THE PBMR STEADY STATE AND COUPLED KINETICS CORE THERMAL-HYDRAULICS BENCHMARK TEST PROBLEMS

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ABSTRACT: In support of the Pebble Bed Modular Reactor (PBMR) Verification and Validation (V&V) effort, a set of benchmark test problems has been defined that focus on coupled core neutronics and thermal-hydraulic code-to-code comparisons. The motivation is not only to test the existing methods or codes available for High-temperature Gas-cooled Reactors (HTGRs), but also to serve as a basis for the development of more accurate and efficient tools to analyse the neutronics and thermal-hydraulic behaviour for design and safety evaluations in future.

The reference design for the PBMR268 benchmark problem is derived from the 268 MW PBMR design with a dynamic central column containing only graphite spheres. Several simplifications were made to the design in order to limit the need for any further approximations when defining code models. During this process, care was taken to ensure that all the important characteristics of the reactor design were preserved. The definition and initial phases of the benchmark were performed under a cooperative research project between NRG, Penn State University (PSU) and PBMR (Pty) Ltd. However, participation has recently been extended to include Purdue University and INEEL, and will be opened as soon as the benchmark definition has reached maturity. The benchmark has also been formally proposed as an OECD benchmark problem and approval is expected this year.

In this paper, the benchmark definition and the different test cases are described in some detail. Phase 1 focuses on steady-state conditions with the purpose of quantifying differences between code systems, models and basic data. It also serves as the basis to establish a common starting condition for the transient cases. In Phase 2, the focus is on performing coupled kinetics/core thermal-hydraulics test problems with a common cross-section and material property sets. The six events selected are described, and examples of some results are included to illustrate the behaviour of the transients.

KEYWORDS: PBMR, Test Problems, Benchmark, Kinetics, Verification & Validation, VSOP, TINTE, PANTHERMIX, NEM, PARCS, MICROX, THERMIX

0. INTRODUCTION

The reference design for the PBMR268 benchmark problem is derived from the 268 MW PBMR design with a dynamic central column containing only graphite spheres. The PBMR design applies a continuous reloading scheme where unloaded fuel spheres, which have not reached the target burn-up, are returned to the top of the core. This multi-pass fuel circulation is often referred to as a MEDUL cycle. For a so-called 10-pass equilibrium core used in this definition, this implies that the fuel spheres will, on average, circulate 10 times through the core before being discarded as spent fuel.

In Section 0 the rationale for the creation of the PBMR benchmark is explained, with attention given to the assumptions made. In Section 0 the geometrical description is included, with core dimensions and material region identified that are defined in Section 0. Although the data defining all cases (such as the detailed material number densities and cross-sections sets) cannot be included in this paper, the description is adequate to enable some cases to be calculated, and to gain insight into the specification.

The benchmark calculations are performed in two phases. Phase 1 focuses on steady-state conditions with the purpose of quantifying differences between the different code systems, models and basic data. The secondary aim is to establish a common starting condition for the transient cases defined as Phase 2. In Section 0 the steady-state cases are defined, while the transient definitions are given in Section -. The transient cross section library philosophy and a description of the tools used to generated the data is provided in Section -. Selected preliminary results are included in Section 0 with the main aim to illustrate the differences in the behaviour of the transient cases. Conclusions and future work is included in Section 0.

1. BENCHMARK PHILOSOPHY AND ASSUMPTIONS

Some benchmark problem definitions and experimental facilities do of course exist for High-temperature Reactors (HTRs), including a few existing for pebble bed reactors. This includes, amongst others, the Proteus pebble bed critical experiments [1] and ASTRA facility [2], [3] as examples of critical assemblies, the HTR-10 reactor in operation at Institute of Nuclear Energy Technology, Tsinghua University, Beijing, China [4], reactors that operated in Germany in the past such as the AVR [5], and several code-to-code comparisons performed as part of the IAEA CRP-5 (Co-ordinated Research Project (CRP) on "Evaluation of HTGR Performance") and similar programmes. Some transient experiments were also performed at the AVR and simulated by TINTE [6]. All of these contributed to the benchmarking and V&V of coupled core neutronics tools used in pebble bed reactor designs. The question thus remains where the current benchmark definition and code-to-code comparison effort can make a contribution.

The focus of the current benchmark test cases is on establishing well-defined transient benchmark cases. Although some work on the benchmarking of core simulation methods (steady-state) has been performed on PBMR designs [7], [8], the focus of the current work is on developing coupled kinetics/core thermal-hydraulics test problems that include both fast (reactivity insertions) and slow (thermal heat up due to decay heat) transients. In order to achieve this, several simplifications were made to the design specification so that the need for any further approximations is limited. It also meant that number densities and macroscopic cross-sections are given for reference equilibrium core cases. Furthermore, a multi-dimensional cross-section library and interpolation routines are to be supplied to participants as the basis for all transient studies. Although this requires source code changes, it circumvents differences due to different sources of cross-section data and preparation. This is perhaps the feature that distinguishes this effort the most from others. This approach is not new and has being applied to Power Water Reactor (PWR) and Boiling Water Reactor (BWR) transient benchmark definitions [9], [10].

During the simplification process, care has been taken to ensure that all the important characteristics of the reactor design are preserved. The simplifications make the core design essentially two-dimensional (r,z). Flow channels within the pebble bed have been simplified to be parallel and at equal speed while the dynamic central column and mixing zone widths were defined to be constant over the total axial height. This implies flattening of the pebble bed's upper surface and the removal of the bottom cone and de-fuel channel that results in a flat bottom reflector. Control rods in the side reflector are modelled as a cylindrical skirt (also referred to as a grey curtain) with a given B10 concentration. A multi-pass fuel circulation (MEDUL with 10 passes) and vertical pebble flow are assumed.

Thermal-hydraulic simplifications include the specification of stagnant helium between the barrel and Reactor Pressure Vessel (RPV) and stagnant air between the RPV and heat sink (outer boundary). The coolant flow is restricted to upwards flow from the inlet below the core within a porous ring in the reflector and downwards flow through the pebble bed to the outlet plenum. No reflector cooling or leakage paths were defined.

TABLE 1. Core geometrical specifications.

