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STATUS OF THE IAEA COORDINATED RESEARCH PROJECT ON HTGR CORE PHYSICS AND THERMAL HYDRAULICS BENCHMARKS (CRP-5)

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ABSTRACT

High-temperature Gas-Cooled Reactor (HTGR) designs present special computational challenges related to their core physics and thermal-hydraulic characteristics. Neutron streaming and double heterogeneities are examples of the core physics challenges, while porous gas flow in a high-temperature and high-burnup ceramic core, characterize some of the thermal-hydraulic challenges. In an effort to address such potential calculation uncertainties, computational benchmarks have been proposed in the framework of an IAEA Coordinated Research Project (CRP-5) and results by several participating international institutes are used for the purpose of code-to-code as well as code-to-experiment comparisons. The benchmark problems cover areas such as core criticality, power and burnup distribution, safety analysis, decay heat removal and turbo-machinery performance. Experimental data have been made available by the HTTR of Japan, the HTR-10 of China, the ASTRA of Russia as well the Pebble Bed Micro Model (PBMM) facility of South Africa. This paper summarizes the status of this work. The results of finalized benchmarks will be presented.

INTRODUCTION

Modular High Temperature Gas-cooled Reactor (HTGR) designs are one of the reactor concepts being considered by some for future nuclear power plant deployment. Gas-cooled reactor design concepts have been evolving since the 1940's and in recent years there have been a surge of global interest in their modular variants due to their claimed features of enhanced safety and improved economics. HTGR fuel is characterized by a spherical kernel of TRISO coated particles, with a diameter in the range of 500 microns or less and featuring three levels of coating. In the block design, the coated fuel particles are bonded within a graphite matrix to form cylindrical compacts while in the PBMR design, the particles are also imbedded in a graphite matrix but in the form of spherical pebbles. The combination of coated fuel particle design, graphite moderator and helium coolant, gives the HTGR design its distinctive features of high core thermal capacity and high core outlet coolant temperatures, allowing increased efficiency in both electricity production as well as process heat applications such as hydrogen production.

Validation of HTGR core physics and thermal analysis codes helps assess calculation uncertainties and is therefore considered an essential design and licensing step. In this regard, an IAEA coordinated research projects (CRP-5), was launched in 1998, with the objective of validating existing HTGR core calculation methods, by benchmarking them against each other and against available test data. 11 member states have been participating in the benchmark activities. Test data for the CRP has been provided by the HTTR reactor in Japan, the HTR-10 reactor in China and the ASTRA facility in Russia. In addition, additional benchmarks are being considered for the GT-MHR and the PBMR core designs. The core physics codes used vary from detailed Monte Carlo methods to a combination of cell transport and core diffusion models. Streaming effects, double-heterogeneities, impurities and the random distribution of coated fuel particles in the graphite matrix are some of the challenges encountered in these calculations. Thermal analysis codes on the other hand, varied from custom finite volume or finite element methods to Computational Fluid Dynamic (CFD) codes. Challenges here included three-dimensional geometry effects and

validity of thermal correlations such as Graphite thermal conductivity.
The following institutions from the member states have been participated in the CRP activities

1. Institute of Energy Technology (INET), Beijing, China

2. SACLAY (CEA), Gif-sur-Yvette, France,

3. Research Center Juelich (ISR), Juelich, Germany

4. National Nuclear Energy Agency (BATAN), Serpong, Indonesia

5. Japan Atomic Energy Research Institute (JAERI), Oarai, Japan

6. Nuclear Energy And Consaltancy (NRG), Petten, Netherlands

7. Korea Atomic Energy Institute (KAERI), Rep. Of Korea

8. OKBM/Kurchatov Institute, Nizhny Novgorod, Russian Federation

9. Pebble Bed Modular Reactor (PBMR), Centurion, South Africa

10. Nuclear Engineering Department, Hacettepe University, Ankara, Turkey

11. AMEC NNC Limited, Cheshire, U.K

11. Oak Ridge National Laboratory (ORNL), Oak Ridge, TN, U.S.A.

Details of the benchmarks problems and results obtained by different participants are published by the IAEA. The first TECDOC covering benchmark problems related to initial testing of the HTTR and HTR-10 was published in 2003 [1]. A second TECDOC covering benchmark problems related to ASTRA, HTR-10, PBMR, PBMM, and the Pebble Box is under preparation.

