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Meeting Report of the

3^{rd} Research Coordination Meeting (RCM) of the CRP1496

on

Benchmarking against Experimental Data of the Neutronic and Thermalhydraulic Computational Methods and Tools for Operation and Safety Analysis for Research Reactors

International Atomic Energy Agency

Aix-en-Provence, France

5 -9 December 2011

Vienna, Austria ; December 2011

CONTENTS

A. Background and Objectives

The topics covered by this CRP directly support the overall objectives of the Project 3.2.5.1 (i.e. to enhance the safety of research reactors), through promotion of the use in safety analysis of computational tools validated against experimental data and enhancing of knowledge sharing. The CRP also supports the Project 1.4.2.2 (i.e. to support research reactor modernization and innovation), by collecting and sharing relevant information and the Project 1.4.2.1 (i.e. to enhance utilization and applications of research reactors) in terms of validation of computer codes important to various applications of research reactors. The topics of the CRP as well as the subjects of the individual contracts and agreements remain relevant to the safety, operation, and application of the research reactors in the participating institutions.

The output of the CRP will be an IAEA publication describing its results and conclusions. The expected research outcomes are:

- Transferred know-how in the area of research reactor numerical analysis, including design, safety analysis, operation, and utilization;
- Set of experimental data for benchmarking of the neutronic and thermal-hydraulic computer codes;
- Benchmarked neutronic and thermal-hydraulic computer codes against experimental results;
- Identified user effect on the results predicted by the computer codes;
- Enhanced capabilities of the CRP participants in performing research reactor numerical analysis and safety assessment.

According to the CRP Action Matrix, which was developed in the first RCM, and following the individual work plans, the activities of the Chief Scientific Investigators (CSIs) to date focused on benchmarking against experimental data of the neutronic and thermalhydraulic computer codes that are available at their institutions against the experimental results collected during this CRP.

The main objectives of this 3rd RCM were

- To discuss the results achieved to date for the individual Research Contracts/Agreements
- To evaluate the progress made towards achievement of the CRP objectives and outcomes
- To adjust the work plans of the individual Research Contracts/Agreements for the remaining CRP duration
- To revise the draft publication under preparation as an output of the CRP
- To prepare draft meeting report.

The meeting also included a technical tour to the construction sites of the research reactor RJH and experimental fusion reactor ITER located at CEA Cadarache centre.

B. Meeting Achievements

The meeting was attended by 30 participants representing 15 Member States and 2 international organizations, namely IAEA and NEA/OECD. Mr D. Ridikas, NAPC-PS and Mr A. M. Shokr, NSNI-RRSS were Scientific Secretaries of the meeting. Detailed list of all participants is provided in

Annex VI. The meeting was hosted by the CEA, in Aix-en-Provence, France, and hence was opened by Mr Jerome Estrade, Section Head of Jules Horowitz Reactor from CEA, followed by welcome remarks of the IAEA Scientific Secretaries. After a brief self-introduction of the participants the Agenda proposed by the Scientific Secretaries was adopted (see Annex V). Mr. A. Olson (ANL, USA) and Mr. R. Prinsloo (NECSA, South Africa) were respectively designated as Chairperson and Rapporteur of the Meeting.

The first two days of the meeting included a brief review by the IAEA representatives of the CRP objectives and the expected results, followed by technical presentations by the project CSI's as well as observers on the status of the work and results achieved so far. Summary reports of all individual country achievements are included in Annex VII.

The second afternoon of the meeting was dedicated to technical visits to the construction sites of JHR and ITER at CEA Cadarache centre and the meeting participants expressed their gratitude to the CEA and IAEA for including these interesting visits in the schedule.

Day three of the meeting included a number of additional presentations from invited experts and observers. Amongst others, topics included an overview of a set of tools developed by the NEA/OECD for describing, searching and analysing reactor neutronics benchmarks (Mr I. Hill), an overview of Fukushima accident analysis (Mr C. Allison) and an overview of various code systems and approaches by participants from France and Korea. Finally some new data from Brazil RR were presented and will be available for the benchmarks.

During the afternoon session, the participants revised the CRP Action Matrix and adjusted the corresponding individual work plans for the remaining period of the CRP, as given in detail in Annex I. An effort was made to confirm that sufficient results are available for each of the 10 benchmark problems, and to ensure that all results will be submitted by either June 2012, or at the latest October 2012. All updates of benchmark specifications, including experimental data, are required by the end of January 2012. Other deadlines and action plan are given in Annex II.

The 4th and the last RCM was thus suggested to be held around February 2013. The meeting further discussed the responsibilities regarding collation of results for each benchmark, and the CSI's responsible for these tasks were confirmed. The day was concluded with specific breakout sessions where data providers could clarify specific issues posed by involved participants.

Thursday morning was opened with some general discussion, giving participants an opportunity to raise issues not yet covered during the meeting thus far. The below items resume these discussions and decisions taken:

- The participants were requested to formulate some preliminary technical conclusions based on analysis performed thus far. These conclusions were included in the participant feedback reports in Annex VII

- The possibility to identify certain benchmarks as high quality reference benchmarks and as such include some more extensive calculation data for future comparison was discussed. In this regard some criteria should be defined to identify such a subset of high quality benchmarks for further exploration and could lead to more complex benchmarks for building upon the experience gained during this CRP

- The need for establishing a database of thermal-hydraulic benchmarks problems, similar to that which exists at the NEA/OECD for criticality and reactor physics benchmark problems, was highlighted. Such databases do exists for power reactors, but the need for a focussed research reactor counterpart still needs to be developed. This CRP is already addressing this issue in terms of providing 10 well documented benchmark specifications.

- In order to ensure effective knowledge transfer and perform combined analysis of individual results, reported by all CRP members, a dedicated CM will be held in 2012 prior to the final 4th RCM meeting. This CM will collate joint technical conclusions particular to the various codes and benchmark problems performed within the CRP.

Further activities during the last two days focussed on the review and finalisation of important draft documents such as the two IAEA publications, as well as results templates needed for submission of calculation results. Issues raised from the previous meeting report were discussed in detail during the final sessions of the meeting, and the following was highlighted:

- Shortcomings in MCNP's capabilities in calculating kinetic parameters. Some efforts and studies were made in this regard (two participants addressed it explicitly), and good experience was shared with the rest of participants.
- Distribution of "best" input models for various benchmarks may be shared amongst participants at the end of the CRP. The added value of doing so is evident, i.e. the "best" model is made available to all interested participants, if consensus and time allows.

Finally, some time was spent on discussing the possible outcomes and future activities following this CRP. Possible venues for the final RCM in February/March 2013 were discussed, where Greece and Italy were mentioned as possibilities. The last session of the meeting was dedicated to the formulation of the meeting conclusions and recommendations and to consolidate a draft of the meeting report. All meeting materials, including presentations of the individual participants, were distributed to the meeting participants. Additional copies are available on request from the Scientific Secretaries.

