

Chapter 5

RESULTS, CONCLUSIONS AND RECOMMENDATIONS

This Coordinated Research Programme provides the unique opportunity for Member States to both share in the code-to-code evaluation of selected benchmark problems, and also to compare their individual analyses to tests performed on the HTTR and HTR-10 reactors.

This chapter includes a collation of the benchmark problem results obtained by each organization coupled with the actual results achieved on the HTTR and HTR-10 test reactors. It also includes a synopsis by the Chief Scientific Investigators as to the areas of uncertainty and diverse modeling options that may have contributed to differences in individual Member State results as well as recommendations for code and model improvements that can be applied to future reactor design and development activities.

5.1 HTTR REACTOR PHYSICS BENCHMARKS

This section contains a review of the HTTR core physics benchmark problem results obtained by individual Member States. In many cases, the participating organizations performed these benchmarks using both diffusion and Monte Carlo methodologies which are reported herein on separate tables in section 5.1.2.

5.1.1. Collation of Codes and Models

Analyses by individual CSIs related to the HTTR reactor physics benchmark problem calculations included a wide variety of codes, models and methods. These are described in detail in chapter 2 by each Member State. A summary listing of the models and methods utilized for diffusion and Monte Carlo calculations are provided below.

France utilized the reactor physics code system SAPHYR. This includes several codes developed at CEA including APOLLO2 (transport), based on a data base produced with THEMIS/NJOY, CRONOS2 (diffusion-transport) and FLICA4 (3D thermal hydraulics). The Monte-Carlo code TRIPOLI4 was also used.

Indonesia utilized the CITATION model of SRAC-EWS.

Turkey utilized the MCNP-4B code in the HTTR calculations.

USA core physics calculations were performed utilizing the MCNP-4A Monte-Carlo code.

Tables 5.1 and 5.2 present additional details on codes and models utilized by selected Member States for diffusion and Monte Carlo calculations, respectively.

Table 5.1. Analysis method and model for diffusion calculations

Items	Germany	Russia	Japan	Netherlands	
	FZJ	OKBM	HTTR	NRG	IRI
Nuc. Data File	JEF-2.2	ENDF/B6	ENDF/B-4	JEF2.2	JEF2.2
Fuel Cell Code	TOTMOS	WIMS-D/4	DELIGHT	WIMS	SCALE4
Theory	Col.	S4	Col.	Col.	Transport
Model	Cyl.	Cyl.	Cyl.	Cyl.	Cyl.
No. of Groups	123	69	40	69	172
BP Cell Code	TOTMOS DORT	WIMS-D/4	TWOTRAN	WIMS	SCALE4
Theory	Transport	S4	Transport	Col.	Transport
Model	Cyl.	Cyl.	Cyl.	Hex.	Cyl.
No. of Groups	123	69	6	16	172
Core Cal. Code	CITATION	JAR-3D	CITATION-1000VP	PANTHER	BOLD-VENTURE
Model	Tri. (24mesh)	Tri. (6 mesh)	Tri. (24mesh)	Hex.	
No. of Groups (Fast +Thermal)	26	1 + 1	6 (3+3)	2	13
Cut-off Energy (eV)	1.86	0.625	2.38	2.1	2.1

Col. = Collision Probability
Cyl. = Cylindrical

Table 5.2. Analysis method and model description for Monte Carlo calculations

Items	Russia		Japan	Netherlands	
	RRC KI	IBRAE	HTTR	NRG	IRI
Nuc. Data File	DLC/MCUDAT-1.0	ENDF/B6	JENDL-3.2	--	JEF2.2
Energy Struct.	Continuous	Continuous	Continuous		Group
Code	MCU	MCNP 4A	MVP	--	KENO V.a
History	200	2000 (up to 16000)	20000		10000
Batches	5000	1000	150		200
Skipped-Batches	1	10	5		1

5.1.2. Collation of Benchmark Analysis Results

A collation of the results obtained by each Member State for the HTTR core physics benchmark problems are provided in Tables 5.3 through 5.12. Investigation of these benchmarks by many of the participating organizations included analyses utilizing both diffusion and Monte Carlo methodologies. Also included are the actual experimental results obtained on the HTTR.