HTTR REACTOR PHYSICS BENCHMARKS

The High Temperature Engineering Test Reactor (HTTR) is a 30-MW graphite moderated and helium gas cooled test reactor, operated by JAEA in Japan. Figure 1 shows the core fuel zone arrangement for the HTTR.
The benchmark problem is related to the start-up physics tests and thermal hydraulics of the HTTR. Relevant benchmark problem cases were

1. Initial criticality (HTTR-FC) Phase 1: The determination of the number of fuel columns necessary for the first criticality with clockwise fuel loading one-by-one.
2. Initial criticality (HTTR-FC) Phase 2: Same as the previous case. Additionally, air in graphite void, revised impurity content in dummy block, and aluminum in the temporary neutron detector holders are considered.
3. Control rod position at criticality (HTTR-CR): Control rod insertion depths are evaluated at the critical condition.
4. Excess reactivity (HTTR-EX): The excess reactivity is evaluated for the previous three cases.
5. Scram reactivity (HTTR-SC): The scram reactivity is evaluated for (i) all reflector control rods inserted at the critical condition, (ii) all control rods, reflector and core, are inserted at the critical condition
6. Isothermal temperature coefficient (HTTR-TC): Isothermal temperature coefficients for the fully loaded core are evaluated.

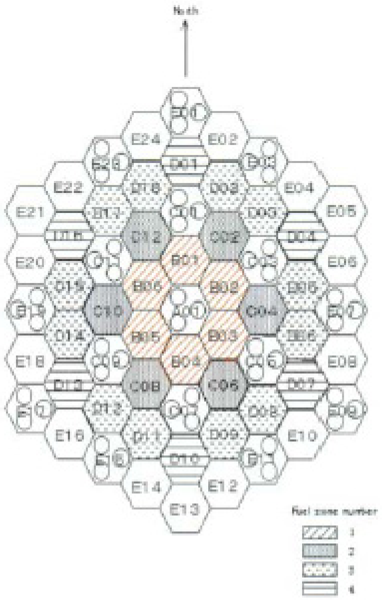

Figure 1. Cross sectional view of HTTR fuel zones

Participants from eight member states contributed towards these benchmark problems. The code system utilized by the participants from Japan consists of the cell codes DELIGHT (collision probability), TWOTRAN-II (transport), and CITATION-1000VP (diffusion theory reactor core analysis). Furthermore, multiplication factors for different core configuration were obtained using the Monte Carlo transport code MVP using the nuclear data of JENDL-3.2. The contribution of France was obtained by the reactor physics code system SAPHYR which mainly consists of APOLLO2 (transport), CRONOS2 (diffusion-transport), and FLICA4 (3D thermal-hydraulics). The Monte Carlo code TRIPOLI4 was also used throughout the study. The participants from Germany involved in the benchmark problems using a series of neutronics codes. A 123 group cross section library was generated based on the JEF-2.2 nuclear data files. The Dancoff factors to account the double heterogeneity were evaluated by the ZUT code. The shielded cross sections were calculated by the NITAWL module of the AMPX-77 system. 1-D cell calculations and 2-D discrete ordinate transport calculations were performed by the TOTMOS and DORT codes, respectively. Eigenvalues and flux distributions in the core were evaluated by the CITATION code. Neutron streaming corrections were made by the multi-group integral transport theory code MARCOPOLO. The Indonesian contribution was based on the SCRAC-EWS code system which utilized the CITATION module for diffusion calculations in the core using cross sections generated by the CELL module. The contribution from the Netherlands was based on the calculations using the modules of SCALE (BONAMI, NITAWL, KENO-Va, BOLD VENTURE, and XSDRNPM) codes system and the reactor code PANTHER-5.0. The participants from the Russian Federation used diffusion codes WIMS-D/4 and JAR for benchmark calculations and Monte Carlo transport codes MCU and MCNP as reference. The participants from Turkey utilized the Monte Carlo transport code MCNP-4B and continuous energy cross sections from the ENDF/BV library throughout their calculations. Similarly, the contribution from the United States of America was based on the calculations performed by the MCNP-4A with cross sections of the ENDF/BV.