Description	Unit	Value
Equivalent core outer radius	m	1.75
Cylindrical height of the core (Flattened core surface at the top and flat bottom reflector).	m	8.5
Total core volume	m^3	81.779
Dynamic central column radius filled with graphite only spheres	m	0.786
Mixing zone outer radius with 50:50 mixture of fuel and graphite	m	1.109
Fraction of graphite spheres in the core	%	~30
Effective height of the upper void cavity	m	0.25
(levelled core surface to bottom of top reflector).		
Effective annular thickness of the inner reflector block (graphite).	m	0.75
Equivalent annular thickness of the outer reflector block (carbon).	m	0.25
Inner radius of the core barrel	m	2.87
The wall thickness of the core barrel	m	0.05
The inner radius of the RPV	m	3.0
The wall thickness of the RPV	m	0.17
Radius of cooling system/20 ˚C temperature isothermal boundary	m	4.17
Radii of five material meshes in core (5 radial meshes in core)	m	0.786 / 1.109
		1.357 / 1.566
		1.750
Axial material mesh: 8.5 m/17 meshes	m	0.5
Outlet plenum inner/outer diameter/height	m	0 / 1.75 / 0.5
Inlet plenum inner/outer diameter/height	m	2.15 / 2.35 / 0.5
He up-flow skirt/porous region inner/outer radius	m	2.15 / 2.35
Distance from bottom of core to top of the inlet plenum	m	1.0
Centre line axial distance between inlet and outlet plenum	m	1.0
Top inlet plenum inner/outer diameter/height	m	0.0 / 2.35 / 0.5
Distance from bottom of top reflector to bottom of top inlet plenum	m	1.0
Porous region (vertical flow) in top reflector inner/outer diameter	m	0.0 / 1.75
(region 2)		
Total height of top reflector (including top plenum)	m	1.5
Carbon block height at top of top reflector (above top inlet plenum)	m	0.25
Total height of bottom reflector (distance from top of bottom plate to bottom of core)	m	2.75
Top/bottom steel plate thickness	m	0.3

	0	78.6	110.9	135.7	156.6	175	181	194	215	235	250	275	287	292	300	317	417	418
-231		78.6	32.3	24.8	20.9	18.4	6	13	21	20	15	25	12	5	8	17	100	1
-230	1	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	18
-200	30	13	13	13	13	13	13	13	13	13	13	13	13	12	15	16	17	18
-175	25	7	7	7	7	7	7	7	7	7	7	7	11	12	15	16	17	18
-125	50	9	9	9	9	9	9	9	9	9	3	7	11	12	15	16	17	18
-75	50	2	2	2	2	2	3	3	3	6	3	7	11	12	15	16	17	18
-25	50	2	2	2	2	2	3	3	3	6	3	7	11	12	15	16	17	18
0	25	1	1	1	1	1	3	5	3	6	3	7	11	12	15	16	17	18
50	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
100	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
150	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
200	50	a	b	c	d	e	3	5	3	6	3	7	11	12	15	16	17	18
250	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
300	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
350	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
400	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
450	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
500	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
550	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
600	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
650	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
700	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
750	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
800	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
850	50	a	b	c	d	e	3	3	3	6	3	7	11	12	15	16	17	18
900	50	4	4	4	4	4	3	3	3	6	3	7	11	12	15	16	17	18
950	50	4	4	4	4	4	3	3	3	6	3	7	11	12	15	16	17	18
1000	50	4	4	4	4	4	3	3	3	8	3	7	11	12	15	16	17	18
1050	50	4	4	4	4	4	3	3	3	3	3	7	11	12	15	16	17	18
1100	50	10	10	10	10	10	3	3	3	3	3	7	11	12	15	16	17	18
1125	25	7	7	7	7	7	7	7	7	7	7	7	11	12	15	16	17	18
1155	30	14	14	14	14	14	14	14	14	14	14	14	14	12	15	16	17	18
1156	1	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19	18

Neutronics

a	Dynamic central column (graphite sheres only)
b	Mixture region with 50:50 fuel : graphite spheres
c	3rd flow channel with only fuel spheres
d	4th flow channel with only fuel spheres
e	Outer flow channel with only fuel spheres
1	Void area above the pebble bed
2	Top reflector graphite
3	Side reflector graphite
4	Bottom reflector graphite
5	Control rod grey skirt / graphite
6	Side reflector graphite (with helium channels)
7	Carbon side reflector / insulation

Thermal Hydraulics

a	Pebble bed
b	Pebble bed
c	Pebble bed
d	Pebble bed
e	Pebble bed
1	void area above core with vertical helium flow
2	Top reflector: downwards helium flow
3	Side reflector (no helium flow)
4	Bottom reflector: downwards helium flow
5	Control rod region / no flow
6	Side reflector: with upward flow in helium channels
7	Carbon top / side / bottom reflector / insulation
8	Bottom inlet plenum
9	Top Inlet plenum
10	Bottom Outlet plenum
11	Stagnant helium between side reflector and barrel
12	Barrel
13	Top plate
14	Bottom plate
15	Stagnant helium between barrel and RPV
16	RPV
17	Air outside RPV
18	Heat sink
19	Adiabatic boundary condition
20	Adiabatic boundary condition

FIGURE 1.Geometry description of the PBMR268 test case.

Other simplifications include the assumption that all heat sources (from fission) will be deposited locally, i.e. in the fuel, and that no other heat sources exist outside the core (for example neutron absorption in the control rod region). Simplifications are also made in the material thermal properties, in as far as constant values are employed or specific correlations are employed.

2. GEOMETRICAL DESCRIPTION

The reactor geometrical layout and dimensions are summarized in *TABLE 1* and *FIGURE 1*.

3. MATERIALS, PROPERTIES AND CROSS-SECTIONS

Pebble fuel spheres with low enriched (8%) uranium-oxide triso-coated particles and with a loading of 9 g per fuel sphere are used in the benchmark definition. A pebble packing fraction of 0.61 is assumed. Only one void region, being the 25 cm between the top of the pebble-bed and the bottom of the top reflector, is present. In the diffusion calculation, directional dependent diffusion coefficients are used to represent the neutron streaming effects. A factor is multiplied to the diffusion coefficient for the r and z direction, 0.1 for r and 0.5 for the z direction. The radius of the region is taken as the diffusion coefficient. The fuel and material definitions are given in

TABLE 2.

TABLE 2. Fuel and graphite sphere specifications.

Description	Unit	Value
Fuel pebble		
Fuel pebble outer radius	cm	3.0
Thickness of fuel free zone	cm	0.5
Density of graphite in fuel sphere	$g.cm^{-3}$	1.75
Total heavy metal loading per fuel pebble (equilibrium fuel)	g	9
Enrichment for equilibrium fuel (weight percentage)	$^{w}/_{o}$	8.0
U-235 content for equilibrium fuel	g/fs	0.72
Coated Particle		
Fuel kernel diameter	μm	500
Particle material type		UO_2
UO_2 density	$g.cm^{-3}$	$\bar{x}$ = 10.4
Coating Material		C / C/ SiC / C
Layer thickness: Buffer/PyC/SiC/Outer PyC	μm	95 / 40 / 35 / 40
Buffer layer densities	$g.cm^{-3}$	1.05 / 1.9 / 3.18 / 1.90
Graphite Spheres		
Sphere outer radius	cm	3.0
Density of graphite	$g.cm^{-3}$	1.75
Other		
The reflector graphite density for all graphite structures	$g.cm^{-3}$	1.75
The carbon bricks density for the outer/back reflector block	$g.cm^{-3}$	1.6
Density of RPV and Core Barrel: Iron	$g.cm^{-3}$	7.8

The equivalent B-10 concentration for the control rods, modelled as a gray curtain, is given in *TABLE 3*. The control rods are inserted 200 cm into the core and are also assumed to extend 25 cm above the core (corresponding to the void area above the pebble bed)

The thermal properties of the materials are included in *TABLE 4*. The properties were simplified to remove temperature and fast fluence dependence to allow straightforward implementation and comparison. It is, however, important to note that in particular the thermal conductivity of the pebble-bed and reflector graphite is dependent on irradiation damage and temperature (history and instantaneous), and that these effects must be included if core design calculations are to be performed.

