The OECD/NEA/NSC PBMR Coupled Neutronics/Thermal Hydraulics Transient Benchmark: The PBMR-400 Core Design

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Abstract

The Pebble Bed Modular Reactor (PBMR) is a High-Temperature Gas-cooled Reactor (HTGR) concept to be built in South Africa. As part of the verification and validation program the definition and execution of code-to-code benchmark exercises are important.

The Nuclear Energy Agency (NEA) of the Organisation for Economic Cooperation and Development (OECD) has accepted, through the Nuclear Science Committee (NSC), the inclusion of the Pebble-Bed Modular Reactor (PBMR) coupled neutronics/thermal hydraulics transient benchmark problem in its program.

The OECD benchmark defines steady-state and transients cases, including reactivity insertion transients. It makes use of a common set of cross sections (to eliminate uncertainties between different codes) and includes specific simplifications to the design to limit the need for participants to introduce approximations in their models.

In this paper the detailed specification is explained, including the test cases to be calculated and the results required from participants.

KEYWORDS: Pebble bed modular reactor, PBMR, Coupled Neutronics, Thermal Hydraulics, Transient, OECD Benchmark

1. Introduction

The Nuclear Energy Agency (NEA) of the Organisation for Economic Cooperation and Development (OECD) has accepted the Pebble-Bed Modular Reactor (PBMR) coupled neutronics/thermal hydraulics transient benchmark problem as part of their program.

The PBMR is a High-Temperature Gas-cooled Reactor (HTGR) concept, which has attracted the attention of the nuclear research and development community. The deterministic neutronics, thermal-hydraulics and transient analysis tools and methods available to design and analyse PBMRs have, in many cases, lagged behind the state of the art compared to other reactor

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technologies. This has motivated not only the testing of existing methods for HTGRs but also the development of more accurate and efficient tools to analyse the neutronics and thermal-hydraulic behaviour for the design and safety evaluations of the PBMR. In addition to the development of new methods, this includes defining appropriate benchmarks to verify and validate the new methods in computer codes.

The scope of the benchmark described in this paper is to establish a well-defined problem, based on a common given set of cross sections, to compare methods and tools in core simulation and to do thermal hydraulics analysis with a specific focus on transient events through a set of multi-dimensional computational test problems.

In addition, the benchmark activity has the following objectives:

1. Establish a standard benchmark for coupled codes (neutronics/thermal-hydraulics) for PBMR design
2. Code-to-code and methods comparison using a common cross-section library – this is very important for the Verification and Validation part of the PBMR licensing process
3. Obtain a detailed understanding of the phenomena that is important to model during the different transient events
4. Benefit from the use of different methods and also different approaches to the test exercises
5. Obtain an understanding of the limitations of the tools and the effects of approximations introduced
6. Organize special sessions at conferences or a special issue of a publication to give exposure to HTGR methods and designs
7. Serve as the vehicle for future benchmarks based on experimental facilities or eventually the PBMR demonstration unit.

Of course the OECD benchmark is not the first effort to verify the methods and codes used for the PBMR design or more generally for HTGRs. The tools and methods available today have been validated continuously against experiments and operating reactors throughout their development, although the formalized procedures required today were not in general use. Some benchmark problem definitions and experimental facilities do of course exist for High-temperature Reactors (HTRs), including a few for pebble bed reactors. Examples of these are the Proteus pebble bed critical experiments [1] and ASTRA facility [2,3] - which are critical assemblies, the HTR 10 reactor in operation at the Institute of Nuclear Energy Technology, Tsinghua University, Beijing, China [4], reactors that operated in Germany in the past such as the AVR (Arbeitsgemeinschaft Versuchsreaktor GmbH – a 15 MWe pebble bed experimental reactor built at the FZJ site, Jülich and operated from December 1967 till 1988), and several code-to-code comparisons performed as part of the IAEA CRP-5 (Co-ordinated Research Project (CRP) on "Evaluation of HTGR Performance") [5,6] and similar programmes. Some transient experiments were also performed at the AVR [7] and others include simulations with codes still in use today [8]. All of these contributed to the benchmarking and V&V of coupled neutronics/core thermal-hydraulics tools used in pebble bed reactor designs.

