temperatures following a DLOFC with SCRAM (local & non-local heat) - no cooling in the reflectors.

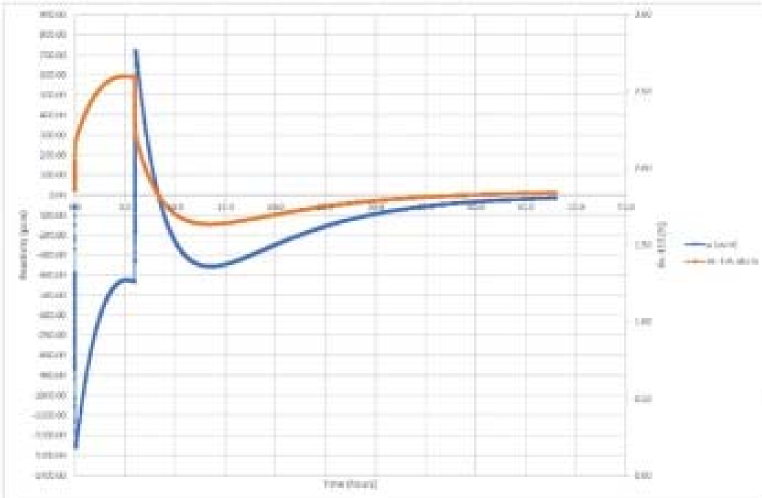

Fig. 19. 100–40-100 load following (reactivity vs Xe-135 vs time).

In Fig. 20 the reactivity vs power percentage is shown for a 100 %-40 %-100 % Load Following. When the reactor steps from 40 % power back to 100 % power there is a spike in the reactivity. The spike is 719.38 per cent mille (pcm). The control rods need to be able to counter this spike in reactivity from their reactivity difference between their nominal positions to the removed conditions. This difference is called the dynamic rod worth.

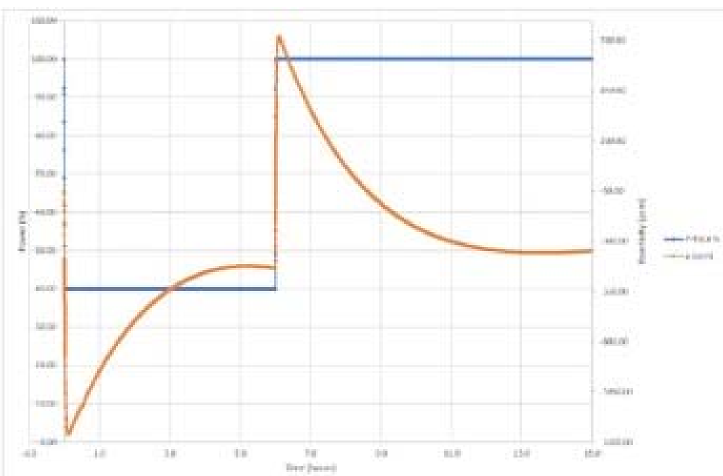

Fig. 20. Power vs reactivity (over 15 h).

As can be seen in Fig. 21 the Xe-135 % increases as the power level decreases due to the lower neutron flux at 40 % power. There are less neutrons at lower power levels which are available to interact with xenon, thus an increase in xenon is seen. At higher power levels there are higher neutron flux that consume xenon. It peaks at the point where the reactor steps back up to 100 % power.

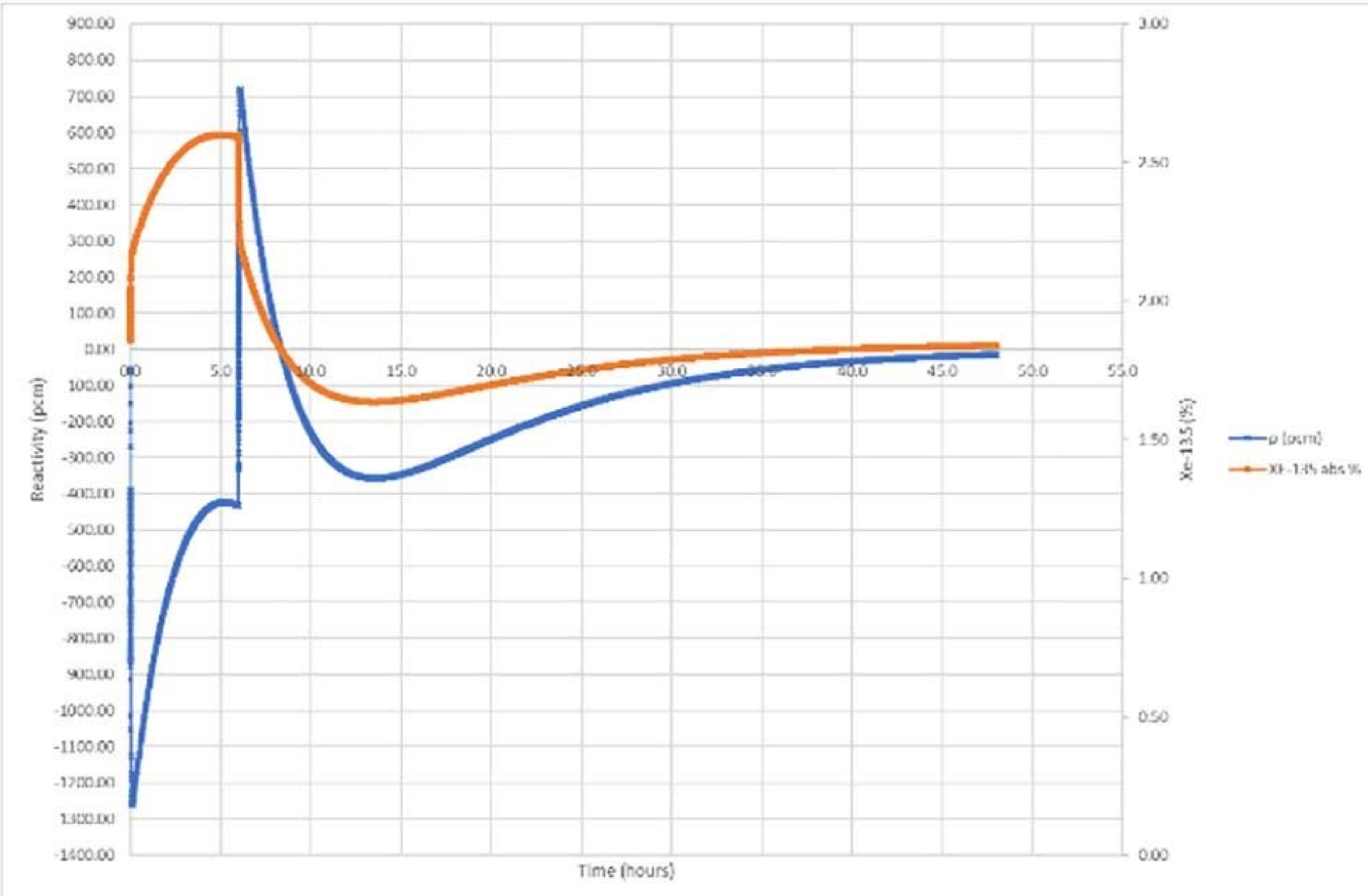

Fig. 19. 100–40-100 load following (reactivity vs Xe-135 vs time).