At the very end gratitude and appreciation was expressed by the IAEA representatives as well by all meeting participants to the host organizers for their considerable efforts and continuous support provided during the meeting.

C. Summary of Conclusions

The activities of the 3rd RCM showed a significant progress in achieving the objectives of the CRP. The status of facility and experiment descriptions (benchmark specifications) were once more reviewed and final comments and clarifications were gathered prior to drafting the first IAEA publication containing this information. Regarding participant progress, all actions plans were reviewed and the targets set during the previous meeting are on track. Minor adjustments were made to the action matrix (Annex I).

During this meeting, the draft of the first IAEA publication, the table of contents of the second publication, and the results templates for participant submissions were reviewed and accepted with minor adjustments. All benchmarks have the required minimum contributions (some still in progress) and participants have made at least the minimum required number of commitments (participation in 2 benchmarks).

Given the scope of this CRP, which includes ten benchmark problems (with at least three experiments each) covering broad subjects related to research reactor safety, operation and utilization, the participants were informed during 2011 that the CRP was extended by an additional year. The final RCM for this CRP is set for February or March 2013, with one additional consultancy planned during 2012 to summarise and collate joint conclusions. Participants have to submit final individual reports by either June (1^{st} reporting period) or October 2012 (2^{nd} and the last reporting period), and all benchmark specifications have to be finalised by February 2012.

The inclusion of uncertainties in the final reports, in both experimental and calculated data, were discussed and it was reiterated that information related to uncertainties should be included as much as possible/available.

In addition, the following conclusions were drawn by participants:

- The CRP is helpful to the research reactor community **thanks to the data providers** for making this effort possible. **The CRP has provided a set of data and results which were clearly missing from literature.**

- The various benchmark problems have been challenging and have provided an excellent opportunity for good practice and lessons learned. Although the CRP has achieved a great deal in gathering relevant benchmarks and perform preliminary analysis on all of these, it was noted that **interaction between neutronics and thermal-hydraulics components of these benchmarks were still not optimal.** In most cases these disciplines were treated rather independently and suggestions for improvement include either coupled calculations, or at least coupled approaches by neutronic and thermal-hydraulic analysts.

- Benchmark development, followed by fine detail and fine scale modelling took significant effort and justified extension of the CRP for an additional year. However, t**here is a need to continue dedicated communication efforts between participants and relevant data providers**.

- The comparison planned within this CRP between individual submissions by the participants and joint benchmarking efforts is an added value of this project in terms of evaluating both user effects and models used in the codes employed. In the analysis of submitted results, when possible, **a clear distinction should be made between the evaluation of the code versus the evaluation of the user effect. In addition, feedback to code developers in the process will be also valuable. Therefore, continuation of joint activities after closure of this CRP (Q2 2013) is advisable**

- It is strongly recommended to **include input models, i.e. input files, in the final publication (at least electronically)** so that the future users could build upon the lessons learned.

- The benchmarks performed so far show that static **multi-dimensional neutronics, with point kinetics and multi-channel thermal-hydraulics is a commonly used approach for the reactor problems in this benchmark exercise.** 3D time-dependent solutions could be evaluated for selected transients to define the added value of such detailed modelling.

- The benchmarks performed so far show that **neutronics modelling has proven to be reasonably accurate**; **obtaining good agreement for thermal-hydraulics analysis is more challenging as** similar problems were experienced by many of the users.

- It would be useful for multiple group **analysts to be involved in the design and analysis of future experiments for the sake of code benchmarking and validation.** Information on such future experiments could be collected and shared through the present CRP web-portal.
- The definition of quality of the benchmarks is a difficult task, and often shortcomings are only found out during advanced stages of the modelling. Nevertheless the **supplied benchmarks have reached an acceptable level of completeness and certainly add significant value to the research reactor community.**
- **Valuable understanding is being gained by code developer(s) about how to guide code users within the research reactor community**, and feedback has lead to new development directions.
- **Good communication between analysts, data suppliers and code developers should be continued and facilitated** to ensure that a final high quality product is delivered.

D. Summary of Recommendations

During the final day of the meeting the participants were encouraged to formulate relevant recommendations regarding follow up actions and additional efforts related to this CRP. In particular, the following issues were raised:

1. To **define and initiate an effort on fully coupled 3D reactor kinetics and thermal-hydraulics benchmark with pre-defined transients**. Such a benchmark could add significant value to determine the state of the art of modern code systems against recent research reactor designs.
2. To **consider the development of training material/course based on the data and models used/developed during the CRP; promote the data base to interested parties outside the CRP as soon as it is finalised/published.**
3. To **allow future submissions to the IAEA database in regard to both additional benchmarks as well as further calculated results on existing benchmarks**, and hence consider updated publication in the future.
4. To **organize code or benchmark specific user group meetings or facilitate other means of communication among the participants** to enhance the quality of the final submissions to the second IAEA publication.
5. To **consider a research reactor based depletion benchmark exercise as a potential future initiative**, in particular to address issues related to reactor utilization, source term definition, waste quantification, etc. Organization of a CM on this topic is recommended.
6. **Access the NEA/OECD databases ICSBEP and IRPHEP for non-NEA/OECD member states is possible provided that member states are contributing to these databases by submitting their neutronic benchmark specifications for inclusion**. Such initiatives are encouraged and should be facilitated by the IAEA.

7. **To encourage data providers to examine the NEA/OECD uncertainty guidelines and consider applying them to their neutronics benchmark problems. These guidelines are available through the NEA/OECD web page.**

ANNEX I. REVISED ACTION MATRIX

Available Exp. data from	Neutronics							Depletion/Activation		Thermalhydraulics		
	Criticality (K_{eff}, K_{inf})	flux shape/ profile	Flux spectrum	Control rod worth	Reactivity effects	Reactivity co-efficients	Kinetic para-meters	Depletion (fuel, burnable poisons)	Activation outside the core	Steady state temp.	loss of flow transient	Reactivity insertion transient
OPAL	ARG, AUL, PAK, SAF, KOR	ARG, AUL, PAK, SAF, KOR	ARG, AUL, PAK, SAF, KOR	ARG, AUL, PAK, SAF, KOR		ARG, AUL, PAK, SAF	ARG, AUL, PAK, SAF	ARG, AUL, SAF				
ETRR-2 Some data is missing	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF			ARG, AUL, GRE, EGY, SYR, SAF	ARG, AUL, GRE, EGY, SYR, SAF	ARG, AUL, GRE, EGY, SYR, SAF
WWR-SM Some data is missing	EGY, USA+UZB	EGY, USA+UZB		EGY, USA+UZB	EGY, USA+UZB	EGY, USA+UZB	EGY, USA+UZB			EGY, USA+UZB	EGY, USA+UZB	EGY, USA+UZB
MNSR-Y			PAK, SYR, USA+ GHA	PAK, SYR, USA+ GHA	PAK, SYR, USA+ GHA	PAK, SYR, USA+ GHA	PAK, SYR, USA+ GHA			PAK, SYR, USA+ GHA		PAK, SYR, USA+ GHA
MINERVE	FRA, FRA2, ROM	FRA, FRA2, ROM	FRA, FRA2, ROM	FRA, FRA2, ROM	FRA, FRA2, ROM							
MNR	ARG, CAN, SYR, SAF	ARG, CAN, SYR, SAF		ARG, CAN, SYR, SAF	ARG, CAN, SYR, SAF	ARG, CAN, SYR, SAF						
RSG-GAS Some data to be clarified										ARG, GRE, EGY, SYR	ARG, GRE, EGY, SYR	
SPERT III	GHA,	GHA,	GHA,	GHA,	GHA,	GHA,	GHA,					GHA,