The number of fuel columns needed to reach criticality was evaluated in the benchmark problem HTTR-FC. The experimental results are given in Table 1 together with the results supplied by the participants.

Table 1. Results of HTTR-FC benchmark

Participant	# of columns	k-eff
Experimental	19	
Diffusion calculation		
Japan	17	1.0005
France	17	1.0061
Germany	18	1.008
Indonesia	18	1.0058
Russia	16	1.005
Monte Carlo calculation		
Japan	18	1.0061
France	18	1.0085
Netherlands	17	1.0062
Russia (IBRAE)	16	1.006
Russia	17	1.004
Turkey	15	1.005

The results of the second benchmark problem HTTR-CR are given in Table 2.

Table 2 Results of HTTR-CR benchmark

Participant	Control rod position at criticality (mm)		
	18 col.	24 col.	30 col.
Experimental		2215	1775
	Diffusion calculation		
Japan	3035	2055	1665
France			1787
Netherlands			1615
Russia			1660
	Monte Carlo calculation		
Japan	2810	2080	1800
France			1779
Netherlands			1705
Russia (IBRAE)	2590	1950	1700
Russia (RRCKI)	3060	2010	1540
Turkey	2850	2100	1640
USA			1590

The results of the benchmark problem associated with the excess reactivity are given in Table 3.

Table 3. Results of HTTR-EX

Participant	% Δk/k		
	18 col.	24 col.	30 col.
Experimental		7.7	12.0
	Diffusion calculation		
Japan	1.2	9.2	12.6
France	1.7 – 2.7	9.1 – 9.9	12.0 – 12.7
Germany	0.79	8.6	11.8
Indonesia	0.557	6.472	8.517
Netherlands (NRG)			13.8
Netherlands (IRI)			16.5
Russia (OKBM)	2.68	9.73	11.14
	Monte Carlo calculation		
Japan	0.61	9.06	12.5
France	0.85		12.15
Netherlands (IRI)	2.4		13.8
Russia (IBRAE)	2.7	10.83	13.55
Russia (RRCKI)	1.7	9.8	13.4
Turkey	2.98	10.69	13.53
USA			12.28

The next benchmark results are for the scram reactivity calculations (HTTR-SC). Results are given in Table 4.

Table 4. HTTR-SC benchmark results

Participant	% Δk/k	
	Ref. CR	All CR
Experimental	12.0	46.0
	Diffusion calculation	
Japan	8.3-8.9	44.6
France	10.83	56.31
Netherlands (NRG)		37.5
Russia (OKBM)	8.43	52.37
	Monte Carlo calculation	
Japan	9.53	45.1
France	8.56	46.32
Netherlands (IRI)	9.88	47.78
Russia (IBRAE)	9.61	40.40
Russia (RRCKI)	9.55	50.81
Turkey	7.75	37.96
USA		45.0

The last reactor physics benchmark for HTTR is the calculation of reactivity temperature coefficient (HTTR-TC)). Results are given in Table 5.

Deviation in results can be attributed to the uncertainties in the level of impurities, water and air or nitrogen content of graphite pores, the use of different cross section sets, modeling differences, neutron streaming effect in diffusion calculations.

HTTR THERMAL HYDRAULIC BENCHMARKS

The HTTR thermal hydraulic benchmark problems are the vessel cooling system (VCS) and the transient behavior of the loss of offsite electrical power (LP) at 15 and 30 MW. In the first case, the amount of heat removed by the VCS at 30 MW

Table 5. Results of HTTR-TC.