The current transient benchmark specification with all the different transient cases is the first activity based on the 400 MWth PBMR design with its annular core design and fixed graphite central column. In previous work, based on the older PBMR 268 MWth dynamic central column design [9], similar transient cases were analyzed, but this was restricted to only a few participants. The specific behaviour of the PBMR 400 MWth design required a new definition and the OECD acceptance has given it wider exposure. The lessons learned from these initial efforts are now applied to the OECD benchmark.

2. The Benchmark Definition

The reference design for the PBMR-400 benchmark problem is derived from the PBMR 400MWth design of the demo unit. A detailed description of the plant design, and specifically the reactor core neutronics design, has been published [6,10,11]. Several simplifications were made to the design in this specification in order to minimize the need for any further approximations in the simulation models. During this process care was taken to ensure that all the important characteristics of the reactor design were preserved. This ensures that the results from the benchmark will be representative of the actual design's characteristics.

2.1 Simplifications Introduced

The simplifications made for the benchmark problem make the core design essentially two-dimensional (r,z). It includes 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. Flow channels within the pebble bed have been simplified to be parallel and at equal speed. Control rods in the side reflector are modeled as a cylindrical skirt (also referred to as a grey curtain) with a given B_{10} concentration. Only one of the transient cases, the single control-rod ejection event, requires a three-dimensional model. In this case, an equivalent boron concentration is defined for a specific mesh or region where the control rods are situated.

Thermal-hydraulic simplifications include the specification of stagnant helium (no mass flow) between the side reflector and barrel and the barrel and Reactor Pressure Vessel (RPV). Stagnant air (no mass flow) is defined between the RPV and heat sink (outer boundary). The coolant flow is simplified to the main engineered flow paths, i.e. upwards flow from the inlet below the core within a porous ring in the side reflector, and downwards flow through the pebble bed to the outlet plenum. No reflector cooling or leakage paths were defined. In the fixed central reflector the 10 cm hole in the middle, the cooling dowels and cooling slits were also removed. Other engineered coolant flows excluded are the control rod cooling flow, the core barrel leakage flow and the cooling effect of the core barrel conditioning system (CBCS) that would keep the barrel temperature within a temperature range during operation.

The effect of excluding specific coolant flows is balanced to some extent by 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 rods). Simplifications are also made in the material thermal properties in as far as constant values or specific correlations are employed.

2.2 Geometrical Description

The benchmark reactor unit geometry description is given in Fig. 1 with the general material layout shown. The mesh dimensions (r, z) as well as the general material regions of the core are shown. The figure represents half of the reactor with a symmetry axis on the zero radius line in the centre of the reactor. The pebble bed core is the 85 cm thick annulus represented by "F". Just above the fuel is a "void" area (that is filled with helium). The graphite reflector has been divided into four regions called the Central Column (CC), the Top Reflector (TR), the