	ROM, USA	**ROM, USA**	**ROM, USA**	**ROM, USA**	**ROM, USA**	**ROM, USA**	**ROM, USA**					**ROM, USA**
SPERT IV	**ALG, AUL, BGD, FRA, FRA2, PAK, SYR**	**ALG, AUL, BGD, FRA, FRA2, PAK, SYR**	**ALG, AUL, BGD, FRA, FRA2, PAK, SYR**	**ALG, AUL, BGD, FRA, FRA2, PAK, SYR**	**ALG, AUL, BGD, FRA, FRA2, PAK, SYR**	**ALG, AUL, BGD, FRA, FRA2, PAK, SYR**	**ALG, AUL, BGD, FRA, FRA2, PAK, SYR**					**ALG, AUL, BGD, GRE, FRA, FRA2, ITA, PAK, SYR**
IEA-R1 **New data to be included**										**ALG, ARG, BRA, GRE, Korea, SYR, ITA, BGD**	**ALG, ARG, BRA, GRE, Korea, SYR, ITA, BGD**	

The letters in the Table indicate a CSI from:

Algeria (ALG), Argentina (ARG), Australia (AUL), Bangladesh (BGD),
Canada (CAN), Egypt (EGY), France-CEA (FRA), Ghana (GHA),
Italy (ITA), Pakistan (PAK), Romania (ROM), South Africa (SAF),
Syria (SYR), United States of America (USA), Uzbekistan (UZB).

The BRA, GRE, FRA2, and KOR represent the observers from Brazil, Greece, France-IRSN, and Republic of Korea respectively.

* Red colour stands for individual results/reports submission by 1 October 2012. Otherwise, the reports should be submitted by 1 June 2012.

** Yellow highlights indicate that benchmark description still needs some clarification/input. This should be finalized by 1 February 2012.

ANNEX II. LIST OF THE FOLLOW-UP ACTIONS

Deadline	Action	Responsible
20/12/2011	Clarification on procedure for publications in international journals under CRP	IAEA
01/02/2012	Submission of all final specifications and data needed to perform the benchmarks (submission of contributions to the TECDOC-1 on RR specifications, experiment description and associated experimental data)	10 benchmark data providers
May 2012	Finalization of the 1st CRP publication	IAEA + experts
01/06/2012	Submission of benchmarking results: individual reports + result reporting sheets agreed for the 1st reporting period	All
15/07/2012	Submission consolidated report draft regarding the 1st reporting period	Result consolidation team
01/10/2012	Submission of remaining benchmarking results: individual reports + result reporting sheets agreed for the 2nd and final reporting period	All
15/11/2012	Submission consolidated report draft regarding the 2nd and the last reporting period (full draft of the consolidated results report)	Result consolidation team
Dec. 2012	1st CM to discuss the full draft of the consolidated results report.	IAEA + experts
Mar. 2013	4th and the last RCM to discuss/agree on the full draft of the consolidated results report, wrap up of CRP	All (Vienna, Greece or Italy)
May 2013	2nd CM to discuss and finalise the 2nd CRP publication	IAEA + experts

Experimental data from	CSI responsible for consolidation of the results (neutronics)	CSI responsible for consolidation of the results (thermal-hydraulics)
OPAL	AUS, ARG	Not applicable
ETRR-2	EGY, ARG	GRE
WWR-SM	UZB+USA	UZB+USA
MNSR-Y	SYR	SYR
RSG-GAS	Not applicable	GRE
MINERVE	FRA	Not applicable
MNR	CAN+SAF	Not applicable
SPERT III	USA	USA
SPERT IV	CAN+SAF	CAN+GRE
IEA-R1	Not applicable	BRA+SYR

* Red colour stands for urgent clarification and confirmation from engaged partners.

ANNEX III. TABLE OF CONTENTS FOR THE CRP PUBLICATION PART II (BENCHMARKING RESULTS)

1. Foreword (countries, etc.)
2. Introduction/Background
3. Short description of tools/codes (all tools for all experiments)
    a. Code 1
    b. Code 2
    c. …
4. Short description of Facility 1: short description to be provided, Ref. is made to the CRP publication Part I
    a. Experiment 1
        i. Short description of experiment
        ii. Summary and comparison of benchmark results
        iii. Discussion of benchmark results
        iv. Conclusions and recommendations

        (based on results obtained by different codes/users for the same experiment)
    b. Experiment 2

…

5. Facility 2 (equivalent to bullet 4)

…

6. Summary and Conclusions

    (based on consensus/agreement reached during the final discussions, see bullet a.iv, above
7. Recommendations
8. References
9. Annexes…

ANNEX IV. TABLE OF CONTENTS FOR THE FINAL REPORT OF THE INDIVIDUAL COUNTRY REPORTS

1. Introduction/Background
2. Description of tools/codes (all tools for all experiments)
   a. Code 1
   b. Code 2
   c. …
3. Facility 1: short description to be provided, Ref. is made to the CRP publication Part I
   a. Experiment 1
      i. Short description of experiment, Ref. is made to the CRP publication Part I
      ii. Description of models and methods, nodalisation scheme, assumptions or adopted values, other relevant input information
      iii. Benchmark results
      iv. Discussion of benchmark results
      v. Conclusions and recommendations
   b. Experiment 2

…

4. Facility 2 (equivalent to bullet 3)

…

5. References

List of Tables & Figures

Glossary

ANNEX V. MEETING AGENDA

3rd Research Coordination Meeting (RCM) of the IAEA CRP1496 on Benchmarking against Experimental Data of the Neutronic and Thermalhydraulic Computational Methods and Tools for Operation and Safety Analysis for Research Reactors

5 - 9 December 2011, CEA Cadarache, Aix-en-Provence, France

Monday, 5 December 2011		
09:00-09:30	- Welcome by the IAEA and CEA - Self introduction of the participants - Status of the CRP and expected output of the 3rd RCM - Adoption of the Meeting Agenda - Election of Chairperson/Rapporteur	IAEA & CEA ALL IAEA ALL ALL
09:30-10:30	Presentations of the Chief Scientific Investigators (CSIs) on the status of the individual Research Contracts/Agreements, including results achieved & evaluation of the progress in relation to the achievement of the CRP objectives & outcomes (30 min each)	Argentina1+Argentina2
10:30-11:00	Coffee break	
11:30-13:00	Presentations of the CSIs: continued (30 min each)	Bangladesh, Canada, Greece
13:00-14:00	Lunch Break	
14:00-15:30	Presentations of the CSIs: continued (30 min each)	Ghana, Italy, Romania
15:30-16:00	Coffee Break	
16:00-17:30	Presentations of the CSIs: continued (30 min each)	Australia1+Australia2, France, France2