TABLE 3. Control rod specifications.

Description	**Unit**	**Value**
Thickness of gray-curtain region representing control	m	0.13
Distance between core outer diameter and inner diameter of control rod gray-curtain region	m	0.06
Homogenised number density of B-10 representing control system	# $barn^{-1}$ cm^{-1}	3.2E-6
Density of graphite in the gray-curtain region	$g.cm^{-3}$	1.75

TABLE 4. Thermal hydraulic parameters.

Description	**Unit**	**Value**
Hydraulic diameters		
Risers channels: Side porous region	cm	19.0
Top porous region	cm	5.0
Bottom porous region	cm	4.5
Top void area	cm	350
Inlet (top and bottom) and outlet plenums	cm	50
Thermal Conductivity		
Pebble bed (effective)	W m^{-1} K^{-1}	Zehner-Schlünder
Simplified with constant thermal conductivity, heat capacity and with zero fast fluence dose		
Reflector, graphite spheres and fuel graphite. (actual value is a function of temperature, fast fluence and irradiation temperature)	W m^{-1} K^{-1}	20.0 (assume constant)
Carbon bricks	W m^{-1} K^{-1}	7.0
Reactor Pressure Vessel (SA 508) (constant)	W m^{-1} K^{-1}	38.0
Core Barrel (Type 316 SS) (constant)	W m^{-1} K^{-1}	17.0
Fraction of contact area between spheres (vs total contact area)	%	1.6
Specific Heat Capacity		
Fuel, graphite spheres, reflector graphite and Carbon (or 3.02 $W.s.cm^{-3}.K^{-1}$ for a density of 1.75 $g.cm^{-3}$)	J. kg^{-1} K^{-1}	1725
Reactor Pressure Vessel (or 4.095 $W.s.cm^{-3}.K^{-1}$ for a density of 7.8 $g.cm^{-3}$)	J. kg^{-1} K^{-1}	525
Core Barrel (or 4.212 $W.s.cm^{-3}.K^{-1}$ for a density of 7.8 $g.cm^{-3}$)	J. kg^{-1} K^{-1}	540
Specific heat capacity of helium (constant)	J. kg^{-1} K^{-1}	5 195

Description	**Unit**	**Value**
The emissivity of all materials in definition		0.8

The regions where helium coolant flows occur are defined as porous regions (containing graphite and helium), but no reduction should be made in the graphite density during the neutronic calculation. In contrast, adjustments were made in the 'density' in the thermal hydraulic specification to take the coolant flow channels into account. Void fractions of 0.2 (packing fraction = 0.8) were assumed for all the porous regions where helium flow is defined. In *TABLE 5*, the coolant flow characteristics and flow path are defined. No bypass flow is defined, and therefore the total coolant mass will flow through the pebble bed.

TABLE 5. Main flow characteristics.

Description	Unit	Value
He inlet/outlet temperature	ºC	500/900
Total inlet mass flow rate	kg/s	129
Inlet pressure	kPa	7 000
Coolant flow – flow is into inlet plenum (← 8), up into the He flow skirt (↑ 6), into top plenum (← 9), through porous top reflector and void (↓ 2 ↓ 1); through core (↓ a-e); through porous bottom reflector (↓ 4); into bottom/outlet plenum (→10).		

Three sets of homogenized atomic or number densities are provided representing fresh fuel, equilibrium fuel (with only the uranium isotopic distribution) and the equilibrium core isotopic distribution given for the 17 x 5 material regions in the core (as a homogenized mixture), including all the higher elements. The benchmark definition also includes sets of macroscopic cross-sections given at specific temperatures to be used in the steady-state cases, while a multi-dimensional macroscopic cross-section table, tabulated as a function of fuel temperature, moderator temperature, geometrical buckling and xenon concentration are provided for the transient cases. A FORTRAN module that performs interpolation in this table at the correct (input) conditions, is also included. The detailed tables with number densities and cross-sections can of course not be included in this paper but can be made available.

4. STEADY-STATE TEST CASES

Several cases were defined at different temperatures (300 K – 1 500 K), with either fresh fuel or at equilibrium cycle conditions (the applicable homogenized number densities are specified and cross-sections have to be generated by the participants). Other cases make use of given homogenized macroscopic cross-sections for both fresh and equilibrium fuel, and at various temperatures. The combination of test cases was devised to test effects due to different cross-section data, and also to test the differences between methods (for example, finite-difference and nodal), when macroscopic cross-sections are given. A case to test the thermal hydraulics, where the heat sources are defined, was also included.

In all cases, results such as power, flux and temperature profiles are to be presented on a given mesh for easy comparisons. A two-group structure with a thermal cut off at 2.1 eV was selected.

4. 1. Case N-1: Fresh Fuel and Cold Conditions

The steady-state solution is to be found for the model description and the following conditions:

- All fuel is fresh (9 g HM and 8 w/o enriched)
- Cold conditions (300 K) for all materials

- Cross-sections to be generated by participants

4. 2. Case N-2: Equilibrium Cycle with Given Number Densities

The steady-state solution is to be found for the model description and the following conditions:

- The given equilibrium uranium, graphite and structural number densities are used (no fission products or higher elements)
- Constant temperature conditions (600 K and 900 K) for all materials
- Cross-sections to be generated by participants

4. 3. Case N-3: Fresh Fuel with Given Cross-sections

The steady-state solution is to be found for the model description and the following conditions:

- Constant temperature conditions at 300 K and 900 K.
- Make use of single set of tabulated set of cross-sections (at the appropriate temperature) - Cross-sections for 300 K to 1 500 K in 300 K steps are provided as part of the detailed benchmark definition for all materials.