Figure 1: Core layout and identification

	0	10	41	73.6	80.55	92.05	100	117	134	151	168	185	192.95	204.45	211.4	225	243.6	260.6	275	287.5	292.5	310	328	462	463
-235		10	31	32.6	6.95	11.5	7.95	17	17	17	17	17	7.95	11.5	6.95	13.6	18.6	17	14.4	12.5	5	17.5	18	134	1
-200	35	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	TP	He	RPV	Air	RCCS
-150	50	CC	CC	CC	CC	RSS	CC	TR	TR	TR	TR	TR	SR	RCS	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
-100	50	CC	CC	CC	CC	RSS	CC	TR	TR	TR	TR	TR	SR	RCS	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
-50	50	CC	CC	CC	CC	RSS	CC	TR	TR	TR	TR	TR	SR	RCS	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
0	50	CC	CC	CC	CC	RSS	CC	V	V	V	V	V	SR	RCS	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
50	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	IP	RCS	IP	IP	IP	RC	SR	He	CB	He	RPV	Air	RCCS
100	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
150	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
200	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
250	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
300	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
350	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
400	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
450	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
500	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
550	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
600	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
650	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
700	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
750	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
800	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
850	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
900	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
950	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
1000	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
1050	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
1100	50	CC	CC	CC	CC	RSS	CC	F	F	F	F	F	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
1150	50	CC	CC	CC	CC	RSS	CC	BR	BR	BR	BR	BR	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
1200	50	CC	CC	CC	CC	RSS	CC	BR	BR	BR	BR	BR	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
1250	50	CC	CC	CC	CC	RSS	CC	BR	BR	BR	BR	BR	SR	RCS	SR	SR	SR	RC	SR	He	CB	He	RPV	Air	RCCS
1300	50	CC	CC	CC	CC	CC	CC	BR	BR	BR	BR	BR	SR	SR	SR	SR	SR	IP	SR	He	CB	He	RPV	Air	RCCS
1350	50	CC	CC	CC	CC	CC	CC	BR	BR	BR	BR	BR	SR	SR	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
1400	50	CC	CC	CC	CC	CC	CC	OP	OP	OP	OP	OP	SR	SR	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
1450	50	CC	CC	CC	CC	CC	CC	BR	BR	BR	BR	BR	SR	SR	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
1500	50	CC	CC	CC	CC	CC	CC	BR	BR	BR	BR	BR	SR	SR	SR	SR	SR	SR	SR	He	CB	He	RPV	Air	RCCS
1535	35	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	He	RPV	Air	RCCS

CORE LAYOUT DEFINITIONS

F	REACTOR CORE CONTAINING THE FUEL
V	HELIUM GAP BETWEEN FUEL AND TOP REFLECTOR: VOID
CC	CENTRAL REFLECTOR: GRAPHITE
TR	TOP REFLECTOR: GRAPHITE
BR	BOTTOM REFLECTOR: GRAPHITE
SR	SIDE REFLECTOR: GRAPHITE
RCS	REACTOR CONTROL SYSTEM CHANNEL : GRAPHITE / GREY CURTAIN AREA
RSS	RESERVE SHUTDOWN SYSTEM CHANNEL : GRAPHITE / GREY CURTAIN AREA
IP	INLET PLENUM TOP / BOTTOM : GRAPHITE
RC	RISER CHANNEL IN SIDE REFLECTOR : GRAPHITE
OP	OUTLET PLENUM BOTTOM : GRAPHITE
He	STAGNANT HELIUM
TP	TOP PLATE : IRON : ADIABATIC BOUNDARY
BP	BOTTOM PLATE : IRON : ADIABATIC BOUNDARY
CB	CORE BARREL : IRON
RPV	REACTOR PRESSURE VESSEL : IRON
Air	STAGNANT AIR
RCCS	REACTOR CAVITY COOLING SYSTEM : 20C TH BOUNDARY
	NEUTRONIC BOUNDARY CONDITIONS

Bottom Reflector (BR) and the Side Reflector (SR). The central column has a radius of 100 cm and the side reflector a thickness of 90 cm. Beyond the side reflector is a helium gap (He) between the graphite blocks of the SR and core barrel (CB). This is followed by another Helium gap (He) and the RPV. Outside the RPV an air gap of 134 cm is found before the Reactor Cavity Cooling System (RCCS), which is defined as the outer boundary of the benchmark model.

Above the top reflector of 150 cm the 35 cm thick Top Plate (TP) is the upper boundary. Similarly the Bottom Plate (BT) forms the lower boundary of the problem 400 cm below the bottom reflector. The effective core height is 1100 cm with a helium filled cavity at the top of 50 cm. The core effective volume that can be filled with fuel therefore is 83.7155 m^3.