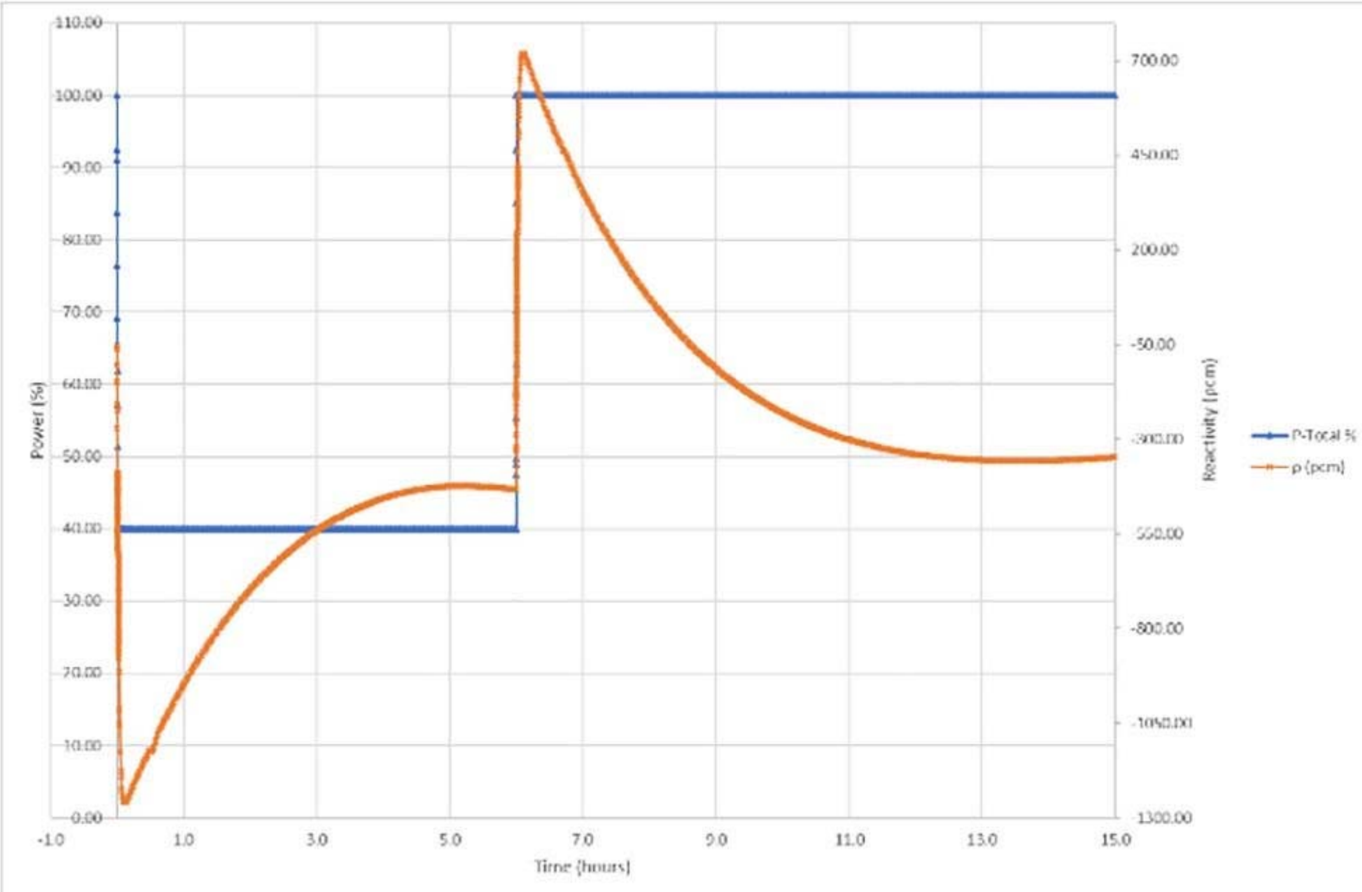

Fig. 20. Power vs reactivity (over 15 h).

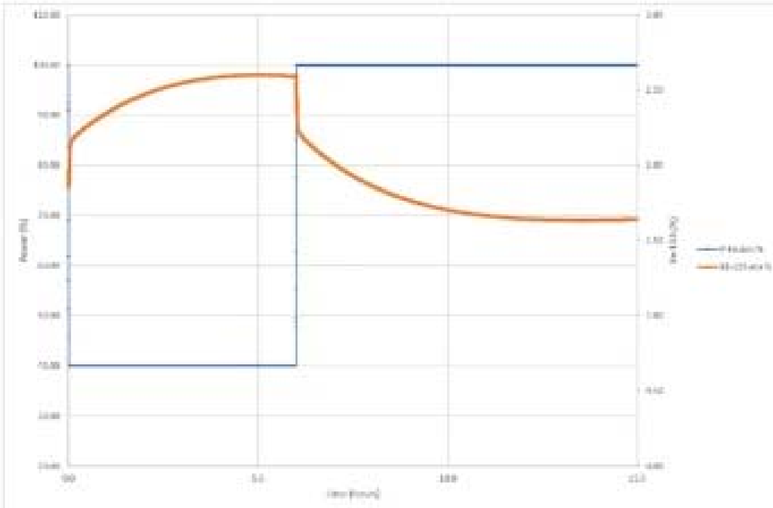

Fig. 21. Power vs Xe-135 (%) (over 15 h).

Fig. 22 shows the reactivity vs xenon % over 15 h, enlarged at the 6-hour mark. The xenon transient after the load follow transient does return to its original value see also Fig. 19 for the full view.

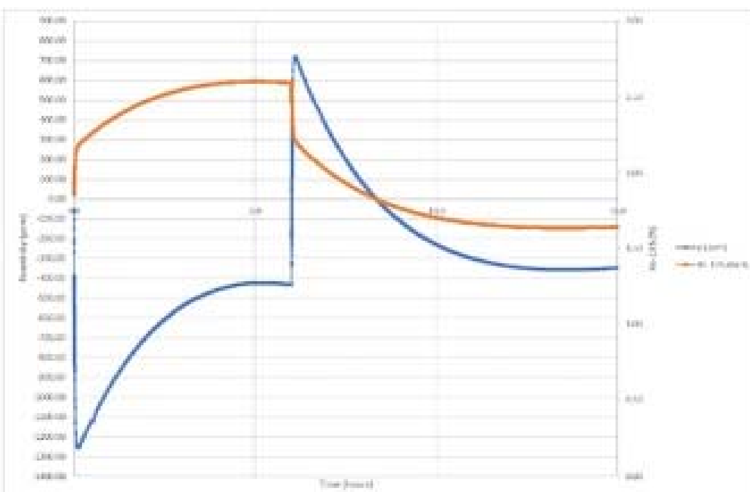

Fig. 22. Reactivity vs Xe-135 (%) (over 15 h).

In Fig. 22 the reactivity vs Xe-135 absorption % is shown for the LF of 100 %-40 %-100 %. The reactivity decreases once the 100 %-40 % power reduction occurs as a result of the Xenon build-up due to the decay of the Iodine fission product that was produced earlier before the power reduction. The control rods need to be withdrawn slightly from the core to allow excess reactivity to counter this xenon build-up. Table 5 shows the k_{eff} values for VSOP and MGT at equilibrium conditions, the difference in k_{eff} is also shown Δ (k/k) for the two codes.