5.1.2.1. Initial Criticality (HTTR FC)

The number of fuel columns are evaluated for first criticality, with the fuel columns charged from the outer region of the core. They are loaded clockwise, one by one. A small

excess reactivity at the first criticality is also evaluated. The following effects are considered in Phase 2 to improve benchmark problem calculation accuracy: 1) Air in void of graphite, 2) Revised impurity contents in dummy block, 3) Aluminum in the temporary neutron detector holders. Tables 5.3 and 5.4 provide the results for the HTTR FC benchmark problem for diffusion and Monte Carlo calculations, respectively.

Table 5.3. HTTR FC (diffusion calculation)

Member state	Number of fuel columns	Keff	Excess(%dk/k)
Japan	17	1.0005	0.05
France	17	1.0061	0.61
Germany	18	1.008	0.79
Indonesia	18	1.0058	0.577
Russia (OKBM)	16	1.005	0.498
Experimental results	19		

Table 5.4. HTTR FC (Monte-Carlo Calculation)

Member state	Number of fuel columns	Keff	Excess(%dk/k)
Japan	18	1.0061	0.61
France	18	1.0085	0.85
Netherlands(IRI)	17	1.0062	0.62
Russia(IBRAE)	16	1.006	0.596
Russia(RRCKI)	17	1.004	0.398
Turkey	15	1.005	0.50
Experimental results	19		

5.1.2.2. Control Rod Position at Criticality (HTTR CR)

The control rod insertion depths are evaluated at the critical condition for three cases. All control rod insertion levels are adjusted on the same level except three pairs of control rods in the most outer region in the side reflectors. These three pairs of control rods should be fully withdrawn for the calculation at: 1) 18 columns (thin annular core), 2) 24 columns (thick annular core), 3) 30 columns (fully-loaded core). Tables 5.5 and 5.6 provide the results for the HTTR CR benchmark problem for diffusion and Monte Carlo calculations, respectively.

Table 5.5. HTTR CR (diffusion calculation)

Member State	Control rod position at critical (mm)		
	18 col.	24 col.	30 col.
Japan	3035	2055	1665
France			1787
Netherlands (NRG)			1615
Russia (OKBM)	2710	1960	1660
Experimental results		2215	1775

Table 5.6. HTTR CR (Monte Carlo calculation)

Member State	Control rod position at critical (mm)		
	18 col.	24 col.	30 col.
Japan	2810	2080	1800
France			1779
Netherlands (IRI)			1705
Russia (IBRAE)	2590	1950	1700
Russia (RRCKI)	3060	2010	1540
Turkey	2850	2100	1640
USA			1590
Experimental results		2215	1775

5.1.2.3. Excess Reactivity (HTTR EX)

The excess reactivity is evaluated for the three cases mentioned above. The room temperature of 300 K is to be assumed as the moderator and fuel temperatures for the benchmark problem. One atmospheric pressure of helium is to be used as the primary coolant condition. Tables 5.7 and 5.8 provide the results for the HTTR EX benchmark problem for diffusion and Monte Carlo calculations, respectively.

Table 5.7. HTTR EX (diffusion calculation)

Member State	% dk/k		
	18 col.	24 col.	30 col.
Japan	1.2	9.2	12.6
France	1.7 to 2.7	9.1 to 9.9	12.0 to 12.7
Germany	0.79	8.6	11.8
Indonesia	0.577	6.472	8.517
Netherlands (NRG)			13.8
Netherlands (IRI)			16.5
Russia (OKBM)	2.68	9.73	11.14
Experimental results		7.7	12.0

Table 5.8. HTTR EX (Monte Carlo calculation)

Member State	% dk/k		
	18 col.	24 col.	30 col.
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	2.981	10.689	13.525
USA			12.28
Experimental results		7.7	12.0

5.1.2.4. Scram Reactivity (HTTR SC)

The Scram reactivity is evaluated for the following two cases: 1) All reflector CRs are inserted at the critical condition, 2) All CRs in reflector and core are inserted at the critical condition. The core condition for this benchmark problem is as follows:

- Fully-loaded core (30 column fuel core)
- Fresh fuel core

Tables 5.9 and 5.10 provide the results for the HTTR SC benchmark problem for diffusion and Monte Carlo calculations, respectively.