The helium coolant path is bordered by the solid lines shown within Fig. 1. The side reflector contains the lower inlet plenum (IP) where the coolant gas enters and flows upward through the Riser Channels (RC) into the upper inlet plenum (IP). The gas then flows down through the core, through slits in the bottom reflector into the outlet plenum from where it enters into the power conversion unit (PCU). The side reflector also contains the Reactivity Control System channels (RCS) while the Central Column contains the Reserve Shutdown System channels (RSS). In the current benchmark definition the RSS is never filled with any neutron absorber material and is therefore treated the same as the rest of the Central Column. Only the upper few meshes of the RCS are typically filled by the control rods (see shaded area) while the rest of the axial meshes are then assumed to be the same as the side reflector.

2.3 Cross-section Library

An important characteristic of the benchmark is the use of a common set of cross sections. The data is represented as various sets of two-group macroscopic cross-section data representing different fuel regions (with different mixtures of burned pebbles), reflector and control rod regions. The cross sections are stored in multi-dimensional tables that include cross terms. The cross sections are tabulated as a function of (and all combinations of) five state-parameters, namely fuel temperature, moderator temperature, fast and thermal buckling (representing the leakage spectral effects) and xenon concentration. All cross terms dependencies are included implying linear interpolation on a 5-dimensional space.

All of the fuel material cross-section tables have data for the five state parameters with cross section data at four fuel temperatures, seven moderator temperatures, three fast bucklings, three thermal bucklings and three Xenon number densities. The non-fuel materials have no fuel temperature or xenon variations. The ranges chosen for each parameter were selected based on the reactor conditions for normal operation as well as for accident conditions. The current data ranges are given in Tab. 1.

Table 1: State parameters and ranges used in the benchmark cross-section library

Parameter	Range of values
Fuel temperature	300K, 800K, 1400K, 2400K
Moderator temperature	400K, 600K, 800K, 1100K, 1400K, 1800K, 2400K
Fuel regions:	
Fast buckling	-1.0×10^{-6}, 1.0×10^{-4}, 4.0×10^{-3}
Thermal buckling	-1.4×10^{-3}, -2.0×10^{-5}, 5.0×10^{-5}
Reflector regions:	
Fast buckling	-6.5×10^{-3}, -1.0×10^{-4}, 0.0
Thermal buckling	-1.1×10^{-3}, 5.0×10^{-5}, 1.0×10^{-4}
Xenon concentration	0.0, 3.78×10^{-13}, 9.44×10^{-8} [#/barn cm]

Tests to confirm the appropriateness of the state-parameter set and its range, and to ensure that the number of data points is adequate for linear interpolation, must still be finalized for all transient cases. One such example is that the fuel temperature should be interpolated as the square root of the temperatures.

The use of the fuel and moderator temperatures as state parameters to tabulate cross sections are well known for most reactor types while the use of xenon, or other parameters to quantify the spectrum effects are also often used. The environmental effects are very important in graphite-moderated reactors, and specifically pebble-bed reactors with their mixture of fuel pebbles,. This is largely due to the large mean free path lengths and the transparency of the fuel pebbles. Therefore, it is thus important to introduce the environment effects in the fine group spectrum analysis by representing the cell leakage in some way. The method used in VSOP99 and TINTE [12] represents the broad-group leakage from the core diffusion calculation as buckling terms in the fine group cell calculation (GAM module of VSOP99, Tispec preparation for Tinte). For implementation in the benchmark specification the following definition is used.

Buckling is defined as follows: $\beta^2 = \frac{L}{D.\phi.V}$ [cm^{-2}];

where β^2 is the Buckling; L is the total out-leakage from a given mesh or region; D the diffusion coefficient; ϕ the average flux and V the region volume. Note that a net inflow of neutrons will lead to a negative leakage value and thus a negative buckling. In the case of a negative buckling a necessary condition for a positive flux in each fine group is that $\left|D\beta^2\right| < \Sigma_r$.

This condition can easily be violated if, for example, an assumption of a constant fine-group buckling is used. In the two-group representation used for the benchmark, this should not be a problem but such cases had to be circumvented in the library creation process.