Tuesday, 6 December 2011		
09:00-11:00	Presentations of the CSIs: continued (30 min each)	France3, South Africa, Syria, USA
11:00-11:15	Coffee break	
11:15-12:15	Transportation of the participants by bus to CEA Cadarache	
12:15-13:45	Lunch at “Château de Cadarache”	
14:00-15:00	Visit of Jules Horowitz Reactor (RJH) construction site	
15:00-15:30	Transportation of the participants by bus – change of site	
15:30-16:30	Visit of ITER construction site	
16:30-17:30	Return by bus to Hotels in Aix-en-Provence	

Wednesday, 7 December 2011		
09:00-09:45	NEA activities in the area of the International Reactor Physics Experiments Benchmark (IRPhE) project, including demonstration of the related Database Tool	Mr I. Hill, NEA/OECD
09:45-10:30	Presentation on the use of RELAP Code: recent experience in research reactor safety analysis after Fukushima accident	Mr C.M. Allison, USA
10:30-11:00	Coffee break	
11:00:12:00	Presentation on CATHARE code: experience in computer code benchmarking for research reactor analysis	Ms B. Noel , CEA France
12:00-12:30	Experience in computer code benchmarking for research reactor analysis at KAERI	Mr S. Park and Mr B. Lee
12:30-13:00	New experimental results of the instrumented fuel element at Brazilian RR	Mr P.E. Umbehaun, Brazil
13:00-14:00	Lunch Break	
14:00-15:30	Consolidation of the status and progress made in relation to the achievement of the CRP objectives	All
15:30-16:00	Coffee Break	
16:00-17:30	Consolidation of the status and progress made in relation to the achievement of the CRP objectives	All
19:00	Hospitality event in Aix-en-Provence	

Thursday, 8 December 2011		
09:00-10:30	Adjustment of the work plan for the last year of the CRP: Consolidation of the CSIs' individual work plans based on the developed CRP action matrix and on the availability of new experimental data	All
10:30-11:00	Coffee break	
11:00:13:00	Continuation of the morning session	All
13:00-14:00	Lunch Break	
14:00-15:30	Discussion, critical review and finalization of the 1st publication on "10 RR specifications, experiments description and experimental results" as the 1st output of the CRP	All
15:30-16:00	Coffee Break	
16:00-17:30	Discussion, finalization of detailed table of contents, assignment of responsibilities for separate sections of the 2nd publication on "Code to experiment comparison" as a major output of the CRP	All

Friday, 9 December 2011		
09:00-10:30	Discussion and drafting the RCM report	All
10:30-11:00	Coffee break	
11:00:12:00	Drafting and drafting of the RCM conclusions/recommendations Finalization of the RCM meeting report	All
12:00	Closing of the Meeting	

ANNEX VI. LIST OF PARTICIPANTS

3rd Research Coordination Meeting (RCM) of the IAEA CRP1496 on Benchmarking against Experimental Data of the Neutronic and Thermalhydraulic Computational Methods and Tools for Operation and Safety Analysis for Research Reactors

5 - 9 December 2011, CEA Cadarache, Aix-en-Provence, France

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ANNEX VII. RCM PHOTOS

RCM participants

Construction site of Jules Horowitz Reactor

ANNEX VIII. SUMMARY OF WORK STATUS FROM PARTICIPANTS

Australia

Australia has revised the original documents for the specification of the OPAL reactor and experiments and associated measurements following comments and feedback from other participants and included data on OPAL operation to allow calculations related to fuel burnup to replace the FRJ-2 data originally expected. In addition templates have been provided for the collation of OPAL related results to assist in reporting, analysis and comparisons.

All calculation results related to OPAL experiments have been completed. It is possible that some of these will be updated as more sophisticated calculation methodologies are developed and refined. The results obtained were in good agreement with experimental values.

The SPERT IV neutronic calculations are almost complete with some refinement expected following discussions at the 3rd RCM. SPERT IV transient analyses using RELAP will be finalised shortly and a new set of results will be provided based on PARET using more appropriate critical heat flux correlations available in PARET. Some details regarding location of voiding strips within the coolant channels for the uniform void coefficient are required. Clarification of the physical quantity represented by the term flux (thermal or total neutron flux or Co activity) is needed. Assessment of transient results from RELAP indicate that for large reactivity insertion accidents temperature and power values are significantly overestimated. For more extreme reactivity insertions the solution did not converge.

Steady state thermalhydraulic calculations for ETTR-2 have been completed but analysis of transients is outstanding. It is expected that this work will be completed within the extended period of the CRP. Some specific details regarding fuel power peaking of the core and specific pump characteristics would assist in the development of a thermalhydraulic model although the current model adopts nominal values for such reactors.

Brazil

A description of the facility IEA-R1 reactor was presented (fuel assembly, pool, equipment, isometric piping), axial power distribution (for lateral and central plates) for neutronic and thermal-hydraulic modeling. Only some specific details are pending, but can be provided before the end of February.

The main commitment made was to provide data from a new experiment performed with the Instrumented Fuel Assembly. A new series of experiments were performed with the instrumented fuel assembly in the IEA-R1 research reactor and presented in this meeting. These experiments were performed at 4.0 MW with a box around the core and without the box.

The main conclusion is that the installation of a box around the core, reduced the temperature gradient between central plate and lateral plate, indicating that the flow in external channels is smaller without the box. Therefore, the installation of a box around the core is a good solution to eliminate cross flow and reduce the difference in temperature between internal and external plates.

Another important conclusion is that the installation of a box around the core does not affect the temperatures of fluid and clad in the transient of Loss of Flow Accident (Phenomena of Natural Circulation).

Greece

By the end of the 2nd year the following, Greece has undertaken and completed the following actions:

- Submission of RSG-GAS results template on 28/09/2010
- Submission of ETRR-2 results template on 28/09/2010

- Submission of SPERT-IV results template on 18/01/2011
- Submission of Progress Report in collaboration with Canada on 05/08/2011
- Participation in RELAP5 workshop during 31/01/2011-11/02/2011
- Finalized modeling of RSG-GAS, ETRR-2 and SPERT-IV reactors

  Journal papers
- S. Chatzidakis, A. Ikonomopoulos, S.E. Day (2011)"PARET-ANL modeling of a *SPERT-IV experiment under different DNB correlation",* Nuclear Technology, Volume 177, pp. 119 – 132, January 2012
- S. Chatzidakis, A. Ikonomopoulos, D. Ridikas (2011)*"Evaluation of RELAP5/MOD3 behavior against loss of flow experimental results in two research reactor facilities"*, Nuclear Engineering and Design, under review

Technical conclusions drawn so far:

- RSG-GAS and ETRR-2 steady state modelling employed the RELAP5 code comes in excellent agreement with the experimental results since the deviation is < 4%
- The model estimations for the cladding temperature show that RELAP5/MOD3 can provide very accurate predictions for thermocouples located within the core channels
- Beyond the core boundaries RELAP5/MOD3 estimates deviate from the experimental values especially under natural convection
- The SPERT-IV analysis performed employed the PARET-ANL code for the entire range of reactivity insertions (0.80 - 2.14$) producing transients with periods ranging from 980 to 7.0 msec under various coolant flow conditions
- A systematic comparison with the experimental measurements has been performed
- Transients showed very good agreement with the experimental results for reactivity insertions up to 1.2 $ (period > 50msec)
- rRMSE was used to identify and quantify the regions where significant variations from the experimental results were experienced
- **Attention ought to be paid to the DNB correlation selection**

3rd year commitments:

- By June 2012, the IEA-R1 reactor modelling results will be provided to the IAEA and the results consolidators
- By June 2012 the final report containing all the simulation results will be provided to the IAEA
- Consolidate results for RSG-GAS, ETRR-2 and SPERT-IV reactors.

France-CEA

a) CEA commitments

CEA is involved in two benchmarks. The first one is the interpretation of Valmont program in MINERVE for which CEA is the provider of experimental data. The second one is SPERT IV program for both neutronics and thermal hydraulic simulations

b) Current status

Regarding Valmont in MINERVE, all the simulation have been carried out, however the non physical asymmetry observed on the calculation of the axial fission rate distributions needs further investigation.

For SPERT IV, all the neutronics configurations have been studied except spectrum factor. For some of them additional calculations are needed to be able to normalize the results as required in the template. All the thermal hydraulic transients have been analyzed but due to the important discrepancies between calculation and measurement further improvement of the modeling will be done. However, the draft of the EXCEL template is available with all the results obtained.

c) Modification of MINERVE specification

The MINERVE specifications will be corrected before the end of January 2012 to represent the VALMONT fuel pin I annular, with 5mm internal diameter.

d) Results

MINERVE

The different experiments have been analyzed using the Tripoli4 French Monte Carlo code version 4.7 and the JEFF3.1.1 nuclear data library. All the calculation results are consistent with experimental ones, within 2 experimental standard deviations. Particularly the spectral indices, reactivity worth, conversion ratio and radial power distribution are very well calculated.

However the non physical asymmetry observed on the calculation of the axial fission rate distributions needs further investigation.

Spert IV neutronics

The simulations were carried out using TRIPOLI4 version 4.7 and the JEFF3.1.1 nuclear data library. All parameters are in very good agreement with the measurements. The critical configuration with 25 loaded fuel elements is calculated with accuracy better than 100 pcm. The kinetics parameters (ratio between neutron life time and beta effective) deviate by 2.5% from the experimental measurements. The reactivity coefficient, void and temperature, are consistent with the experimental measurements and it is the same for the rod worth. There are more discrepancies for the axial flux distribution mainly at the bottom of the fuel assembly. This is probably due to the approximation in the modeling of the bottom structure witch is not well known. All the calculated neutronics parameters are used for the thermal hydraulic simulation but some data are lacking. It is the case for the axial position of the transient rod before ejection. In this case the given efficiencies were assumed and the corresponding positions were calculated.

Spert IV Thermal hydraulics

The objective of this presentation was to give some preliminary results of the thermalhydraulics analysis of the SPERT IV tests using the CATHARE2 code. In the SPERT IV programme, 39 tests were performed with reactivity injections ranging between 0.80 and 1.91$, without and with forced flow rates up to 5000 gpm at ~1.5 bar and with an inlet water temperature of ~20°C. The facility is nodalized by an inlet pipe, a volume below the core, the 25 fuel assemblies were modelled by 6 independent hydraulics channels (with associated fuel plates), two volumes to represent the upper and the lateral parts of the pool. The power distributions in the core, the temperature and the void reactivity effects were calculated using TRIPOLI4 code (in very good agreement with the data). We observed an important overestimation of the peak power and of the cladding temperature in these preliminary calculations. This could be due to an underestimation of the HT exchanges and of the associated reactivity feedback. Sensitivity calculations were done to analyse the effect of different parameters. This analysis shows that:

- The HT exchanges and the CHF have an important effect on the tests with large reactivity injections > 1.40$,
- The uncertainty on the value of the injected reactivity has an important effect on the tests with low reactivity injections < 1.40$,
- The peak cladding temperature was still overestimated using an imposed nuclear power but to a lesser extent in comparison to the preliminary calculations.

As a conclusion, this work should be continued in order to better explain the discrepancies between the calculations and the experiments.

Ghana

a) An overview of the commitments made**:** to study the neutronics and thermal-hydraulic benchmarks for MNSR in Syria and SPERT III using MCNP5 and PARET/ANL and the complete conversion analysis for Ghana Research Reactor-1.

b) The currents status and progress against these commitments: The neutronic and thermal-hydraulic benchmarks for Syrian MNSR have been completed. The transient parameters obtained from PARET/ANL for 3.6 mk insertion of reactivity have been found as follows: peak power of 80.3567 kW, mass flow rate of 15.9780 $kg/s/m^2$, inlet temperature of 64.0063 °C, fuel cladding temperature of 84.0105 °C, fuel centre-line temperature of 85.3006 °C and outlet temperature of 64.0063 °C. Steady state computations performed with PARET/ANL at 15 kW resulted in flowrate of 13.1590 $kg/s/m^2$, inlet temperature of 46.6385 °C, fuel cladding temperature of 52.2516 °C, fuel centre-line temperature of 52.4930 °C and outlet temperature of 46.6385 °C. Steady state computations performed with PARET/ANL at 30 kW resulted in flowrate of 17.8580 $kg/s/m^2$, inlet temperature of 61.4137 °C, fuel cladding temperature of 70.8531 °C, fuel centre-line temperature of 71.3358 °C and outlet temperature of 61.4137 °C.

The neutronic model for SPERT III has been established requiring clarifications on the concentrations of the materials in the facility. The initial computation with the model resulted in k-inf of 1.07266 with standard deviation of 0.00063, resulting in reactivity of 67.7381 mk where mk = 10^{-3} $\Delta k/k$. The k-inf with prompt neutrons only resulted in 1.06533 with standard deviation of 0.00063. The β-eff was found to be $(6.414 \pm 0.001897) \times 10^{-3}$.

<u>c) A concise list of data/experimental</u> related shortcomings which should be added to the specifications (or at least should be discussed with the providers or other analysts). The team requires the material composition of the components of the SPERT III reactor.

<u>d) The most important technical conclusions</u> you can draw from the analysis work performed thus far. MCNP5 and PARET/ANL have been adequate in simulating the various parameters of the two reactors so far. The safety analysis of GHARR-1 for the conversion from HEU to LEU has been performed.