4. 4. Case T-1: Equilibrium Cycle with given Power/Heat Sources

Make use of the thermal hydraulic properties and model description and the following conditions:

- The provided flat power/heat source to be used - a simplified axially averaged power profile was constructed from the relative power distribution in the five core regions. The values are:
  - Central column/Mixing zone/Fuel 1/Fuel 2/Fuel 3/= 0.0/0.79/1.43/1.39/1.39
- Calculate the temperatures, pressure drops and flow rates
- Assume fresh fuel Zehner-Schlünder pebble bed effective thermal conductivities (resultant values to be provided to be used as input if required)

4. 5. Case C-1: Fresh Core with Combined Neutronics- Thermal- hydraulics case

Make use of the neutronics and thermal hydraulic properties and model description and the following conditions:

- The fresh core is used
- Perform neutronic calculation to obtain power profiles/heat sources
- Calculate the temperatures, pressure drops and flow rates
- Repeat neutronics – thermal hydraulic feedback
- Make use of own cross-sections

4. 6. Case D-1: Equilibrium Cycle with Combined Neutronics Thermal Hydraulics Calculation

Make use of neutronics model description and the following conditions:

- The equilibrium uranium, higher isotopes, graphite and structural number densities are used (no fission products or higher elements)
- Make use of temperature-dependent tabulated set of cross-sections
- Calculate the temperatures, pressure drops and flow rates
- Cross-section interpolation routines and provided tabulated cross-section data should be implemented in the codes and used

5. TRANSIENT TEST CASES

All transient cases will start from the steady-state case based on common cross-section sets (Case D-1). For the steady state solution the spatial maps of the maximum and average fuel temperature, maximum and average moderator temperature, power density, relative pressure, mass flow and thermal conductivity are the requested output, while single parameter results include the axial offset and eigenvalue (k-eff). Results must be given for certain time histories, while the more detailed maps, as defined above for the steady-state solution, are also to be given at specific times (identified in the detailed benchmark definition per case) and specific points of interest during the transients.

Six case studies, covering the range from slow thermal-hydraulic to fast neutronic transients, are defined. The slow transients rely largely on the thermal properties of the reactor materials, and will therefore test these phenomena separate from the kinetics and the reactor reactivity feedback mechanisms tested in the postulated fast reactivity events. A typical load follow case, which largely follows the xenon behaviour, was also included. The cases are given in the following paragraphs, together with a short description of the event, and required quantities to be provided as results. Preliminary results as an illustration of the behaviour of each transient, as calculated with TINTE [6], are included in Section 0.

5. 1. Case 1: Depressurized Loss of Forced Cooling (DLOFC) without SCRAM

A depressurization is assumed to occur linearly over a period of 13 s, with a resultant loss of forced cooling over the same period. The effects of natural convection should be included in this case, and since no SCRAM is assumed, re-criticality will occur.

The transient should be followed for 100 h, and at given times, the maximum and average fuel temperature maps, maximum and average moderator temperature maps, power and heat density distribution and axial offset, relative pressure and mass flows should be edited for comparisons. In addition to the standard maps given at prescribed time points, the results should be compared at the following two events:

- The time point when the maximum fuel temperature is reached
- The time point when the reactor attains re-criticality.

5. 2. Case 2: Depressurized Loss of Forced Cooling (DLOFC) with SCRAM

A depressurization is assumed to be the same as for Case 1 to occur linearly over a period of 13 s. At this time all control rods are inserted linearly over 3 s to SCRAM the reactor. The effects of natural convection should be included as before, but in this case, since a SCRAM did take place, the reactor is expected to remain shut down, and therefore no re-criticality will occur. Results required are as for Case 1, but the detailed maps should also be given after the SCRAM, i.e. at 16 s.

5. 3. Case 3: Pressurized Loss of Forced Cooling (PLOFC) with SCRAM

A reduction in reactor helium pressure from nominal (70 bar) to the system equalization pressure of 40 bar is specified to occur linearly over 13 s. Similarly, a linear reduction in the reactor coolant mass flow from nominal (129 kg/s) to 0.1 kg/s is specified over the same period. This is followed by a total linear insertion of the control rods over 3 s (the SCRAM). Results required are as for Case 2.

5. 4. Case 4: Load Follow 100%-40%-100%

The insertion depth of the control rods in normal operation allows xenon override during power changes. In this case, the xenon concentration and reactivity changes due to load follow are investigated. A reduction in the power from the 100% steady-state operation to 40% for a period of 6 h is defined. After 6 h, the power returns to 100%. Note that this case does not involve moving the control rods, and therefore the core will not be calculated to be critical during the transient. The reactivity should therefore be given as part of the results.

In more detail, the power change is achieved by a linear reactor power level reduction from nominal 268 MW (100%) to 107.2 MW (40%) over 8 s, and a simultaneous reduction in the reactor coolant mass flow from nominal (129 kg/s) to 51.6 kg/s (40% of nominal) over 8 s. The mass flow ramp is also assumed linear. After 6 h, the opposite actions are applied to increase the power and flow rate to 100% again, also in 8 s.

The transient history (from t=0 to t=54 h) must be given for the following parameters:

- maximum and average fuel temperatures
- maximum and average moderator temperatures
- fission power
- reactivity
- average xenon concentration in the entire spatial mesh
- the xenon concentration in two meshes: one top mesh with coordinates (r=156 cm, z=150 cm), and one bottom mesh with coordinates (r=156 cm, z=750 cm).

The values of these parameters must be indicated at the time points as requested.

5. 5. Case 5: Fast Reactivity Insertion

An event had to be postulated to simulate fast reactivity insertion by simulating a Control Rod Ejection (CRE) scenario. Although this scenario is not realistic, it was important to define a scenario where a large amount of reactivity is added over a very short time, so that the code capabilities to simulate fast reactivity excursions could be evaluated. The withdrawal of all 24 control rods over a 0.1 s duration was therefore assumed. The control rods are withdrawn from the normal operation insertion depth of 200 cm below the pebble bed.

The transient should be followed for 60 s. Spatial maps of the maximum and average fuel temperature, maximum and average moderator temperature and power density should be edited at t = 0.05, 0.1, 0.15, 0.2, 0.5, 1.0 s.

5. 6. Case 6: Overcooled Helium Inlet

This case simulates a bypass valve opening, with ‘cold’ helium being injected into the core inlet plenum. A temperature ramp of 50 ºC (i.e. 10% of nominal inlet temperature) is applied over 10 s, without changing any other reactor parameters such as mass flow, pressure or control rod positions. It is assumed that a reactor protection system would cause the valve to close again after 300 s, and the temperature would return to nominal value, again over 10 s.

The transient history is to be followed for an hour, and the following parameters edited for comparison:

- Maximum and average fuel temperatures
- Maximum and average moderator temperatures
- Axial offset
- Fission power
-

6. CROSS-SECTION LIBRARY GENERATION

For the steady state test cases, the number density and cross-section sets were generated with VSOP [12] and PANTHERMIX [13] (using WIMS). The selection of four state parameters to represent the cross-sections in multi-dimensional tables made the use of VSOP and PANTHERMIX less attractive, since the cross-section generation process is integrated with the core and fuel management system. The availability and experience on the MICROX-2 [14] code at PSU made it a natural choice to generate the required cross-section sets. It also adds the possibility to compare the generated cross-sections with the VSOP and PANTHERMIX data, although this falls outside the scope of this benchmark problem.