Participant	$(\% \Delta k/k)/k \times 10^{-4}$				
	320	360	400	440	470
Experimental	-1.3 to -1.4				
	Diffusion calculation				
Japan	-1.15 to -1.30 over the entire range				
France	-1.5 to -1.6 between 300 and 420 K				
Netherlands NRG	-1.52 average				
Russia OKBM	-2.23	-2.19	-1.97	-1.82	-1.81
	Monte Carlo calculation				
Japan	-1.23	-1.66	-1.63	-1.56	-0.91
Netherlands IRI	-1.47 average				
Russia IBRAE	-1.95	-1.73	-1.65	-1.77	-1.45
Russia RRCKI	-1.1	-1.7	-0.9	-1.8	-1.3
Turkey	-1.2 @450 K				
USA	-1.3 to -1.4				

power is to be determined together with the associated temperature profile on the surface of the side panel. The analytical simulation on the transient behavior of the reactor and plant during the loss of off site power for HTTR is sought at 15 and 30 MW operation.

The first benchmark problem was analyzed with the SSPHEAT code using the finite element method (FEM) by the participants from Japan. The ACCORD code was used to estimate the hot plenum block temperature, reactor inlet and outlet coolant temperatures, primary coolant pressure, reactor power and heat removal by the auxiliary heat exchanger. South Africa participated in this benchmark exercise using computational fluid dynamics (CFD) package FLOWNET, predecessor of the FLOWNEX [3]. The US contribution is based on the ORNL Graphite Reactor Severe Accident Code (GRSAC). The code was upgraded to provide more detailed analysis of the HTTR. The multi purpose finite element code CAST3M developed at CEA was used for this benchmark by the participants from France.

The results for the HTTR-VC benchmark problem are shown in Table 6.

Table 6. Results of HTTR-VC

Particaipant		Analytical				Exp. Res.
		Japan	Russia	USA	France	
9 MW	VCS heat removal	0.2 MW	0.133 MW	0.18 MW	0.178 MW	0.22 MW
	RPV temp ºC (El. 19-27 m)	170	165	159		170
30 MW	VCS heat removal	0.77 MW	0.494 MW	0.67 MW	0.555 NW	0.81 MW
	RPV temp. ºC (El. 19-27 m)	370-380	330-360	330		340-360

HTR TEST MODULE CORE PHYSICS BENCHMARKS

The HTR-10 is a 10-MW pebble bed gas cooled test rector operated by INET in China. A cross sectional view of is given in Figure 2. Core physics benchmarks proposed for the HTR-10 include the evaluation of initial criticality, control rod worth for the initial and the full core, and temperature coefficients of reactivity.

Countries contributing to this benchmark included China, Indonesia, Japan, Russian Federation, the Netherlands, the United States of America, Turkey, and France, and South Africa. .

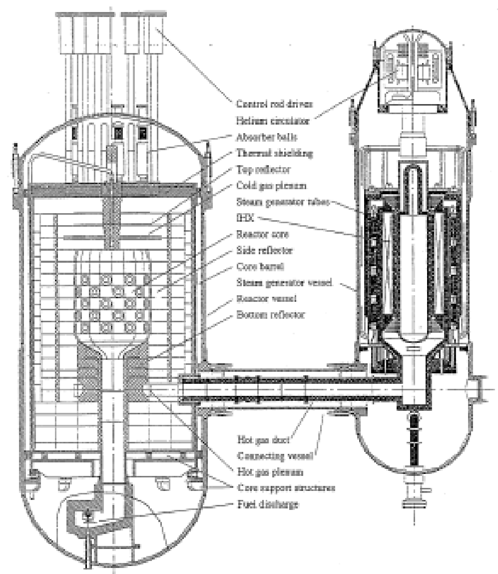

Figure 2. The HTR-10 cross sectional view

The results reported by the various participants are given in Table 7 for the loading height estimation. The experimental critical loading height achieved was123.06 cm.