Argentina

Overview of the commitments made

The commitments set during the 2nd Research Coordinated Meeting are described in the following paragraphs:

Neutronic area:

For MNR:

- Calculation of simulated void experiments to estimate the local void coefficients
- Calculation of pool temperature experiment to estimate fuel and other core components temperature coefficients

For WWR-SM:

- Simulation of different neutronic benchmark experiments

For OPAL reactor:

- Deeper analysis of the axial flux due to a lack of agreement between experimental and calculated values

For ETRR-2 reactor:

- Neutronic benchmark including: criticality (k-eff, k-inf), flux shape/profile, control rod worth, reactivity effects, reactivity coefficients and kinetic parameters

Thermal-hydraulic area:

- Simulation of a negative reactivity insertion in the ETRR-2 reactor
- Modeling of the RSG-GAS reactor, steady state and comparison of LOFA results
- Modeling of the IEA-R1 reactor, steady state and comparison of LOFA results

Current status and progress of activities

Neutronic area:

For MNR the following tasks were done:

- Modeling and estimation of void experiments – Critical configurations were modeled and the local void coefficients were estimated
- Pool temperature experiment – Critical configurations were modeled and fuel and other core components temperature coefficients were estimated separately.

The analysis of the criticality offset in HEU cores and the use of thermal- hydraulic feedback in the pool experiment, still remain to be performed.

For WWR-SM, no tasks were performed as experimental data was not supplied.

For OPAL reactor, the deeper analysis showed a misinterpretation of the experimental data. From this analysis some suggestions were done to correct the measurements.

For ETRR-2 the neutronic benchmark calculation is a pending issue

Thermal-hydraulic area:

During the 3rd Research Coordinated Meeting the tasks previously stated were fulfilled meaning that there is no pending task. However, some sensitivity analysis could be performed or new calculations could be achieved based on new data presented by Brazil team during the meeting.

Missing data

Neutronic area:

For MNR no missing data is reported. However, additional experimental results such as local void coefficients and temperature coefficients would be useful to improve the comparison.

Thermal-hydraulic area:

No missing data for the RSG-GAS reactor although there is a lot of information which results confusion. An input data set will be agreed among the benchmark participants.

For IEA-R1 reactor information shall be supplied for the value used for the normalized axial power density profile of the instrumented fuel assembly.

The corresponding specification of the new LOFA transient shall be provided ASAP.

Conclusions

Neutronic area:

For MNR:

- ✓ LEU core experiments are properly modeled both control rods calibration and axial wire activation rates normalized to peak
- ✓ Void experiment critical configurations were modeled with some problems
- ✓ Pool temperature experiment was properly modeled

For OPAL:

- ✓ Calculated reaction rates properly represent the measured values far from the control rods
- ✓ The difference observed in the wires next to control rods could be due to a shielding between wires and control rods that was not modeled
- ✓ The lack of Cadmium factors measurement at the end of the fuel plates resulted in unrealistic values of the thermal flux. This resulted in a weak agreement between calculated and experimental values. When Cd factors are properly measured this disagreement disappears

Thermal-hydraulic area:

- ***Simulation of a negative reactivity insertion in the ETRR-2 reactor***

  The benchmarking agreement achieved is good in all relevant parameters of SS and, particularly, in the coolant temperatures during the transient. The temperature differences between measured and calculated values are within the accuracy of the instruments.
- ***Modeling of the RSG-GAS reactor, steady state and comparison of LOFA results***

The RELAP5 calculation model shows good agreement with the facility both for the SS and LOFA experimental results. Although transient calculations overestimate the maximum cladding temperatures measurements they are conservative for safety assessment.

Input data should be verified as in some cases it is over-specified and/or confusing.

- ***Modeling of the IEA-R1 reactor, steady state and comparison of LOFA results***

  There is a good agreement between measured and calculated coolant temperatures for the different SS presented.

  Differences in the wall temperatures could be attributed either (i) to the heat transfer correlation that underestimates the heat transfer coefficient; (ii) power density averaged over 6 plates while the temperature is measured on a single plate and/or (iii) a deficient contact between the thermocouple and the cladding surface.

  In the forced convection regime calculated values are conservative so they are applicable for safety assessment while in natural circulation regime temperatures are underestimated either, due to the "averaged channel effect" or 3D effects within the core.

Syria

❖ Overview of the commitments made

Based on the performed neutronic and steady state thermal hydraulic analyses and the achieved results during the last two years 2009-2010 the 3rd year was devoted to complete neutronic analyses for SPERT IV experiments and transient thermal hydraulic benchmark calculation for the reactors ETRR-2, RSG-Gas, IEA-R1and SPERT-IV.

❖ The currents status and progress against these commitments

1. Finalization of SS and LOFA analysis for ETRR2 and RSG-GAS reactors using MERSAT and RELAP Codes.
2. Completing the neutronic analyses for SPERT IV experiment using the code MCNP .
3. Developing MERSAT input-deck for TH analyses of SPERT-IV and performing 1 benchmark set of 0.8 $.
4. Completing Benchmark analysis of the Brazilian IEA-R1 reactor using of MERSAT code and incorporating additional calculation with the code PARET (was not committed previously).
5. Code-to-code benchmarking (results comparison with other teams) for the safety analysis of ETRR2, RSG-GAS, SPERT IV and IEA-R1.
6. Preparation of interim report.

A concise list of data/experimental related shortcomings which should be added to the specifications (or at least should be discussed with the providers or other analysts)

<u>for the Indonesian RSG-GAS reactor:</u>

1) Water height over the core, core inlet and outlet plenum areas and bypass area;
2) Geometrical information about the natural circulation flaps (with dimension);
3) Radial distribution of power density across the reactor core;
4) Radial power density factor in IFE-10 and IFE-11 ;
5) Experimental data for channel pressure drop versus flow rate.
6) Regarding T2 it is specified as clad temperature in the average channel, whereas its real position is in the hot channel (according to the Indonesian report).
7) T3 is not used at all.

<u>For the Brazilian IEA-R1 reactor:</u>

Radial peaking factor

The **most important technical conclusions** you can draw from the analysis work performed thus far. The performed benchmark calculation proved to be a very useful approach for the validation of the selected computational tools. In particular, the TH experiments performed on RSG-Gas and IEA-R1 using instrumented fuel elements provide good opportunity to validate in particular the heat transfer package being used in the advanced TH codes RELAP and MERSAT. Besides, the related code-to-code comparisons could provide a useful approach with which to assess the prediction capability and ranking of the applied codes regarding the selected TH effects. This method supports the objective of the ongoing CRP for qualification of TH codes for safety analysis of RR. It is recommended for future activities to focus on a certain benchmark problem (integral effect) that covers various TH and neutronic effects (with good experimental documentation). The selected problem can be considered as a standard one that should be calculated by various teams using different TH codes.