Table 5.9. HTTR SC (diffusion calculation)

Member State	% dk/k	
	Ref.CR	All CR.
Japan	8.3 CR-block and 8.94 for CR-hex	44.6
France	10.83	56.31
Netherlands (NRG)		37.5
Russia (OKBM)	8.43	52.37
Experimental results	12.0	46.0

Critical position at 480K, C, R1, R2 and R3 calculated at 1825 mm(full out) by Japan

Table 5.10. HTTR SC (Monte Carlo calculation)

Member State	% dk/k	
	Ref. CR	All CR.
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
Experimental Results	12.0	46.0

5.1.2.5. Isothermal Temperature Coefficient (HTTR TC)

Isothermal temperature coefficients for the fully-loaded core are evaluated from the effective multiplication. The critical control rod positions are changed with temperature elevation in actual reactor operation. However, the control rod position is not to be changed in the calculation to obtain the reactivity difference. Critical control rod positions are to be evaluated at a temperature of 480K. Tables 5.11 and 5.12 provide the results for the HTTR TC benchmark problem for diffusion and Monte Carlo calculations, respectively.

Table 5.11. HTTR TC (diffusion calculation)

Member State	% dk/k/K x 10 ⁻⁴					
	290	320	360	400	440	470
Japan	-1.15 to - 1.39 over entire range					
France	-1.5 to -1.6 between 300 and 420K					
Netherlands (NRG)	-1.52(Average)					
Russia (OKBM)		-2.33	-2.19	-1.97	-1.82	-1.81
Experimental results	-1.3 to -1.4*					

*Evaluated from measured control rod positions and calculated control rod worth curve

Table 5.12. HTTR TC (Monte Carlo calculation)

Member State	% dk/k/K x 10 ⁻⁴					
	290	320	360	400	440	470
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@ 450K					
USA	-.75 @ 550K					
Experimental results	-1.3 to -1.4					

5.1.3. Discussion of Results and General Conclusions

The Chief Scientific Investigators cited the following areas of uncertainty and different modeling options that may have contributed to differences in the results obtained by the participating organizations. These include:

1. Uncertainties in the level of impurities in the dummy blocks
2. Uncertainties in water and air or nitrogen content of graphite pores
3. Uncertainties in Monte Carlo modeling of coated fuel particles and differences in geometry representation, which may include any of the following options:
 - Explicit geometry
 - Regular array placement
 - Random placement
 - Statistical geometry
 - Homogeneous representation of coated particle region, explicit geometry elsewhere.
4. Choice of selected cross section data library and version (JEF, ENDF, JENDL, etc.)
5. Uncertainties in the modeling of neutron streaming with diffusion methods
6. Difficulty in modeling harmonics in thin annular cores with diffusion methods. This can be mitigated by:
 - Detailed leakage feedback
 - Use of fine group constants or super cell calculations

5.1.4. Recommendations

A comparative review and analysis of the results obtained by the individual Member States was performed by the CSIs at the 4th Research Coordination Meeting. The following recommendations are suggested for incorporation into future reactor research and development activities:

1. Perform a comparison of results from different END/FB libraries (END/FB-VI-4 with older ones). Old libraries have poor graphite scattering data. This will help evaluate magnitude of error.
2. Further investigation of coated fuel particle modeling.
3. Additional experiments and analyses for temperature and burnup dependence of temperature coefficients.
4. Investigation of streaming especially in empty control rod channels and of methods used to calculate anisotropic diffusion coefficients for whole core calculations.
5. Two core physics benchmarks will be proposed after authorization from JAERI, namely:
 - HTTR – PCR: Calculation of control rod insertion depth at 15 and 30 MW powers.
 - HTTR – PTC: Calculation of temperature coefficients at 15 and 30 MW powers.

5.2 HTTR THERMAL HYDRAULIC BENCHMARKS

This section includes a summary of the Member State analyses and the experimental results for the thermal hydraulic benchmark problems of vessel cooling and loss of off-site electric power on the HTTR.

5.2.1. HTTR Vessel Cooling (HTTR-VC)

The vessel cooling benchmark problem included participation by four Member States (Japan, Russia, France and the USA). Reactor data was provided by JAERI for HTTR operation at 9 MW and 30 MW (850°C avg. reactor outlet temp.).