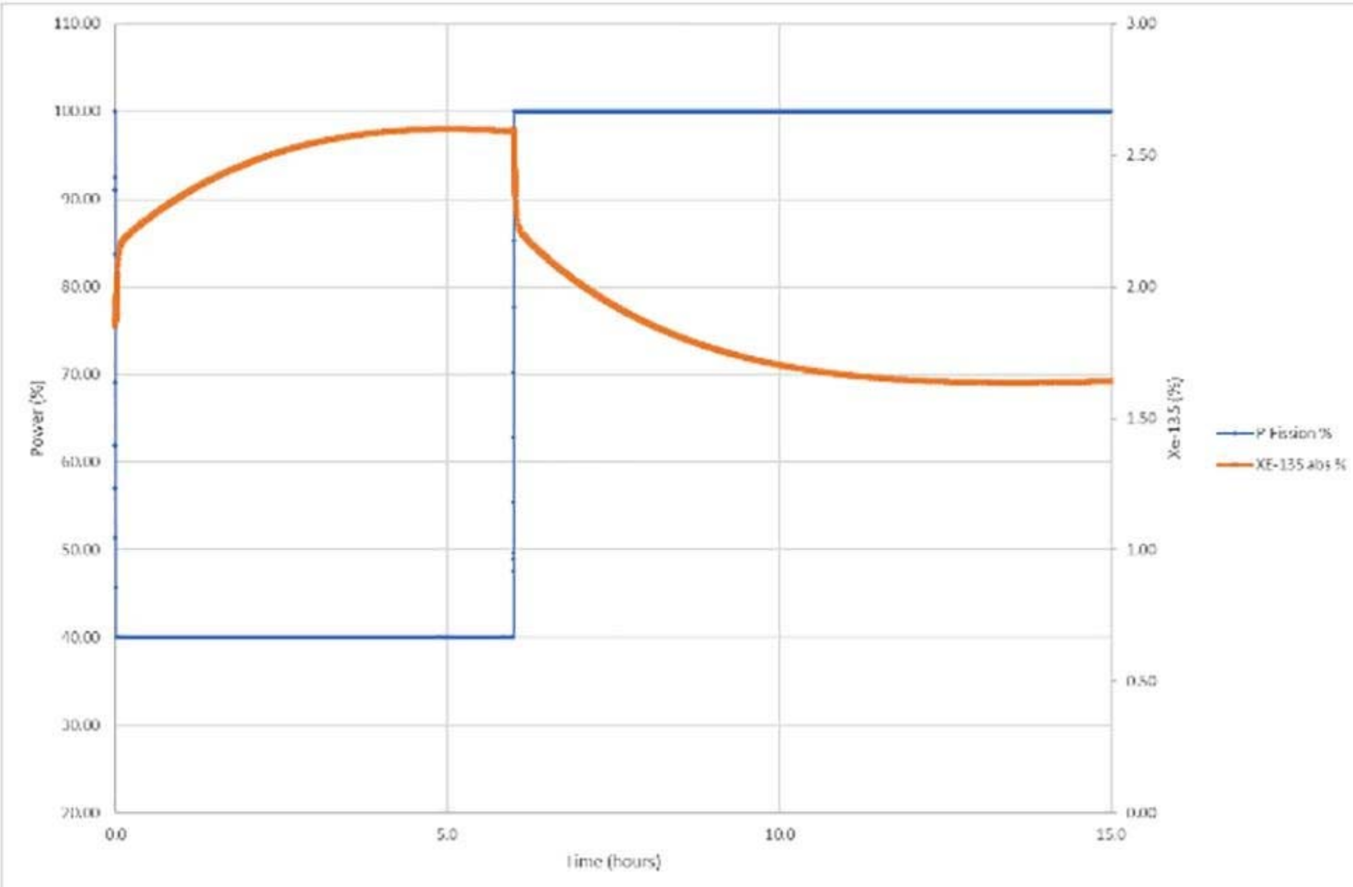

Fig. 21. Power vs Xe-135 (%) (over 15 h).

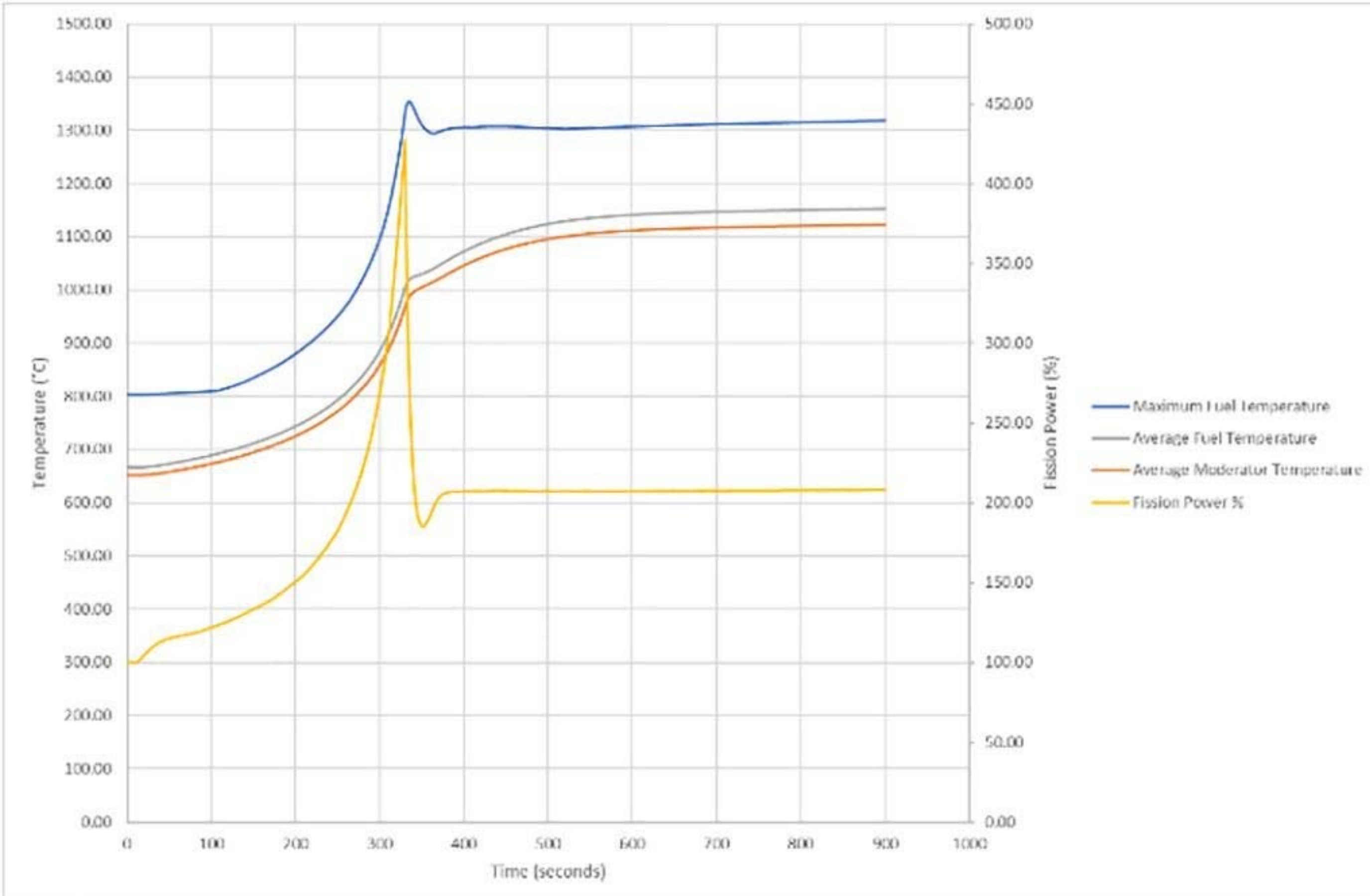

Fig. 22. Reactivity vs Xe-135 (%) (over 15 h).

Table 5. MGT control rod position k_{eff} values.

VSOP (k_{eff})	MGT (k_{eff})	Δ (k/k) (pcm)
0.9992	0.9882	1100.6512

Table 6 shows the MGT k_{eff} values for the control rods situated at the equilibrium (nominal position), rods out position and rods inserted position to highlight the differences in reactivity of the control rods at various positions.

Table 6. VSOP and MGT equilibrium (nominal position) k_{eff} values.

Description	k_{eff}
Equilibrium (Nominal Position)	0.9882
All Rods Out (ARO)	1.0124
All Rods In (ARI)	0.8726

Notes: * ARI = All Rods In.

* ARO = All Rods Out.

The reactivity difference for the control rod positions is shown in Table 7. This shows the difference in k_{eff} for the control rods fully withdrawn compared to the equilibrium (nominal position), between the nominal position and the rods fully inserted as well as between the rods fully withdrawn and fully inserted.

Table 7. MGT control rod worth values.

Description	Δ (k/k) (pcm)
ARO – Equilibrium Nominal Position (Static Rod Worth)	2417.6542
ARI - Nominal Position	−11556.5318
ARI - ARO	−13974.1861
Dynamic Rod Worth (Peak in Reactivity ρ on 100–40-100 LF Graph)	719.3800

The control rod positions are identified to see whether a 100 %-40 %-100 % LF is indeed possible.

The CRs need to be able to cover the spike in reactivity when the reactor is ramped back up to 100 % power from 40 % power and the CRs also need to have enough excess reactivity from a specific insertion depth that when the xenon poisoning occurs the CRs can be withdrawn without the reactor becoming sub-critical in a LF due to the increase in Xe-135.

The All Rods Out (ARO) minus the nominal control rod position provides a static rod worth of 2417.65 pcm which will in fact counter the dynamic rod requirement of 719.38 pcm as stated before. A 100 %-40 %-100 % LF is easily possible. A greater reduction in load following is possible due to the difference between the static and dynamic rod worth, for example a 100 %-25 %-100 % LF but this was not part of the study. The excess reactivity is also shown here with a $\Delta k_{eff} \sim 1700$ pcm (difference between static and dynamic rod worth).