The cross-section library is dependent on fuel temperature, moderator temperature, the xenon concentration, and geometrical buckling representing the environment spectrum and leakage feedback effects. The exposure is not currently included, since a snapshot at equilibrium cycle is used as the starting point for the transients, and burn-up can be ignored for the duration of the transients. The approach is similar to that followed in TINTE, although some additional dependence is available in TINTE.

MICROX-2 is an integral transport theory code. It solves the B1 neutron balance equations in a one-dimensional two-region unit cell. The two regions are coupled by collision probabilities based on spatially flat neutron emission. Dancoff factors and bucklings correct the one-dimensional cell calculations for multi-dimensional lattice effects. MICROX-2 prepares the broad group neutron cross-sections for use in diffusion and transport codes.

MICROX-2 has three geometry options: spherical, cylindrical and planar (slab). In spherical geometry (the geometry used for the pebble cell), the two regions of the pebble are distinguished as follows: Region 1 consists of the inner fuel region. This is the 5 cm diameter region that contains the fuel kernels in the graphite matrix. The graphite shell and the coolant regions are homogenized into one region to make Region 2. Refer to *FIGURE 2*.

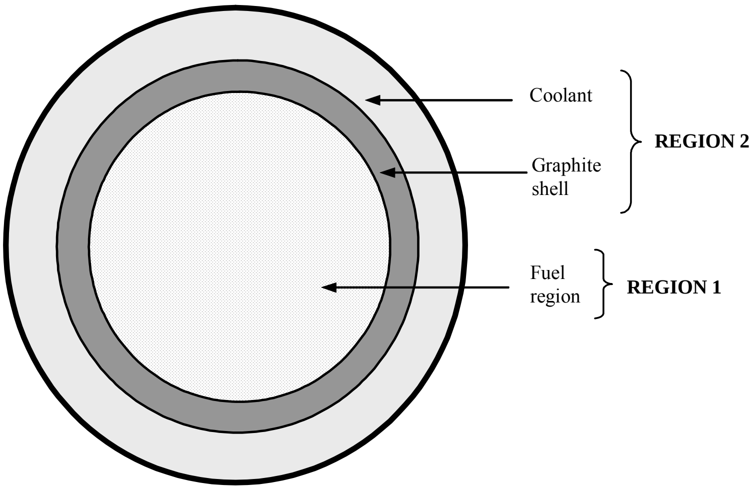


FIGURE 2. MICROX-2 Spherical Geometry Model

The earlier version of the code was developed by General Atomics. Further development work and testing was carried out at the Paul Scherrer Institute in Switzerland.

7. SELECTED RESULTS

Results of some of the cases in Phase I have been reported [11]. The execution of the benchmark during the definition phase was important to fine-tune the benchmark definition by identifying shortcomings, drawing up results reporting structures and investigating the method to prepare and represent the cross-sections.

Some additional results are shown in this paper. Results for the steady-state Case D-1 are shown. This represents the starting point for all the transients. Note that the results for the transient case are for illustration only, and do not represent reference results, since the calculations performed with TINTE make use of a TINTE cross-section library that was created with isotopic data from VSOP. So far, results have not been generated using the interpolation routines and reference cross-section tables that are in the process of being created.

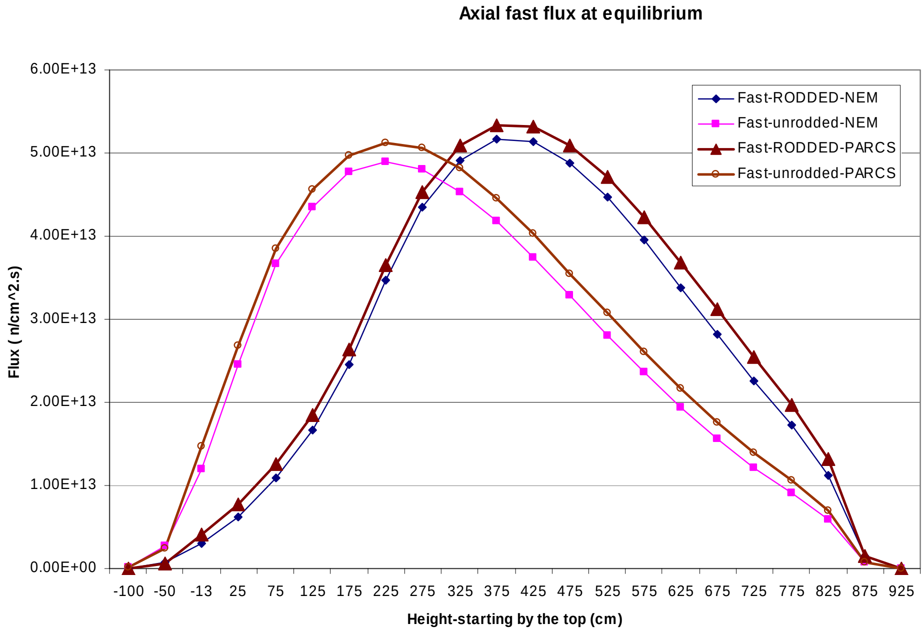
7. 1. Case D-1 - Equilibrium Cycle with Combined Neutronics Thermal Hydraulics Calculation

Case D-1 was analyzed with the neutronics codes PARCS [15] and NEM [16] coupled with the two different versions of the thermal hydraulics code THERMIX. These results are preliminary since both code systems are under continued development. The thermal and fast axial flux and power profiles predicted by PARCS and NEM for Case D-1 are shown in FIGURE 3 and FIGURE 4, respectively, and a comparison of the predicted eigenvalues and temperatures are shown in TABLE 6.

As indicated in the Tables and Figures, there are some noticeable differences in the predictions of the two codes. In particular, the difference in the eigenvalue for both the rodded and unrodded cases is about 1000 pcm. Efforts are ongoing to isolate the reasons for the differences.

TABLE 6. Comparison of PARCS and NEM Results for Case D-1.

Description	PARCS	NEM
k-eff (unrodded)	1.04930	1.05874
k-eff (rodded)	1.03289	1.04038
Maximum solid temperature (ºC)	1042	1075
Position of maximum solid temperature (r/z) (cm)	156.6/850	130.0/850
Maximum gas temperature (ºC)	1036	1064
Position of maximum gas temperature (r/z) (cm)	156/900	130/862