5.2.1.1. Collation of HTTR-VC Codes and Models

Japan

Heat removal of the Vessel Cooling System was calculated using the SSPHEAT code which was developed to analyze the temperature distribution in the in-core structure test section (T2) of HENDEL with complicated passages of helium flow [3-2]. The helium flow was simulated by a thermal-flow element, which is a uniaxial element in three-dimensional space with the ability to conduct heat and transmit fluid between its nodal points. The element has two parameters, temperature and pressure, at each nodal point. Within the computer code, the model is solved by the finite element method (FEM).

Russia

SM-1, GTAS-M and DUPT codes were used for computation of power transferred from the reactor vessel to the reactor cavity cooling system and temperature distribution on the side panel. The SM-1 code is intended for computation of transient temperatures in

structures with arbitrary geometry and based on solution of the heat conduction equation by heat balance method.

USA

The ORNL Graphite Reactor Severe Accident Code (GRSAC) was used for the IAEA CRP-5 HTTR-VC and HTTR-LP benchmark problems, utilizing both steady state and dynamic code features. These calculations relate to the HTTR initial rise to power sequence and safety demonstration tests. An existing HTTR model in GRSAC [3-15] was upgraded to provide more detail in certain critical areas.

France

A model of the High Temperature Engineering Test Reactor was developed for the benchmark concerning the evaluation of the Performance of Vessel Cooling System during normal operation. The CAST3M code [3-17] has been used to model the HTTR. CAST3M is a multi-purpose finite element code developed at CEA, which allows 3D, 2D and R-Z axisymmetric calculations.

5.2.1.2. Collation of HTTR-VC Benchmark Analysis Results

Table 5.13 summarizes the calculated and experimental results for VCS power removal at 30 and 100% power levels.

Table 5.13: Comparison between analytical results and experimental results

Country		Analytical results				Experimental Results
		Japan	Russia	USA	France	
9MW operation	VCS heat removal	0.2 MW	0.133 MW	0.180 MW	0.178 MW	0.22 MW
	RPV temperature (EL. 19-27 m)*	~170°C	165°C	159°C		~ 170°C
30MW operation	VCS heat removal	0.77 MW	0.494 MW	0.67 MW	0.555 MW	0.81 MW
	RPV temperature (EL. 19-27 m)*	370-380°C	330-360°C	330°C		340-360°C

* EL = Elevation (See Figure 3.6)

5.2.1.3. Discussion of Results and General Conclusions

In all cases, power removed by the VCS was underestimated by the calculations. JAERI had observed that hot cavity air leakage and circulation behind the cooling panels was considerably greater than initially expected. This degraded the effectiveness of radiation shields that were to reduce power removed by the VCS. Modeling of the effects of this leakage problem appeared to underestimate this effect to a greater or lesser extent in each of the calculations. The models used ranged from very detailed CFD calculations (France) to simplified empirically-derived models based on the JAERI scaled VCS experiment analyzed in CRP-3 (US).

Predictions of maximum vessel temperatures were generally good; however, the vessel temperatures are more dependent on conditions within the vessel than on VCS performance. The predictions for VCS power at the two operating conditions ranged from ~10% low to ~40% low compared to the measured values. This indicates a typical uncertainty range for VCS performance predictions – based on previous experience with CRP-3 benchmarks for the JAERI VCS mockup experiments. Clearly, additional experience in VCS performance calculations would be useful.

5.2.2. HTTR Loss of Off-site Electric Power (HTTR-LP)

The loss of off-site electrical power from HTTR operating conditions of 15 and 30 MW was the basis for this benchmark problem. Four Member States (Japan, Russia, South Africa and the USA) participated in this activity.

5.2.2.1. Collation of HTTR-LP Codes and Models

Japan

The pre-estimation results of the benchmark problem concerning the loss of off-site electric power simulation of the HTTR were determined utilizing the ‘ACCORD’ code and included the transition of the hot plenum block temperature, reactor inlet and outlet coolant temperatures, primary coolant pressure, reactor power and heat removal of the auxiliary heat exchanger. The estimated duration is for 3600s from the beginning of the loss of off-site electric power.