4.4.3. Control rod Withdrawal

The CRW transient which is a slow withdrawal of all of the control rods at approximately 1 cm/second is shown in Fig. 23 and Fig. 24. Such a transient takes about 300 s. These

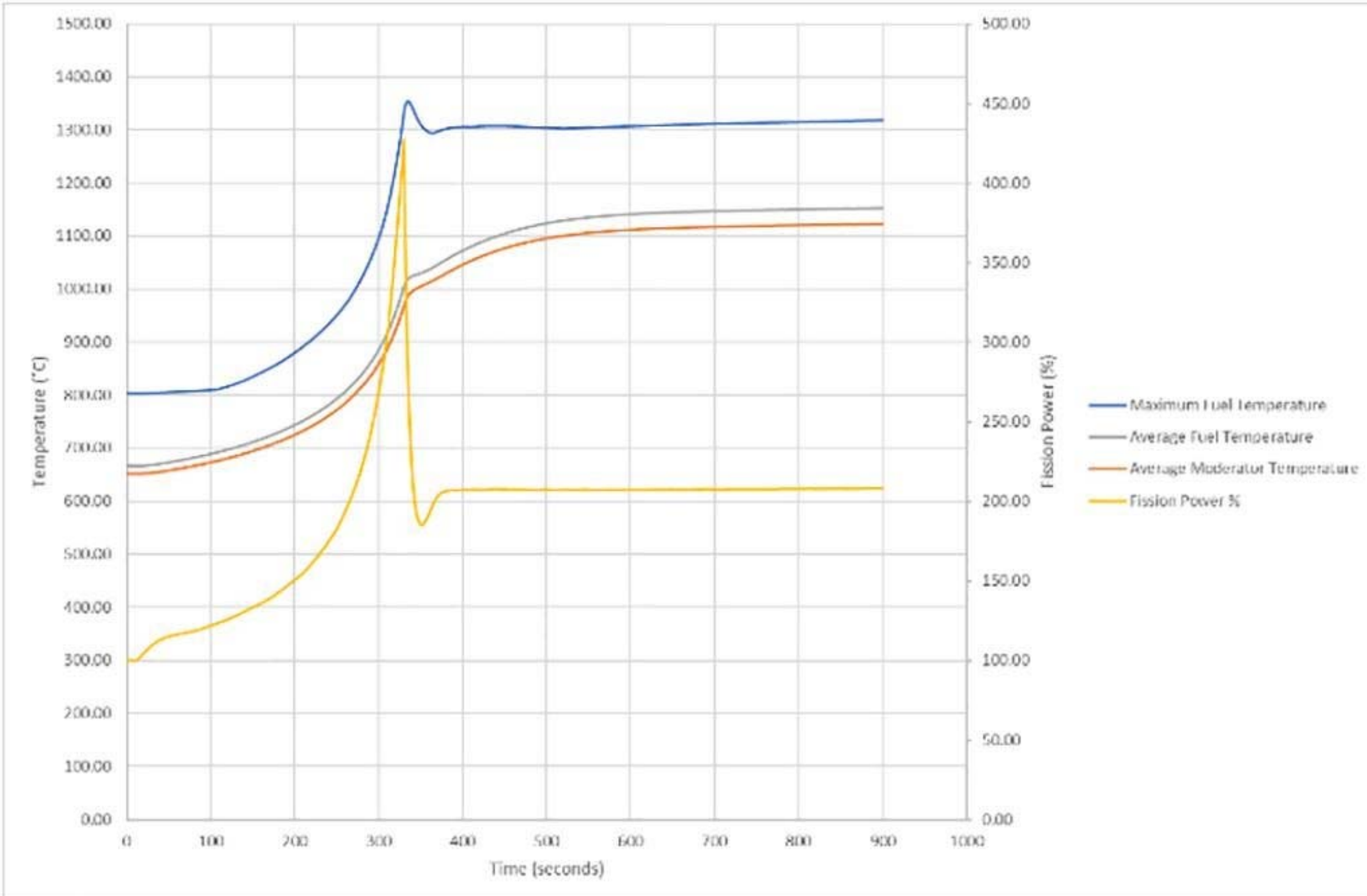

Fig. 23. Spike in power and temperature after a Control Rod Withdrawal (CRW) of all the control rods of the RCS.

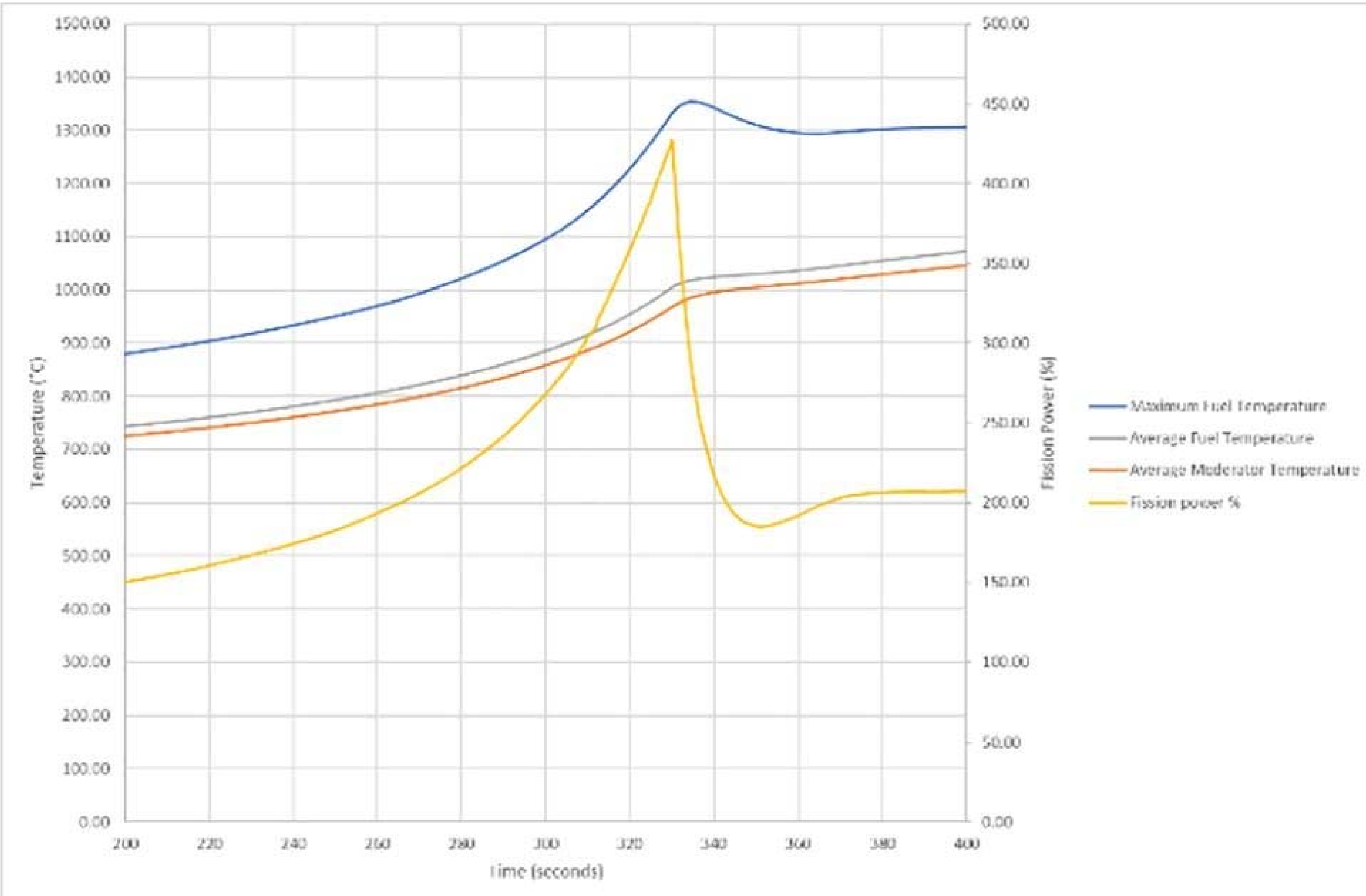

Fig. 24. Spike in power and temperature after a Control Rod Withdrawal (CRW) of all the control rods of the RCS. (enlarged version).

figures show that within 30 s the power would spike to 427 % and the maximum fuel temperature would be around 1354 °C, which is within limits of the well-known 1600 °C fuel damage limit.

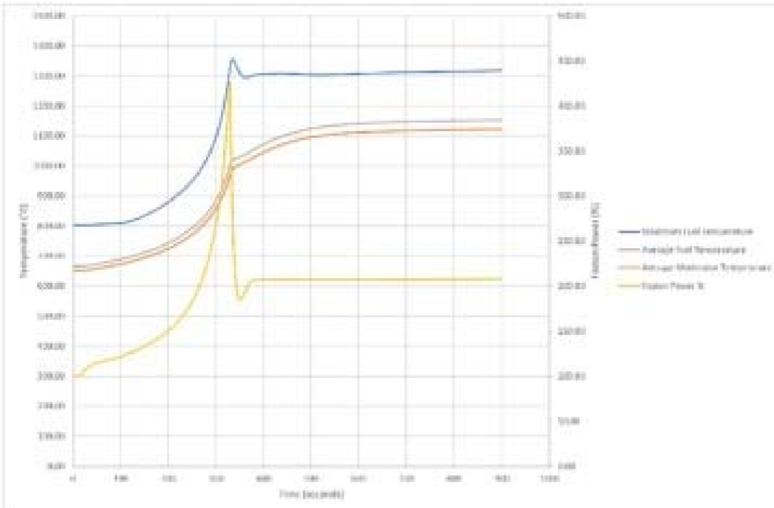

Fig. 23. Spike in power and temperature after a Control Rod Withdrawal (CRW) of all the control rods of the RCS.