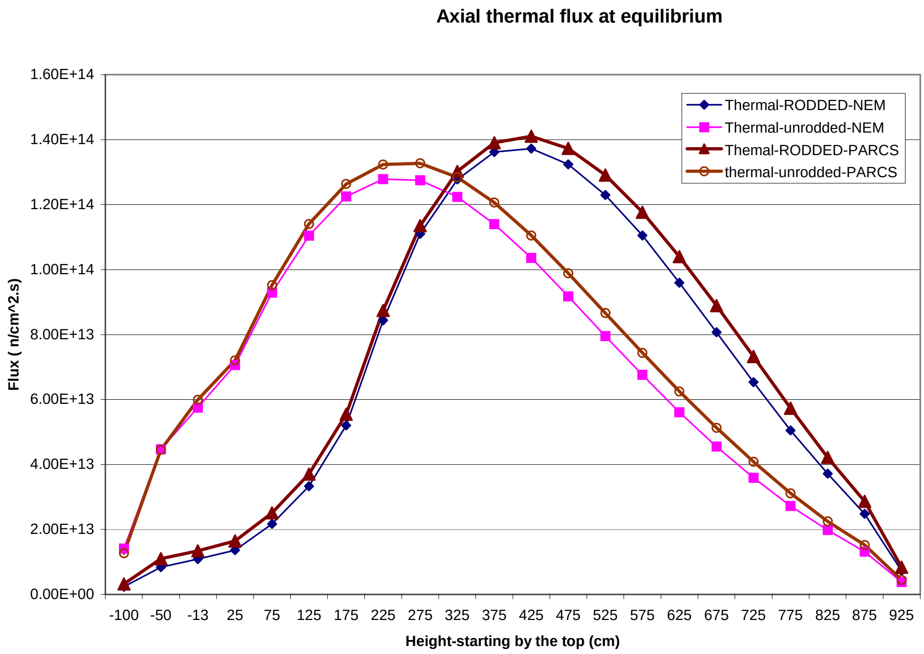
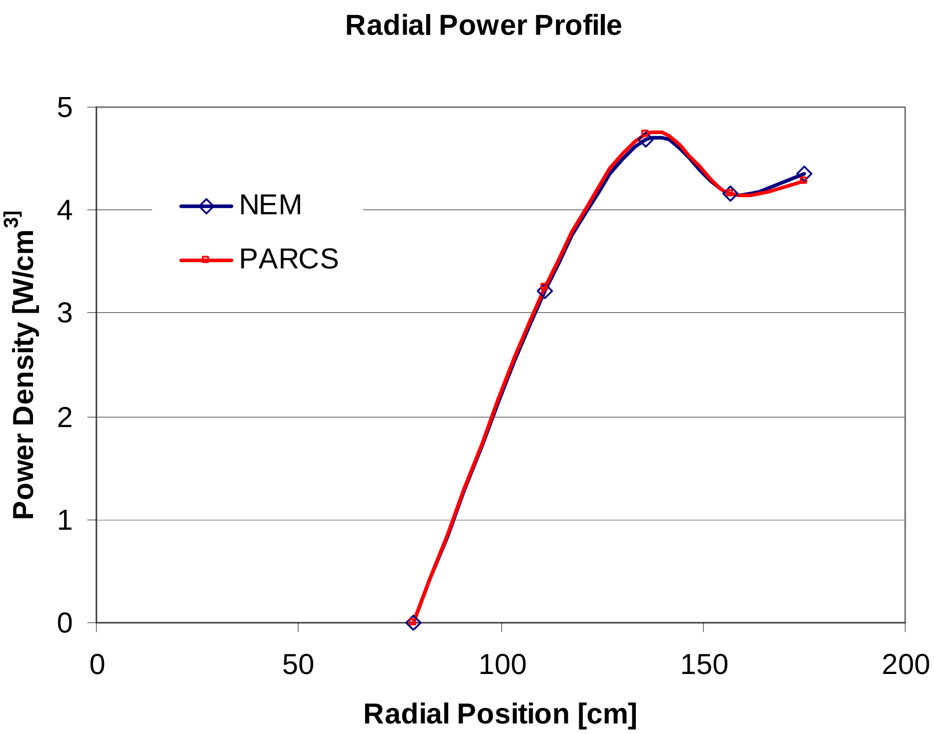


FIGURE 3. Thermal and Fast Axial Flux Profiles for Case D-1



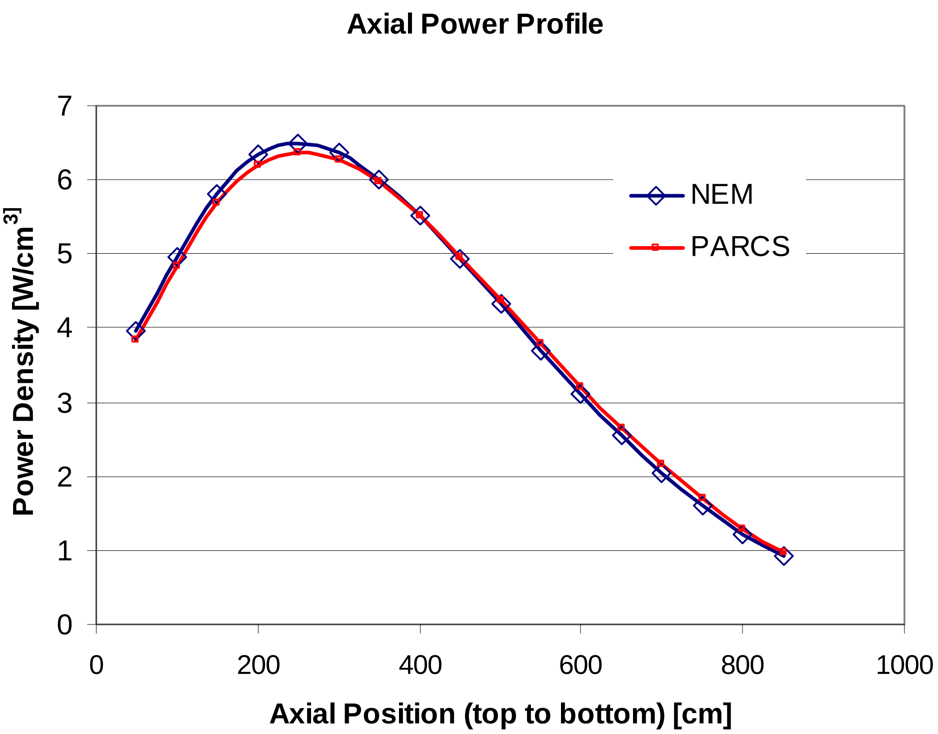


FIGURE 4. Comparison of Radial and Axial Power Profiles for Case D-1

7. 2. Case 1 – Case 3: Loss of Flow and Depressurization Cases

The first three transient cases have very similar time behaviour, with the transient developing over several days. Due to the loss of active cooling, the only heat removal mechanism is the passive thermal conduction and radiation through the reactor vessel to the defined heat sink. The reactor structures and fuel will therefore heat up initially, as the decay heat is more than the heat removed.

Since the decay heat decreases all the time, a maximum fuel temperature is reached, followed by the gradual cool-down of the reactor. In *FIGURE 5*, the maximum fuel temperature is shown for Cases 1 to 3. The lack of a SCRAM in Case 1 leads to re-criticality after 70 h – some detail is shown in *FIGURE 6* – while the presence of helium at equalization pressure in Case 3 results in a considerably lower fuel temperature peak.