Another benchmark problem was to evaluate the temperature coefficient at 20℃ (B21), 120℃ (B22), and 250° C (B23) for the HTR-10 for the full core and under helium atmosphere. The results of this benchmark study are given in Table 8.

Another benchmark case was a full core control rod worth evaluation in two parts, one with ten fully inserted control rods (B31) and the second involving one fully inserted control rod (B32). The results of this benchmark case are summarized in Table 9.

Table 7. The HTR-10 critical loading height benchmark results (cms)

Participant	Original		Revised	
Experiment	123			
	Diff/Trans.	Monte Carlo	Diff/Trans.	Monte Carlo
China	125.8	126.1	122.558	122.874
France				115.36 117.37
Germany	124.2 126.8		121.0 123.3	
Indonesia	107 120			
Japan	113			
Russia	136	137.3		
South Africa			122.537	
Turkey	119.27	129.7 135.3		
USA		127.5 128		

Table 8. The HTR-10 temperature coefficient benchmark problem results

Part	B21 (20℃)		B22 (20℃)		B23 (23℃)	
	D/T	M	D/T	M	D/T	M
China	1.1358	1.1381	1.1262		1.1111	
France		1.1568 1.1474				
Germany	1.1468 1.1368		1.1334 1.1232		1.1160 1.1054	
Indonesia	1.2193 1.1381		1.1983 1.1149		1.1748 1.0844	
Netherlands	1.1176		1.1085		1.0963	
Russia	1.1182	1.1076	1.1079	1.1093	1.0927	1.1094
South Africa	1.1286		1.1196		1.1047	
Turkey		1.0941 1.0809		1.0802 1.0380		1.0671 1.0035
USA		1.1319 1.1298		1.1279		1.1245

Another benchmark case for HTR-10 was a control rod worth evaluation for the initial core. This benchmark involves the calculation of the reactivity worth of ten fully inserted control rods (B41) under helium atmosphere and at 20ºC for a loading height of 126 cm and the differential rod worth of one control rod (B42) for the same conditions. The results are tabulated in Table 10.

Table 9. HTR-10 control rod worth benchmark results

Participant	B31 (%)		B32 (%)	
	D/T	MC	D/T	MC
China	14.46	15.31	1.277	1.34
France		13.06 13.44		1.35 1.31
Germany	15.73		1.48	
Japan	18.0			
Netherlands	11.86			
Russia	15.50	17.90		
Turkey		18.73 21.88		2.53 4.60
USA		16.50 16.56		

Table 10. The HTR-10 control rod worth benchmark for the initial core

Participant	B41 (%)		B42 (%)	
	D/T	MC	D/T	MC
China	17.23	18.28	1.540	1.572
France		13.66 13.80		1.52
Germany	19.31		1.86	
Netherlands	19.31		1.86	

HTR TEST MODULE THERMAL HYDRAULICS BENCHMARKS

Three benchmark problems on thermal hydraulics analysis of the HTR-10 reactor have been defined The first one deals with the steady state temperature distribution at full power with the initial configuration. The other two problems are associated with the transient response of the HTR-10, namely, the loss of primary flow without scram and control rod withdrawal without scram benchmarks. The contribution from Turkey is based on the analysis of steady state temperature distribution of the HTR-10 using the CFD code FLUENT. The same problem was approached by the participants from France using the ARCTURUS code, the CFD module of the CAST3M code.

This benchmark problem is still under progress, and therefore, interim results will not be presented.

ASTRA CRITICAL FACILITY BENCHMARKS

The ASTRA Critical Facility is located at the Russian Research Centre – Kurchatov Institute. It has been used for experimental neutron physics investigations of the PBMR. The main idea behind the ASTRA facility is to carry out experiments to validate codes.

The cross sectional views of the ASTRA facility are shown in Figures 3 and 4. The octagon shape core consists of three regions; the outer fuel zone, intermediate mixing zone, and inner graphite reflector zone. The core is surrounded by circular graphite reflector region. Fuel elements are identical with the one used in power reactors in size. However, the fuel content per sphere is less.