Russia

Russia used the VGM-code for the HTTR-LP benchmark. This code is intended for calculating normal and emergency transients in nuclear power plants cooled by water or helium.

South Africa

The HTTR system was modeled and analysed using the code “Flownet Nuclear” which is comprised of a graphite-moderated prismatic block reactor, a primary cooling circuit, secondary cooling circuit and an auxiliary cooling circuit. (helium is used as the working fluid).

USA

The ORNL Graphite Reactor Severe Accident Code (GRSAC) was used for the HTTR-LP benchmark problem. An existing HTTR model in GRSAC [3-15] was upgraded to provide more detail in certain critical areas.

5.2.2.2. Collation of HTTR-LP Benchmark Analysis Results

Due to the time related nature of this benchmark problem, a concise collation of results is not a realistic option for this chapter. Therefore, the resulting curves and data obtained by the participating organizations is referenced below by table, figure and page number.

Japan has detailed the plant conditions for the HTTR before the LP test in Tables 3-22 through 3-28 (pages 182-184). The analyzed LP transient is described in Section 3.1.1.2 with the pre-event scenario for the LP transient depicted as Figure 3.4 (page 173). Japan's experimental results of the loss of the off-site electric power from 15 MW and 30 MW is shown as Figures 3.7 and 3.8 (page 182), respectively.

Russia's analysis of HTTR behaviour during the LP event is provided in a family of curves with Figures 3.20–3.32 for the transient from 15 MW and Figures 3.24–3.27 for the transient from full power (see pages 200–203)

South Africa's analysis using Flownet Nuclear includes Figures 3.28–3.43 (pages 223-225) for the LP event from 15 MW and Figures 3.44-3.49 (pages 226-228) for the transient from 30 MW.

Analyses by the USA are depicted in Figures 3.51 and 3.52 for the LP event from 100% power and Figures 3.53 and 3.54 from 50% power. Figures 3.55 and 3.56 provide the results of HTTR loss of off-site electric power even with no auxiliary cooling flow in the pressurized and depressurized conditions, respectively (pages 237-242).

5.2.2.3. Discussion of Results and General Conclusions

The LP tests proved to be rather uneventful, as predicted, with the core temperatures decreasing gradually from the start of the transient. Following the reactor scram and main circulator coastdown, the two auxiliary cooling system circulators started up and ran for the first 40 minutes. After that, one of the two auxiliary circulators was stopped to reduce core thermal stresses associated with a rapid cooldown.

Analyses of the cooldown transients by Russia (VGM code), Japan (ACCORD code), South Africa (Flownet Nuclear code), and USA (GRSAC code) were presented and found to be in general agreement with the experimental results. For the 30 MW case, however, a discrepancy was observed between the Japanese and South African calculations and measurement of auxiliary cooler heat removal rate versus that predicted by both the Russian and USA codes. There are also some discrepancies between the calculated cooldown rates for the core. These differences are being investigated by the parties involved.

5.3 HTR-10 REACTOR PHYSICS BENCHMARKS

This section contains a review of the HTR-10 core physics benchmark problem results obtained by individual Member States. This includes organizations from China, France, Germany, Indonesia, Japan, the Netherlands, Russia, South Africa, Turkey and the USA. In many cases, the participating organizations performed these benchmarks using both diffusion and Monte Carlo methodologies which are reported herein in separate tables.

5.3.1 Analysis Methods and Models

This section includes an overall summary of the codes and models utilized by the participating Member States in their investigation of the benchmark problems associated with HTR-10 core physics

China employed the VSOP code system for calculation of HTR-10 criticality using the diffusion approach. The code system includes GAM for the calculation of fast and epithermal spectrums and THERMOS for the calculation of thermal spectrum. The finite mesh diffusion code CITATION in the code system calculates the eigenvalue problem in four energy groups and in two or three dimensional reactor geometry. Cross-sections of the resolved and unresolved resonances are generated by the ZUT-DGL code. For the Monte Carlo Calculation, the code version of MCNP-4A has been used in the criticality calculation. Nuclear data are based on ENDF/B-V.

The French reactor physics code system SAPHYR used by CEA. SAPHYR gathers several codes developed at CEA like APOLLO2 [4-46] (transport) based on a database produced with THEMIS/NJOY, CRONOS2 (diffusion-transport), and FLICA4 (3D-thermal hydraulics), which are interconnected. The Monte-Carlo code, TRIPOLI4 [4-47], has also been used throughout the study.