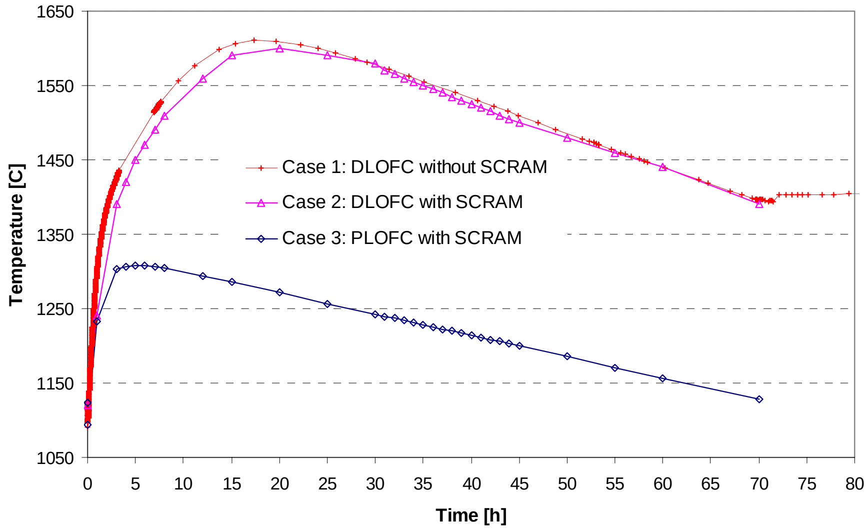


FIGURE 5. Maximum fuel temperature behaviour for transient cases 1 to 3.

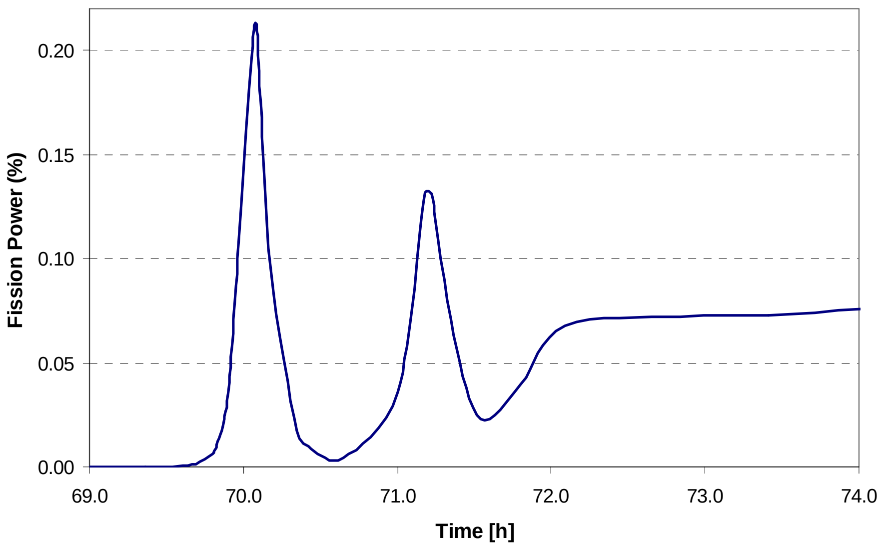


FIGURE 6. Case 1: Fission power behaviour at re-criticality.

7. 3. Case 4 Load Follow

The load follow ability of the PBMR reactor is simulated in this case, and since the inlet and outlet temperatures are largely unchanged during the power changes, the main effect on reactivity is the xenon build-up, decay and burn-up during the transient. In *FIGURE 7*, the typical behaviour of the external reactivity required to keep the reactor critical is shown. This will be managed by moving the control rods, but for the case definition, no control rod movements were defined.

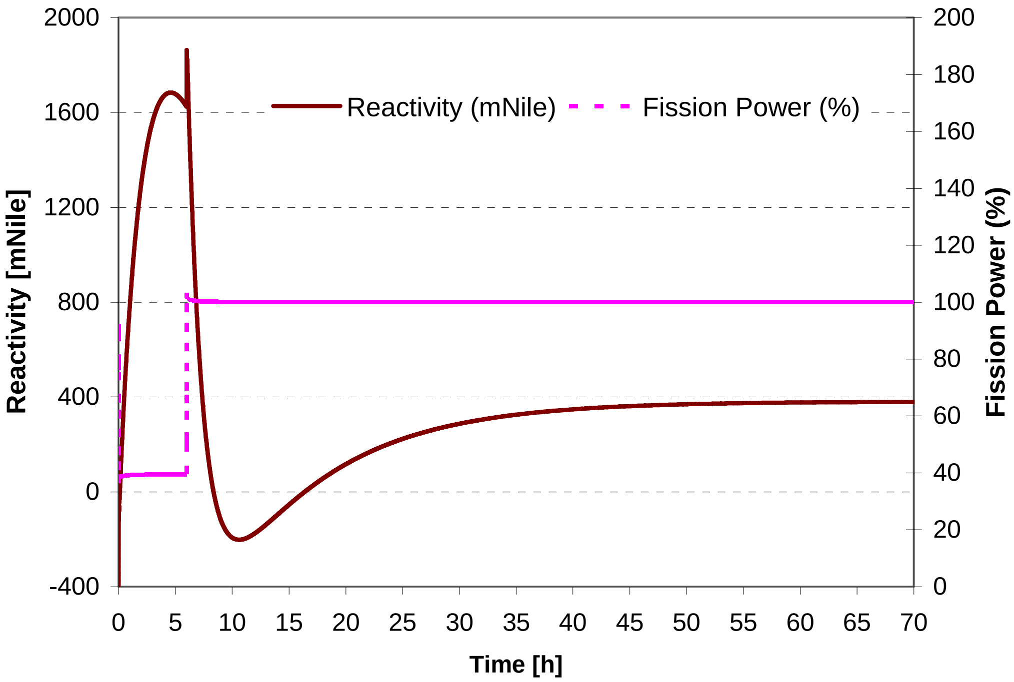


FIGURE 7. Case 4: Reactivity and fission power behaviour during 100%-40%-100% load follow.

7. 4. Case 5 Control Rods Ejection

The sample results considered so far were all for long time periods where the reactivity temperature coefficients and other kinetics parameters do not play such a large role. Transient Case 5 was specifically postulated to have an unrealistically large reactivity insertion over a very short time. The correct implementation of kinetics equations and parameters can thus be verified by this problem, while the detailed thermal hydraulic modelling of the fuel kernel and pebble will also be important.

FIGURE 8 also illustrates the typical behaviour of the transient case. A large increase in the fission power and associated rapid increase in the maximum fuel temperature is shown over the few seconds of the transient. The large negative fuel temperature reactivity coefficient limits the power excursion to a few seconds, and after 20 s the power has returned to 103%.