USA

1. SPERT-III

The benchmark specification for the reactor is complete. A test matrix is provided of experimental conditions for reactivity insertion events for various initial conditions of power, pressure, flow, and inlet temperature. The experimentally measured results of power vs. time, energy release vs. time, and feedback reactivity vs. time have been digitized from published figures. USA has agreed to finish the analysis by 1 June 2012. We have agreed, as the CSI, to consolidate results from other organizations (Romania) per Annex IV of the previous CRP. Graphical output will be standardized in EXCEL format. A question was raised as to how many assemblies are in the core: is it different for cold tests vs. hot? USA will respond.

2. WWR-SM (in collaboration with Uzbekistan)

There was no representative from Uzbekistan at this RCM. Dr. Olson agreed to assist in determining the status of this work, and to assist in completing it if possible.

3. MNSR (in collaboration with Ghana)

This work is being provided by Ghana. The USA assisted in preparing the neutronics model for calculating pin powers and feedback coefficients from temperature change or coolant density change from void. Point kinetics parameters beta-eff and prompt neutron lifetime have been computed. The USA also assisted in preparing the steady-state thermal hydraulics model for the PLTEMP code. The USA also assisted in PARET transient analysis by modifying an existing model that was developed for Nigeria. The benchmark case is a typical limiting reactivity insertion transient for this class of reactor. The analysis is complete. Work remains to prepare the report.

Other notes:
Ian Hill of NEA/OCD Paris asked for any SPERT I documents that USA/ANL may have.
Simon Day (Canada) provided three design drawings with manufacturing tolerances that USA will add to the SPERT III benchmark specification.

Italy

Overview of the commitments made

The objective of the present activity concerns the application of the coupled codes technique to research reactor safety analysis and its assessment. This goes through the development of all required steps ranging from cross section generation through unit cells modeling, and full core simulation under steady state and transient conditions.

For this purpose the cross section code WIMSD5, the 3D Neutron Kinetic (NK) diffusion code PARCS and the Thermal–Hydraulic (TH) System Code (THSC) RELAP5/Mod3.3 code are to be used. The model was successfully applied to generate the steady state and the dynamic behavior under RIA of the IAEA standard MTR benchmark HEU core.

The currents status and progress against these commitments
Activities are ongoing, the previously developed nodalization are to be modified to comply with the SPERT-IV experimental data using the developed methodology.

Planned further steps:

- Completion of the ongoing analysis on SPERT-IV experiment using the PARCS and RELAP5/3.3 Coupled codes.
- Performing transient thermalhydraulic analysis of SPERT-IV experiment with PARCS-RELAP5/3.3.
- Preparation of the final report of SPERT-IV and submission of the results of the benchmark calculations.
- Interest in the IEA-R1 benchmark calculations.

A concise list of data/experimental related shortcomings which should be added to the specifications (or at least should be discussed with the providers or other analysts)

The most important technical conclusions you can draw from the analysis work performed thus far. The demonstration of the applicability of qualified best-estimate system codes to Research Reactor accident analysis, based on the benefits of the experience available from NPP, constitutes the challenge of this work. During our previous work, a demonstration of utilization of a best estimate system code (BE) such as RELAP5/3.2 has shown great benefits of using standardized computer codes. The purpose of this activity is a tentative demonstration of the application of coupled code techniques to research reactor safety analysis.

In this light, a 3D-Neutron Kinetics and Thermal Hydraulic Code (3D-NKTH) is developed based on coupling the cross section generation code WIMSD5, the 3D NK diffusion code PARCS and the thermal hydraulic system code (THSC) RELAP5/3.3 code. This system will be applied to SPERT-IV.

The code has been successfully applied to the IAEA benchmark HEU core and verified by comparison to tabulated results for steady state and for a hypothetical rod ejection accident.

South Africa

a) An overview of the commitment made:
   a. OPAL benchmark problem (neutronics)
      OSCAR-4, MCNP
   b. MNR benchmark problem (neutronics)
      OSCAR-4
   c. ETRR2 benchmark problem (neutronics and thermal-hydraulics)
      OSCAR-4, RELAP

b) The current status and progress against these commitments:

Benchmark	Status of data	Code	Model building	Experiments modeled
OPAL	Facility: Complete Experiment: Complete (some clarification needed)	OSCAR-4	Complete	Critical cases, Foils, CRC, burn-up
		CUCGP	Complete	Critical cases, foils, CRC

		MCNP	Complete	Criticals, foils
MNR	Facility: Complete Experiment: Some clarification needed	OSCAR-4	Complete for 54A	CRC, radial flux wires, axial flux wires
ETRR-II	Facility & Experiment: Some clarification needed	OSCAR-4	Partial	CRC (assumptions made)
		RELAP	Partial	

c) The current status of data/experimental related shortcoming which should be added to the specifications (at many instances there was lack of data and that led to some interpretation which might differ from other participants);
   a. Opal: clarification of the correct location of the cold neutron source and other experimental facilities.
   b. MNR:
      i. CRC experimental data
         - Need to agree on a consistent way to interpolate the S-curves
      ii. Radial Flux Mapping
         - Need to investigate the source of large differences
      iii. Axial Flux Measurements
         - Detector model for non-fuel components needs to be implemented
      iv. Still need to do calculations for the miscellaneous reactivity measurements
   c. ETRR-2
      i. Control rod calibration data
      ii. Incomplete descriptions of other core configurations
      iii. For all cores configurations: critical rod positions not given
      iv. Axial and radial power peaks
      v. Primary pump data to simulate LOFA

d) Technical conclusions from analysis work performed thus far:
   a. General
      i. The method of comparisons e.g. normalization to core average and simple error estimations, might not be consistent with the other participants.
   b. MNR
      i. The shortcoming in accurate number densities supplied in the benchmark does not seem to detract from the quality of the benchmark problem
      ii. A large uncertainty seems to exists in the control rod calibration data supplied for rods 1 – 5, the source of which should be identified
      iii. A k-eff bias of around 1500 pcm exists, but with a very narrow variation around this value.
      iv. The model comparisons to flux shapes and critical cases seems to agree quite well with experimental data
   c. OPAL
      i. A Monte Carlo, full core transport and deterministic diffusion solution are presented for this problem and generally all approaches prove quite accurate.
      ii. The burnup case is particularly neutronically challenging given the Cd absorber wires in the fuel design.

Romania

a) Commitments were made for SPERT-III and MINERVE
b) SPERT-III – 90% accomplished. Model created for neutronics, comparison with provided data realized and calculation of kinetic parameters and insertion of reactivity transients. Only some more verification of modeling for the tests that showed discrepancies with experimental data.
MINERVE – 90% accomplished. Need only to run again the radial power distribution, with new dimensions for the Valmont pin.
c) Currently, data are clear
d) For SPERT-III: Good results for some of the test series. Overestimation of peak power and released energy for cold test series. For MINERVE: good agreement with static neutronic data provided.

Korea

In the 2nd CRP meeting, KAERI made a commitment to accomplish thermal hydraulic analyses of the experiments of the IEA-R1, RSG-GAS, and ETRR2 using RELAP5/MOD3.

KAERI has done the thermal hydraulic analyses of the experiments of the IEA-R1 and RSG-GAS using RELAP5/MOD3.3, and presented the results in this meeting. In addition, KAERI has started the neutronic analysis for the OPAL experiment using the MACARD code, and preliminary results are presented.