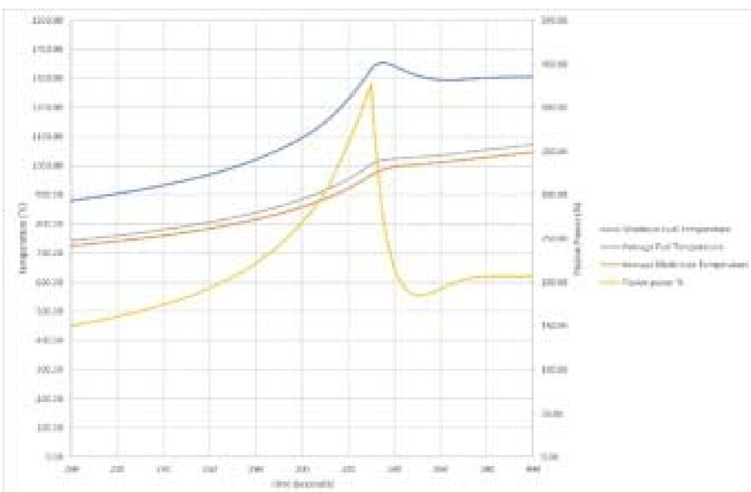

Fig. 24. Spike in power and temperature after a Control Rod Withdrawal (CRW) of all the control rods of the RCS. (enlarged version).

This represents the maximum centreline fuel temperatures seen by the centreline fuel in the middle of the reactor.

4.4.4. Control rod Ejection

The CRE transient which is an extremely fast unrealistic removal of all of the control rods shown in Fig. 25 and Fig. 26 occurs within 10 s which is physically not possible or is a beyond design base accident. But these figures show that within 30 s with a removal/ejection

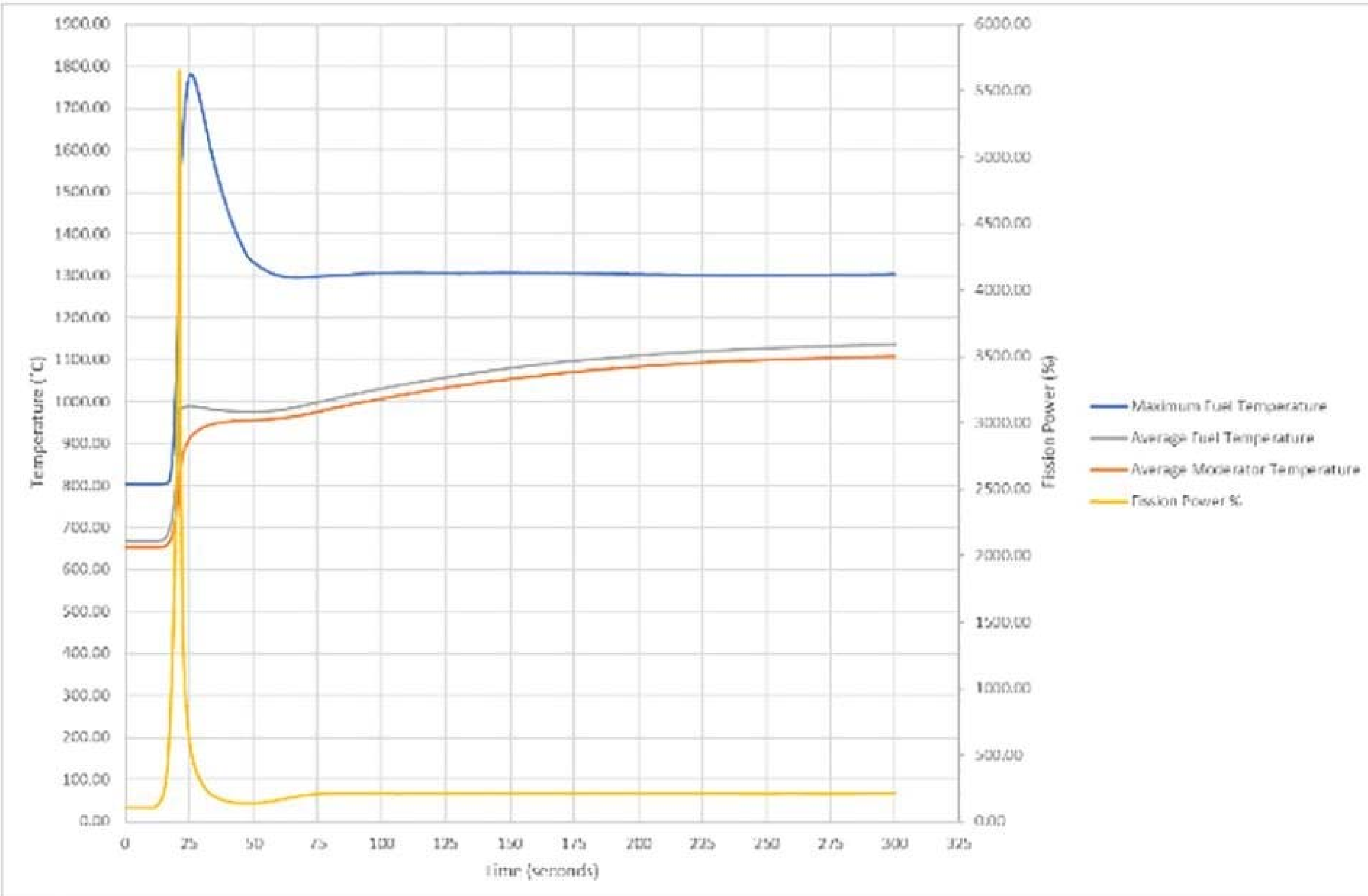

Fig. 25. Spike in power and fuel temperature following a Control Rod Ejection (CRE) of all the control rods of the RCS.