In Germany, the HTR-10 benchmark problems were calculated using the following parts of the VSOP code system: the ZUT [4-51], GAM-1 [4-52], THERMOS [4-53], and the CITATION [4-54] code.

BATAN of Indonesia used the WIMS/D4 nuclear design code in calculating HTR-10 first criticality. In addition, a code system consisting of DELIGHT [3-10], TWOTRAN-II [3-11] and CITATION-1000VP [3-12] codes were also used in collaboration between Indonesia and Japan.

In Japan, the HTR-10 core physics calculations were carried out using the HTTR nuclear evaluation code system. This code system consists of DELIGHT, TWOTRAN-II and CITATION-1000VP codes. The DELIGHT code is a one dimensional cell burnup code. Nuclear data was based on ENDF-IV, and III. TWOTRAN-II code was used for control rod cell calculation. CITATION-1000VP code was used for two-dimensional core calculations.

In the Netherlands the HTR-10 was modeled in the PANTHERMIX code, a combination of the 3-D diffusion reactor code PANTHER 5.1 coupled to the 2-D thermal hydraulics code THERMIX./DIREKT. The nuclear data necessary for the PANTHER code was generated by means of the WIMS8 code system.

Russia utilized the WIMS-D/4 code for the diffusion model. This was used to calculate the few-group macrosections characterizing the fuel cells and reflector blocks. The main results of diffusion approximation were obtained in two-group with a thermal cut-off energy of 0,625 eV. The JAR-code of 3D reactor calculation was used for estimation of multiplication coefficients. The nuclear data file was from ENDF/B6. The MCNP4A code was used for the Monte Carlo calculations with the nuclear data files from ENDF/B6 and NJOY.

At Hacettepe University in Turkey, Monte Carlo calculations were carried out using the KENOva module of the SCALE4.4 code system. Diffusion calculations were performed using VSOP'94.

The MCNP4B code was employed in the USA for criticality analysis using ENDF/B-VI cross-section data evaluated at 300 K, and the University of Texas at Austin (UTXS) cross-section library for the temperature-dependent calculations.

South Africa used the VSOP-PBMR design code with DATA2 to model the heterogeneous pebble fuel, the 2-D BIRGIT for reactor geometry and THERMIX for spectral temperature

5.3.2 Collation of Results

The results collation deals with two cases: (1) calculation results of the original defined benchmark problems, and (2) calculation results of the revised (deviated) benchmark problems. The differences between these two cases lie in the following: in the revised (deviated) benchmark problems, the following parameter changes have been considered in comparison to the original defined benchmark problems:

- ✧ Density of dummy balls: 1.73 → 1.84 g/cm³
- ✧ Boron equivalent of impurities in dummy ball: 1.3 → 0.125 ppm
- ✧ Core atmosphere: Helium → Air; Temperature: 20°C → 15°C

Available experimental results are also included in the collation tables.

Table 5-14: Collation of Results for Benchmark Problem B1 (Given in loading height, cm)

	Original Benchmark Problems		Revised (Deviated) Benchmark Problems	
	Diffusion/Transport	Monte Carlo	Diffusion/Transport	Monte Carlo
China	125.8	126.1	122.558	122.874
France ¹	-	-	-	115.36 117.37
Germany ²	124.2 126.8	-	121.0 123.3	-
Indonesia ³	107 120	-	-	-
Japan	113	-	-	-
Netherlands	125.3	-	122.1	-
Russia	136	137.3	-	-
South Africa	-	-	122.537	-
Turkey ⁴	119.27	129.7 135.3	-	-
USA ⁵	-	127.5 128	-	-
<i>Experimental result of critical loading height: 123.06cm. It is noted that the experimental conditions are those conditions for the revised (deviated) benchmarks except the temperature is 15°C instead of 20°C (or 27°C).</i>				

1. The first row of data is obtained with simplified PB modeling, and the second row of data with improved PB modeling.
2. The first row of data is obtained with 2-dimensional VSOP, and the second row of data with 3-dimensional VSOP.
3. The first row of data is obtained with the DELIGHT code, and the second row of data with SRAC code.
4. The first row of data in the Monte Carlo approach is obtained with the ENDF/B-IV nuclear data set, and the second row of data with ENDF/B-V nuclear data set.
5. The first row of data is obtained with the UTXS nuclear data set, and the second row of data with ENDF/B-VI nuclear data set.