Four calculational benchmark problems have been considered. The first task is to evaluate the fuel height necessary to achieve criticality. The next two tasks involve control rods. Worth of control rods is to be determined depending on the position of the side reflector. Interference between control rods with different combinations are also calculated as the third benchmark problem. Finally, the last problem considers the investigation of critical parameters with varying height of the pebble bed.

South Africa, Turkey, France, China, and Indonesia contributed to the benchmark problems on the ASTRA facility.

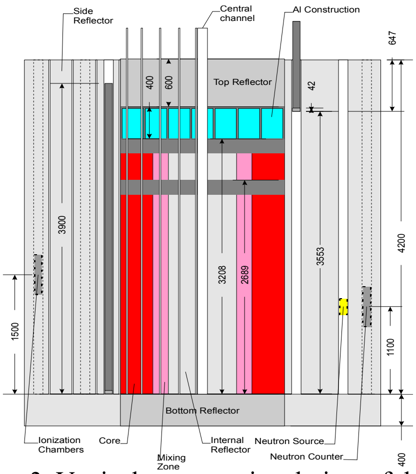

Figure 3. Vertical cross sectional view of the ASTRA Critical Facility

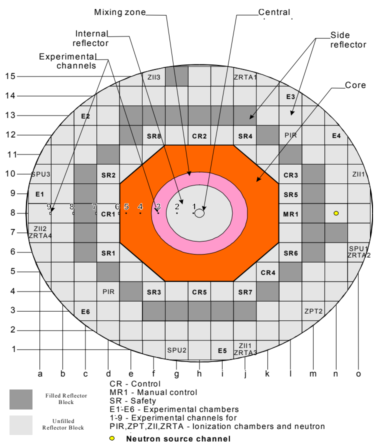

Figure 4. Horizontal cross sectional view of the ASTRA Critical Facility.

PBMR BENCHMARKS

The PBMR benchmark problems have been designed to analyze neutronics and thermal-hydraulics characteristics of the PBMR-400. The Pebble Bed Modular Reactor (PBMR) is expected to be built and operated in South Africa within the next decade. It will be the first commercially operated pebble bed power high temperature reactor.

Several pebble bed reactors such as AVR, THTR-300 have been operated in the past. The development of PBMR began in the early 1990s. The basic idea was to implement direct gas turbine using Brayton cycle. The initial design was with a dynamic central column containing graphite spheres as in the ASTRA critical facility. However, it was later replaced with a solid central column as central reflector.

The calculational model of the PBMR in 2-D is given in Figure 5. The thermal power of the reactor is 400 MW. Spherical fuel pebbles are loaded with 61% packing ratio. The enrichment of the fuel is 9.6%. The core of the PBMR is represented by a thin and long cylindrical region divided into a number of axial and radial nodes. The side, bottom, and top reflectors are also discretized.The flow path of Helium (He) coolant is represented explicitly. He enters into the core from the top at 503˚C and leaves from the bottom at 900˚C.

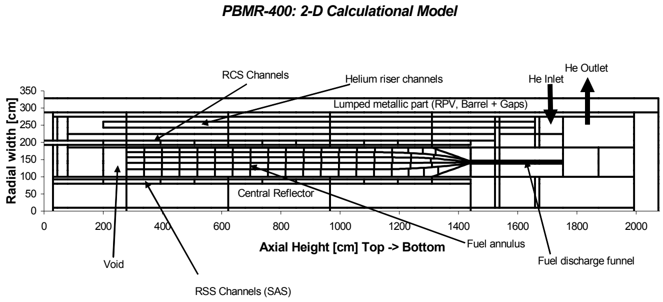

Figure 5. The PBMR computational model

The developmental status of the neutronics and thermal hydraulics analysis tools capable of the analysis of the PBMR is not like those for other reactor technologies. It has been the driving force for the benchmark problems to make comparative analysis of different approaches and methodologies. In order to make analyses tractable, certain simplifications have been introduced. These simplifications include geometrical details as well as flow paths and heat transfer characteristics. Relevant correlations for the thermal conductivity of pebble bed and the other components are also provided.