KAERI's modeling of the thermal hydraulic tests concludes:

- RELAP5/MOD3.3 predicts the overall trend of the coolant and cladding temperatures measured despite a lack of information on the experiment,
- Detailed information on the experiment and measurement uncertainties is required for a better code validation.

For a better validation of RELAP5/MOD3, KAERI proposes that clarification of the following data is required:

- Radial peaking factor to calculate the power of the instrumented fuel assembly (IFA),
- Detailed geometric data on the core to calculate the coolant volume and loss coefficients for the IFA, other fuel assemblies, fuel assembly bypass region, fuel matrix plate, and core outlet plenum in the core modeling,
- Extended data of the pump coast-down flow down to zero flow.

Canada

Overview of Commitments Made & Status

Research Reactor	Deliverable	Date	Status
SPERT IV D-Core	Facility Specification	January 2011	Complete (R02)
SPERT IV D-Core	Benchmark Specification	January 2011	Complete R01
SPERT IV D-Core	Results Template	January 2011	Complete (R01)
SPERT IV D-Core	Data Collection	-	Pending
MNR	Facility Specification	January 2011	Complete (R02)
MNR	Benchmark Specification	January 2011	Complete (R01)
MNR	Results Template	January 2011	Complete (R01)
MNR	Calculations	-	Pending (5% to date)
MNR	Data Collection	-	Pending

Details on Current Status and Progress

The SPERT IV D-Core Facility Specification (R02) requires revision to correct erratum with regards to absorber material atom densities. This correction has already been supplied to the CRP group and is

already incorporated in the draft project document. For completeness the SPERT IV Facility Specification will be reissued (R03) with this change.
With regards to the MNR Benchmark Specification, INVAP has requested additional information (if available) regarding MNR experimental results, specifically, rod calibration data for experimental cores related to the void and temperature experiments. If the data is available it will be supplied and added to the Benchmark Specification as a re-issue (R02).
Data collection is pending but dependent on user groups finalizing their results. France 1 (CEA) have supplied SPERT IV results but will likely finalize/revise their transient work.

Data/Experimental Concerns
See above regarding additional MNR experimental information. Results to date from groups analyzing the MNR problem suggest the quality of the Rod Calibration data may be questionable. The problem is also challenged by available information on burnup distribution.
The SPERT IV D-Core problem is limited in some respects due to lack of available technical specifications and others by lack of experimental detail. The problem could be improved if more information can be located. Specific points include lack of information on core plenum and support structure as they relate to thermal-hydraulic nodalization, and local placement of void plates for uniform void measurement.
Section on experimental uncertainties can be added to both MNR and SPERT IV Benchmark Specifications.

Technical Conclusions
Groups have yet to reach final conclusions, but results to date from the various analysis groups for the SPERT IV D-12/25 transient analysis suggest serious challenges for simulating modest to severe RIA events. This is obviously an important conclusion for the RR community, particularly topical given the current license requirements of the JHR to include BORAX-scenario analysis. (Sidebar: this project is calling the SPERT IV transient analysis "thermal hydraulics". I believe this is misleading as the experiment most certainly involves complex coupling of neutronics and thermal-hydraulics).
A suggestion for the CRP report, as raised by G. Braoudakis (OPAL), is to include a summary of recommendations for future experiments in the context of benchmarking analysis tools, based on experience of the various user groups in this CRP exercise. This could take the form of a short appendix covering both the neutronics and thermal-hydraulic points of interest (not forgetting the coupling of the two areas). This would be a valuable contribution.

Bangladesh

a) Overview of the commitments are
   a. to perform the neutronics and thermal hydraulics calculations of SPERT IV 12/25 reactor using the Monte Carlo Code MVP and the thermal hydraulic code EUREKA-2/RR respectively,
   b. similar calculations will be done for IEA-R1 within the estimated time of IAEA after this meeting
b) Current status and progress against the commitments:

Task	Tools	Achievements
Neutronics of SPERT IV	Monte Carlo Code MVP	90 %
Thermal hydraulics of SPERT IV	EUREKA-2/RR	20 %
Similar calculations will start for IEA-R1	MCNP or MVP or EUREKA-2/RR	Under investigation

c) Concise list of data/experimental related shortcomings may be needed based on the above status. So we communicate to the responsible person (Simon Day, for SPERT-IV).
d) We conclude that from the neutronics analysis of SPERT-IV reactor that the MVP code is reliable for neutronics calculation.

France-IRSN

IRSN has completed calculations for the SPERT-IV case for neutronics and for thermal-hydraulic parts.
Neutronics calculations were performed by the Monte-Carlo tools of the SCALE 6.0 code package.
SCALE (Standardized Computer Analyses for Licensing Evaluation) is developed and maintained by Oak Ridge National Laboratory under contract with NRC and DOE to perform reactor physics, criticality safety, radiation shielding, and spent fuel characterization for nuclear facilities and transportation/storage package designs. This code package is able to perform the calculations of the criticality, the depletion and the reactivity coefficients in an arbitrary geometry in multi-group neutron field representation. In addition to the other tools based on the Monte-Carlo modeling it has well developed perturbation theory modules to be used for sensitivity coefficients calculations, for uncertainty propagation and for kinetic parameters determination. The code has been validated thoroughly against numerous experimental cases with different geometry and spectra, so concerning validity of reactivity studies SCALE is well qualified too. But SCALE and especially TSUNAMI and TSAR modules could be used exactly for static discrepancies propagation onto the transient calculations.

The perturbation calculation opportunities allowed performing highly accurate spatial power and moderator density reactivity estimation. The kinetic parameters had been evaluated with perturbation theory as well. Control rod worths were done with coherent movement of rods as a bank.

In order to provide data for thermal hydraulic and coupled transient analysis the power distribution was studied with different positions of the Transient Rod. The power distribution calculations included evaluation of the balance of heat generation between fuel, clad and coolant.

Thermal hydraulic computation was performed for imposed power evolution with given (calculated by SCALE) power spatial distribution.

All of the calculations were performed with the ASTEC code package ASTEC code System of codes for severe accident simulation was co-developed by IRSN and GRS Version V2.1 under development (including plate-type fuel).

The peculiarity of the calculation had been done was the control the CHF for regimes with burn-up calculations. Modeling of SPERT-IV assemblies includes Single channel for one assembly 12 plates for fuel assemblies and 6 plates for control assemblies, there were 15 axial meshes 10 radial meshes inside fuel (UAl) and 10 radial meshes inside cladding (Al). The boundary conditions were the constant temperature for inlet and mass flow rate. Power pulse shape vs. time was taken from the SPERT-IV experimental database. The spatial distribution had been calculated before by the SCALE code.

The consistency of coupled modeling calculations had been examined with reconstruction of the feedback reactivity from the comparison between derived from the experimental power evolution reactivity and the reactivity feedback synthesized upon the calculated density and specific moderator worth spatial distributions.