The library contains all the cross sections and kinetic parameters needed to perform the steady-state and transient cases. The current data set includes the following data for fast and thermal energies:

1. Diffusion coefficient, [cm]
2. Macroscopic absorption, [cm-1]
3. Macroscopic nu-fission, [cm-1]
4. Macroscopic fission, [cm-1]
5. Macroscopic Scattering (to the other group) [cm-1]
6. Inverse velocity, [s.cm-1]
7. Fraction Beta(1) of delayed neutrons, [dimensionless]
8. Fraction Beta(2) of delayed neutrons, [dimensionless]
9. Fraction Beta(3) of delayed neutrons, [dimensionless]
10. Fraction Beta(4) of delayed neutrons, [dimensionless]
11. Fraction Beta(5) of delayed neutrons, [dimensionless]
12. Fraction Beta(6) of delayed neutrons, [dimensionless]
13. Kappa, Energy release per fission, [MeV]
14. Microscopic absorption of Xenon , [cm2]
15. Iodine yield [dimensionless]
16. Xenon yield [dimensionless]

The last two quantities are not energy dependent and are given after the fast and thermal set of cross sections and data on the library. The five-dimensional interpolation routines required to read and interpolate the data are provided to the benchmark participants.

3. The Calculational Cases

3.1 Steady State Cases

The steady-state calculational cases were designed to test the correct implementation of the benchmark definition and boundary conditions in the participants' codes in a systematic way. For example, in Exercise 1 a simplified cross section set is provided to verify the correct implementation of the cross-section lookup tables into the different code packages. It also enables participants to use card input cross-section data that is available in many codes. Participants must perform mesh refinement calculations to ensure a converge solution and report the k_{eff}, two-group fluxes and region powers in the spreadsheet template provided.

Exercise 2 is defined to test the implementation of the thermal hydraulic data in the different codes. A two-dimensional heat source distribution is provided as input and used to verify the thermal-hydraulic performance of the models. This includes a wide variety of correlations and constants that includes the pebble-bed effective conductivity, material heat capacities, helium coolant mass flow distribution and the heat transfer from the fuel spheres to the coolant, to name a few. Results that need to be compared include the outlet gas temperature, pressure drop over the core, average and detailed two-dimensional maps of the fuel, graphite and gas temperatures.

A final steady-state case, Exercise 3, is an integrated test of the combined neutronics thermal-hydraulics models. It utilizes the multi-dimensional cross-section library, which includes full feedback on fuel and moderator temperatures, the leakage conditions of the material or cross-section spectrum region and the xenon concentration. This also requires a model that determines the iodine and xenon concentrations explicitly. Exercise 3 represents the starting condition of all the transient cases and a good agreement in these results is essential to assure a good platform for the transient cases.

3.2 Transient Cases

The focus of the benchmark is on the modeling of the transient behaviour of the PBMR core. Six exercises, covering the range from slow to fast neutronic transients, as well as feedback effects from thermal-hydraulic parameters and fission products, are included. A list of the test exercises, some with various sub-cases, with a short description of each is given below:

1. Depressurized Loss of Forced Cooling (DLOFC) without SCRAM
   The event is a Depressurized Loss of Forced Cooling in a very short time. A linear reduction in reactor inlet coolant mass flow from 192.7 kg/s to 0.0 kg/s is assumed over 13 seconds. There is no external flow assumed after this step. During the same time a linear reduction in the reactor helium outlet pressure from ~ 90 bars to 1 bar is postulated. The effects of natural convection are to be excluded in this case for simplicity. Since no SCRAM signal is assumed re-criticality should occur after some time.
2. Depressurized Loss of Forced Cooling (DLOFC) with SCRAM
   The event is the same as Exercise 1 but with a reactor SCRAM after the depressurization phase of 13 seconds. At that time all control rods are fully inserted over 3 seconds to SCRAM the reactor. No re-criticality is expected but all other conditions and assumptions remain unchanged.
3. Pressurized Loss of Forced Cooling (PLOFC) with SCRAM
   The event is a Pressurised Loss of Forced Cooling (PLOFC) with SCRAM. The effects of

natural convection are included in this case. The time sequence of the event is similar to Exercise 2 with a linear reduction of the mass flow to zero over 13 seconds. In this case a linear reduction in reactor helium outlet pressure from ~90 bars to 60 bars takes place over 13 seconds. After the pressure equalization phase is completed, natural convection may start that will lead to some internal mass flow. No external mass flow is allowed. Also after 13 seconds all control rods are fully inserted over 3 seconds to SCRAM the reactor.