Benchmark cases may be summarized as

1. Fresh fuel and cold conditions (Case F-1): Core is loaded with fresh fuel and temperature is 300 K. Participants are asked to use their choice of cross sections
2. First core loading (Case F-2): Core composition, 1/3 fuel spheres and 2/3 graphite spheres. Temperatures will be 300, 600, and 900 K.
3. Equilibrium cycle calculation (Case E-1): Neutronic calculations with thermal-hydraulic feedback (fuel, reflector, and coolant temperatures) will be done to obtain power profiles.

4. Depressurized loss of forced cooling (Case T-1): As in the previous case but force cooling is eliminated.

There are four participants in this benchmark exercise. South Africa and China used the VSOP code package. However, South Africa's code has been modified, upgraded, and named the VSOP-NTC. The contribution from Turkey was based on the neutronics calculations using the MCNP and thermal hydraulics calculations using the computational fluid dynamic code FLUENT. The US contribution is based on the code system GRSAC.

This benchmark results have not been finalized yet and the results are not presented here.

PEBBLE BED MICRO MODEL (PBMM) BENCHMARKS

The Pebble Bed Micro Model (PBMM) is an experimental facility which contains a functional model of the power conversion unit (PCU) of the PBMR. It is located at the North West University, Potchefstroom, South Africa. The unit is operated with the Brayton power cycle. Nitrogen is used as the working fluid and acts as energy carrier. The unit is equipped with three separate shafts; one for the high pressure (HP) compressor/turbine pair, one for the low pressure compressor/turbine pair, and one for the power turbine and generator. A recuperator is used to recover heat to be wasted and then that is transferred elsewhere in the system. Thus, overall efficiency of the systems is increased. The schematic layout of the system is shown in Figure 6.

The idea behind the PBMM facility is to provide opportunities for code verification and validation opportunity for complex thermal-fluid systems and dynamics components. Four benchmark problem cases were introduced.

1. Steady state benchmark: The idea is to evaluate steady state operational parameters for the given set of operational parameters. Steady state runs have been performed for two different suction pressures.
2. Transient test 1 (nitrogen injection): Nitrogen is to the system into the upstream of the pre-cooler. The transient response of the system is evaluated upon the increase of nitrogen inventory.
3. Transient test 2 (load rejection) If the load on the generator decreases, the output of the turbine should be reduced. This is accomplished by opening the gas cycle bypass valve. The response of the system for different valve openings is analyzed in this benchmark problem.
4. Transient test 3 (Opening of compressor bypass valves): The power output of the cycle can also be reduced by opening the compressor bypass valves. The system response is analyzed for the case of specific amount of openings in two compressor bypass valves for a given period of time.

South Africa,, France, and Turkey have participated in this benchmark study. The South African participants used the 1-D computational fluid dynamics code package FLOWNEX. The CATHARE system code developed by CEA, EDF, IRSN, and FRAMATOME-ANP was extensively used by the participant from France to analyze the PBMM benchmarks problems. The participants from Turkey utilized the computation environment MATLAB-SIMULINK.
The benchmark study associated with the PBMM is not yet complete.