3.3.2.1. Initial Criticality (Benchmark Problem B1)

This benchmark problem involved calculating the amount of loading (given in loading height, starting from the upper surface of the conus region) for the first criticality: $K_{\text{eff}} = 1.0$ under the atmosphere of helium and core temperature of 20 °C, without any control rod being inserted. A collation of results by Member States is provided in Table 5-14.

Table 5-15: Collation of Results for Benchmark Problem B2, (original)

	B21 (K_{eff} at 20 or 27°C)		B22 (K_{eff} at 120°C)		B23 (K_{eff} at 250°C)	
	D/T	M	D/T	M	D/T	M
China	1.1197	-	1.1104	-	1.0960	-
Germany ¹	1.13725 1.12665	-	1.12404 1.11331	-	1.10693 1.09588	-
Indonesia ²	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	1.0933	1.0927	1.0794
Turkey ³	-	1.0941 1.0809	-	1.0802 1.0380	-	1.0671 1.0035
USA ⁴	-	1.1319 1.1298	-	1.1279	-	1.1245

D/T: Diffusion and Transport approach, M: Monte Carlo approach

1. The first row of data is obtained with 2-dimensional VSOP, and the second row of data with 3-dimensional VSOP.
2. The first row of data is obtained with the DELIGHT code, and the second row of data with SRAC code.
3. The first row of data in the Monte Carlo approach is obtained with the ENDF/B-IV nuclear data set, and the second row of data with ENDF/B-V nuclear data set.
4. The first row of data is obtained with the UTXS nuclear data set, and the second row of data with ENDF/B-VI nuclear data set.

Table 5-16: Collation of Results for Benchmark Problem B2, Revised (deviated)

	B21 (K_{eff} at 20 or 27°C)		B22 (K_{eff} at 120°C)		B23 (K_{eff} at 250°C)	
	D/T	M	D/T	M	D/T	M
China	1.1358	1.1381	1.1262	-	1.1111	-
France ¹	-	1.15679 1.14737	-	-	-	-
Germany ²	1.1468 1.1368	-	1.1334 1.1232	-	1.1160 1.1054	-
South Africa	1.12861	-	1.11956	-	1.10469	-

D/T: Diffusion and Transport approach, M: Monte Carlo approach

1. The first row of data is obtained with simplified PB modeling, and the second row of data with improved PB modeling.
2. The first row of data is obtained with 2-dimensional VSOP, and the second row of data with 3-dimensional VSOP.

3.3.2.2. Temperature Coefficient (Benchmark Problem B2)

This benchmark problem was to calculate the effective multiplication factor K_{eff} of the full core (5m^3) under helium atmosphere and core temperatures as follows: 20°C (B21), 120°C (B22) and 250°C (B23) respectively, without any control rods being inserted. A collation of results by Member States is provided in Table 5-15 and 5-16 for the original and deviated benchmark, respectively.

3.3.2.3. Control Rod Worth for Full Core (Benchmark Problem B3)

This problem included calculating the reactivity worth of the ten fully inserted control rods (B31), and of one fully inserted control rod (B32, the other rods are in the withdrawn position) under helium atmosphere and a core temperature of 20°C for the full core. A collation of results by Member States is provided in Table 5-17.

Table 5-17. Collation of Results for Benchmark Problem B3

	Original Benchmark Problems				Revised (Deviated) Benchmark Problems			
	B31 (%)		B32 (%)		B31 (%)		B32 (%)	
	D/T	M	D/T	M	D/T	M	D/T	M
China	15.24	16.56	-	1.413	14.46	15.31	1.277	1.343
France ¹	-	-	-	-	-	13.06 13.44	-	1.35 1.31
Germany	16.6	-	1.56	-	15.73	-	1.48	-
Japan	18.0	-	-	-	-	-	-	-
Netherlands	11.86	-	-	-	-	-	-	-
Russia	15.50	17.90	-	-	-	-	-	-
Turkey ²	-	18.73 21.88	-	2.53 4.60				
USA ³	-	16.50 16.56	-	-	-	-	-	-

D/T: Diffusion and Transport approach, M: Monte Carlo approach

1. The first row of data is obtained with simplified PB modeling, and the second row of data with improved PB modeling.
2. The first row of data in the Monte Carlo approach is obtained with the ENDF/B-IV nuclear data set, and the second row of data with ENDF/B-V nuclear data set.
3. The first row of data is obtained with the UTXS nuclear data set, and the second row of data with ENDF/B-VI nuclear data set.