4. 100%-40%-100% Power Load Follow
   The event simulates load follow in the plant with a fast change in the power from 100% to 40%, and after some time, back to 100%. The main effect is, of course, the xenon transient due to the power changes, but other feedback effects such as the Doppler temperature also play a role. Two scenarios should be considered. In the first no control-rod movement is allowed, while in the second scenario the control-rods are moved to maintain a critical core within a given reactivity bandwidth. No decay heat effects will be taken into account during the transient so that the heat is only from fission. To assess the Xenon behaviour the Xenon concentrations during these two cases are included in the output.
5. Reactivity Insertions by Control Rod Withdrawal (CRW) and a beyond design event Control Rod Ejection (CRE)
   The exercise defines fast reactivity insertion by simulating different CRW and CRE scenarios at hot full power conditions. Note that no decay heat effects will be taken into account during the transient. Since only the core is included in this specification the changes in the inlet and outlet conditions due to the power conversion unit are not included and therefore the inlet mass flow rate, inlet temperature and outlet pressure should be kept constant at nominal conditions. Four different cases are to be analyzed. They are (i) Withdrawal of all 24 control rods at the maximum speed of 1 $cm.s^{-1}$; (ii) Ejection of all 24 control rods over a 0.1 second duration; (iii) Ejection of a single control rod over a 0.1 second duration; and (iv) Ejection of 6 control rods in one quarter of the core over a 0.1 second duration. Sub-cases ii to iv were selected to include the sub-prompt and super-prompt cases, even though these events are not possible on the plant and thus only of academic value.
6. Cold Helium Inlet Event
   This exercise simulates a bypass valve opening, with “cold” Helium being injected into the core inlet plenum. A temperature ramp of 50 ^{o}C (i.e. 10% of nominal inlet temperature) is applied over 10 seconds, without changing any other reactor parameters like mass flow, pressure or control rod positions. Thus the reactor inlet temperature is reduced linearly from nominal (500 ^{o}C) to 450 ^{o}C over 10 seconds. It is postulated that a reactor protection system would cause the valve to close again after 300 seconds, and the temperature would return to the nominal value, again over 10 seconds. Note that no decay heat effects will be taken into account during the transient.

For all the transient cases the focus of the results are on maximum and average fuel, moderator and gas temperatures and coolant flow behaviour. For the reactivity excursions the core fission power evolution, as well as at selected time snapshots the peak power, power density profiles, and axial offset are of interest. These results are to be presented as a function of time or at specific time points or events.

4. Concluding Remarks, Status and Future Work

The OECD PBMR Coupled Neutronics/Thermal Hydraulics Transient Benchmark based on the PBMR-400 Core Design has reached a mature level of definition and has been well-supported. Two official meetings have already taken place at the OECD/NEA headquarters in Paris. The first workshop was held on 16th and 17th June 2005 at the OECD headquarters in Paris, followed by a second workshop on the 26th and 27th of January 2006 at the NEA headquarters. At the first meeting, attended by 24 participants from 12 countries, the benchmark was introduced and discussed in detail. For the next meeting missing detail and clarifications were added to the specification, and results templates were provided. Results of the first two steady-state exercises were presented and discussed during the second meeting. These results, from thirteen different participating groups, are the subject of an accompanying paper [13]. The third formal workshop is planned for 1 and 2 February 2007. The current plan is to finalize the benchmark in 2009.

The benchmark exercise is an important initiative to verify the different implementations of HTR physics phenomena especially during postulated transients. It already plays an important role in the verification of the codes and methods used to evaluate pebble bed reactor transient events. Many of the phenomena and issues discussed as part of the benchmark exercises provide insight into the shortcomings of some of the methods used today and will also be used to create the basis for future more advanced methods and codes.

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