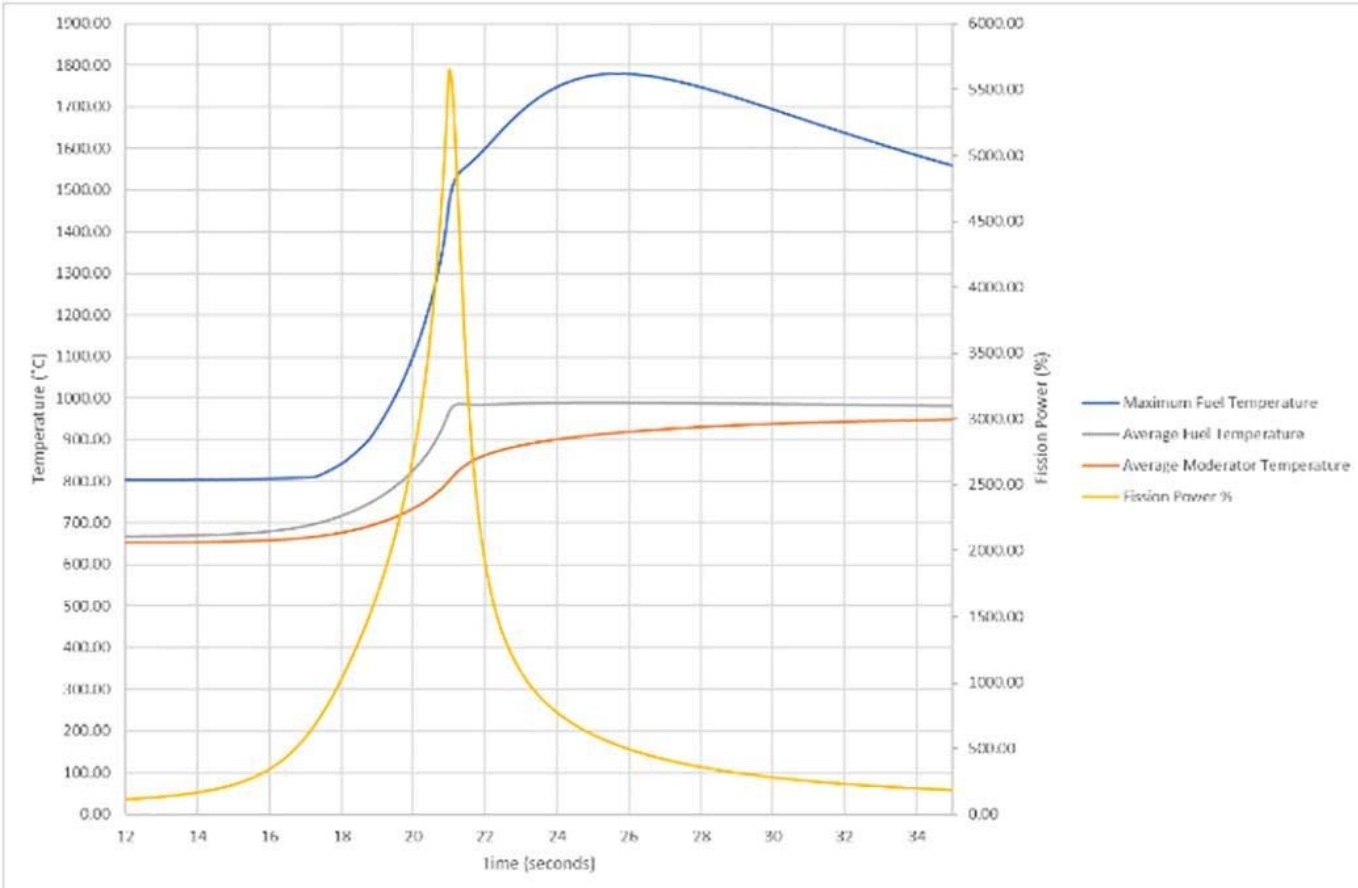

Fig. 26. Spike in power and fuel temperature following a Control Rod Ejection (CRE) of all the control rods of the RCS. (enlarged version).

of all the CRs the power would spike to 5647 % in power and the maximum fuel temperature would rise to 1780°C for a period of less than 30 s.

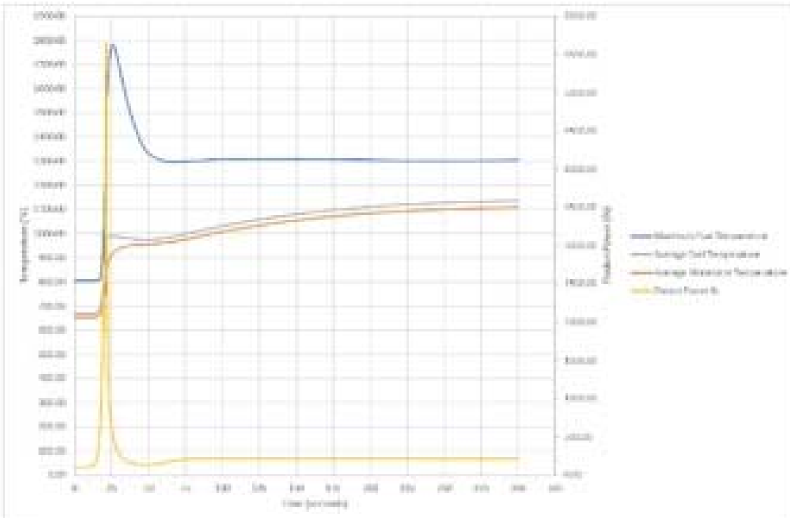

Fig. 25. Spike in power and fuel temperature following a Control Rod Ejection (CRE) of all the control rods of the RCS.

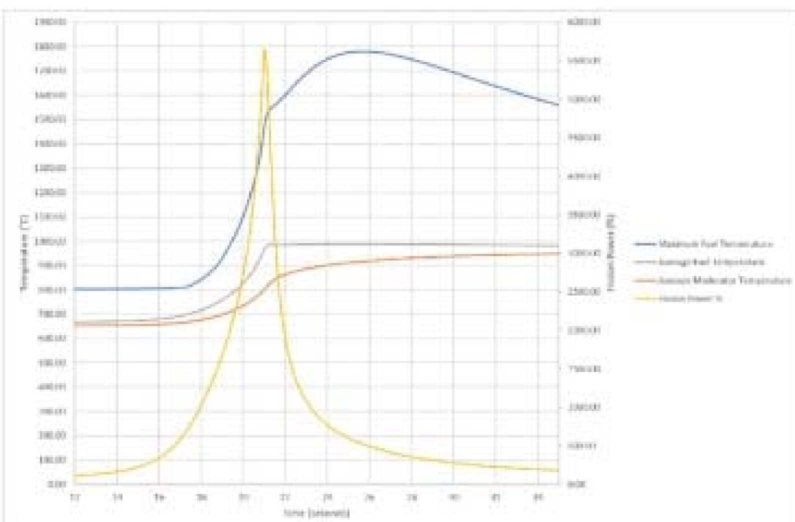

Fig. 26. Spike in power and fuel temperature following a Control Rod Ejection (CRE) of all the control rods of the RCS. (enlarged version).

This represents the maximum centreline fuel temperatures in the middle of the reactor, this is 1 % to 2 % of the total volume. This would not lead to significant fuel damage.

4.4.5. DLOFC without SCRAM

In Fig. 27 a DLOFC without SCRAM is shown for the local-heat case. A normal DLOFC with Scram (local heat) results in a maximum fuel temperature of around 1563.30 °C. For the DLOFC without SCRAM (local heat) the maximum fuel temperature is around 1598 °C. This is about 34.7 °C hotter due to the Control Rods (CRs) not being inserted. They were initially

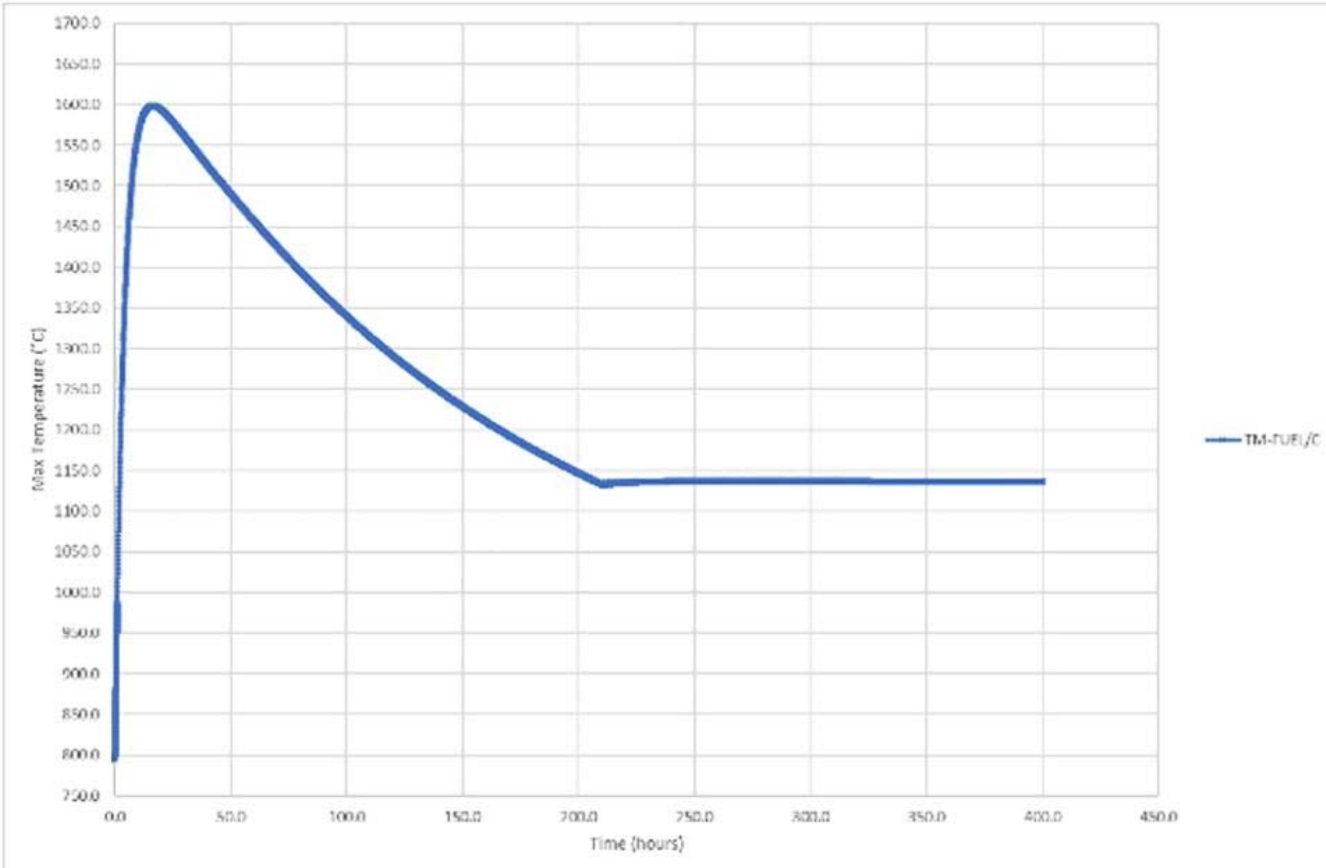

Fig. 27. Maximum fuel temperatures following a DLOFC without SCRAM (local heat) – no cooling.