3.3.2.4. Control Rod Worth for the Initial Core (Benchmark Problem B4)

This benchmark problem involves calculation of the reactivity worth of the ten fully inserted control rods (B41) under helium atmosphere and core temperature of 20°C for a loading height of 126cm, and the differential worth of one control rod (B42, with the other rods in the withdrawn position). The differential reactivity worth is calculated when the lower end of the rod is at the following axial positions: 394.2cm, 383.618cm, 334.918cm, 331.318cm, 282.618cm, 279.018cm, 230.318cm.) under helium atmosphere and core temperature of 20°C for a loading height of 126cm. A collation of results by Member States is provided in Table 5-18.

Table 5-18: Collation of Results for Benchmark Problem B4

	Original Benchmark Problems				Revised (Deviated) Benchmark Problems			
	B41 (%)		B42 (%)		B41 (%)		B42 (%)	
	D/T	M	D/T	M	D/T	M	D/T	M
China	18.27	19.36	1.619	1.793	17.23	18.28	1.540	1.572
France ¹	-	-	-	-	-	13.66 13.80	-	1.52
Germany	20.50	-	1.97	-	19.31	-	1.86	-
Netherlands	13.61	-	-	-	-	-	-	-

Experimental result of one rod (S3 rod) at initial core (B42): 1.4368%. It is noted that the experiment result is not directly to be compared with calculated data, as the control rod movement in the experiment is not exactly the same as benchmark definition. For detailed reasons please refer to Section 3.2.1.6.

D/T: Diffusion and Transport approach, M: Monte Carlo approach

1. The first row of data is obtained with simplified PB modeling, and the second row of data with improved PB modeling.

5.3.3. Analysis and Conclusion

It is readily observed that the analyses by the Member State organizations can be factually divided as follows:

1. There are cases where good agreement exists between (1) different approaches (2) different participating institutions (3) calculation and experiment.
2. Generally and statistically speaking, the differences are huge between (1) different approaches (2) different participating institutions (3) calculation and experiment.

An analysis of why there are large statistical differences in the results between participating organizations is as follows:

1. Some cell calculations for preparing control rod cross-sections are more appropriate for rod array evaluations rather than single rod worth, due to boundary conditions used.
2. Control rod evaluations in partially fuelled core where some parts of the rod are adjacent to fuel and some he/air.
3. Uncertainty in the modeling of neutron streaming (directional diffusion calculations are required)
4. Choice of selected cross section library (JEFF, ENDF, JENDL, etc.)
5. Water content of graphite pores (assured zero but some may remain)
6. Investigation of library dependent cross-sections for temperature coefficients
7. Effective fuel homogenization methods for deterministic approach
8. Uncertainties in the modeling of fuel with Monte Carlo calculations including:
 - Explicit geometry
 - (ア) Coated particles, uniform distribution of particles is important throughout the pebbles
 - (イ) Pebbles, placement and distribution is important
 - Some analysis were performed using homogeneous cross-sections representing the fuel region containing particles, explicit geometry was used elsewhere as a hybrid method
 - The modeling of the fuel/dummy ball ratio

5.3.4. Recommendations

A comparative review and analysis of the results obtained by the individual Member States was performed by the CSIs at the 4th Research Coordination Meeting. The following recommendations are suggested for incorporation in future reactor research and development activities:

1. Perform a comparison of results from different ENDFB libraries (ENDF/B-VI-4 with older ones). Old libraries have poor graphite scattering data. This will help evaluate magnitude of error.
2. Further investigation of pebble modeling
3. Investigation of streaming especially in empty channels in graphite reflectors. Need to determine how to prepare directional diffusion coefficient for whole core calculations.