7. 5. Case 6: Cold Coolant Reactivity Excursion

The expected effect of cold helium entering the PBMR test case is illustrated in *FIGURE 9*. Contrary to water reactors, the colder helium has no real direct effect on the neutronics, but indirectly through the cooling of the moderator within the fuel pebbles, and then the fuel kernels. The event leads to only a moderate increase in the total power before the coolant temperature is returned to normal. During the power increase to less than 104%, an increase of less than 50 ºC in the maximum fuel temperature was seen. The transient, although much slower than the large reactivity insertion, returns to normal conditions in a matter of hours.

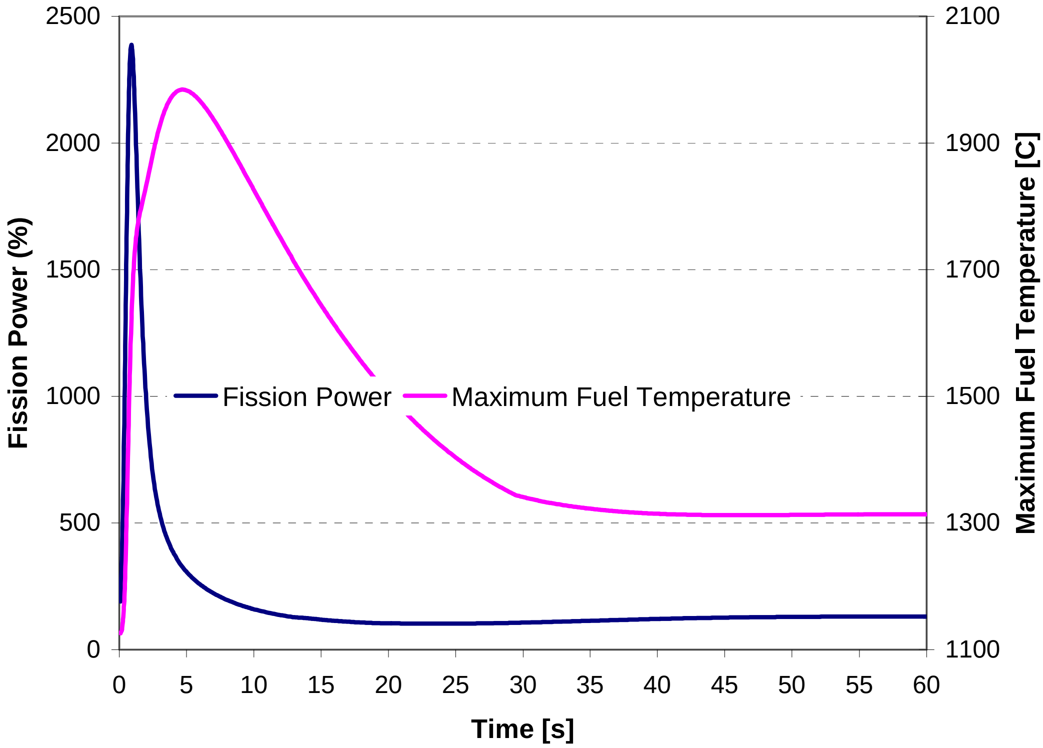


FIGURE 8. Case 5: Maximum fuel temperatures and fission power behaviour.

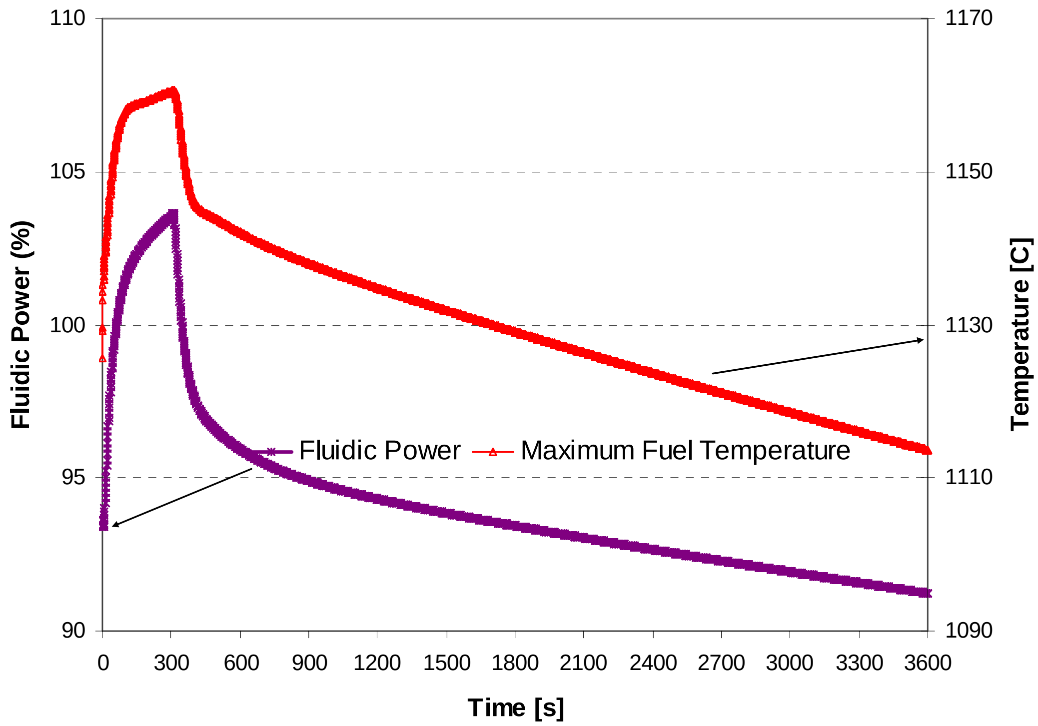


FIGURE 9. Case 6: Power removed by coolant and maximum fuel temperature behaviour following the flow of over-cooled helium through the core.

8. CONCLUSIONS

The PBMR steady state and coupled kinetics core thermal-hydraulics benchmark test problems have been described in detail in this paper. The problems are well defined as many of the problems have been performed by the initiating participants. The test cases include several steady-state cases while representative transient cases with both short and long time evolutions were included. The final tasks are to create cross-section sets to be represented in multi-dimensional tables and then to prepare the final benchmark definition documentation. The benchmark has also been formally proposed as an OECD benchmark problem and approval is expected this year. The benchmark calculation will therefore be open for wider participation.

The use of given cross-section and detailed definition of other parameters makes the benchmark effort more valuable since it can be used to perform detailed code to code comparisons since most uncertainties in data and definitions should be excluded. This will add value to the Verification and Validation efforts and status of the neutronic and thermal-hydraulic simulators. The establishment of a reference set of results remains unresolved and should be pursued further.

The benchmark test cases can also be the basis for testing more advanced calculational and cross-section preparation models. Some of these are cross section preparation and representation, approaches to modelling the multi-pass fuel management schemes, and the fuel pebble and UO_2 kernel thermal transport models. Furthermore, the benchmark effort has already generated cooperation on other aspects and the sharing of resources.

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