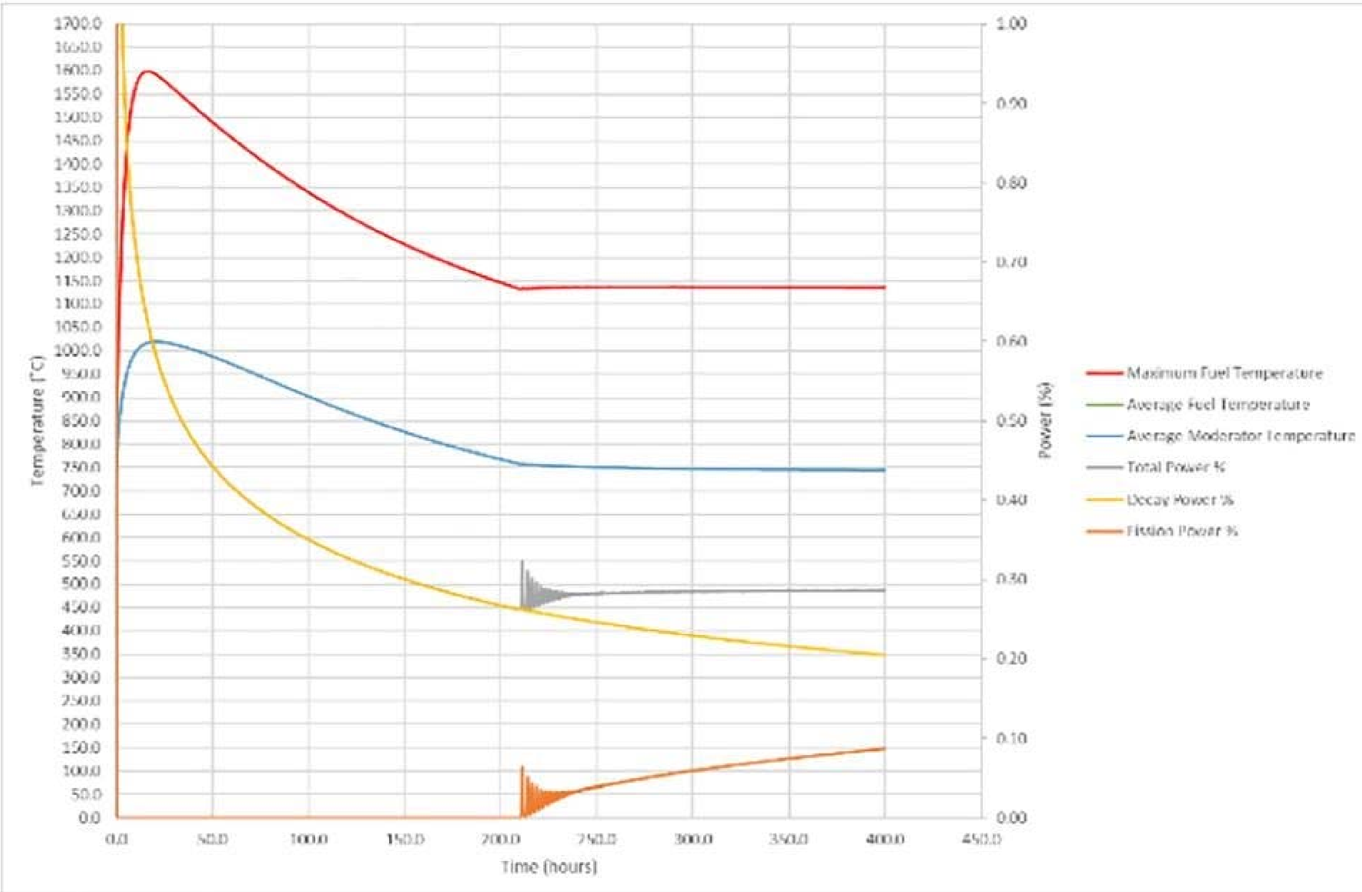

Fig. 28. Power vs fuel temperature for a DLOFC without SCRAM (local heat) – no helium for cooling.

inserted relatively deep and that is why this simulation with no SCRAM had little effect. The control rods need to only move up slightly out of the high flux zone resulting in a lower spike of reactivity and a lower rise in the maximum fuel temperature.

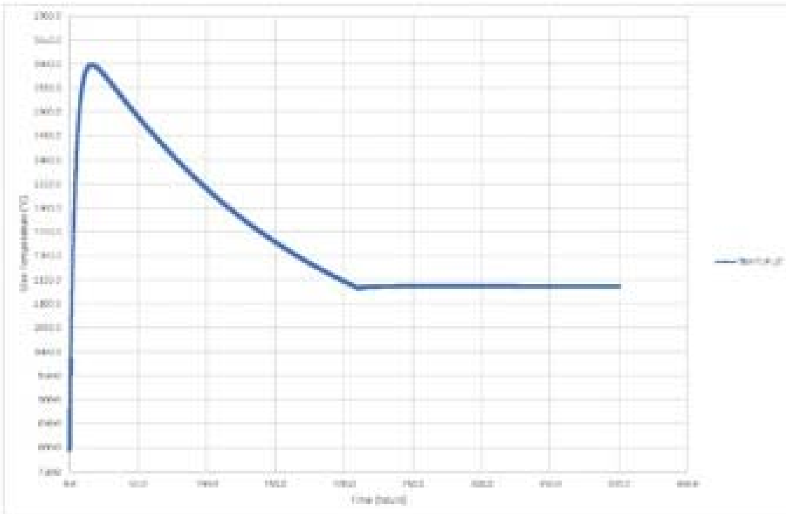

Fig. 27. Maximum fuel temperatures following a DLOFC without SCRAM (local heat) – no cooling.

In Fig. 28 it can be seen that if all coolant is lost with a DLOFC without SCRAM the reactor will regain criticality at approximately 210 h after the start of the event. This assumes that the control rods remain in their original withdrawn positions.

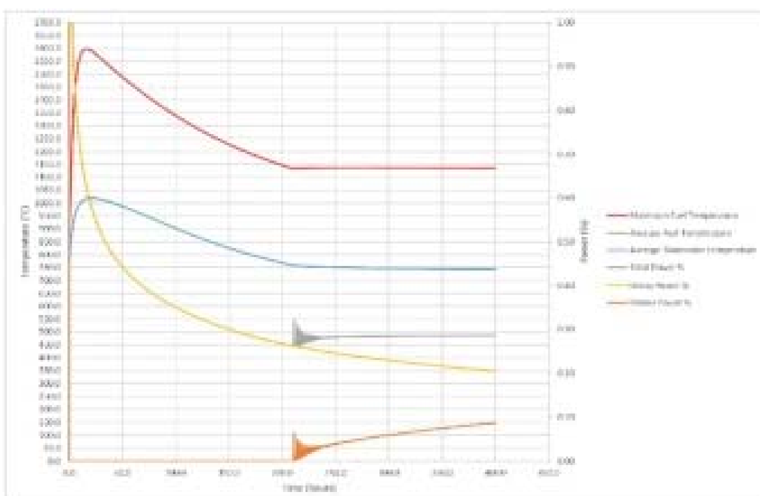

Fig. 28. Power vs fuel temperature for a DLOFC without SCRAM (local heat) – no helium for cooling.

Fig. 29 is an enlarged view of Fig. 28 and shows that the maximum fuel temperature will remain at approximately 1287 °C when the reactor regains criticality. The average fuel temperature would remain at approximately 860 °C. This reactor regains criticality at approximately 210-hour after the start of the event due to cool down of the core.