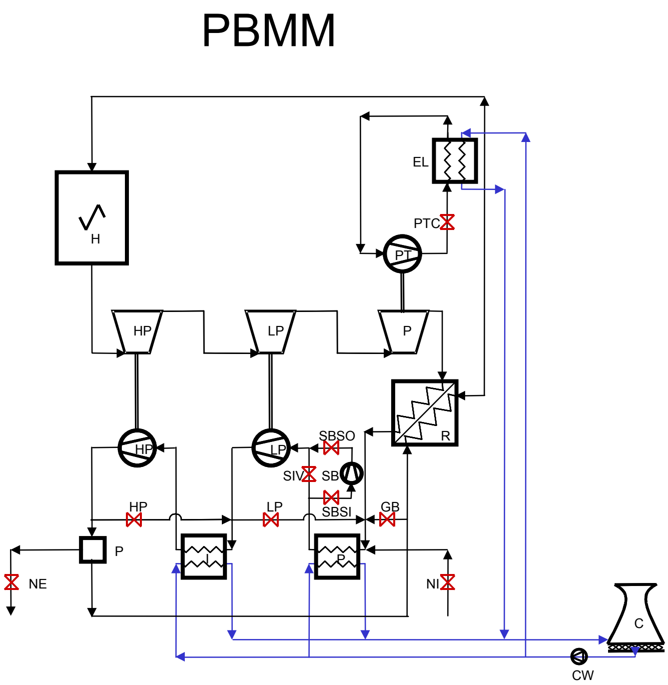

Figure 6. Schematic layout of PBMM.

GT-MHR BENCHMARKS

The GT-MHR is one of the HTGR gas turbine plant designs using the Brayton cycle. The electrical efficiency is around 47%. The GT-MHR reactor design and development involves an international effort. The participants of this project include The Ministry of Atomic Energy of Russia, General Atomics from the USA, Framatome of France, and Fuji Electric of Japan. There is an ongoing effort to burn weapon grade plutonium in the GT-MHR. The core of GT-MHR is consisted of hexahedral fuel assemblies. Each fuel assembly is different than the other with respect to its fuel composition. Thus, fuel management strategies can be employed based local flux, burnup, and isotopic configuration. The core region is surrounded by inner and outer reflector structures which are also hexahedral graphite blocks. The GT-MHR cross sectional view is shown in Figure 7.

The GT-MHR benchmark problems can be categorized in two parts; cell calculations and reactor calculations. In the case of cell calculations, the following benchmarks have been proposed:

1. variation of k_∞ as a function of for a unit cell for the given irradiation conditions,
2. Isotopic compositions as a function of burnup at the given effective full power days

For core-wide calculations, the following benchmarks have been proposed:

1. isothermal reactivity coefficients as a function of temperature at the beginning and end of the cycle
2. control rod worth in the active core at the beginning and end of the cycle
3. control rod worth in the side reflector at the beginning and at the end of the cycle

are considered.

The participants from Russia used Monte Carlo codes MCU and MCNP to analyze GT-MHR benchmark problems. The second group of participations is from France and they used the APOLLO2 code system in their analysis.

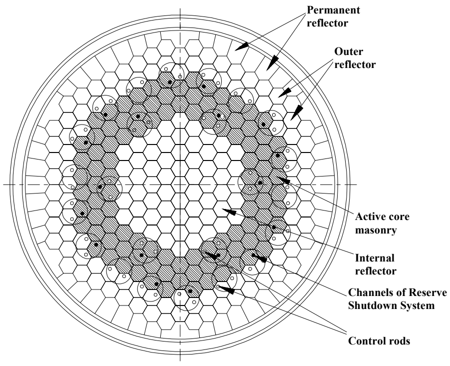

Figure 7. GT-MHR Core layout

CONCLUDING REMARKS

Benchmark problems on HTGR core physics and thermal-hydraulics have been addressed within the framework of an IAEA coordinated research project, CRP-5, with the participation of 13 international institutes. Test data has been made available to the CRP by various test facilities, namely the HTTR, HTR-10, ASTRA and PBMM facilities. In addition, several code-to-code benchmarks have been defined.

The benchmarks provided a useful framework for assessing HTGR core physics uncertainties related to challenges posed by streaming effects, double-heterogeneities, impurities and the random distribution of coated fuel particles in the graphite matrix. Results overviewed here show some variations which can be attributed to modeling differences, uncertainties in cross section, geometry and composition data. Thermal benchmarks also pose challenges related to geometry effects, graphite thermal property changes with irradiation and the range of potential heat transfer regimes, including conduction, convection and radiation.

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