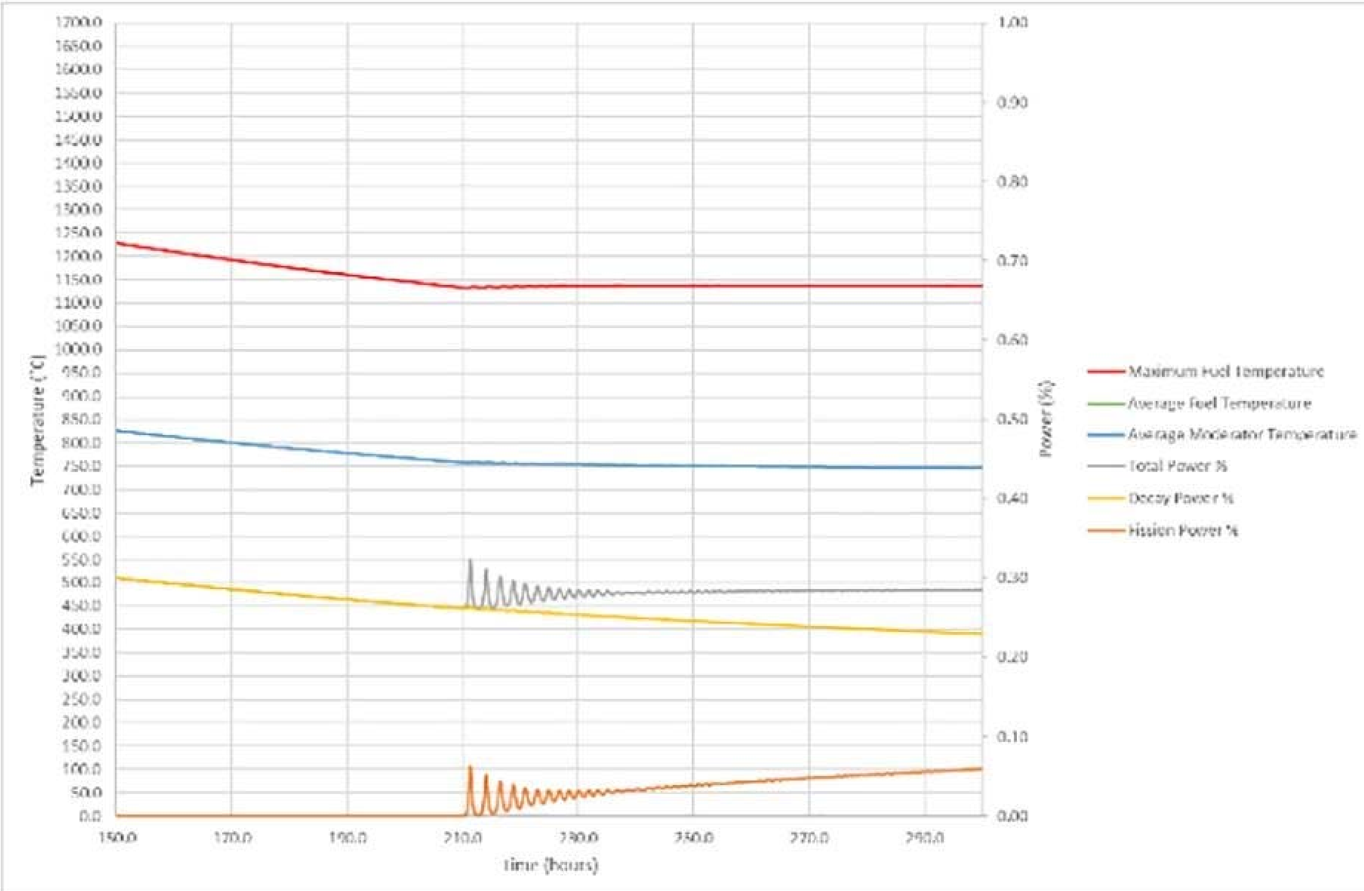

Fig. 29. Power vs fuel temperature for a DLOFC without SCRAM (local heat) – no helium cooling, enlarged Image.

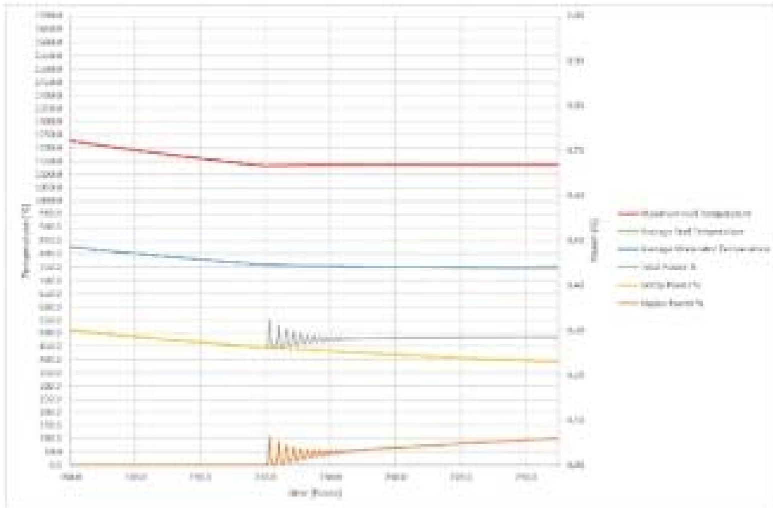

Fig. 29. Power vs fuel temperature for a DLOFC without SCRAM (local heat) – no helium cooling, enlarged Image.

4.4.6. VSOP & MGT comparison

It can be seen in Table 8 that the maximum, average and maximum to average temperatures for the fuel and moderator in both VSOP and MGT are similar. This shows that the two codes are in relatively good agreement with one another.

Table 8. VSOP & MGT comparison (normal operations).

Parameter	**Unit**	**VSOP 99**	**MGT**	**Error %**
Multiplication factor (k_{eff})	—	1.0000	0.9882	1.1756
Maximum power density	MW/m^3	7.95	7.35	7.54
Average power density	MW/m^3	3.613	3.080	14.757
Maximum fuel temperature	°C	822	800	2.676
Average fuel temperature	°C	655.5	669.592	2.149
Average fuel surface temperature	°C	639	639.047	0.007
Average moderator temperature	°C	661	769.023	16.342
Helium inlet temperature	°C	250	250	0
Helium outlet temperature	°C	752	748.7	0.438

5. Conclusions

The neutronic and thermal–hydraulic design of the HTMR100 reactor was assessed. The study proves that the two codes the VSOP99 suite of codes to obtain equilibrium results and the second code MGT transient dynamic accident analysis code yield similar results for HTMR100 with regards to fuel centreline temperatures, outer fuel sphere temperatures as well as moderator temperatures. The reactivity values for the two codes are also in good agreement with a 1.17 % difference between them. The DLOFC temperatures for the two codes use a different approach to calculate the maximum fuel temperatures but provide similar results. The codes are thus functioning well, are in agreement and the reactor model can be claimed to be accurate.

The HTMR100 reactor utilizing uranium dioxide UO_2 fuel at 10 wt% enrichment (U-235) content, 10 g of heavy metal per fuel element does indeed produce the targeted 80 000 MW_D/T_{HM} average burnup as specified for the OTTO fuel cycle.

The transient safety analysis performed proves that the reactor is indeed safe and that for the design base events the fuel temperatures remain below the accepted maximum temperature of 1600°C for the oxide-based UO_2 fuel. This is true as all the design-base events do indeed show that the centreline fuel temperatures remain below the set value of 1600°C. In addition, the temperature-volume analysis of the design-base events show that only a fraction (~2%) of the fuel reaches temperatures between 1500 °C − 1600 °C in a DLOFC event.

The beyond design base events which will rarely ever occur do exceed the set value of 1600°C for the fuel, but only for short periods of time while the transient takes place. This will not lead to fuel damage as this occurs for 30 s or less.

The HTMR100 reactor can be claimed to be passively safe. Water ingress analyses are still to be conducted for the HTMR100.

CRediT authorship contribution statement

W.A Boyes: Conceptualization, Methodology, Data curation, Formal analysis, Investigation, Visualization, Writing - original draft. **J.F.M Slabber:** Supervision, Methodology, Validation, Writing - review & editing. **D.W Boyes:** Conceptualization, Software, Funding acquisition, Project administration, Resources, Supervision, Validation, Writing - review & editing.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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Data availability

Data will be made available on request.

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